14.1.9 Loss of External Electrical Load

Accident Description

The loss of external electrical load event is defined as a complete loss of steam load or a turbine trip from full power without a direct reactor trip. This anticipated transient is analyzed as a turbine trip from full power as this bounds both events: the loss of external electrical load and turbine trip. The turbine trip event is more severe than the total loss of external electrical load event since it results in a more rapid reduction in steam flow.

For a turbine trip, the reactor would be tripped directly (unless below approximately 10% power) from a signal derived from either the turbine auto-stop oil pressure or a closure of the turbine stop valves. The automatic steam dump system accommodates the excess steam generation. Reactor coolant temperatures and pressures do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser were not available, the excess steam generation would be dumped to the atmosphere. Additionally, main feedwater flow would be lost if the turbine condenser were not available. For this situation, steam generator level would be maintained by the auxiliary feedwater (AFW) system.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. The plant would be expected to trip from the reactor protection system (RPS). A continued steam load of approximately 5% would exist after a total loss of external electrical load because of the steam demand of plant auxiliaries.

In the event of a large loss of load in which the steam dump valves fail to open or a complete loss of load with the steam dump operating, the main steam safety valves (MSSVs) may lift and the reactor may be tripped by any of the following signals: high pressurizer pressure, high pressurizer water level, overtemperature ΔT (OTΔT), overpower ΔT (OPΔT), or low-low steam generator water level. The steam generator shell-side pressure and reactor coolant temperatures will increase rapidly. However, the pressurizer safety valves (PSVs) and MSSVs are sized to protect the reactor coolant system (RCS) and steam generators (SGs) against overpressure for all load losses without assuming the operation of the steam dump system. The steam dump valves will not be opened for load reductions of 10% or less, but may open for larger load reductions. The RCS and main steam system (MSS) steam relieving capacities were designed to ensure safety of the unit without requiring automatic rod control, pressurizer pressure control, steam bypass control systems, or a reactor trip on turbine trip.

Method of Analysis

The loss of load transients are analyzed using the RETRAN computer code. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and main steam safety valves. The code computes pertinent plant variables including temperatures, pressures, and power levels.

The loss of load accident is analyzed: (1) to confirm that the PSVs and MSSVs are adequately sized to prevent overpressurization of the primary RCS and MSS, respectively; and (2) to ensure that the increase in RCS temperature does not result in a departure from nucleate boiling (DNB) in the core. The RPS is designed to automatically terminate any such transient before the DNB ratio (DNBR) falls below the limit value.

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power with no credit taken for a direct reactor trip on turbine trip. This assumption will delay reactor trip until conditions in the RCS cause a trip on some other signal. Thus, the analysis assumes a worst case transient and demonstrates the adequacy of the pressure relieving devices and plant-specific RPS setpoints assumed in the analysis for this event.

Of the three cases analyzed, one is performed to address DNB concerns, one ensures that the peak primary RCS pressure remains below the design limit (2750 psia) and the final case confirms that the peak MSS pressure remains below 110% of the SG shell design pressure (1210 psia). The major assumptions for these cases are summarized as follows:

a. For the case analyzed to demonstrate that the core thermal limits are adequately protected (beginning of cycle (BOC) reactivity feedback conditions with automatic pressurizer pressure control), the loss of load accident is analyzed using the revised thermal design procedure (RTDP). For this case, initial core power, reactor coolant temperature, and reactor coolant pressure are assumed to be at the nominal values consistent with steady-state full power operation. Uncertainties in initial conditions are included in determining the DNBR limit value. For the case analyzed to demonstrate the adequacy of the primary pressure relieving devices (BOC reactivity feedback conditions without automatic pressurizer pressure control), the loss of load accident is analyzed using the standard thermal design procedure (STDP). For this case, initial core power and reactor coolant temperature are assumed at the maximum values consistent with steady-state full power operation, including allowances for calibration and instrument errors. Initial pressurizer pressure is assumed at the minimum value for this case, since it delays reactor trip on high pressurizer pressure and results in more severe primary side temperature and pressure transients. This results in the maximum power difference for the loss of load. Similar to the primary RCS overpressurization case, the MSS overpressurization case is analyzed assuming the STDP assumptions with respect to initial conditions and uncertainties and also assumes BOC reactivity feedback conditions. However, it differs from the primary RCS overpressurization case in that automatic pressurizer pressure control is assumed in order to delay reactor trip.

b. The loss of load event results in a primary system heatup and is therefore conservatively analyzed assuming minimum reactivity feedback consistent with BOC conditions. This includes assuming a moderator temperature coefficient (MTC) value consistent with BOC hot-full power (HFP) conditions (i.e., zero MTC) and a least negative Doppler power coefficient (DPC). Maximum feedback (end of cycle (EOC)) cases that were previously considered in the USAR are no longer analyzed since they have been determined (as part of the Westinghouse methodology for the analysis of this event) to be non-limiting with respect to the minimum DNBR, peak primary RCS pressure, and peak MSS pressure.

c. It is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

d. No credit is taken for the operation of the steam dump system or SG power-operated relief valves (PORVs). The SG pressure rises to the safety valve setpoints, where steam release through the MSSVs limits the secondary side steam pressure to the setpoint values. As detailed in Section 5.1.0.4, staggered lift setpoints were modeled for the MSSVs in the loss of load licensing basis analysis, including a +3% accumulation and +1% setpoint tolerance. By maximizing the pressure transient in the MSS, the saturation temperature in the SGs is maximized, resulting in limiting pressure and temperature conditions in the RCS.

e. Three cases are analyzed:

1. For the case analyzed for DNB, automatic pressurizer pressure control is assumed. Thus, full credit is taken for the effect of the pressurizer spray and PORVs in reducing or limiting the primary coolant pressure. Safety valves are also available and are modeled assuming a - 1% setpoint tolerance.

2. For the case analyzed for primary RCS overpressure concerns, it is assumed that automatic pressurizer pressure control is not available. Therefore, no credit is taken for the effect of the pressurizer spray or PORVs in reducing or limiting the primary coolant pressure. Safety valves are assumed operable, but are modeled assuming a +1% setpoint tolerance. The effects of the PSV loop seals are also conservatively modeled in the analysis.

3. For the case analyzed for MSS overpressure concerns, it is assumed that automatic pressurizer pressure control is available. Credit is taken for the effect of the pressurizer spray and PORVs in reducing or limiting the primary coolant pressure, thus conservatively delaying the actuation of the RPS until an OTΔT reactor trip signal is generated. Delaying the reactor trip ensures that the energy input to the secondary system, and subsequently the MSS pressure, is maximized.

f. Main feedwater flow to the SGs is assumed to be lost at the time of turbine trip. No credit is taken for AFW flow since a stabilized plant condition will be reached before AFW initiation is normally assumed to occur for full-power cases. However, the AFW pumps would be expected to start on a trip of the main feedwater pumps. The AFW flow would remove core decay heat following plant stabilization.

g. The analysis is performed for operation with 422V+ fuel and a maximum steam generator tube plugging level (uniform) for Kewaunee of ≤10%.

Results

The transient responses for a total loss of load from full power operation are shown in Figures 14.1.9-1 through 14.1.9-16 for the three cases assuming BOC reactivity feedback conditions with and without automatic pressurizer pressure control (pressurizer spray and PORVs).

Figures 14.1.9-1 through 14.1.9-6 show the transient responses for the total loss of steam load at BOC (minimum feedback reactivity coefficients) assuming full credit for the pressurizer spray and PORVs to calculate the transient DNBR response. Following event initiation, the pressurizer pressure and average RCS temperature increase due to the rapidly reduced steam flow and heat removal capacity of the secondary side. The peak pressurizer pressure and water volume and RCS average temperature are reached shortly after the reactor is tripped by the OTΔT trip function. The DNBR initially increases slightly, then decreases until the reactor trip is tripped, and finally, following reactor trip, increases rapidly. The minimum DNBR remains well above the safety analysis limit value. The MSSVs actuate to limit the MSS pressure below 110% of the SG shell design pressure.

The total loss of load event was also analyzed assuming the plant to be initially operating at full power at BOC with no credit taken for the pressurizer spray or PORVs to maximize the primary RCS pressure response. Figures 14.1.9-7 through 14.1.9-11 show the transients for this case. The neutron flux remains relatively constant prior to reactor trip, while pressurizer pressure, pressurizer water volume and RCS average temperature increase due to the sudden reduction in primary to secondary heat transfer. The reactor is tripped on the high pressurizer pressure trip signal. In this case the PSVs are actuated and maintain the primary RCS pressure below 110% of the design value. The MSSVs actuate to limit the MSS pressure below 110% of the SG shell design pressure.

Figures 14.1.9-12 through 14.1.9-16 show the transient responses for the total loss of steam load at BOC (minimum feedback reactivity coefficients) assuming full credit for the pressurizer spray and PORVs to maximize the MSS pressure response. Following event initiation, the pressurizer pressure and average RCS temperature increase due to the rapidly reduced steam flow and heat removal capacity of the secondary side. The peak pressurizer pressure and water volume and RCS average temperature are reached shortly after the reactor is tripped by the OTΔT trip function. The MSS pressure increases, resulting in the actuation of the first three MSSVs, and then decreases rapidly following reactor trip. The MSSVs actuate to limit the MSS pressure below 110% of the SG shell design pressure.

Conclusions

The results of the analyses show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the primary RCS or MSS. Pressure relieving devices that have been incorporated into the plant design are adequate to limit the maximum pressures to within the safety analysis limits, i.e., 2750 psia for the primary RCS and 1210 psia for the MSS. The integrity of the core is maintained by operation of the RPS, i.e., the minimum DNBR is maintained above the safety analysis limit value.

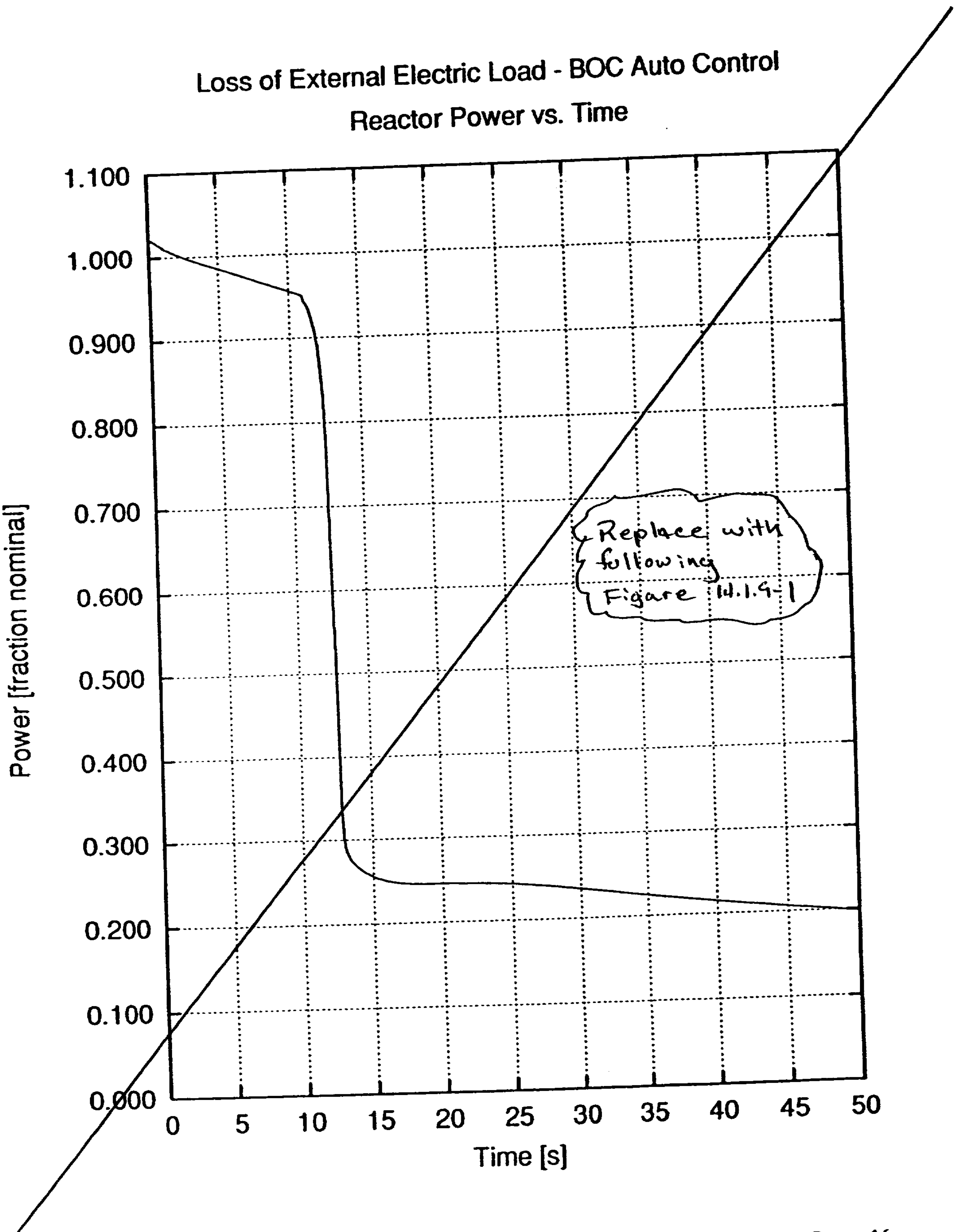

Figure 14.1.9-1

Loss of External Electrical Load With Auto Pressure Control (DNB Case)

Nuclear Power vs. Time

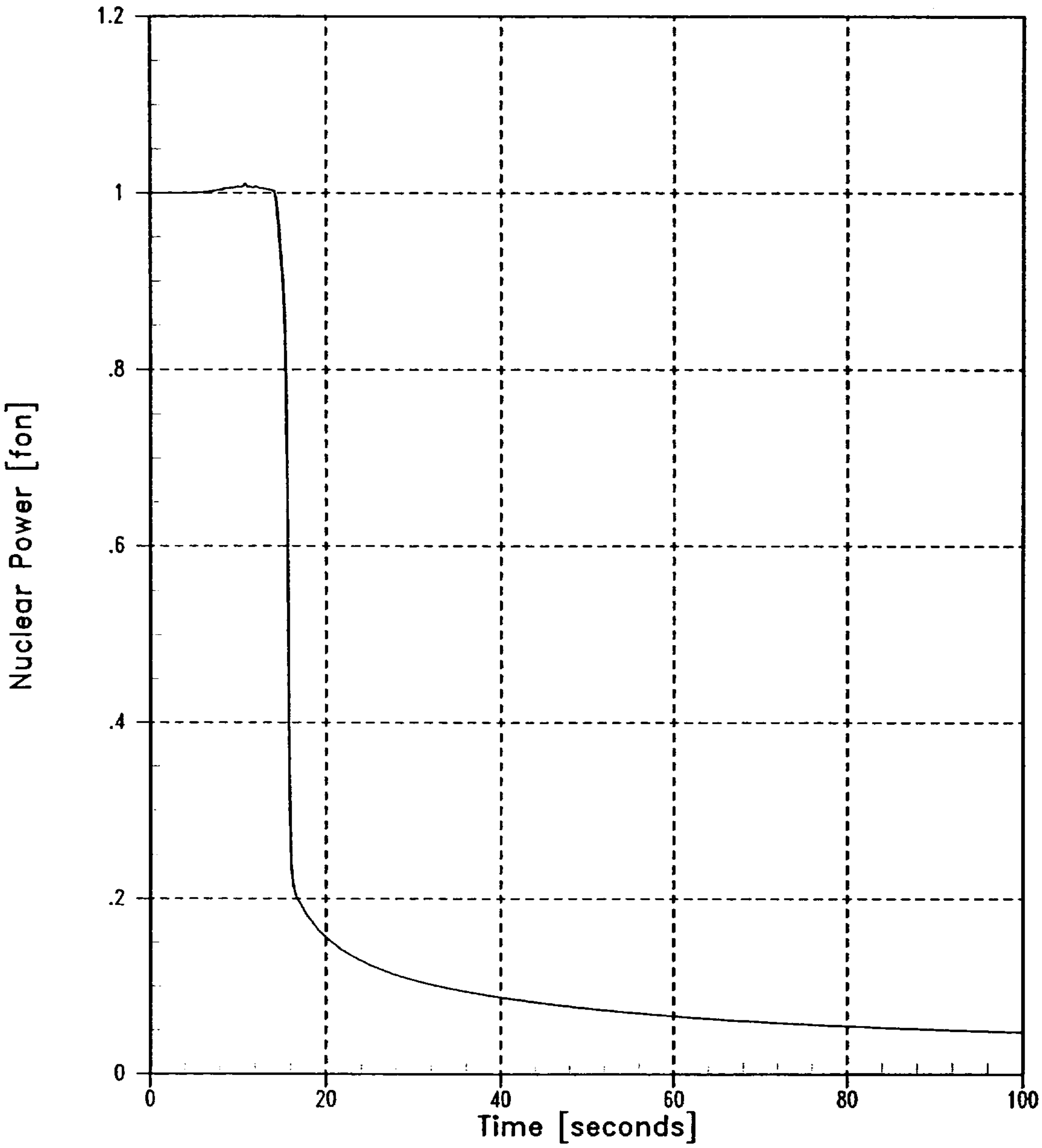

Figure 14.1.9-1

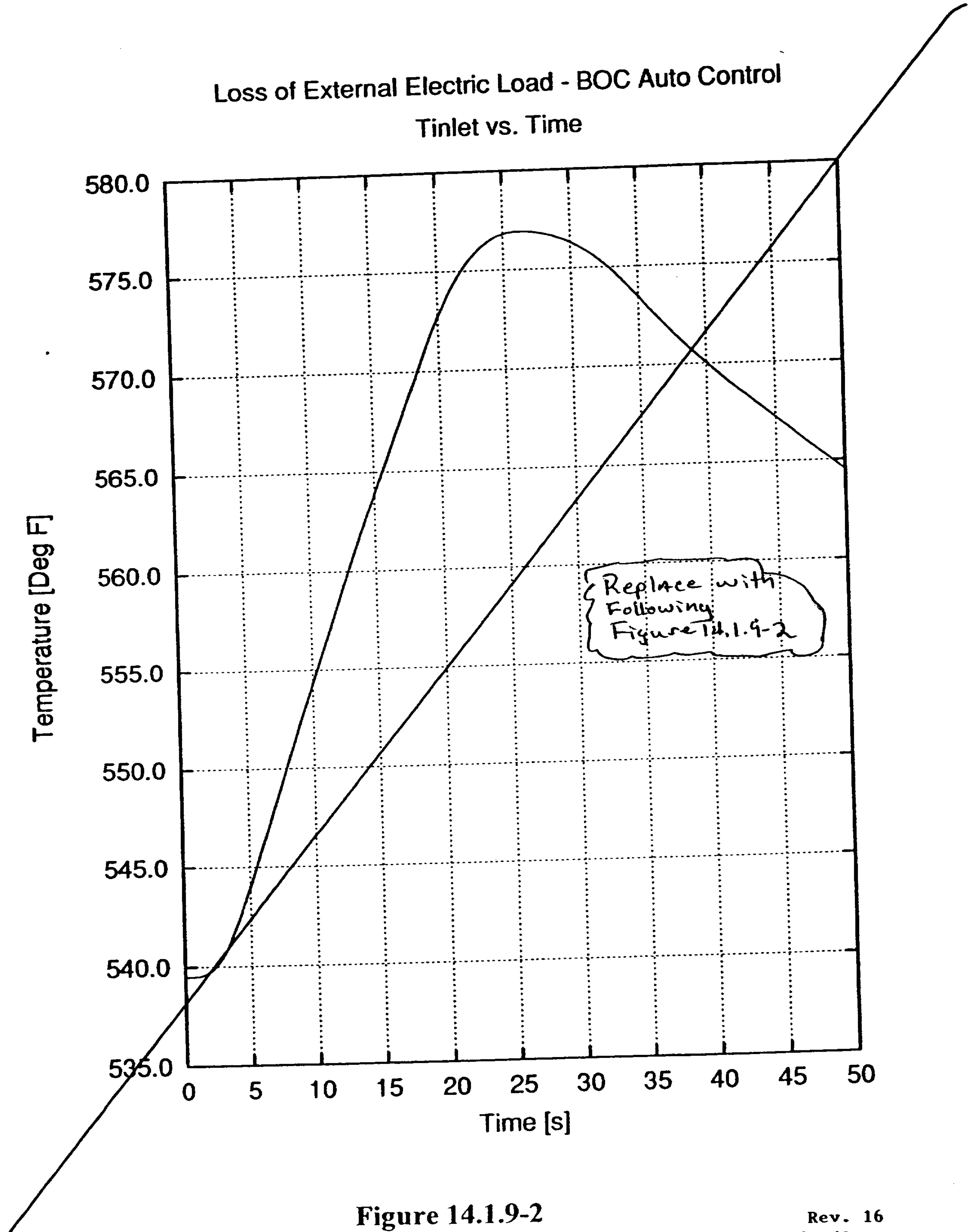

Figure 14.1.9-2

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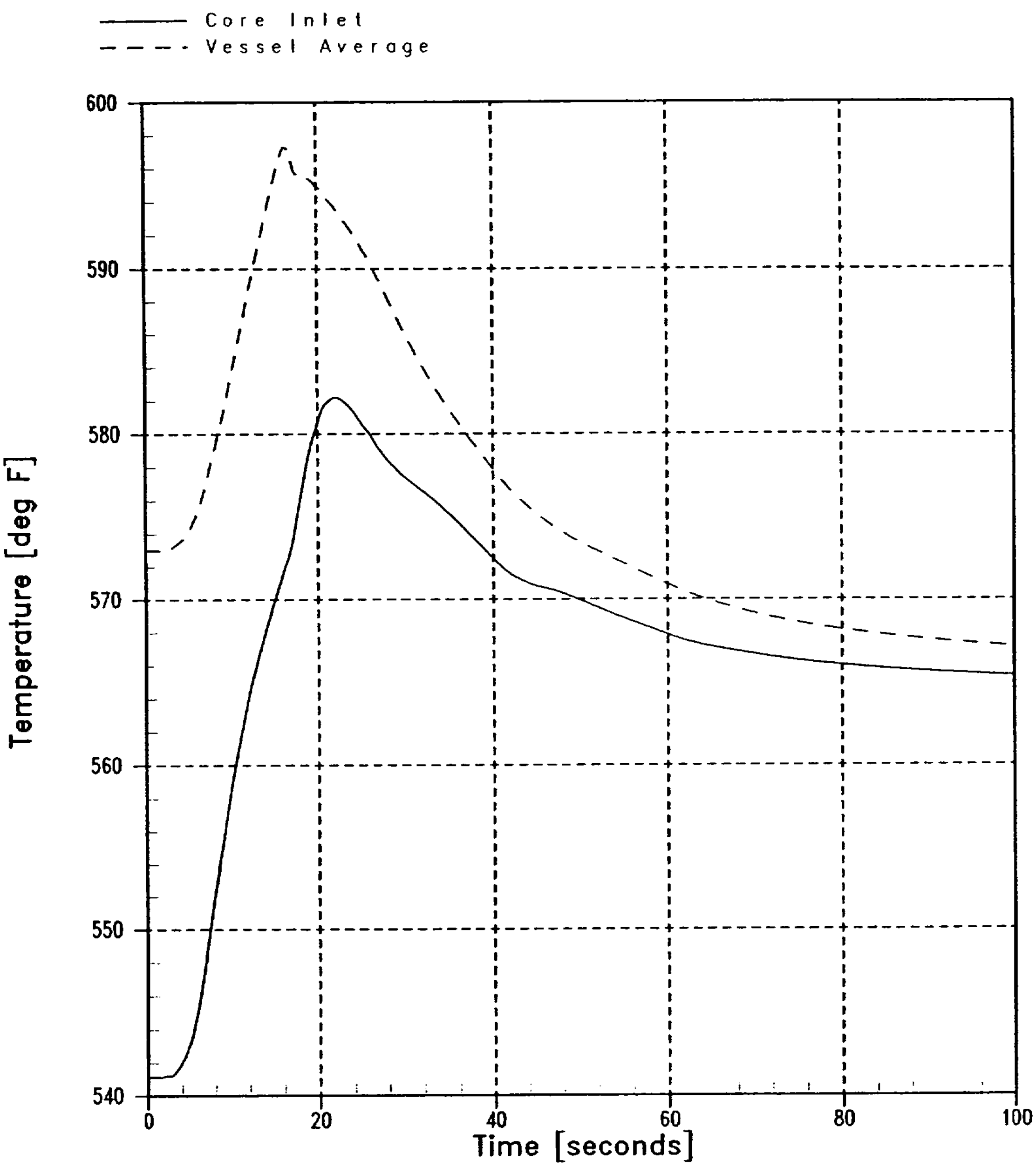
Loss of External Electrical Load With Auto Pressure Control (DNB Case)

Vessel Average and Core Inlet Temperature vs. Time



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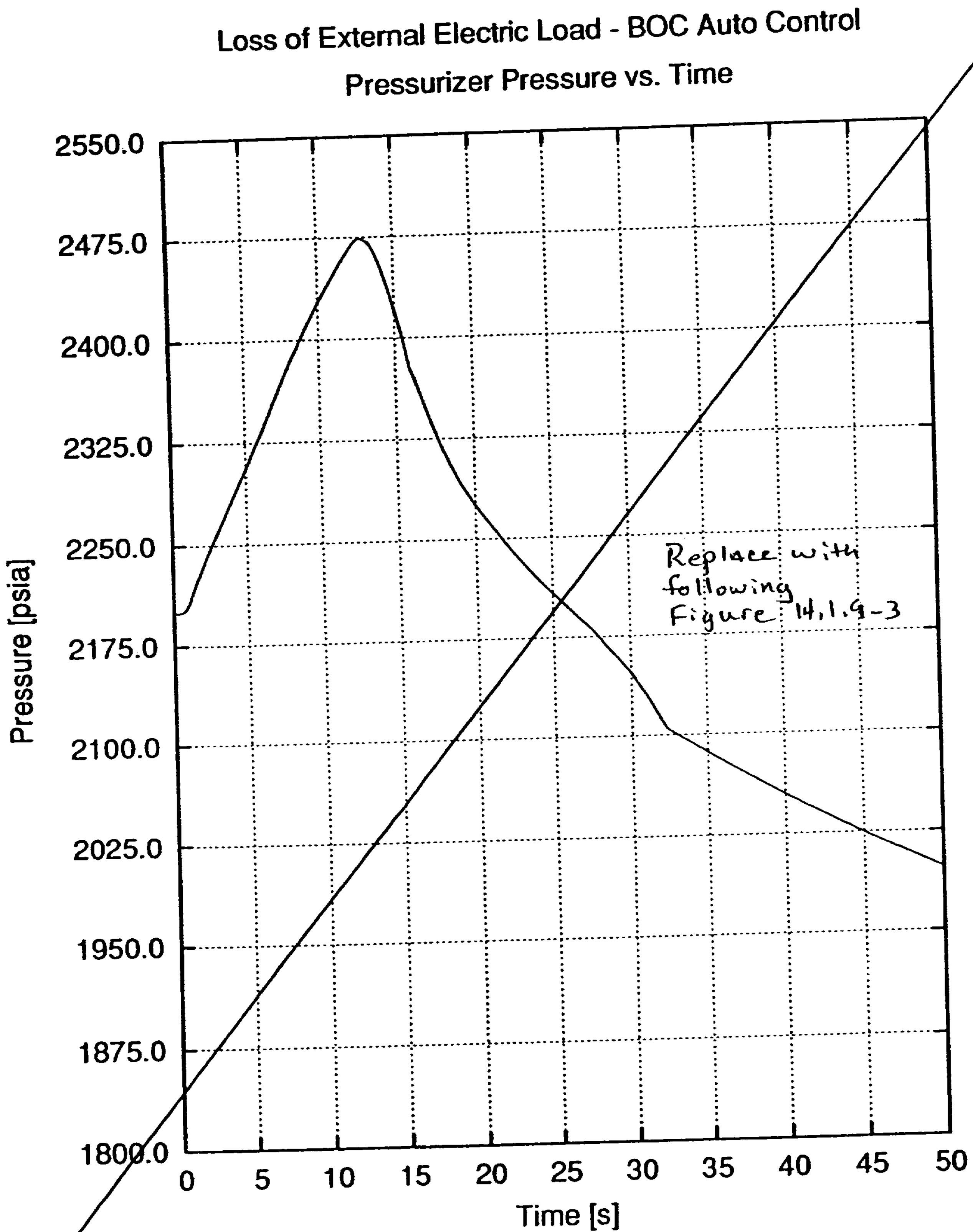

Figure 14.1.9-3

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Loss of External Electrical Load With Auto Pressure Control (DNB Case)

Pressurizer and RCP Exit Pressure vs. Time

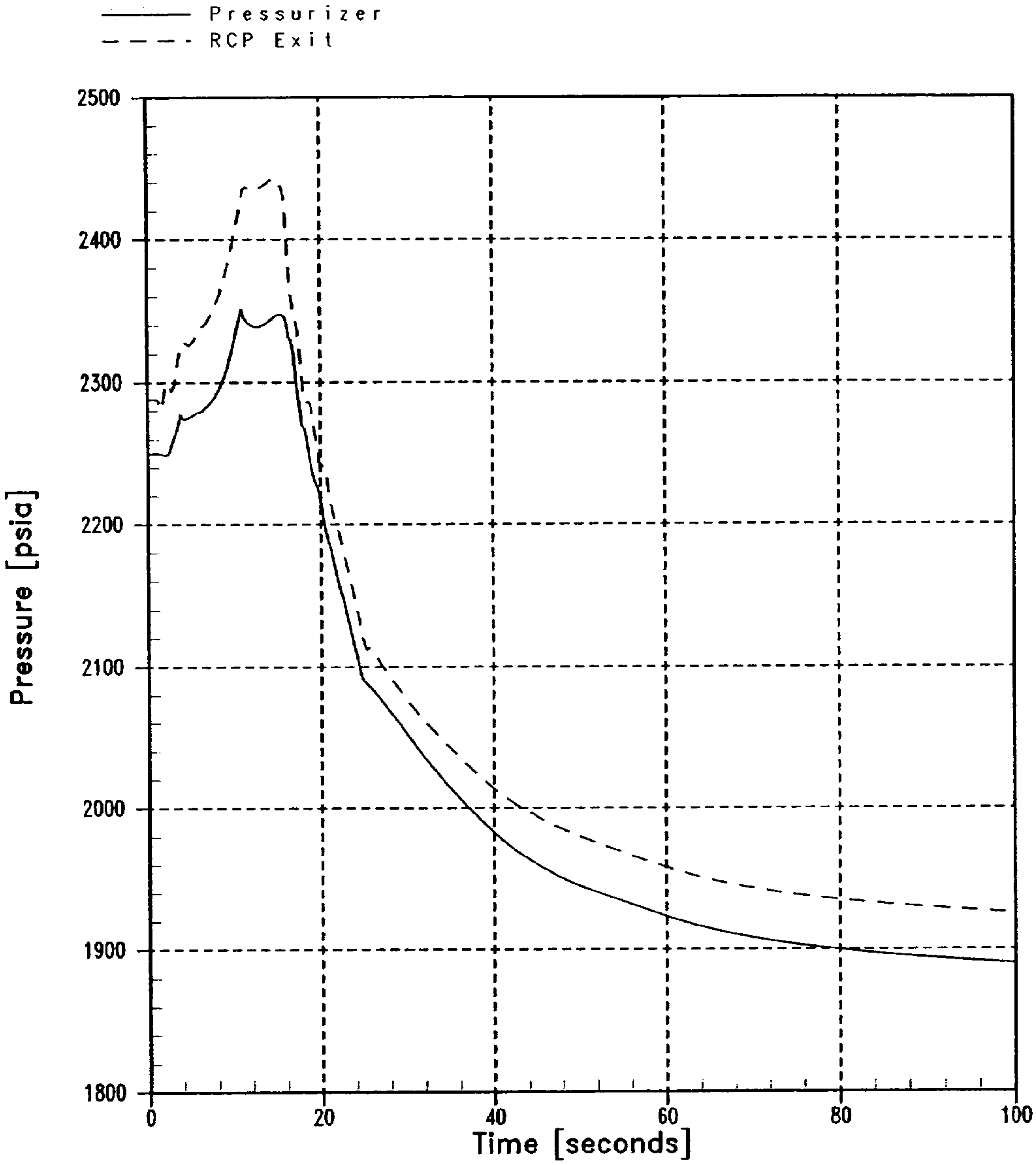

Figure 14.1.9-3

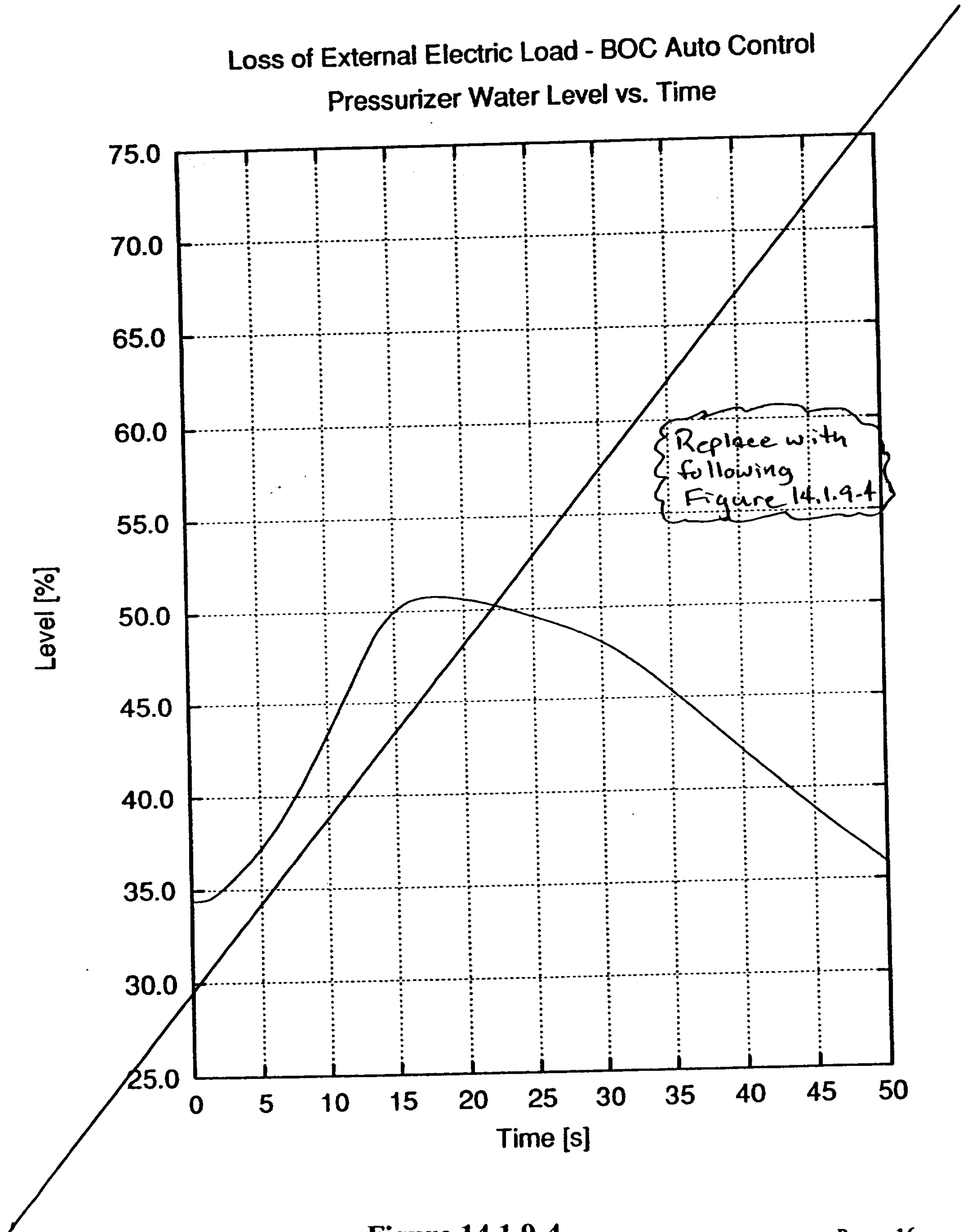

Figure 14.1.9-4

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Loss of External Electrical Load With Auto Pressure Control (DNB Case)
Pressurizer Water Volume vs. Time

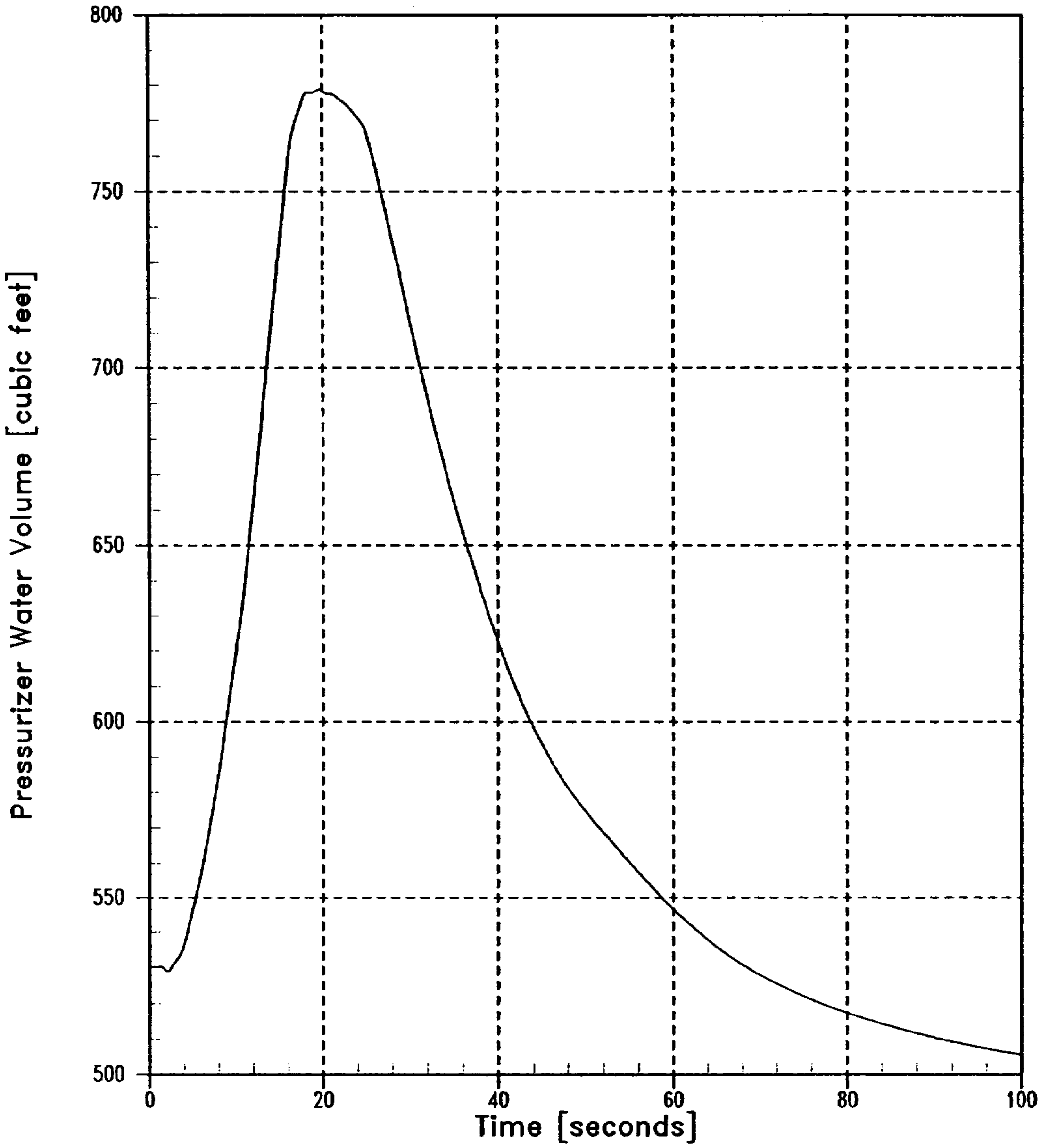


Figure 14.1.9-4

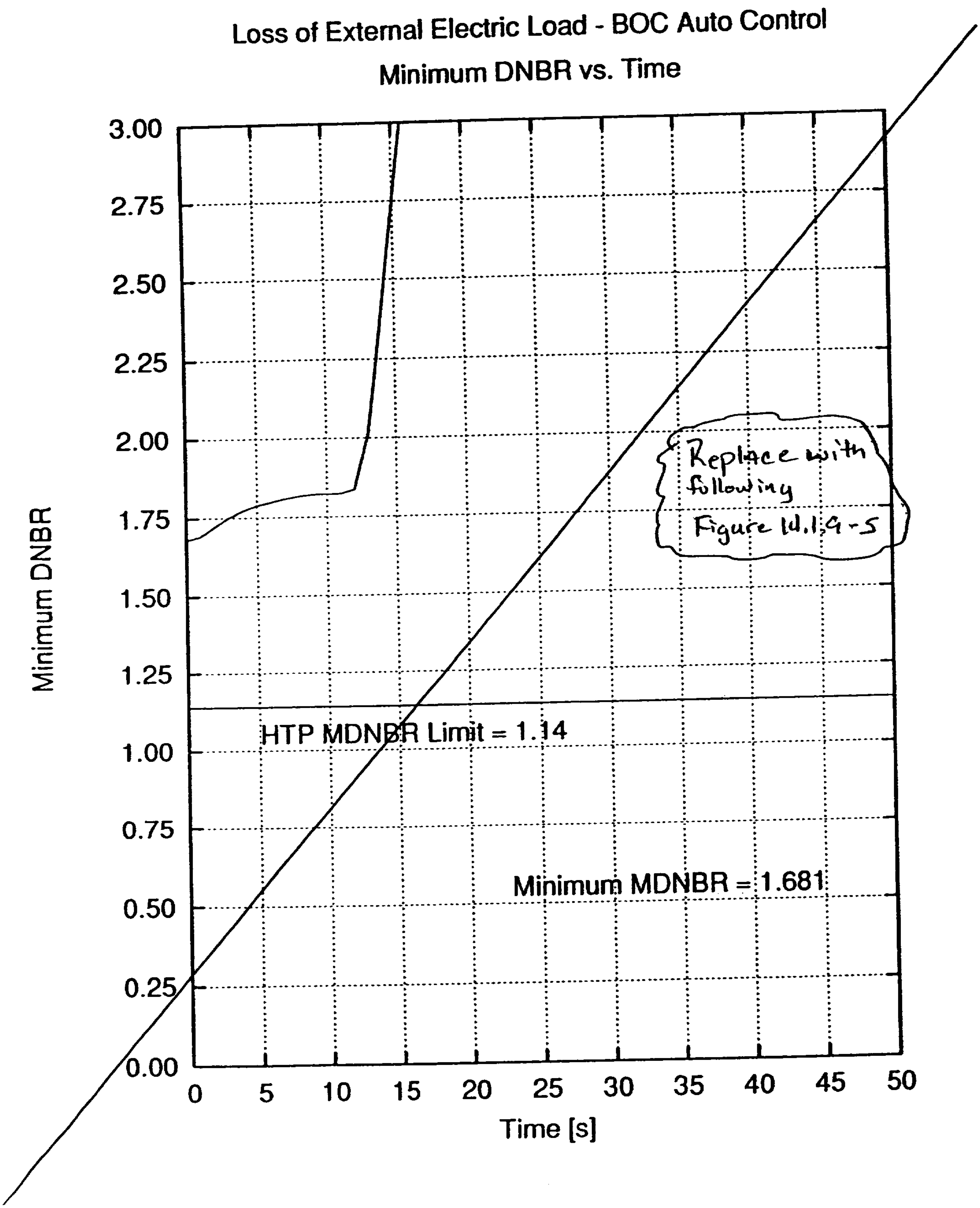
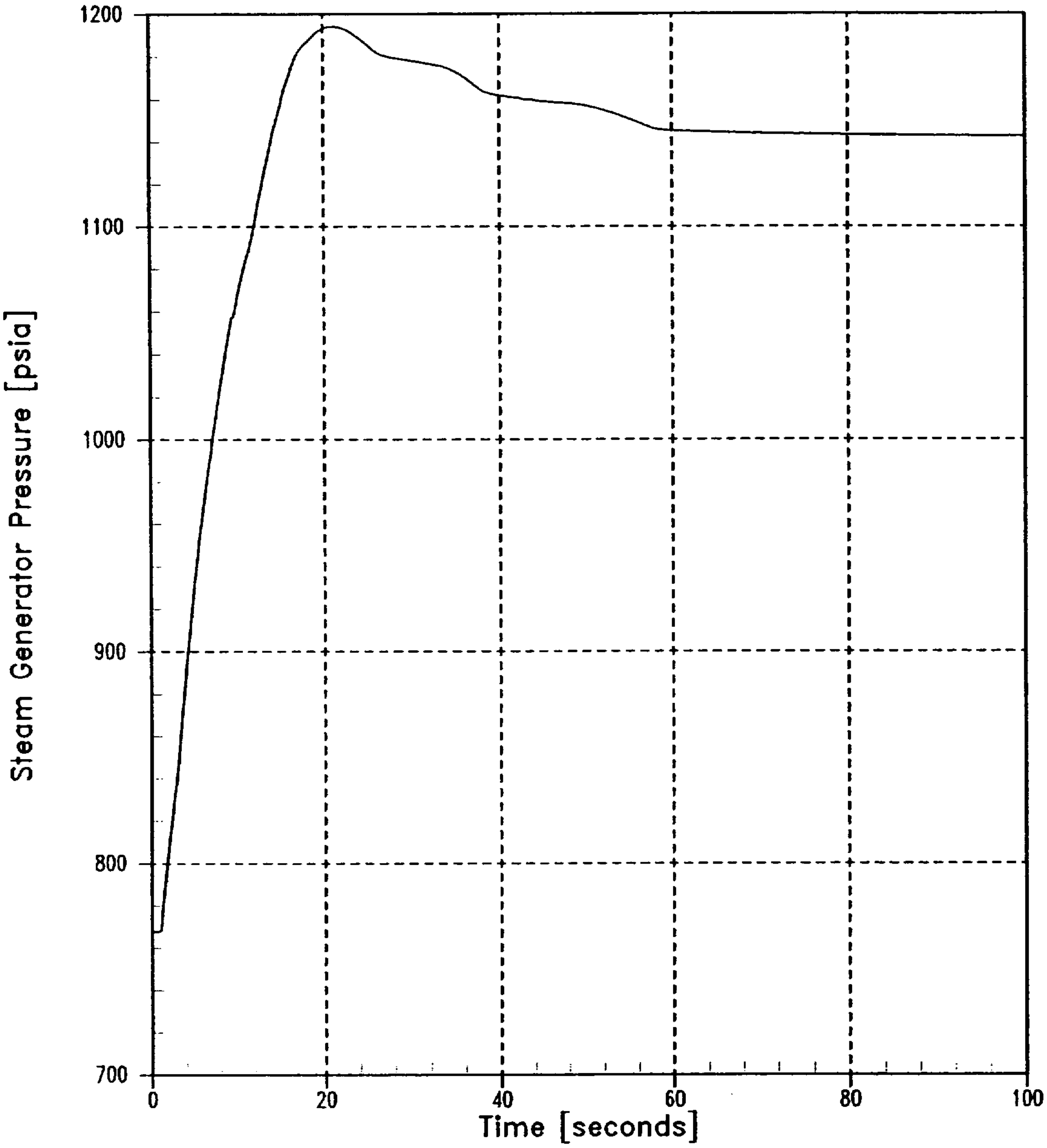


Figure 14.1.9-5

Loss of External Electrical Load With Auto Pressure Control (DNB Case)

Steam Generator Pressure vs. Time



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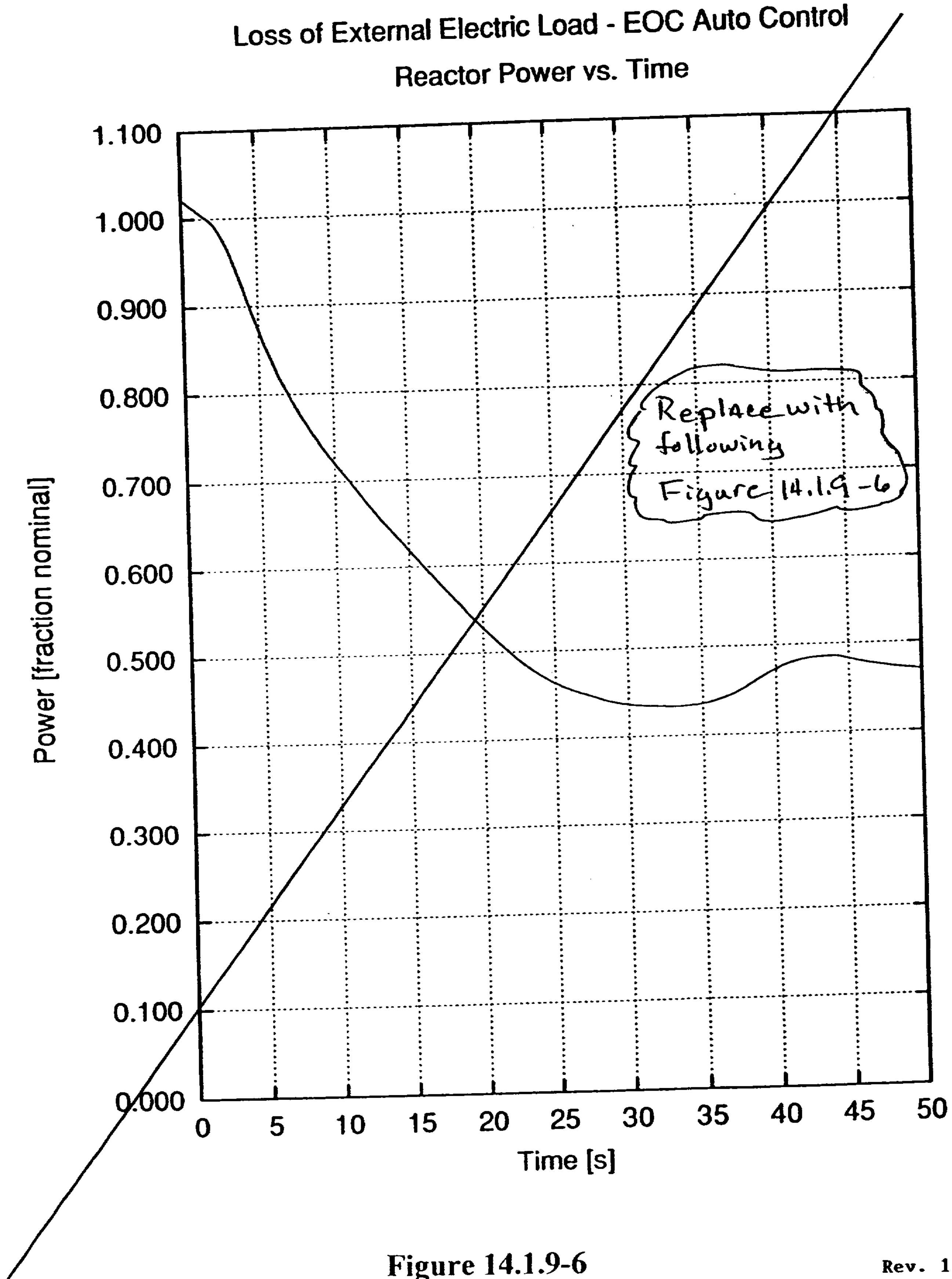
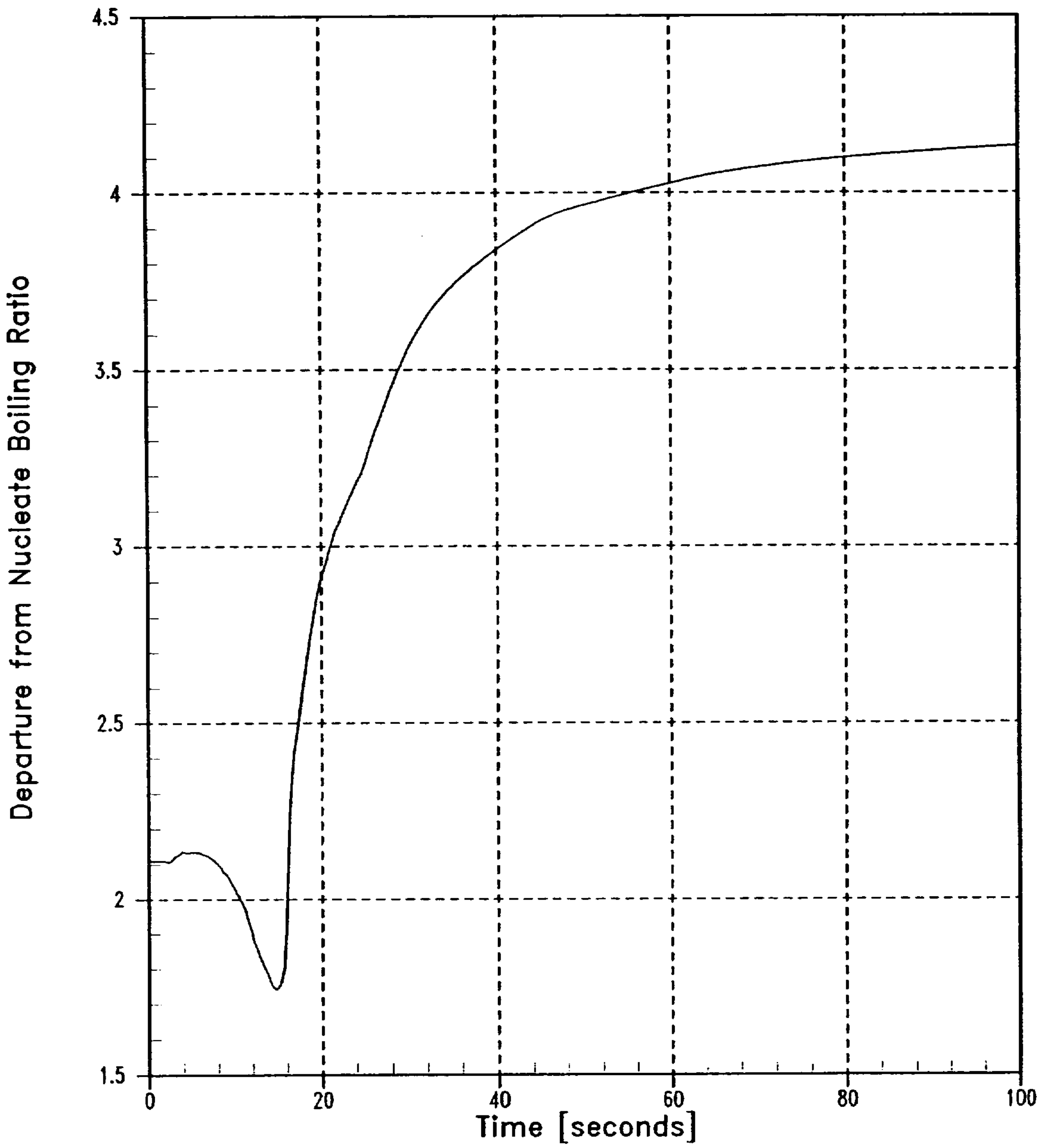


Figure 14.1.9-6

Loss of External Electrical Load With Auto Pressure Control (DNB Case)

DNBR vs. Time



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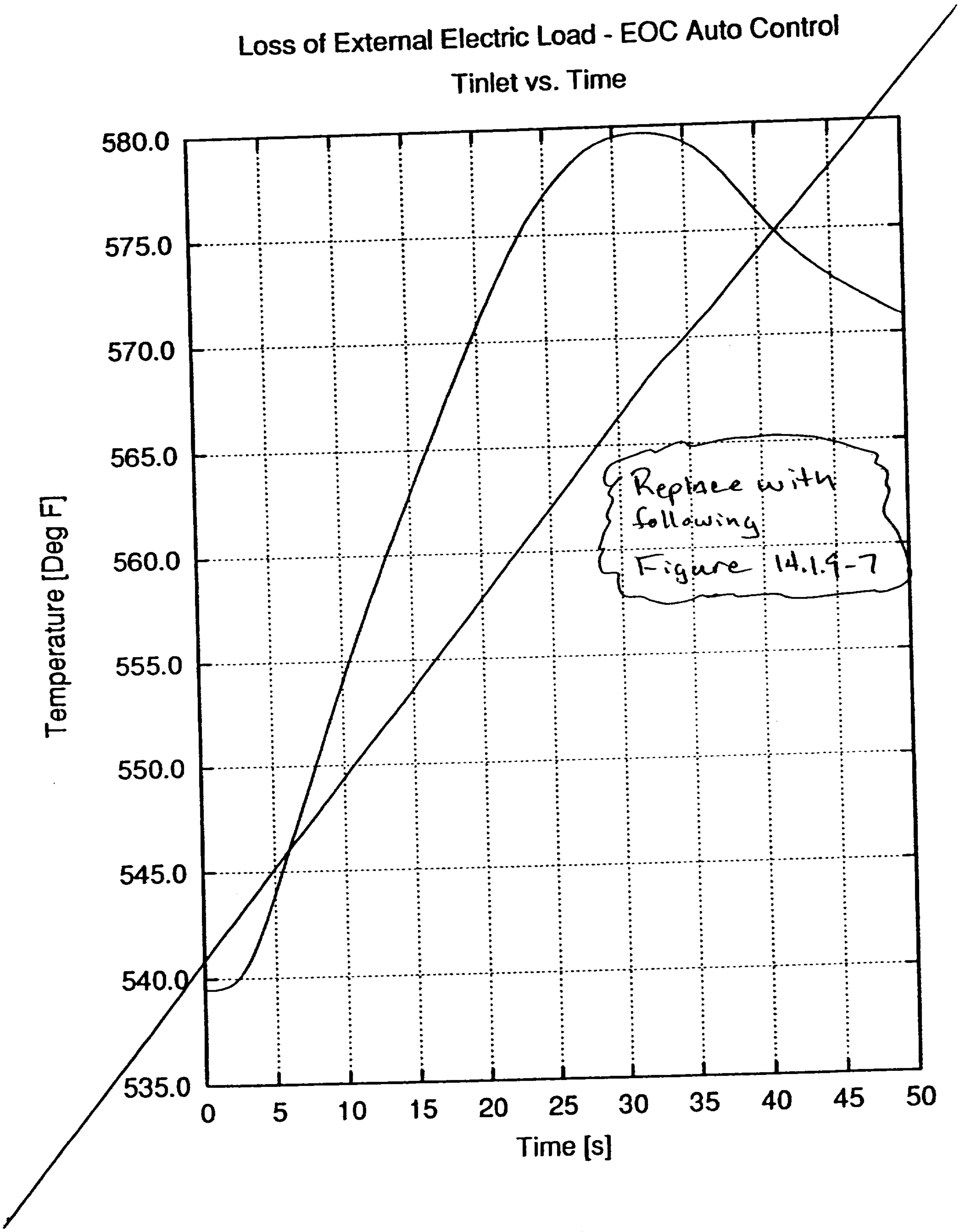

Figure 14.1.9-7

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Loss of External Electrical Load Without Auto Pressure Control (RCS Overpressure)

Nuclear Power vs. Time

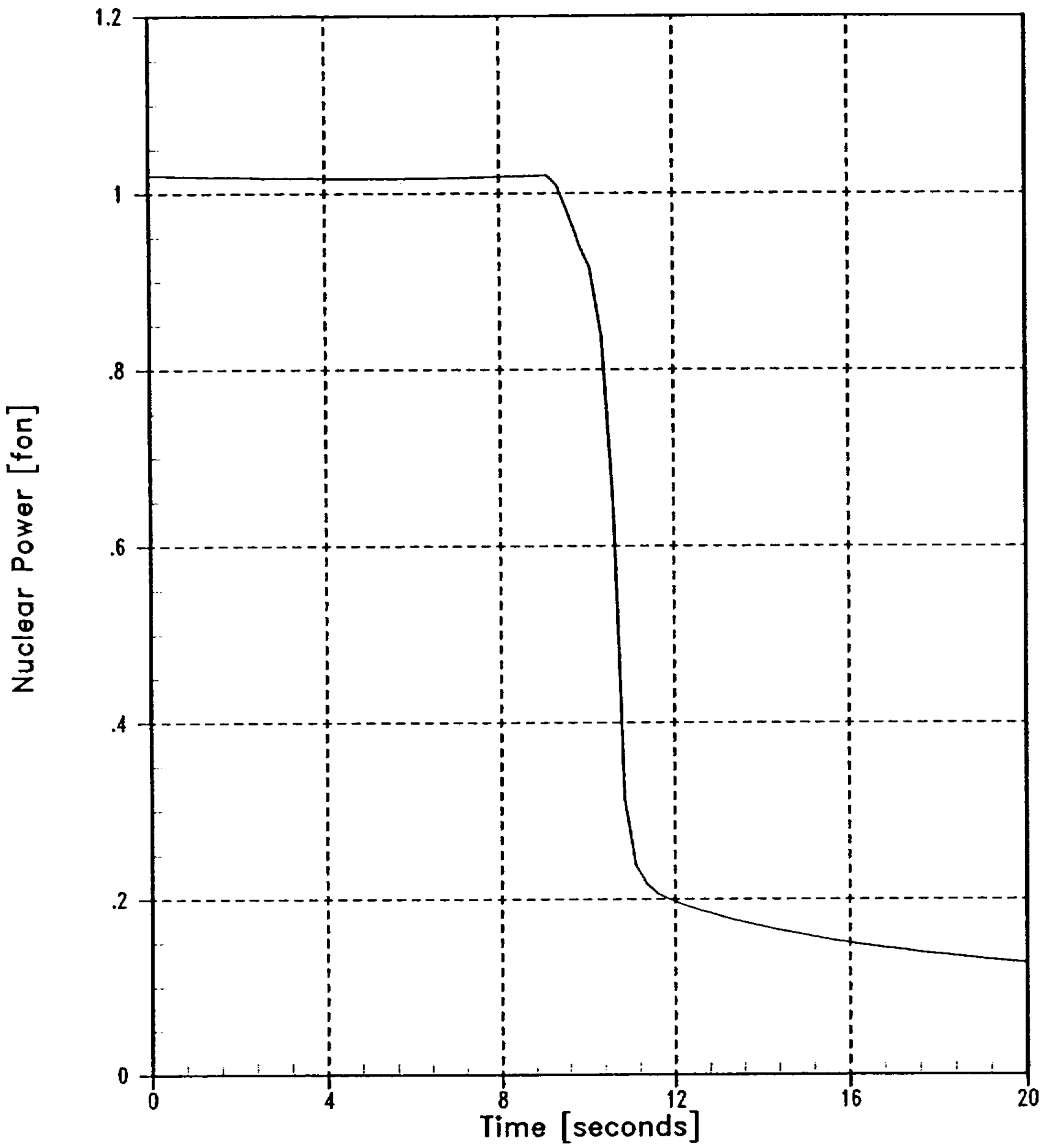

Figure 14.1.9-7

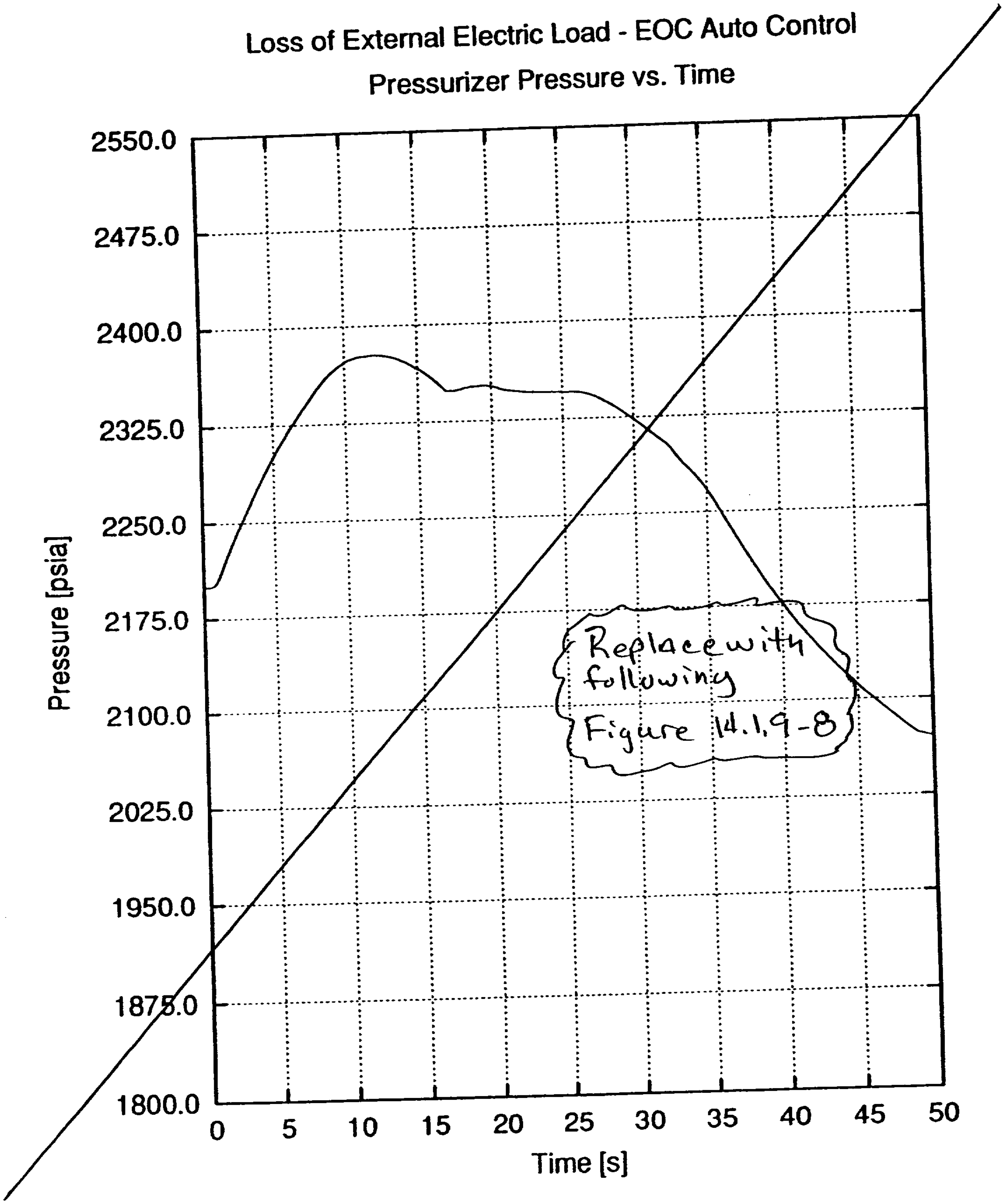

Figure 14.1.9-8

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Loss of External Electrical Load Without Auto Pressure Control (RCS Overpressure)

Vessel Average and Core Inlet Temperature vs. Time

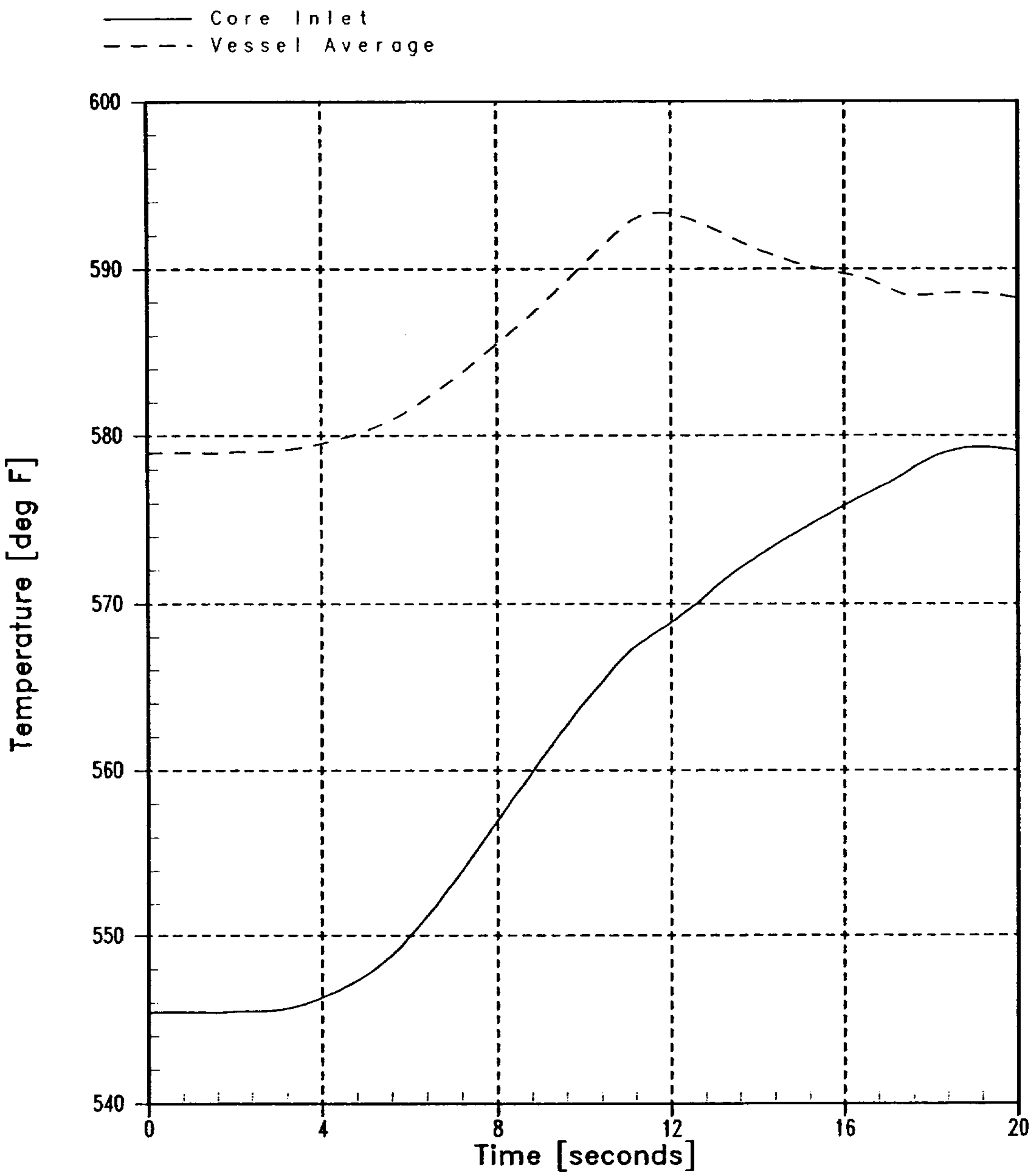

Figure 14.1.9-8

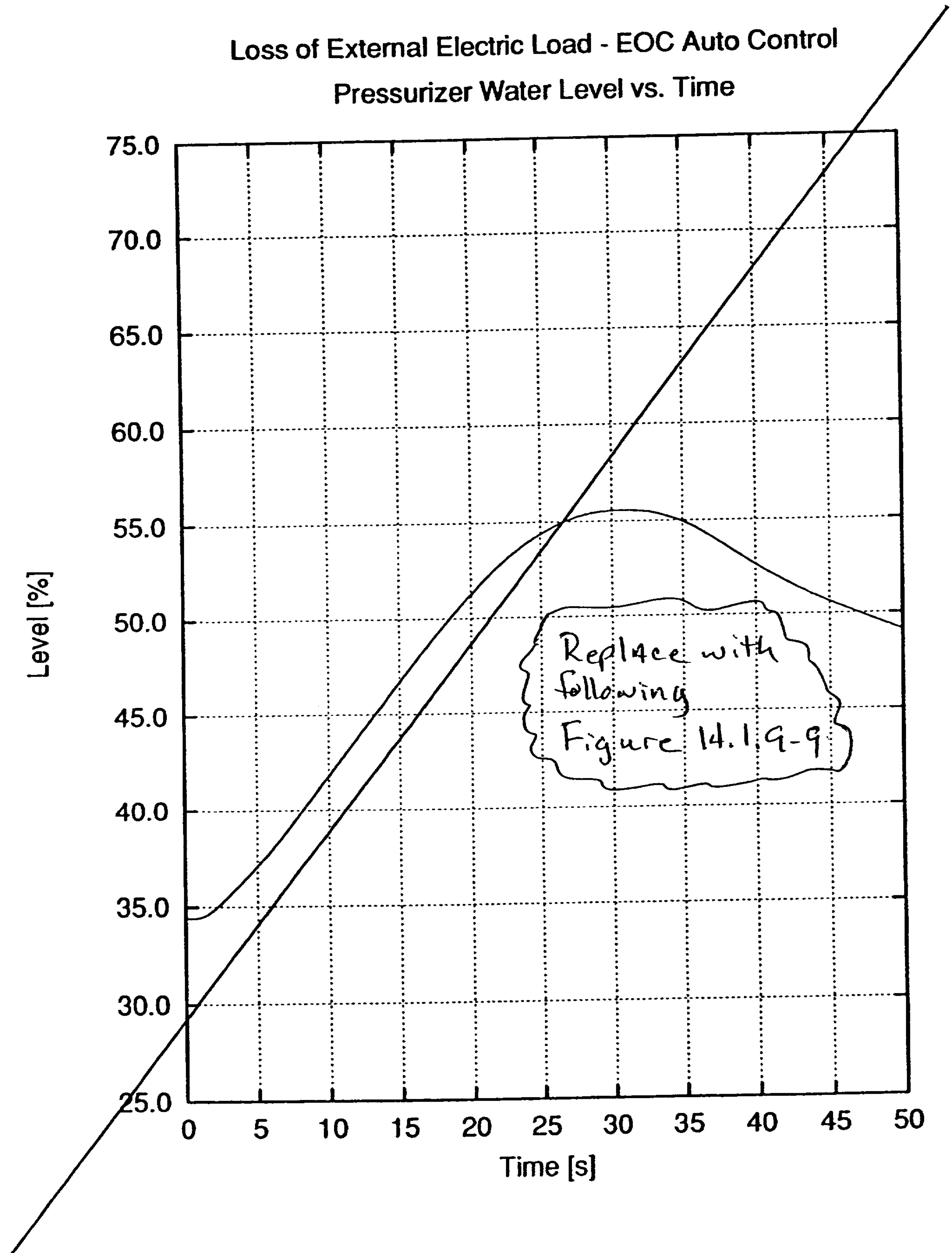

Figure 14.1.9-9

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Loss of External Electrical Load Without Auto Pressure Control (RCS Overpressure)

Pressurizer and RCP Exit Pressure vs. Time

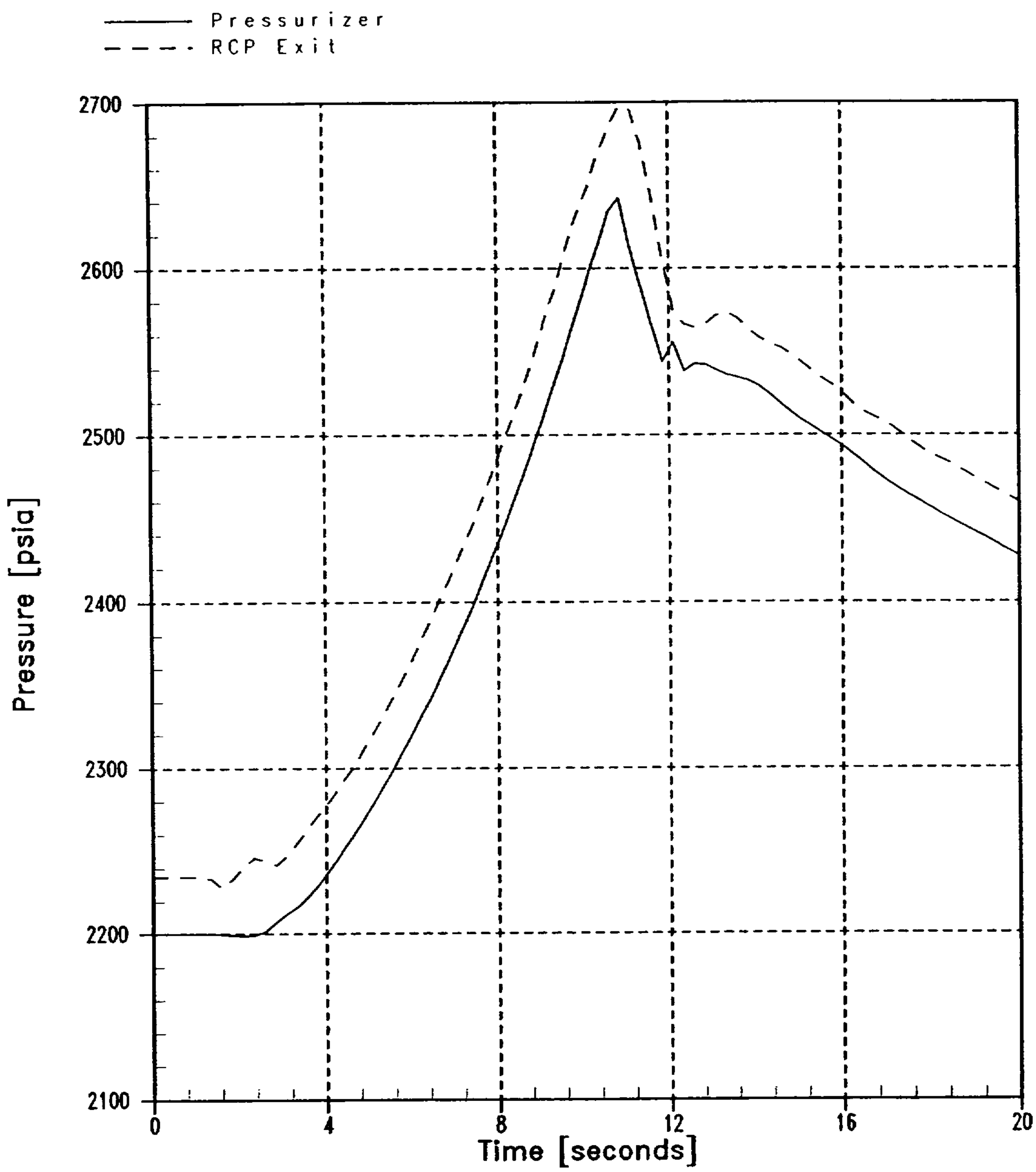

Figure 14.1.9-9

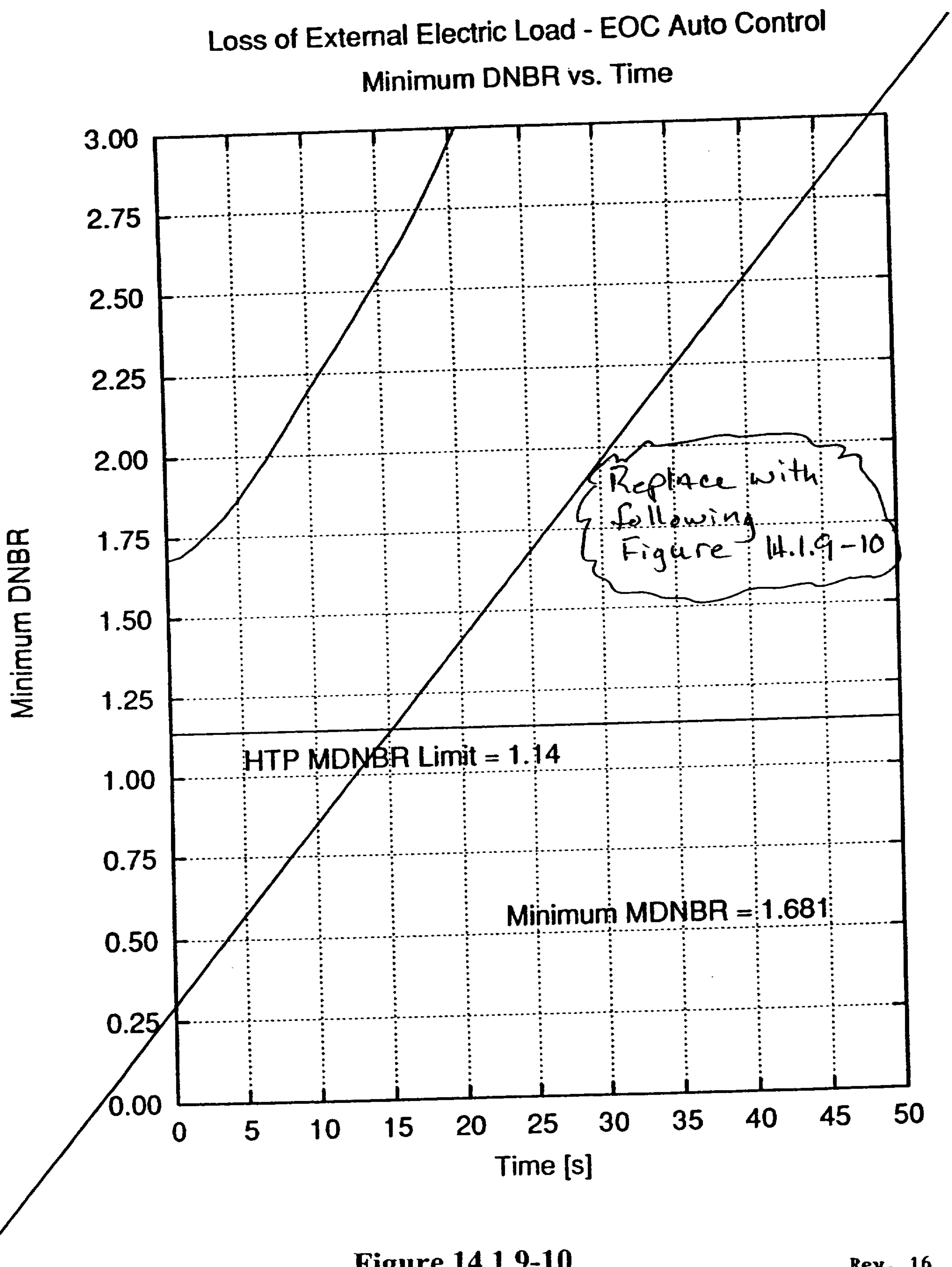

Figure 14.1.9-10

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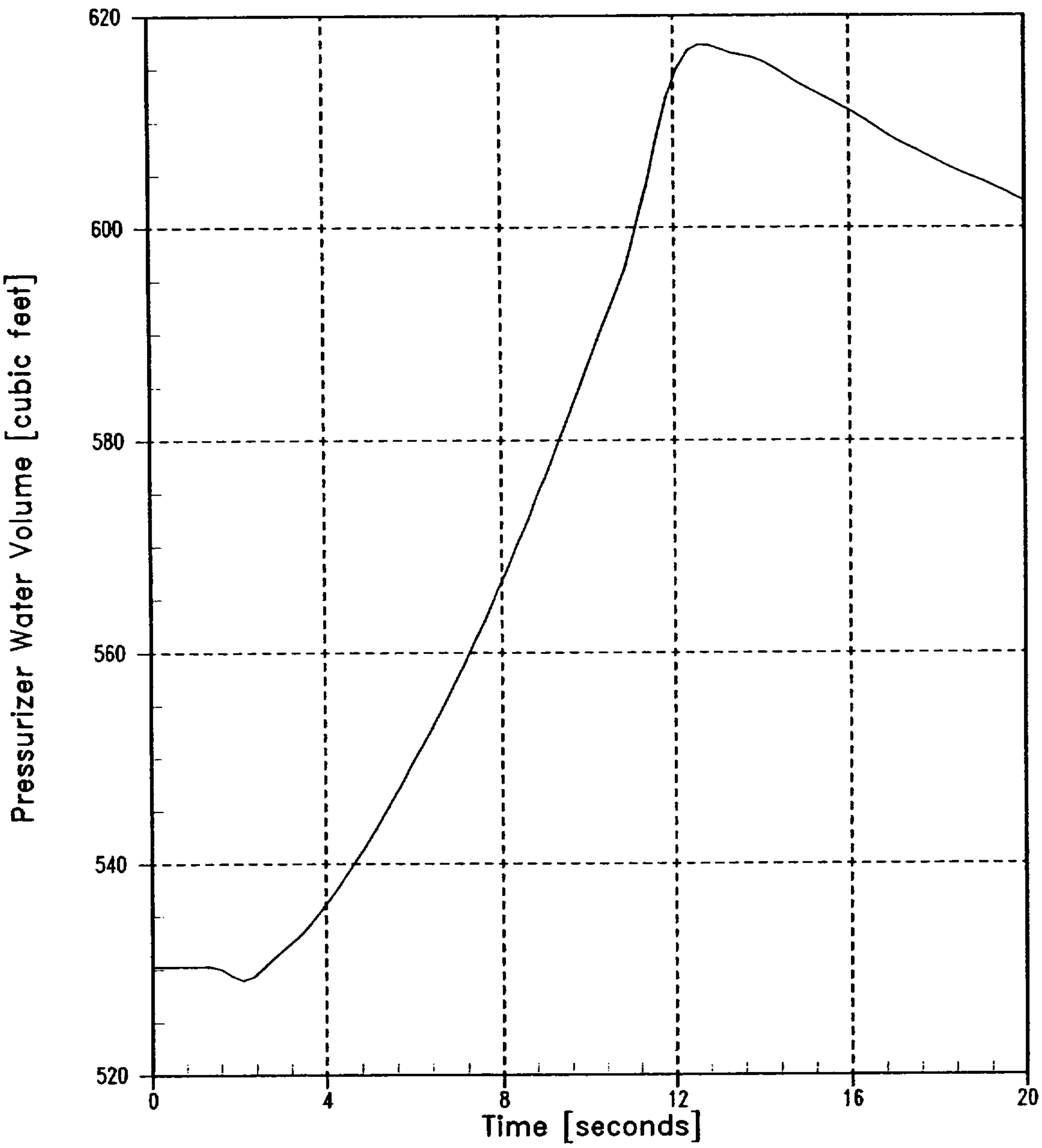
Loss of External Electrical Load Without Auto Pressure Control (RCS Overpressure)

Pressurizer Water Volume vs. Time



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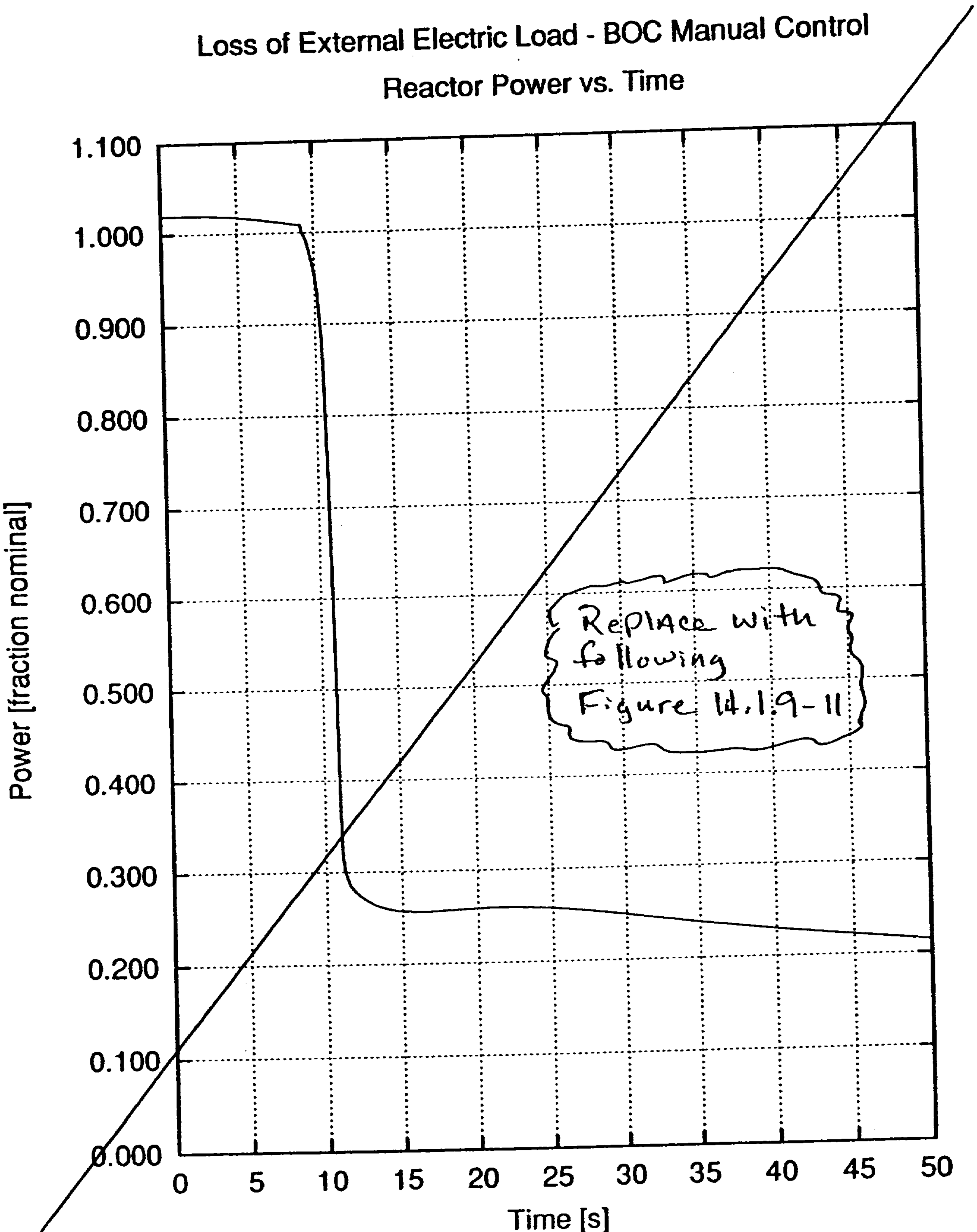

Figure 14.1.9-11

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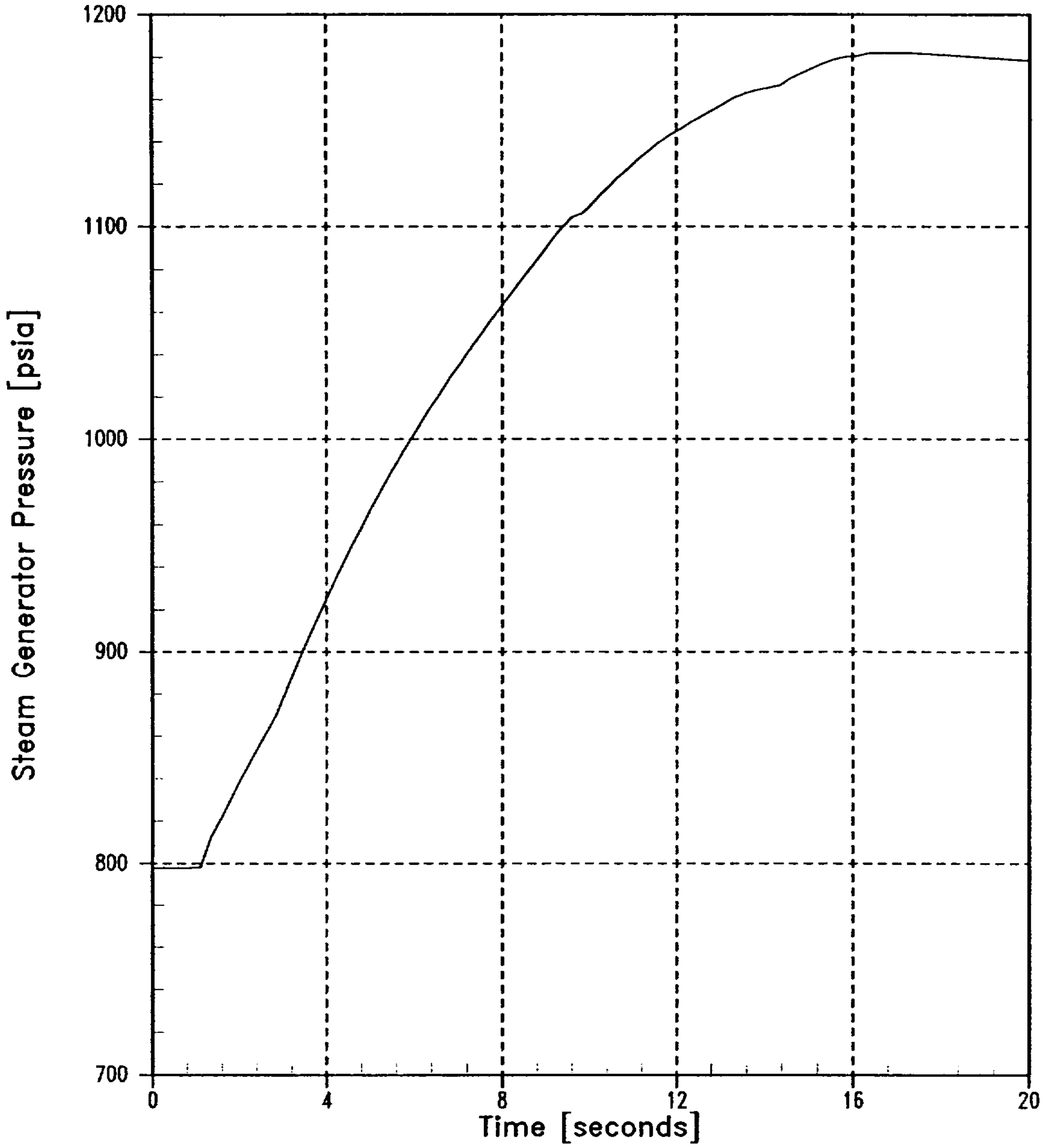
Loss of External Electrical Load Without Auto Pressure Control (RCS Overpressure)

Steam Generator Pressure vs. Time



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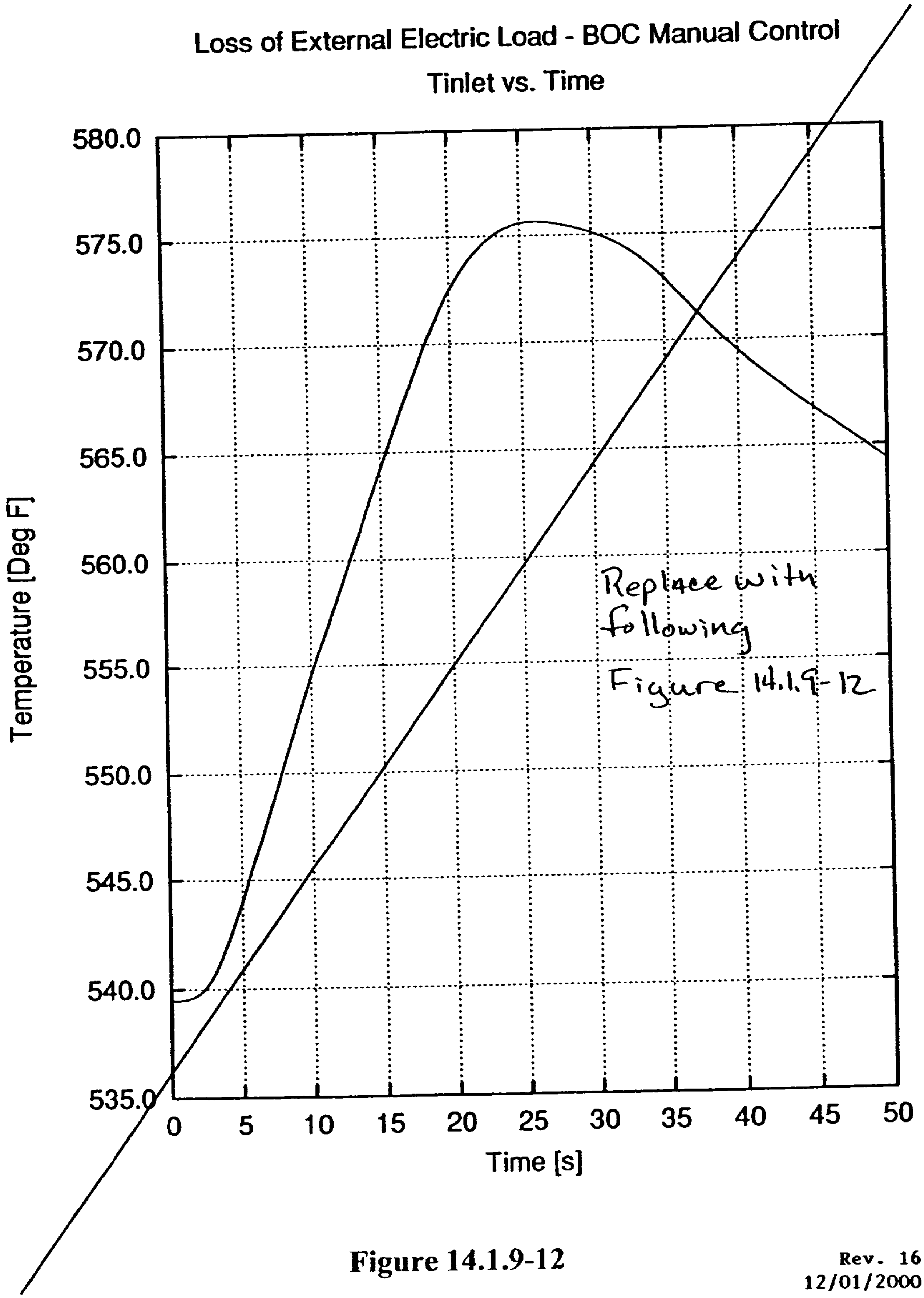

Figure 14.1.9-12

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Loss of External Electrical Load With Auto Pressure Control (MSS Overpressure)

Nuclear Power vs. Time

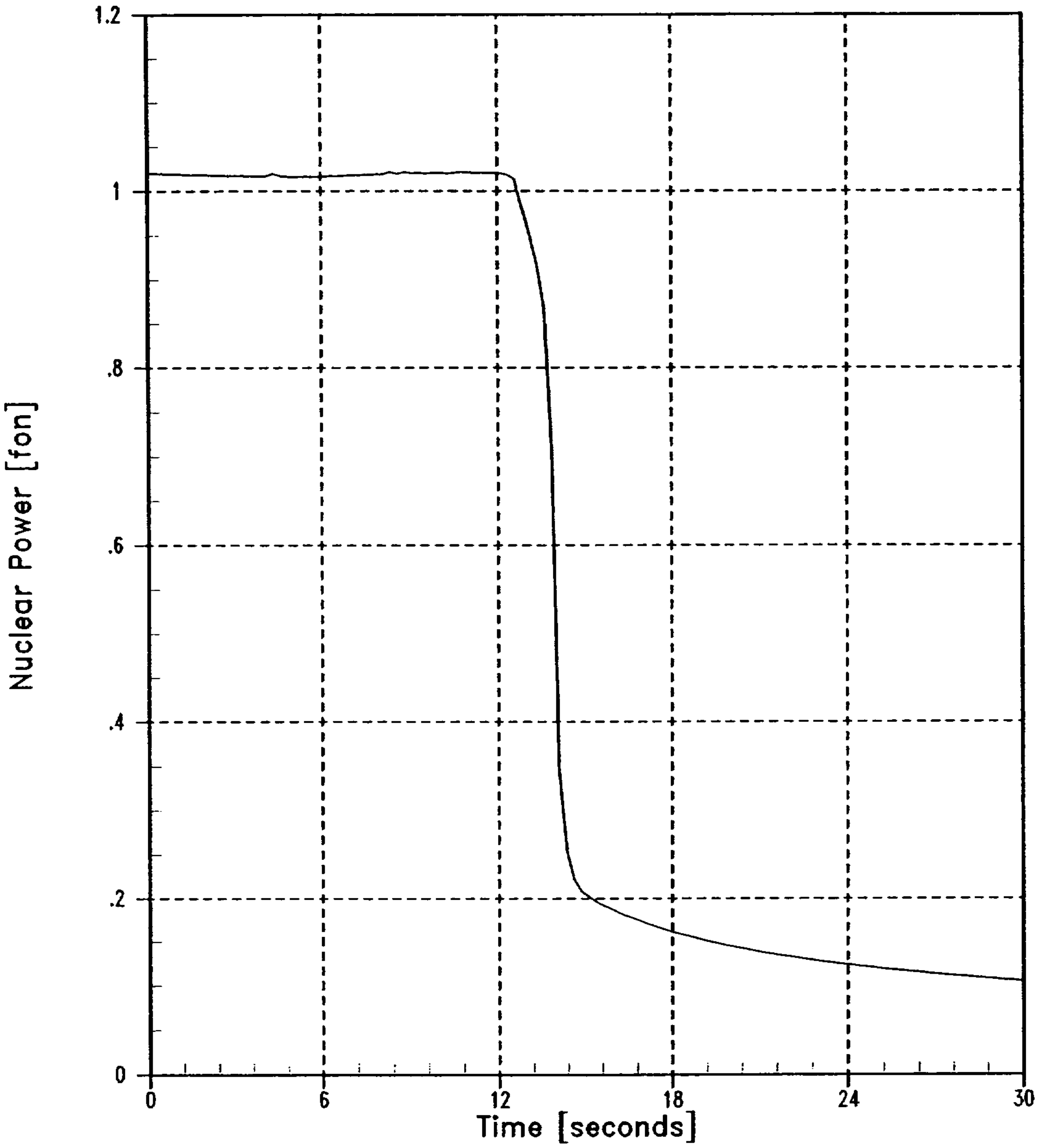

Figure 14.1.9-12

Loss of External Electric Load - BOC Manual Control

Pressurizer Pressure vs. Time

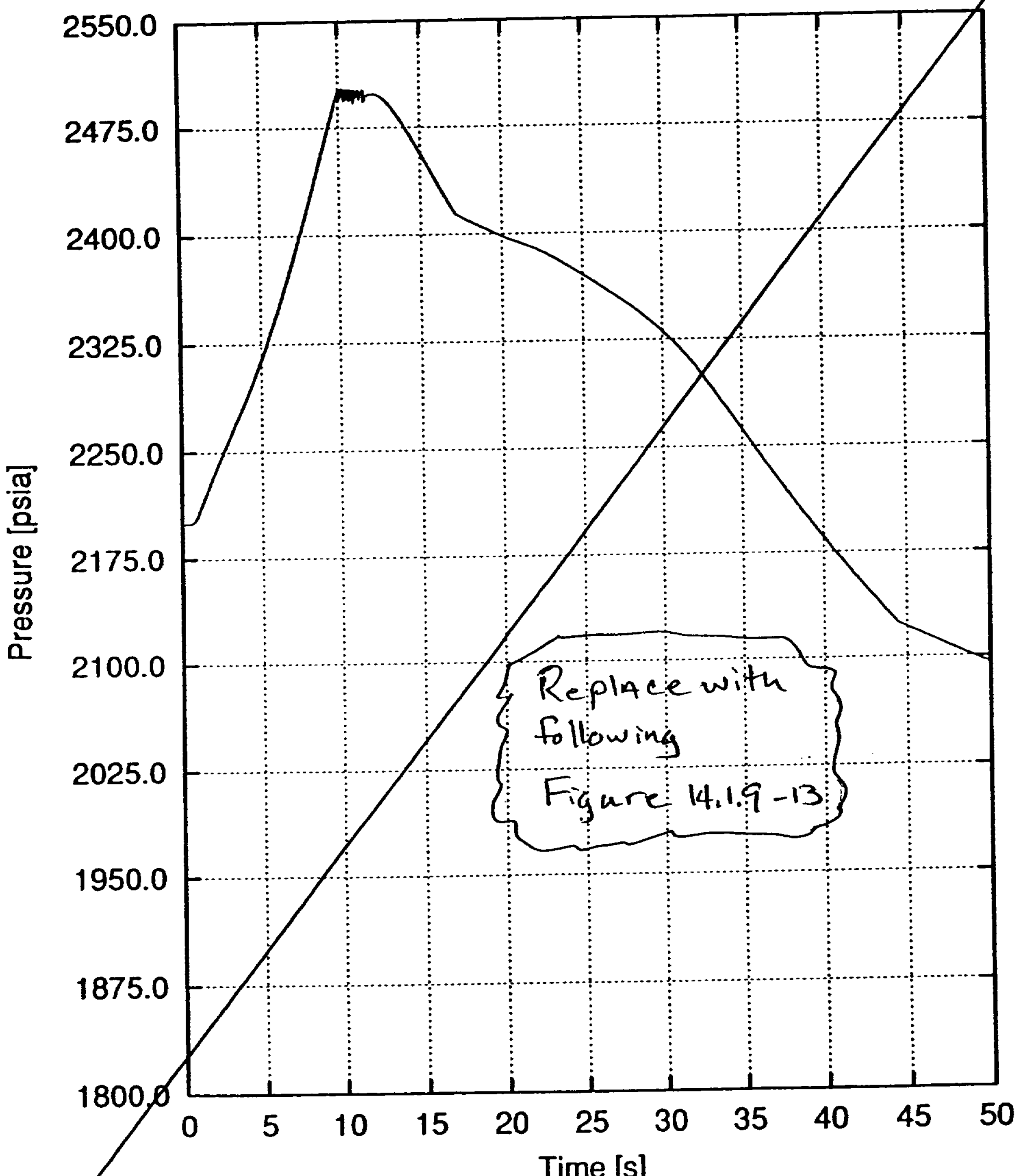

Figure 14.1.9-13

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Loss of External Electrical Load With Auto Pressure Control (MSS Overpressure)

Vessel Average and Core Inlet Temperature vs. Time

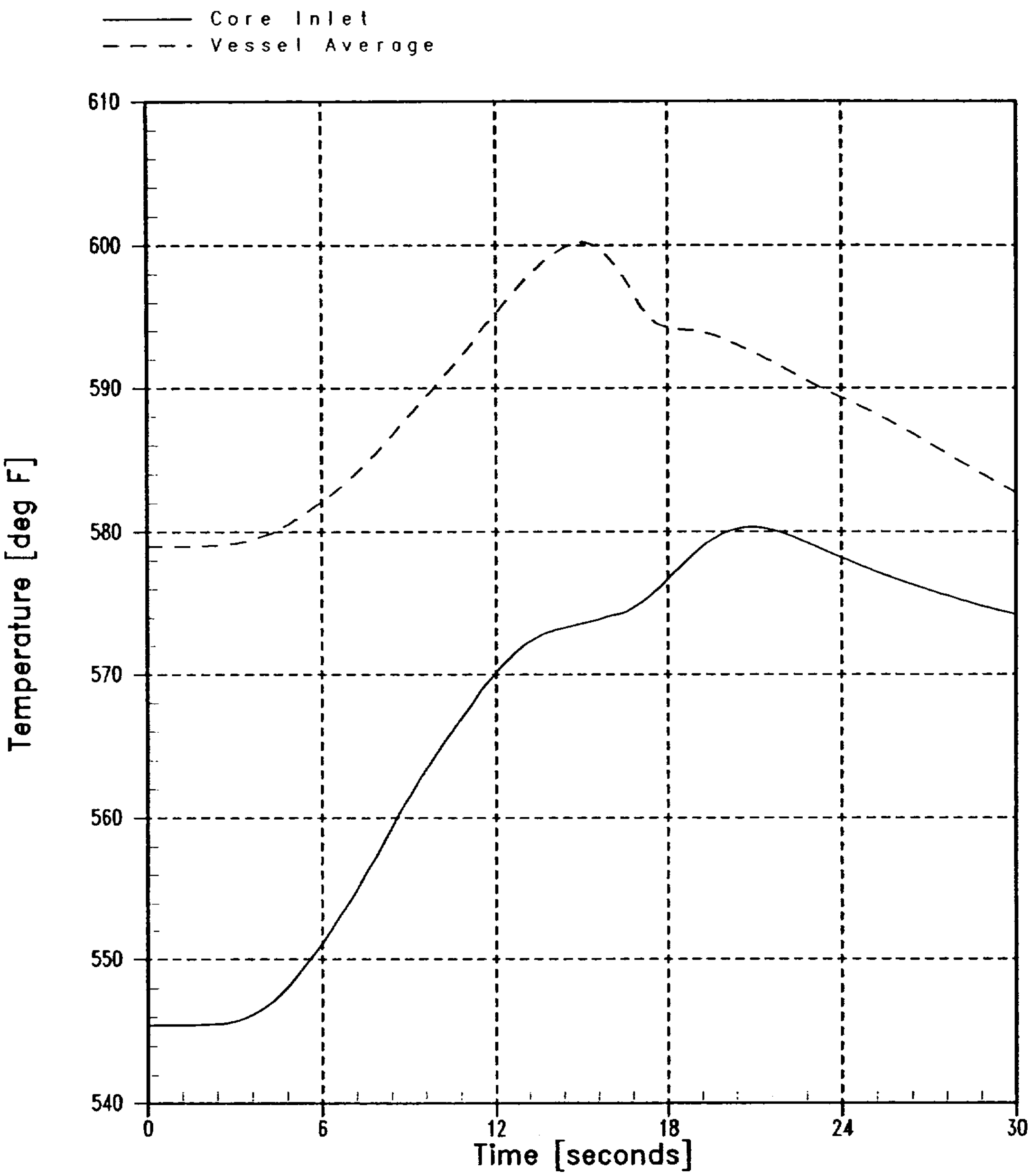

Figure 14.1.9-13

Loss of External Electric Load - BOC Manual Control

Pressurizer Water Level vs. Time

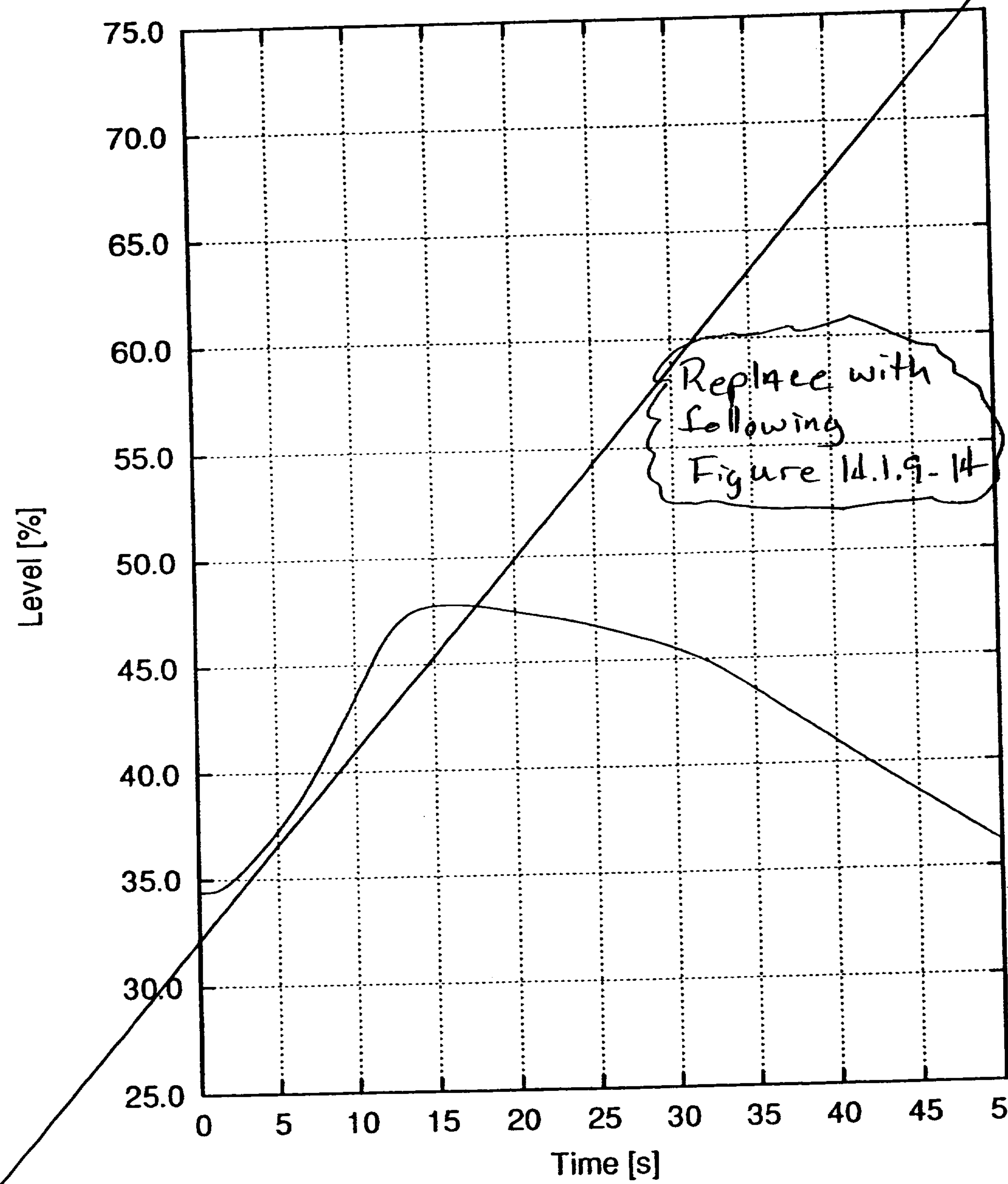

Figure 14.1.9-14

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Loss of External Electrical Load With Auto Pressure Control (MSS Overpressure)
Pressurizer and RCP Exit Pressure vs. Time

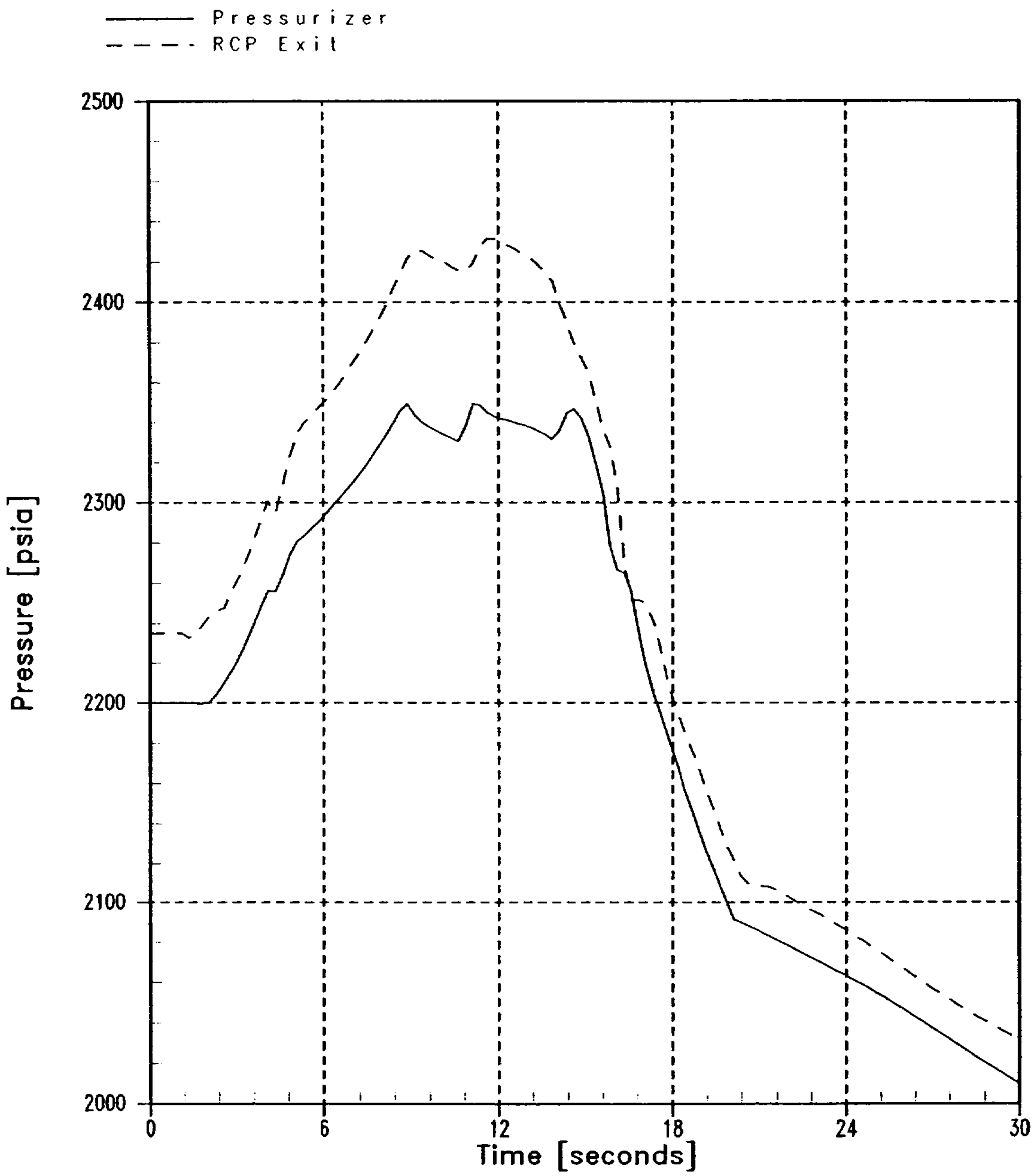


Figure 14.1.9-14

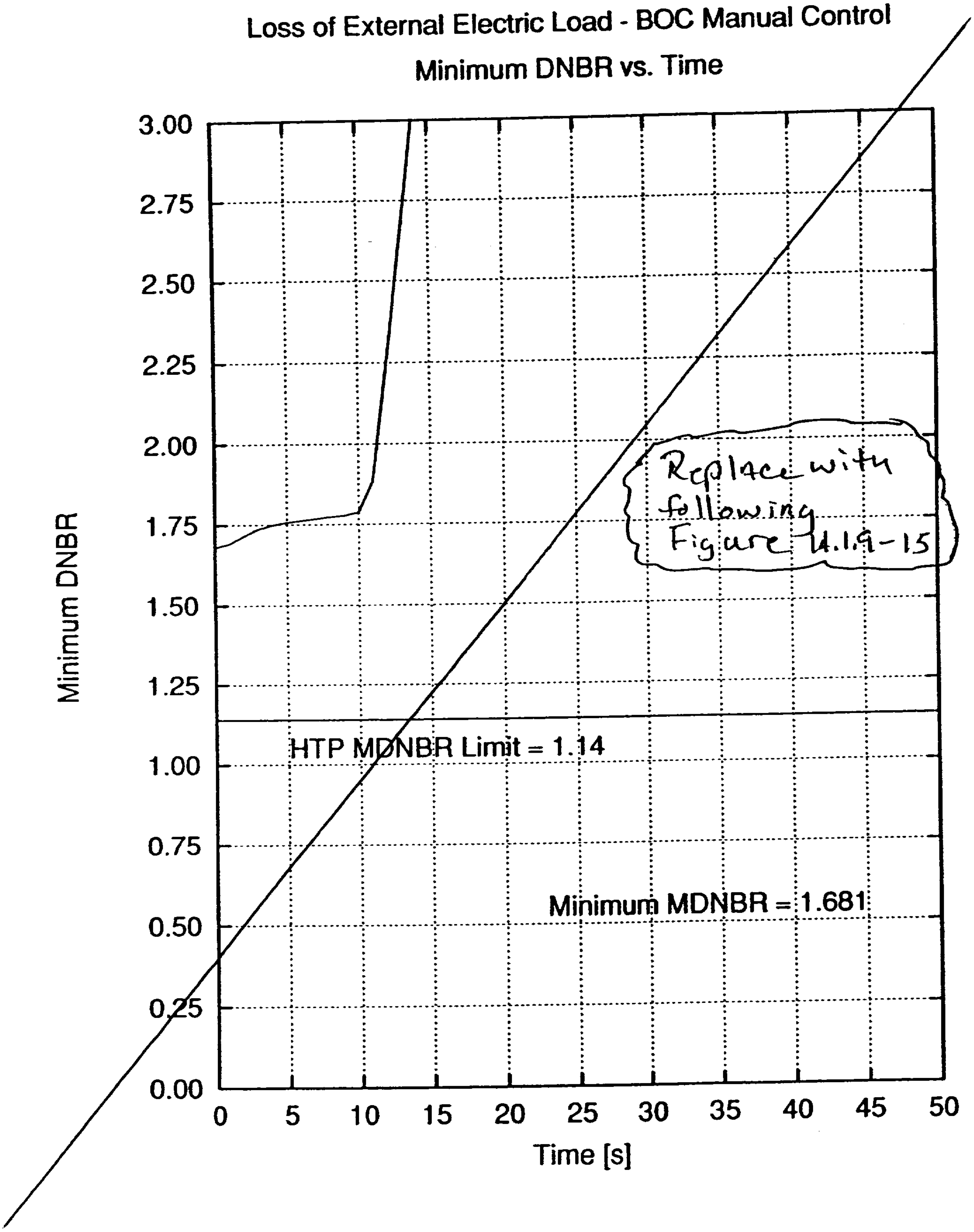
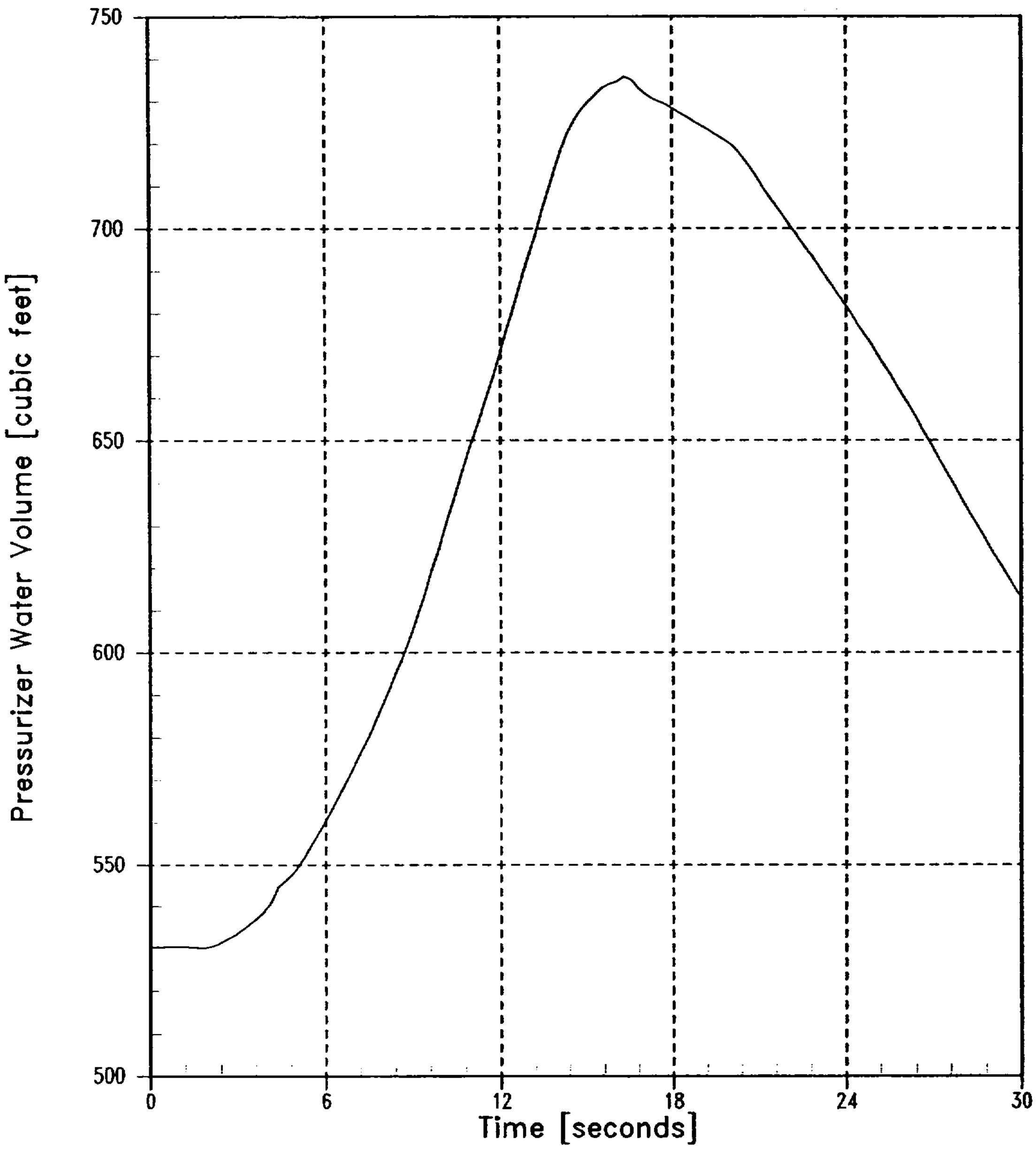


Figure 14.1.9-15

Loss of External Electrical Load With Auto Pressure Control (MSS Overpressure)
Pressurizer Water Volume vs. Time



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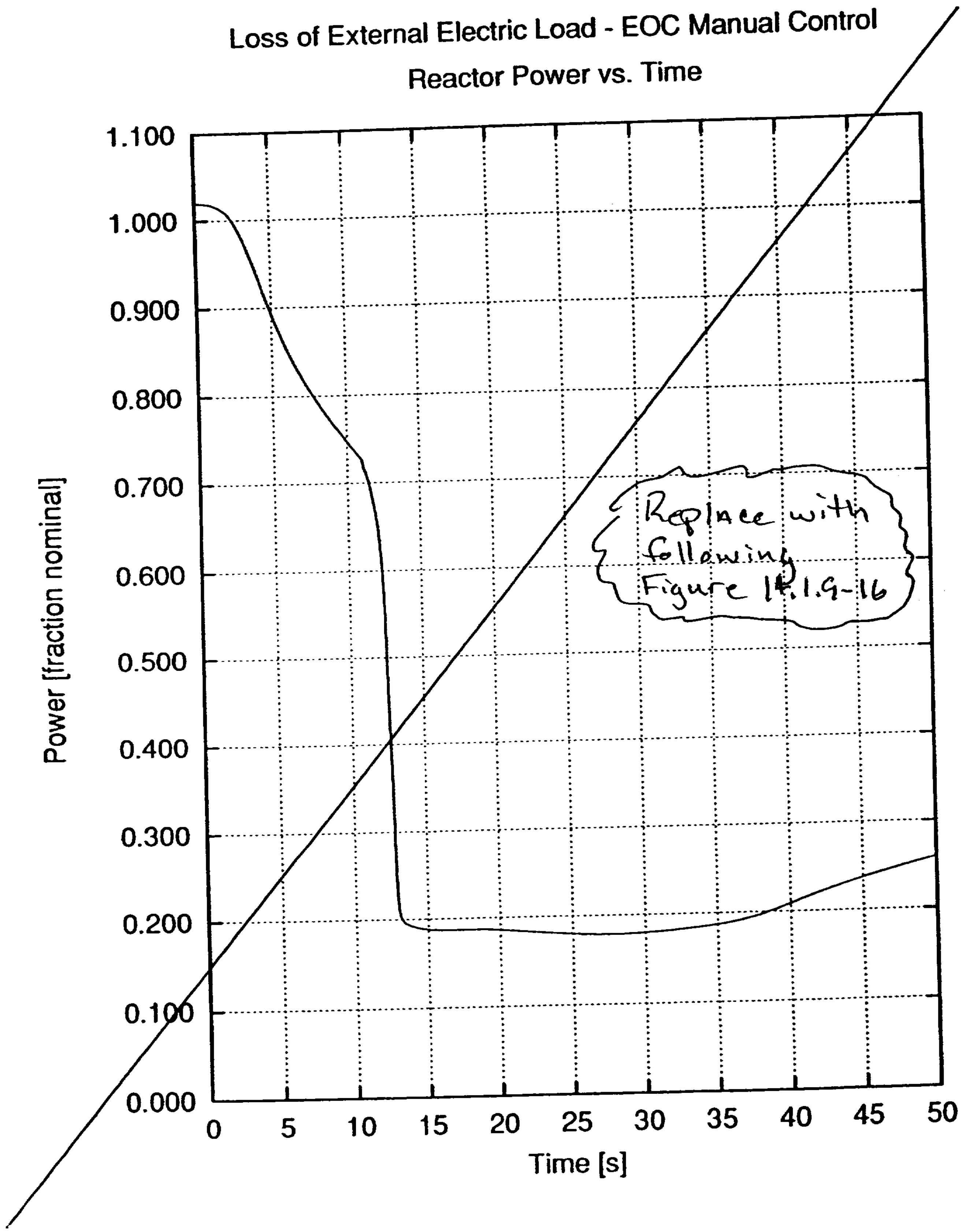

Figure 14.1.9-16

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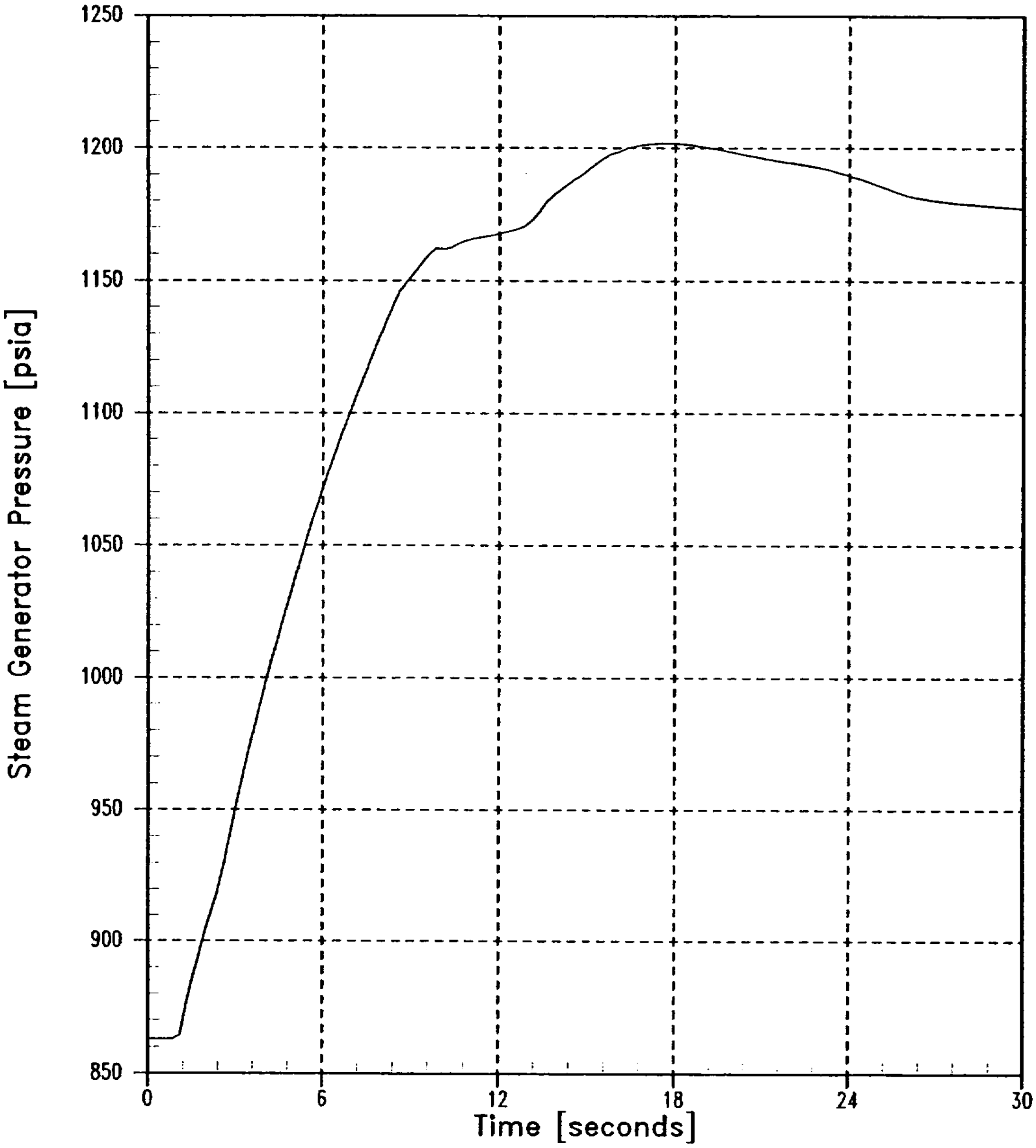
Loss of External Electrical Load With Auto Pressure Control (MSS Overpressure)
Steam Generator Pressure vs. Time



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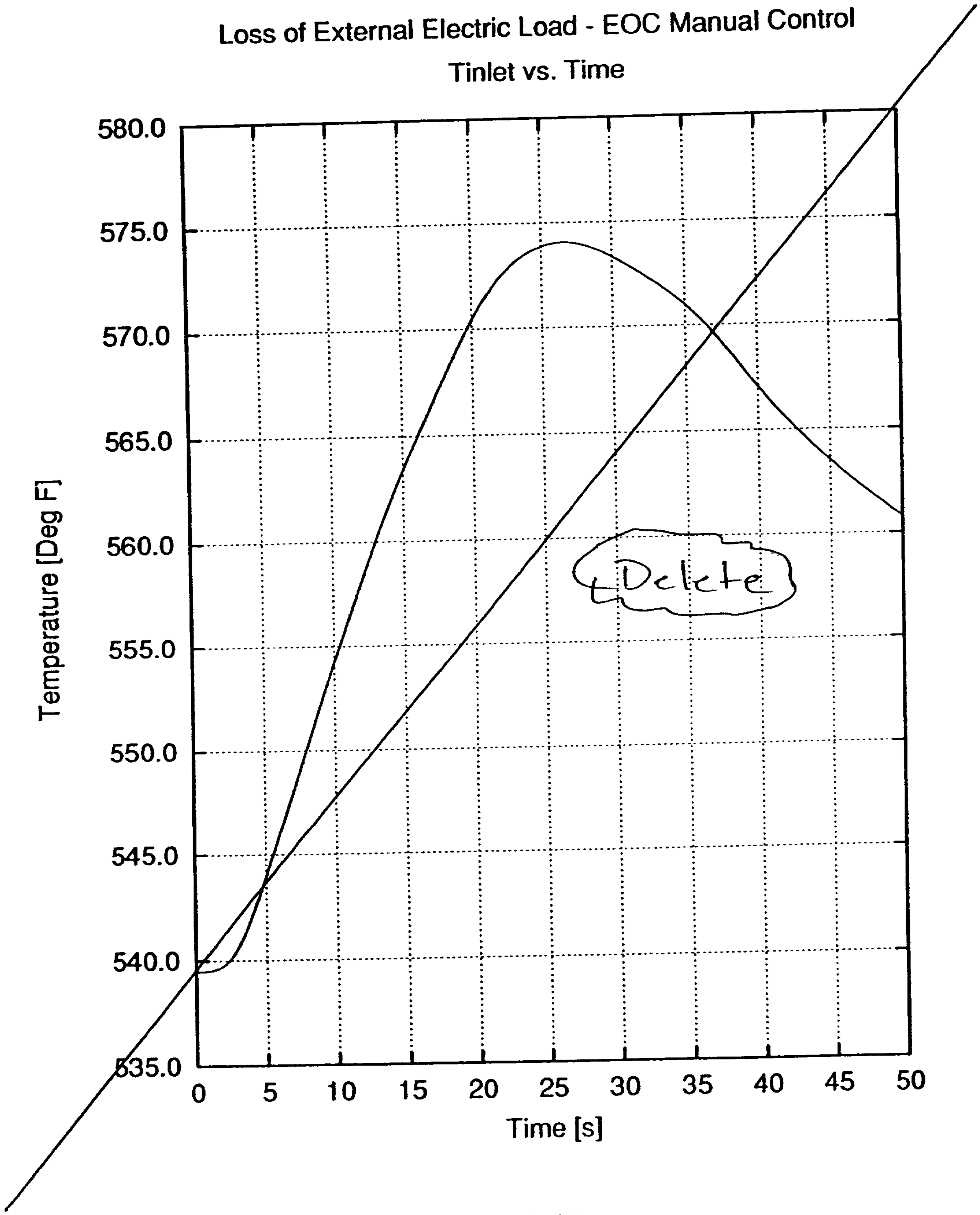

Figure 14.1.9-17

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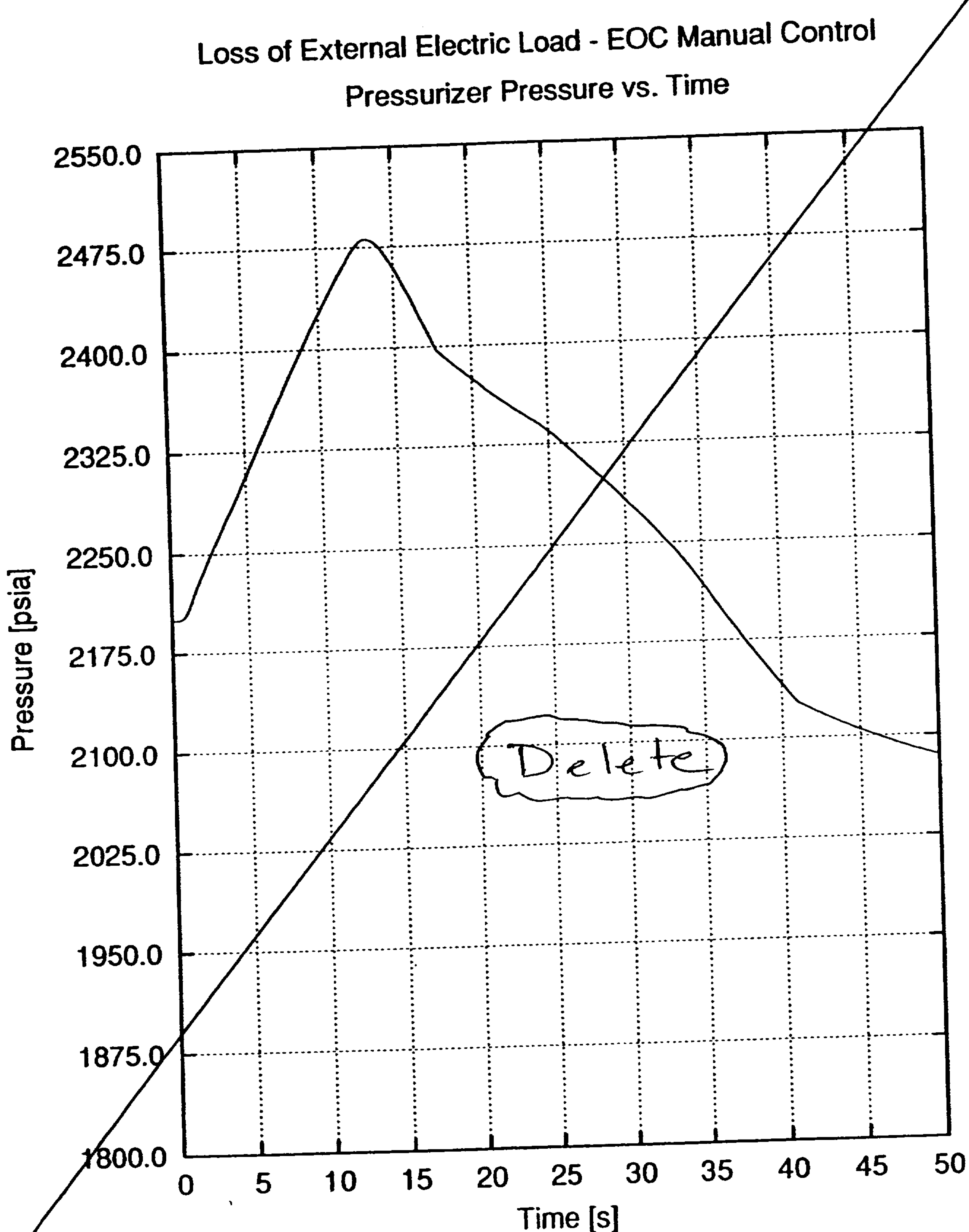

Figure 14.1.9-18

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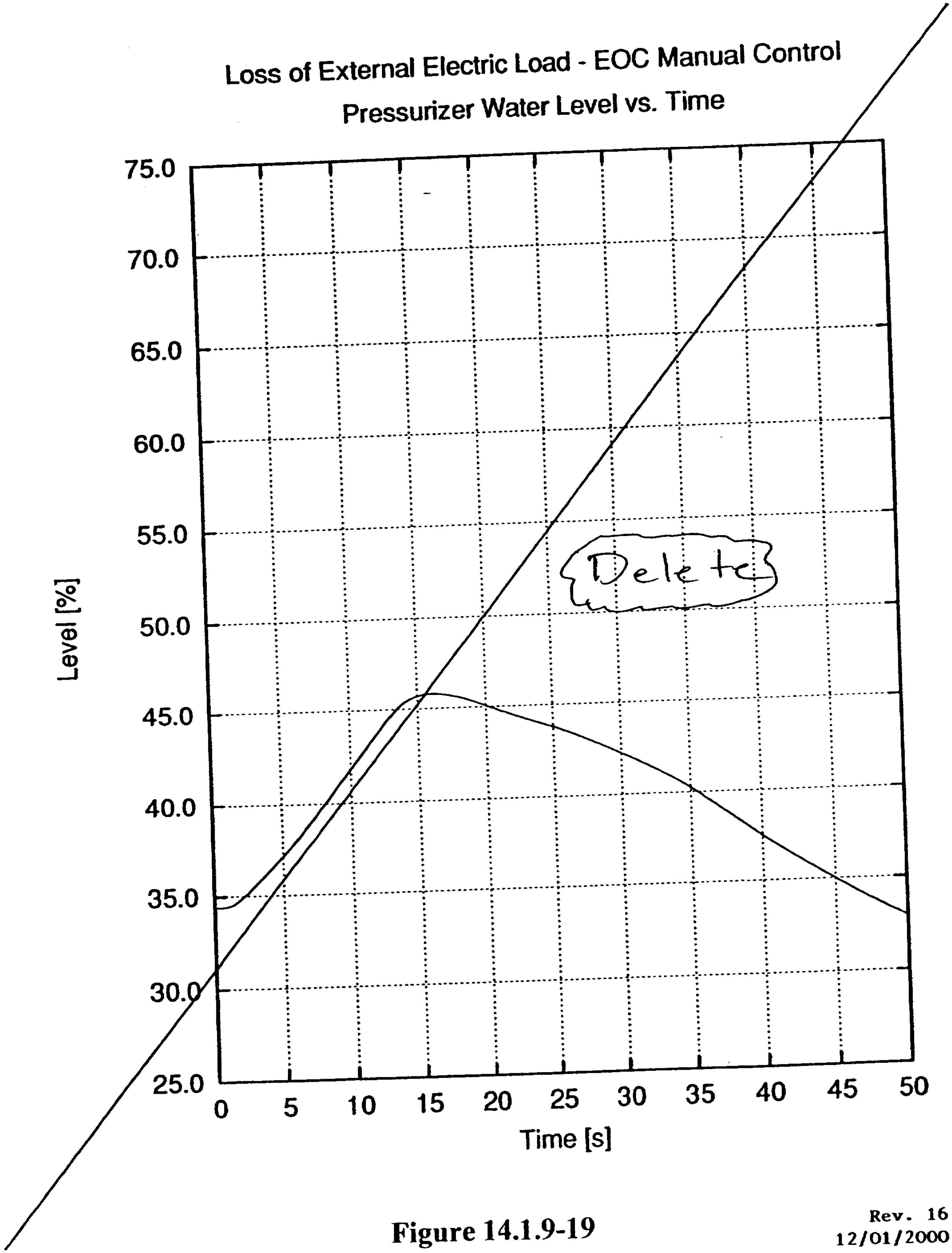

Figure 14.1.9-19

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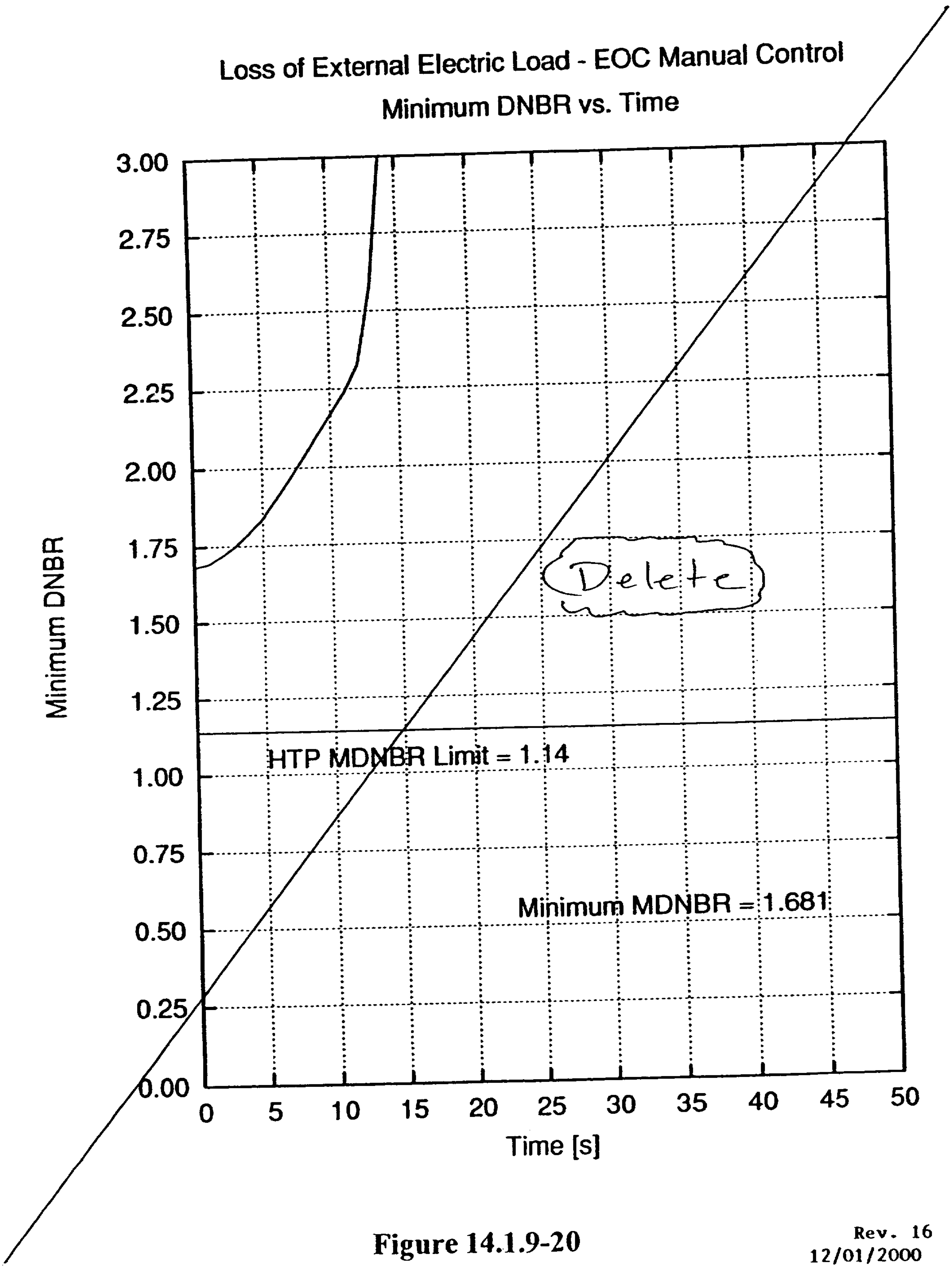

Figure 14.1.9-20

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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.

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	2474/2750	1184/1210
EOC Manual Control	1.681/1.14	2481/2750	1182/1210
EOC Auto Control	1.681/1.14	2377/2750	1198/1210

Section 14.1.9 changes suggested earlier.

Conclusions

Out of scope of LONF

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

Replace

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.

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

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

Replace

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

   d) Loss of off-site power, or

   e) Steam generator AMSAC low-low level, or

   f) Manually

4. One turbine driven pump which is started automatically on:

   a) Low-Low level in both steam generators, or

   b) Loss of voltage on both 4 kV. buses, or

   c) Steam generator AMSAC low-low level, or

   d) Manually

The motor-driven auxiliary feedwater pumps are supplied power by the diesel generators if a loss of outside power occurs. The turbine-driven pump uses steam from the secondary system. The turbine exhausts the secondary steam to the atmosphere. The auxiliary feedwater pumps take suction directly from the condensate storage tank for delivery to the steam generators.

Three auxiliary feedwater pumps are provided in the plant (two motor driven and one turbine driven). Necessary protection against consequences of a loss of normal feedwater including that caused by loss of off-site power is therefore available, allowing for an active failure on one of the operable auxiliary feedwater pumps even when one of the pumps is out-of-service.

When all three pumps are operable there is considerable backup in equipment and control to insure that reactor trip and automatic auxiliary feedwater flow occur following loss of normal feedwater.

Method of Analysis

Replace

The analysis was performed using a digital simulation of the plant to show that following a loss of normal feedwater, the Auxiliary Feedwater System is adequate to remove stored and residual heat.

The following assumptions are made:

1. The initial steam generator water level (in both steam generators) when the reactor trip occurs is assumed to be at the Low-Low level tap. This is conservative, because this level would result in a reactor trip and automatic initiation of the auxiliary feedwater flow.

2. The plant is initially operating at 102% of 1650 MWt.

3. Off-Site power is not available, resulting in natural circulation flow in the Reactor Coolant System.

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

5. Only one motor-driven auxiliary feedwater (AFW) pump is available 630 seconds after the accident is initiated.

6. Auxiliary feedwater is delivered to only one steam generator, at a flow rate of 176 gpm.

7. Secondary system steam relief is through the self-actuated safety valves. Nominal safety valve settings and rated safety valve flow capacities (Section 10.2.2) are assumed. To maximize the pressurizer insurge and Reactor Coolant System heatup, the safety valve blowdown settings are assumed to be at 0%. (Steam relief would in fact, be through the power-operated relief valves or condenser dump valves for most cases of loss of normal feedwater. However, these are assumed to be unavailable in the analysis).

Results

Figures 14.1.10-1 through 14.1.10-5 show the plant parameters following a loss of normal feedwater accident with the assumptions listed above. Following the reactor and turbine trip from full load, the water level in the steam generators falls due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. The auxiliary feedwater pump is delivering flow 630 seconds following the initiation of the low-low level trip, thus reducing the rate of water level decrease. The capacity of the auxiliary feedwater pump is such that the water level in the steam generator 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 primary system relief or safety valves.

From Figures 14.1.10-1 through 14.1.10-5, it can be seen that at no time is the tube sheet uncovered in the steam generator receiving auxiliary feedwater flow and at no time is there

Replace

water relief from the pressurizer. If the auxiliary feed delivered is greater than that of one motor driven pump, the initial reactor power is < 102% of 1650 MWt, or the steam generator water level in one or both steam generators is above the Low-Low level trip point at the time of trip, then the result is a steam generator minimum water level higher than shown and an increased margin to the point at which reactor coolant water relief occurs.

The following table shows the comparison of the important calculated safety parameters (Calculated Value/Acceptance Criterion):

	MDNBR	RCS Pressure (psia)	MS Pressure (psia)
Loss of Feedwater	1.681/1.14	2500/2750	1165/1210

Conclusion

The loss of normal feedwater does not result in any adverse condition in the core, because it does not result in water relief from the pressurizer relief or safety valves, nor does it result in uncovering the tube sheets of the steam generator being supplied with water.

14.1.11 ANTICIPATED TRANSIENTS WITHOUT SCRAM

No Changes

An Anticipated Transient Without Scram (ATWS) is a postulated anticipated operational occurrence (such as loss of feedwater, loss of condenser vacuum, or loss of off-site power) that is accompanied by a failure of the Reactor Protection System (RPS) to shut down the reactor.

The Kewaunee Nuclear Power Plant was originally licensed based on the results of a study of ATWS presented in WCAP 7486 (see Reference 9). The conclusions of this study are that there is very little likelihood of failure to trip the reactor and that even in the hypothetical case of no protective reactor trip, there is no gross fuel damage. WCAP 8330 presented the results of generic ATWS analysis for 2, 3, and 4 loop Westinghouse plants. The results of these analyses showed that the consequences of an ATWS were acceptable as long as the turbine was tripped and AFW initiated in a timely fashion. Acceptable consequences are defined as RCS pressure remaining below 3200 psig and no fuel failure. The results of the analyses in WCAP 8330 also showed that the most severe ATWS transients were those which entailed a loss of main feedwater. Subsequent to the operational license at KNPP and based on the studies cited above, additional ATWS protection was required as described below.

The Code of Federal Regulations (CFR), Section 10 CFR 50.62 (Reference 10) specifies ATWS mitigation system requirements. The Westinghouse Owners Group developed a set of conceptual ATWS Mitigating System Actuation Circuitry (AMSAC) designs (Reference 11). The AMSAC actuation on low steam generator water level design has been implemented, with the exception that AMSAC is armed at all power levels (the "c-20 permissive" signal is not used). The logic of AMSAC is to trip the turbine and start all three auxiliary feedwater pumps when low-low steam generator water level signals are present on 3 of 4 channels for a specified time period. However, as discussed in Section 6.6, manual initiation of auxiliary feedwater may be required at low power levels (< 15%). The level setpoint and time delay criteria are described in Reference 11.

The NRC Safety Evaluation Report (Reference 12) and a subsequent NRC Special Inspection Report (Reference 13) reviewed the Kewaunee design and installation against 14 key elements for compliance. The NRC concluded that the Kewaunee AMSAC is acceptable and in compliance with the ATWS rule, 10 CFR 50.62.

In 1998, in response to an engineering evaluation of the AFW system, a plant design change added a Diverse Scram System (DSS). The DSS is initiated on a signal from the existing AMSAC system and de-energizes the Rod Drive MG Set exciter field. Removing the Rod Drive MG set exciter field will interrupt power to the control rod grippers, allowing the control rods to free fall into the core, ending the ATWS event.

The DSS was installed to ensure the AFW pumps would continue to run throughout a loss of main feedwater ATWS. The DSS in conjunction with the AMSAC system will end the transient before the AFW flow to the steam generators increases to a point where AFW pump NPSH could be lost. The loss of main feedwater ATWS, mitigated by the DSS and AMSAC system, was analyzed using a similar methodology as the loss of main feedwater transient described in Section 14.1.10.

No changes

The original AMSAC submittal to the NRC was amended to include the DSS. The NRC Safety Evaluation Report (Reference 15) concluded that the Kewaunee DSS design was acceptable. The WPSC Safety Evaluation for the original AMSAC and the DSS included a review of the 14 key elements of ATWS compliance used by the NRC. This review concluded that the original AMSAC design reviewed by the NRC was unaffected by the addition of the DSS.

14.1.12 LOSS OF AC POWER TO THE PLANT AUXILIARIES

In the event of a complete loss of off-site power and a turbine trip, there will be a loss of power to the plant auxiliaries, i.e., the reactor coolant pumps, main feedwater pumps, etc. The events following a loss of off-site power with turbine trip are described in the sequence below.

Replace

a. The reactor is tripped and plant vital instruments are supplied by the emergency power sources.

b. The diesel generators start on loss-of-voltage on the 4kV buses to supply plant vital loads.

c. As the steam system pressure subsequently increases, the steam system power-operated relief valves are automatically opened to the atmosphere. Steam dump to the condenser is assumed not available because of loss of the circulating water pumps.

d. If the steam flow rate through the power-operated relief valves is not sufficient (or if the power-operated relief valves are not available), the steam generator self-actuated safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual heat produced in the reactor.

e. As the no-load temperature is approached, the steam power-operated relief valves (or self-actuated safety valves if the power-operated relief valves are not available for any reason) are used to dissipate the residual heat and to maintain the plant at the hot shutdown condition.

The auxiliary feedwater system is started automatically on loss of off-site power. The turbine-driven auxiliary feedwater pump uses steam from the secondary system and exhausts to the atmosphere. The motor-driven auxiliary feedwater pumps are supplied by power from the diesel generators. The pumps take suction directly from the condensate storage tank for delivery to the steam generators. The auxiliary feedwater system ensures feedwater flow upon loss of power to the plant auxiliaries. The flow rate assumed in the safety analysis for each of the two motor-driven auxiliary feedwater pumps and the turbine-driven auxiliary feedwater pump is the same as that discussed in Section 14.1.1.10.

The turbine-driven pump can be tested at any time by admitting steam to the turbine driver. The motor-driven pumps also can be tested at any time. The auxiliary feedwater control valves can be operationally tested whenever the plant is at hot shutdown and the remaining valves in the system are operationally tested when the pumps are tested.

Replace

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. The natural circulation flow was calculated for the conditions of equilibrium flow and maximum loop-flow impedance.

In response to Generic Letter 81-21, the ability to cool down via natural circulation without voiding the upper head of the reactor vessel was reviewed. The NRC concluded in Reference 14 that Kewaunee has adequately demonstrated the ability to cooldown without voiding the reactor vessel head and determined that sufficient condensate supply exists to support its cooldown procedures.

The average temperature, pressurizer water volume, and steam generator level assuming the most conservative initial plant conditions and equipment availability are shown in Figures 14.1.10-1, 14.1.10-2, 14.1.10-3, and 14.1.10-4 for a loss of normal feedwater including a loss of off-site power, and reactor coolant system natural circulation. It is shown in Section 14.1.10 that a loss of normal feedwater from any cause including a loss of off-site power does not result in water relief from the pressurizer relief or safety valves.

Conclusion

The loss of off-site power to the plant auxiliaries does not cause any adverse condition in the core since it does not result in water relief from the pressurizer relief or safety valves nor does it result in the loss of the steam generator(s) as a heat sink for residual heat removal.

LOSS OF NORMAL FEEDWATER

REPLACE

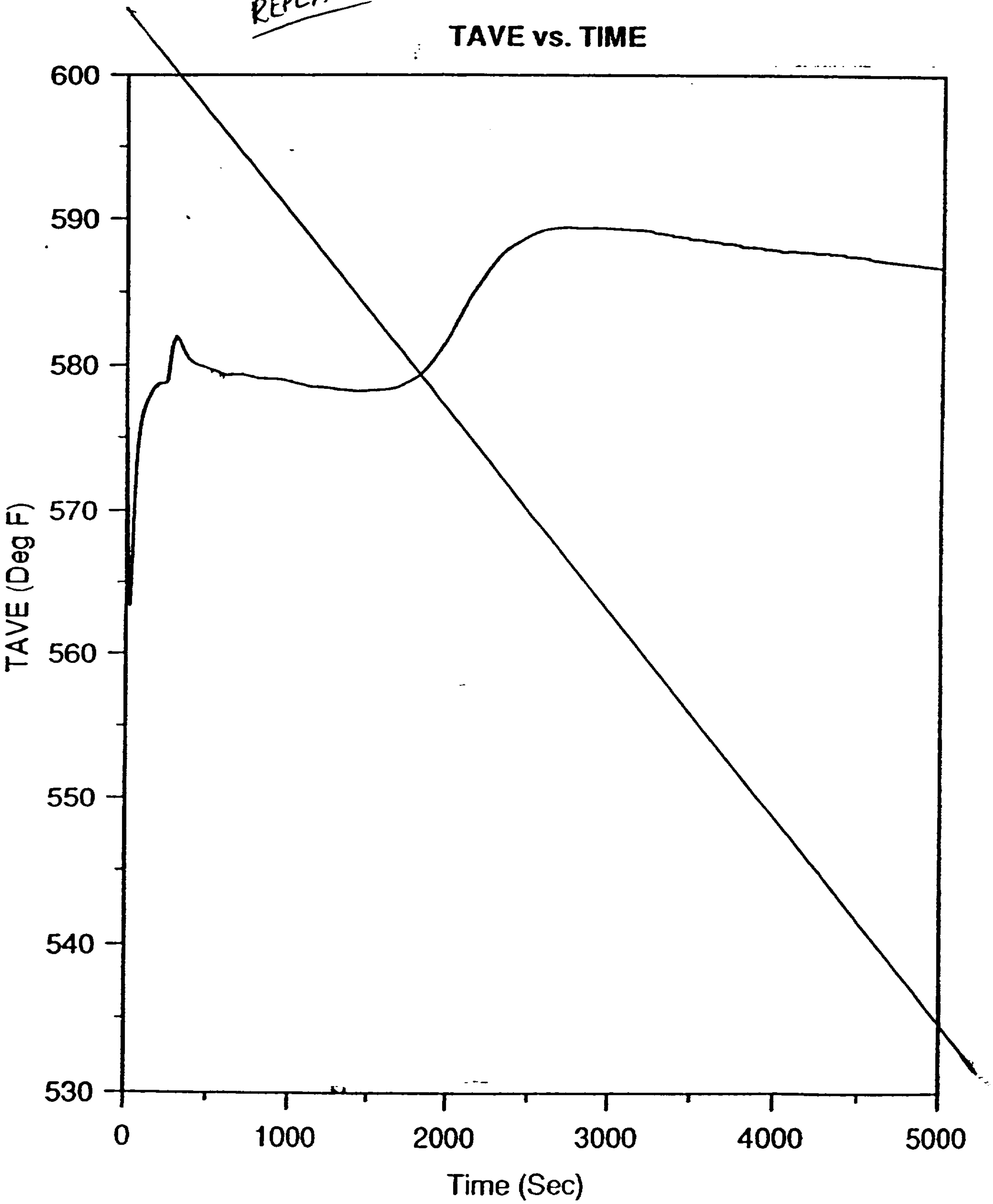

FIGURE 14.1.10-1

Rev. 15
05/01/99

LOSS OF NORMAL FEEDWATER

REPLACE

PRESSURIZER LIQUID VOLUME vs. TIME

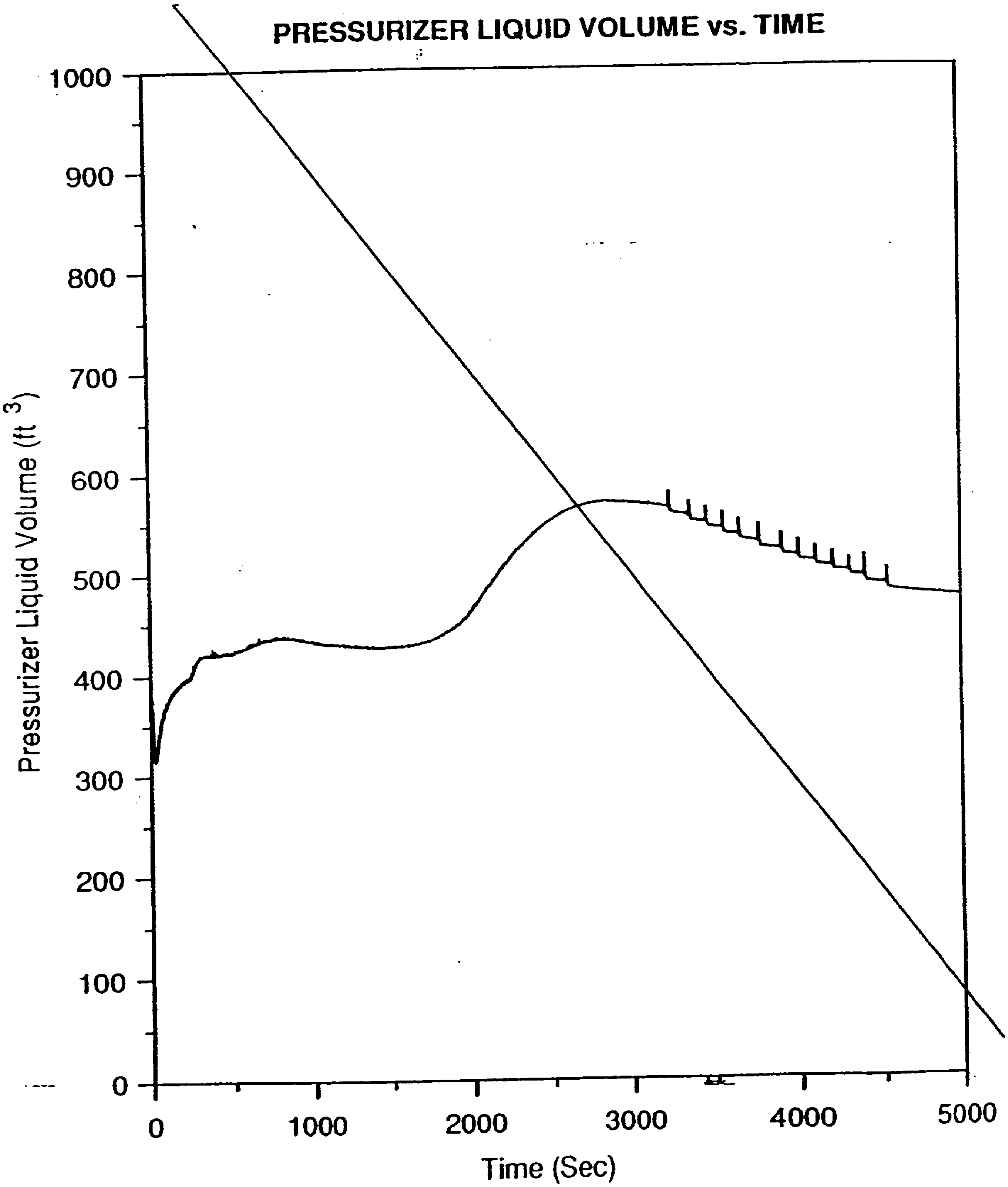

FIGURE 14.1.10-2

Rev. 15
05/01/99

LOSS OF NORMAL FEEDWATER

REPLACE

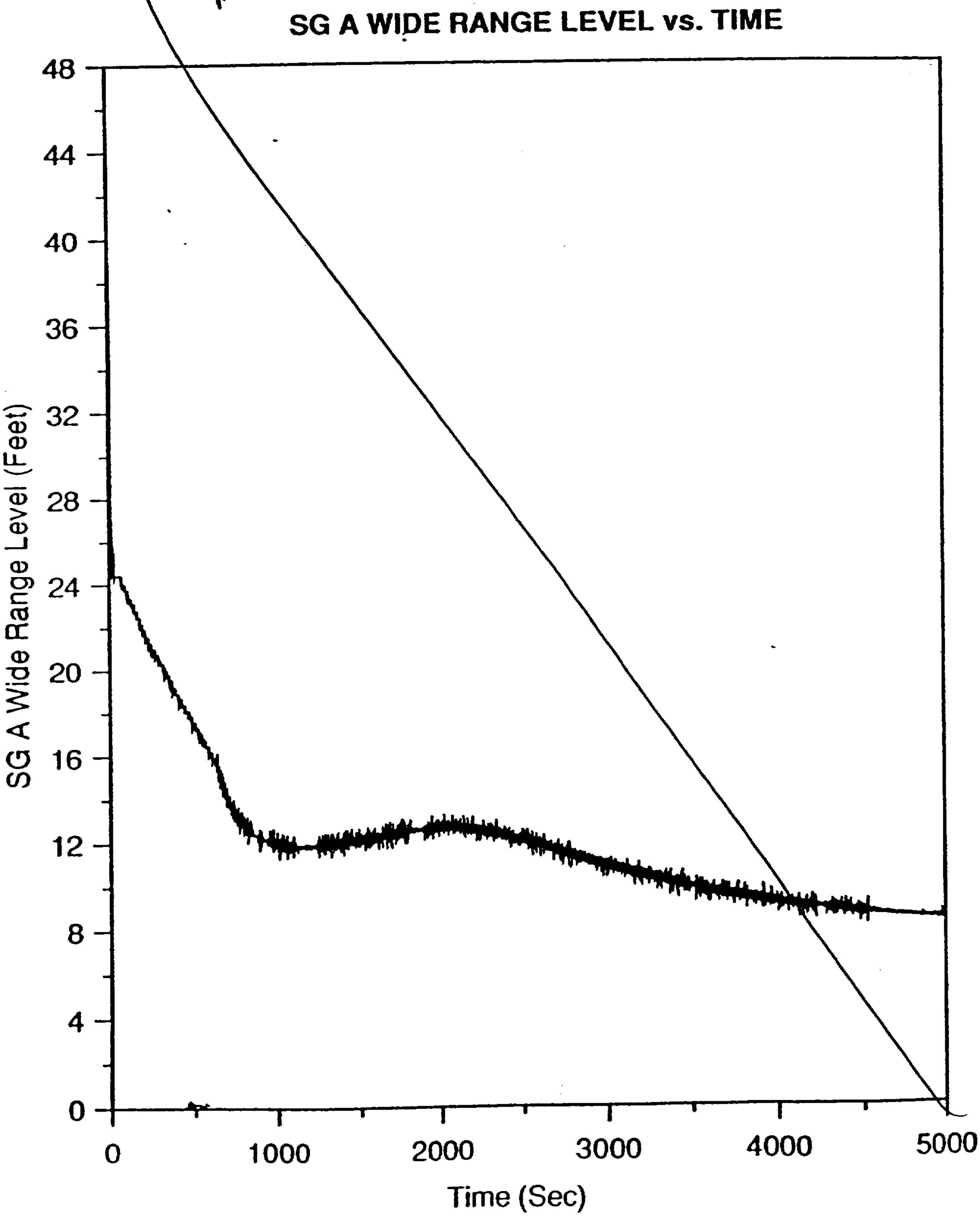

FIGURE 14.1.10-3

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LOSS OF NORMAL FEEDWATER

REPLACE

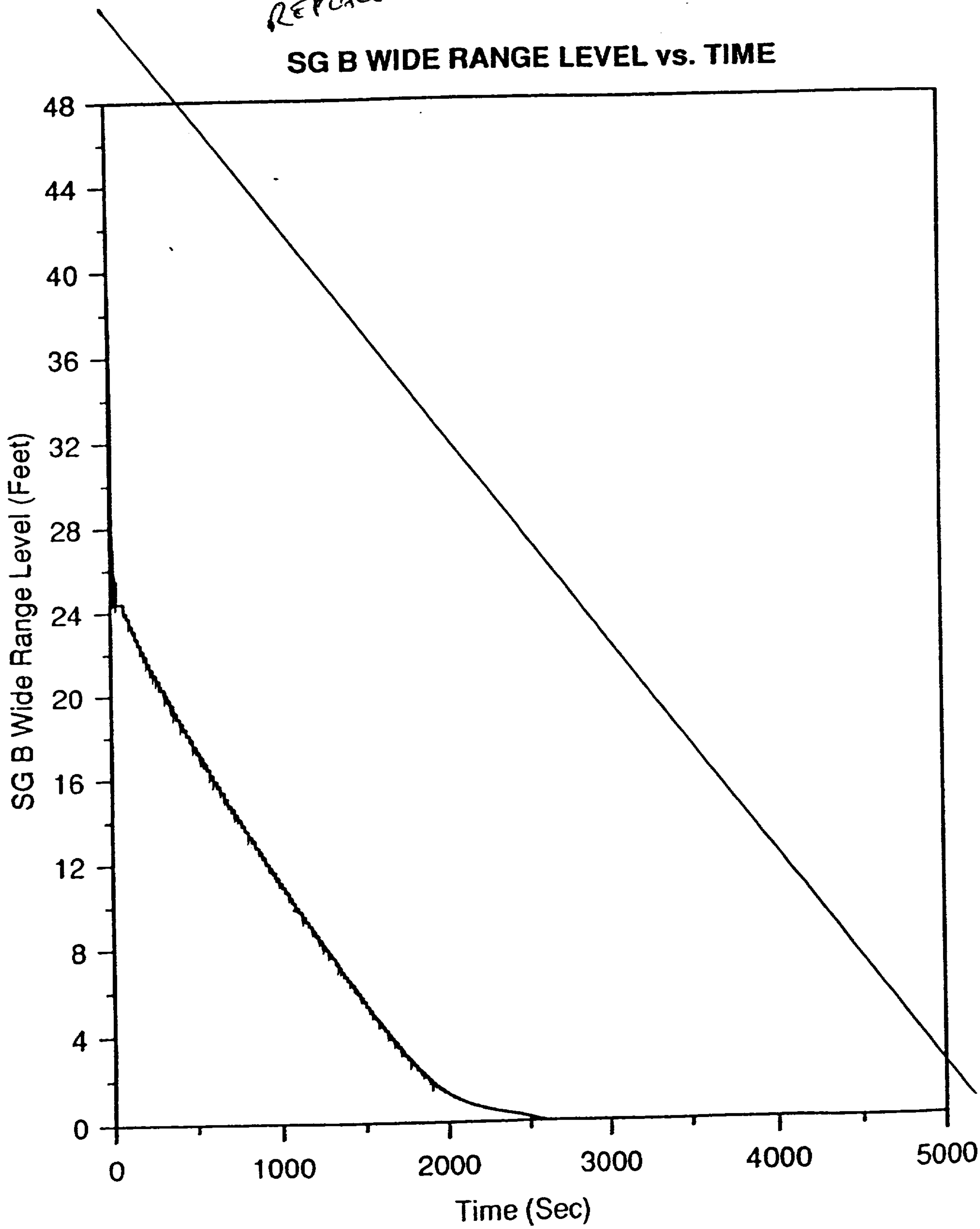

FIGURE 14.1.10-4

LOSS OF NORMAL FEEDWATER

REPLACE

PRESSURIZER PRESSURE vs. TIME

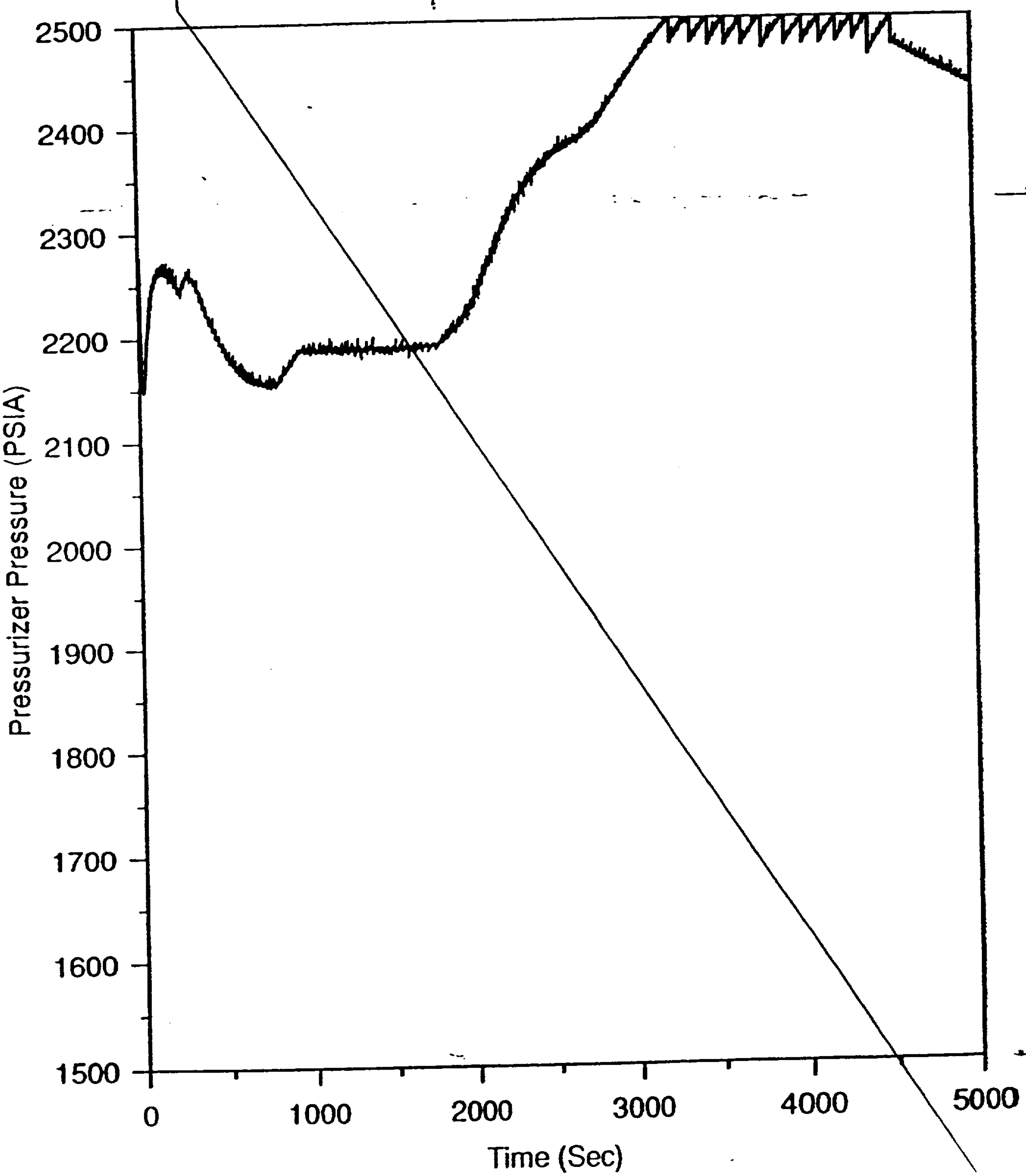

FIGURE 14.1.10-5

Rev. 15
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14.1.10 Loss of Normal Feedwater

Accident Description:

A loss of normal feedwater (from a pipe break, pump failure, or valve malfunction) results in a reduction of the ability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, core damage could possibly occur from a sudden loss of heat sink. If an alternate supply of feedwater were not supplied to the plant, residual heat following reactor trip and reactor coolant pump heat would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the reactor coolant system (RCS). A significant loss of water from the RCS could conceivably lead to core damage. Since the reactor is tripped well before the steam generator heat transfer capability is reduced, the primary system never approaches a condition where the departure from nucleate boiling ratio (DNBR) limit may be violated.

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

a. Reactor trip on low-low water level in either steam generator.

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

c. Two motor-driven auxiliary feedwater (AFW) pumps, which are started on:

   1. Low-low level in two-out-of-three level channels in either steam generator
   2. Opening of both feedwater pump circuit breakers
   3. Any safety injection signal
   4. Loss of offsite power
   5. Steam generator anticipated transient without scram (ATWS) mitigation system actuation circuitry (AMSAC) low-low level
   6. Manual actuation.

d. One turbine-driven AFW pump, which is started on:

   1. Low-low level in two-out-of-three channels in both steam generators
   2. Loss of voltage on both 4 kV buses.
   3. Steam generator AMSAC low-low level
   4. Manual actuation.

The AFW system is started automatically on the signals described above. Below 15% of rated thermal power (RTP), selected AFW valves (AFW-2A & AFW-2B and AFW-10A & AFW-10B) can be placed in the closed position, thereby precluding AFW flow to the steam generators. For this condition, manual operator action to re-initiate AFW flow after it has been isolated has been justified. The motor-driven AFW pumps (MDAFWP) are supplied by the diesel generators if a loss of offsite power occurs, and the turbine-driven AFW pump (TDAFWP) utilizes steam from the secondary system and exhausts the steam to the atmosphere. The AFW pumps take suction from the condensate storage tank (CST) for delivery to the steam generators.

The analysis shows that following a loss of normal feedwater, the AFW system is capable of removing the stored energy, residual decay heat and reactor coolant pump heat, thus preventing overpressurization of the RCS and a loss of water from the reactor core.

Method of Analysis:

The loss of normal feedwater transient is analyzed using the RETRAN computer code. The code simulates the neutron kinetics, RCS including natural circulation, pressurizer, pressurizer relief and safety valves, pressurizer heaters, pressurizer spray, steam generators, feedwater system and main steam safety valves. The code computes pertinent plant variables including steam generator mass, pressurizer water volume and reactor coolant average temperature.

The major assumptions are summarized below.

a. The plant is initially operating at 102% of 1780 MWt.

b. Reactor trip occurs on steam generator low-low water level at 0% of narrow range span (NRS). Turbine trip occurs coincident with reactor trip.

c. A conservative core residual heat generation is assumed, based on the ANS 5.1-1979 decay heat model plus 2 sigma.

d. One minute after the low-low steam generator water level setpoint is reached, the AFW system provides 176 gpm of flow split equally between the two steam generators (equal split is the limiting case). This AFW flow assumption is conservative with respect to the worst case scenario for available AFW flow during a loss of normal feedwater event, as the TDAFWP (single failure) and the second MDAFWP are assumed to be unavailable. The AFW enthalpy is assumed to be 90.8 BTU/lbm (120°F and 1100 psia).

e. Secondary system steam relief is achieved through the main steam safety valves (MSSVs), which include a + 2% setpoint tolerance, a 5 psi ramp for the valve to pop open and a pressure difference from the steam generator to the safety valves of approximately 42 psi. Steam relief through the steam generator power-operated relief valves (PORVs) or condenser dump valves is assumed to be unavailable.

f. The initial reactor coolant average temperature is assumed to be 6°F higher than the nominal full power value of 573.0°F because this results in a greater expansion of the RCS water during the transient, thus, resulting in a higher pressurizer water level.

g. The initial pressurizer pressure is assumed to be 50 psi above the nominal value.

h. Normal reactor control systems are not assumed to function. However, the pressurizer PORVs, pressurizer heaters and pressurizer spray are assumed to operate normally. This assumption results in a conservative transient with respect to the peak pressurizer water level. If these control systems did not operate, the pressurizer safety valves would maintain peak RCS pressure around the actuation setpoint throughout the transient.

i. Credit is assumed for the operators to trip the reactor coolant pumps at 15 minutes following reactor trip, thereby minimizing the overall heat that the AFW system must remove from the RCS.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the reactor protection system (RPS) and engineered safeguards features (ESF) (e.g., the AFW system) to remove long-term decay heat, stored energy and reactor coolant pump heat, thus preventing excessive heatup or overpressurization of the RCS. As such, the assumptions used in the analysis are designed to minimize the energy removal capability of the system and maximize the possibility of water relief from the RCS by maximizing the expansion of the RCS inventory, as noted in the assumptions listed above.

Results:

Figures 14.1.10-1 through 14.1.10-6 show the significant plant responses following a loss of normal feedwater.

Following the reactor and turbine trip from full load, the water level in the steam generators falls due to the reduction of the steam generator void fraction and because steam flow through the MSSVs continues to dissipate the stored and generated heat. One minute after the initiation of the low-low level trip, flow from the available motor-driven AFW pump is credited, thus reducing the rate of water level decrease in the steam generators.

The capacity of one motor-driven AFW pump is such that the water level in the steam generators does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat, stored energy and reactor coolant pump heat without water relief through the pressurizer PORVs or safety valves. Figure 14.1.10-4 shows that at no time is there water relief from the pressurizer as the peak pressurizer water volume is less than the limit of 1010.10 ft^3. Plant procedures may be followed to further cool down the plant. The maximum MSS pressure is less than 110% of the steam generator design pressure. The RCS overpressurization limit is also not challenged during this transient. However, note that the pressurizer PORVs are assumed to be operable so as to maximize the potential for pressurizer filling. This event is bounded by the Loss of External Electrical Load (Section 14.1.9) with respect to peak RCS and MSS pressures.

Conclusions:

The results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the MSS. The AFW capacity is sufficient to dissipate core residual heat, stored energy and reactor coolant pump heat such that reactor coolant water is not relieved through the pressurizer relief or safety valves. Pressure relieving devices that have been incorporated into the plant design are adequate to limit the maximum pressures to within the safety analysis limits, i.e., 2750 psia for the primary RCS and 1210 psia for the MSS.

Loss of Normal Feedwater
Nuclear Power vs. Time

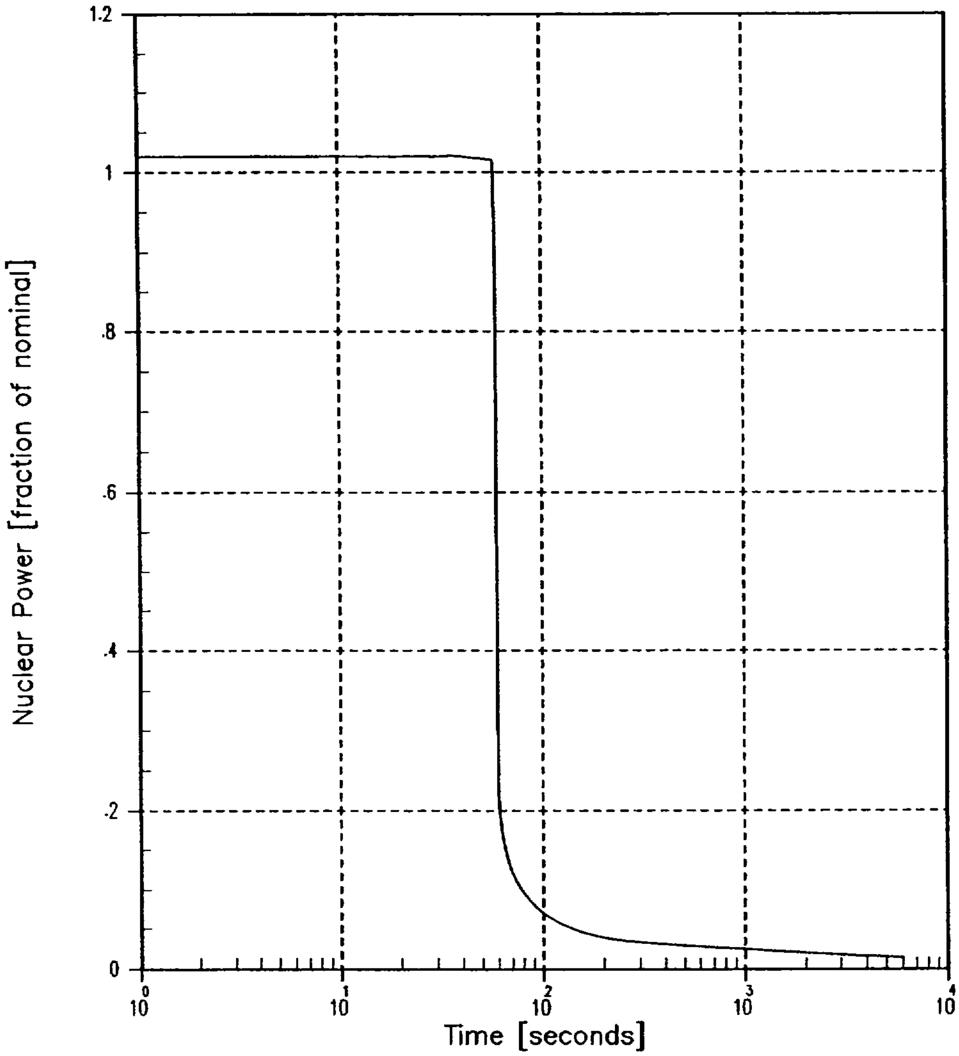

Figure 14.1.10-1

Loss of Normal Feedwater
Vessel Average Temperature vs. Time

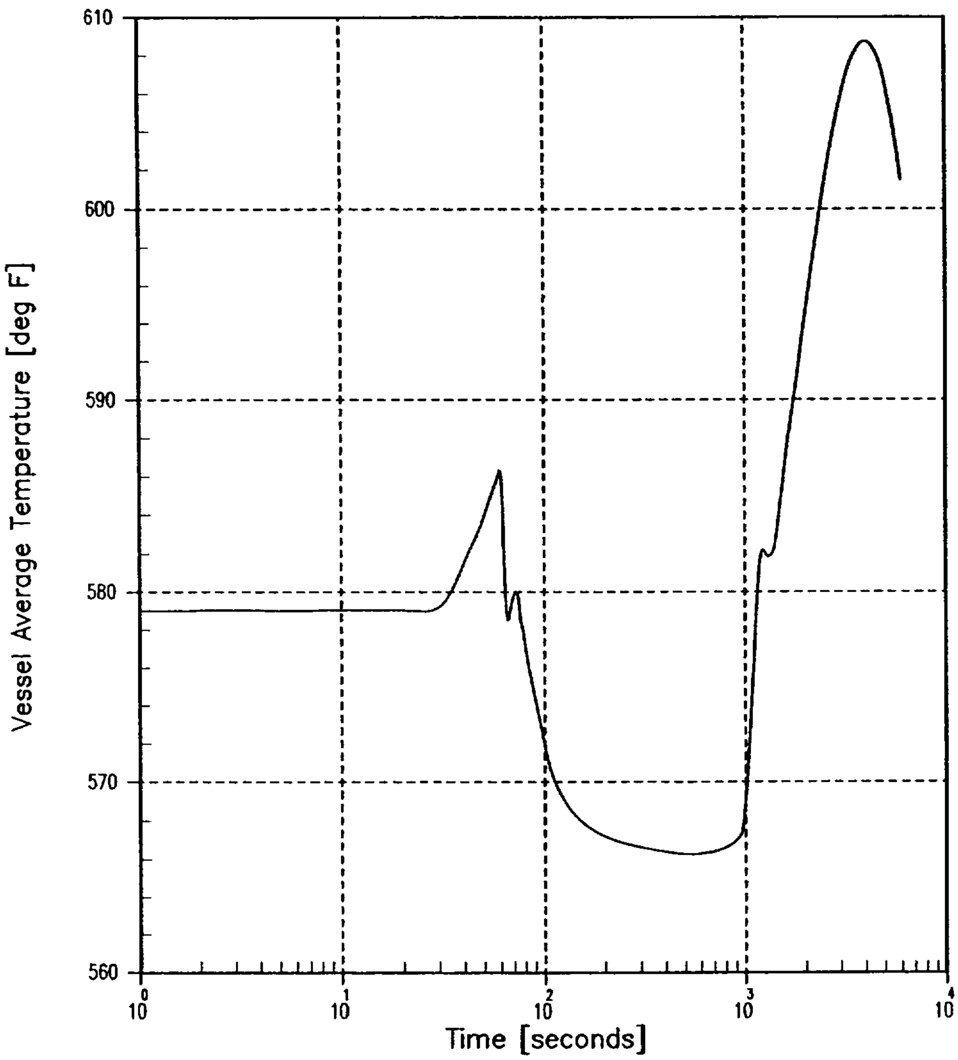

Figure 14.1.10-2

Loss of Normal Feedwater
Pressurizer Pressure vs. Time

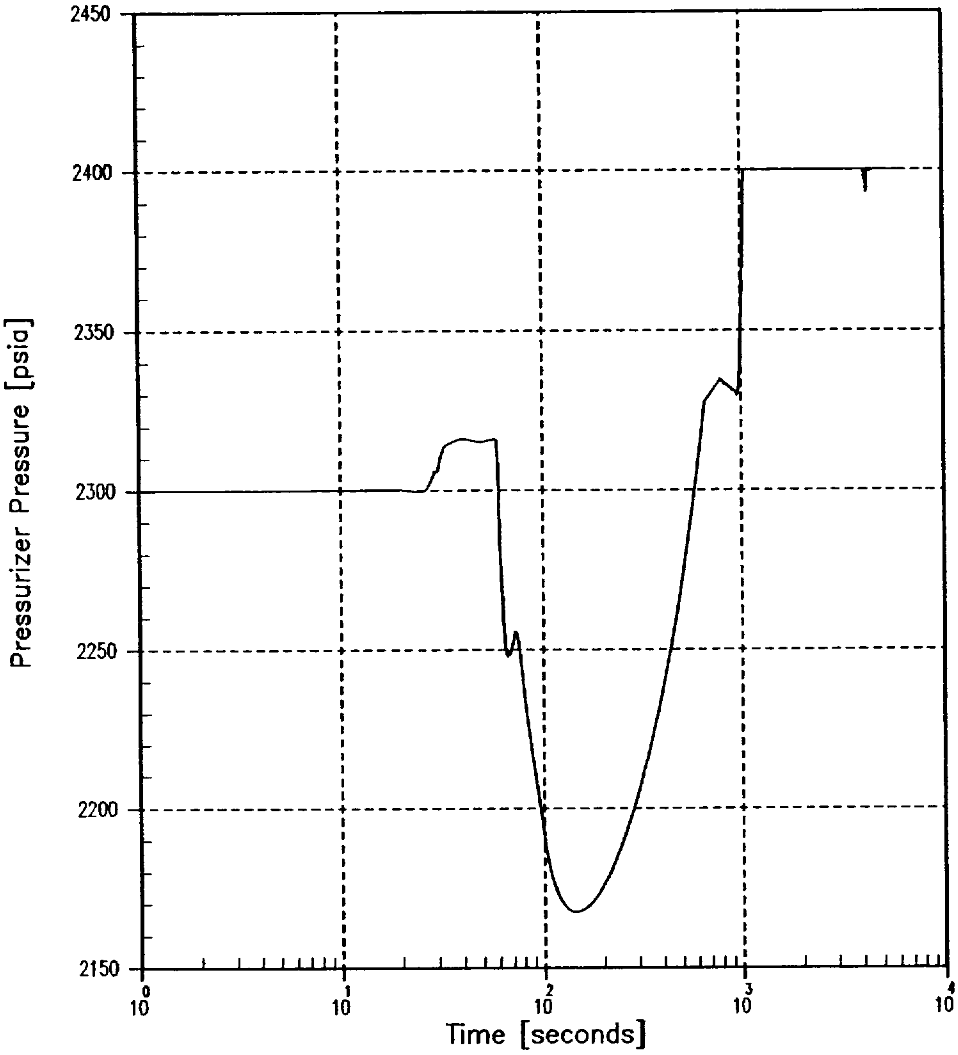

Figure 14.1.10-3

Loss of Normal Feedwater
Pressurizer Water Volume vs. Time

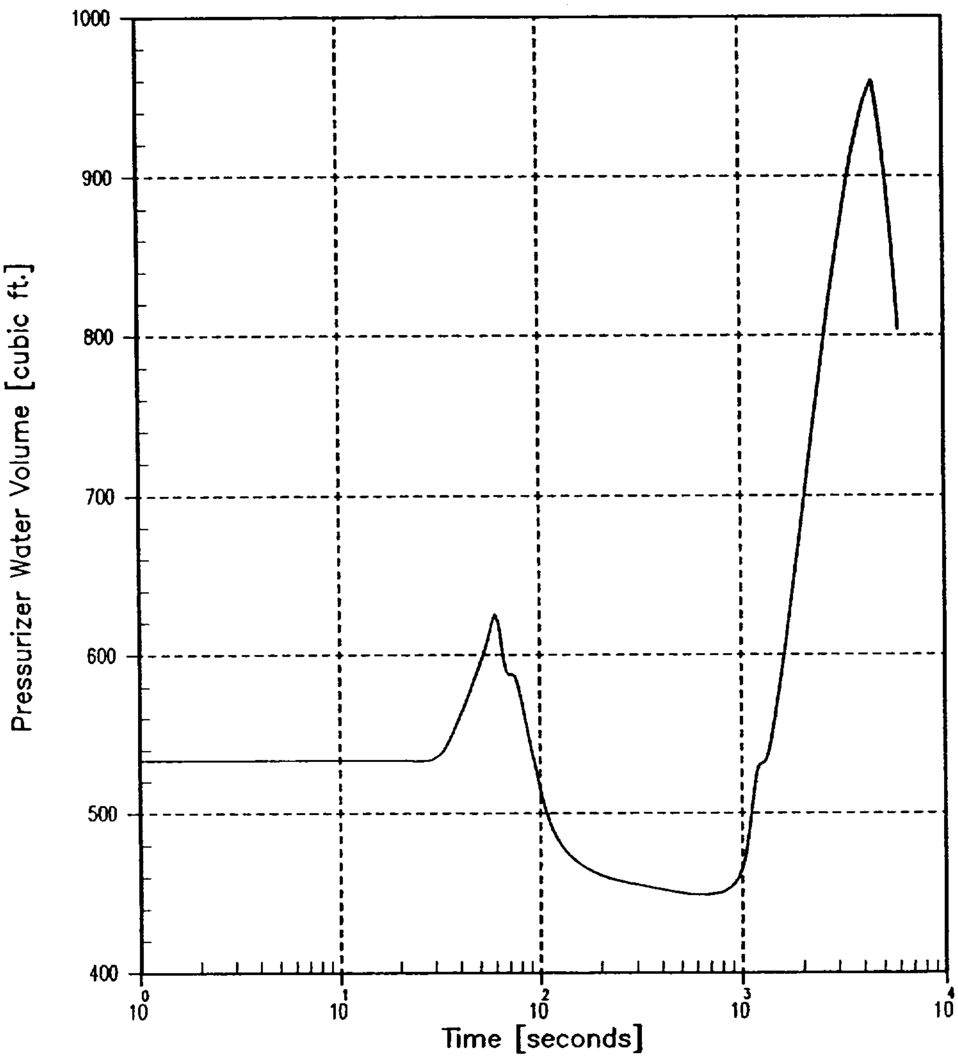

Figure 14.1.10-4

Loss of Normal Feedwater
Steam Generator Pressure vs. Time

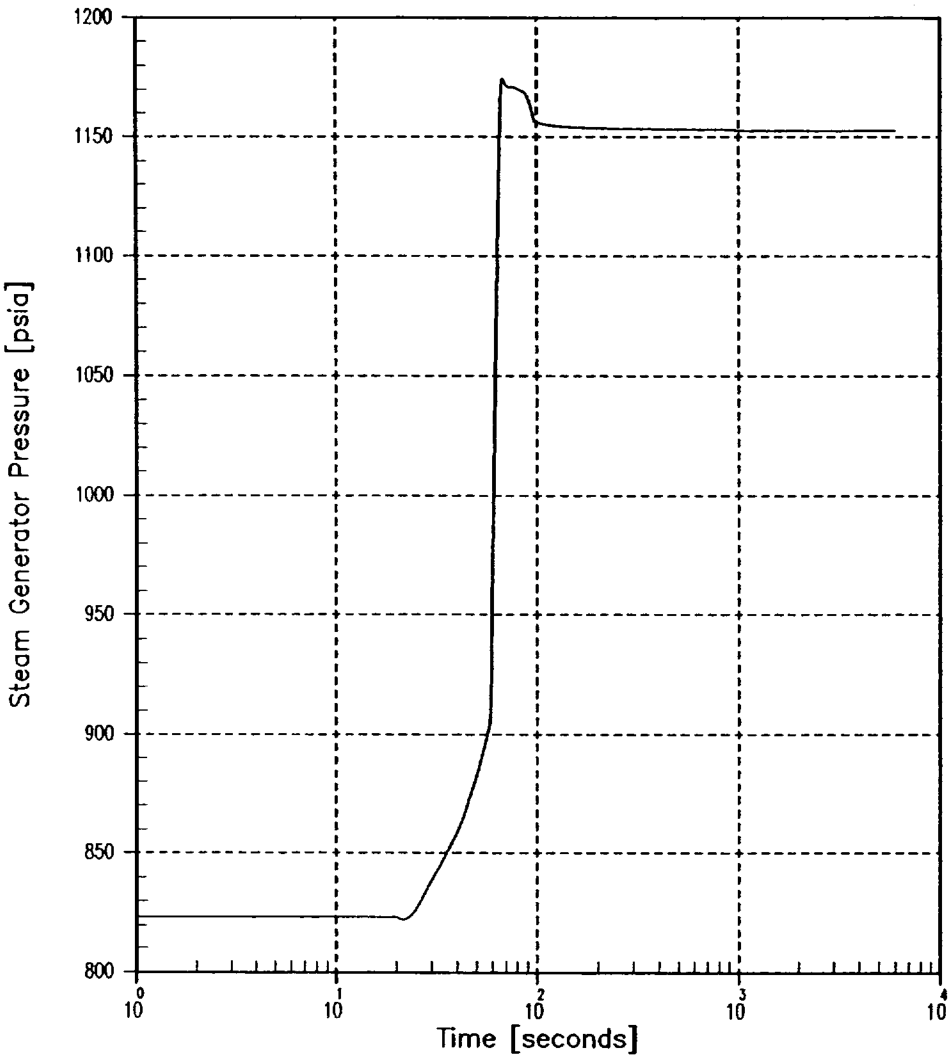

Figure 14.1.10-5

Loss of Normal Feedwater
Steam Generator Mass vs. Time

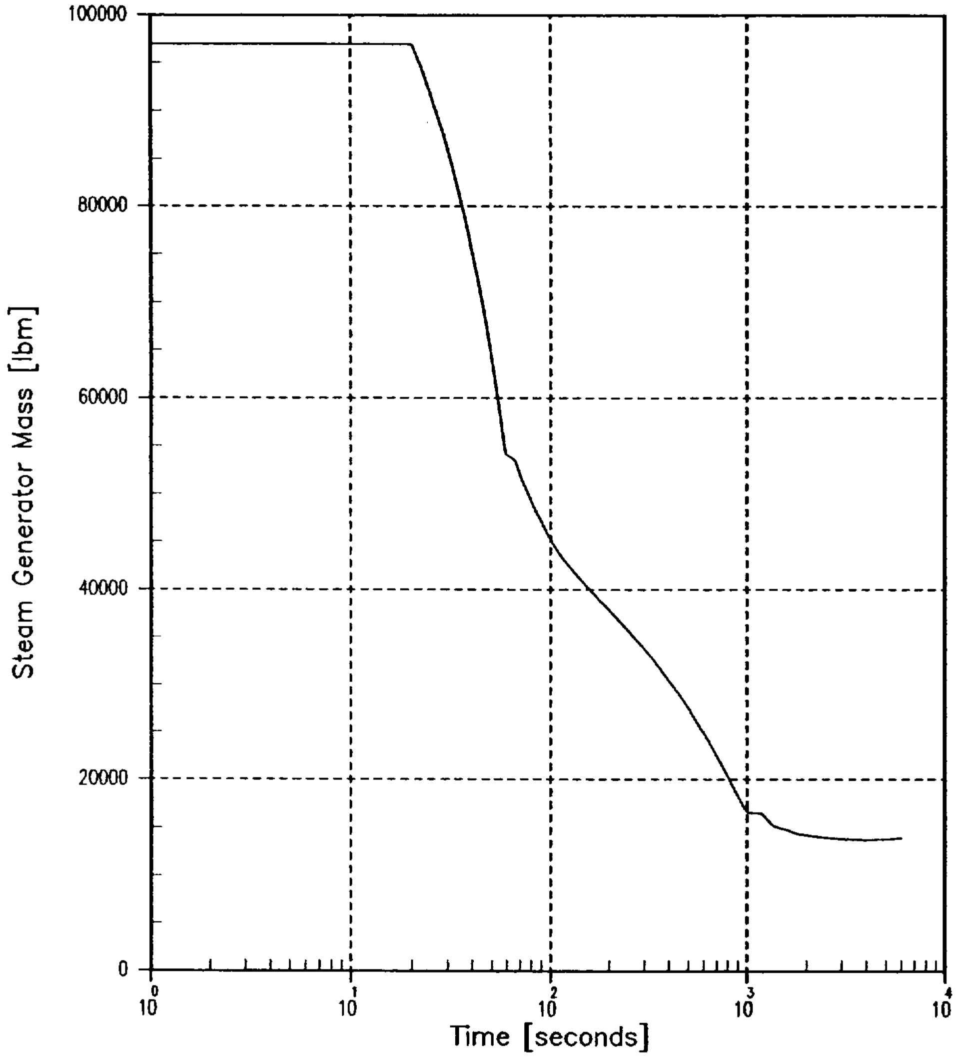

Figure 14.1.10-6

14.1.12 Loss of All AC Power to the Plant Auxiliaries

Accident Description:

A complete loss of non-emergency AC power results in the loss of all power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite AC distribution system.

The events following a loss of AC power with turbine and reactor trip are described in the sequence listed below.

1. Plant vital instruments are supplied from emergency power sources.

2. Steam dump to the condenser and steam generator PORVs are unavailable; therefore, the main steam safety valves lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat.

3. As the no-load temperature is approached, the steam generator PORVs (or the safety valves, if the PORVs are not available) are used to dissipate the residual decay heat and maintain the plant at the hot shutdown condition.

4. The standby diesel generators, started on loss of voltage on the plant emergency busses, begin to supply plant vital loads.

The AFW system is started automatically, as discussed in the loss of normal feedwater analysis (Section 14.1.10). The TDAFWP utilizes steam from the secondary system and exhausts to the atmosphere. The motor-driven AFW pumps are supplied by power from the diesel generators. The pumps take suction directly from the condensate storage tank for delivery to the steam generators.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops. Following the RCP coastdown caused by the loss of AC power, the natural circulation capability of the RCS removes residual and decay heat from the core, aided by the AFW in the secondary system.

Method of Analysis:

The loss of all AC power to the station auxiliaries transient is analyzed using the RETRAN computer code. The code simulates the neutron kinetics, RCS including natural circulation, pressurizer, pressurizer relief and safety valves, pressurizer heaters, pressurizer spray, steam generators, feedwater system and main steam safety valves. The code computes pertinent plant variables including steam generator mass, pressurizer water volume and reactor coolant average temperature.

Major assumptions made in the loss of all auxiliary AC power analysis are summarized below. :

a. The plant is initially operating at 102% of the 1780 MWt.

b. Reactor trip occurs on steam generator low-low level at 0% of narrow range span. Turbine trip occurs coincident with reactor trip.

c. A conservative core residual heat generation based on ANS 5.1-1979 decay heat plus 2 sigma is assumed.

d. The amount of heat transfer assumed to occur in the steam generators following the reactor coolant pump coastdown is based on RCS natural circulation conditions.

e. One minute after the low-low steam generator water level setpoint is reached, the AFW system provides 176 gpm of flow split equally between the two steam generators (equal split is the limiting case). The AFW flow assumption is conservative with respect to the worst case scenario for available AFW flow during a loss of all auxiliary AC power event, as the TDAFWP (single failure) and the second MDAFWP are assumed to be unavailable. The AFW enthalpy is assumed to be 90.8 BTU/lbm (120°F and 1100 psia).

f. Secondary system steam relief is achieved through the main steam safety valves, which include a +2% setpoint tolerance, a 5 psi ramp for the valve to pop open and a pressure difference from the steam generator to the safety valves of approximately 42 psi. Steam relief through the steam generator power-operated relief valves (PORVs) or condenser dump valves is assumed unavailable.

g. The initial reactor coolant average temperature is assumed to be 6°F lower than the nominal value of 573.0°F because this results in a greater expansion of the RCS water during the transient, thus, resulting in a higher pressurizer water level.

h. The initial pressurizer pressure is assumed to be 50 psi above its nominal value. An additional 0.1-psi uncertainty has been determined to be negligible.

i. Nominal reactor control systems are not assumed to function. However, the pressurizer PORVs, pressurizer heaters and pressurizer spray are assumed to operate normally. This assumption results in a conservative transient with respect to the peak pressurizer water level. If these control systems did not operate, the pressurizer safety valves would maintain peak RCS pressure around the actuation setpoint throughout the transient.

The assumptions used in the analysis are similar to the loss of normal feedwater (Section 14.1.10) except that power is assumed to be lost to the reactor coolant pumps due to the reactor trip.

Results:

Figures 14.1.12-1 through 14.1.12-6 show the significant plant responses following a loss of all AC power to the station auxiliaries event.

The first few seconds after the loss of power to the reactor coolant pumps will closely resemble the simulation of the complete loss of flow accident (FSAR Section 14.1.8), where core damage due to rapidly increasing core temperature 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 peak pressurizer water volume is less than the limit of 1010.10 ft^3. The maximum steam generator pressure calculated was less than 110% of the design pressure of 1085 psig. The RCS overpressurization limit is not challenged during this transient. However, note that the pressurizer PORVs are assumed to be operable so as to maximize the potential for pressurizer filling. This event is bounded by the Loss of External Electrical Load (Section 14.1.9) with respect to peak RCS and MSS pressures.

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

Conclusions:

The results of the analysis show that a loss of all AC power to the station auxiliaries does not adversely affect the core, the RCS, or the MSS. The AFW capacity is sufficient to dissipate core residual heat. Consequently, reactor coolant is not relieved through the pressurizer relief or safety valves. Pressure relieving devices that have been incorporated into the plant design are adequate to limit the maximum pressures to within the safety analysis limits, i.e., 2750 psia for the primary RCS and 1210 psia for the MSS.

Loss of AC Power to the Plant Auxiliaries
Nuclear Power vs. Time

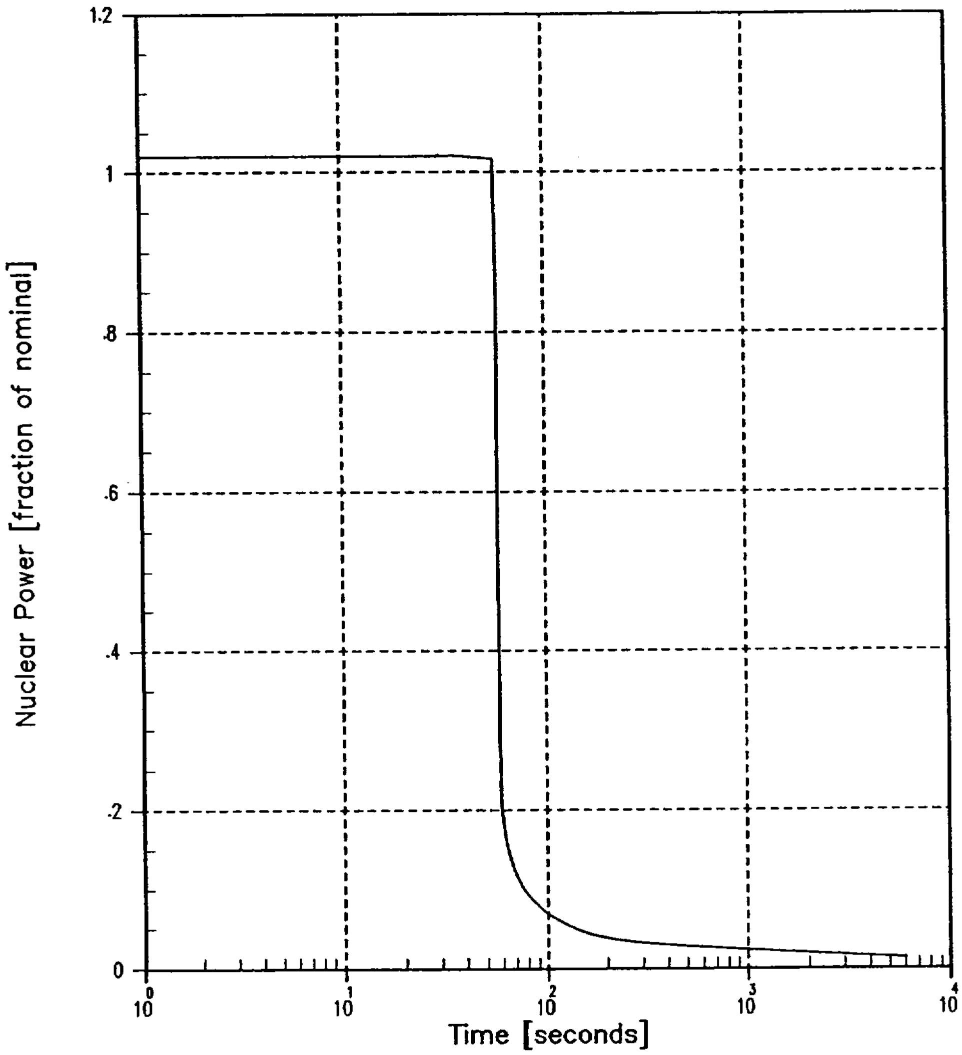

Figure 14.1.12-1

Loss of AC Power to the Plant Auxiliaries
Vessel Average Temperature vs. Time

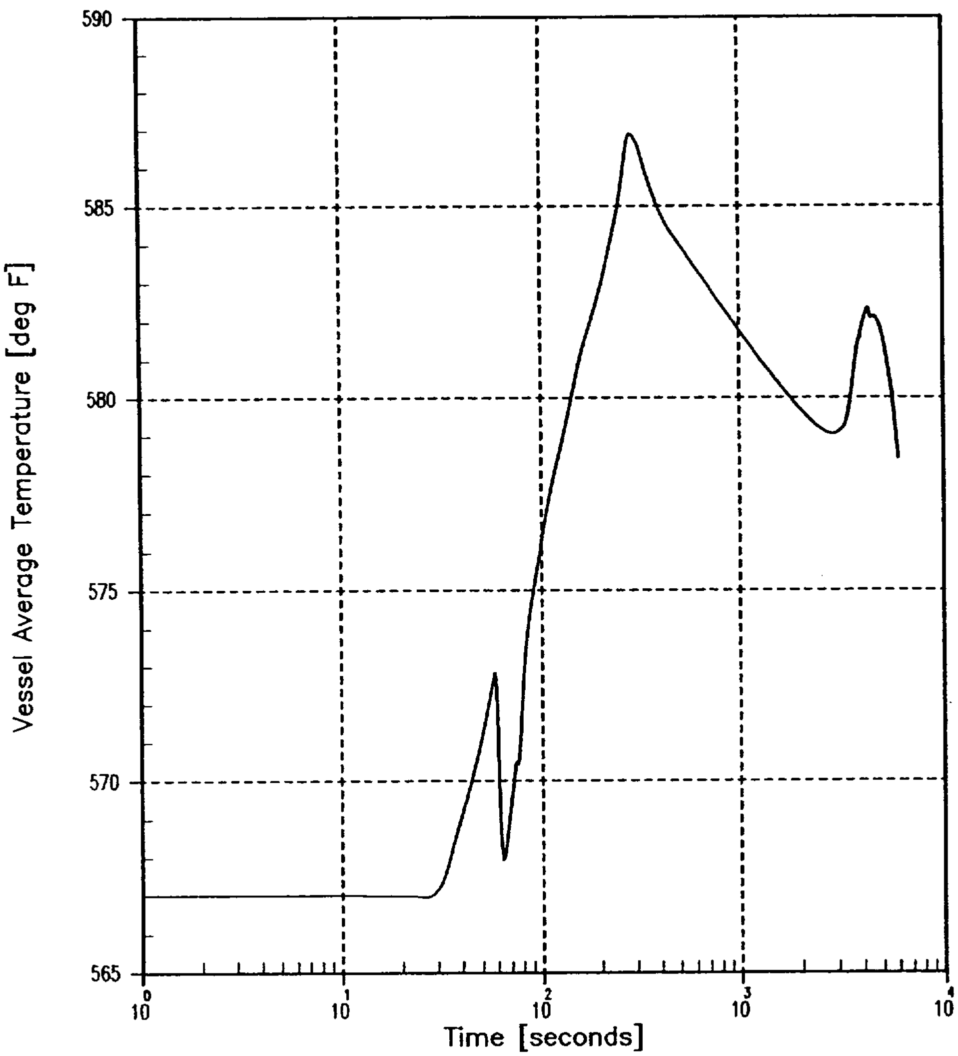

Figure 14.1.12-2

Loss of AC Power to the Plant Auxiliaries
Pressurizer Pressure vs. Time

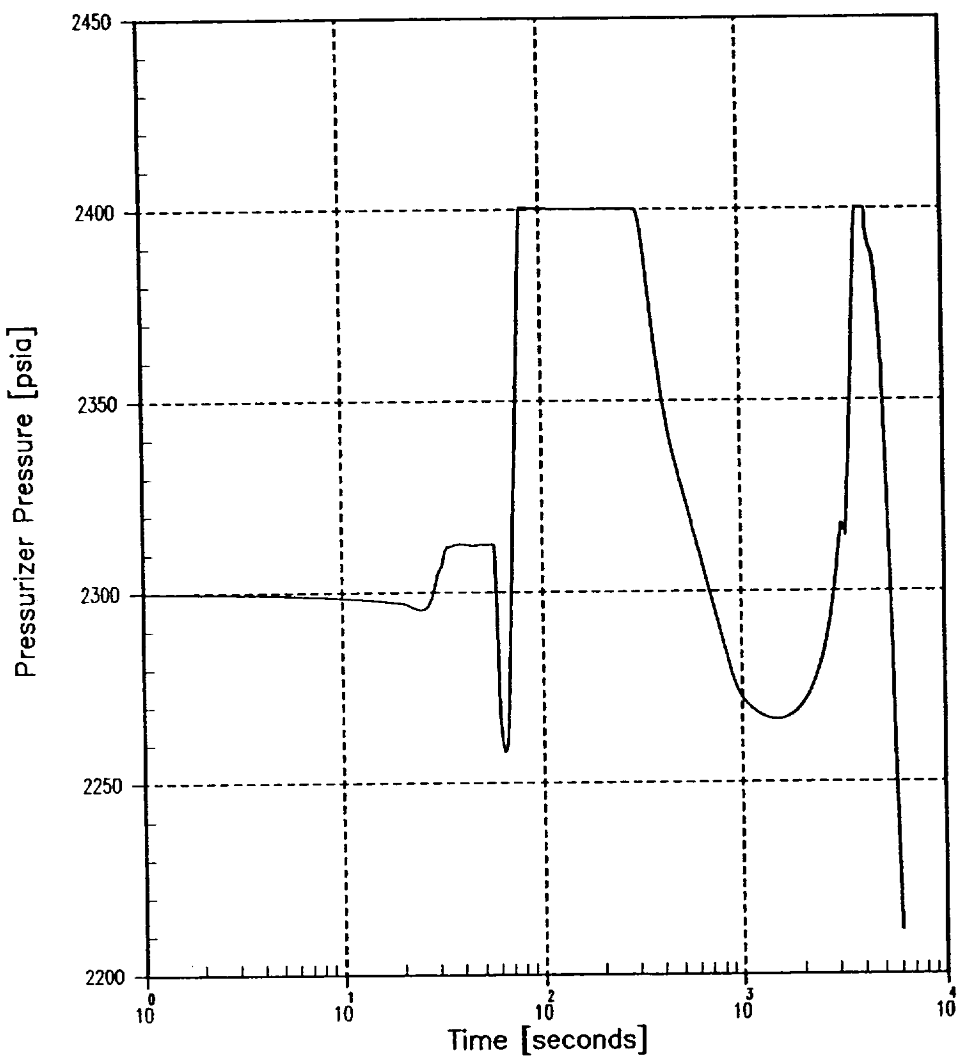

Figure 14.1.12-3

Loss of AC Power to the Plant Auxiliaries
Pressurizer Water Volume vs. Time

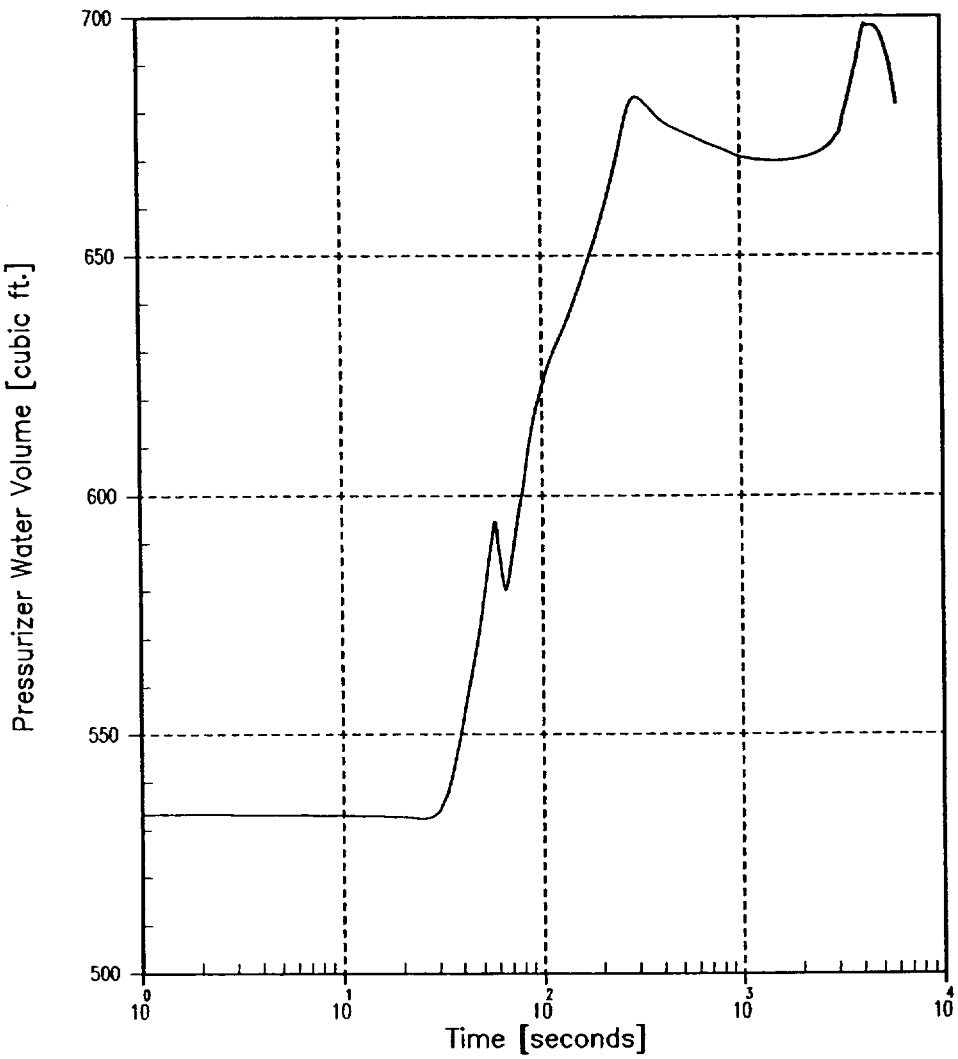

Figure 14.1.12-4

Loss of AC Power to the Plant Auxiliaries
Steam Generator Pressure vs. Time

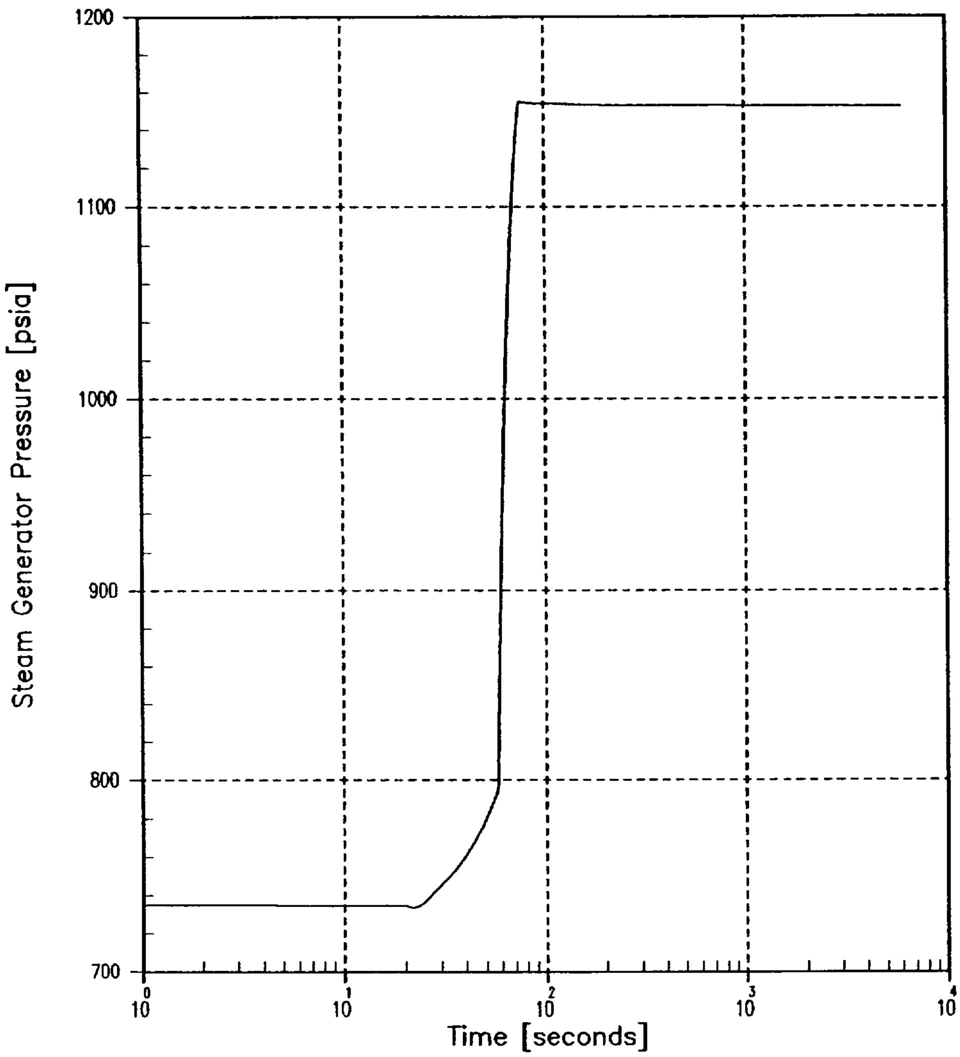

Figure 14.1.12-5

Loss of AC Power to the Plant Auxiliaries
Steam Generator Mass vs. Time

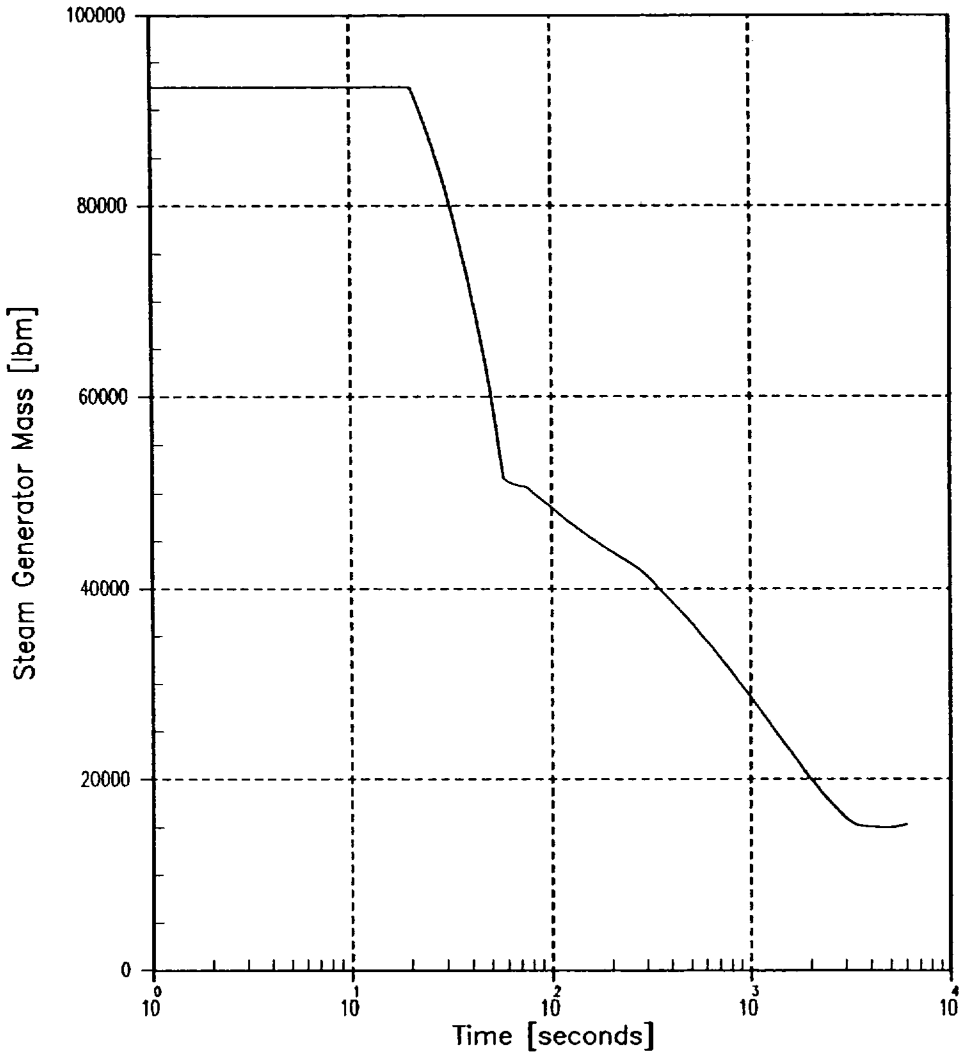

Figure 14.1.12-6

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14.2 STANDBY SAFETY FEATURES ANALYSIS

Adequate provisions have been included in the design of the plant, and its standby engineered safeguards to limit potential exposure of the public to below the guidelines of 10 CFR 100 for situations which have a very low probability of occurrence, but which could conceivably involve uncontrolled releases of radioactive materials to the environment. The situations, which have been considered, are:

- Fuel Handling Accidents
- Accidental Release of Waste Liquid
- Accidental Release of Waste Gases
- Rupture of a Steam Generator Tube
- Steam Line Break
- Rupture of a Control Rod Drive Mechanism Housing - RCCA Ejection
- Turbine Missile Damage to Spent Fuel Pool

14.2.1 FUEL HANDLING ACCIDENTS

NOT TA SCOPE

The following fuel-handling accidents are evaluated to ensure that no hazards are created:

a. A fuel assembly becomes stuck inside the reactor vessel

b. A fuel assembly or Rod Cluster Control Assembly (RCCA) is dropped onto the floor of the reactor refueling cavity or spent fuel pool

c. A fuel assembly becomes stuck in the penetration valve

d. A fuel assembly becomes stuck in the transfer tube or the carriage becomes stuck.

Causes and Assumptions

The possibility of a fuel handling incident of the severity considered in the analysis is very remote because of the many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in nuclear safety. Also, before any refueling operations begin, verification of complete RCCA insertion is obtained by weighing each control rod drive mechanism individually to verify that the control rods are disengaged from the control rod drive mechanisms. Boron concentration in the coolant is raised to the refueling concentration and verified by sampling. Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core sub-critical with all RCCAs withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications.

As the vessel head is removed, a visual check is made to verify that RCCA drive shafts are free of the mechanism housings.

NOT TA SCOPE

After the vessel head is removed, the RCCA drive shafts are disconnected from their respective assemblies using the manipulator crane and the shaft-unlatching tool. A spring scale is used to indicate that the drive shaft is free of the RCCA as the lifting force is applied.

The fuel handling manipulators and hoists are designed so that fuel cannot be raised above a position which provides adequate shield water depth for the safety of operating personnel. This safety feature applies to handling facilities in both the containment and in the spent fuel pool area. In the spent fuel pool, the design of storage racks and manipulation facilities is such that:

Fuel at rest is positioned by positive restraints in a safe, always sub-critical, geometrical array, with no credit for boric acid in the water.

Fuel can be manipulated only one assembly at a time.

Violation of procedures, by placing one fuel assembly in juxtaposition with any group of assemblies in racks does not result in criticality.

Crane facilities do not permit the handling of heavy objects, such as a spent fuel-shipping container, over the spent fuel storage area. A detailed description of crane movement limitations appears in Section 9.5.

Adequate cooling of fuel during underwater handling is provided by convective heat transfer to the surrounding water. The fuel assembly is immersed continuously while in the Refueling Cavity or Spent Fuel Pool.

Even if a spent fuel assembly becomes stuck in the transfer tube, the fuel assembly is completely immersed and natural convection maintains adequate cooling to remove the decay heat. The fuel handling equipment is described in detail in Section 9.5.

Two Nuclear Instrumentation System source-range channels are continuously in operation and provide warning of any approach to criticality during refueling operations. This instrumentation provides a continuous audible signal in the containment, and would annunciate a local horn and an annunciator in the plant control room if the count rate increases above a preset low level.

Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core sub-critical by at least 5% $\Delta k/k$ with all RCCAs inserted. At this boron concentration, the core would also be more than 2% sub-critical with all control rods withdrawn.

All these safety features make the probability of a fuel-handling incident very low. Nevertheless, it is possible that a fuel assembly could be dropped during the handling operations. Therefore, this incident is analyzed both from the standpoint of radiation exposure and accidental criticality.

Special precautions are taken in all fuel handling operations to minimize the possibility of damage to fuel assemblies during transport to and from the spent fuel pool and during installation in the reactor. All handling operations of irradiated fuel are conducted under

NOT TA SCOPE

water. The handling tools used in the fuel handling operations are conservatively designed and the associated devices are of a fail-safe design.

In the fuel storage area, the fuel assemblies are spaced in a pattern that prevents any possibility of a criticality accident.

The motions of the cranes, which move the fuel assemblies, are limited to a low maximum speed. Caution is exercised during fuel handling to prevent the fuel assembly from striking another fuel assembly or structures in the containment or fuel storage building.

The fuel handling equipment suspends the fuel assembly in the vertical position during fuel movements, except when the fuel is moved through the transport tube.

The design of the fuel assembly is such that the fuel rods are restrained by grid clips which provide a total restraining force of approximately 40 pounds on each fuel rod at the end of life. The force transmitted to the fuel rods during normal handling is limited to the (grid frictional) restraining force and is not sufficient to breach the fuel rod cladding. If the fuel rods are not in contact with the fuel assembly bottom nozzles, the rods would have to slide against the 40-pound friction force. This would dissipate an appreciable amount of energy and thus limit the impact force on the individual fuel rods.

If one assembly is lowered on top of another, no damage to the fuel rods would occur that would breach the cladding. Considerable deformation would have to occur before the fuel rods would contact the top nozzle adapter plate and apply any appreciable load to the rods. Based on the above, it is unlikely that any damage would occur to the individual fuel rods during handling.

If during handling and subsequent translational motion the fuel assembly should strike against a flat surface, the fuel assembly lateral loads would be distributed axially along its length with reaction forces at the grid clips and essentially no damage would be expected in any fuel rods.

Analyses have been made assuming that fuel assembly is dropped vertically and strikes a rigid surface and where one fuel assembly is dropped vertically on another. The analysis of a dropped fuel assembly striking a rigid surface considers the stresses in the fuel cladding and any possible buckling of the fuel rods between the grid supports. The results show that the buckling load at the bottom section of the fuel rod, which would receive the highest loading, is below the critical buckling load and the stresses are below the yield stress. For the case in which a fuel assembly is assumed to be dropped on top of another assembly, the impact load is transmitted through the top nozzle and the RCCA guide tubes of the struck assembly before any of the loads reach the fuel rods. As a result, a significant amount of kinetic energy is absorbed by the top nozzle of the struck assembly and bottom nozzle of the falling assembly, thereby limiting the energy available for fuel rod deformation. The results of this analysis indicated that the buckling load on the fuel rods is below the critical buckling load and stresses in the cladding are below yield.

Prototype fuel assemblies have been subjected to 3000 pounds of axial load without excessive lateral or axial deformation. The maximum column load expected to be experienced in service

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is approximately 1000 pounds. This information is used in the fuel handling equipment design to establish the limits for inadvertent axial loads.

For the purposes of evaluating the environmental consequences of a fuel-handling incident, a conservative upper limit of damage is assumed by considering the cladding rupture of all rods in one complete fuel assembly. The remaining fuel assemblies are so protected by the storage rack structure that no lateral bending loads would be imposed.

Activity Release Characteristics

For the assumed accident, there is a sudden release of the gaseous fission products held in the gap between the pellets and cladding of one fuel assembly. The low temperature of the fuel during handling operations precludes further significant release of gases from the pellets themselves after the cladding is breached. Molecular halogen release is also greatly minimized due to their low volatility at these temperatures. The strong tendency for iodine in vapor and particulate form to be scrubbed out of gas bubbles during their ascent to the water surface further reduces the quantity released from the water surface.

The fuel assembly gap activity was conservatively calculated assuming an initial heavy metal loading of 411 kg of Uranium at 5 weight percent U-235. The plant is assumed to be operated at 1721 MW_{th}. Additionally, all of the rodlets in the assembly are assumed to be operated at a peak radial power ratio of 1.70 and a range of burnups up to 60 GWD/MTU. Activity levels corresponding to 48 GWD/MTU were identified as limiting. The iodine gap activities are assumed to be 12% of total fuel iodine activity. The Kr-85 gap activity is assumed to be 30% of the total fuel Kr-85 activity. All other noble gas gap activities are assumed to be 10% of the total fuel noble gas activity. The noble gas and iodine fission products calculated to be present in the fuel rod gap at 100 hours following shutdown are given in Table D.3-2 of Appendix D.

In examining the expected behavior of fission product halogens released from the damaged fuel cladding gap to the spent fuel pool or reactor cavity water, it was predicted that a significant portion of the halogens would be absorbed into the solution from the bubbles and fixed gases. Early experiments indicate that gaseous iodine (admixed with fixed gases) is readily transferred to the aqueous boric acid solution and that a high stripping efficiency results in a pool decontamination factor (DF) on the order of 10^3, i.e., only 1/1000 of the incident halogen reaches the pool surface and is available for release to the environment.

Studies have been performed by Westinghouse to confirm this stripping efficiency with laboratory tests that better represent the conditions of the assumed accident (Reference 1). An experimental arrangement, consisting of a 9-inch diameter vertical column containing boric acid solution at a depth of up to 7.5 feet, was utilized in the study. At the bottom of the column, a gas injection vessel was provided to permit the introduction of a fixed gas (nitrogen) containing iodine vapor at the design basis concentration. The gas mixture was injected in the solution and the stripping efficiency for the molecular iodine was determined by inventory of the fraction, which escaped from the aqueous solution, compared to the quantity retained in the solution. The bubble size was controlled, as was the solution depth, so that the relationship between DF and these variables could be determined.

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The results of these studies indicate that the boric acid solution is an efficient getter for the iodine and that a DF for the 40-foot deep spent fuel pool is in excess of 500 for gas bubble diameters up to 1 inch.

As an extension of the laboratory studies, a full-scale fuel assembly mockup was fabricated to permit the examination of bubble diameters that would result from the damage of all fuel tubes in an assembly. The mockup simulated in exact detail the cross section of an assembly and was fabricated in a manner to permit the simultaneous release of gap gases from all fuel tubes into a deep (25 feet) water pool.

Close examination of the gas bubbles rising from the mockup demonstrates that, for the most part, the bubble diameters are near 1 inch and below. Some few exceptions are noted, resulting from coalescence of a fraction of the smaller bubbles giving rise to diameters larger than 1 inch, but only after several feet of bubble travel through the pool water. These large-scale tests confirm the high stripping efficiencies for the halogens and that DFs of greater than 500, as indicated by the quantitative laboratory-scale tests, are reasonably expected values.

Method of Analysis

The volatile gaseous activities associated with the fuel handling accident could be released either inside the Containment Building or in the Auxiliary Building. Both of these areas have ventilation systems in operation under administrative control during fuel handling operations. Radioactivity monitors provide continuous indication of radiation levels and signal evacuation of these areas on high alarm. The Containment Building high-level alarm automatically closes the purge supply and exhaust ducts. Administrative evaluation of the containment activity would determine when purging could be resumed. A high-level alarm on the Auxiliary Building Vent Monitor would automatically activate the Zone Special Ventilation (SV) System with subsequent absolute and charcoal filtration. This system is described in Section 9.

The fuel handling accident in containment with the personnel air lock doors open has been determined to be the limiting accident. In this analysis, all of the rods of one assembly (179 rodlets) are assumed to be damaged releasing the entire gap activity. Scrubbing of iodine by the borated water results in a decrease in the radioiodine activity available for release. A conservative value of 0.01 for scrubbing by the water is assumed. The activity released from the water surface mixes into the containment atmosphere. Following the containment ventilation system isolation, there is a limited driving force for a release to the environment even with the personnel air lock doors open. However, a conservative value for containment discharge of 6000 scfm is assumed. Additionally, a conservative value for containment volume is assumed to be 1.00E6 ft^3. Taking no credit for containment closure the release is assumed to continue for 2 hours at which point the release is stopped. Using these assumptions (i.e., volume and flow) a reduction of 0.5 is applied to the activity release.

Dispersion of this activity is computed using the Gaussian plume dispersion formula and taking credit for building wake dilution. A wind velocity of 1.5 meters per second is assumed to remain

NOT TA SCOPE

in one direction for the duration of the accident under Pasquill F conditions. The dispersion characteristics are discussed in Section 2.7 and curves, corrected for building wake effect by the volumetric source method, are presented on Figure 2.7-5. The site boundary, Exclusion Area Boundary χ/Q, dispersion factor is [redacted]E–4 sec/m^3. The Low Population Zone χ/Q is 3.977E–5 sec/m^{33}.

The dose to the thyroid has been determined to be limiting. With the ICRP 30 dose conversion factors applied and the above mentioned conservative inputs the thyroid doses at the site boundary and low population zone are 24.4 and 4.4 rem respectively.

Thus, it is concluded that a dropped fuel assembly would present no criticality hazard and would result in radiation levels at the site boundary and low population zone that are well below the 10 CFR 100 guidelines. This analysis was reviewed and approved by the NRC in Reference 2.

14.2.2 ACCIDENTAL RELEASE-RECYCLE OF WASTE LIQUID

Accidents in the Auxiliary Building that result in the release of radioactive liquids are those that involve the rupture or leaking of system pipe lines or storage tanks. The largest vessels are the three liquid holdup tanks, sized such that two tanks can hold more than one reactor coolant liquid volume, used to store the normal recycle or water fluids produced. The contents of one tank are passed through the liquid processing train while the other tanks are being filled.

All liquid waste components except the reactor coolant drain tank are located in the Auxiliary Building and any leakage from the tank or piping will be collected in the building sump to be pumped back into the liquid waste system. The building sump and basement volume are sufficient to hold the full volume of a liquid holding tank without overflowing to areas outside the building. This also is true for the tanks in the Auxiliary Building.

The holdup tanks are also equipped with safety pressure relief and designed to accept the established seismic forces at the site. Liquids in the Chemical and Volume Control System flowing into and out of these tanks are controlled by manual valve operation and governed by prescribed administrative procedures.

The volume control tank design philosophy is similar in many respects to that applied for the holdup tanks. Level alarms, pressure relief valves and automatic tank isolation and valve control assure that a safe condition is maintained during system operation. Excess letdown flow is directed to either the holdup tanks via the reactor coolant drain tank or the volume control tank. The waste holdup tank is a horizontal tank, which is continuously maintained at atmospheric pressure. Its vent is routed to the atmosphere through the Auxiliary Building exhaust ducts.

The potential hazard from these process or waste liquid releases is derived only from the volatilized components. The releases are described and their effects summarized in Section 14.2.3.

NOT TA SCOPE

The evaluation of the credibility of the accidental release of radioactive fluids above maximum normal concentration (4E–5 μCi/cc) from the Waste Disposal System discharge is based upon the following review of waste discharge operating procedure, monitoring function description, monitor failure mode and the consequences of a monitor failure.

The process for discharging liquid wastes is as follows:

a. A batch of waste is collected in one Steam Generator Blowdown Treatment tank (capacity 10,000 gal); other lesser volume tank(s) can be used and follow the same process;

b. The tank, or tanks, is (are) isolated;

c. The tank(s) contents are recirculated to mix the liquid;

d. A sample is taken for radiochemical analysis;

e. If analysis indicates that release can be made within permissible limits, the quantity of activity to be released is recorded on the basis of the liquid volume in the tank(s) and its activity concentration. Each tank or batch is assessed for its radiological impact prior to and/or after each release. If release can not be made within permissible limits, the waste is returned for additional cleanup. Then the process begins again.

f. To release the liquid, the last stop valve in the discharge line (which is normally locked shut) must be unlocked and opened; a second valve, which trips shut automatically on high radiation signal from the effluent monitor, must be opened manually; a pump for the tank being released must be started manually and a flow rate established. The release flow rate is set at or below the maximum release flow rate as listed on the Radiological Liquid Waste Discharge Permit. Liquid is now being pumped to the discharge canal.

As the operating procedure indicates, the release of liquid waste is under administrative control. The effluent monitor is provided to maintain surveillance over the release.

The effluent monitor is provided with the following features:

a. A check source is provided to permit the operator to check the operation of the monitor before discharge from the control room.

b. If the monitor falls off scale at any time, an alarm condition is indicated in the control room and the waste disposal discharge valve is tripped closed automatically.

c. If the AC power supply to the monitor fails, a high radiation alarm is annunciated. The trip valve also closes.

d. The normally closed radiation trip valve fails closed.

NOT TA SCOPE

It is concluded that the administrative controls imposed on the operator combined with the safety features built into the equipment provide a high degree of assurance against accidental release of waste liquids.

Should a complete failure of any tank located in the Auxiliary Building occur, its contents remains in this building. Any subsequent discharge of radioactive liquid to the lake is be conducted under the controls described above and does not result in activity concentrations in excess of the limits given in the Off-Site Dose Calculation Manual (ODCM).

Dilution of off-site liquid releases is discussed in Section 2.6.4.

14.2.3 ACCIDENTAL RELEASE-WASTE GAS

Gas Decay Tank Rupture

Causes and Assumptions

The gas decay tanks contain the gases vented from the Reactor Coolant System the volume control tank, and the liquid holdup tanks. Sufficient volume is provided in each of four tanks to store the gases evolved during a reactor shutdown. The system is adequately sized to permit storage of these gases for forty-five days prior to discharge.

This period is selected as the maximum foreseeable holdup time because in this period the shorter-lived radioactive gaseous isotopes received by the waste system will have decayed to a level, which is less significant than that of long-lived Kr^{85}.

The waste gas accident is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive xenon and krypton fission gases that are stored in the waste gas storage system. Failure of a gas decay tank or associated piping could result in a release of this gaseous activity. This analysis shows that even with the worst expected conditions, the off-site doses following release of this gaseous activity would be very low.

The leakage of fission products through cladding defects can result in a buildup of radioactive gases in the reactor coolant. Based on experience with other operational closed cycle, pressurized water reactors, the number of defective fuel elements and the gaseous activity in the coolant is expected to be low. The principal source of radioactive gases in the Waste Disposal System is the bleeding of effluents from the Reactor Coolant System.

Nonvolatile fission product concentrations are greatly reduced as the cooled Reactor Coolant System liquid is passed through the purification demineralizers. (The removal factor for iodine, for example, is at least 10). The decontamination factor for iodine between the liquid and vapor phases, for example, is expected to be on the order of 10,000. Based on the above analysis and operating experience at Yankee-Rowe and Saxton, activity stored in a gas decay tank consists of the noble gases released from the processed coolant with only negligible quantities of the less volatile isotopes.

NOT TA SCOPE

The components of the waste gas system are not subjected to any high pressures or stresses, are Class I design (see Appendix B), and are designed to the standards given in Table 11.1-2. A rupture or failure is highly unlikely. However, a rupture of a gas decay tank was analyzed to define the hazard caused by a malfunction in the radioactive waste disposal system.

Activity Release Characteristics

The activity in a gas decay tank is taken to be the maximum amount that could accumulate from operation with cladding defects in 1% of the fuel elements. This is at least ten times the expected number of defective fuel elements. The maximum activity is obtained by assuming the noble gases, xenon and krypton, are accumulated with no release over a full core cycle. This postulated amount of activity, one Reactor Coolant System equilibrium cycle inventory, is given in Appendix D. This value is particularly conservative because some of this activity would normally remain in the coolant, some would have been dispersed earlier through the vent, and the shorter-lived isotopes would have decayed substantially.

Samples taken from gas storage tanks in pressurized water reactor plants in operation show no appreciable amount of iodine.

To define the maximum doses, the release is assumed to result from gross failure of any process system storage tank, here represented by a gas decay tank giving an instantaneous release of its volatile and gaseous contents to the atmosphere.

Volume Control Tank Rupture

Causes and Assumptions

The volume control tank contains fission gases and low concentrations of halogens, which are normally a source of waste gas activity, vented to a gas decay tank. The iodine concentrations and volatility are quite low at the temperature, pH, and pressure of the fluid in the volume control tank. The same assumptions detailed in the preceding subsection apply to this tank. As the volume control tank and associated piping are not subjected to any high pressures or stresses, failure is very unlikely. However, a rupture of the volume control tank is analyzed to define the limit of the exposure that could result from such an occurrence.

Activity Release Characteristics

Rupture of the volume control tank is assumed to release all the contained noble gases and 1% of the halogen inventory of the tank plus that amount contained in the 40-gpm flow from the demineralizers, which would continue for up to five minutes before isolation would occur. The 1% halogen release is a very conservative estimate of the decontamination factor expected for these conditions.

Based on 1% fuel defects, the activities available for release are given in Table D.6-1.

NOT TA SCOPE

Method of Analysis

In calculating off-site plume centerline exposure, it is assumed that the activity is discharged to the atmosphere at ground level and is dispersed as a Gaussian plume downwind taking into account building wake dilution.

No credit is taken for the buoyant lift effect of the hydrogen present in the released gas. Dispersion coefficients based on the on-site meteorology program are used. A wind velocity of 1.5 meters per second is assumed to remain in one direction for the duration of the accident under Pasquill F conditions. The dispersion characteristics are discussed in Section 2.7.4 and curves corrected for building wake effects by the volumetric source method, are present on Figure 2.7-8.

The following parameters have been used in the dose assessment:

- A 0-2 hour χ/Q value of 2.23E−4 sec/m³
- Breathing rate equal to 3.47E−4 m³/sec
- The effective decay energies for noble gases found in Table D.8-1.
- The ICRP 2 thyroid dose conversion factors for iodine inhalation.
- The volume control tank specific activities are found in Table D.6-1.
- The gas decay tank activities are found in Table D.7-1.

Summary of Calculated Doses

The following tabulation summarizes the whole body and thyroid doses at the site boundary (exclusion distance), consistent with a receptor on the plume centerline.

	Thyroid Dose	Whole Body Dose
Gas Decay Tank Rupture	Negligible	0.327 rem
Volume Control Tank Rupture	1.06E−3 rem	0.082 rem
10 CFR 100 Guidelines	300 rem	25 rem

It is concluded that a rupture in the waste gas system or in the volume control tank would present no undue hazard to public health and safety.

14.2.4 STEAM GENERATOR TUBE RUPTURE

Accident Description

The accident examined is the complete severance of a single steam generator tube (for additional information on steam generator tubes, see Sections 14.3.10 and 14.3.11), with the reactor at power. This accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the Reactor Coolant System. In the event of a coincident loss of off-site power, or failure of the condenser dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power operated relief valves.

NOT TA SCOPE

The activity that is available for release from the system is limited by:

1. Activities in the steam generator secondary that are a consequence of operational leakage prior to the complete tube rupture.

2. The activity concentration in the reactor coolant, which is conservatively assumed to arise from 1% defective fuel clad.

3. Operator actions to isolate the mixed primary and secondary leakage to atmosphere.

The steam generator tube material is Inconel 600 and, as the material is highly ductile, it is considered that the assumption of a complete severance is conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the Steam and Power Conversion System is subject to continuous surveillance and an accumulation of minor leaks that cause the activity to exceed the limits established in the Technical Specifications is not permitted during reactor operation.

The operator determines that a steam generator tube rupture has occurred, and identifies and isolates the faulty steam generator on a restricted time scale in order to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the faulty unit. The recovery procedure is carried out on a time scale that ensures that break flow to the secondary system is terminated before water level in the faulty steam generator rises into the main steam line. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily. Consideration of the indications provided on the control board together with the magnitude of the break flow leads to the conclusion that the isolation procedure can be completed within thirty minutes of accident initiation.

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube rupture:

1. Pressurizer low-pressure and low-level alarms are actuated and, prior to plant trip, charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side there is a steam flow/feedwater flow mismatch before trip, as feedwater flow to the faulty steam generator is reduced due to the additional break flow which is now being supplied to that generator.

2. Loss of reactor coolant inventory leads to falling pressure and level in the pressurizer until a reactor trip signal is generated by low pressurizer pressure. Resultant plant cooldown following reactor trip leads to a rapid change of pressurizer level, and the safety injection signal, initiated by low pressurizer pressure, follows soon after the reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates auxiliary feedwater addition; as discussed in Section 6.6; manual initiation of auxiliary feedwater may be required at low power levels.

3. The steam generator blowdown liquid monitor and the air-ejector radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system.

NOT TA SCOPE

4. The plant trip automatically shuts off steam supply to the turbine and if off-site power is available the condenser steam dump valves open permitting steam dump to the condenser. In the event of a coincident loss of off-site power, the condenser steam dump valves automatically close to protect the condenser. The steam generator pressure rapidly increases resulting in steam discharge to the atmosphere through the steam generator safety and/or power-operated relief valves.

5. Following plant trip, the continued action of auxiliary feedwater supply and borated safety injection flow (supplied from the Refueling Water Storage Tank) provide a heat sink, which absorbs some of the decay heat. Thus, steam bypass to the condenser, or in the case of loss of off-site power, steam relief to atmosphere, is attenuated during the thirty minutes in which the recovery procedure leading to isolation is being carried out.

6. Safety injection flow results in increasing pressurizer water level. The time after trip at which the operator can clearly see returning level in the pressurizer is dependent upon the amount of operating auxiliary equipment.

Results

In determining the mass transfer from the Reactor Coolant System through the broken tube, several conservative assumptions are made as follows:

a. Plant trip occurs automatically as a result of low pressurizer pressure.

b. Following the initiation of the Safety Injection Signal, both Safety Injection Pumps are actuated and continue to deliver flow for thirty minutes.

c. After plant trip the break flow equilibrates at the point where incoming safety injection flow is balanced by outgoing break flow as shown in Figure 14.2.4-1. The resultant break flow persists from plant trip until thirty minutes after the accident.

d. The steam generators are controlled at the safety valve setting rather than the power-operated relief valve setting.

e. The operator identifies the accident type and terminates break flow to the faulty steam generator within thirty minutes of accident initiation.

The above assumptions lead to a conservative estimate of 120,000 lbs. for the total amount of reactor coolant transferred to the faulty steam generator as a result of a tube rupture accident.

Environmental Consequences of a Tube Rupture

The occurrence of a steam generator tube rupture, followed by immediate loss of off-site electrical power, has an extremely low probability. The effects have, however, been analyzed and the results show that the public health and safety are not endangered.

NOT TA SCOPE

The chronology of events subsequent to the tube failure is discussed above.

In assessing the consequences of the assumed accident, the inventory of halogens and noble gases available for release from the faulty steam generator is based on the following:

1. The activity concentration in the reactor coolant is assumed to arise from continuous operation with 1% defective fuel clad (see Table D.4-1).

2. It is assumed that the plant has been operating with a 5 gpm primary to secondary leak rate for a period of time sufficient to establish radionuclide equilibrium in the secondary loop without credit for blowdown treatment.

3. The corresponding iodine activity in the secondary system is the sum of the equilibrium value due to the pre-existing tube leak and that amount transferred with the reactor coolant due to the complete tube rupture. The value of this activity is 209 Ci of I^{131} equivalent, which is assumed at a uniform concentration in the secondary coolant.

4. The total noble gas inventory available for release is the sum of the equilibrium value due to the pre-existing tube leak, and that amount transferred with reactor coolant due to the complete tube rupture, and is equal to 21,700 Ci of Xe^{133} equivalent.

5. All releases are made by atmospheric steam dump from the faulty steam generator with an assumed decontamination factor of 10 applied to the iodines.

In calculating off-site plume centerline exposure, it is assumed that the activity is discharged to the atmosphere at ground level and is dispersed as a Gaussian plume downwind taking into account building wake dilution.

Dispersion coefficients based on the on-site meteorology program are used. A wind velocity of 1.5 meters per second is assumed to remain in one direction for the duration of the accident under Pasquill F conditions. The dispersion characteristics are discussed in Section 2.7.4 and curves corrected for building wake effects by the volumetric source method, are presented in Figure 2.7-8.

The following parameters have been used in the dose assessment:

- A 0-2 hour χ/Q value of 2.23E–4 sec/m^3
- Breathing rate equal to 3.47E–4 m^3/sec.
- An I^{131} equivalent dose conversion factor equal to 1.48E+6 rem/curie.
- A Xe^{133} dose conversion factor equal to 5.27E–2 rem–m^3/curie/sec.

Summary of Calculated Doses

The following tabulation summarizes the two-hour whole body and thyroid doses at the exclusion distance, consistent with a receptor on the plume centerline.

NOT TA SCOPE

	Thyroid Dose	Whole Body Dose
Steam Generator Tube Rupture	2.4 rem	0.26 rem
10 CFR 100 Guidelines	300 rem	25 rem

It is concluded that the complete failure of a steam generator tube preceded by a long-term leak history prior to its failure would present no undue hazard to public health and safety.

In 1992, Westinghouse completed a study (Reference 3) addressing the radiological consequences of steam generator tube bundle uncovery coincident with a steam generator tube rupture, following a reactor trip. The results of the study indicated that there was little effect on radiological release due to tube uncovery, and that the 10 CFR 100 limits continued to be met. It was concluded that steam generator tube uncovery did not have significant impact on the accident analysis for steam generator tube rupture, and that no modifications to the analysis were necessary. A Westinghouse letter (Reference 4) transmitted the Westinghouse and NRC resolution stating that the issue was closed.

Recovery Procedure

The immediately apparent symptoms of a tube rupture accident such as falling pressurizer pressure and level, and increased charging pump flow are also symptoms of small steam-line breaks and loss of coolant accidents. [redacted] is therefore important for the operator to determine that the accident is a rupture of a steam generator tube to carry out the correct recovery procedure. The steam generator tube rupture is uniquely identified by high condenser air ejector radiation, high steam generator blowdown radiation, high steam line radiation, and [redacted] feedwater flow to the ruptured steam generator before the reactor trip. When the operators observe these indications, they enter the steam generator tube rupture recovery procedure.

The operators perform the following steps, which lead to isolation of the ruptured steam generator and termination of the leak.

1. Identify the ruptured steam generator by observing a higher level or higher radiation levels in one steam generator.

2. Isolate the ruptured steam generator by closing the main steam isolation valve and other smaller valves.

3. Stop auxiliary feedwater flow to the ruptured steam generator when the narrow range level returns to scale.

4. Control auxiliary feedwater flow in the intact steam generator so that the narrow range level remains on scale.

5. If off-site power is available, use condenser steam dumps to cool the Reactor Coolant System to enable RCS pressure to be reduced below the pressure of the ruptured steam generator. If

off-site power is not available, atmospheric steam dumps or steam generator power-operated relief valves are used.

6. If off-site power is available, depressurize the RCS to below the pressure of the ruptured steam generator using pressurizer spray valves. If off-site power is not available, the reactor coolant pumps would not be running, making spray unavailable. In this case pressurizer power-operated relief valves or auxiliary spray are used for the depressurization.

7. Stop safety injection pumps.

8. Cool the Reactor Coolant System to cold shutdown. The ruptured steam generator is depressurized by either backfill into the RCS, blowdown into the Steam Generator Blowdown Treatment System, or steam dump into the condenser or atmosphere.

OUT OF SCOPE

After the Residual Heat Removal System (RHR) is in operation, the condensate accumulated in the secondary system can be sampled and processed.

There is ample time to carry out the above recovery procedure such that isolation of the ruptured steam generator is established before water level rises into the main steam lines. The available time scale is improved by the termination of auxiliary feedwater flow to the faulty steam generator. Normal operator vigilance therefore assures that excessive water level is not attained.

14.2.5 STEAM LINE BREAK

No changes

Accident Description

A steam-line break results in an uncontrolled steam release from a steam generator. The steam release results in an initial increase in steam flow, which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of coolant temperature and pressure. In the presence of a negative coolant temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive RCCA is assumed stuck in its fully withdrawn position, there is an increased probability that the core becomes critical and returns to power. A return to power following a steam line break is a potential problem mainly because of the high hot channel factors that exist when the most reactive RCCA is assumed stuck in its fully withdrawn position. Assuming the most pessimistic combination of circumstances, which could lead to power generation following a steam-line break, the core is ultimately shut down by boric acid injection delivered by the Emergency Core Cooling System.

The analysis of a steam line break is performed to demonstrate that:

1. Assuming a stuck RCCA, with or without off-site power, and assuming a single failure in the engineered safety features, there is no consequential damage to the primary system and the core remains in place and intact.

2. Energy release to the containment from the worst steam line break does not cause failure of the containment structure.

~~3. There will be no return to criticality after reactor trip, for a break equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief or safety valve.~~

Add USAR Insert 14.2.5-1

The following systems and components provide the necessary protection against steam line breaks:

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

   - Two-out-of-three low pressurizer pressure signals.
   - Two-out-of-three low-pressure signals in either steam line.
   - Two-out-of-three high containment pressure signals.

2. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the Safety Injection Signal.

3. Redundant isolation of the main feedwater lines: Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action, which closes the main feedwater valves, a safety injection signal rapidly closes all feedwater control valves, trips the main feedwater pumps, and closes the feedwater pump discharge valves.

4. Trip of the fast-acting main steam isolation valves (MSIVs). These valves are designed to close in less than five seconds on:

   - The coincidence of a Safety Injection Signal with either Hi-Hi steam flow from the respective steam line (one-out-of-two per line) or Hi steam flow from the respective steam line (one-out-of-two per line) in coincidence with Lo-Lo T_{avg} (two-out-of-four).
   - Two-out-of-three Hi containment pressure signals.

   NOT TA SCOPE

   Each steam line has a fast-closing MSIV with a downstream non-return check valve. These four valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For example, in the case of a break upstream of the MSIV in one line, closure of either the non-return check valve in that line or the MSIV in the other line will prevent blowdown of the other steam generator. This arrangement precludes blowdown of more than one steam generator inside the containment and thus prevents structural damage to the containment.

5. Each ~~main~~ steam generator has ~~line incorporates~~ a ~~16-inch diameter venturi type~~ flow restrictor that is integral to its outlet nozzle. ~~which is located inside the containment.~~ The ~~venturi~~ flow restrictors serve to limit the rate of release of steam.

USAR Insert 14.2.5-1

A limited amount of fuel failure is not prohibited in order to ensure that the core remains in place and intact. However, the steam line break analysis described herein demonstrates compliance with the more restrictive acceptance criterion of no DNB for any break assuming the most reactive rod stuck in its fully withdrawn position.

Analysis - Core Response

The analysis of the steam line break has been performed to determine:

1. The core heat flux and Reactor Coolant System temperature and pressure resulting from the cooldown following the steam line break. A full plant digital computer simulation has been used.

2. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital computer calculation has been used to determine if DNB occurs for the core conditions computed in (1) above.

The following conditions are assumed to exist at the time of a steam-line break.

1. 1.3 ~~2~~% end-of-life shutdown margin at no-load, equilibrium xenon conditions, with the most reactive rod stuck in its fully withdrawn position.

2. The negative moderator coefficient corresponding to the end-of-life rodded core with the most active rod in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The k vs. temperature relationship corresponding to the negative moderator temperature coefficient used is shown in Figure 14.2.5-1. In computing the power generation following a steam-line break, the local reactivity feedback from the high neutron flux in the region of the core near the stuck control rod has been included in the overall reactivity balance. The local reactivity feedback is composed of Doppler reactivity from the high fuel temperatures near the stuck control rod and moderator feedback from the high water enthalpy near the stuck rod. The effect of power generation in the core on total core reactivity is shown in Figure 14.2.5-2.

3. Minimum capability for injection of high concentration boric acid solution corresponding to the most restrictive single failure in the Safety Injection System. This corresponds to the flow delivered by one high-head Safety Injection Pump (see Figure 14.2.5-3). Boric acid concentration delivered to the reactor coolant loops corresponds to the minimum concentration of the Refueling Water Storage Tank, which is 2400 ppm. In addition, the passive accumulators were modeled with a minimum boron concentration of 1850 ppm.

4. To maximize the reactor cooldown steam generator tube plugging is at 0% ~~and a conservatively high reactor coolant system flow is assumed.~~

5. Hot channel factors corresponding to the worst stuck rod at end-of-core life. The hot channel factors depend upon the core power, temperature, pressure, and flow.

6. Two steam line break scenarios ~~Three break sizes~~ have been considered in determining the core response transient.

   a) Complete severance of a pipe ~~downstream of the steam flow restrictor,~~ with the plant initially at no-load conditions and ~~both reactor coolant pumps running.~~ offsite power available.

b) Complete severance of a pipe ~~inside the containment at the outlet of the steam generator,~~ with the plant at no-load conditions and ~~both reactor coolant pumps running.~~ with a loss of offsite power.

~~c) A break equivalent to the inadvertent opening of one steam generator safety valve at 1100 psi from one steam generator with off site power available.~~

~~d)~~ Initial hot shutdown conditions were considered for all of the above cases since this represents the most pessimistic initial condition for ~~the accident~~ maximizing the return-to-power after reactor trip. Should the reactor be critical at the time of a steam-line break, it would be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power, the reactor coolant system contains more stored energy than at no-load, the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam-line break before the no-load conditions of reactor coolant system temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis that assumes no-load condition at time zero. However, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of the Reactor Coolant System cooldown are less for steam breaks occurring at power.

7. Steam break flow is given by $W_{Break} = A_{Break}(t) * G(h, P)$ where $A_{Break}(t)$ is the specified break area as a function of time and $G(h, P)$ is the mass flow rate per unit area. G is evaluated from Moody flow tables (Reference 5). For a break in the steam line perfect moisture separation in the steam generator is assumed.

8. The non-return check valves were neglected to conservatively allow blowdown from both steam generators up to the time of steam line isolation.

REPLACE WITH USAR Insert 14.2.5-2

Results

The results presented are a conservative indication of the events that would occur assuming a main steam-line break, since it is postulated that all of the conditions described above occur simultaneously.

Figures [redacted] 14.2.5-4 through [redacted] 14.2.5-8 show the results following a main steam-line break (complete severance of the pipe) ~~downstream of the flow restrictor,~~ at initial no-load conditions and with outside power available. ~~Core heat flux increases and is stabilized by the negative reactivity feedbacks from rising fuel temperatures and increased enthalpy in the region of the stuck rod.~~

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

Figures [redacted] 14.2.5-9 through [redacted] 14.2.5-13 show results from a main steam-line break (complete severance of the pipe) at ~~the exit of a steam generator upstream of the flow restrictor. The sequence of events is similar to those~~ initial no-load conditions and with a loss of offsite power.

REPLACE WITH USAR INSERT 14.2.5-3

~~described above except that criticality is attained earlier due to the more rapid cooldown, and a higher maximum core average heat flux is attained.~~

~~Figures 14.2.5-14 through 14.2.5-18 show the transient resulting from a break equivalent to an inadvertent opening of one S/G safety valve at 1100 psi from one steam generator.~~

A detailed DNBR analysis is performed for the case with offsite power available. ~~break upstream and break downstream of the flow restrictor cases.~~ DNBR is calculated for the core conditions that existed at the time of maximum core heat flux. A conservatively high value for hot channel factor (F_H) is also assumed. ~~The following table shows the comparison of the important calculated safety parameters to its acceptance criteria (Calculated Value/Acceptance Criterion):~~

REPLACE WITH USAR INSERT 14.2.5-4

~~MSLB~~	~~MDNBR~~
~~Downstream of flow restrictor~~	~~3.106/1.45~~
~~Upstream of flow restrictor~~	~~[redacted]/1.45~~

Analysis - Containment Response

OUT OF SCOPE

There are four major factors that influence the release of mass and energy following a steam-line break. These are the initial steam generator fluid inventory, primary to secondary heat transfer, protective system operation, and the state of the secondary fluid blowdown. The following is a list of those plant variables that determine the influence of each of these factors.

- Plant Power Level
- Main Feedwater System Design
- Auxiliary Feedwater System Design
- Break Type, Area, Location
- Availability of Offsite Power
- Steam Generator Design
- Safety System Failures
- SG Reverse Heat Transfer and Reactor Coolant System Metal Heat Capacity

All of these variables are considered in the analyses and are conservatively selected based on Kewaunee plant design.

Steam-line break analysis cases are described based on a specific set of five parameters in the following manner:

1. Power Level: 0, 30, 70, and 102% for the rated power level.

2. Break Size and Location:

   a) Location is either upstream or downstream of the steam line flow restrictor. Upstream case is a large (4.29 ft^2) break; downstream cases are 1.4 ft^2 or less.

USAR Insert 14.2.5-2

In addition to negative reactivity feedback from rising fuel temperatures and increased enthalpy in the region of the stuck rod, boron injection from both the passive accumulators and one high-head safety injection pump act to limit the peak core heat flux.

USAR Insert 14.2.5-3

The peak core heat flux calculated for this case is significantly less than that calculated for the case with offsite power available. The loss of offsite power causes the reactor coolant pumps to trip, which reduces the primary-to-secondary heat transfer capability, and thus lessens the severity of the reactor coolant system cooldown.

USAR Insert 14.2.5-4

The case with a loss of offsite power is bounded by the case with offsite power available. The analysis demonstrates that the DNB design basis is satisfied for each main steam line break case, i.e., the minimum DNBR is greater than the limit value.

b) Break size:

OUT OF SCOPE

- Break downstream of the flow restrictor with an effective area equal to that of the flow restrictor ($A_{FR} = 1.4 \text{ ft}^2$).
- Break downstream of the flow restrictor with an area less than A_{FR}. (Break areas ranging from 1.4 ft^2 to 0.1 ft^2 are considered)
- Split break with maximum area for which MSIV isolation results from a containment signal and entrainment does not occur.
- Break upstream from the flow restrictor with an area equal to the SG outlet nozzle (4.29 ft^2) for the broken SG and equal to A_{FR} for the other SG.

3. Single Failures: There are three single failures which are:

- One Feedwater (FW) Regulating Valve fails to isolate. This is denoted as R.
- One Main Steam Isolation Valve (MSIV) fails to isolate. This is denoted as M.
- One Containment Safeguards Train (one containment safeguard train is: one internal containment spray train and two containment fan cooler units) fails to activate. This is denoted as N.

4. Off-Site Power: Cases with and without the availability of off-site power are considered.

5. Entrainment: The quality of steam exiting the break is explicitly modeled and is dependent on break size and power level.

Based on the above parameters, steam-line break analysis cases are designated as follows:

- Break Size (Units of ft^2)
- Single Failure
  R - FW Reg Valve Failure
  M - MSIV Failure
  N - Containment Safeguards System Failure
- Off-Site Power
  Y - Yes
  N - No
- Entrainment
  Y - Yes
- Power Level
  0 - 0%
  3 - 30%
  7 - 70%
  2 - 102%

OUT OF SCOPE

For identification purposes, the cases are represented by a six number/letter identification tag. For example:

14NYY3 represents the steam line break case with:

14	=	1.4 ft^2 break
N	=	single active failure is one containment safeguards train
Y	=	off-site power is available
Y	=	entrainment is modeled
3	=	initial power level is 30%

Further descriptions of the methods for steam-line break analysis follow:

1. The main feedwater flow is calculated using the following assumptions:

   a) The feedwater pumps are running at full speed at the start of the transient and are tripped off on the safety injection signal. A conservative flow coastdown is modeled.

   b) The condensate pumps are running at full speed throughout the transient.

   c) The regulating valve for the unfaulted Loop remains at its initial position until the time at which it strokes to its fully-closed position at a rate of 5%/sec following an isolation signal. At that time, the valve is closed instantaneously.

   d) The behavior of the regulating valve for the faulted loop is assumed to begin opening at t = 0.0 sec at an 8%/sec rate until the time the isolation signal occurs. It is held at that position until the time at which it strokes to its fully-closed position at a rate of 5%/sec following the isolation signal. At that time, the valve is closed instantaneously. For cases with a regulating valve failure, isolation is produced by closure of the FW isolation valve. The assumption used for the isolation valve is that it begins to close, at the time of the isolation signal, from full-open at a rate of 1.18%/sec. The initial opening of the regulating valve and the instantaneous FW isolation valve closure at the end of the stroke time are the same as for the case without a regulating valve closure failure.

2. The auxiliary feedwater (AFW) flow split between the two SGs is modeled the AFW is initiated, prior to the time for the activation signal, at full capacity and using a conservatively high enthalpy. All three AFW pumps are assumed to be operating.

3. The core physics parameters are based on a bounding set corresponding to end-of-cycle conditions and minimum technical specification shutdown requirements. The scram worth includes having the most reactive rod stuck out.

4. The dynamic reactor coolant pump model is used, which includes the gravity head and pump heat effects.

5. Conservative setpoints and time delays are used throughout.

OUT OF SCOPE

6. No credit is taken for charging flow.

7. No credit is taken for SG tube plugging.

8. The following considerations are made in modeling the steam lines.

   a) The pressure balancing line is modeled to allow communication between the steam lines in an unrestricted manner.

   b) Main steam isolation for the unfaulted loop is assumed to occur instantaneously at the time required for the non-return check valve to close in the faulted loop, which is 5 seconds after the break occurs.

   c) MSIV failure is modeled as a failure of the non-return check valve in the faulted loop. Steam flow from the unfaulted loop continues until the MSIV in the unfaulted main steam line closes. A closure assumption of 5 seconds is used for the MSIV. The time from the event initiation until MSIV closure signal receipt, plus signal instrumentation delays as applicable to the accident sequence analyzed, is added to the 5 second MSIV closure time assumption. At the time of the MSIV closure, the entire faulted and unfaulted loop steam lines from the MSIV to the turbine and the pressure balancing line are added to the total fluid mass and energy input to containment.

9. Entrainment analysis methods are used to obtain the time dependent quality of the faulted steam-line break flow which is power level and break size dependent. The quality of the unfaulted steam line break flow is conservatively assumed to be 1.0.

10. The turbine is tripped at t = 0.0 seconds for 0% power cases, and prior to or at the actual time of reactor trip for at power cases. These are conservative assumptions that maximize the available steam for blowdown.

11. A constant containment back pressure of 14.7 psia is conservatively assumed in all cases.

12. A conservatively high RCS flow rate is assumed.

13. Steam generator fluid inventory is maximized. Initial steam generator water level is 50.0% nominal narrow range level for all cases.

Results

Figures 14.2.5-19 and 14.2.5-20 present containment pressure and temperature responses for the limiting containment response steam line break analysis cases. The table below shows, for these limiting cases, the peak calculated containment pressure, temperature and the corresponding acceptance criteria. All cases analyzed result in a maximum containment pressure that is less than the containment design pressure limit of 60.7 psia. In addition, the limiting containment temperature profile has been evaluated and it does not create an equipment qualification concern. Although the limiting temperature profile exceeds the containment design temperature of 268°F, containment structural limits are not exceeded. The short duration of the temperature spike and the method of heat transfer to the containment shell precludes shell temperature from exceeding the design temperature.

OUT OF SCOPE

MSLB	Containment Peak Pressure (psia)	Containment Peak Temperature (°F)
14MYY0	60.5/60.7	267.7/330.0
01NYY0	39.3/60.7	298.9/330.0

Conclusions

The analyses have shown that the main steam line break acceptance criteria are satisfied.

Although DNB and possible clad perforation are not precluded in the acceptance criteria, the safety analysis has demonstrated that DNB does not occur~~, provided that core $F_{\Delta H}$ under steam line break conditions is ≤ 5.00.~~

The peak pressure for the limiting containment response cases can not exceed the containment design pressure. The limiting temperature profile also does not create an environmental qualification concern for equipment in containment.

OUT OF SCOPE

Based on the preceding analyses, the radiological significance of a steam line break would depend on the activity levels in the secondary loop of the failed steam generator. The consequence of a long-term 5-gpm leak rate has been considered in Section 14.2.4. However, even if it is conservatively assumed that all of the reactor coolant activity associated with 1% defective fuel cladding is suddenly expelled into the steam generator the resultant thyroid dose at the site boundary would be 4.7 rem and the resultant whole body dose would be 0.51 rem. A decontamination factor of 10 has been applied to the iodine inventory. The consequences of these postulated accidents are well below the guidelines of 10 CFR 100.

14.2.6 RUPTURE OF A CONTROL ROD DRIVE MECHANISM HOUSING (RCCA EJECTION)

Description of Accident

This accident is a result of an extremely unlikely mechanical failure of a control rod mechanism pressure housing such that the Reactor Coolant System pressure would then eject the RCCA and drive shaft. The consequences of this mechanical failure, in addition to being a minor loss-of-coolant accident, may also be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage for severe cases. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high neutron flux signals.

Certain features in Westinghouse pressurized water reactors are designed to preclude the possibility of a rod ejection accident, and to limit the consequences if the accident were to

Main Steam Line Break Variation of Reactivity with Core Temperature at 1000 psia for EOL Rodded Core with One Stuck Rod (Zero Power)

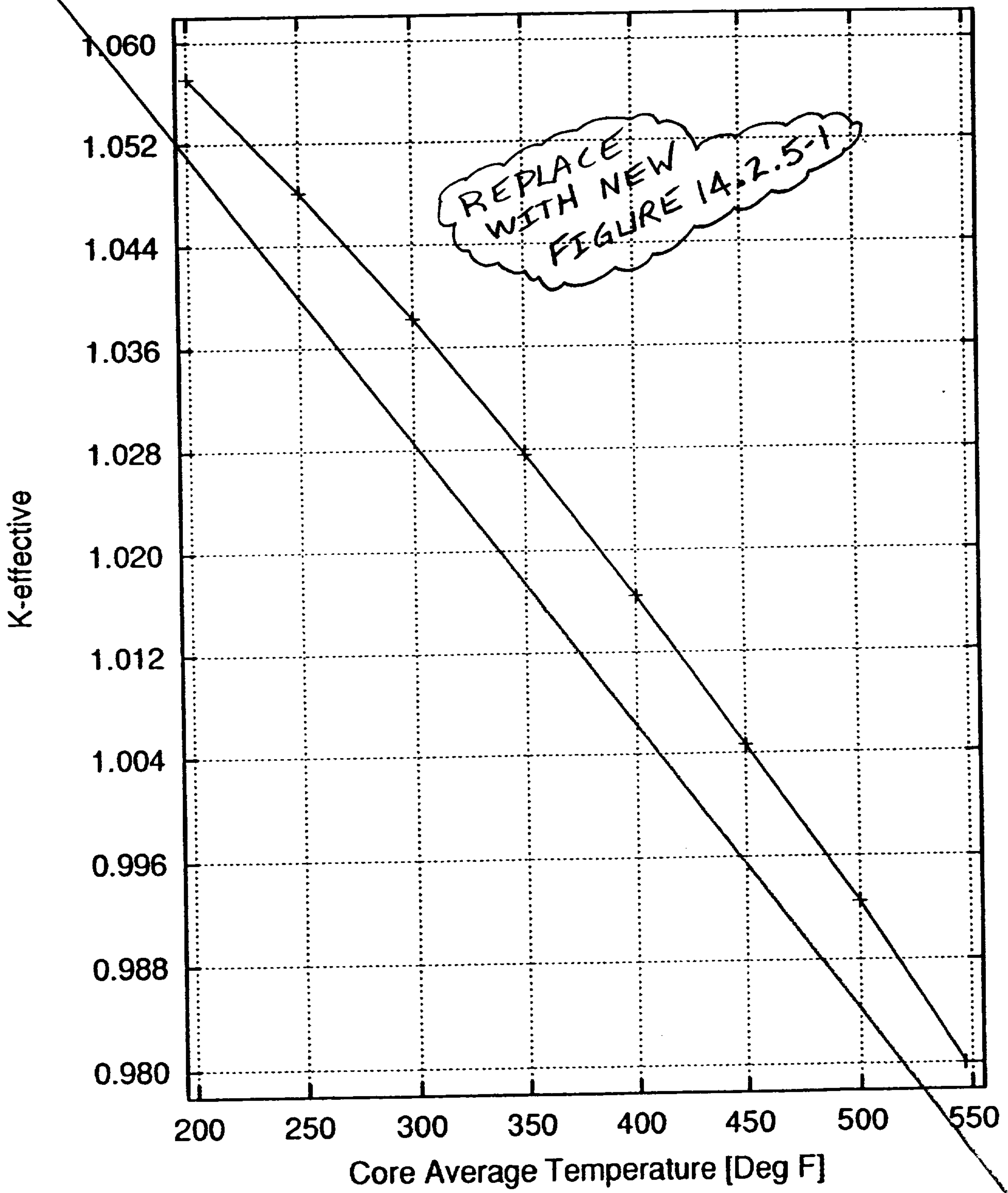

Figure 14.2.5-1

Rev. 16
12/01/2000

MAIN STEAM LINE BREAK

VARIATION OF REACTIVITY WITH POWER AT CONSTANT CORE AVERAGE TEMPERATURE

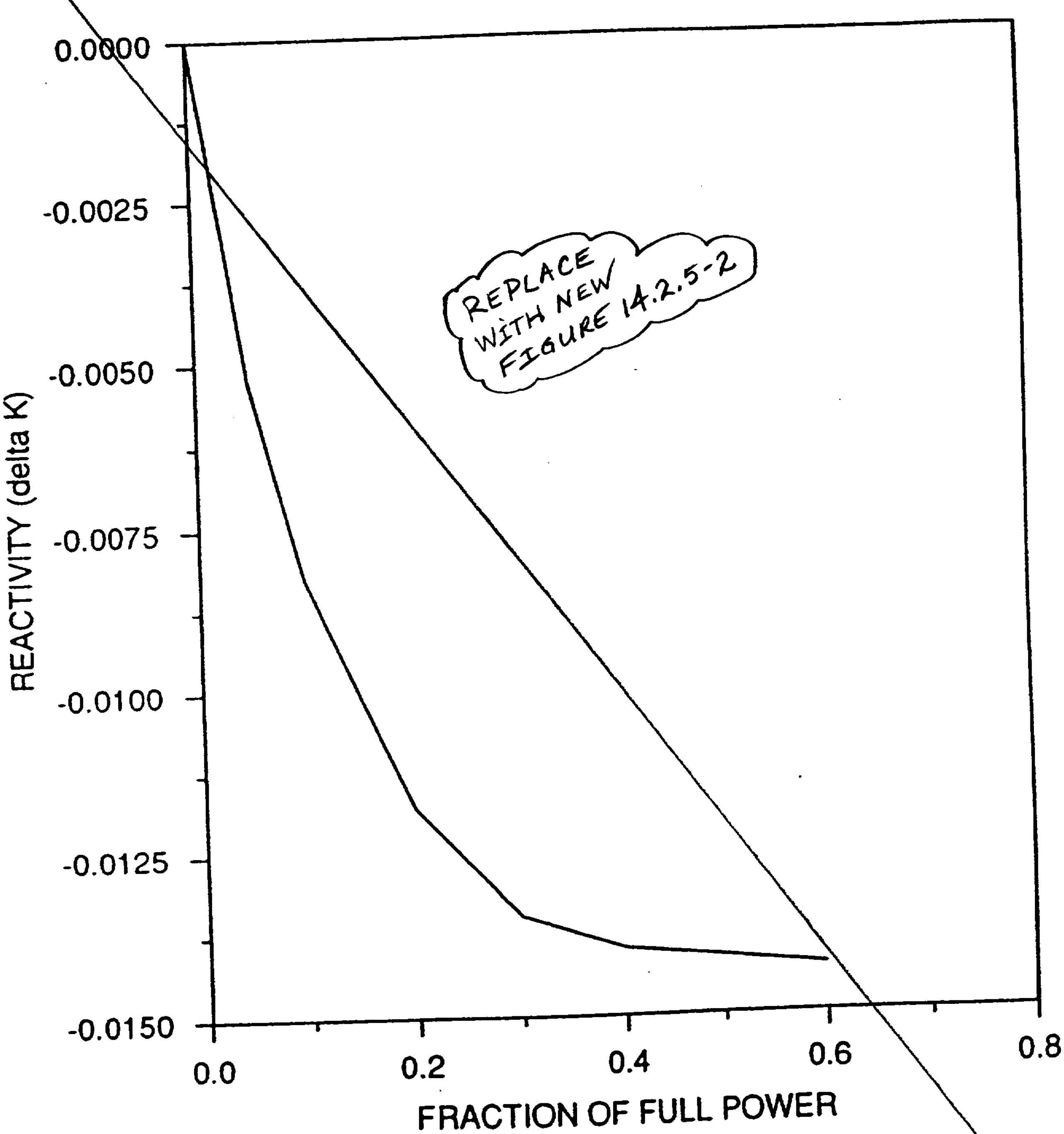

FIGURE 14.2.5-2

Rev. 14
NOV 01 1997

MAIN STEAM LINE BREAK

SAFETY INJECTION FLOW RATE VS REACTOR COOLANT PRESSURE

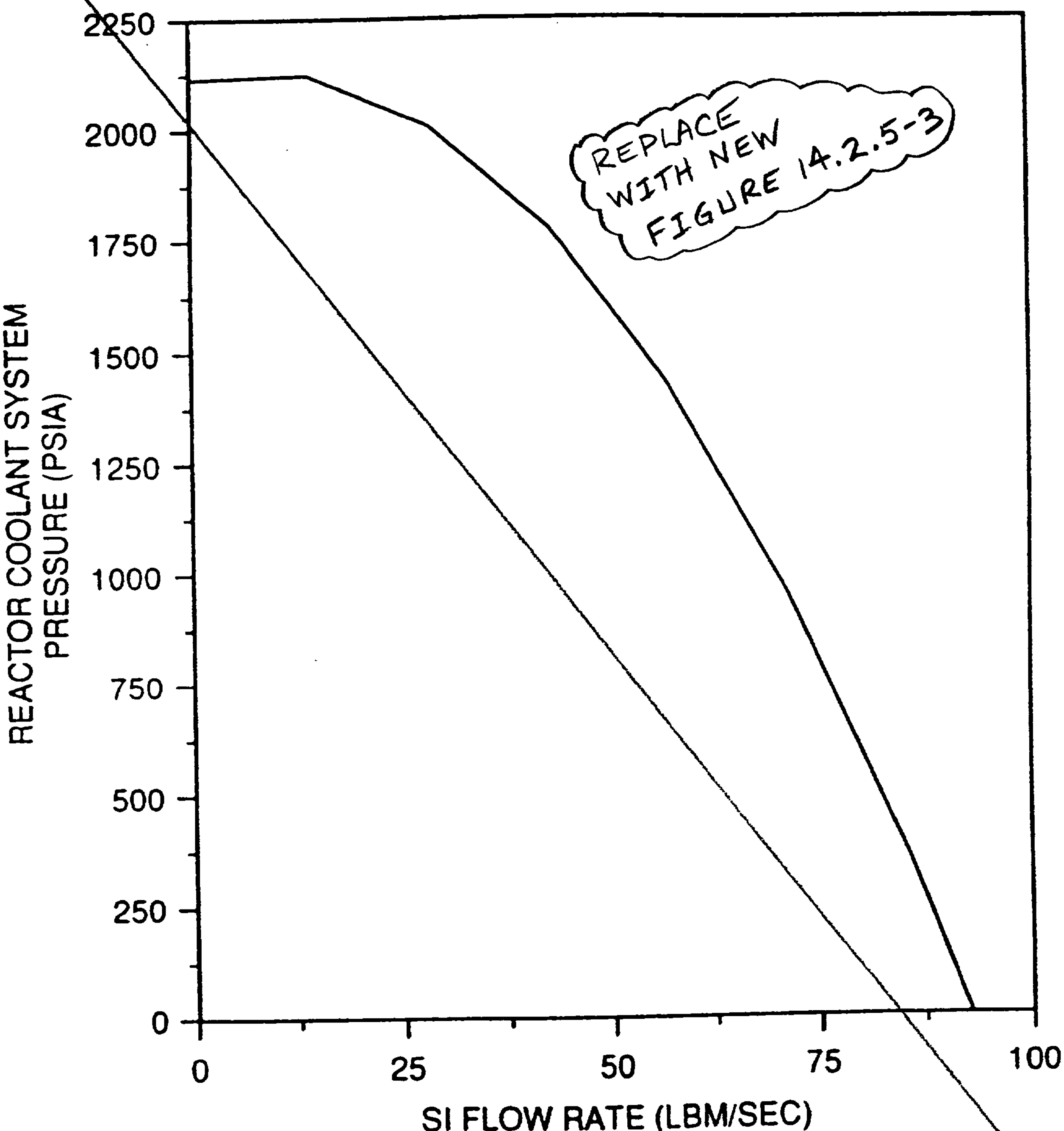

FIGURE 14.2.5-3

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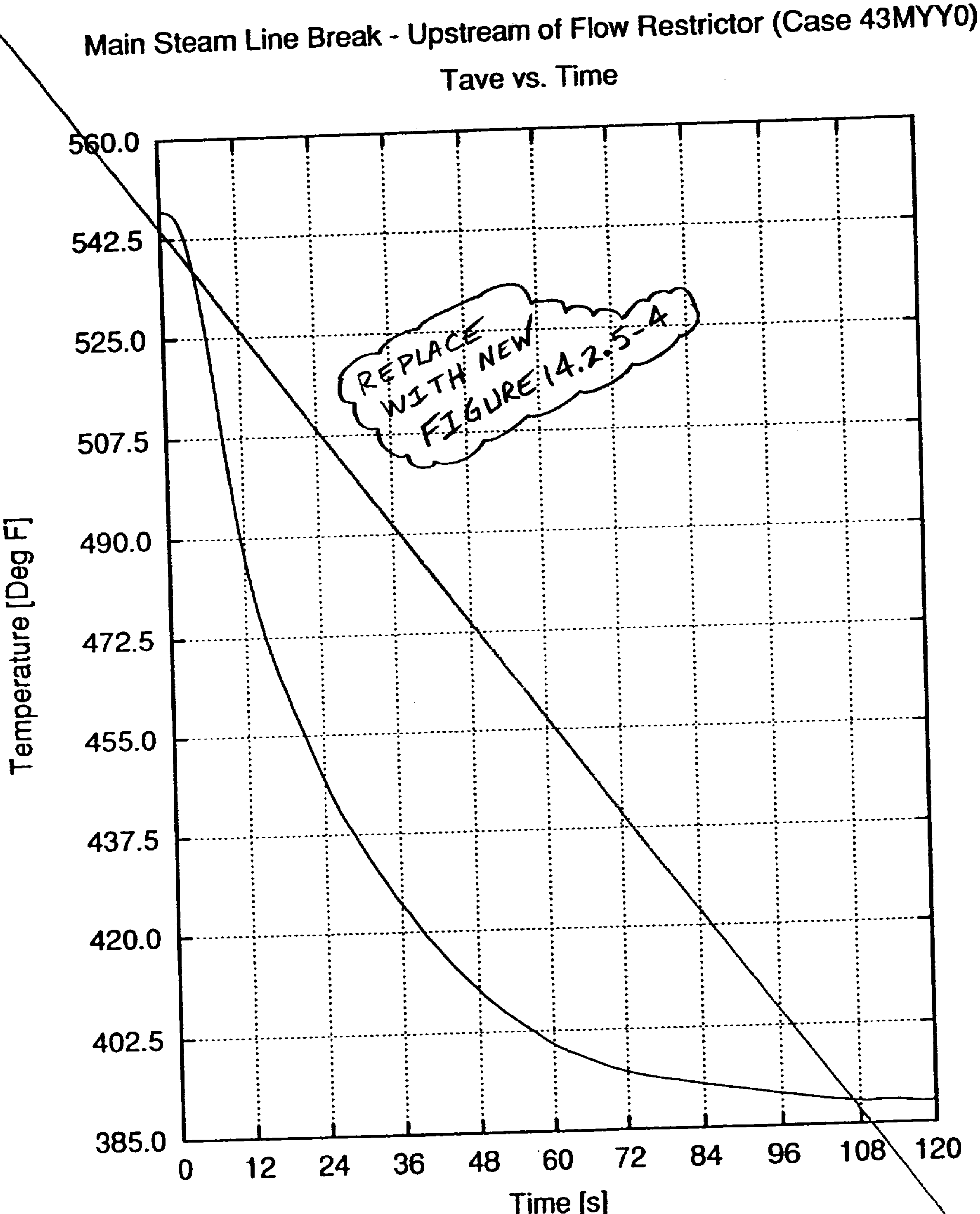

Figure 14.2.5-4

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12/01/2000

Main Steam Line Break - Upstream of Flow Restrictor (Case 43MYY0)

Pressurizer Pressure vs. Time

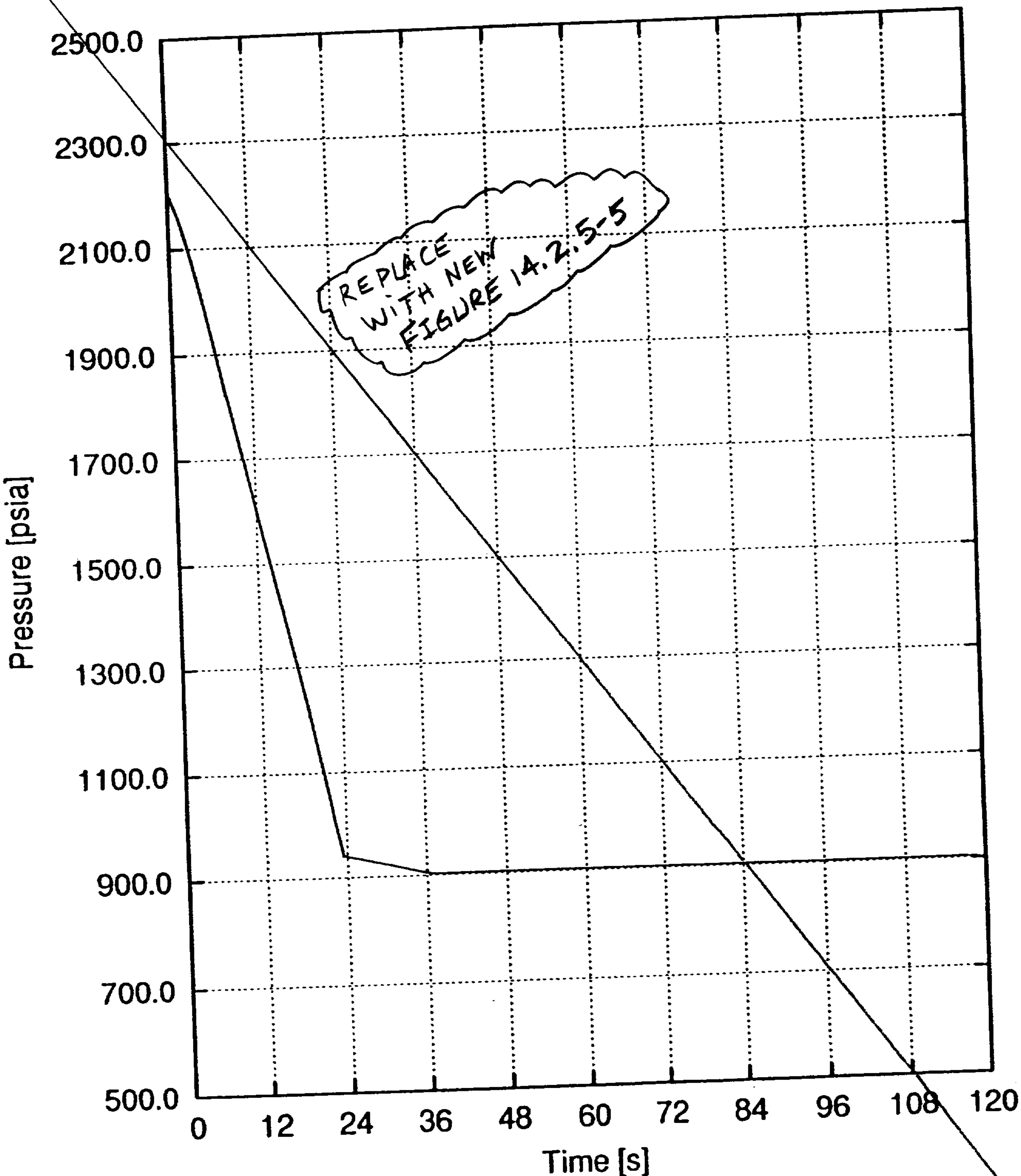

Figure 14.2.5-5

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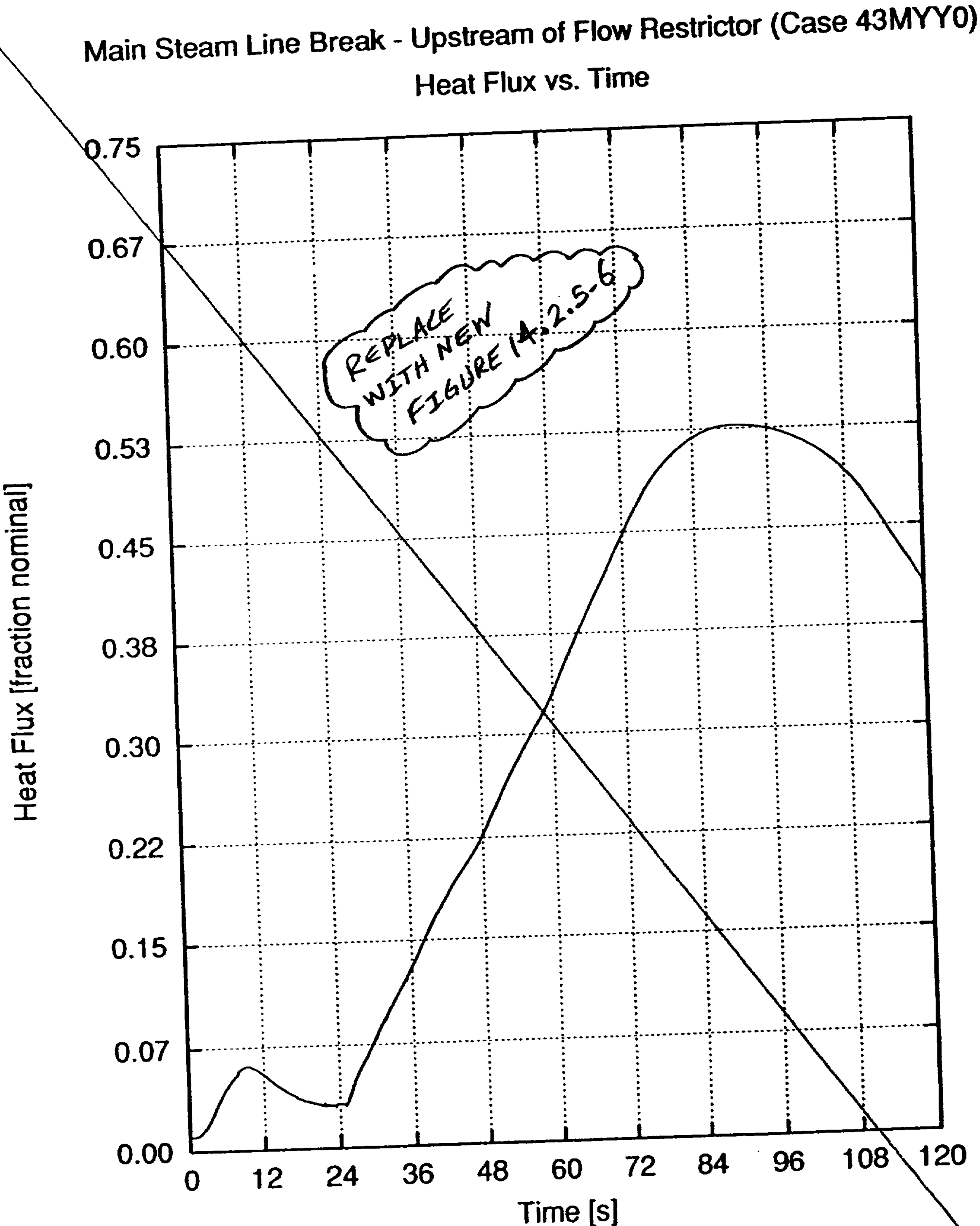

Figure 14.2.5-6

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12/01/2000

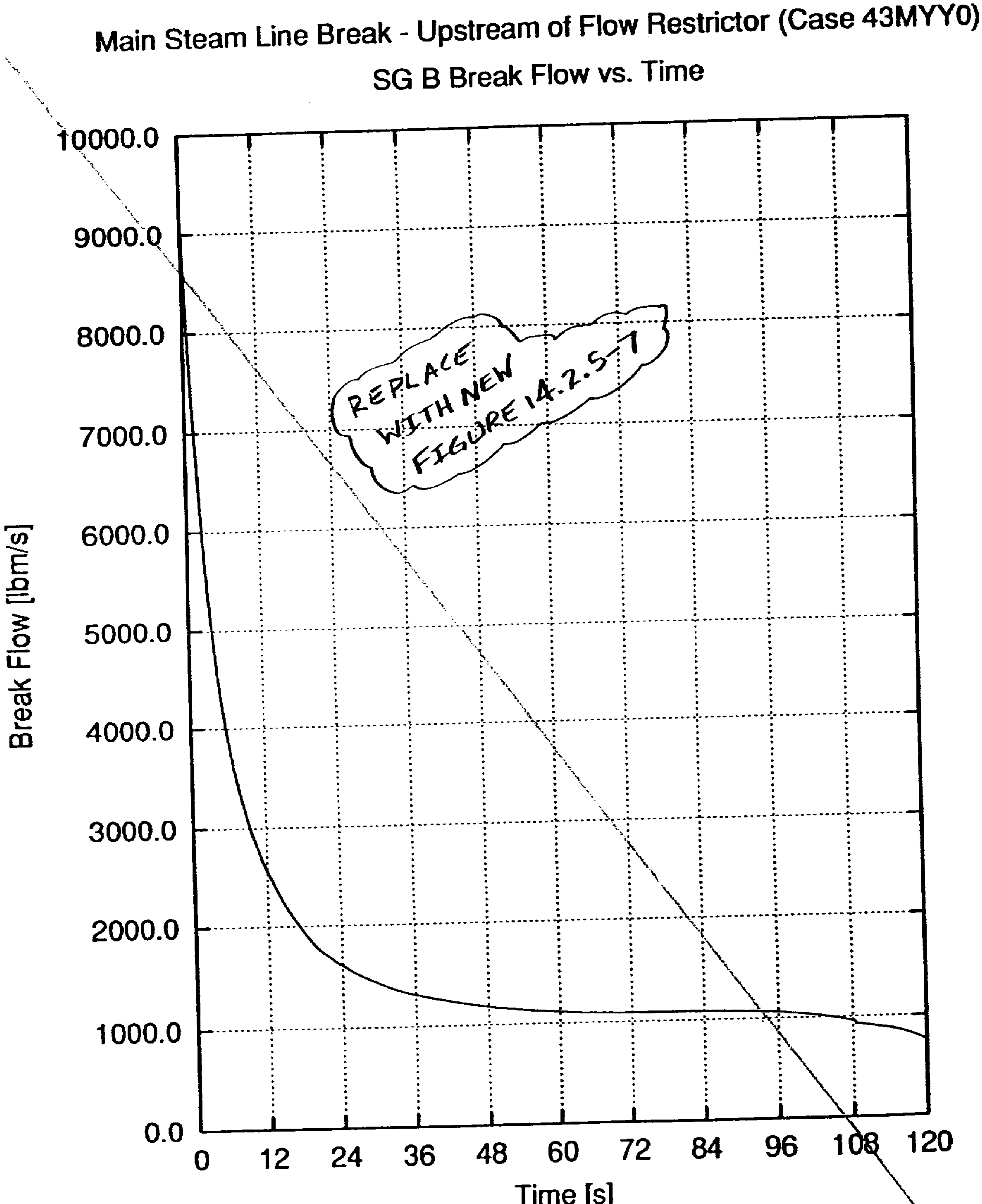

Figure 14.2.5-7

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12/01/2000

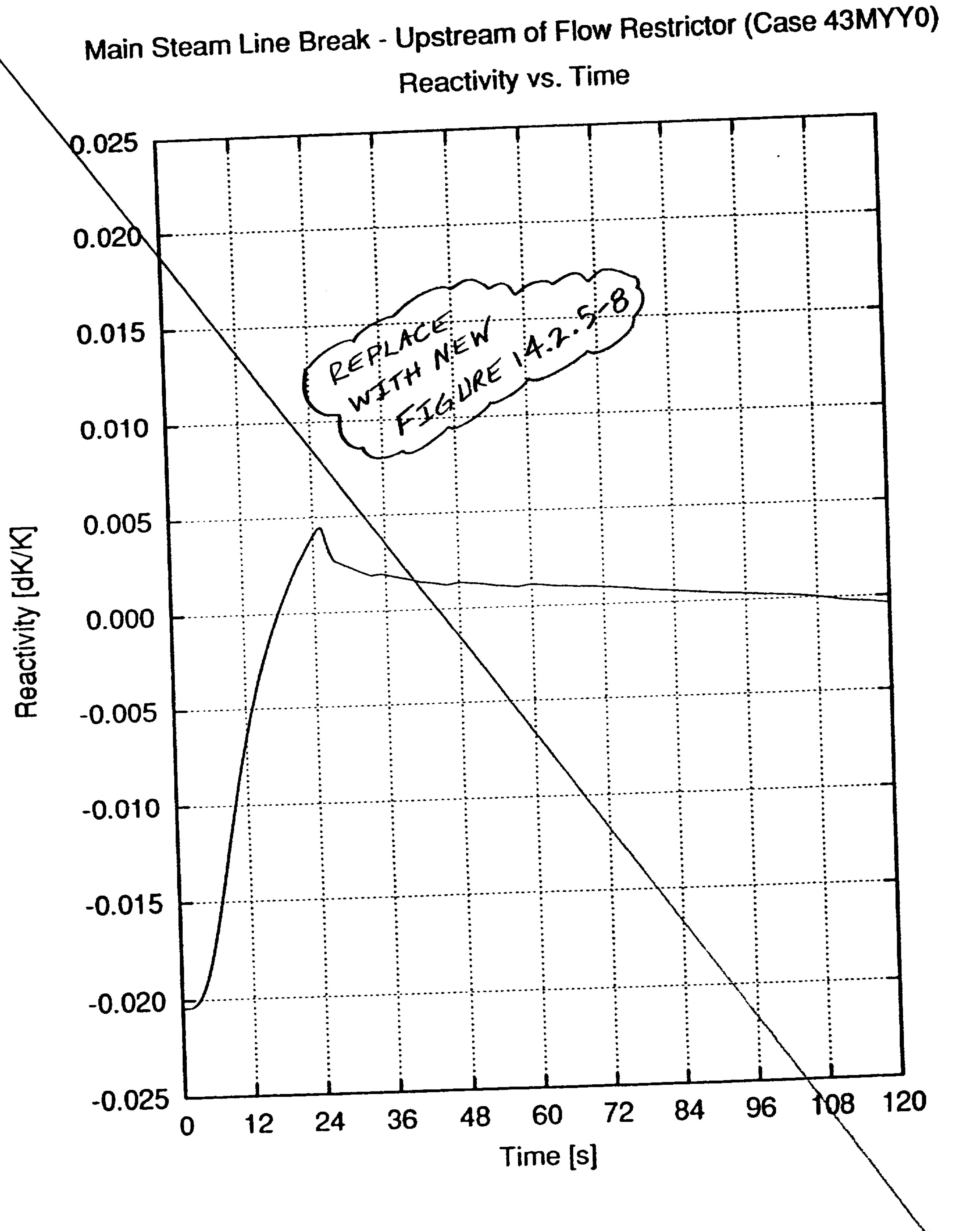

Figure 14.2.5-8

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12/01/2000

MAIN STEAM LINE BREAK

DOWNSTREAM FLOW RESTRICTOR

TAVE vs. TIME

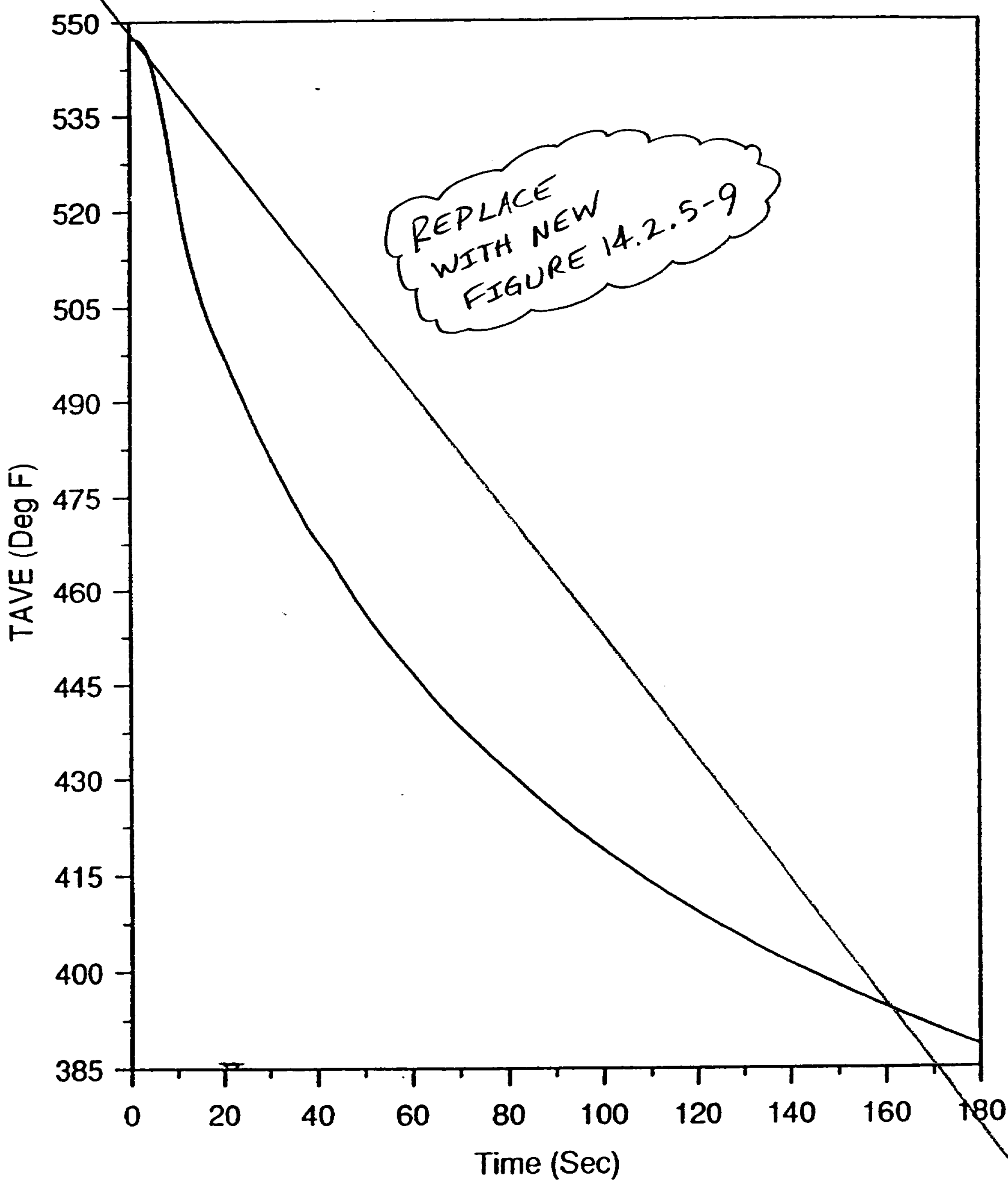

FIGURE 14.2.5-9

Rev. 15
05/01/99

MAIN STEAM LINE BREAK

DOWNSTREAM FLOW RESTRICTOR

PRESSURIZER PRESSURE vs. TIME

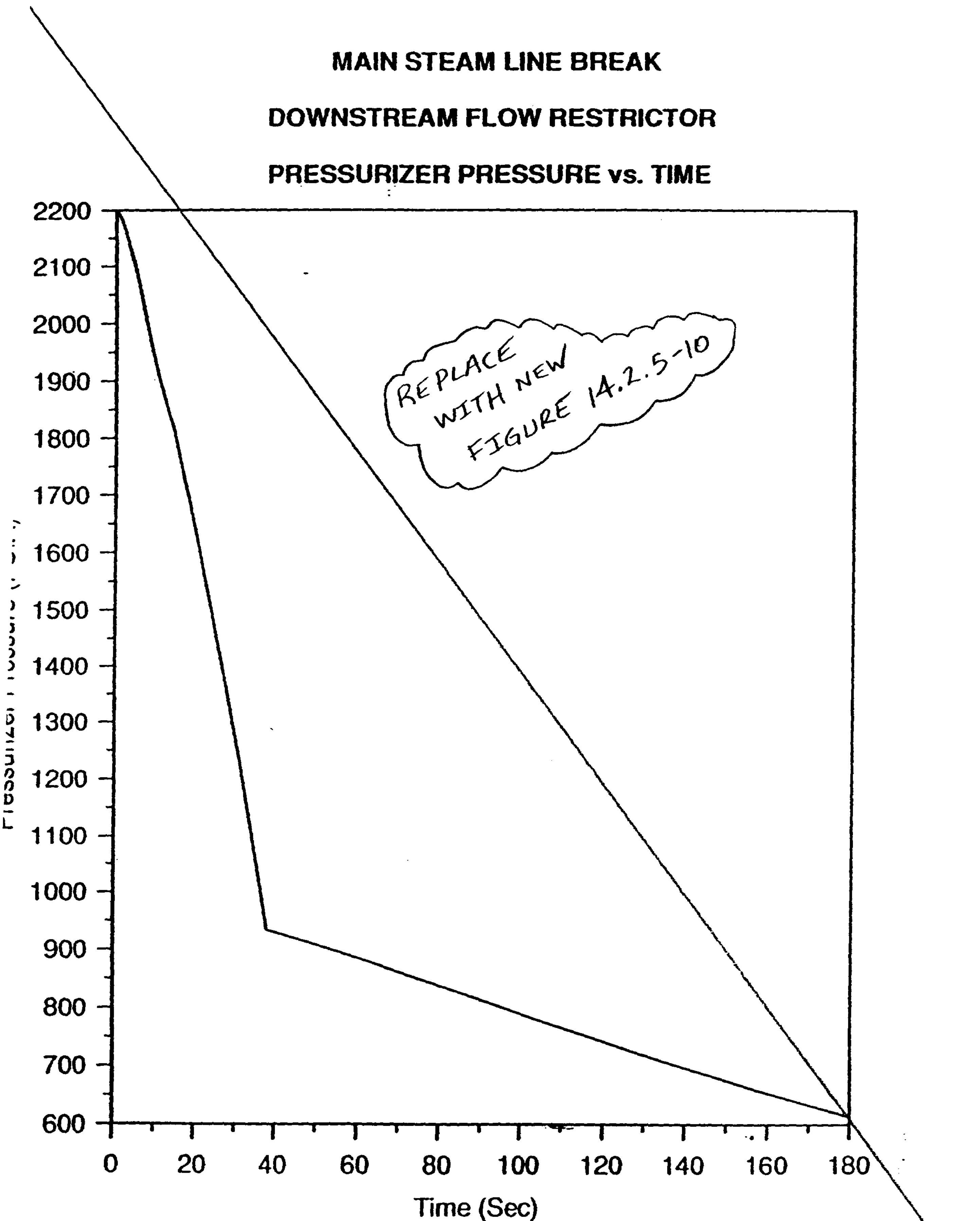

FIGURE 14.2.5-10

Rev. 15
05/01/99

MAIN STEAM LINE BREAK

DOWNSTREAM FLOW RESTRICTOR

HEAT FLUX vs. TIME

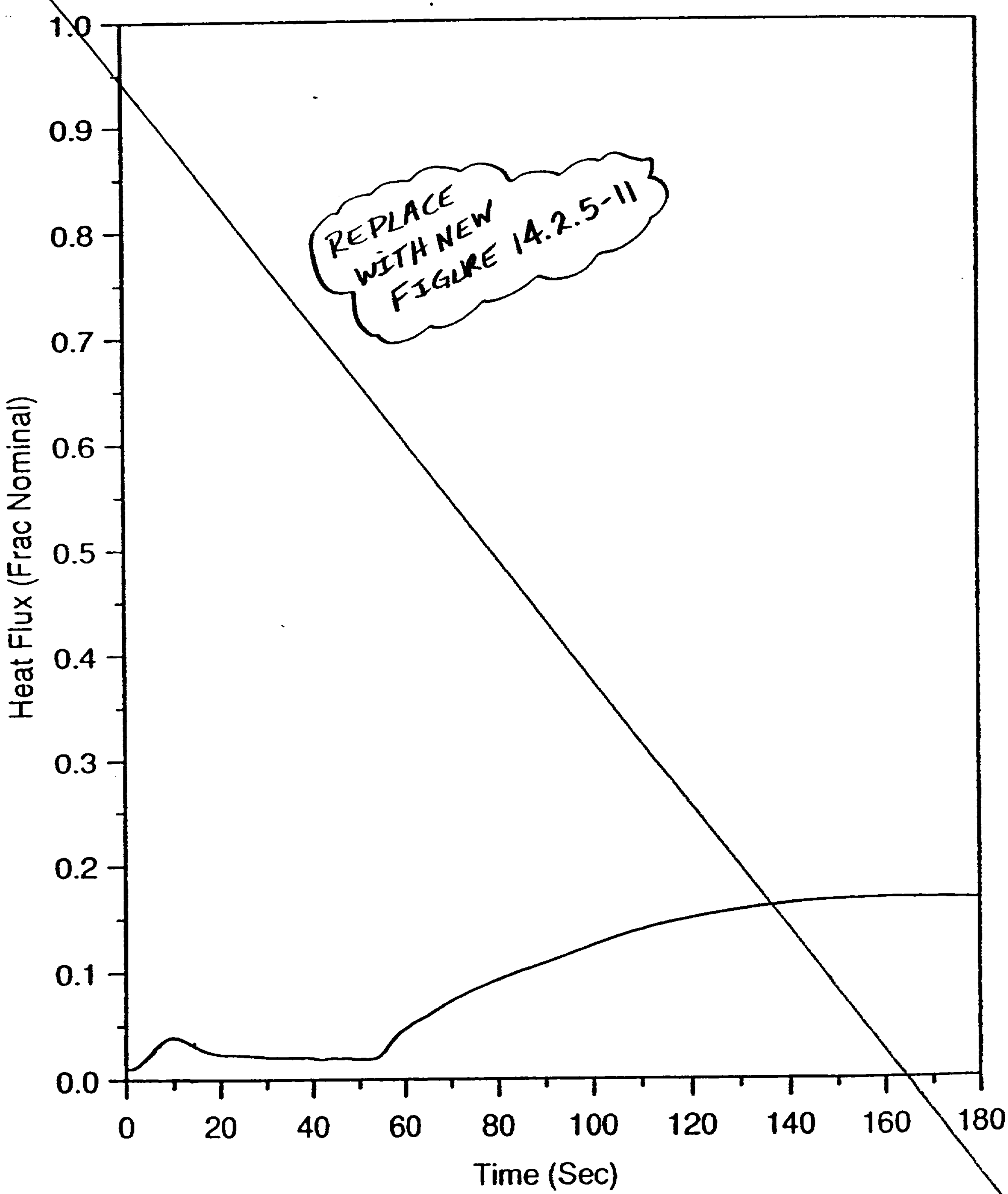

FIGURE 14.2.5-11

MAIN STEAM LINE BREAK

DOWNSTREAM FLOW RESTRICTOR

SG2 BREAK FLOW vs. TIME

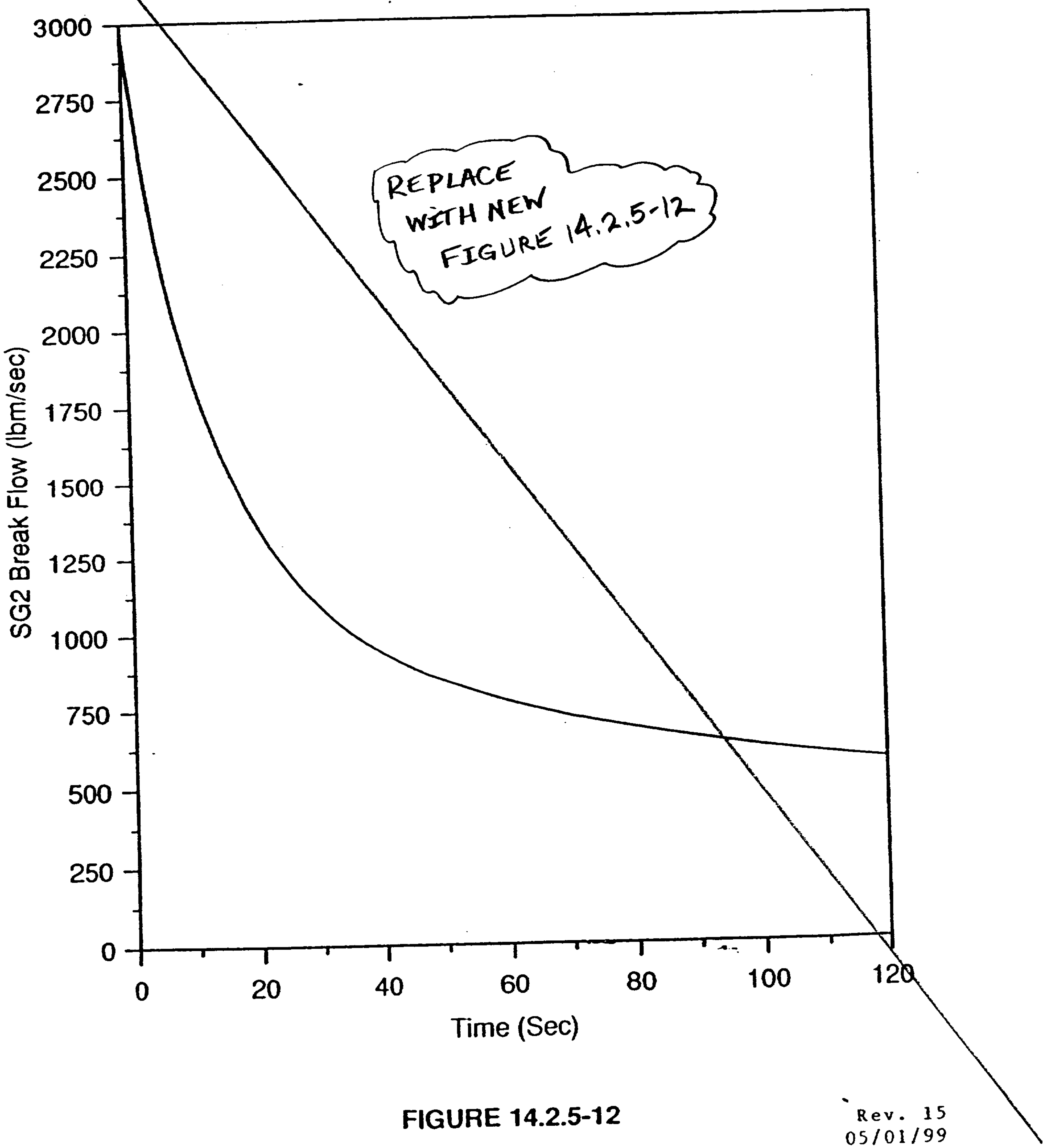

FIGURE 14.2.5-12

MAIN STEAM LINE BREAK

DOWNSTREAM FLOW RESTRICTOR

REACTIVITY vs. TIME

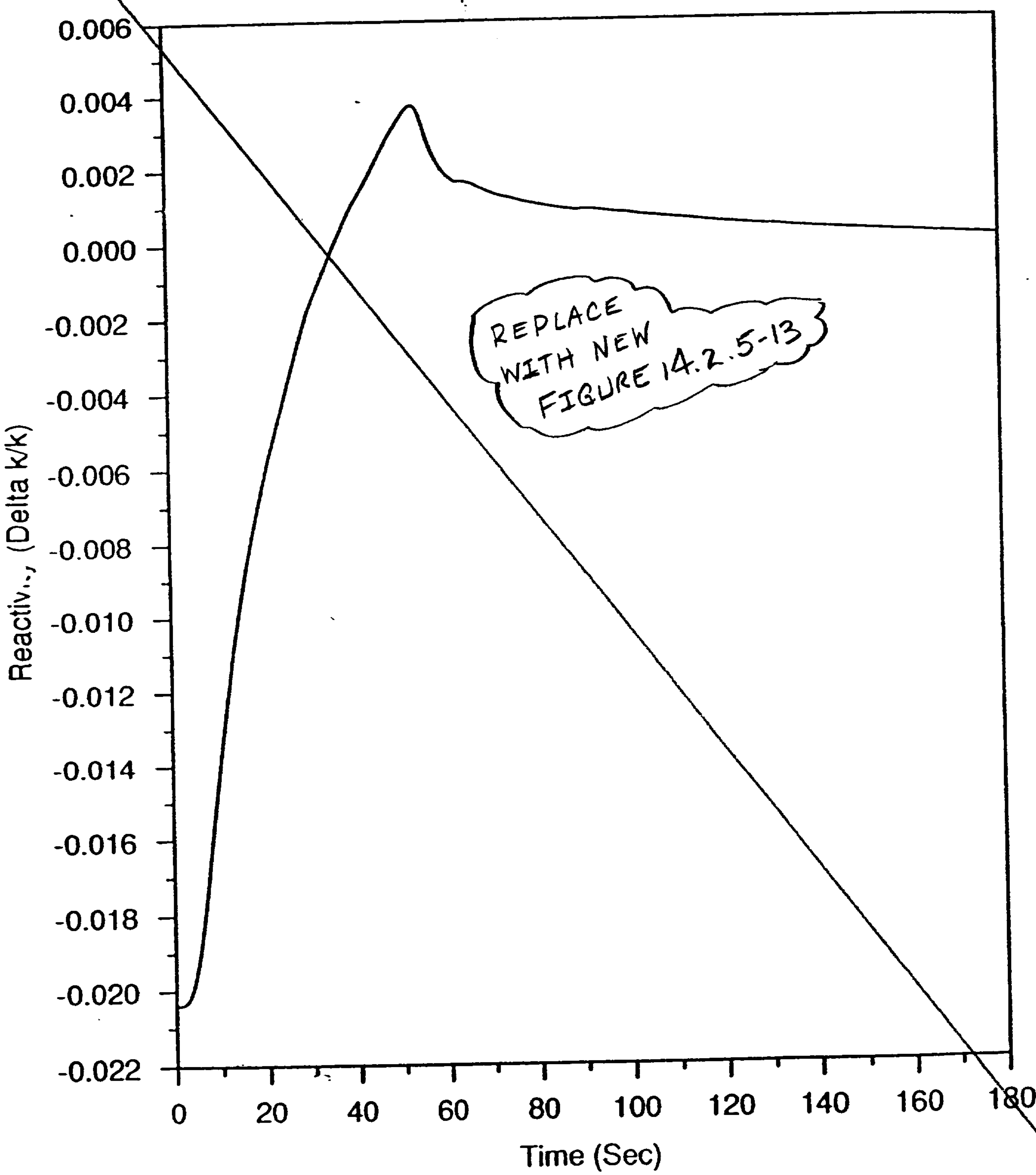

FIGURE 14.2.5-13

Rev. 15
05/01/99

MAIN STEAM LINE BREAK

SPURIOUS OPENING OF SAFETY VALVE

TAVE vs. TIME

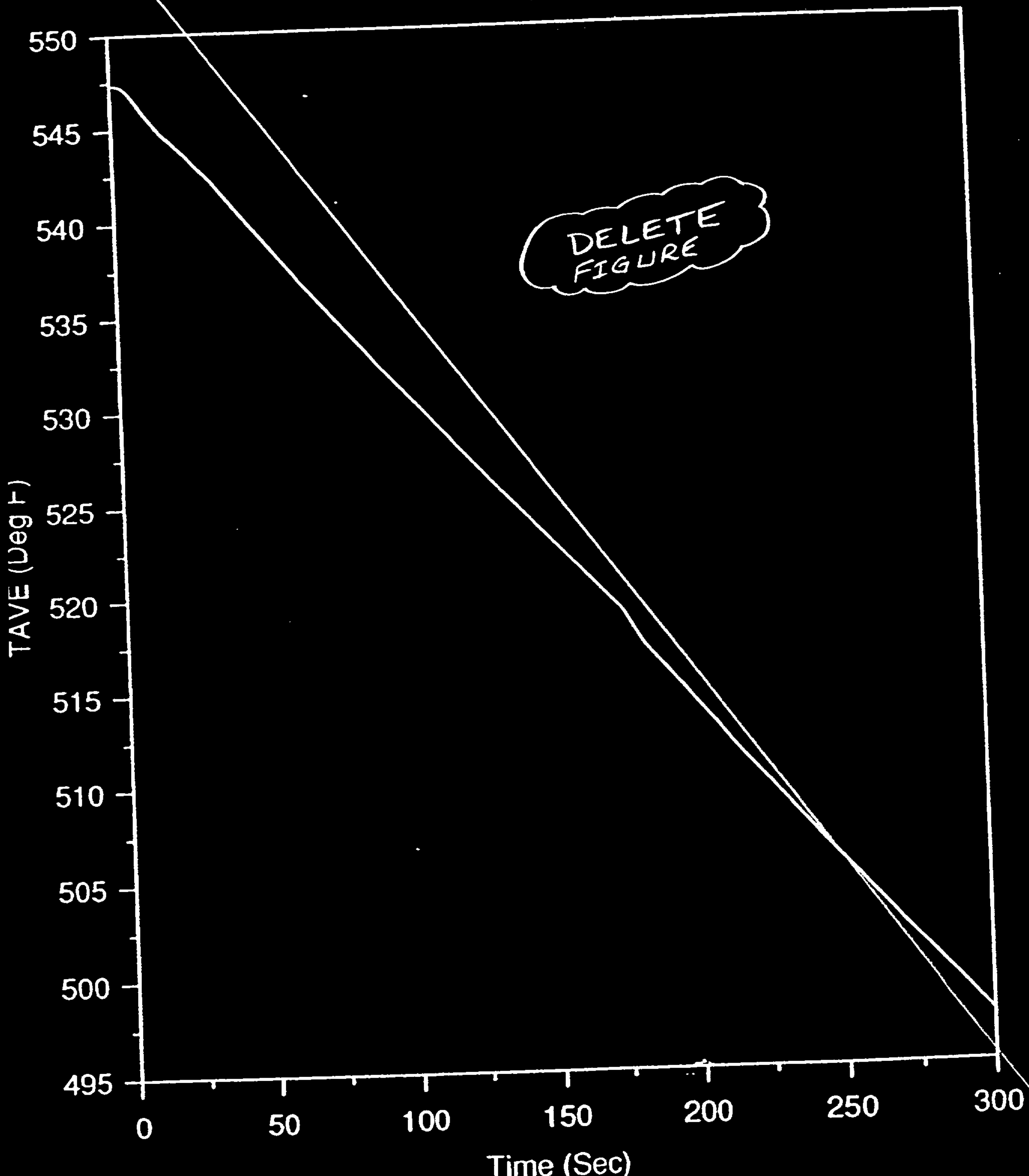

FIGURE 14.2.5-14

MAIN STEAM LINE BREAK

SPURIOUS OPENING OF SAFETY VALVE

PRESSURIZER PRESSURE vs. TIME

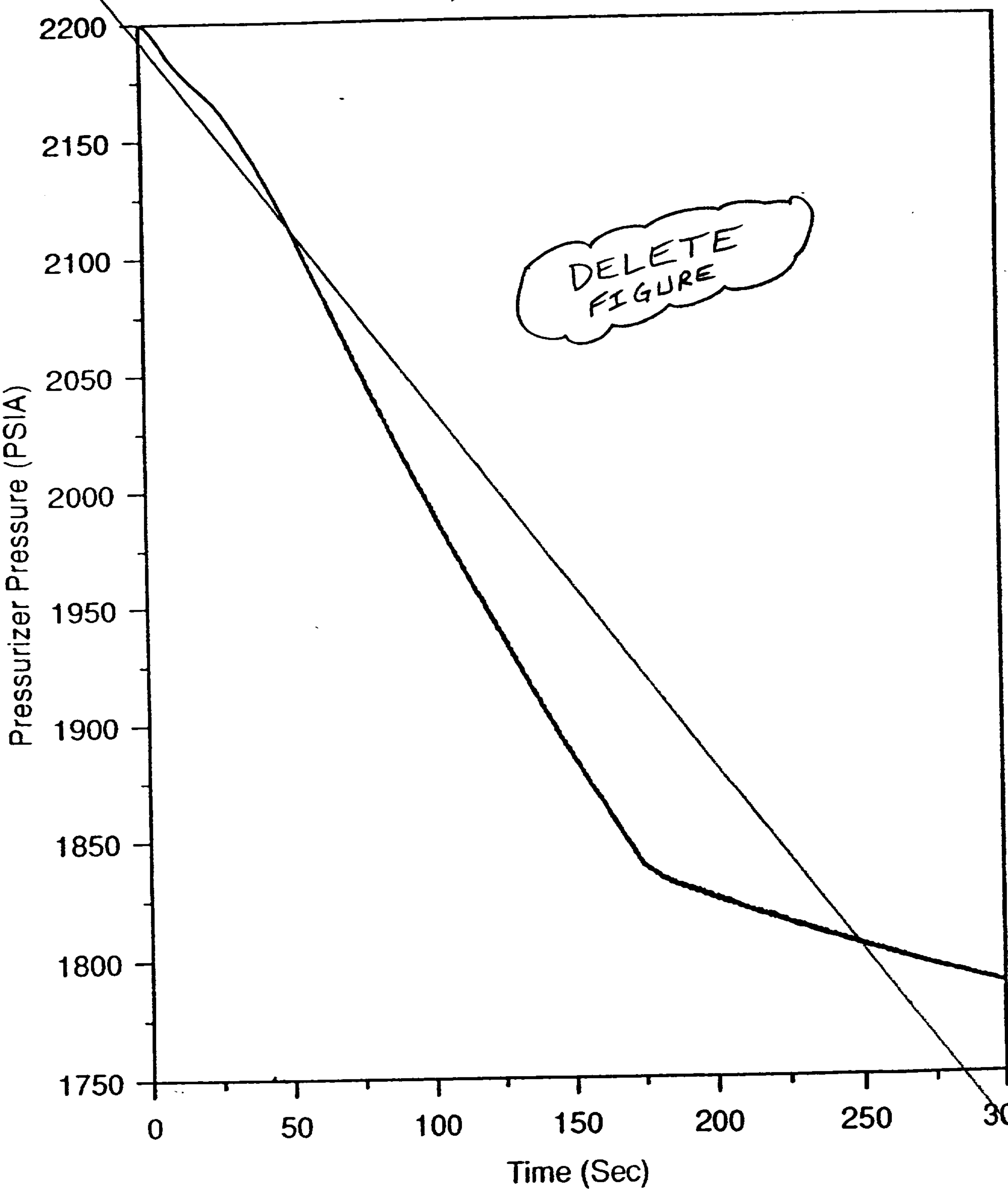

FIGURE 14.2.5-15

Rev. 15
05/01/99

MAIN STEAM LINE BREAK

SPURIOUS OPENING OF SAFETY VALVE

HEAT FLUX vs. TIME

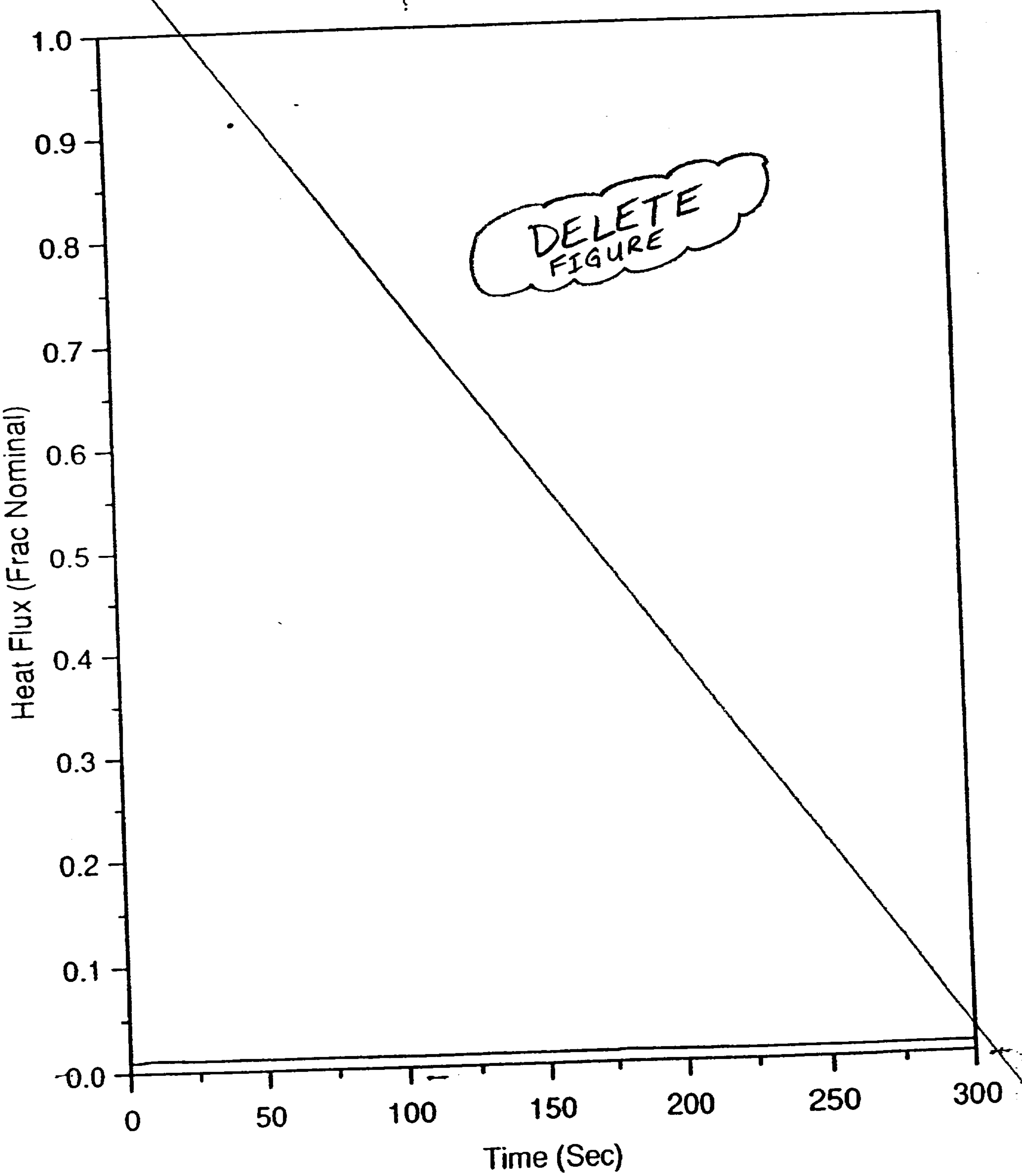

FIGURE 14.2.5-16

Rev. 15
05/01/99

MAIN STEAM LINE BREAK

SPURIOUS OPENING OF SAFETY VALVE

SG2 BREAK FLOW vs. TIME

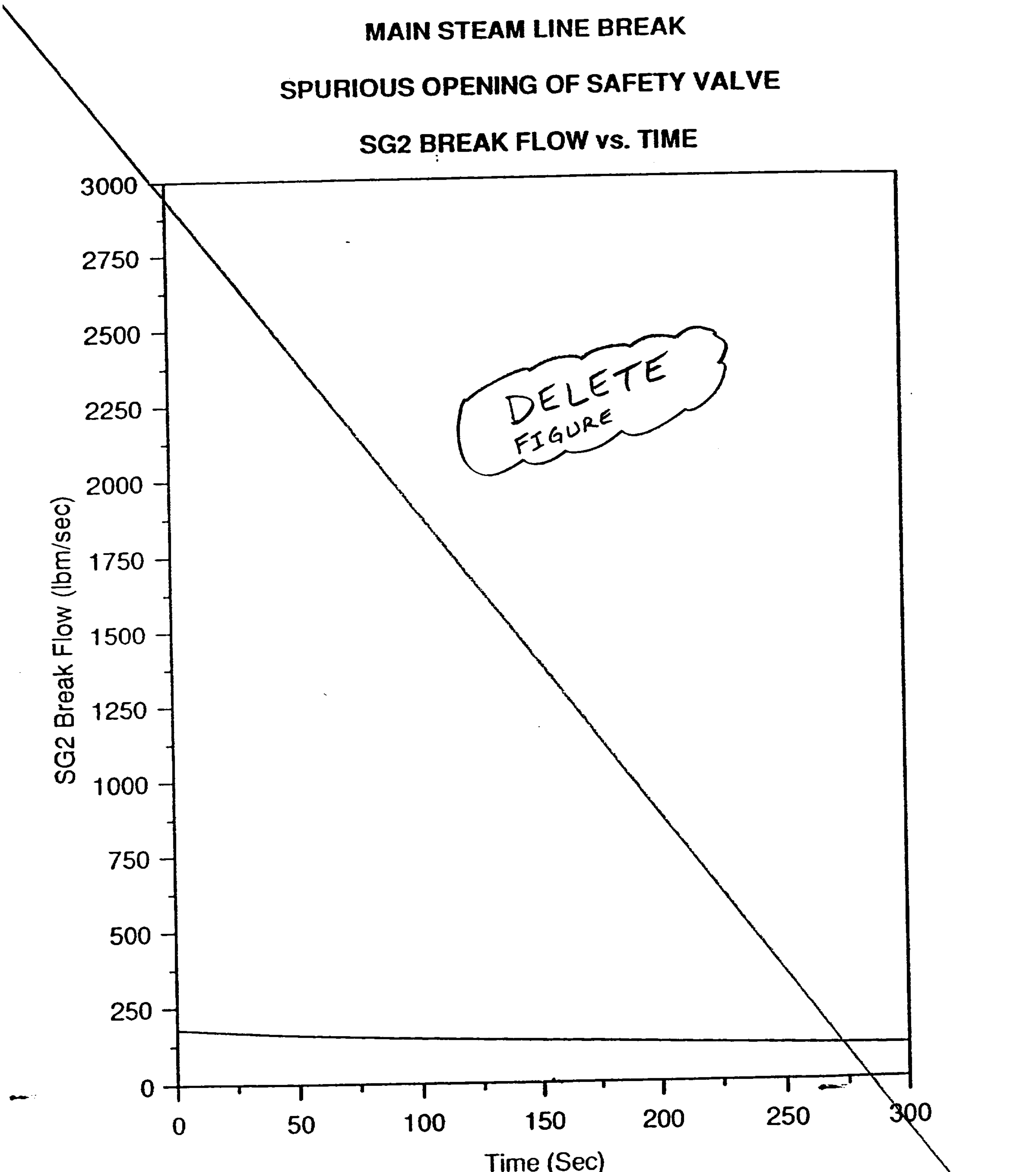

FIGURE 14.2.5-17

Rev. 15
05/01/99

MAIN STEAM LINE BREAK

SPURIOUS OPENING OF SAFETY VALVE

REACTIVITY vs. TIME

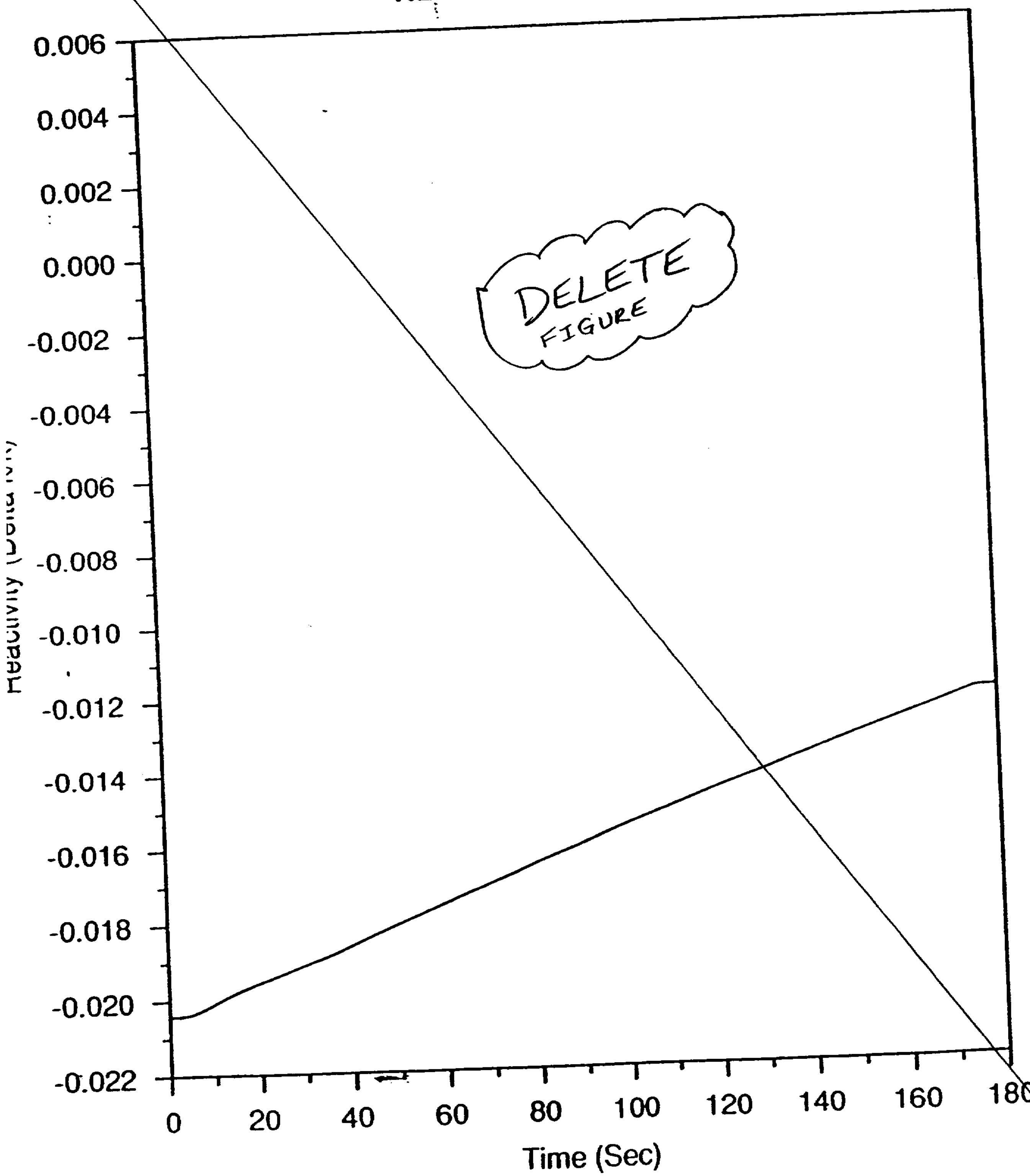

FIGURE 14.2.5-18

Rev. 15
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MAIN STEAM LINE BREAK

CONTAINMENT PRESSURE RESPONSE

OUT OF SCOPE

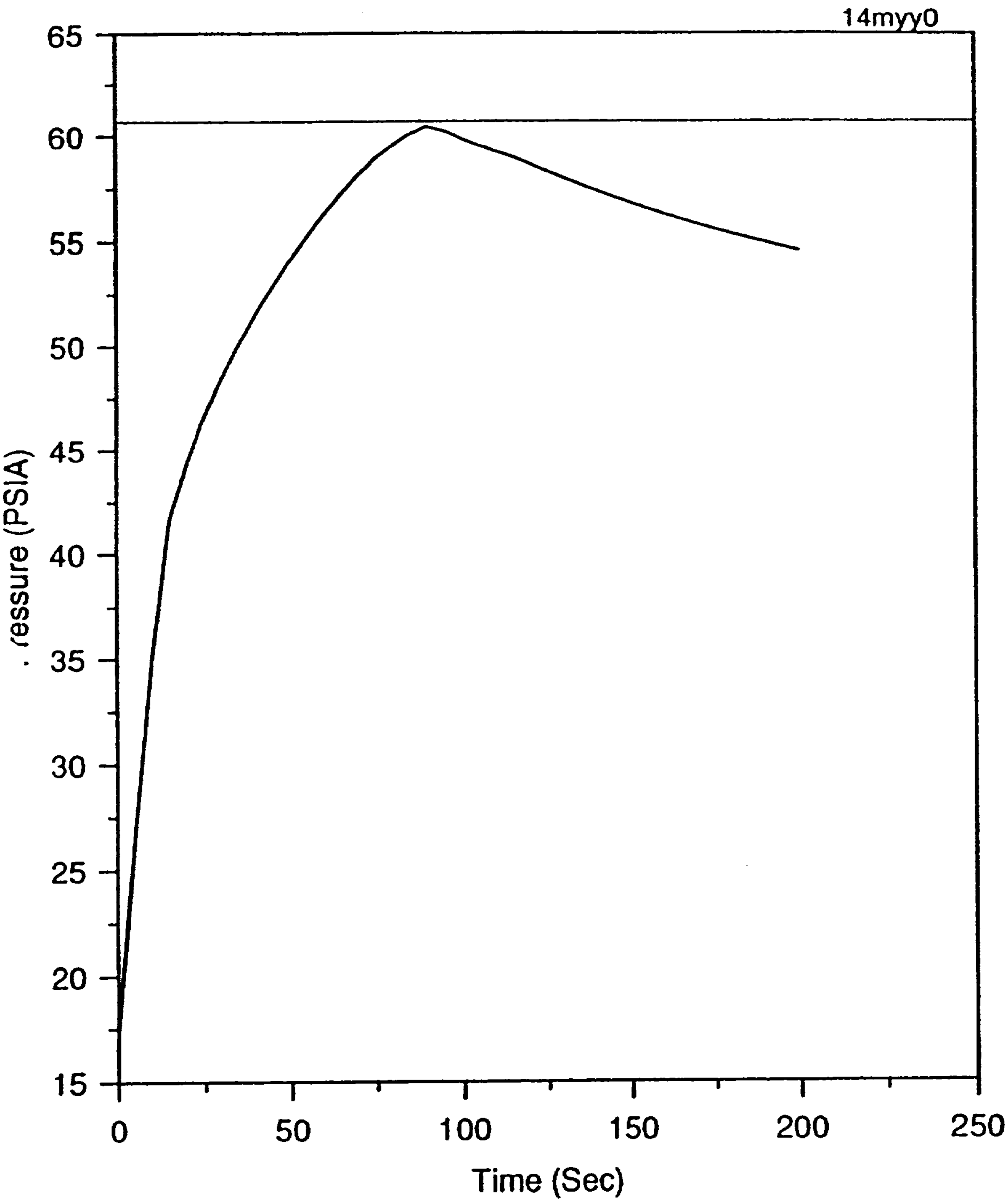

FIGURE 14.2.5-19

Rev. 14

NOV 0 1 1997

MAIN STEAM LINE BREAK

CONTAINMENT TEMPERATURE RESPONSE

OUT OF SCOPE

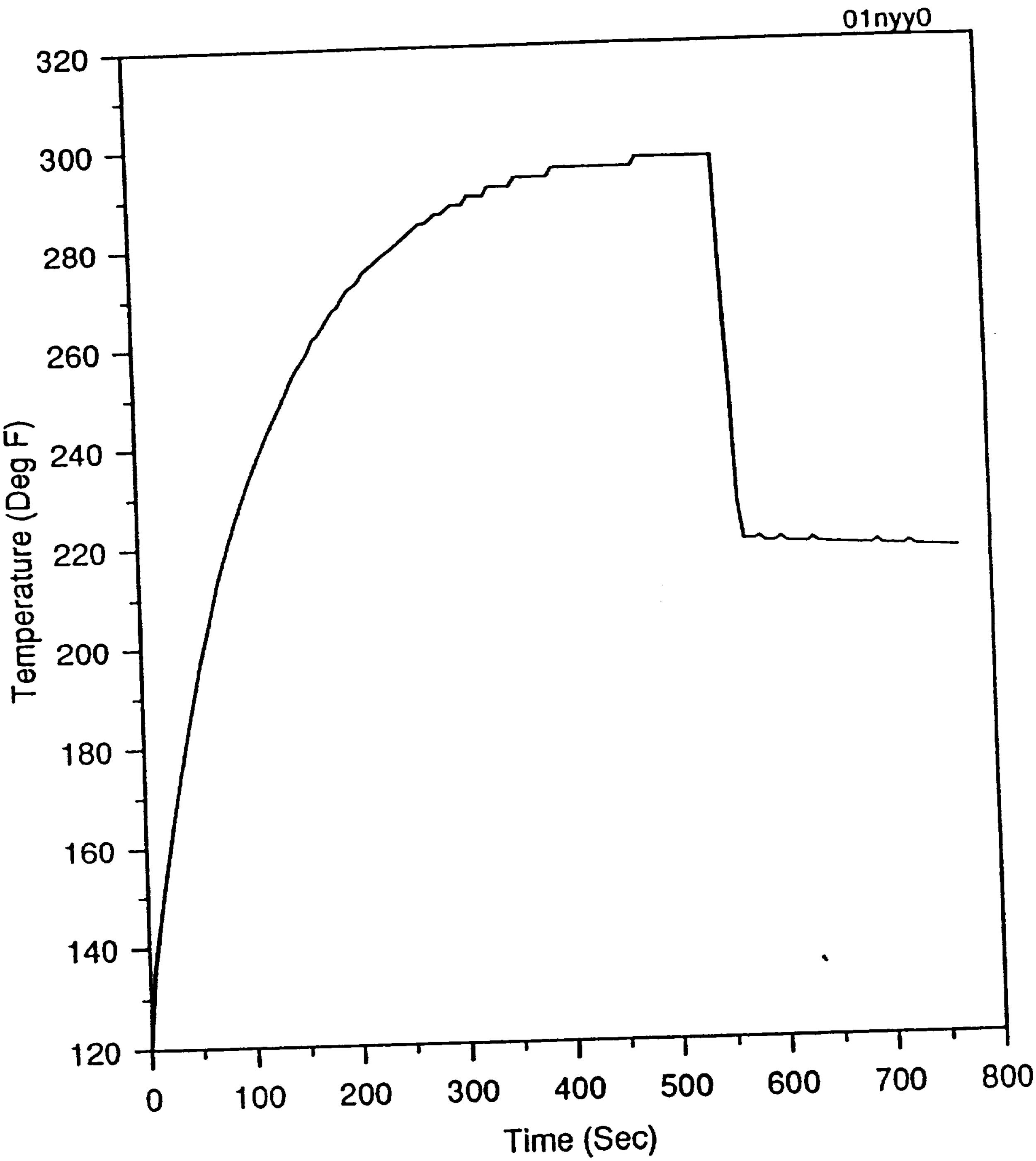

FIGURE 14.2.5-20

Rev. 1

NOV 0 1 1997

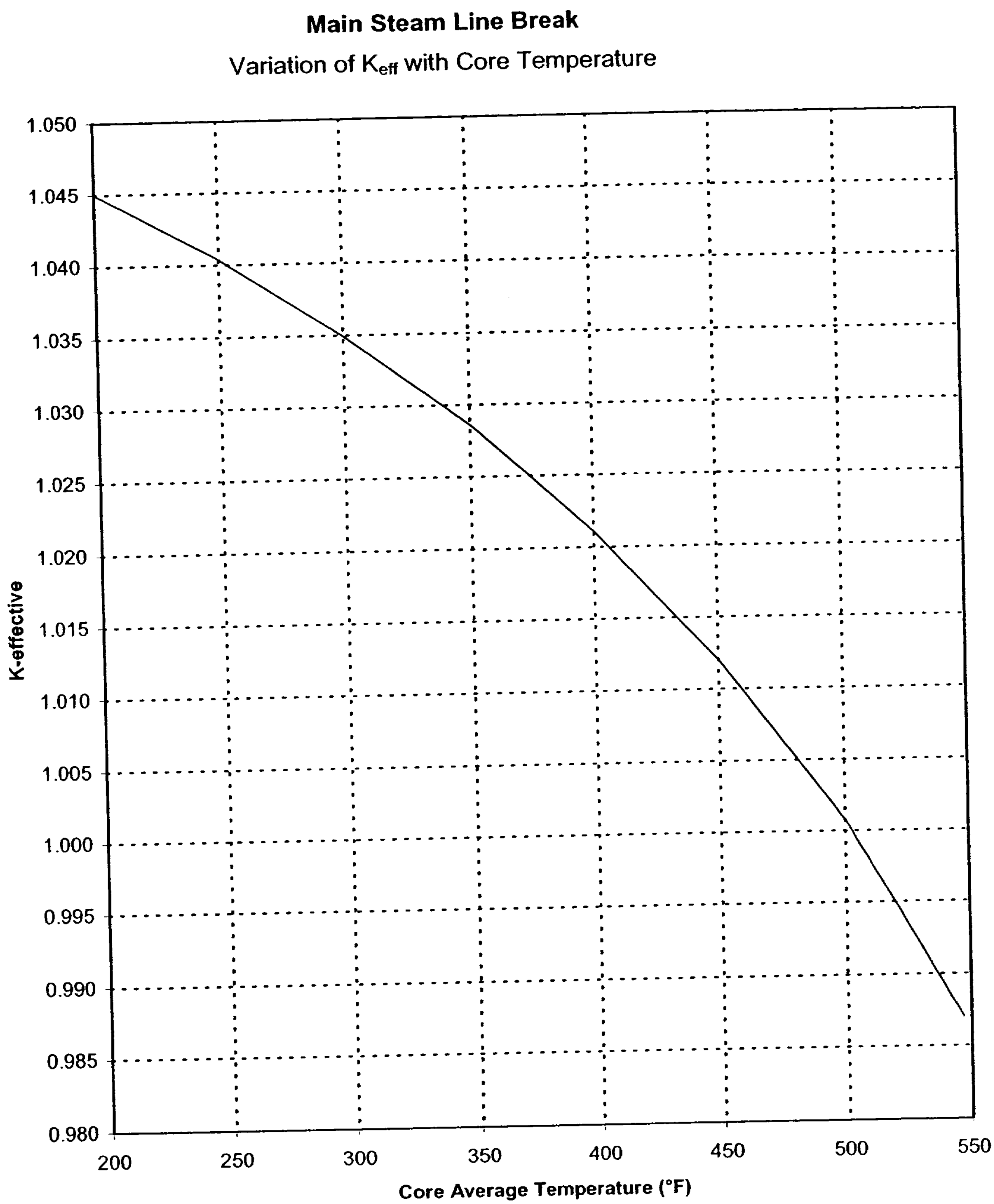

Figure 14.2.5-1

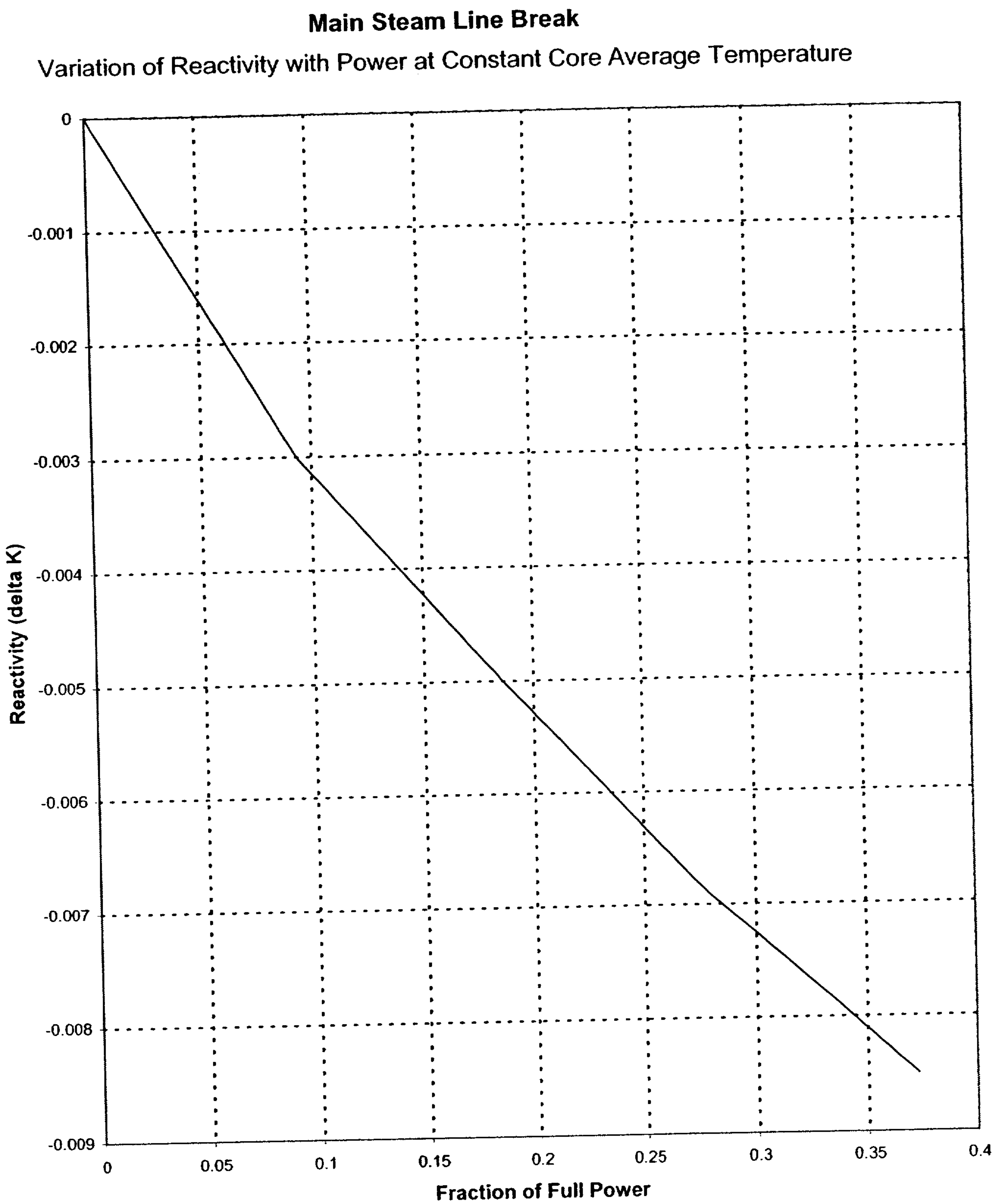

Figure 14.2.5-2

Main Steam Line Break

Safety Injection Flow Rate vs. Reactor Coolant Pressure

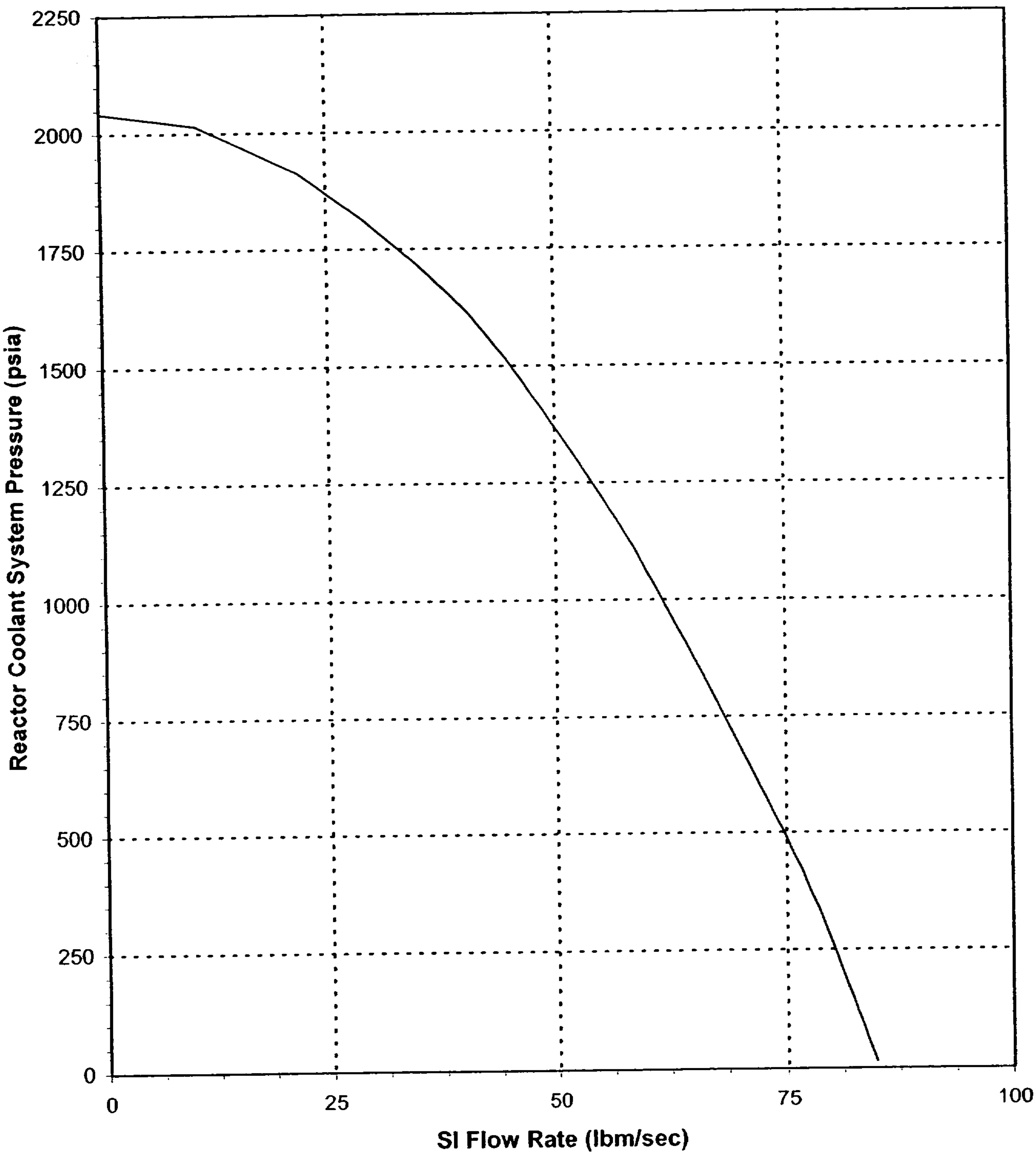

Figure 14.2.5-3

Main Steam Line Break

With Offsite Power Available – Tave vs. Time

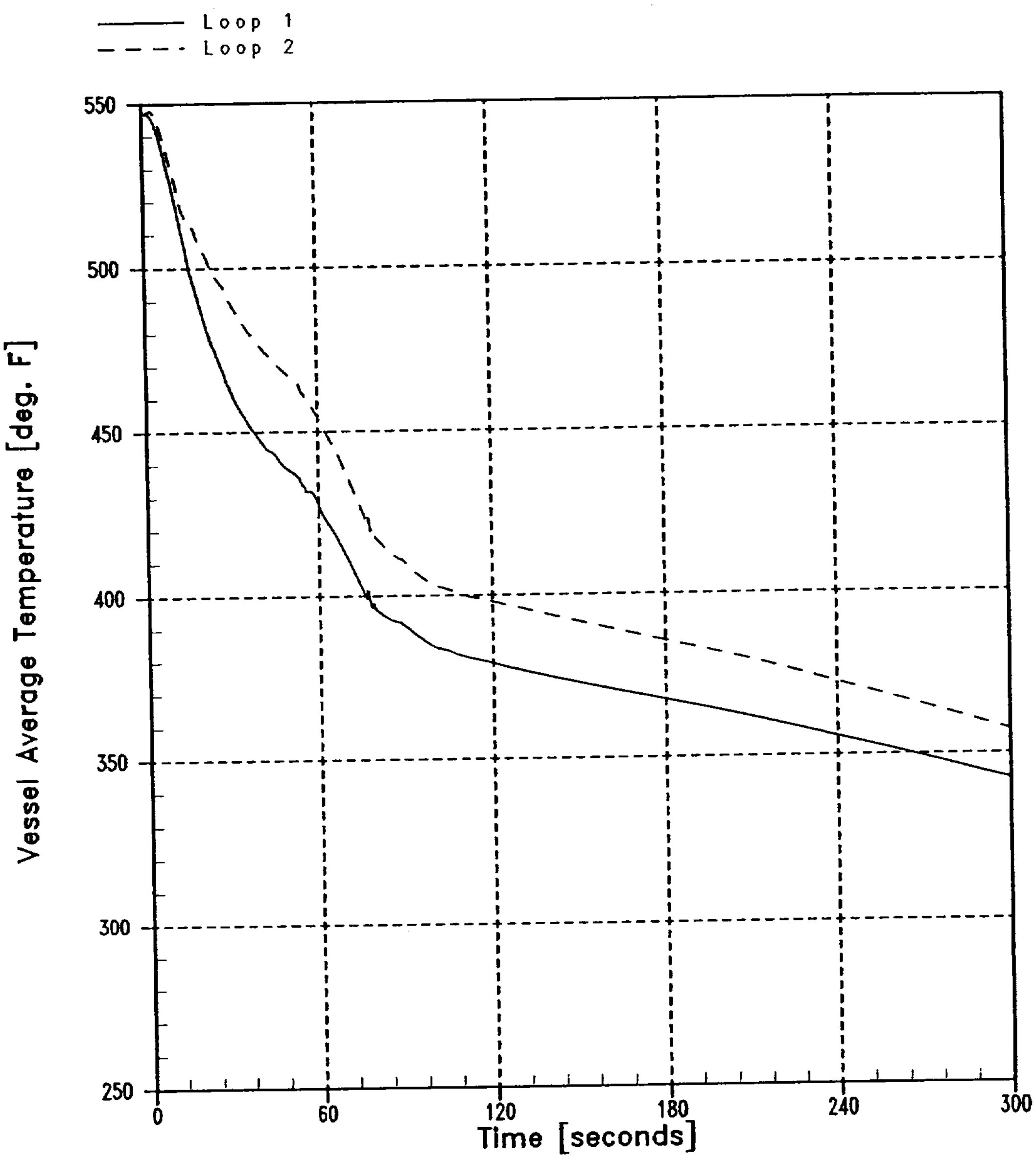

Figure 14.2.5-4

Main Steam Line Break

With Offsite Power Available – Pressurizer Pressure vs. Time

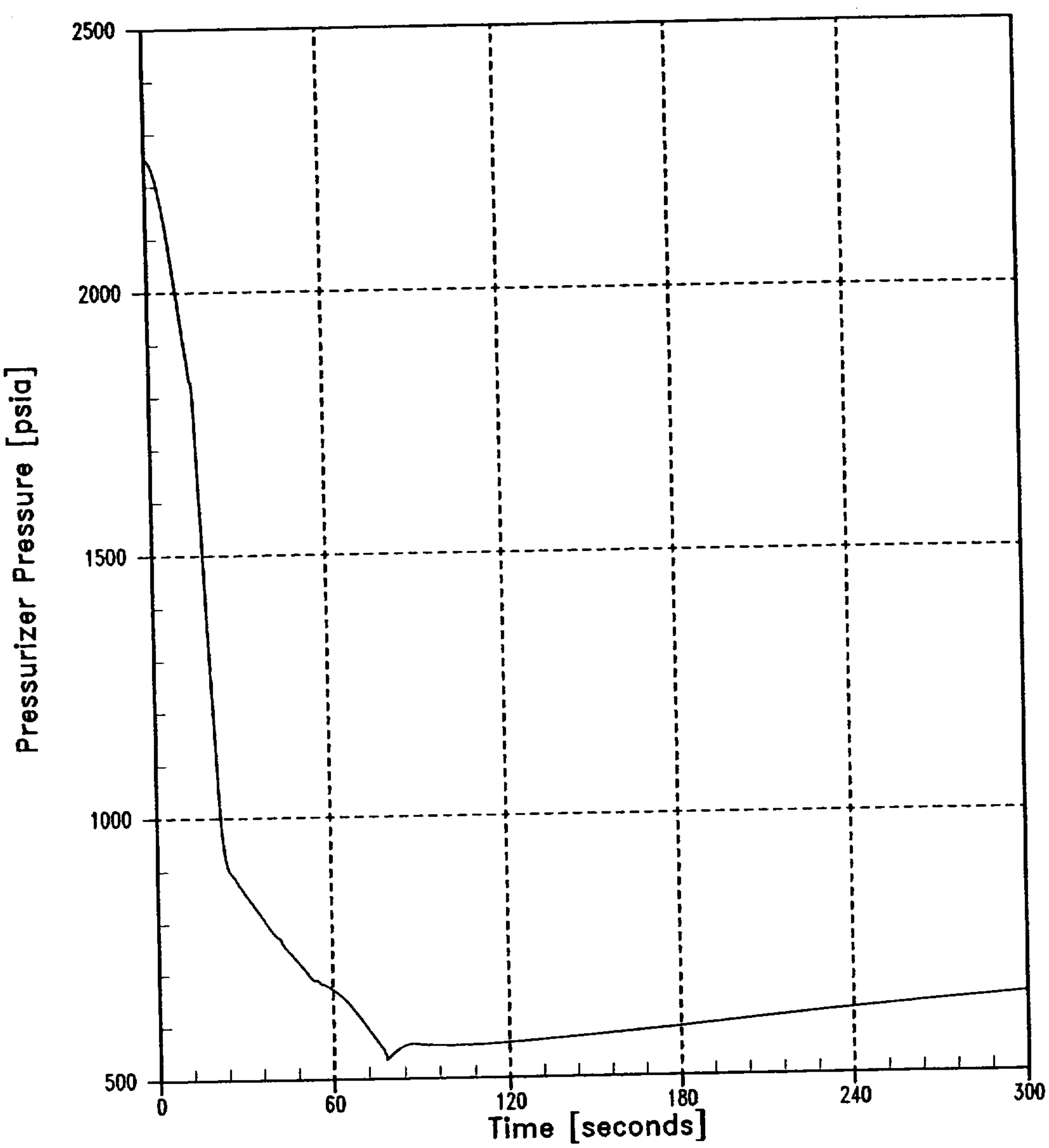

Figure 14.2.5-5

Main Steam Line Break

With Offsite Power Available – Heat Flux vs. Time

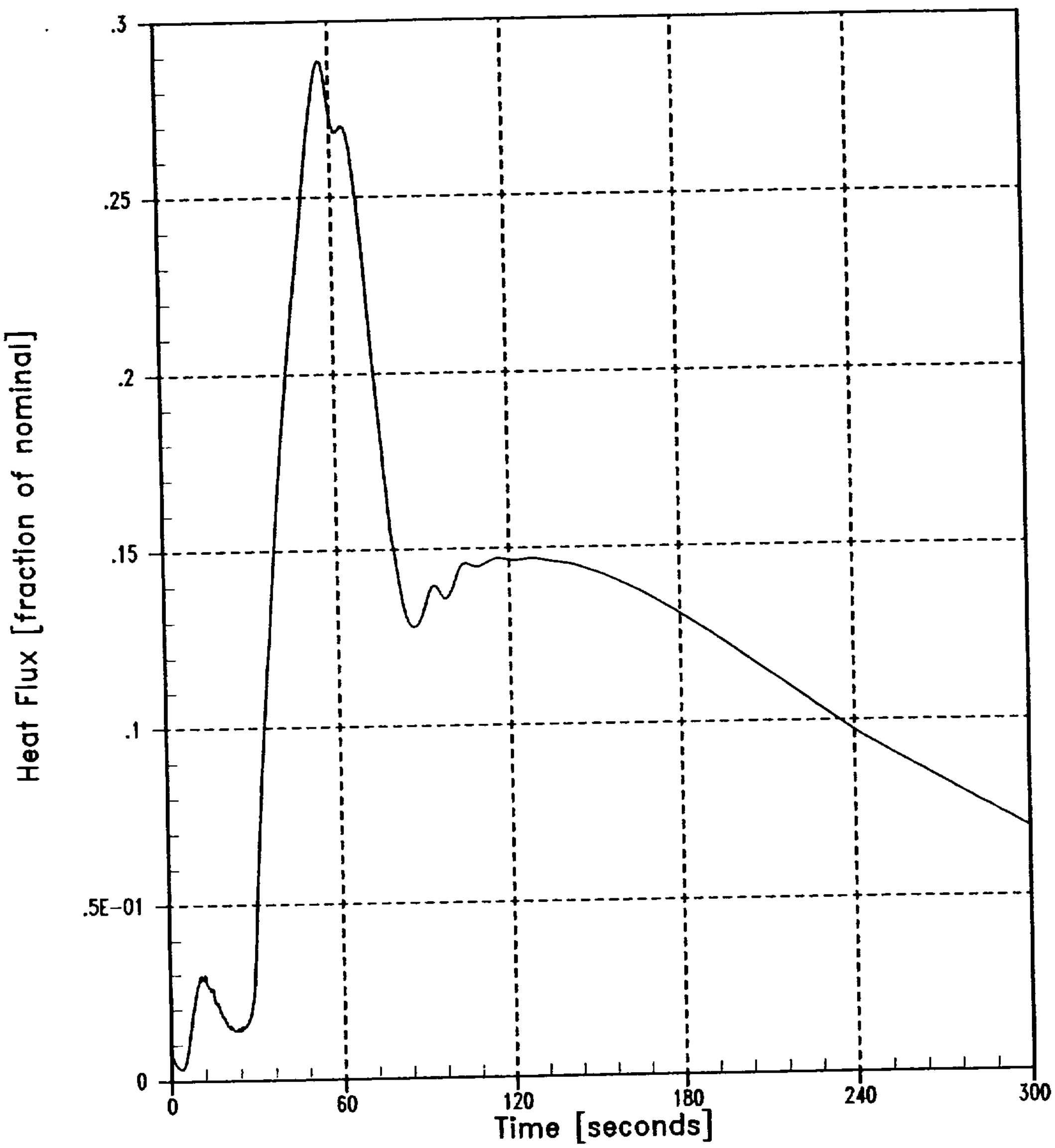

Figure 14.2.5-6

Main Steam Line Break

With Offsite Power Available – SG Outlet Nozzle Flow vs. Time

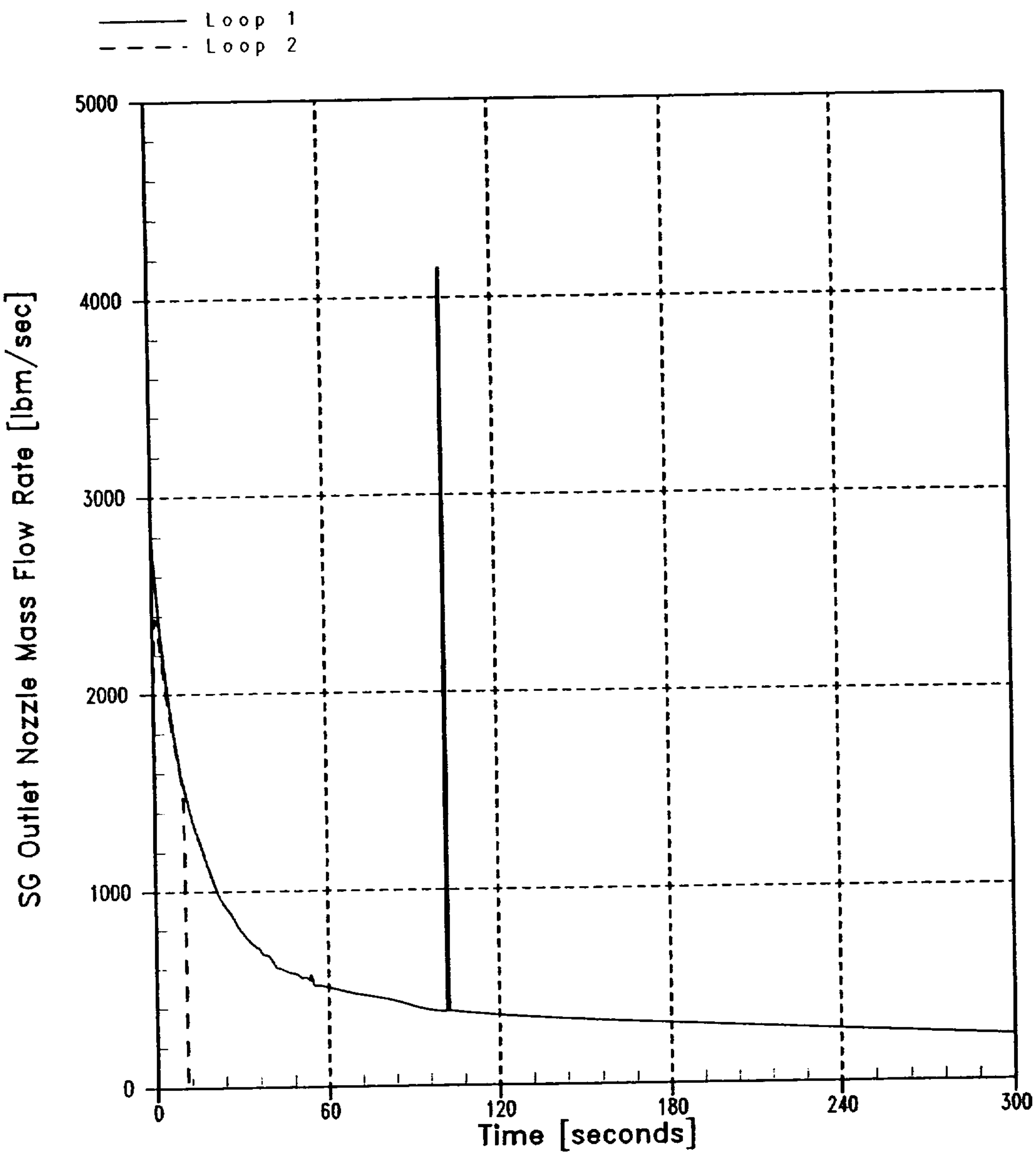

Figure 14.2.5-7

Main Steam Line Break

With Offsite Power Available – Reactivity vs. Time

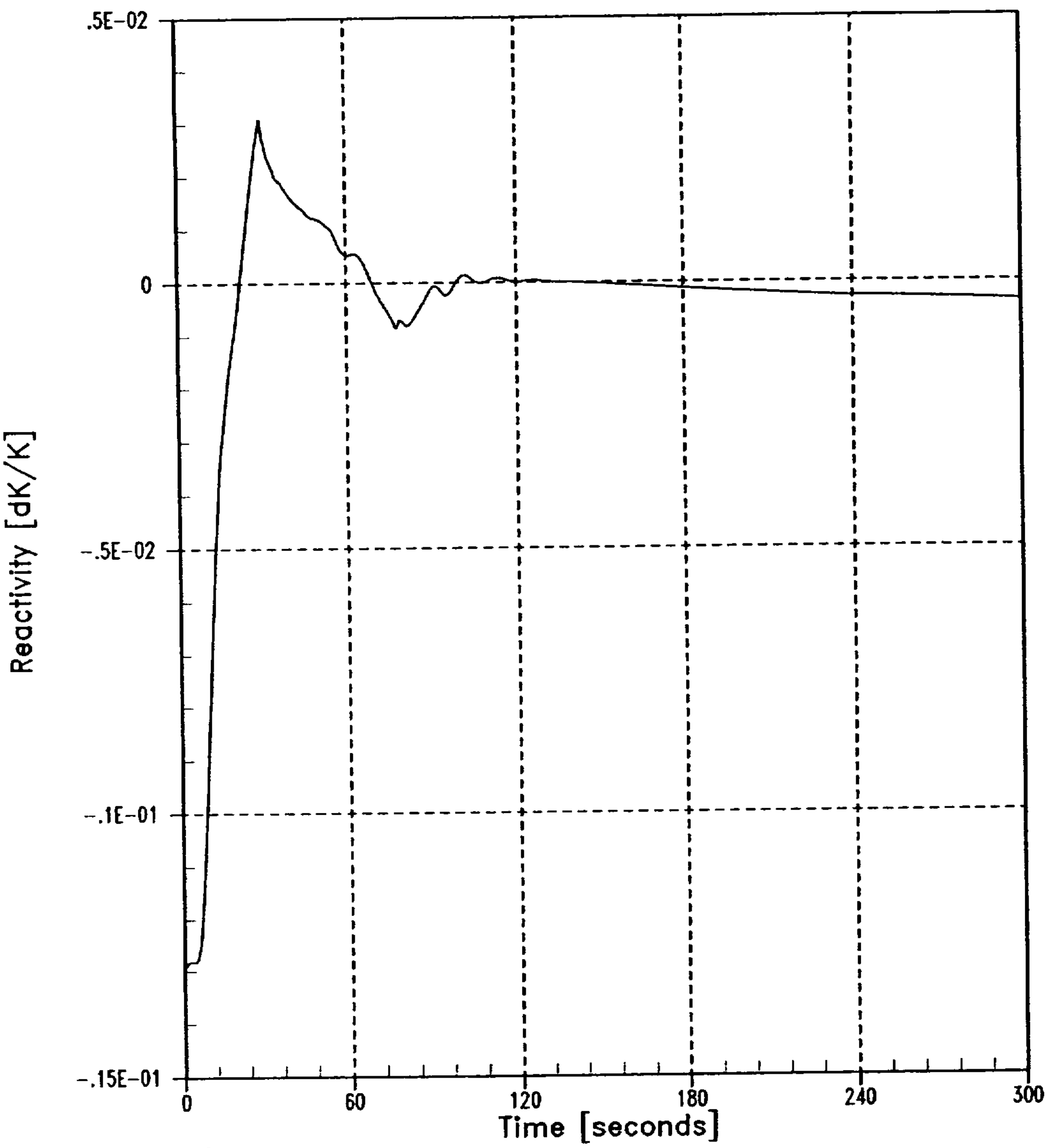

Figure 14.2.5-8

Main Steam Line Break

With a Loss of Offsite Power – Tave vs. Time

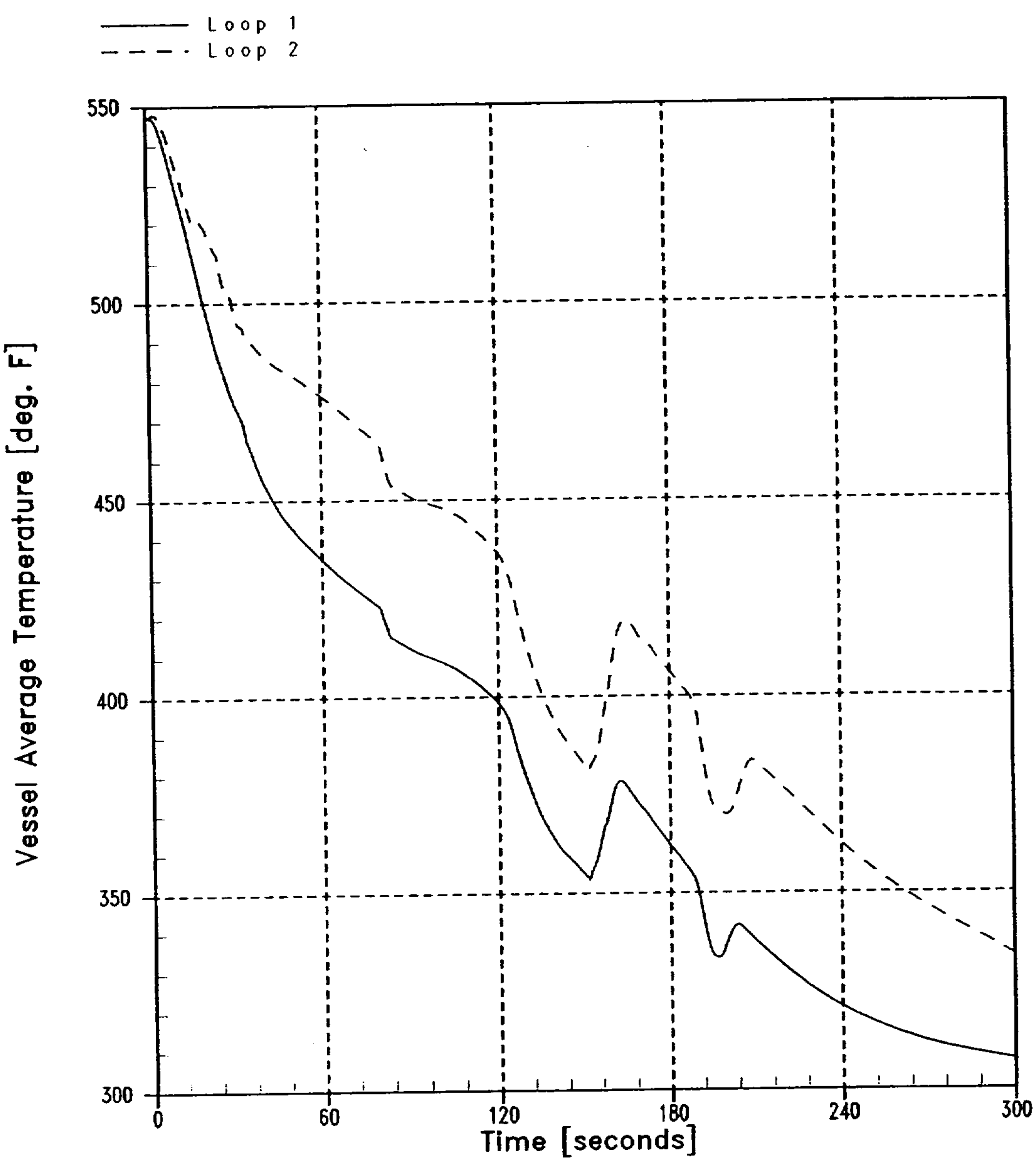

Figure 14.2.5-9

Main Steam Line Break

With a Loss of Offsite Power – Pressurizer Pressure vs. Time

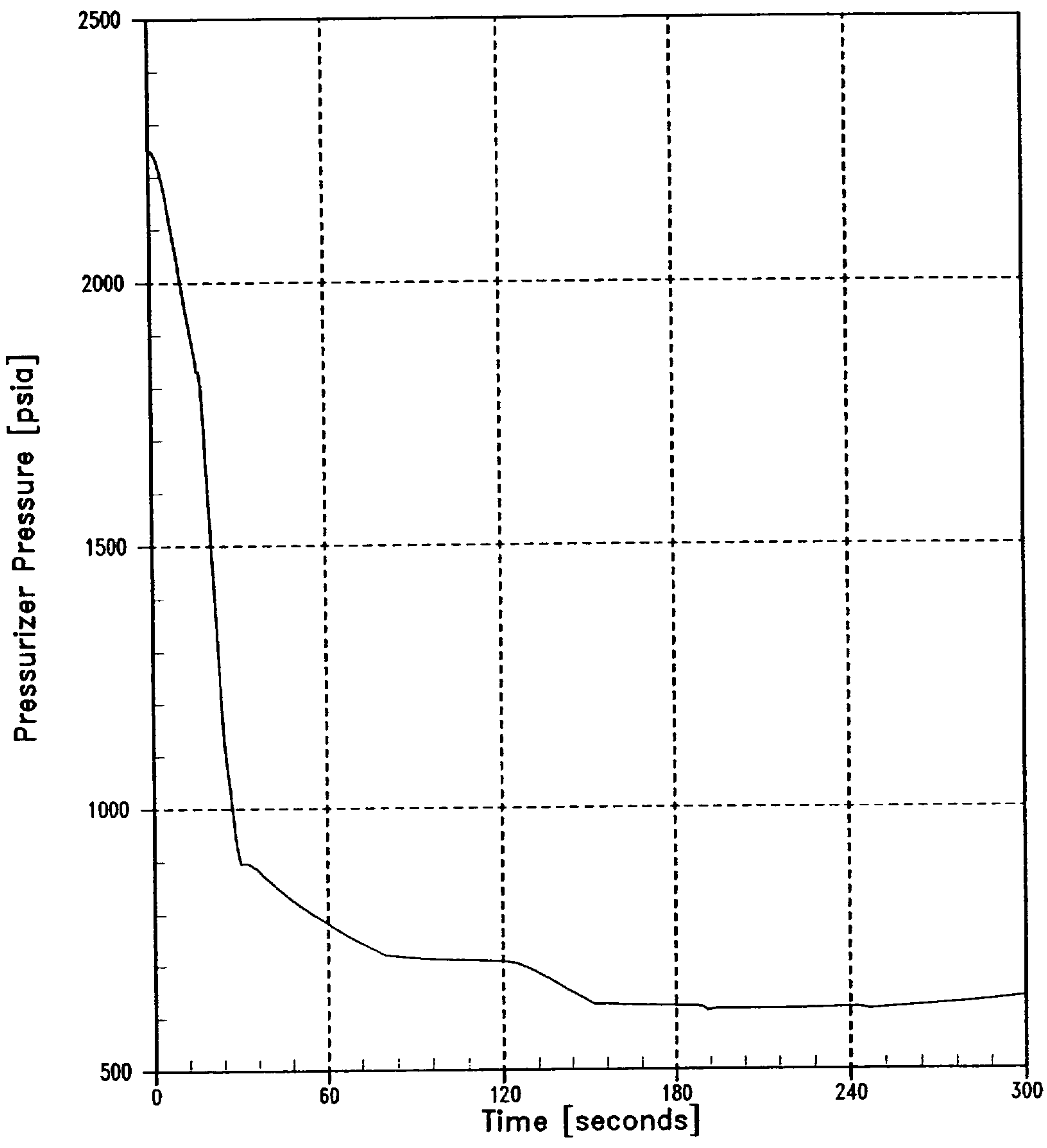

Figure 14.2.5-10

Main Steam Line Break

With a Loss of Offsite Power – Heat Flux vs. Time

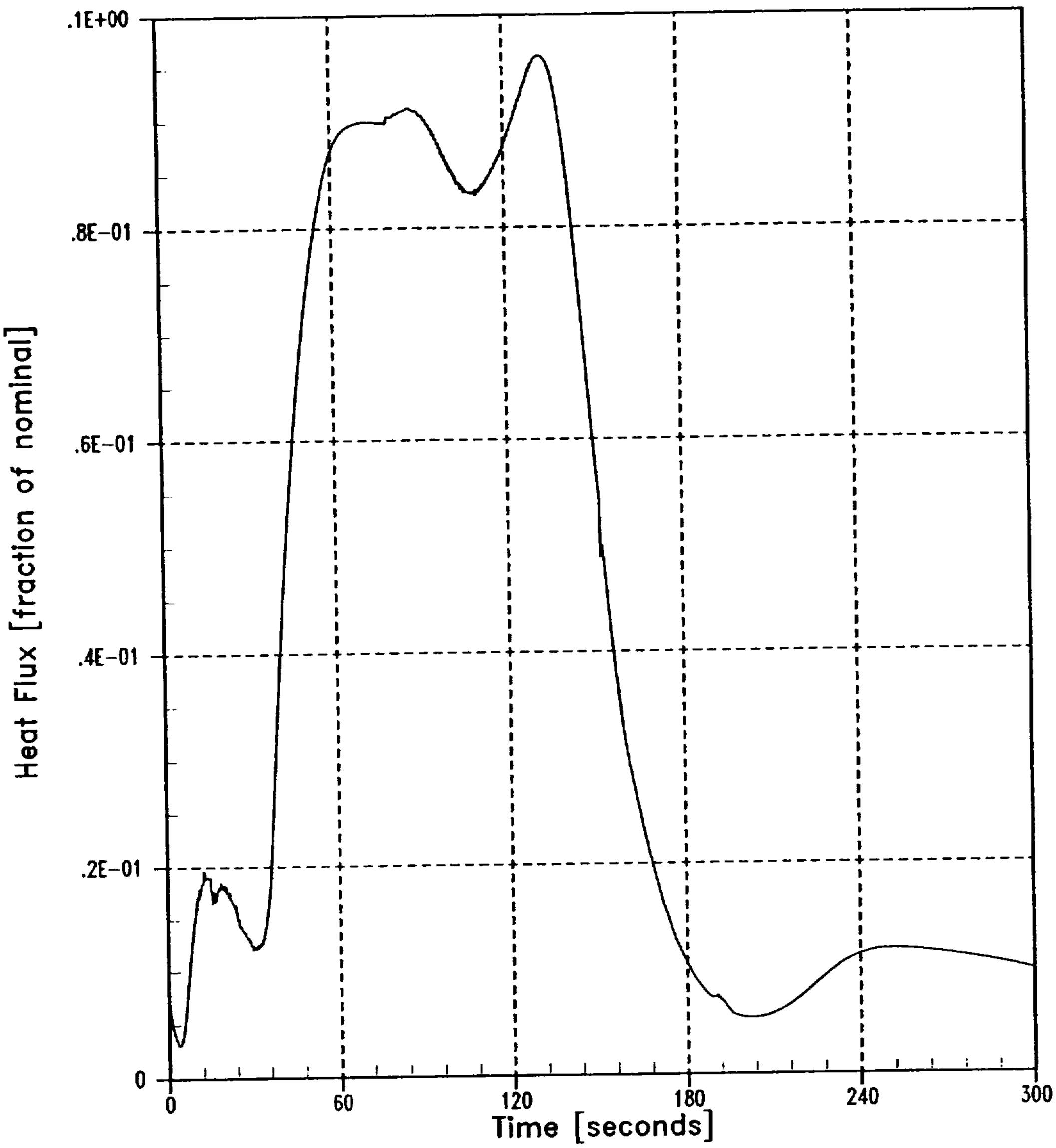

Figure 14.2.5-11

Main Steam Line Break

With a Loss of Offsite Power – SG Outlet Nozzle Flow vs. Time

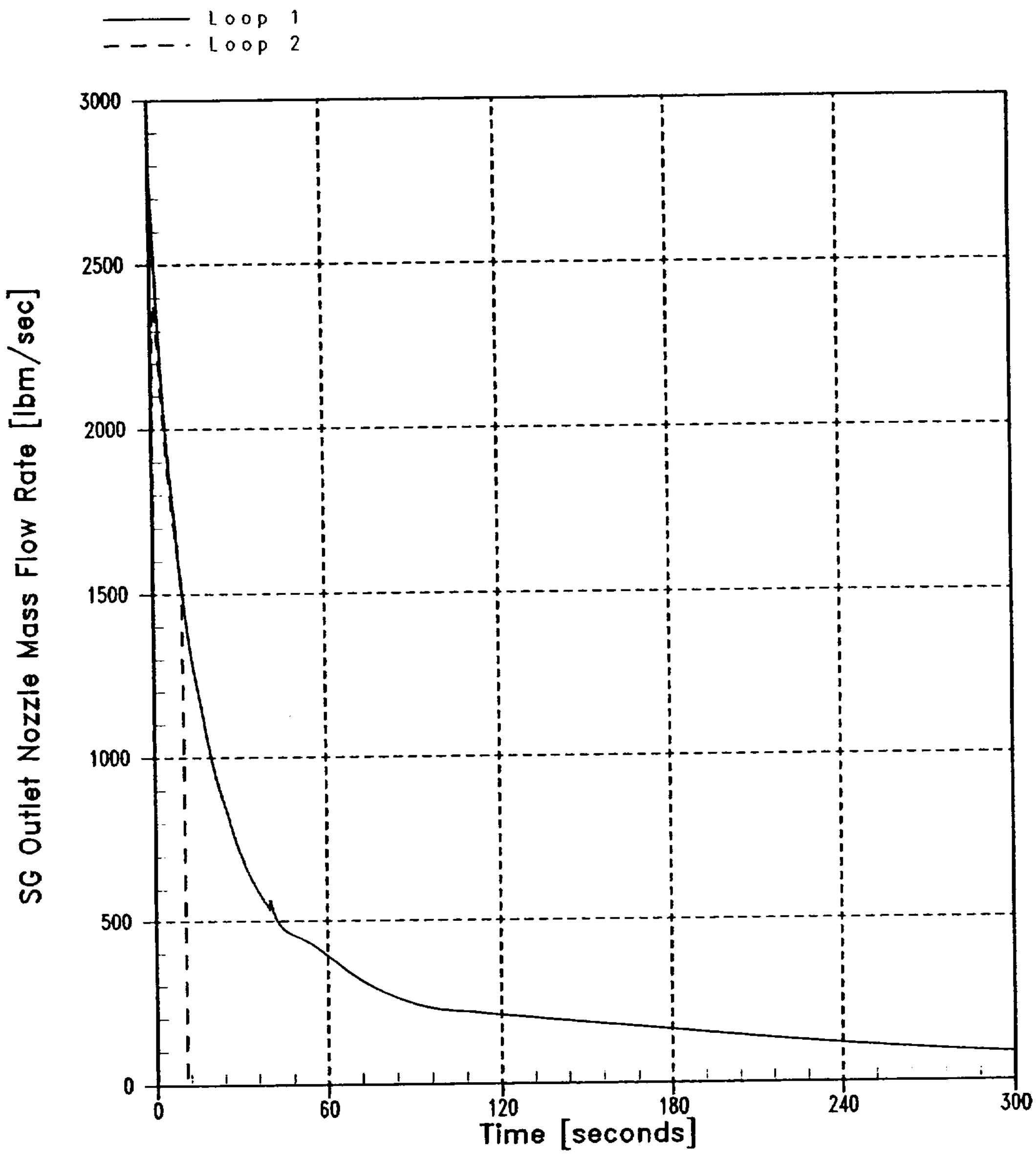

Figure 14.2.5-12

Main Steam Line Break

With a Loss of Offsite Power - Reactivity vs. Time

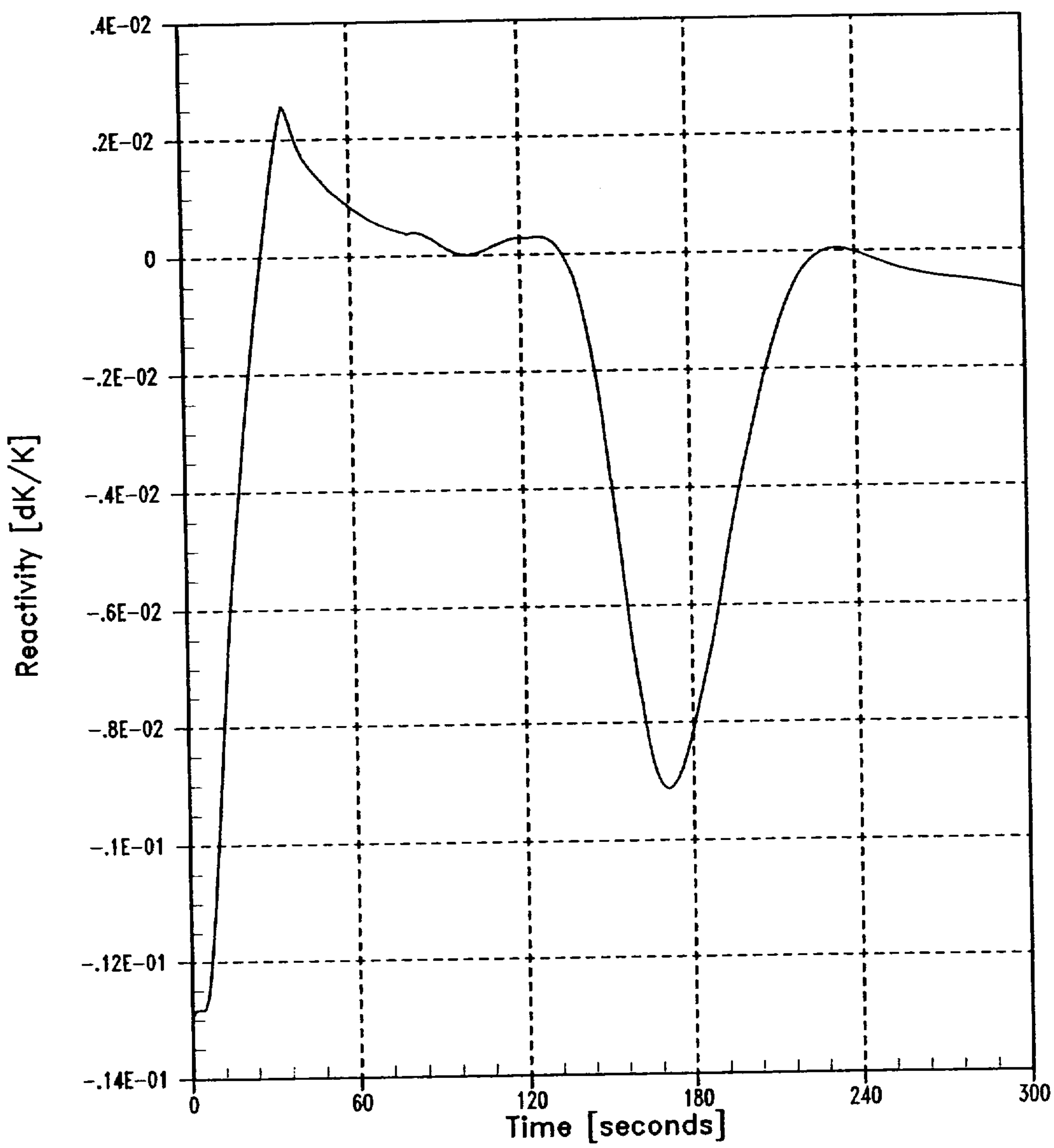

Figure 14.2.5-13

~~concern. Although the limiting temperature profile exceeds the containment design temperature~~ of 268°F, containment structural limits are not exceeded. The short duration of the temperature spike and the method of heat transfer to the containment shell precludes shell temperature from exceeding the design temperature.

MSLB	Containment Peak Pressure (psia)	Containment Peak Temperature (°F)
14MYY0	60.5/60.7	267.7/330.0
01NYY0	39.3/60.7	298.9/330.0

Section 14.2.5 changes suggested earlier.

outside scope of analysis

Conclusions

The analyses have shown that the main steam line break acceptance criteria are satisfied.

Although DNB and possible clad perforation are not precluded in the acceptance criteria, the safety analysis has demonstrated that DNB does not occur, provided that core $F_{\Delta H}$ under steam line break conditions is ≤ 5.00.

The peak pressure for the limiting containment response cases can not exceed the containment design pressure. The limiting temperature profile also does not create an environmental qualification concern for equipment in containment.

Based on the preceding analyses, the radiological significance of a steam line break would depend on the activity levels in the secondary loop of the failed steam generator. The consequence of a long-term 5-gpm leak rate has been considered in Section 14.2.4. However, even if it is conservatively assumed that all of the reactor coolant activity associated with 1% defective fuel cladding is suddenly expelled into the steam generator the resultant thyroid dose at the site boundary would be 4.7 rem and the resultant whole body dose would be 0.51 rem. A decontamination factor of 10 has been applied to the iodine inventory. The consequences of ~~these postulated accidents are well below the guidelines of 10 CFR 100.~~

14.2.6 RUPTURE OF A CONTROL ROD DRIVE MECHANISM HOUSING (RCCA EJECTION)

Description of Accident

This accident is a result of an extremely unlikely mechanical failure of a control rod mechanism pressure housing such that the Reactor Coolant System pressure would then eject the RCCA and drive shaft. The consequences of this mechanical failure, in addition to being a minor loss-of-coolant accident, may also be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage for severe cases. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high neutron flux signals.

Certain features in Westinghouse pressurized water reactors are designed to preclude the possibility of a rod ejection accident, and to limit the consequences if the accident were to occur. These include a sound conservative mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of RCCAs and minimizes the number of assemblies inserted at high power levels.

The mechanical design is discussed in Section 3. An evaluation of the mechanical design and quality control procedures indicates that a failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core should not be considered credible for the following reasons:

1. Each control rod drive mechanism housing is completely assembled and shop-tested at 4100-psi.

2. The mechanism housings are individually hydro-tested as they are installed to the head adapters in the reactor vessel head, and checked during the hydro-test of the completed Reactor Coolant System.

3. Stress levels in the mechanism are not affected by anticipated system transients at power, or by the thermal movement of the coolant loops. Movements induced by the design earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class A components.

4. The latch mechanism housing and rod travel housing are each a single length of forged Type-304 stainless steel. This material exhibits excellent notch toughness at all temperatures that are encountered.

A significant margin of strength in the elastic range, together with the large energy absorption capability in the plastic range, gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy-type rod welds. Administrative regulations require periodic inspections of these (and other) welds.

no change

Even if a rupture of the control rod mechanism housing is postulated, the operation of a chemical shim plant is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with RCCAs inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated by boron changes. Further, the location and grouping of control rod banks are selected during nuclear design to lessen the severity of an ejected assembly. Therefore, should an RCCA be ejected from the reactor vessel during normal operation, there would probably be no reactivity excursion since most of the RCCAs are fully withdrawn from the core, or a minor reactivity excursion if an inserted assembly is ejected from its normal position.

However, it may occasionally be desirable to operate with larger than normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all assemblies is continuously indicated in the control room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one assembly deviates from its bank. There are low and low-low level insertion monitors with visual and audio signals. Operating instructions require normal boration when receiving either alarm. The RCCA position monitoring and alarm systems are described in detail in Section 7.

The reactor protection in the event of a rod ejection accident has been described in WCAP-7306 (Reference 6).

Disregarding the remote possibility of the occurrence of a control rod mechanism housing failure, investigations have shown that failure of a control rod housing due to either longitudinal or circumferential cracking does not cause damage to adjacent housings such that the severity of the initial accident increases.

Due to the extremely low probability of a rod ejection accident, some fuel damage could be considered an acceptable consequence, provided there is no possibility of the off-site consequences exceeding the guidelines of 10 CFR 100. Although severe fuel damage to a portion of the core may in fact be acceptable, it is difficult to treat this type of accident on a sound theoretical basis. For this reason, criteria for the threshold of fuel failure are established, and it is demonstrated that this limit is not exceeded.

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

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

a. Average fuel pellet enthalpy at the hot spot below 200 cal/g.

b. Average clad temperature at the hot spot below the temperature at which clad embrittlement may be expected.

c. Peak reactor coolant pressure much less than that which would cause damage to the Reactor Coolant System.

The ultimate acceptance criteria for this event is that any consequential damage to either the core or the RCS must not prevent long-term core cooling, and that any offsite dose consequences must be within the guidelines of 10 CFR 100. To demonstrate compliance with these requirements, it is sufficient to show that the RCS pressure boundary remains intact, and that no fuel dispersal in the coolant, gross lattice distortions, or severe shock waves will occur in the core. Therefore, the following acceptance criteria are applied to the RCCA Ejection accident:

a. Maximum average fuel pellet enthalpy at the hot spot must remain below 200 cal/g (360 Btu/lbm).

b. Peak RCS pressure must remain below that which would cause the stresses in the RCS to exceed the Faulted Condition stress limits.

c. Maximum fuel melting must be limited to the innermost 10% of the fuel pellet at the hot spot, independent of the above pellet enthalpy limit.

~~The temperature at which clad embrittlement may become a problem for this accident is presently taken to be 2700°F. The peak reactor coolant pressure limit is taken to be 2750 psia, which is much less than that required to strain the reactor vessel or piping to its minimum specified yield strength.~~

Method of Analysis

The analysis of the control rod ejection accident requires modeling of the neutron kinetics coupled with the fuel and clad heat up condition and the thermal hydraulics of the coolant channel. The analysis is performed by first calculating the core average neutronic response and then using the resulting core average power response as a forcing function for the hot spot thermal evaluation.

A 1-D axial kinetics model is used for the analysis of the core average response since it allows for a more realistic representation of the spatial effects of axial moderator feedback, power distribution, and RCCA movement. The moderator reactivity effect is included by correlating reactivity with moderator density, thereby including the effects of coolant temperature, pressure, and voiding. The Doppler reactivity effect is correlated as a function of fuel temperature. The largest temperature rise during the transient, and hence the largest reactivity effects, occurs in channels where the power is higher than average. As a result, when a 3-D space time kinetics calculation is not performed, weighting factors are applied as multipliers to the average channel Doppler reactivity feedback to account for spatial reactivity feedback effects.

The average core energy addition, calculated as described above, is multiplied by the appropriate hot channel factors, and the hot spot analysis is performed using a detailed fuel and clad transient heat-transfer computer code. This computer code calculates the transient temperature distribution in a cross-section of a metal-clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power vs time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A parabolic radial power distribution is used within the fuel rod.

The computer code uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong ~~a transition boiling~~ correlation to determine the film boiling coefficient after DNB (Reference 9). The DNB heat flux is not calculated, instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat-transfer coefficient may be calculated by the code; however, it is adjusted in order to force the full-power steady-state temperature distribution to agree with the fuel heat-transfer design codes.

~~Since the calculations result in maximum fuel enthalpies less than those corresponding to catastrophic fuel failures, the system pressure surge is calculated on the basis of conventional heat transfer from the fuel. The pressure surge model includes prompt heat generation in the coolant, fluid transport in the system, heat transfer in the steam generators, and the action of relief and safety valves. No credit is taken for pressure reduction caused by the assumed failure of the control rod pressure housing.~~

The computer codes used to perform the analyses are identified in Table 14.0-2. Additional details of the methodology are provided in WCAP-7588 (Reference 10).

~~Input parameters for the analysis are conservatively selected on the basis of values calculated for~~ this core. The more important parameters are discussed below. The following table presents the parameters used in this analysis:

Initial Core Conditions	BOC HZP	BOC HFP	EOC HZP	EOC HFP
Ejected Rod Worth, % ΔK	0.91	0.30	0.92	0.42
Delayed Neutron Fraction, %	0.70	0.70	0.50	0.50
Feedback Reactivity Weighting	1.30	1.60	1.30	1.60
F_q After Rod Ejection	8.20	5.03	11.7	5.10
Number of Pumps Operational	1	2	1	2

The values for ejected rod worths and hot channel factors are calculated using a synthesis of one dimensional and two dimensional calculations. No credit is taken for the flux flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the Rod Insertion Limits. An uncertainty of 10% is conservatively added to the ejected rod worth.

The total transient hot channel factor, F_q, is then obtained by multiplying the axial and radial factors, even though the axial peaks are not coincident under the conditions of calculation.

Moderator reactivity assumptions are conservative compared to actual design values. Positive moderator temperature coefficient is assumed for the beginning of cycle zero power case. No weighting factor is applied to the moderator reactivity feedback. The Doppler reactivity is determined as a function of power level and fuel temperature and is conservative when compared to actual design values. A Doppler weighting factor is applied to account for the missing dimensions in the 1-D axial kinetics simulation. This weighting factor is conservative when compared to 3-D space time kinetics computation.

Calculations of the effective delayed neutron fraction (βeff) have yielded values of no more than 0.70% at BOC and no < 0.50% at EOC. The accident is sensitive to β if the ejected rod worth is equal to or greater than β, as in the zero power transients. No uncertainty is applied directly to the value of β since the calculation of the rod worth and hot channel factor is considered very conservative.

The trip reactivity used for the analysis is shown in Figure 14.0.1. The start of rod motion occurs 0.5 seconds after the high neutron flux trip point is reached. This delay is assumed to consist of 0.2 seconds for the instrument channel to produce a signal, 0.15 seconds for the trip breaker to open and 0.15 seconds for the coil to release the rods. The choice of such a conservative insertion rate and delay for rod motion means that there is over 1 second after the trip setpoint is reached before significant shutdown reactivity is inserted into the core. This is a particularly important conservatism for hot full-power accidents.

Results

Results of the beginning and end of life full and zero power rod ejection analyses are shown in ~~Figures 14.2.6-1 through 14.2.6-12. These results are also summarized below. The acceptance~~

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~~criteria on average fuel enthalpy (200 cal/g) and average clad temperature (2700°F) are not~~ exceeded. Therefore, fuel is not expected to be dispersed into the coolant under the most severe conditions of this transient.

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It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, < 15% of the rods entered DNB. (This corresponds to 2% of the core volume.) The position with regard to fission product release is therefore much better than the double-ended coolant pipe break.

A detailed calculation of the pressure surge shows that assuming an initial pressure of 2250 psia, the peak pressure reached in the transient is well within the criteria of 2750 psia and, therefore, no damage to the Reactor Coolant System will occur.

In the region of the hot spot there is a large temperature gradient. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a force tending to bow the mid-point of the rods toward the hot spot. Physics calculations indicate that the net result of this is a negative reactivity insertion. In practice, no significant bowing is anticipated since the structural rigidity of the core is more than sufficient to withstand the forces produced.

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

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

Control Rod Ejection	Max Clad Temp. (°F)	Max Fuel Centerline Temp.(°F)	Max Energy Deposition (cal/g)	RCS Pressure (psia)	MS Pressure (psia)
BOC Full Power	[redacted]/2700	[redacted]/4700	[redacted]/200	[redacted]/2750	863/1210
BOC Zero Power	2555/2700	3925/4700	174/200	[redacted]/2750	1027/1210
EOC Full Power	[redacted]/2700	[redacted]/4700	[redacted]/200	[redacted]/2750	864/1210
~~EOC Zero Power~~	~~2688/2700~~	~~4031/4700~~	~~182/200~~	~~2277/2750~~	~~1022/1210~~

Conclusions

Even on the most pessimistic basis, the analyses indicated that the fuel and clad limits were not exceeded. It was concluded that there was no danger of sudden fuel dispersal into the

The overpressurization of the RCS and number of rods in DNB, as a result of a postulated ejected rod, have both been analyzed on a generic basis for Westinghouse PWRs as detailed in Reference 10.

If the safety limits for fuel damage are not exceeded, there is little likelihood of fuel dispersal into the coolant or a sudden pressure increase from thermal-to-kinetic energy conversion. The pressure surge for this analysis can, therefore, be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

A detailed calculation of the pressure surge for an ejection worth of one dollar at BOL, hot full power, indicates that the peak pressure does not exceed that which would cause stresses in the RCS to exceed their Faulted Condition stress limits. Since the severity of the Kewaunee analysis does not exceed this worst case analysis, the RCCA Ejection accident will not result in an excessive pressure rise or further damage to the RCS.

Reference 10 also documents a detailed three-dimensional THINC-III calculation, which demonstrates an upper limit to the number of rods-in-DNB for the RCCA Ejection accident as 10%. Since the severity of the Kewaunee analysis does not exceed this worst case analysis, the maximum number of rods in DNB following a RCCA Ejection will be less than 10%, which is well within the 15% used in the radiological dose evaluation. The most limiting break size resulting from a RCCA Ejection will not be sufficient to uncover the core or cause DNB at any later time. Since the maximum number of fuel rods experiencing DNB is limited to 15%, the fission product release will not exceed that associated with the guidelines of 10 CFR 100.

In calculating the nuclear power and hot spot fuel rod transients following RCCA Ejection, the following conservative assumptions are made:

a. The RTDP is not used for the RCCA Ejection analysis. Instead, the STDP (maximum uncertainties in initial conditions) is employed. The analysis assumes uncertainties of 2.0% in nominal core power, 6.0°F in nominal vessel T_{avg}, and 50 psi in nominal pressurizer pressure.

b. A minimum value for the delayed neutron fraction for BOC and EOC conditions is assumed which increases the rate at which the nuclear power increases following RCCA Ejection.

c. A minimum value of the Doppler power defect is assumed which conservatively results in the maximum amount of energy deposited in the fuel following RCCA Ejection. A minimum value of the moderator feedback is also assumed. A positive moderator temperature coefficient is assumed for the beginning of cycle, zero power case.

d. Maximum values of ejected RCCA worth and post-ejection total hot channel factors are assumed for all cases considered. These parameters are calculated using standard nuclear design codes for the maximum allowed bank insertion at a given power level as determined by the rod insertion limits. No credit is taken for the flux flattening effects of reactivity feedback.

e. The start of rod motion occurs 0.65 seconds after the high neutron flux trip point is reached.

The analysis is performed to bound operation with Westinghouse 422V+ fuel and a maximum loop-to-loop steam generator tube plugging imbalance of 10%.

Results

Results are presented for the beginning and end of life, full and zero power rod ejection analyses.

<u>BOC, Full Power</u>

Control bank D is assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor are conservatively calculated to be 380 pcm and 4.2, respectively. The maximum fuel stored energy is 167.4 cal/g. The peak hot spot fuel center temperature reaches melting, which is conservatively assumed to occur at 4900°F. However, melting is restricted to less than 10% of the pellet.

<u>BOC, Zero Power</u>

For this condition, control bank D is assumed to be fully inserted and banks B and C are at their insertion limits. The worst ejected rod worth and hot channel factor are conservatively calculated to be 770 pcm and 11.0, respectively. The maximum fuel stored energy is 144.9 cal/g. The peak hot spot fuel center temperature of 3901°F remains below melting, which is conservatively assumed to occur at 4900°F.

EOC, Full Power

Control bank D is assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor are conservatively calculated to be 370 pcm and 5.69, respectively. The maximum fuel stored energy is 170.3 cal/g. The peak hot spot fuel center temperature reaches melting, which is conservatively assumed to occur at 4800°F. However, melting is restricted to less than 10% of the pellet.

EOC, Zero Power

For this condition, control bank D is assumed to be fully inserted and banks B and C are at their insertion limits. The worst ejected rod worth and hot channel factor are conservatively calculated to be 930 pcm and 13.0, respectively. The maximum fuel stored energy is 161.6 cal/g. The peak hot spot fuel center temperature of 4149°F remains below melting, which is conservatively assumed to occur at 4800°F.

A summary of the cases presented above is given in Table 14.2.6-1. The nuclear power and hot spot and cladding temperature transients are presented in Figures 14.2.6-1 through 14.2.6-8.

For all cases, reactor trip occurs very early in the transient, after which the nuclear power excursion is terminated. The reactor will remain subcritical following reactor trip.

Since does not exceed that which would cause stresses to exceed the

coolant. ~~The~~ the pressure ~~surge was shown to be insufficient to exceed 2750 psia, and~~ it was concluded that there was no danger of consequential damage to the primary coolant system. The amount of fission products ~~released~~ as a result of clad rupture during DNB is ~~considerably less than in the case of the double-ended main coolant pipe break (the Design Basis Accident), and~~

further ~~therefore~~ within the guidelines of 10 CFR 100.

14.2.7 TURBINE MISSILE DAMAGE TO SPENT FUEL POOL

DELETED

faulted condition stress limits,

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2. NRC Safety Evaluation Report, "Kewaunee Nuclear Power Plant, Supplemental Safety Evaluation Associated with Amendment No. 132", WO Long (NRC) to ML Marchi (WPSC), Letter No. K-98-124 dated September 3, 1998

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4. Letter WOG 98-066 from KJ Voytell (WOG) to primary representatives (WPS), Transmittal of NRC Closure Letter on the Steam Generator Tube Rupture Uncovery Issue dated March 31, 1993

5. FJ Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture" transactions of the American Society of Mechanical Engineers, February 1965, page 136.

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

7. TG Taxelius, ed. "Annual Report - Spert Project, October 1968, September 1969" Idaho Nuclear Corporation TID-4500, June 1970

8. RC Liimatainen and FJ Testa, "Studies in TREAT of Zircaloy-2-Clad, UO_2 - Core Simulated Fuel Elements" Argonne National Laboratory Chemical Engineering Division Semi-Annual Report, ANL-7225, January-June 1966

9. ~~RF Farman and JO Cermak, "Post DNB Heat Transfer During Blowdown" WCAP-7837, January 1972~~ A.A. Bishop, R.O. Sandberg and L.S. Tong, "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31 (1965).

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

TABLE 14.2.6-1

PARAMETERS USED IN THE ANALYSIS OF THE
ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

Parameters	BOL-HZP	BOL-HFP	EOL-HZP	EOL-HFP
Initial core power level, percent	0	102	0	102
Ejected rod worth, % Δk	0.77	0.38	0.93	0.37
Delayed neutron fraction, %	0.49	0.49	0.43	0.43
Doppler reactivity defect (absolute value), pcm	1000	1000	900	900
Doppler feedback reactivity weighting	2.008	1.139	2.144	1.316
Trip reactivity, percent Δk	1.0	3.5	1.0	3.5
F_Q before rod ejection	N/A	2.5	N/A	2.5
F_Q after rod ejection	11.0	4.2	13.0	5.69
Number of operational pumps	1	2	1	2
Maximum fuel pellet average temperature, °F	3426	3867	3753	3923
Maximum fuel pellet center temperature, °F	3901	4953	4149	4871
Maximum cladding average temperature, °F	2567	2092	2987	2120
Maximum fuel stored energy, cal/g	144.9	167.4	161.6	170.3
Maximum fuel melt, percent	0.0	2.17	0.0	5.89

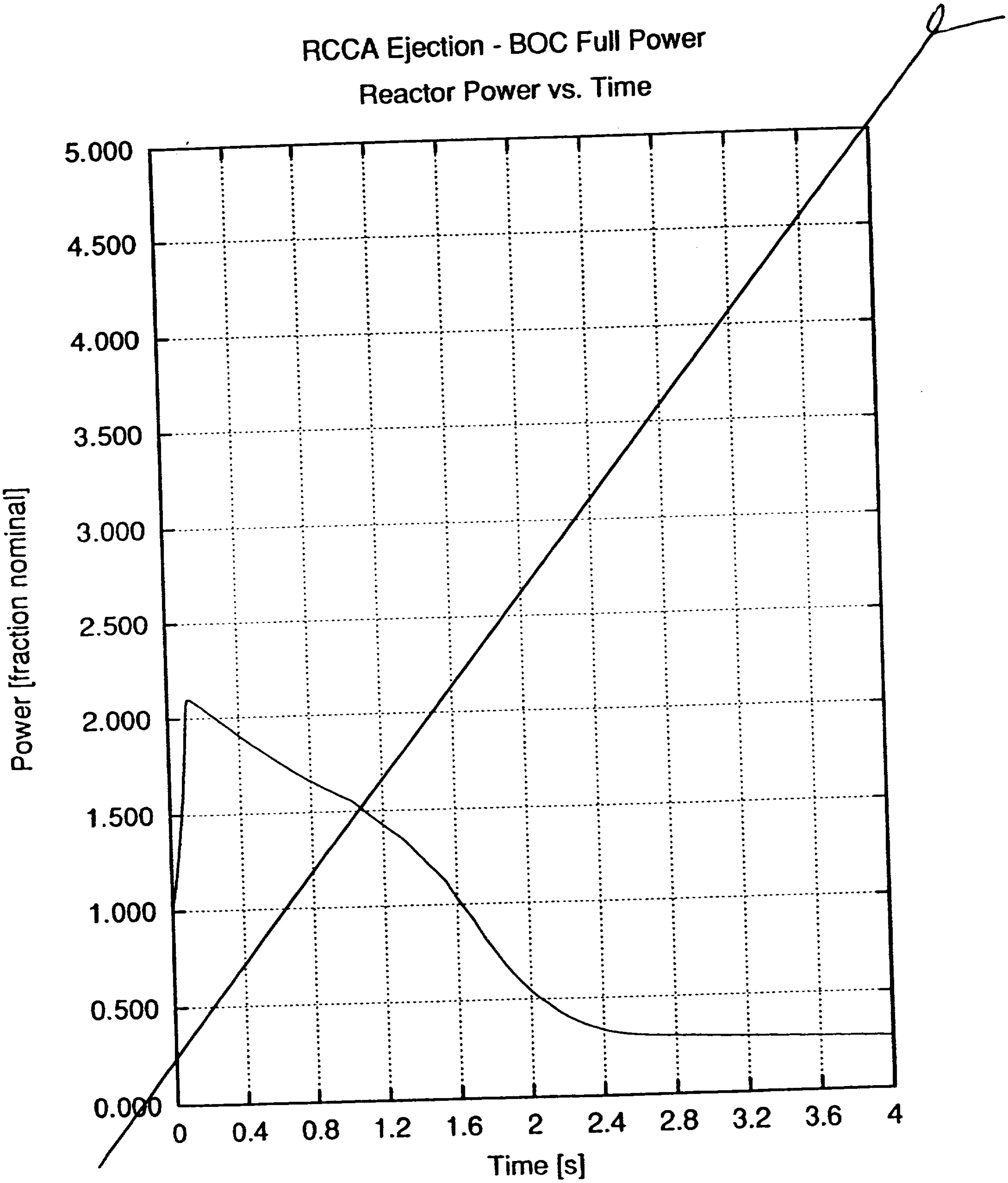

Figure 14.2.6-1

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RCCA Ejection – BOC Full Power
Reactor Power vs. Time

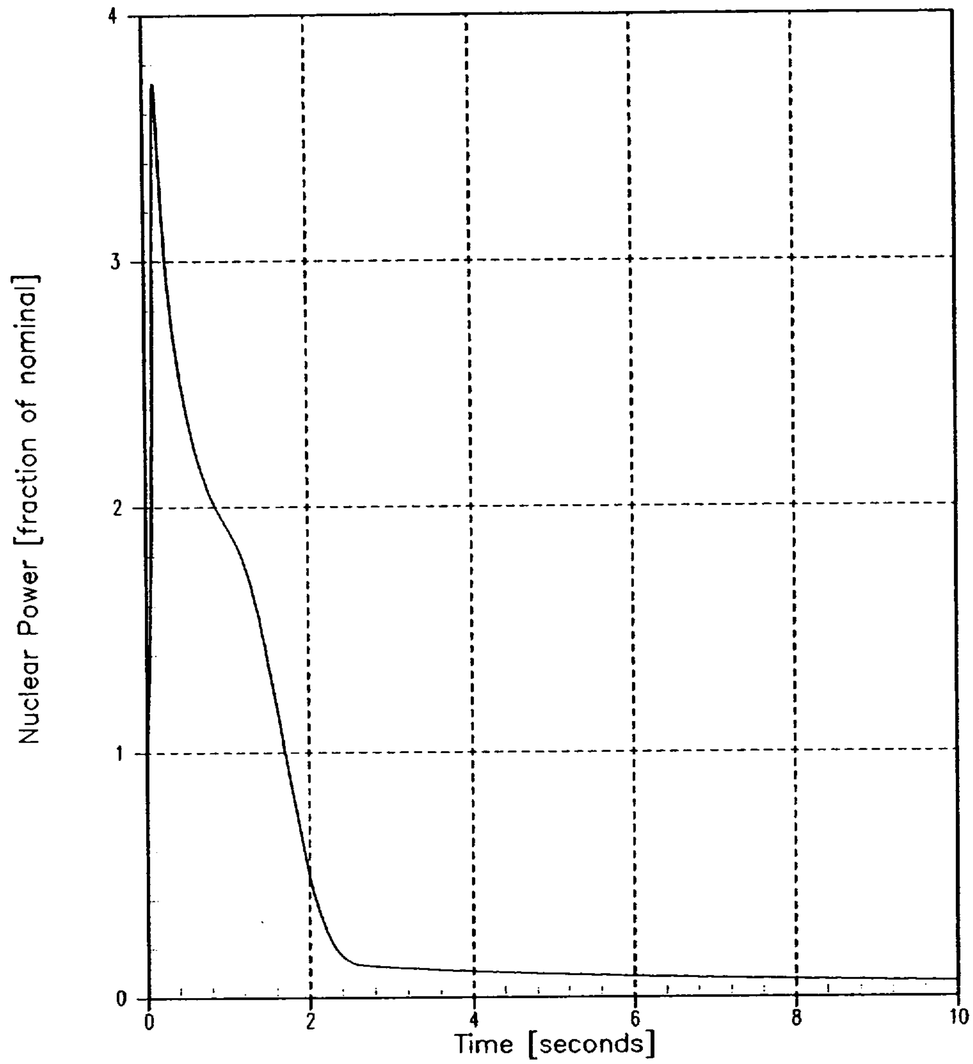

Figure 14.2.6–1

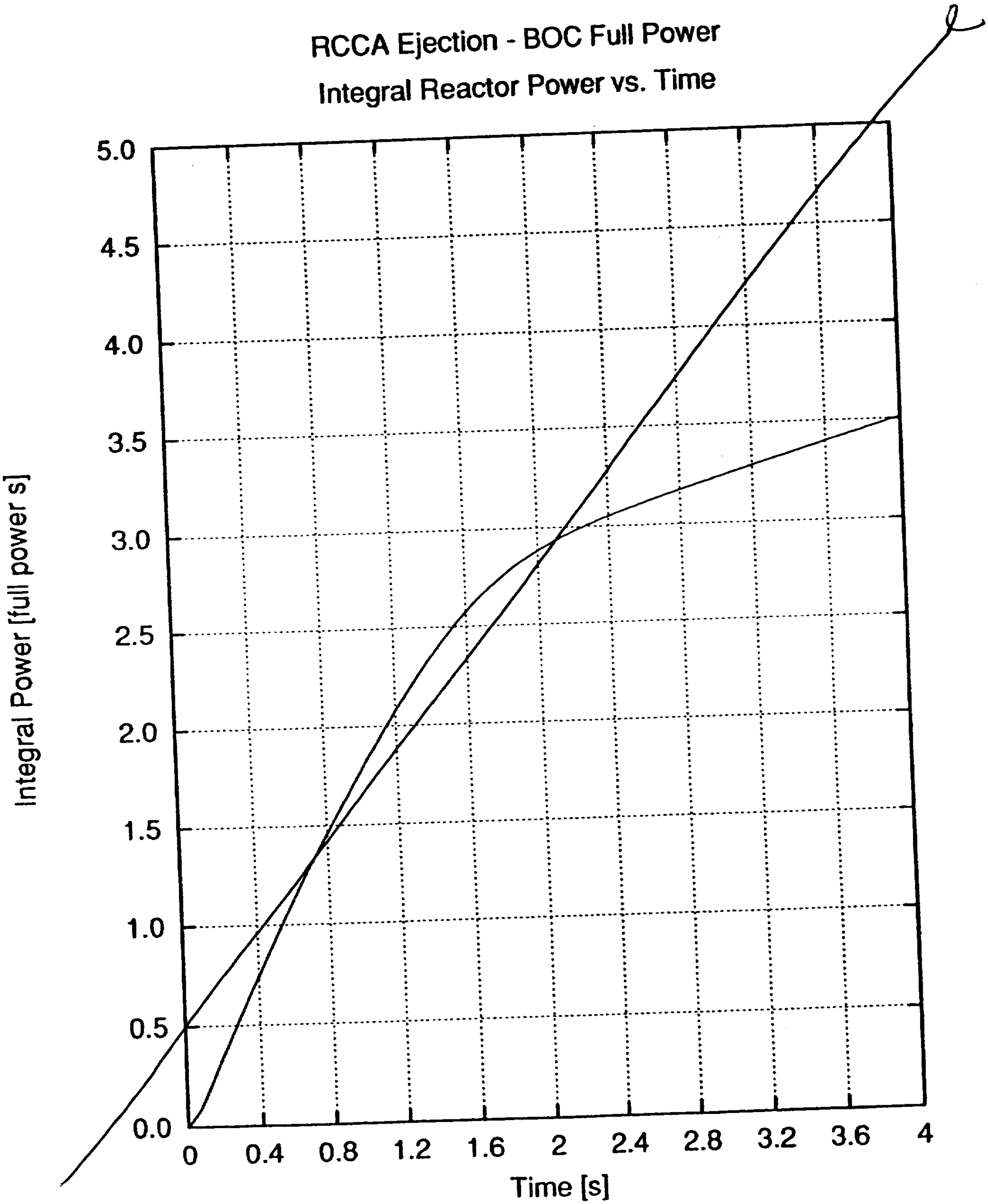

Figure 14.2.6-2

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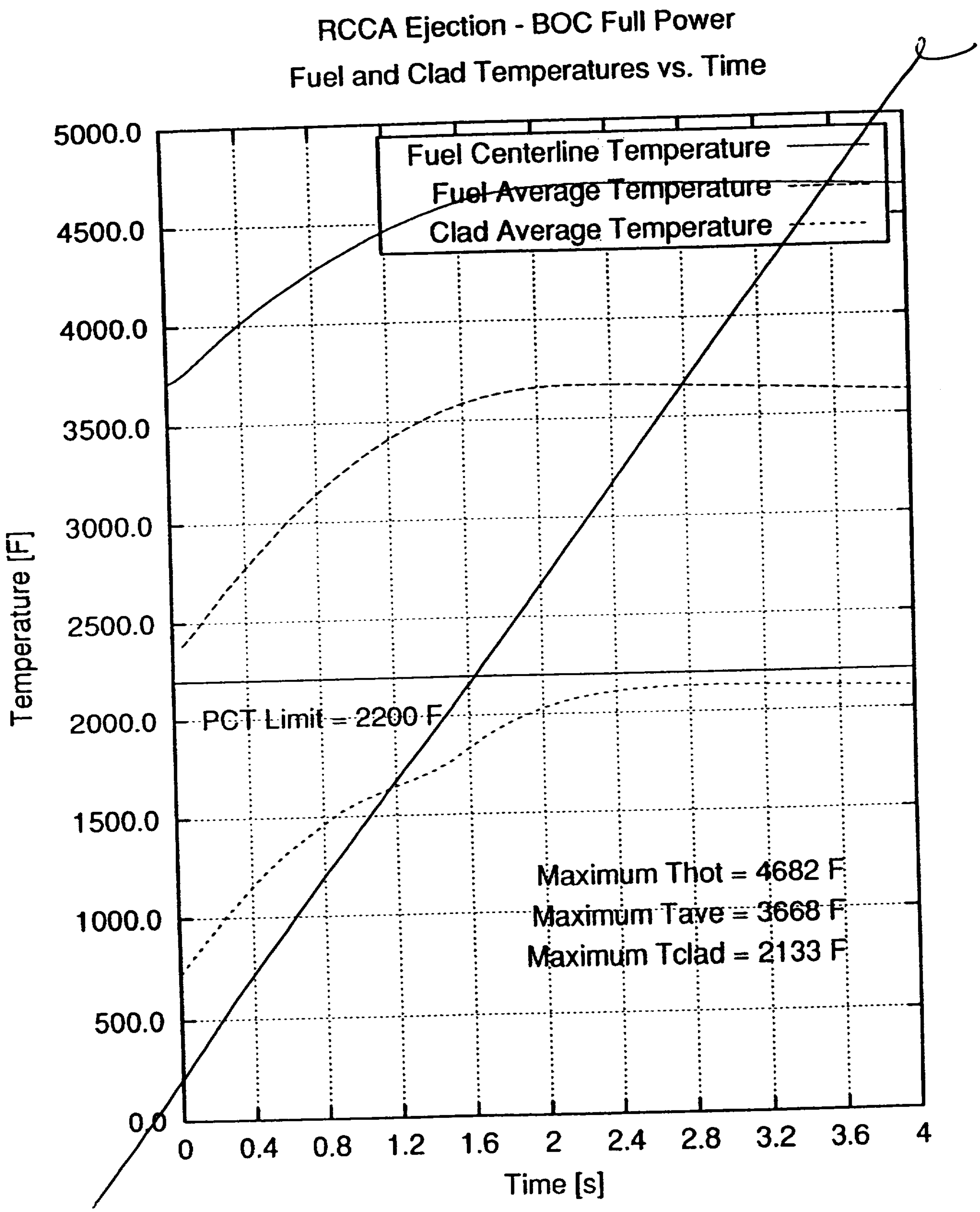

Figure 14.2.6-3

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RCCA Ejection – BOC Full Power
Fuel and Clad Temperatures vs. Time

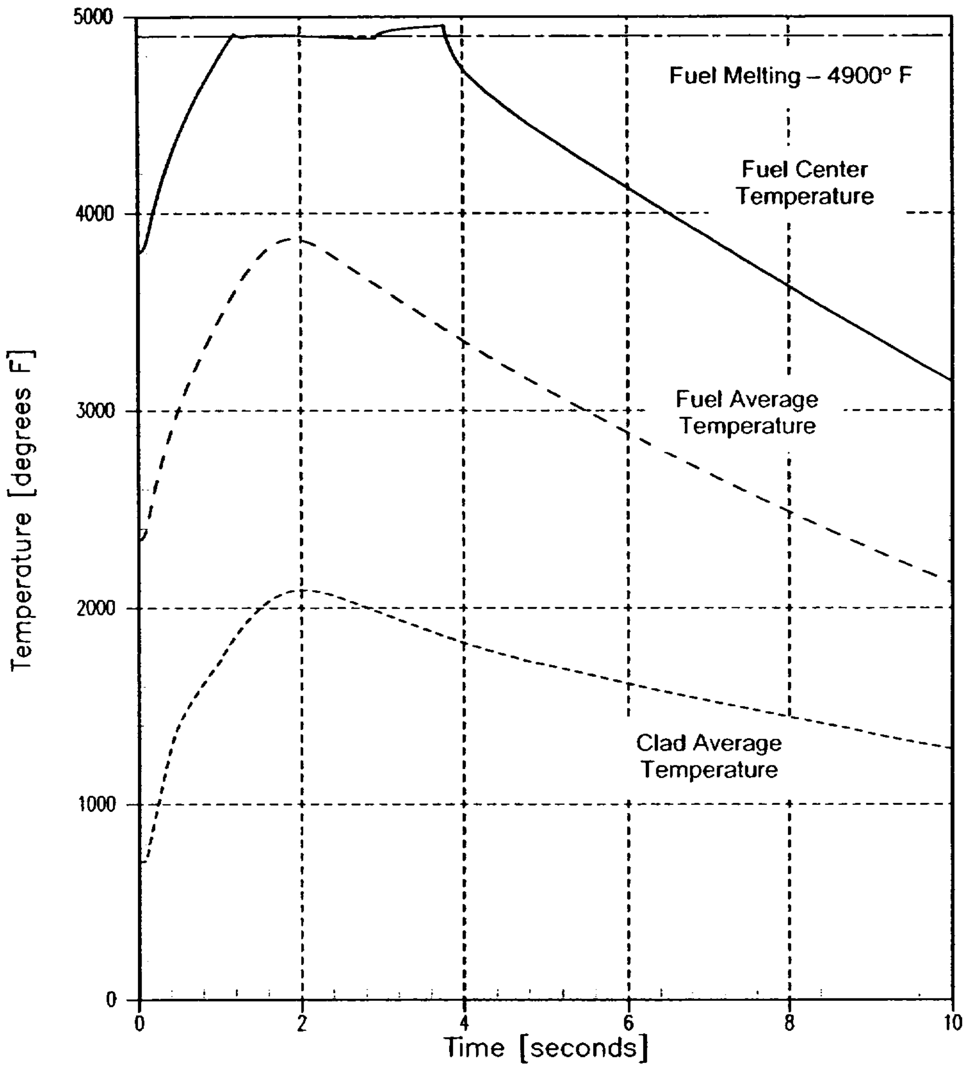

Figure 14.2.6–2

RCCA EJECTION

BOC ZERO POWER

REACTOR POWER vs. TIME

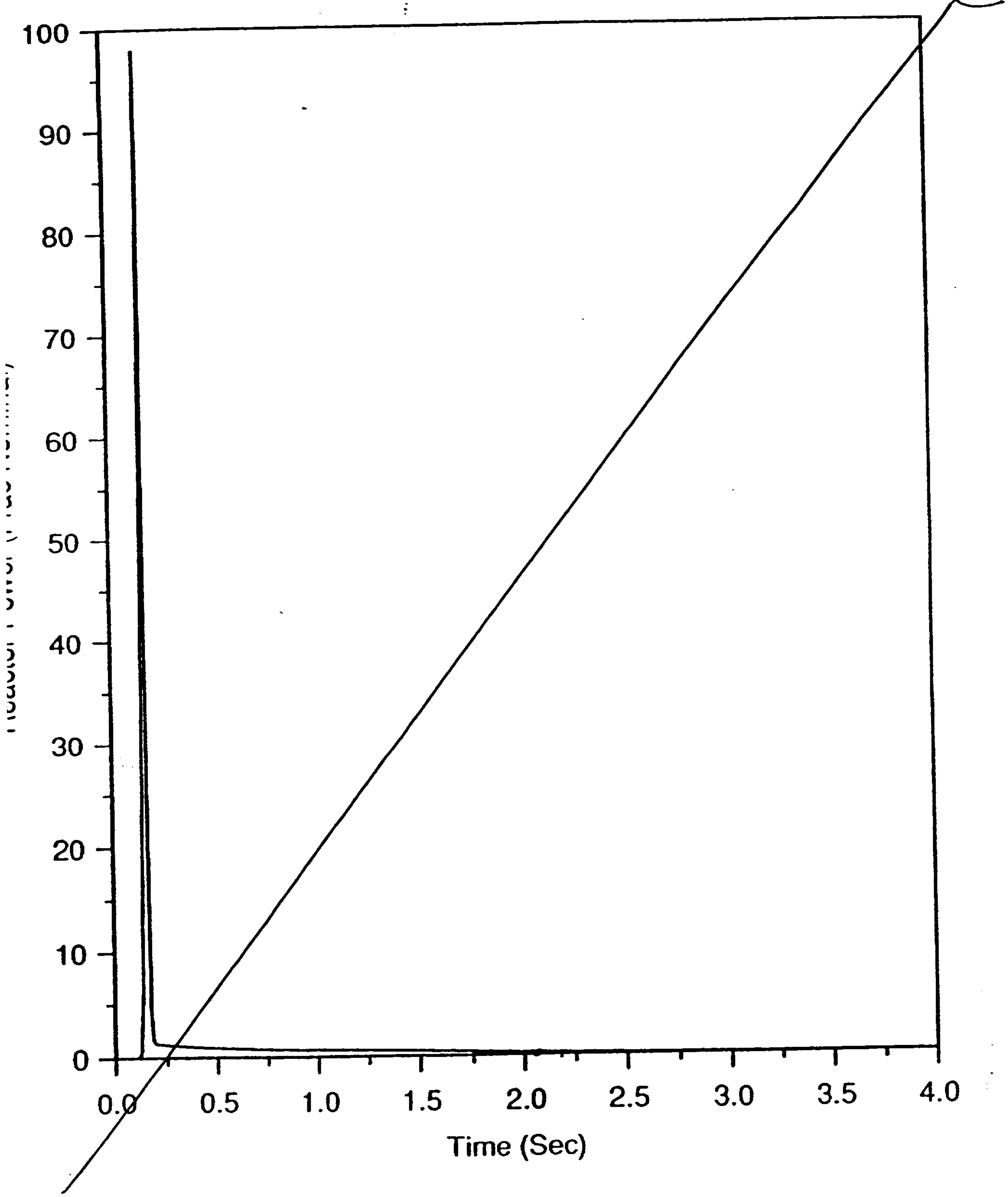

FIGURE 14.2.6-4

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RCCA Ejection – BOC Zero Power
Reactor Power vs. Time

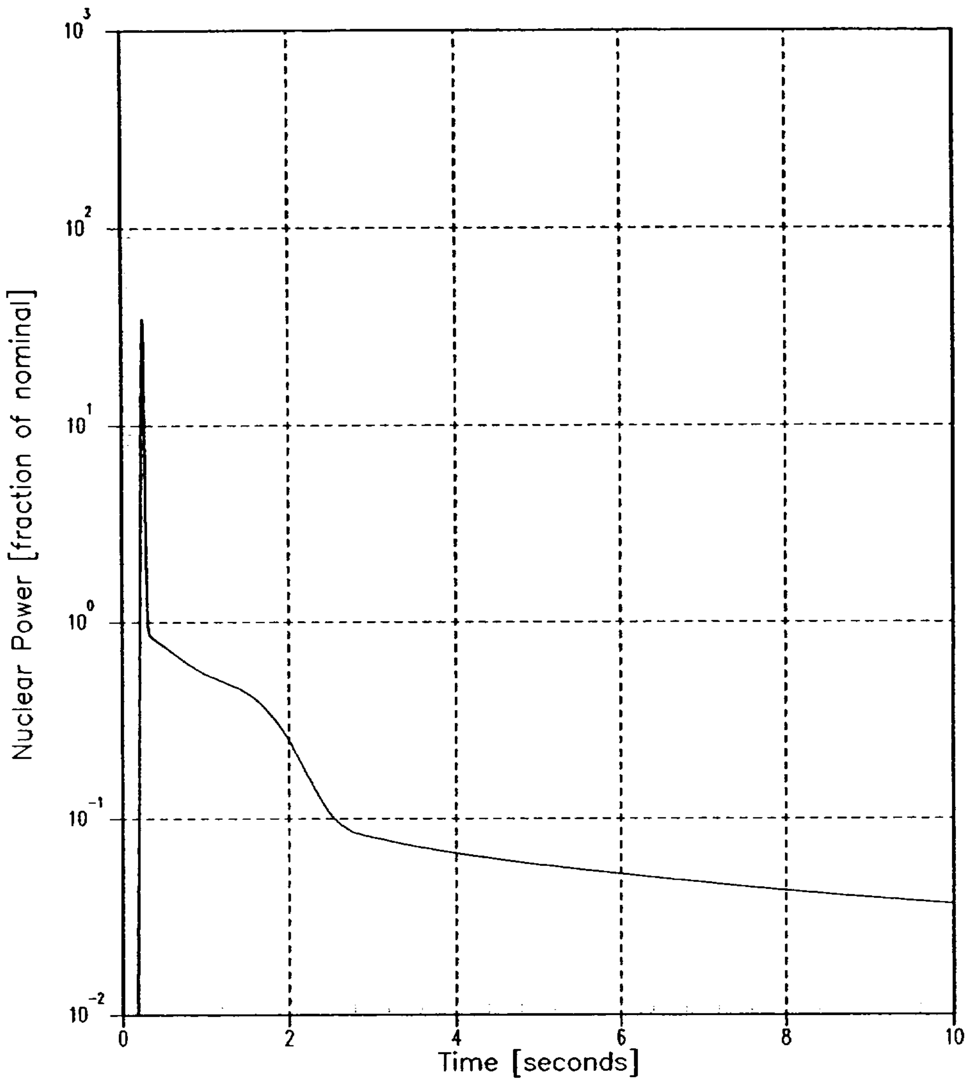

Figure 14.2.6–3

RCCA EJECTION

BOC ZERO POWER

REACTOR INTEGRATED POWER vs. TIME

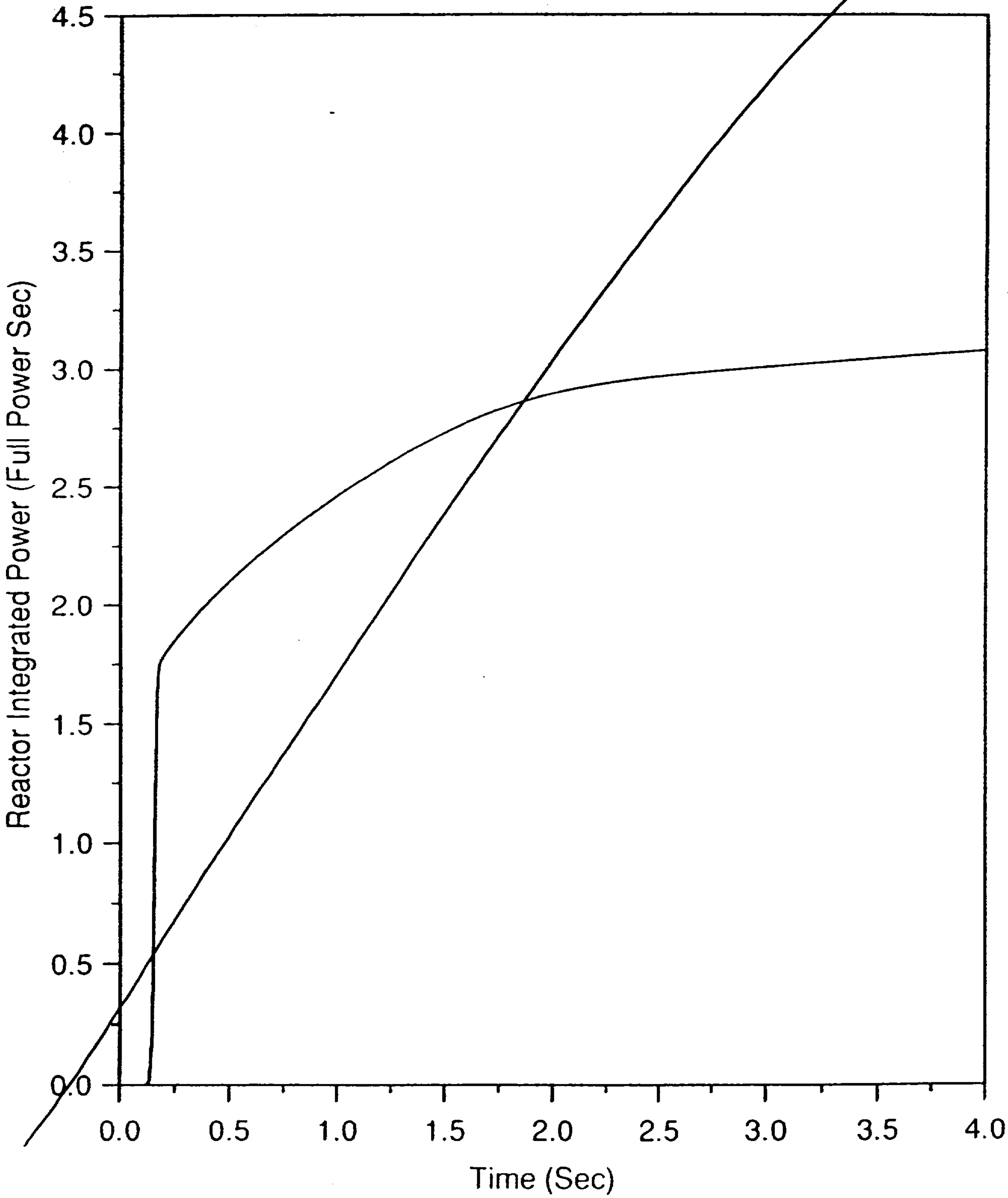

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FIGURE 14.2.6-5

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RCCA EJECTION

BOC ZERO POWER

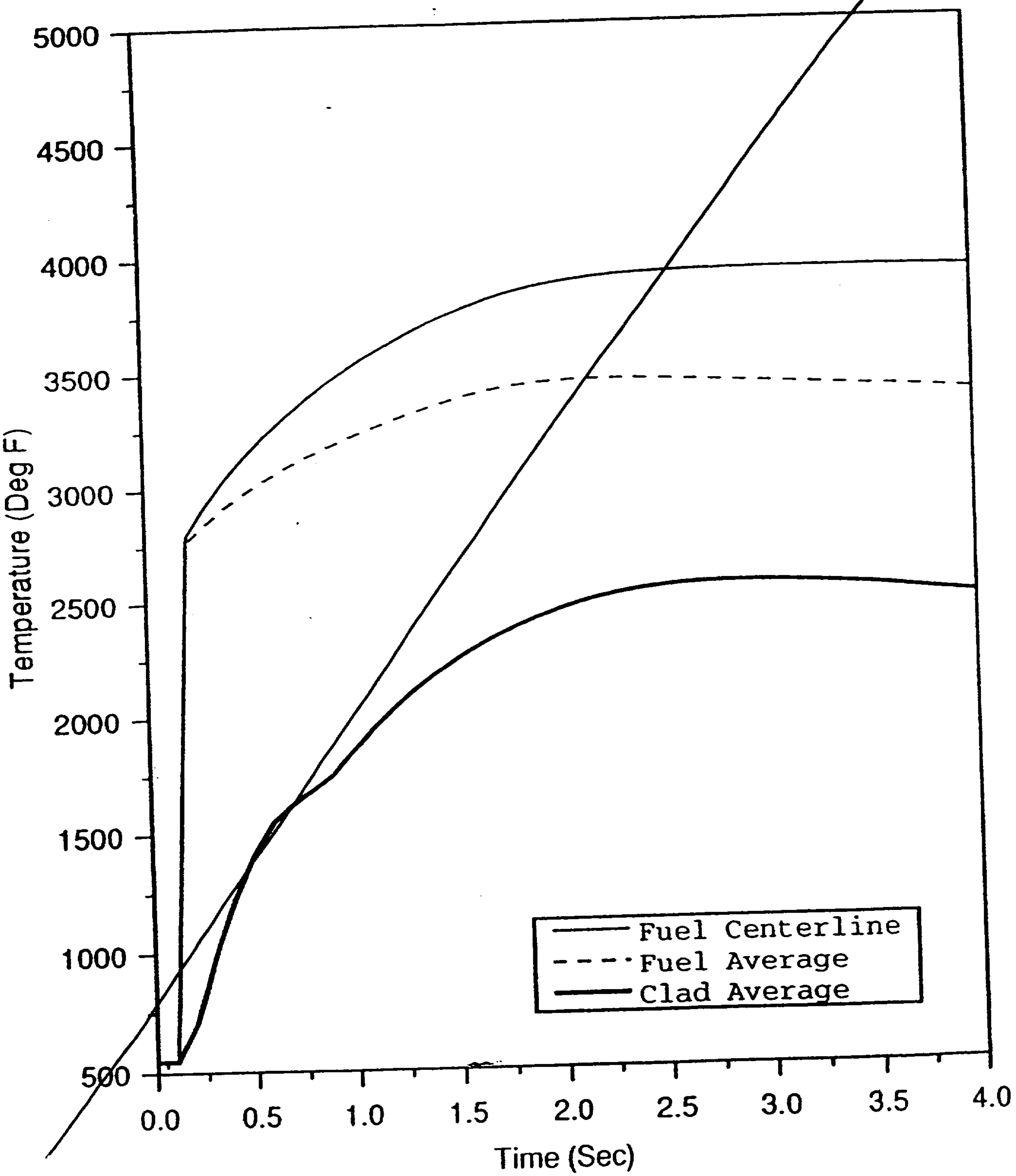

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FIGURE 14.2.6-6

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RCCA Ejection – BOC Zero Power
Fuel and Clad Temperatures vs. Time

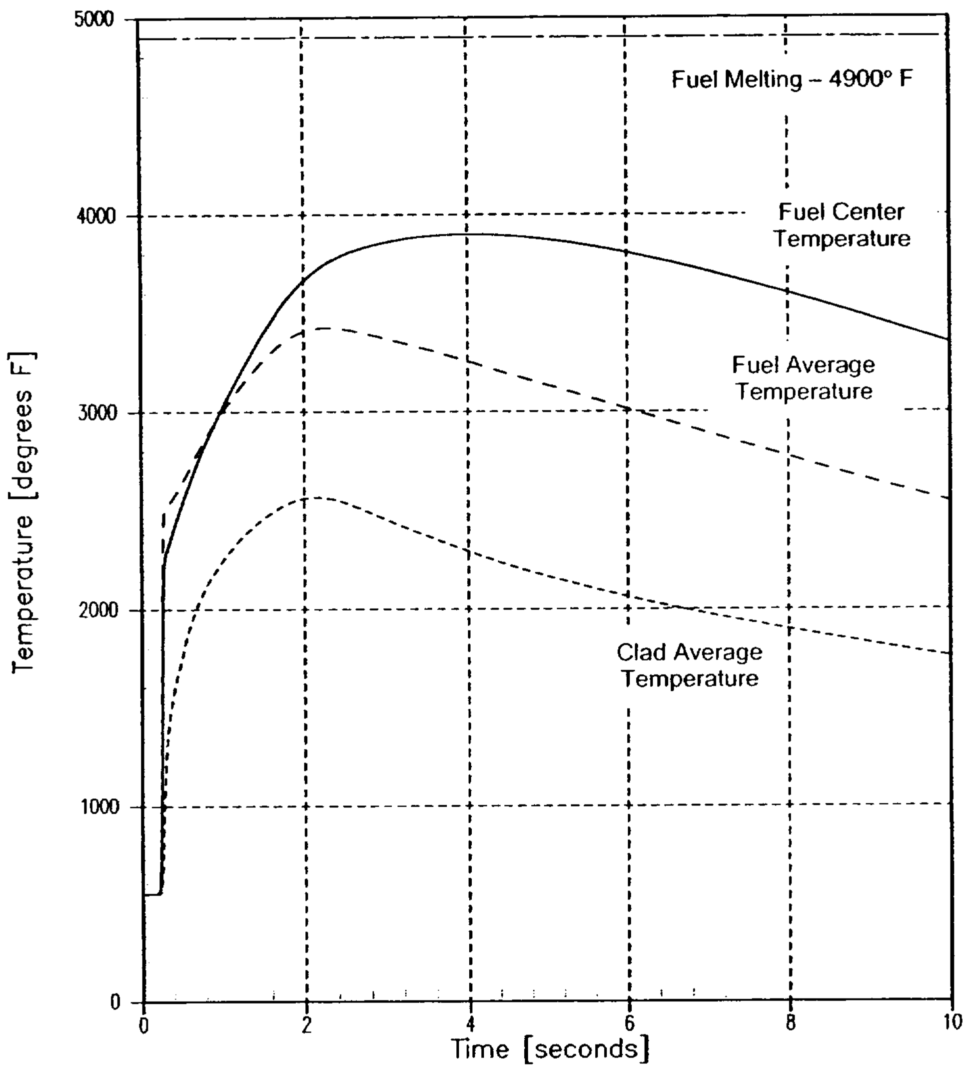

Figure 14.2.6–4

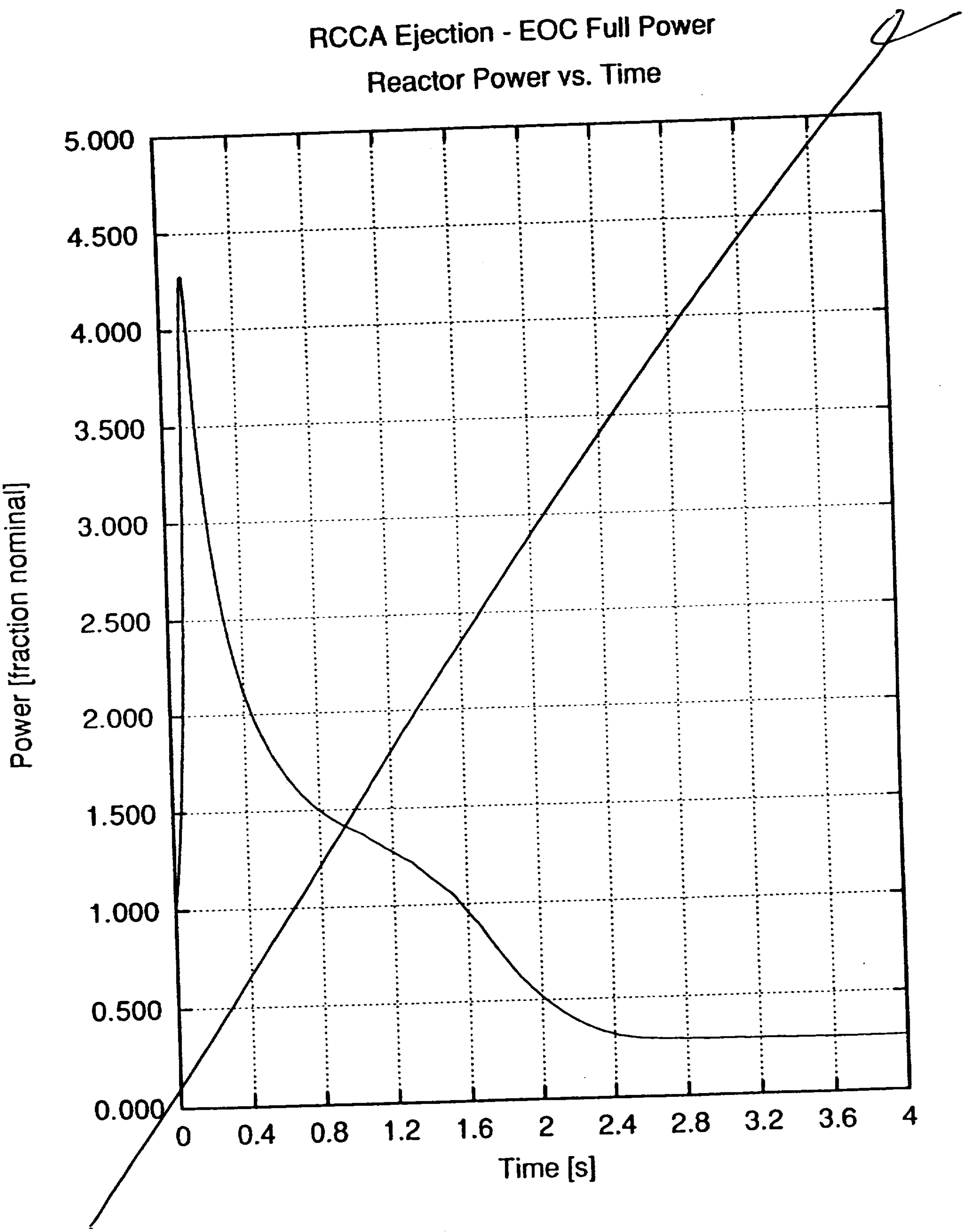

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Figure 14.2.6-7

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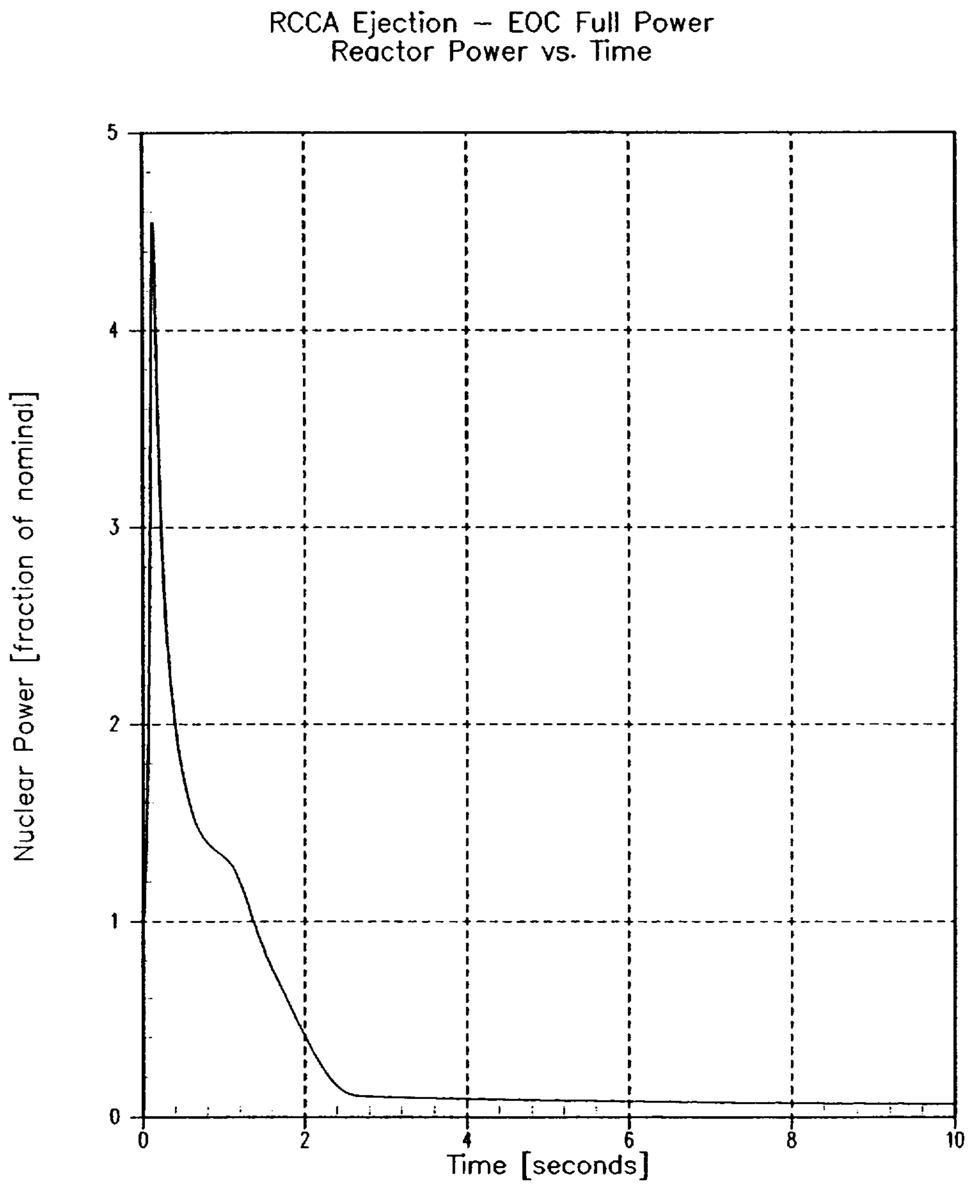

Figure 14.2.6–5

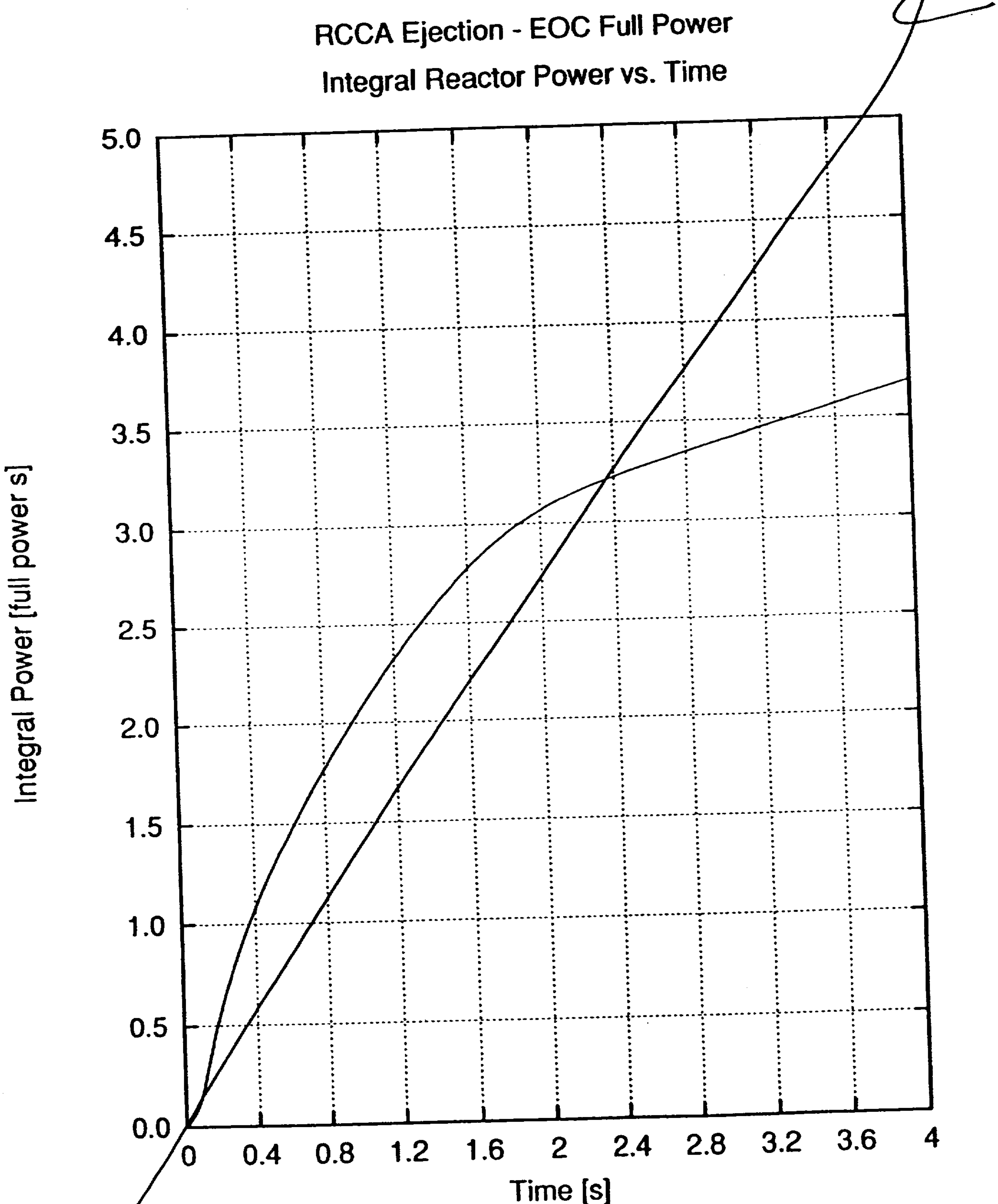

Figure 14.2.6-8

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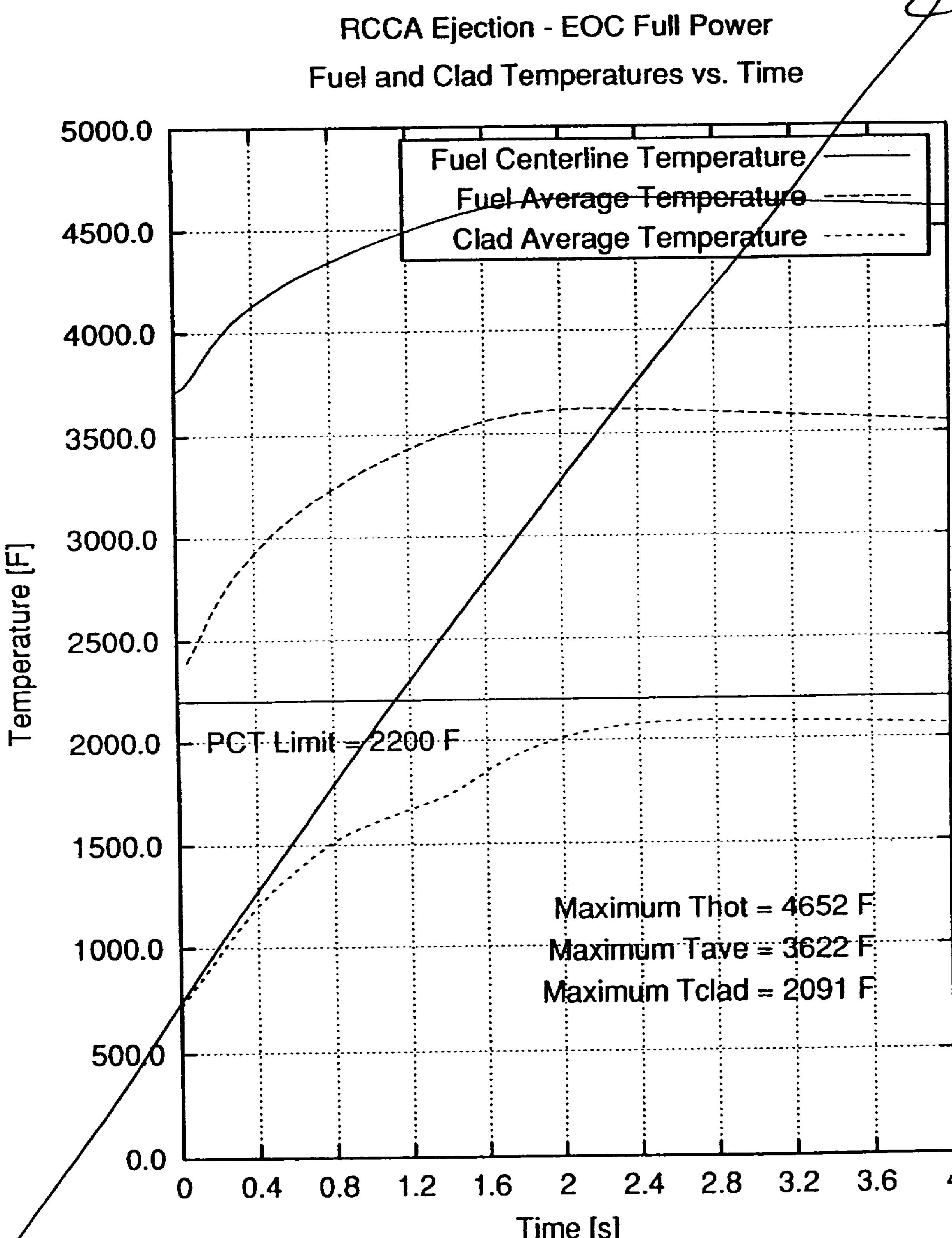

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Figure 14.2.6-9

RCCA Ejection – EOC Full Power
Fuel and Clad Temperatures vs. Time

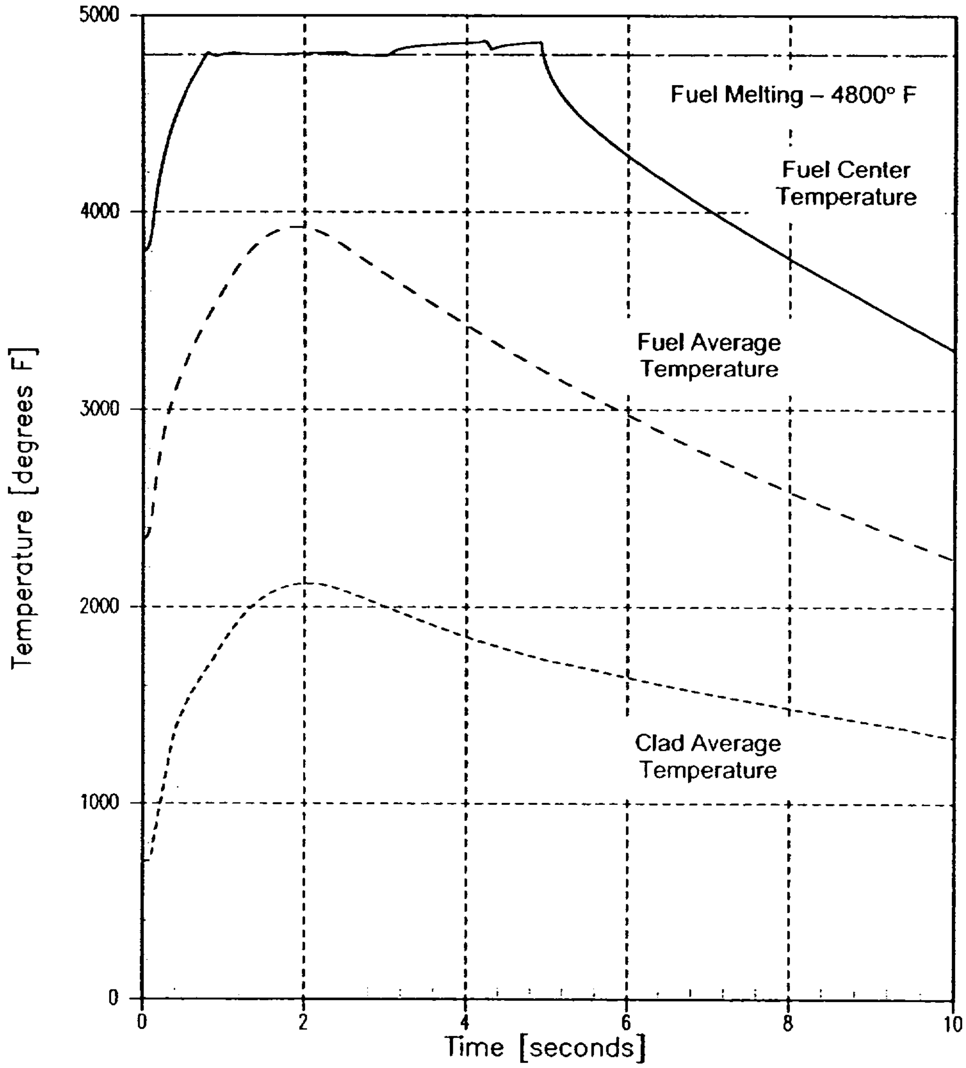

Figure 14.2.6–6

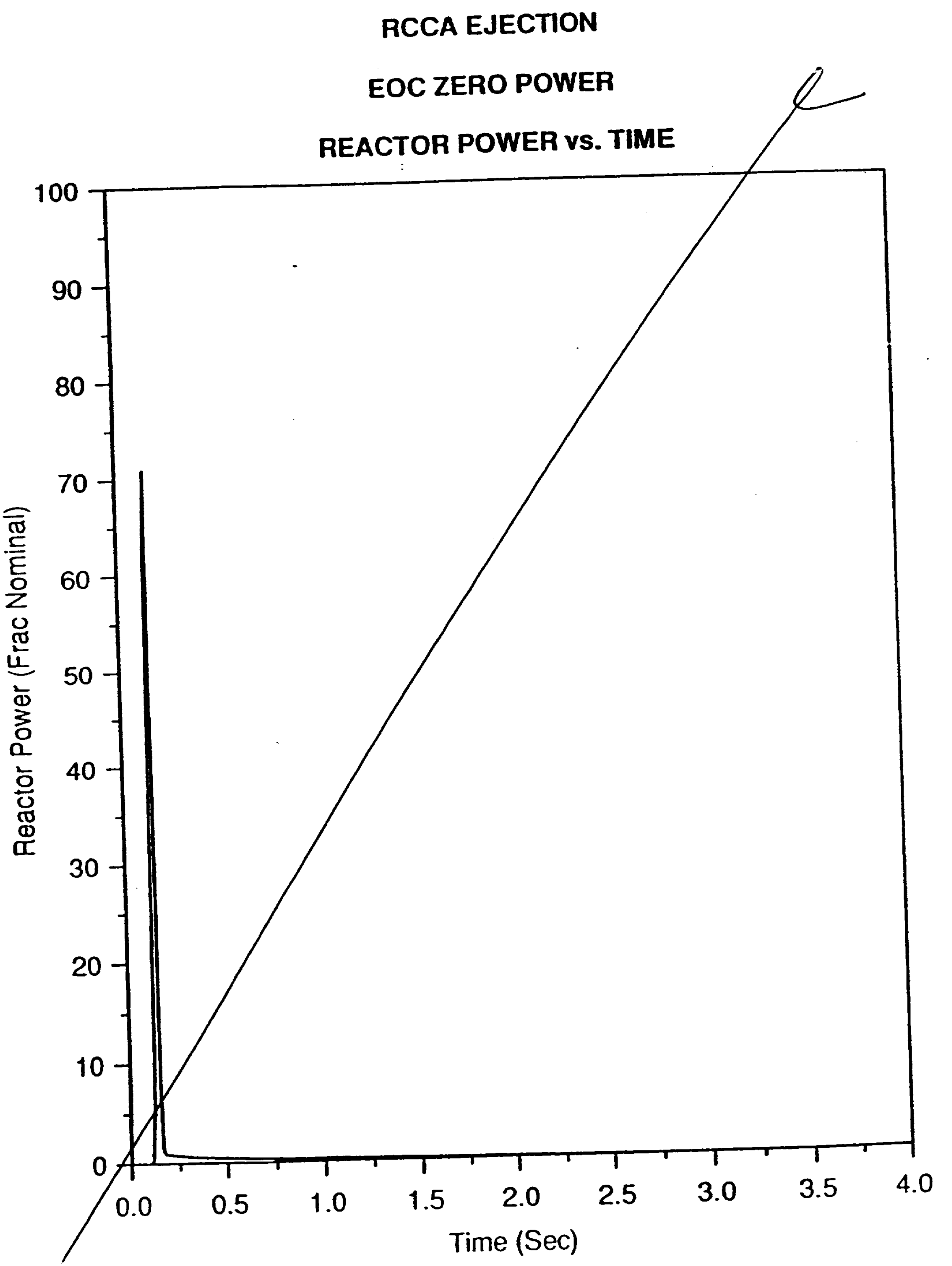

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FIGURE 14.2.6-10

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RCCA Ejection – EOC Zero Power
Reactor Power vs. Time

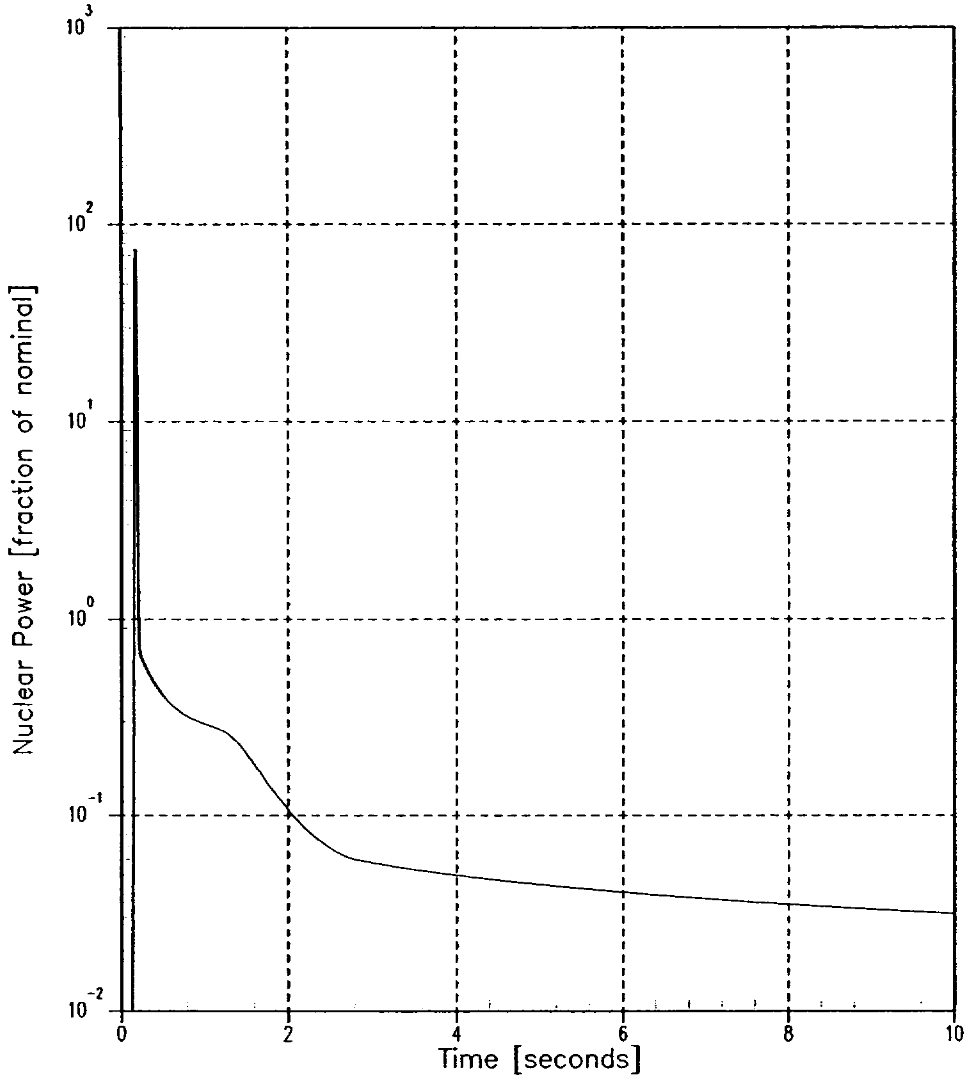

Figure 14.2.6–7

RCCA EJECTION

EOC ZERO POWER

REACTOR INTEGRATED POWER vs. TIME

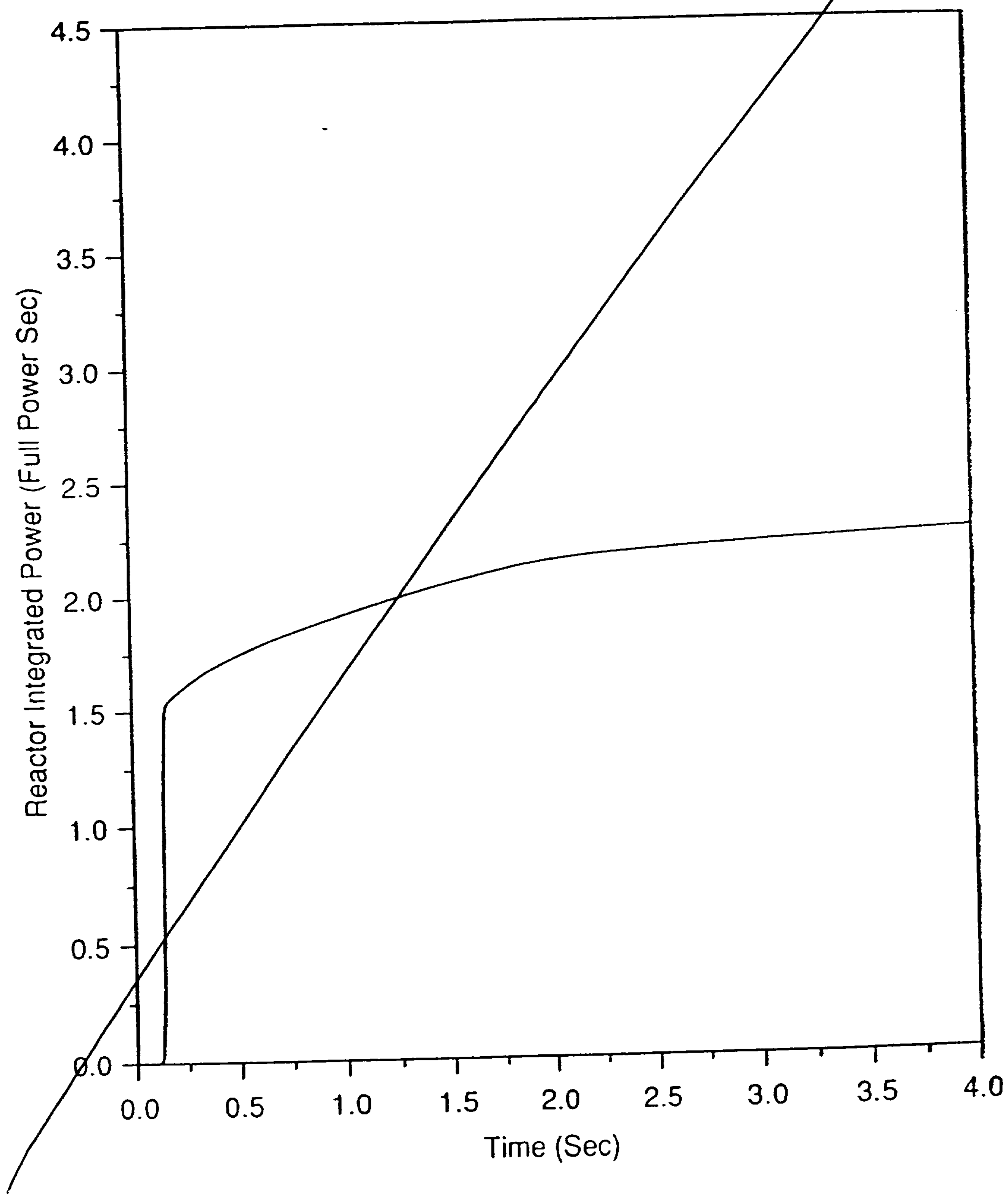

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FIGURE 14.2.6-11

RCCA EJECTION

EOC ZERO POWER

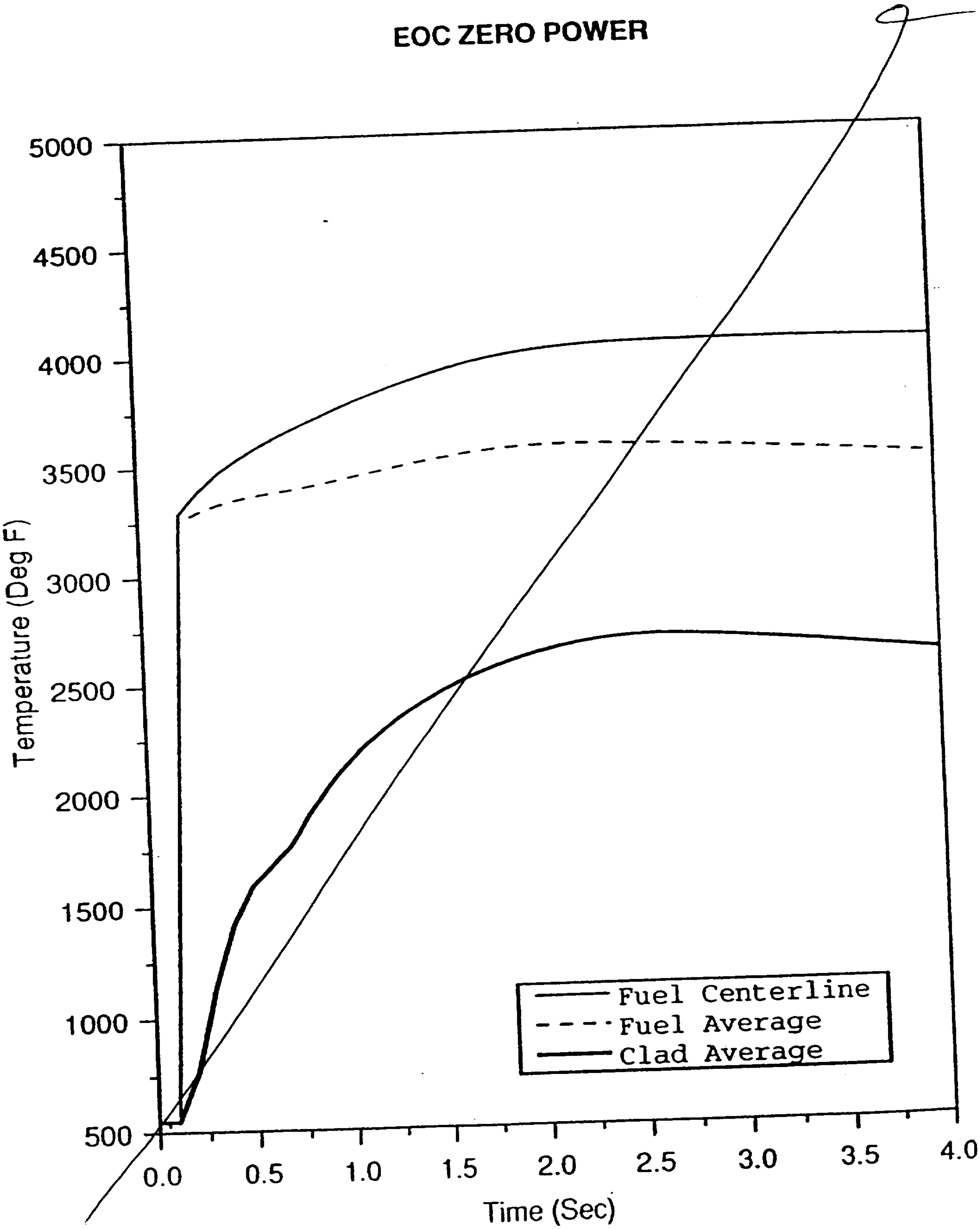

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FIGURE 14.2.6-12

RCCA Ejection – EOC Zero Power
Fuel and Clad Temperatures vs. Time

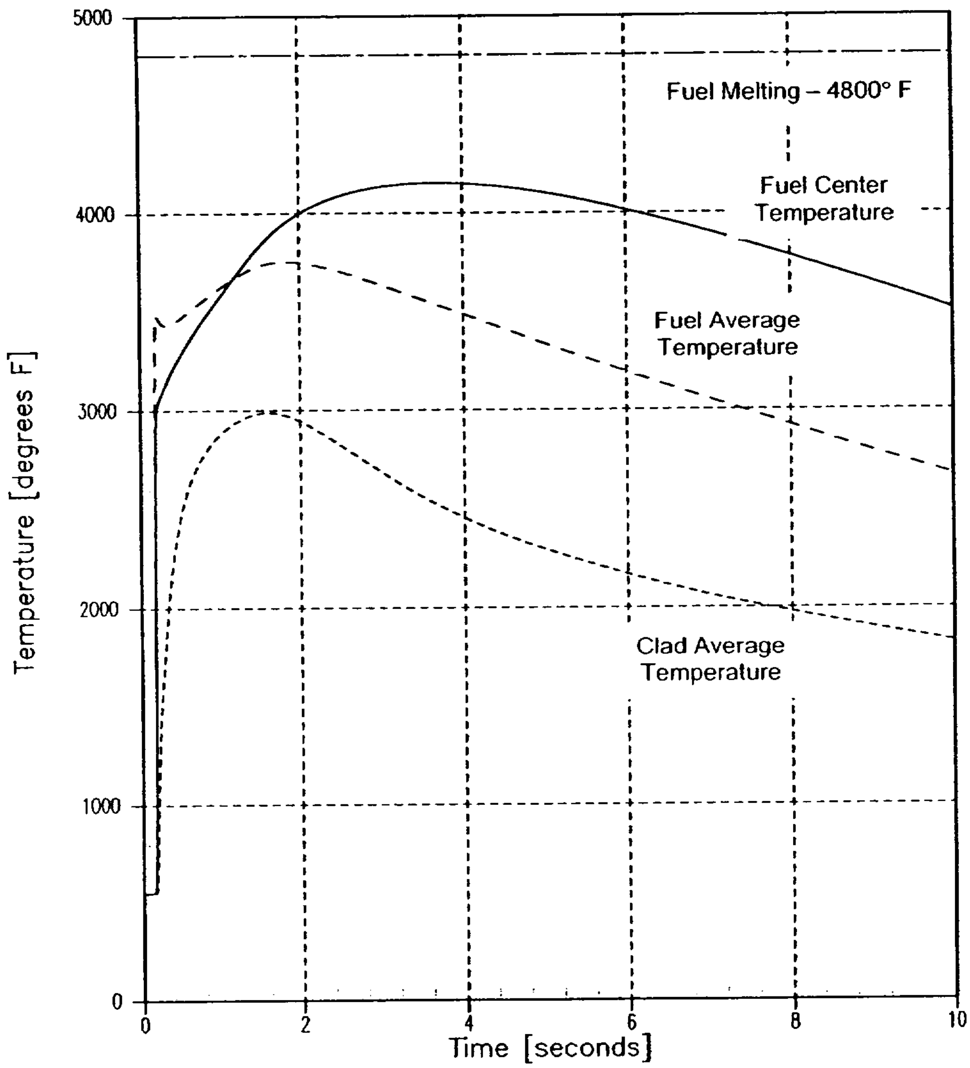

Figure 14.2.6–8