ATTACHMENT I

SAFETY EVALUATION REPORT FOR

H. B. ROBINSON UNIT 2, CYCLE 10

RELOAD APPLICATION

CHAPTER 15 EVENTS

15.0 Introduction And Analytical Techniques

The Carolina Power and Light Company (CP&L) submitted XN-NF-84-74, "Plant Transient Analysis For H. B. Robinson Unit 2 At 2300 MWt With Increased $F^N_{\Delta H}$," in support of its Cycle 10 reload application for H. B. Robinson. XN-NF-84-74 presents the analyses of the Chapter 15 transient and accident events. These analyses were performed by Exxon Nuclear Company, the fuel vendor for the H. B. Robinson plant.

The application for the Cycle 10 reload incorporated plant design changes resulting from steam generator replacements and justification for return to full power operation.

The analytical methodology and the computer models used in the safety analyses have not been approved. The Safety Evaluation Reports (SER) for Cycle 8 and Cycle 9 required the licensee (if it continued to rely on Exxon analyses) to develop a stand-alone analysis methodology which does not infringe upon other vendors' methods. As a consequence, Exxon Nuclear Company (ENC) developed a stand-alone methodology which is at present under staff review.

The computer programs used in the analyses are PTSPWR2, SLOTRAX and RELAP5. The RELAP5 computer program was submitted in response to NRC's small-break LOCA analysis concerns outlined in TMI Action Plan Item

II.K.3.30 (NUREG-0737). The use of this code for mild transient calculations, as applied in XN-NF-84-74, should be acceptable. This code has been developed and applied to transient analyses by the Office of Nuclear Regulatory Research, at the NRC. Generic approval for this code will result from the staff review of TMI Action Item II.K.3.30.

The staff's review of the PTSPWR2 computer program is nearing completion. This code has been significantly modified since its application to Cycle 8 and Cycle 9 reloads. The code has been benchmarked with several LOFT experimental transients, with a RELAP5 analysis, and with an operating plant transient. Our review has progressed sufficiently to conclude that the analyzed events submitted in XN-NF-84-74 will not be significantly altered upon completion of review.

The analytical methods (by which the licensee applies a computer program for a specific event) is documented in XN-NF-84-73(P), "Exxon Nuclear Methodology For Pressurized Water Reactors Analysis Of Chapter 15 Events." This methodology report is still being developed by Exxon and undergoing staff review. Our review of both XN-NF-84-73(P) and XN-NF-84-74 concludes that the calculated results for H. B. Robinson Unit 2 would not be appreciably altered upon our completion of the methodology review. This conclusion is based upon the code validation results and the limiting boundary conditions applied to each event.

ENC has not finalized its methodology for evaluating the consequences of postulated steam line break events. However, by incorporating an integral flow restrictor within the nozzles of the steam generators, the consequences of a postulated steam line break event is significantly reduced. In addition, the limiting operating conditions for a postulated steam line break is at end of cycle (EOC). At this time in operating cycle, the moderator density or temperature coefficient is at its most negative value. This maximizes the potential for return to power from an over-cooling event.

In order to confirm that no fuel failure is anticipated to occur, the staff performed its analysis of a steam line break event for H. B. Robinson. Results of the staff's analysis is documented in Appendix A to this report. CP&L has committed to provide reanalyses of the steam line break events for H. B. Robinson. We require this submittal, including documentation of the methodology, by January 31, 1985. It is our understanding that the analyses will be performed with RELAP5. We require a copy of the RELAP5 input deck for our review.

The loss of feedwater event was analyzed with the SLOTRAX computer code. SLOTRAX is under staff review. Our review indicates that SLOTRAX under-predicts the pressurization of the primary system for the loss of feed-water event. However, the insurge of primary coolant into the pres-surizer is conservatively calculated by the homogeneous equilibrium model in SLOTRAX. The licensee, applying the conservative pressurizer inflow, performed a hand calculation of the peak pressure by assuming isentropic compression of the steam. This analysis is conservative. We require the licensee to provide code validation of SLOTRAX by November 30, 1984. This has not been submitted to the staff as part of the SLOTRAX documentation.

The following sections address the specific events analyzed in XN-NF-84-74.

15.1 Increase In Heat Removal By The Secondary System

15.1.1 Feedwater Malfunctions That Result in a Decrease in Feedwater Temperature

The licensee concluded in technical report XN-NF-83-72 that the excess load event, documented in Section 15.1.3 of XN-NF-83-74, bounds the consequences of the decrease in feedwater temperature event. We find the licensee's assessment acceptable.

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

The licensee concluded in technical report XN-NF-83-72 that the excess load event, documented in Section 15.1.3 of XN-NF-84-74, bounds the overcooling response of the decrease in feedwater temperature event. In addition, the rod withdrawal event, documented in XN-NF-84-74, bounds the reactivity insertion response of the decrease in feedwater temperature event. We find the licensee's assessment acceptable.

15.1.3 Increase In Steam Flow (Excess Load)

Section 15.1.3 of XN-NF-84-74 evaluates the Excess Load Event for H. B. Robinson 2. The maximum step increase in load demand was 10% from full power operation. This was stated to be the maximum capacity of the turbine steam regulating valves from the most degraded DNBR condition.

The Excess Load Event is classified as a Condition II event, an Anticipated Operational Occurrence. The acceptance criteria for this event is that the primary system pressurization remains below 110% of design values; that the DNBR not decrease below 1.17 when applying the XNB correlation; that the radiological consequences be less than 10 CFR 20 guidelines; and that the event should not generate a more serious plant condition without other faults occurring independently.

In assessing this event, the licensee performed two analyses. One analysis minimized the moderator temperature feedback and the second analysis maximized the contribution of the moderator feedback. The conclusions of these analyses showed a negligible difference between the resulting minimum DNBR for the two cases. The analysis with minimum reactivity feedback resulted in a minimum DNBR of 1.331. The analysis with the maximum reactivity feedback resulted in a minimum DNBR of 1.332. Since the calculated minimum DNBR did not decrease below 1.17, no fuel failure was predicted to occur.

The similarity of the minimum DNBR for both events is attributed to the similarities of the thermal-hydraulics during the initial 45 seconds. During this time interval the minimum primary system pressure decreased to 2205 psia, and the core power and core inlet temperature (decreasing by 4°F) behaved similarly for both analyses. Differences in plant responses occurred following the time of minimum DNBR. For the maximized feedback event, the DNBR remained relatively constant near the minimum value as the primary system pressure increased and leveled off at a slightly higher value. The minimum feedback event, however, continued to increase in pressure and in DNBR. This is attributed to the less negative (zero) moderator temperature coefficient. The primary system pressure achieved a peak of 2390 psia. This is well within 110% of the primary system design pressure.

15.1.3.1 Conclusion For The Excess Load Event

The licensee demonstrated conformance to the acceptance criteria for the Excess Load Event, as it applies to H. B. Robinson Unit 2. The methodology used in analyzing the Excess Load Event is acceptable. The applicant used the PTSPWR2/Mod 1 (1984 version) computer program to calculate the thermal-hydraulic systems and core heat flux responses. This code is undergoing staff review and an SER is anticipated by end of calendar year 1984. We have reasonable assurances that upon completion of our review of PTSPWR2/Mod 1, any modification or restrictions placed upon the code would have negligible impact on this analyzed event. We therefore find the analysis of the Excess Load Event acceptable.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Power Operated Relief Valve

The licensee concluded in technical report XN-NF-83-72 that the excess load event, documented in Section 15.1.3 of XN-NF-84-74, bounds the consequences of an inadvertent opening of a steam generator relief valve. The excess load event results in symmetric cooldown of all 3 steam generators. The open atmospheric relief valve results in asymmetric cooldown of the primary system.

Exxon Nuclear Company, the fuel vendor for H. B. Robinson, is developing a new methodology for evaluating steam line break and stuck-open atmospheric relief valve events. This methodology will account for asymmetric thermal-hydraulics within the reactor vessel. This methodology and analysis will be submitted by January 31, 1984 and will be used to confirm that the excess load event is bounding. We find the licensee's response acceptable.

15.1.5 Steam System Piping Failures

The analysis of a postulated steam line break or an inadvertent opening of a steam generator relief or safety valve requires the modeling of thermal-hydraulic asymmetry within the reactor vessel. Previous H. B. Robinson analyses for these events were performed by Westinghouse and by Exxon Nuclear Company.

The analyses performed by Exxon Nuclear Company were determined unacceptable for previous Cycles. The reason was primarily due to insufficient justification for neglecting asymmetry in the thermal-hydraulics within the reactor vessel. Exxon Nuclear is developing its analytical methodology for steam line break analysis. This methodology will use the RELAP5 computer program and model the asymmetric thermal-hydraulics for these events.

For Cycle 10, CP&L replaced the steam generators at H. B. Robinson. These generators have integral flow restrictors designed within their outlet nozzles. The restrictors decrease the minimum cross sectional flow area from 4.7 ft^2 to 1.4 ft^2. These flow restrictors significantly reduce the consequences of a postulated steam line break event. In addition, the limiting operating conditions for a major rupture of a steam line is at end of cycle (EOC). At this time, the moderator density or temperature coefficient is at its most negative value. This provides the greatest potential for return to power.

To confirm that no fuel failure would occur, the staff performed its analysis of a steam line break event for H. B. Robinson. Results of the staff's analysis is documented in Appendix A to this report.

15.1.5.1 Conclusion For The Steam Line Break Events

We have reviewed the licensee's justification for delaying submittal of the steam line break events and find them acceptable. The staff's analysis of the steam line break event for H. B. Robinson Unit 2 showed ample margin to the specified acceptable fuel design limits (SAFDL). Consequently fuel integrity should be maintained.

CP&L has committed to provide reanalyses of the steam line break events for H. B. Robinson. We require this submittal, including documentation of the methodology, by January 31, 1985. It is our understanding that the analyses will be performed with RELAP5. We require a copy of the RELAP5 input deck for our review.

15.2 Decrease In Heat Removal By The Secondary System

15.2.1 Steam Pressure Regulator Malfunction That Result in Decreasing Steam Flow

This event is not applicable to H. B. Robinson Unit 2 since it has no steam line pressure regulators.

15.2.2 Loss of External Electrical Load

Section 15.2.2 of XN-NF-84-74 evaluates the Loss of External Electrical Load event for H. B. Robinson 2. This analysis assumes an instantaneous loss of generator load. Offsite power is not affected for this event and is therefore available for reactor coolant pump operation.

The loss of load event was analyzed twice. In one case, the event was initiated at the limiting conditions for assessing peak primary system pressurization. The second case was initiated at limiting conditions for minimum DNBR considerations. The loss of load event is classified as a Condition II event, an Anticipated Operational Occurrence. The acceptance criteria for this event is that the primary system pressurization remains below 110% of design values; that the DNBR not decrease below 1.17 when applying the XNB correlation; that the radiological consequences be less than 10 CFR 20 guidelines; and that the event should not generate a more serious plant condition without other faults occurring independently.

The analysis of this event was initiated by an instantaneous loss of generator load. The turbine stop valves closed as the turbine tripped. A reactor trip was not credited from the turbine trip. The isolation of the secondary system led to its pressurization. The secondary dump valves were assumed not to function.

The analysis which challenged the primary system overpressurization resulted in a peak pressure of 2661 psi. This pressurization is well below 110% of the primary system design pressure. The event was initiated at 102% of rated power. Conservative multipliers were assumed for the Moderator and Doppler reactivity coefficients. The initial pressurizer water level was biased high and the pressurizer pressure was biased low. The pressurizer spray and PORVs were assumed inoperative. These biases were predetermined based on sensitivity studies to be documented within XN-NF-84-73(P).

The analysis which maximized the challenge to the fuel design limit (minimum DNBR) was biased by increasing the core inlet temperature; decreasing the pressurizer pressure; and crediting operation of the pressurizer sprays and PORVs. This tended to minimize system pressurization. Consequently, the minimum DNBR analysis resulted in a peak primary system pressure of 2310 psi, or 351 psi lower than for the peak pressurization event. This analysis resulted in a minimum DNBR of 1.19, which is greater than the fuel design limit (for the XNB correlation) of 1.17. As a result, fuel integrity is maintained.

15.2.2.1 Conclusion for the Loss Of External Electrical Load Event

The licensee assessed the consequences of a loss of external electrical load event with respect to challenging the primary system pressure response and the fuel design limits. These were presented as two bounding analyses using the PTSPWR2/MOD1 (1984 version) computer code. The results of these analyses are found acceptable.

The PTSPWR2/MOD1 computer code and methodology (documented in XN-NF-84-73(P)) are under staff review. The sensitivity studies which determined the limiting operating conditions (biases) for this event have not been submitted in the XN-NF-84-73(P). We require the licensee to submit these results prior to December 31, 1984.

Our review of the PRSPWR2/MOD1 computer program is nearing completion. We anticipate issuing an SER by December 31, 1984. We have reasonable assurances that upon completion of our review of PTSPWR2/MOD1, any modification or restrictions placed upon the code would have negligible impact on these analyzed events. We therefore find the analysjs of the loss of external electrical load event acceptable.

15.2.3 Turbine Trip

The licensee concluded in technical report XN-NF-83-72 that the turbine trip event is not required to be analyzed since it is bounded by the loss of load event, Section 15.2.2 in XN-NF-84-74. We find the licensee's assessment acceptable.

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

The licensee concluded in technical report XN-NF-83-72 that the subject events are bounded by the loss of load event and need not be analyzed. We find the licensee's assessment acceptable.

15.2.5 Inadvertent Closure of Main Steam Isolation Valves (MSIVs)

The licensee concluded in technical report XN-NF-83-72 that the subject event is bounded by the loss of load event and need not be analyzed. We find the licensee's assessment acceptable.

15.2.6 Loss of Non-Emergency AC Power to the Station Auxiliaries

The licensee concluded in technical report XN-NF-83-72 that the subject event is bounded by the loss of load event and need not be analyzed. We find the licensee's assessment acceptable.

15.2.8 Feedwater System Pipe Break

The licensee concluded in technical report XN-NF-83-72 that the spectrum of steam line break events bounds the consequences of feedwater line break events. This was attributed to the high elevation of the feedwater nozzle. Consequently, mostly steam would be discharged out the break. This was the design basis of the plant and we find the licensee's assessment acceptable.

15.3 Decrease in Reactor Coolant Flow

15.3.1 Loss of Forced Reactor Coolant Flow

Section 15.3.1 of XN-NF-84-74 evaluates the Loss of Forced Reactor Coolant Flow for H. B. Robinson Unit 2. This event was simulated as a loss of electric power to all of the reactor coolant pumps. Offsite power was assumed available.

The loss of forced reactor coolant flow is classified as a Condition II event, an Anticipated Operational Occurrence. The acceptance criteria for this event is that the primary system pressurization remains below 110% of design values; that the DNBR not derrease below 1.17 when applying the XNB correlation; that the radiological consequences be less than 10 CFR 20 guidelines; and that the event should not generate a more serious plant condition without other faults occurring independently.

The licensee has concluded that there exists no active single failure which would result in a more severe overpressurization or lower DNBR for this event. The licensee addressed the concern of overpressurization and minimum DNBR with two calculations. The calculation for maximizing the system pressurization response assumed a high reactor system initial pressure, a high pressurizer level, disabled PORVs, minimum reactor coolant flywheel inertia, high moderator reactivity temperature coefficient, low Doppler reactivity coefficient, and maximum heat transfer coefficient across the fuel gap.

The calculation for minimizing the DNBR assumed low initial primary system pressure, low pressurizer level, PORVs availability, minimum flywheel inertia for the reactor coolant pumps, increased core inlet temperature, high moderator reactivity temperature coefficient, low Doppler reactivity coefficient, and high gap conductance within the reactor fuel rods.

The above biases on operating conditions were determined as part of the methodology development, to be documented in XN-NF-84-73(P). These studies have not been transmitted to the NRC for review. We require the licensee to submit these studies by December 31, 1984.

The analyses were initiated with a pump coastdown from the above operating conditions. The DNBR rapidly decreased with decreasing coolant flow. The reactor coolant temperature then increased (8°F) and expanded into the steam region of he pressurizer. Upon a low coolant flow indication (87% flow from the loop flow detectors), the reactor tripped. The reactor was assumed on manual control to prevent rod insertion upon an increase in coolant temperature. The reactor power reached 105%. The peak primary system pressure, for the maximum pressurization calculation, was 2582 psi. This is well below 110% of design. For the minimum DNBR biased calculation, the peak primary system pressure was 2304 psi, or 278 psi lower. The minimum DNBR for this event decreased to 1.19.

15.3.1.1 Conclusions for the Loss of Forced Reactor Coolant Flow Event

The licensee assessed the consequences of a loss of reactor coolant flow event with respect to challenging the primary system pressure response and the fuel design limits. These were presented as two bounding analyses using the PTSPWR2/MOD1 computer code. The results of these analyses are found acceptable. The peak primary system pressurization was well below 110% of system design and the minimum DNBR was above 1.17 when applying the XNB critical heat flux correlation. As a consequence both primary system and fuel integrity are maintained.

Both the PTSPWR2/MOD1 computer code and methodology of implementation (documented in XN-NF-84-73(P)) are under staff review. The sensitivity studies which determined the limiting operating conditions (biases) for this event have not been submitted as part of XN-NF-84-73(P). We require the licensee to submit these results prior to December 31, 1984. Our review of the PTSPWR2/MOD1 computer code is nearing completion. We anticipate issuing an SER by December 31, 1984. Our review has progressed sufficiently such that we have reasonable assurances that upon completion of our review of PTSPWR2/MOD1, any modification or restrictions placed upon the code would have negligible impact on these analyzed events. We therefore find the analysis of the loss of reactor coolant flow event acceptable.

15.3.2 Flow Controller Malfunction

The H. B. Robinson Unit 2 plant has no primary coolant flow controllers. Therefore, this event is not applicable to H. B. Robinson Unit 2.

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

Section 15.3.3 of XN-NF-84-74 evaluates the consequences of a locked rotor event for H. B. Robinson Unit 2. The event was initiated by an instantaneous seizure of a rotor from one of the primary system reactor coolant pumps.

The locked rotor event is classified as a Condition IV event, a Postulated Accident. The acceptance criteria for the locked rotor event is that the radiological consequences be less than 10 CFR 100 guidelines; the event should not cause a consequential loss of the required functions of the systems needed to cope with the reactor and containment systems; the radially averaged fuel enthalpy be less than 280 cal/gm; all fuel rods which experience a minimum DNBR below the specified acceptable fuel design limit (SAFDL, 1.17 for the XNB critical heat flux correlation) are assumed to fail; and the primary system pressure should not exceed 110% of design.

Two analyses were presented for this event. One analysis maximized the system pressurization and the other minimized the DNBR. Both calculations were initiated by an instantaneous seizure of a rotor from one of the primary system reactor coolant pumps. A reactor trip was initiated by a low flow signal from the affected loop. As the flow decreased, the primary coolant temperature began to rise. With increasing coolant temperature the primary system liquid expanded into the pressurizer, which led to primary system pressurization. Reverse flow in the affected loop occurred one second into the event. This was attributed to continued operation of the two remaining pumps.

The locked rotor calculations resulted in a core flow reduction to 60% of nominal. This occurred 4.0 seconds into the event.

The analysis, which biased the reactor operating conditions to minimize the DNBR, was initialized with a high core inlet temperature; low pressurizer level; low pressurizer pressure; high moderator reactivity temperature coefficient; low Doppler reactivity coefficient; and a high gap conductance to maximize the heat flux at the fuel pin surface. In addition, the PORVs were assumed operational to minimize system pressurization.

The analysis which biased the operating conditions to maximize primary system pressurization was initialized with a high core inlet coolant temperature; a high pressurizer level; high pressurizer pressure; high moderator reactivity temperature coefficient; and a high fuel gap conductance. For this analysis, both the pressurizer and secondary system PORVs were assumed disabled. The system, for this analysis, pressurized to 2524 psia. This is well below 110% of design.

15.3.3.1 Conclusions for the Reactor Coolant Pump Shaft Seizure (Locked Rotor) Event

The licensee assessed the consequences of a seized or locked rotor event for H. B. Robinson Unit 2. Two analyses were performed. One challenged the primary system pressurization response and the other challenged the fuel design limits. Both analyses used the PTSPWR2/MOD1 (1984 version) computer code. The results of these analyses were found acceptable.

With the above biases in operating conditions, a reactor trip signal on low coolant flow was generated 1.25 seconds into the event. As a result of the positive moderator coefficient, reactor power increased to 107.6% of rated. The minimum DNBR of 0.9 occurred shortly after reactor trip (2.17 seconds into the event). All fuel pins which experienced a DNBR below 1.17 were assumed to fail. The licensee calculated the radiological consequences to be less than 10% of 10 CFR 100 limits.

The PTSPWR2/MOD1 computer code is a one-dimensional representation of a nuclear steam supply system. Since the primary system is in a non-compressible state, a potential exists for asymmetric flow distribution across the core. A request was made to the Office of Nuclear Regulatory Research (RES) at NRC to assess the multi-dimensional fluid characteristics of a locked rotor event. In response, RES conducted a generic evaluation of a locked rotor event using the TRAC/PF1 computer program. Results of this evaluation showed negligible asymmetry of the coolant flow distribution across the reactor core.

As a consequence of the one-dimensional hydraulic characteristics of the locked rotor event, the PTSPWR2/MOD1 computer code should be appropriate for such application. The PTSPWR2/MOD1 computer code and methodology of implementation (documented in XN-NF-84-73(P)) are under staff review. The sensitivity studies which determined the limiting operating conditions (biases) for this event have not been submitted in XN-NF-84-73(P). We require the licensee to submit these results prior to December 31, 1984.

Our review of the PTSPWR2/MOD1 computer code is nearing completion. We anticipate issuing an SER by December 31, 1984. Our review has progressed sufficiently to acquire reasonable assurances that upon completion of our review, any modification or restrictions placed upon the code would have negligible impact on these calculations. We therefore find the analysis of the locked rotor event acceptable.

15.3.4 Reactor Coolant Pump Broken Shaft

The licensee concluded in technical report XN-NF-83-72 that the locked rotor event bounds the consequences of the broken shaft event and need not be analyzed. We find the licensee's assessment acceptable.

15.3.4 Reactor Coolant Pump Broken Shaft

The licensee concluded in technical report XN-NF-83-72 that the locked rotor event bounds the consequences of the broken shaft event and need not be analyzed. We find the licensee's assessment acceptable.

15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

Section 15.4.6 of XN-NF-84-74 evaluates boron dilution events for H. B. Robinson Unit 2. The events analyzed were for the following reactor modes of operation: (1) Refueling, (2) Cold shutdown with 3% delta rho shutdown margin and vessel filled to the centerline elevation of the hot legs (required for RHR mixing), (3) Cold shutdown with 1% delta rho shutdown margin and the primary system (excluding the pressurizer) filled with coolant, (4) Hot shutdown, (5) Startup and (6) Power operation.

The rate of dilution of primary system coolant is limited by the capacity of the charging pumps. This corresponds to an addition of 230 gpm of unborated water. For the cold shutdown mode of operation with emptied steam generators, the maximum dilution rate is limited to the capacity of one charging pump, or 77 gpm.

The time for operator action was determined by solving the differential equation for fluid dilution. The critical boron concentration and boron worth were determined with the XTGPWR computer code.

The boron dilution event is classified as a Condition II event, an Anticipated Operational Occurrence. The acceptance criteria for this event is that the primary system pressurization remains below 110% of design values; that the DNBR not decrease below 1.17 when applying the XNB correlation; that the radiological consequences be less than 10 CFR 20 guidelines; and that the event should not generate a more serious plant condition without other faults occurring independently. If operator action is required to terminate the transient, the following minimum time intervals must be available between the time when the alarm announces that dilution is occurring and the time ~~of~~ loss of shutdown margin:

a. During Refueling: 30 minutes.

b. During Startup, cold shutdown hot standby, and power operation: 15 minutes.

15.4.6.1 Conclusions for the Boron Dilution Events

The licensee assessed the minimum time available for operator action to mitigate the consequences of a boron dilution event. The licensee has determined that during refueling, the operators have in excess of 30 minutes to respond and mitigate the dilution process after receiving alarm indications. We find this acceptable.

During startup, cold shutdown, and hot standby operating conditions, the licensee calculated that the operator has in excess of 15 minutes to respond and mitigate the dilution event. We find this acceptable. The dilution event at power operation is bounded by the consequences of the rod withdrawal events. The consequences for these events showed that fuel integrity is maintained (MDNBR is greater than 1.17). We find this acceptable.

15.5 Increases In Reactor Coolant System Inventory

15.5.1 Inadvertent Operation Of Emergency Core Cooling System

The licensee concluded in technical report XN-NF-83-72 that the subject event need not be analyzed for Cycle 10 reload. The licensee argued that the shutoff head of the high head safety injection pumps is 1500 psia, which is well below the trip actuation setpoint of 1850 psia. With regards to the pressurized thermal shock issue, the licensee has an ongoing program, which includes installing part length shielding fuel assemblies to meet the screening criteria for RT_{NDT}. We find the licensee's assessment acceptable.

15.5.2 CVCS Malfunction that Increases Reactor Coolant Inventory

The licensee concluded in technical report XN-NF-83-72 that the subject event need not be analyzed since it is bounded by other events and previously addressed in the updated H. B. Robinson Unit 2 FSAR. We find the licensees assessment acceptable.

15.6 Decrease In Reactor Coolant System Inventory

15.6.1 Inadvertent Opening Of A Pressurizer Safety Or Power Operated Relief Valve

In technical report XN-NF-83-72, the licensee referenced the FSAR design basis analysis of an inadvertent opening of a pressurizer safety valve. The H. B. Robinson Unit 2 licensing basis acceptance criteria for this event is as for postulated accidents. However, the licensee performed an analysis which demonstrated that DNBR would not decrease below the specified acceptable fuel design limit (SAFDL). The calculated minimum DNBR was 1.33, well above the 1.17 SAFDL for the XNB critical heat flux correlation.

We find the licensee's assessment acceptable.

15.6.2 Steam Generator Tube Rupture

Section 15.6.3 of XN-NF-84-74 evaluates the Steam Generator Tube Rupture event for H. B. Robinson Unit 2. This event is initiated with an instantaneous rupture of a steam generator tube, relieving primary system coolant to the shell of the steam generator.

The steam generator tube rupture event is categorized as a Condition IV event, a Postulated accident. The acceptance criteria for this event are as follows:

(1) For a postulated accident with an assumed pre-accident iodine spike in the reactor coolant and for the postulated accident with the highest worth control rod stuck out of the core, the calculated doses should not exceed the guideline values of 10 CFR 100, Section 11.

(2) For the postulated accident with equilibrium iodine concentration for continued full power operation in combination with an assumed accident initiated iodine spike, the calculated doses should not exceed 10% or 2.5 rem and 30 rem, respectively, for the whole-body and thyroid doses.

Challenge to the specified acceptable fuel design limits (SAFDL), or fuel integrity, for the steam generator tube rupture event is bounded by the analysis of the inadvertent opening of a pressurizer relief valve (Section 15.6.1). The analysis of the inadvertent opening of a pressurizer relief valve showed that the minimum DNBR did not decrease below the SAFDL. Consequently, fuel integrity is maintained.

The licensee applied the H. B. Robinson design basis methods for calculating radiological releases for Cycle 10. The only variation in the method was a reanalysis of the primary to secondary coolant break flow for the new steam generator (the steam generators for H. B. Robinson Unit 2 were replaced).

The analysis assumptions for this event assumed loss of offsite power which resulted in steam relief directly to the atmosphere through a stuck open PORV. Operator action at 30 minutes into the event was credited to isolate the affected steam generator.

The RELAP5/MOD1 computer program was used to calculate the primary to secondary flow characteristics and the flow out the atmospheric dump and POR valves. Several break locations were evaluated for limiting conditions. The limiting break location was determined to result adjacent to the hot leg with cold leg fluid temperature conditions. The RELAP5 model nodalization of the steam generator was acceptably detailed. The primary system was modeled as a stand-alone steam generator between the hot and cold legs. The reactor vessel was not modeled. To conservatively bound the possible break and atmospheric release rates, conservative primary system boundary conditions were employed. These included maintaining a constant primary system pressure of 2280 psia and temperature of 536.2°F. Sensitivity studies were performed with a boundary temperature of 614.6°F and combination of 614.6°F at the hot leg and 536.2 °F at the cold leg. The lower temperature case resulted in the maximum flow out the tube.

In addition to the primary to secondary heat transfer, the licensee incorporated an additional energy boundary condition to the secondary system equivalent to 1/3 of the core generated power, including the energy generated by the primary coolant pumps, plus the energy equivalent to 100°F cooling of the primary system. This assumption maximized the mass transferred through the PORVs out to the atmosphere.

To confirm the acceptability of the RELAP5 break flow model, the licensee benchmarked the calculated flow rate with the Moody and Henry/Fauske break flow models. The comparison validated the conservatism of the RELAP5 calculation.

We find the method for calculating break flow characteristics acceptable.

15.6.2.1 Summary for the Steam Generator Tube Rupture Event

The licensee performed a radiological assessment of a postulated steam generator tube rupture event. The licensing design basis assumptions were used in this assessment. This assumption credited operator action to isolate the faulted steam generator 30 minutes into the event.

The contaminated mass entering the atmosphere was conservatively calculated. Since the maximum allowable Tech Spec primary system activity has not been modified since the last FSAR update, the same activity was applied to the analysis for Cycle 10.

The consequential dosage for this event was calculated at 0.6 rem whole body and 3.4 rem thyroid. These are well within the 10 CFR 100 guidelines. We find the analysis of the steam generator tube rupture event acceptable.

APPENDIX - A

CONFIRMATORY STEAM LINE BREAK ANALYSIS IN SUPPORT OF THE H. B. ROBINSON UNIT 2, CYCLE-10 RELOAD APPLICATION

I. INTRODUCTION

The previous H. B. Robinson Unit 2 steam line break analysis was performed by Exxon Nuclear Company using the PTSPWR2 computer program. Exxon Nuclear Company is the fuel vendor for Carolina Power & Light Company (CP&L), the licensee of H. B. Robinson Unit 2.

The PTSPWR2 computer program is a one-dimensional analytical representation of a nuclear steam supply system (NSSS). The program assumes ideal thermal-hydraulic mixing of the coolant entering the reactor vessel from the affected and intact steam generators. In addition, the moderator and Doppler reactivity feedback are obtained from average core thermal-hydraulic conditions.

Proprietary experimental data obtained by the NSSS vendors have shown significant thermal-hydraulic asymmetry of the fluid states within the reactor vessel for expected steam line break conditions. Consequently, the staff requested (as part of the generic review of the PTSPWR2 computer program) Exxon Nuclear Company to refine its analytical methods to account for asymmetric influences and demonstrate acceptability of the PTSPWR2 results. Exxon Nuclear Company is revising its analytical methods to address the above concerns.

The licensee has committed to provide reanalyses of the steam line break event for Cycle 10 by January 31, 1985. This commitment is acceptable.

The bases for accepting a late submittal of the steam line break event are as follows:

(1) H. B. Robinson Unit 2 replaced its steam generators with a new model that incorporates an integral flow restrictor within the outlet nozzle. The flow restrictor significantly reduces the consequences of a major rupture of a steam line,

(2) The limiting consequences of a large steam line break occurs at end of cycle (EOC) when the moderator coefficient is at its most negative value, and

(3) Staff analysis of the steam line break event (guillotine break) showed ample margin to the acceptance criteria for H.B. Robinson Unit 2.

The following documents the staff's analysis of a postulated steam line break event for H. B. Robinson Unit 2.

II. MODEL DESCRIPTION

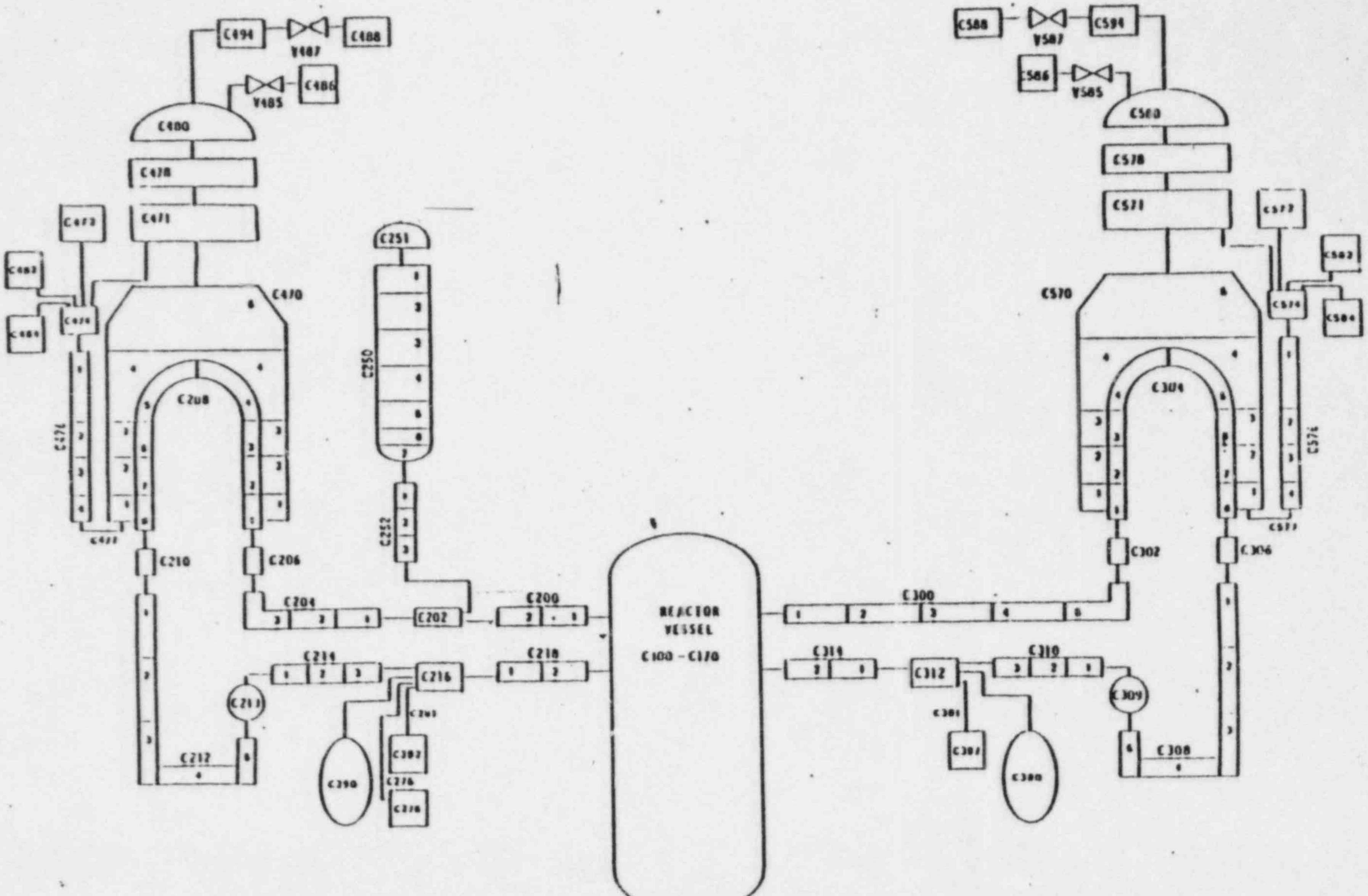
The computer code used for analyzing the steam line break (SLB) event was RELAP5/MOD1.5 Cycle 39. An input deck of a generic 3-loop Westinghouse plant was modified by data supplied by CP&L and ENC to model the H. B. Robinson plant.

II.1 NODALIZATION

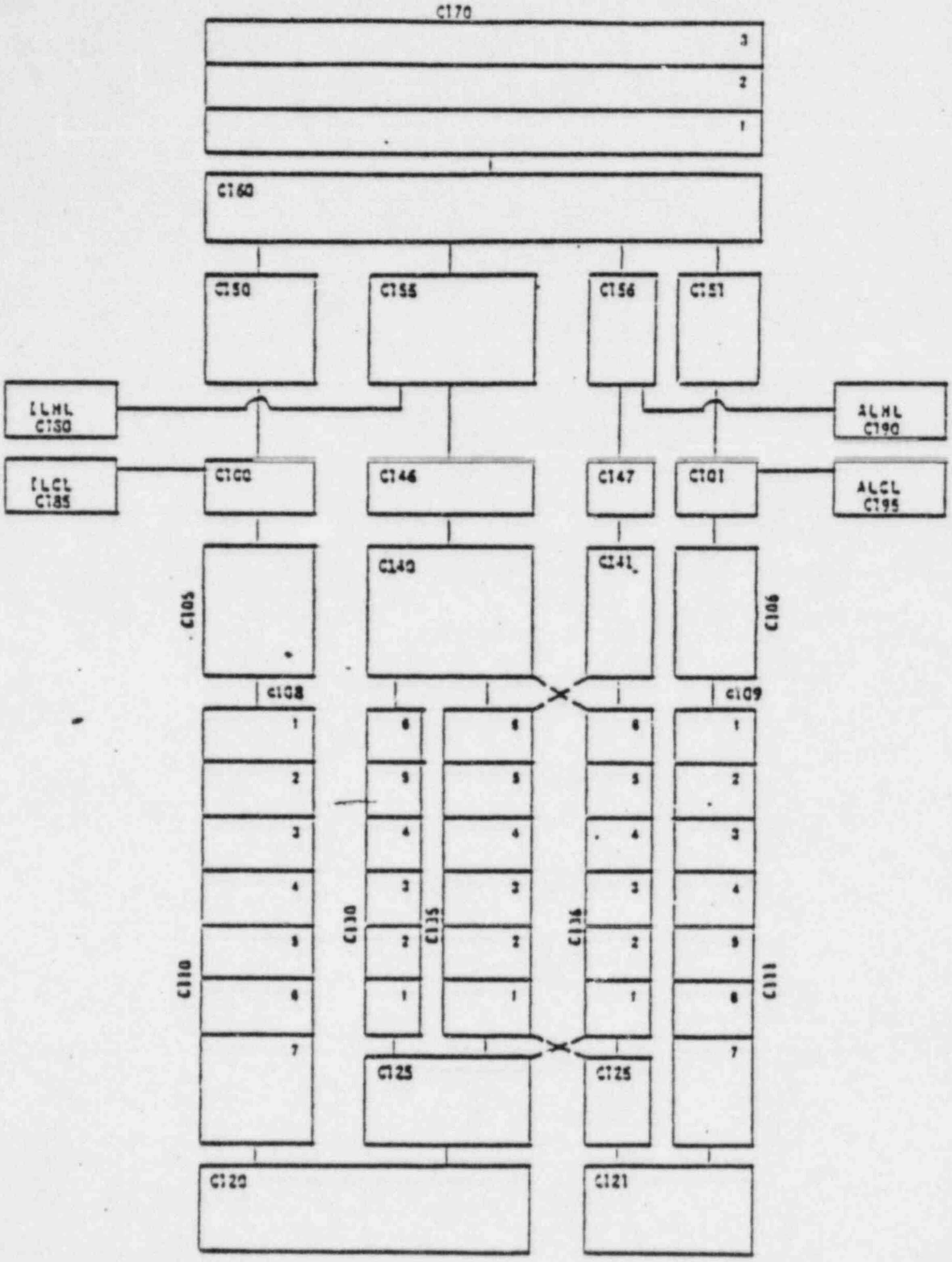
The model nodalization used for the H. B. Robinson SLB calculations is shown in Figs. A-1 and A-2. This nodalization represents the major components and flow paths of the H. B. Robinson 3-loop nuclear steam supply system.

The model consists of two loops. The intact loop is a lumped representation of two loops containing the unaffected steam generators. The pressurizer is connected to the unaffected loop. The affected loop contains the steam generator with the faulted steam line.

Except for the upper head region, the rector vessel was divided into two parallel channels proportioned 2:1. the model incorporated cross flow junctions in the upper and lower plena to simulate thermal-hydraulic coupling between the two core channels.



Nuclear Steam Supply System (NSSS) RELAP5/MOD1.5 Noding Diagram.
FIGURE A-1



Split Vessel RELAP5/MOD1.5 Noding Diagram.
FIGURE A-2

The amount of coupling was experimentally predetermined. Heat slabs representing the primary system metal masses in the vessel, pressurizer and steam generators as well as the metal in the primary coolant piping were included in the model.

II.2 MAJOR ASSUMPTIONS AND INITIAL CONDITIONS

The following major assumptions and initial conditions were used:

1. The system initial conditions prior to initiation of the SLB event are listed in Table A-1.
2. A uniform power profile was used. A power fraction of 0.1667 was assigned to each of the six axial core regions. In addition, these power fractions were weighted 2:1 between the intact core and the affected core regions.
3. Point kinetic reactivity feedback as a function of four parameters was calculated by a control system. The method is similar to that applied by Westinghouse, the reactor vendor for H. B. Robinson.

   (a) Moderator Density Reactivity Feedback

   The moderator density reactivity, as documented in Table A-2 was provided by Exxon Nuclear Company. Each of the six volumes within a core channel provided one-sixth of the total moderator reactivity feedback for that channel (uniform axial weighting). The overall moderator reactivity was given by weighting the affected and

TABLE A-1

STEAM LINE BREAK ANALYSIS INITIAL CONDITIONS

Parameter	Zero Power
Core Power	27.75 MW
Core mass flow	29,166 lb/s
Core T	0.64°F
Cold leg temperature	550.°F
Primary pressure	2251 psia
Secondary pressure	1004 psia
Secondary mass	135,000 lb/steam generator
Steam/Feed flow	---
Boron concentration	0 ppm

TABLE A-2

MODERATOR DENSITY REACTIVITY

Moderator Density (Lb/ft^3)	Reactivity ($)
43.93	-3.71
46.73	0.00
49.35	3.35
51.51	6.19
53.88	8.38
55.67	10.08
57.51	11.41

unaffected core channels in accordance with the Westinghouse methodology.

(b) Doppler Reactivity Feedback

The Doppler contribution to total reactivity was divided into two parts. One part represented the power coefficient at constant moderator temperature (Table A-3), while the other part accounted for the variation in the moderator temperature.

(c) Control Rod Insertion

The control rods, with the exception of the single most reactive rod, have a reactivity worth of -3.61 \$. This reactivity was assumed to be linearly inserted with 0.2 sec. delay at time of reactor trip.

(d) Boron Reactivity Feedback

A core average boron concentration calculated by the RELAP5 control system, was used for the reactivity feedback. It was assumed that the HPI system initiated 13 seconds after a generated SI signal. It was also assumed that borated water did not enter the primary coolant system until the HPI lines were purged of its initial inventory. The clearing of the lines was

TABLE A-3

DOPPLER POWER REACTIVITY

Core Power Density (% of Rated)	Reactivity ($)
0	0.00
5	0.65
10	-1.18
20	-1.96
30	-2.61
40	-3.61

assumed to take 30 seconds. This was based upon a line volume of 30 ft^3 and an injection rate of 1 ft^3/sec. The boron worth is given in Table A-4. The initial boron concentration in the boron injection tank was specified as 21000 ppm. It was conservatively assumed that this concentration decreased exponentially to 10 ppm over a period of 120 sec. This is conservative since the makeup water flowing into the tank is borated at approximately 2000 ppm.

4. The trips and setpoints used in the SLB calculation are listed in Table A-5.
5. The SI injection systems represents a single high pressure injection train. Injection temperature was set at 120°F.
6. All of the main feedwater was diverted to the affected steam generator during the initial 10 sec of the transient. The flow was assumed constant at 3861.1 lbm/sec. The temperature of the feedwater was assumed at 120°F.

TABLE A-4

BORON REACTIVITY COEFFICIENT

Moderator Density (Lb/ft^3)	Boron Coefficient ($/PPM)
43.93	0.020
46.73	0.022
57.51	0.028

TABLE A-5

STEAM LINE BREAK ANALYSIS TRIPS AND SETPOINTS

Trip	Setpoint
1. High steam flow	450 lb/s (40% nominal)
2. Low steam line pressure SI signal	615 psia
3. Low T_{avg}	543°F
4. Low primary pressure	1780 psia
5. Safety injection	(1 and (2 or 3) or 4 of the above trips

7. The reactor coolant pumps remained in operation at a constant speed throughout the transient.
8. A recirculation model was added to the steam generators so that a conservative (perfect) separation could be calculated. By calculating only steam flow out the break, the energy removed from the system is maximized.

III. BENCHMARK ANALYSIS

H. B. Robinson's original steam generators did not have an integral flow restrictor incorporated into their outlet nozzles. The original FSAR analysis, therefore, modeled the break area as 4.6 ft^2. To benchmark the H. B. Robinson RELAP5 model with the FSAR results, a calculation was performed which assumed a 4.6 ft^2 break area (The cross sectional area for the flow restrictor is 1.4 ft^2).

The break was simulated by an instantaneous opening of two flow paths, one connected to each side of the guillotine break (steam generator secondary). The primary break path (connected to the affected steam generator) was sized at 4.6 ft^2, to simulate the unrestricted rupture of a main steam line. The second break path was sized at 1.485 ft^2, to simulate the flow through the restrictor of the broken steam line. Flow from this valve was terminated 10 seconds after break initiation, the assumed closure time of the steam isolation valves (MSIVs).

The event was initiated at 100 seconds (after obtaining steady-state initial conditions). Results of the reactivity, power level, primary pressure, and primary coolant temperatures are shown in Fig. A-3 through A-6, respectively.

The results from the original design basis FSAR analysis are shown in Fig. A-7. Comparisons between the staff calculation and the FSAR design basis analysis are in good agreement.

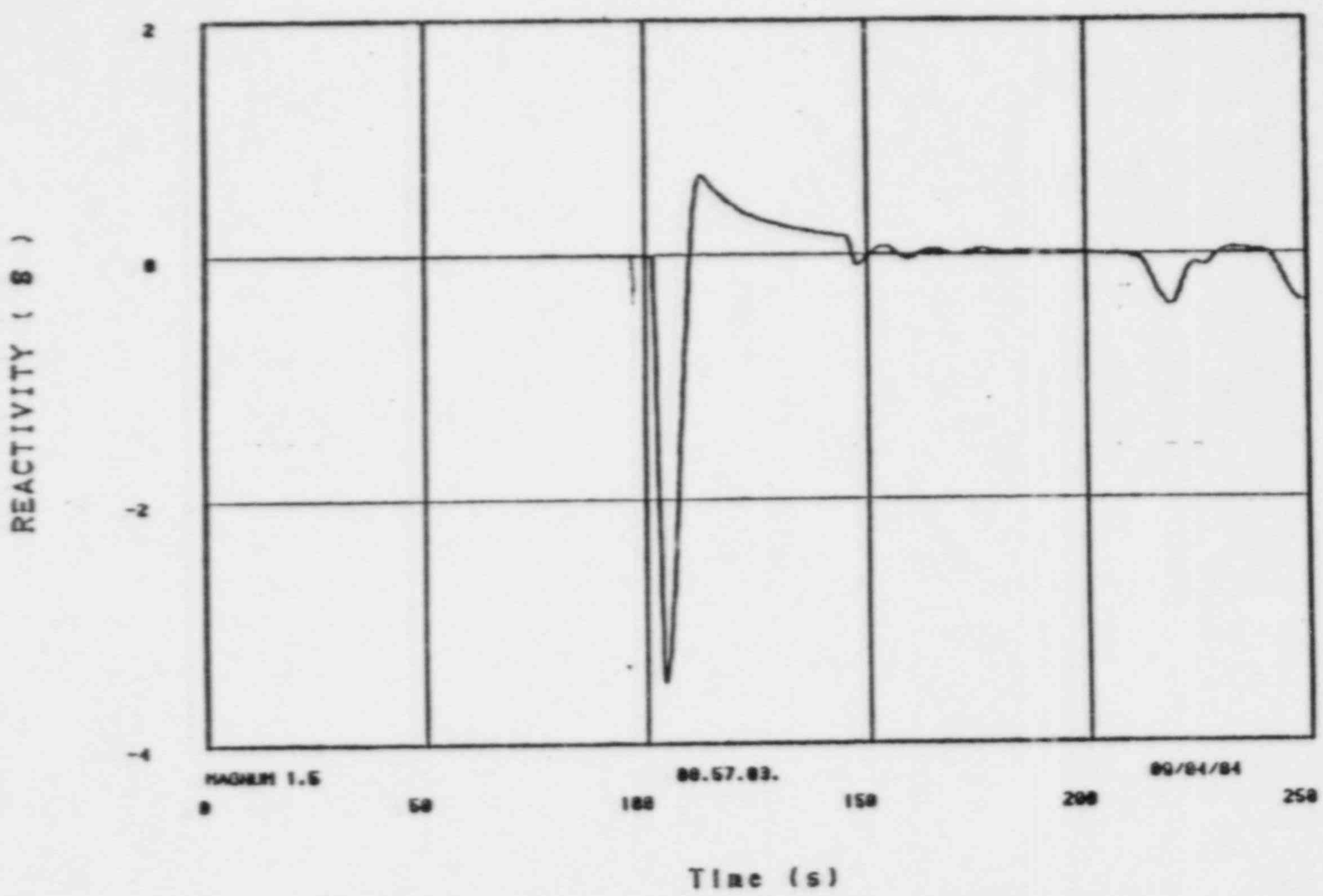
IV. DETERMINATION OF THE LIMITING BREAK

The following cases were analyzed to determine the limiting break location and conditions for H. B. Robinson Unit 2 Cycle 10:

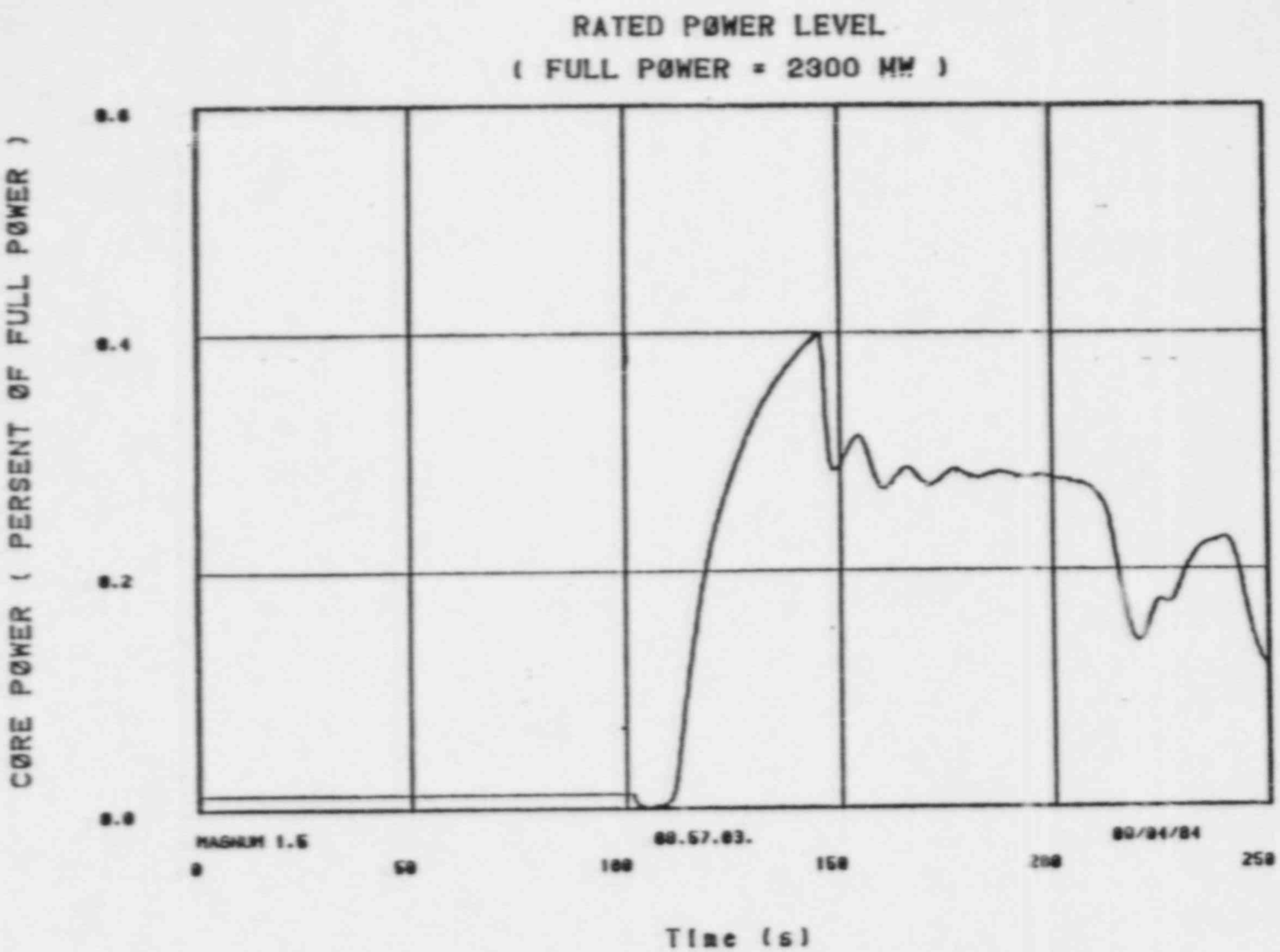
Case 1: Break between the flow restrictors with offsite power available. The blowdown areas are 1.388 ft^2 (affected) and 1.485 ft^2 (intact).

Case 2: Break downstream of both flow restrictors with offsite power available. The blowdown areas are 1.388 ft^2 (affected) and 2.776 ft^2 (intact).

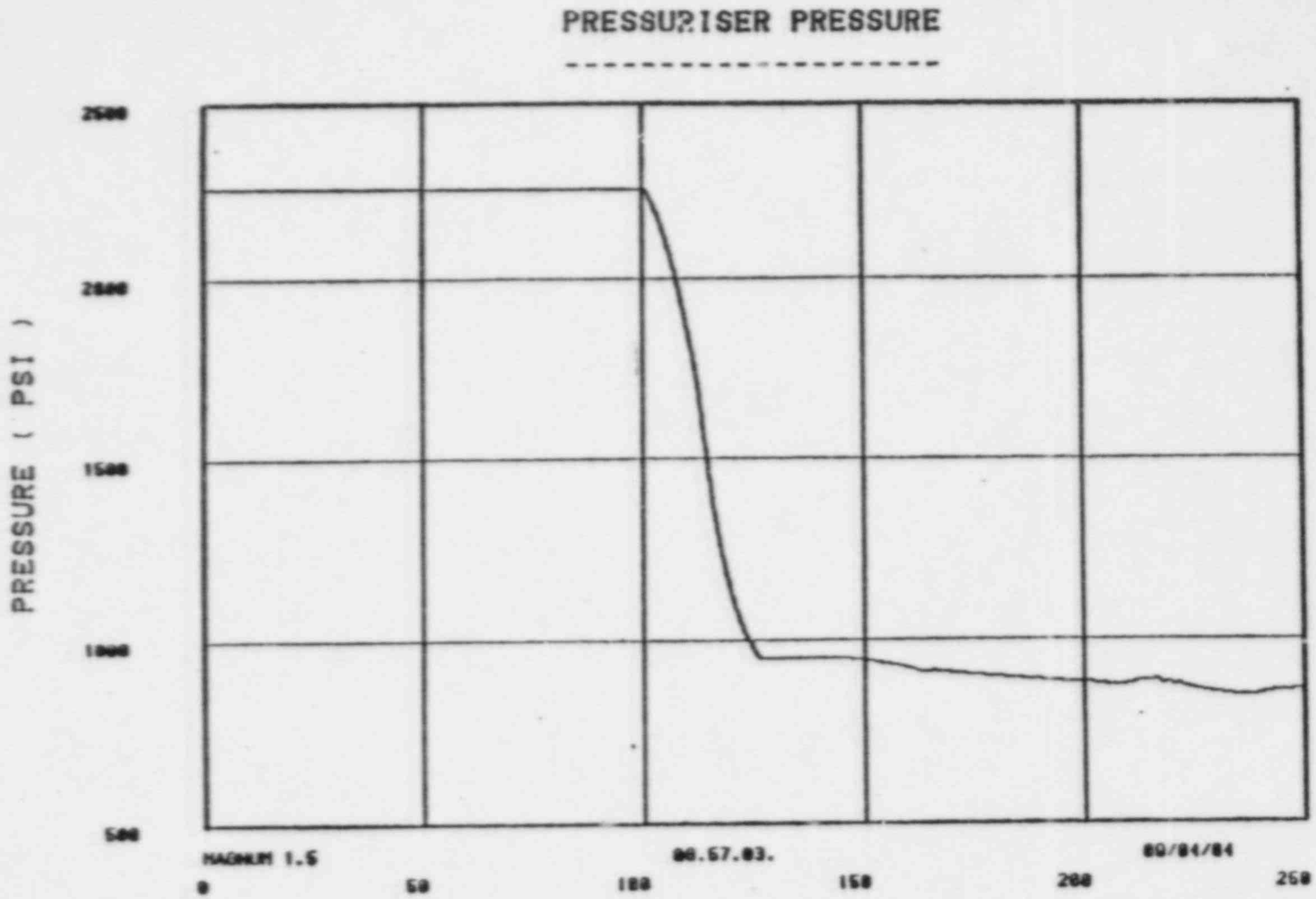
TØTAL REACTIVITY



H.B.RØBINSØN CYCLE 10 RELØAD
STEAM LINE BREAK ANALYSIS
BREAK UPSTREAM ØF FLØW RESTRICTØRS
FIGURE A-3



H.B.RØBINSØN CYCLE 10 RELØAD
STEAM LINE BREAK ANALYSIS
BREAK UPSTREAM ØF FLØW RESTRICTØRS
FIGURE A-4



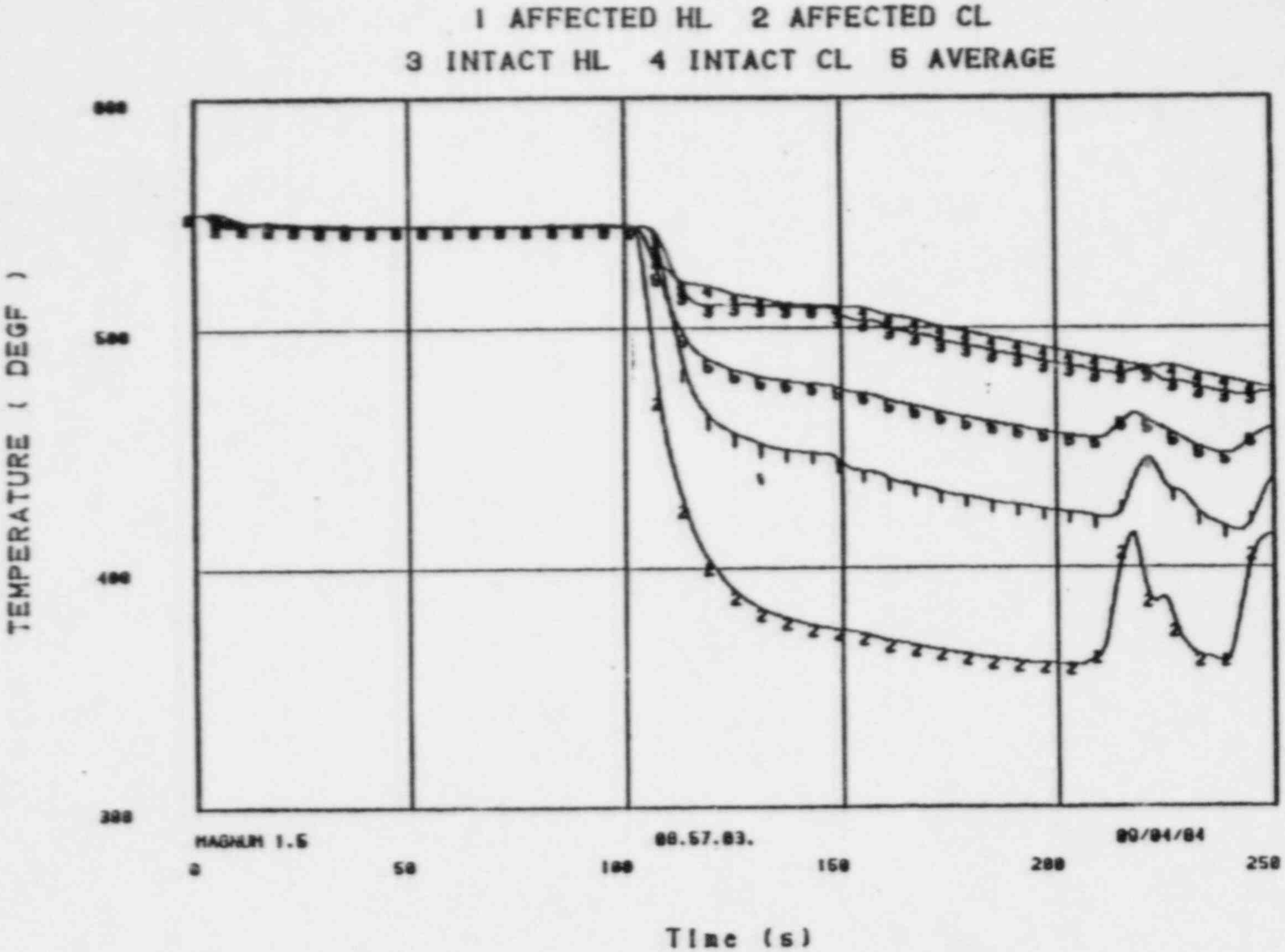
Time (s)

H.B.RØBINSØN CYCLE 10 RELØAD

STEAM LINE BREAK ANALYSIS

BREAK UPSTREAM ØF FLØW RESTRICTØRS

FIGURE A-5



H.B.RØBINSØN CYCLE 10 RELØAD
STEAM LINE BREAK ANALYSIS
BREAK UPSTREAM ØF FLØW RESTRICTØRS
FIGURE A-6

H.B. ROBINSON UNIT 2

ORIGINAL FSAR ANALYSIS

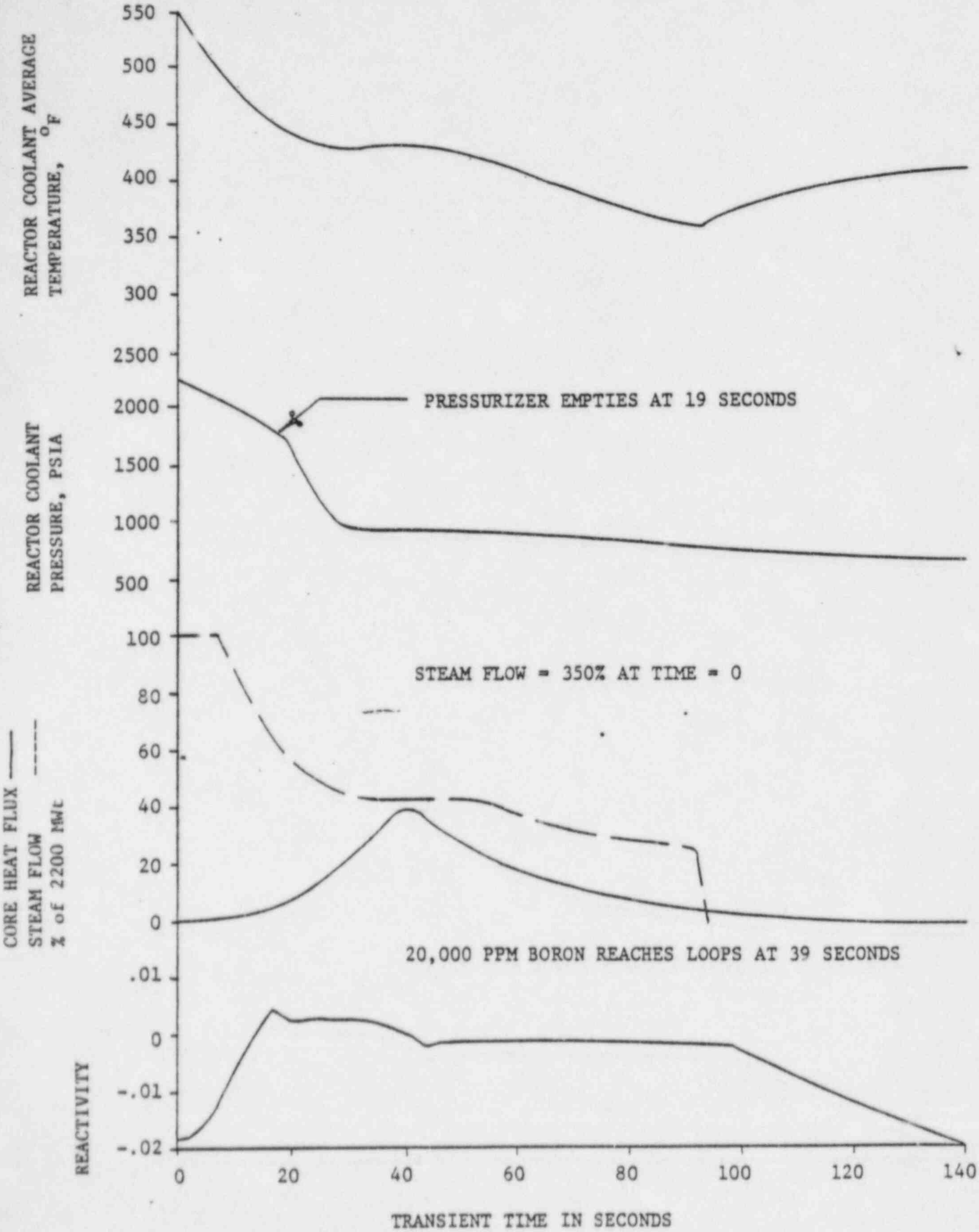


FIGURE A-7

Case 3: Break downstream of both flow restrictors with loss of offsite power at break initiation.

Case 4: Break downstream of both flow restrictors with loss of offiste power at time of reactor trip.

All cases assumed initial hot shutdown conditions. This maximized the liquid mass within the steam generator shell, minimized the core generated decay heat and thereby maximized the overcooling for the event. The peak return to power and time of occurrence for each case is listed in Table A-6.

The results of the steam line break studies showed that the limiting condition occurs for the break downstream of the flow restrictors (greatest cross sectional flow area) with offsite power available. Results for this case are shown in Figures A-8 thru A-19.

V. DESCRIPTION OF THE LIMITING SLB EVENT

As described in the previous section, the limiting steam line break (SLB) event occurred for a break postulated downsteam of the flow restrictors with offsite power available. The sequence of events is given in Table A-7. Various responses in the NSSS are shown in Figures A-8 through A-19.

TABLE A-6

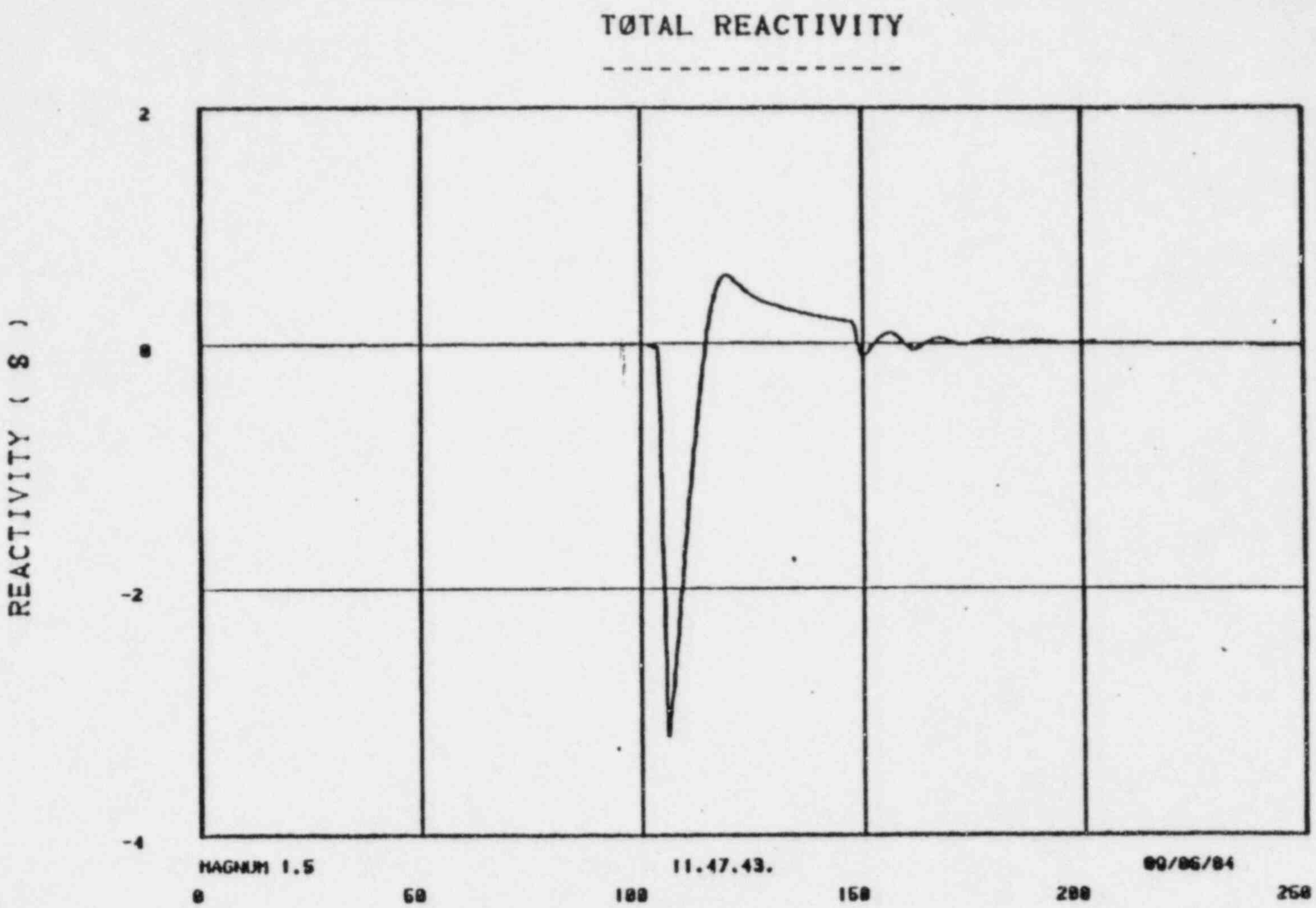
MAXIMUM RETURN TO POWER FOR CASES 1-4

	Maximum Return to Power (% of 2300 MWt)	Time (sec)
Case 1	19.2	48.0
Case 2	22.4	47.4
Case 3	13.1	51.2
Case 4	13.6	50.6

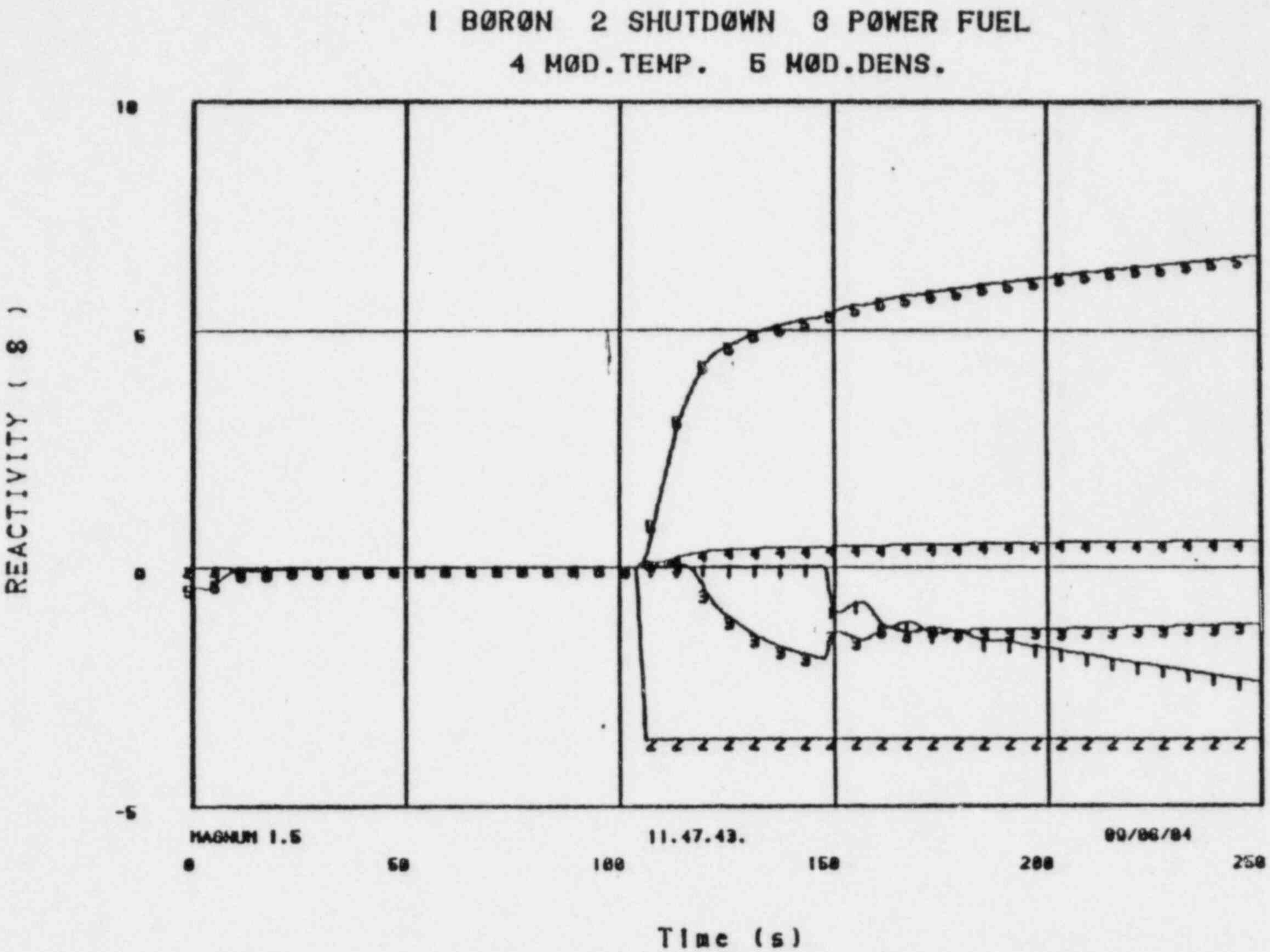
TABLE A-7

SEQUENCE OF EVENTS FOR
THE LIMITING SLB

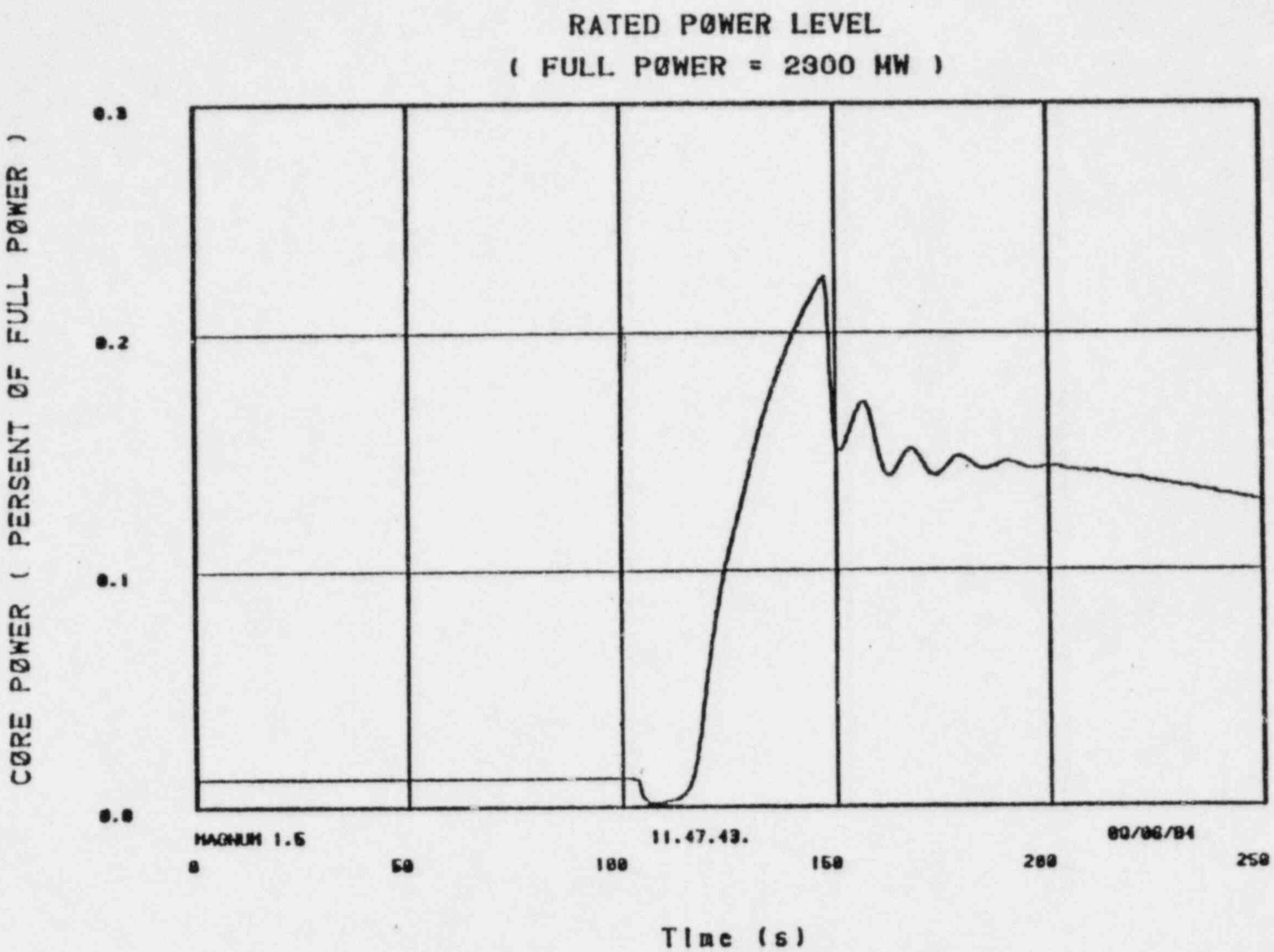
Time (Seconds)	Event
0.00	Break initiation
0.02	High steam flow signal
3.58	SI signal initiated all feedwater diverted to affected steam generator
3.80	Reactor Trip Initiated
5.60	Low steam line pressure signal
10.00	MSIV closed
13.58	Feedwater stopped
14.60	Reactivity becomes positive
16.58	HPI initiated
19.00	Peak reactivity
46.60	SI boron enters core
47.40	Peak Core Power



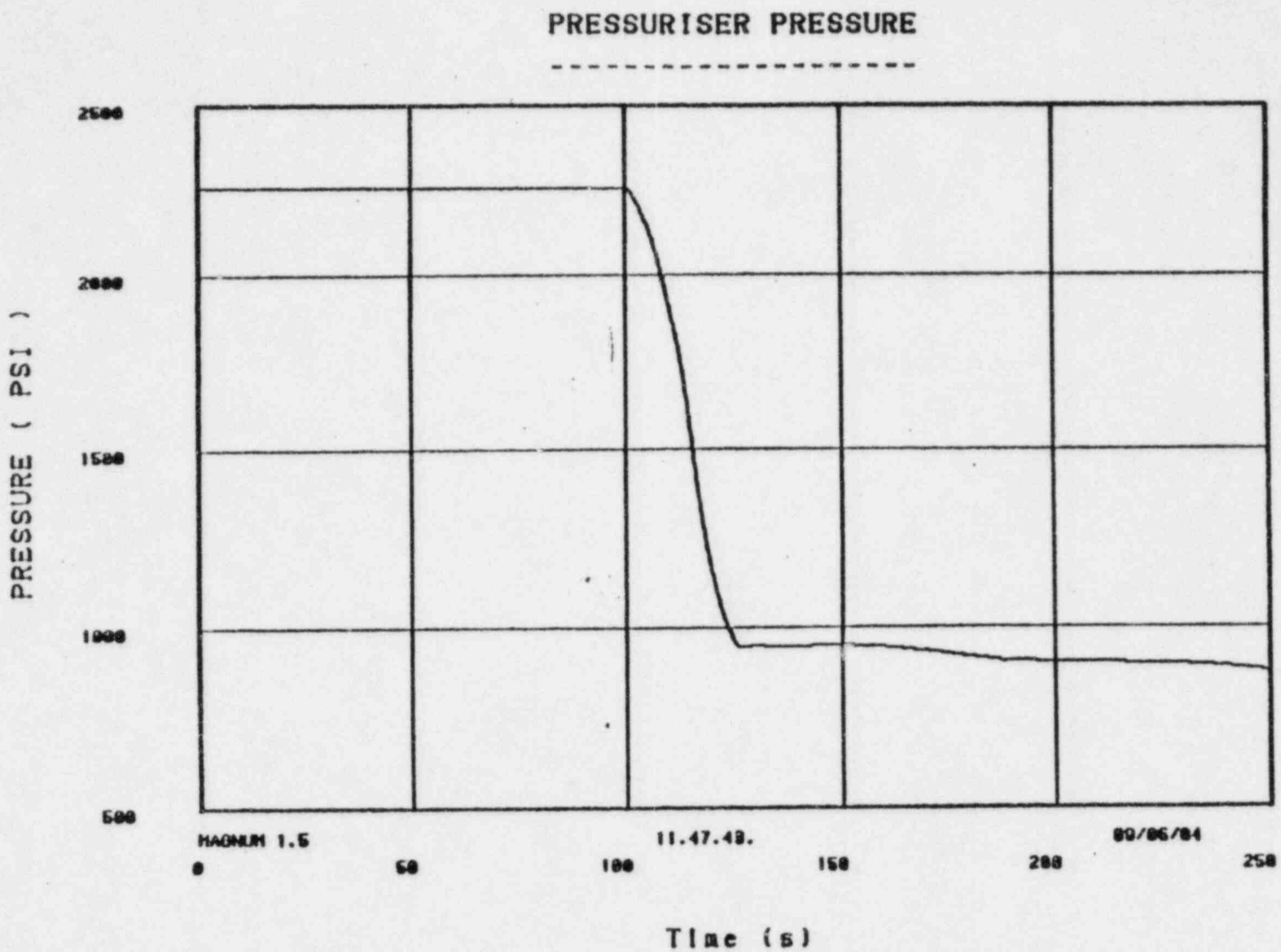
H.B.RØBINSØN CYCLE 10 RELØAD
STEAM LINE BREAK ANALYSIS
BREAK DØWNSTREAM ØF FLØW RESTRICTØRS
FIGURE A-8



H.B.RØBINSØN CYCLE 10 RELØAD
STEAM LINE BREAK ANALYSIS
BREAK DØWNSTREAM ØF FLØW RESTRICTØRS
FIGURE A-9



H.B.RØBINSØN CYCLE 10 RELØAD
STEAM LINE BREAK ANALYSIS
BREAK DØWNSTREAM ØF FLØW RESTRICTØRS
FIGURE A-10



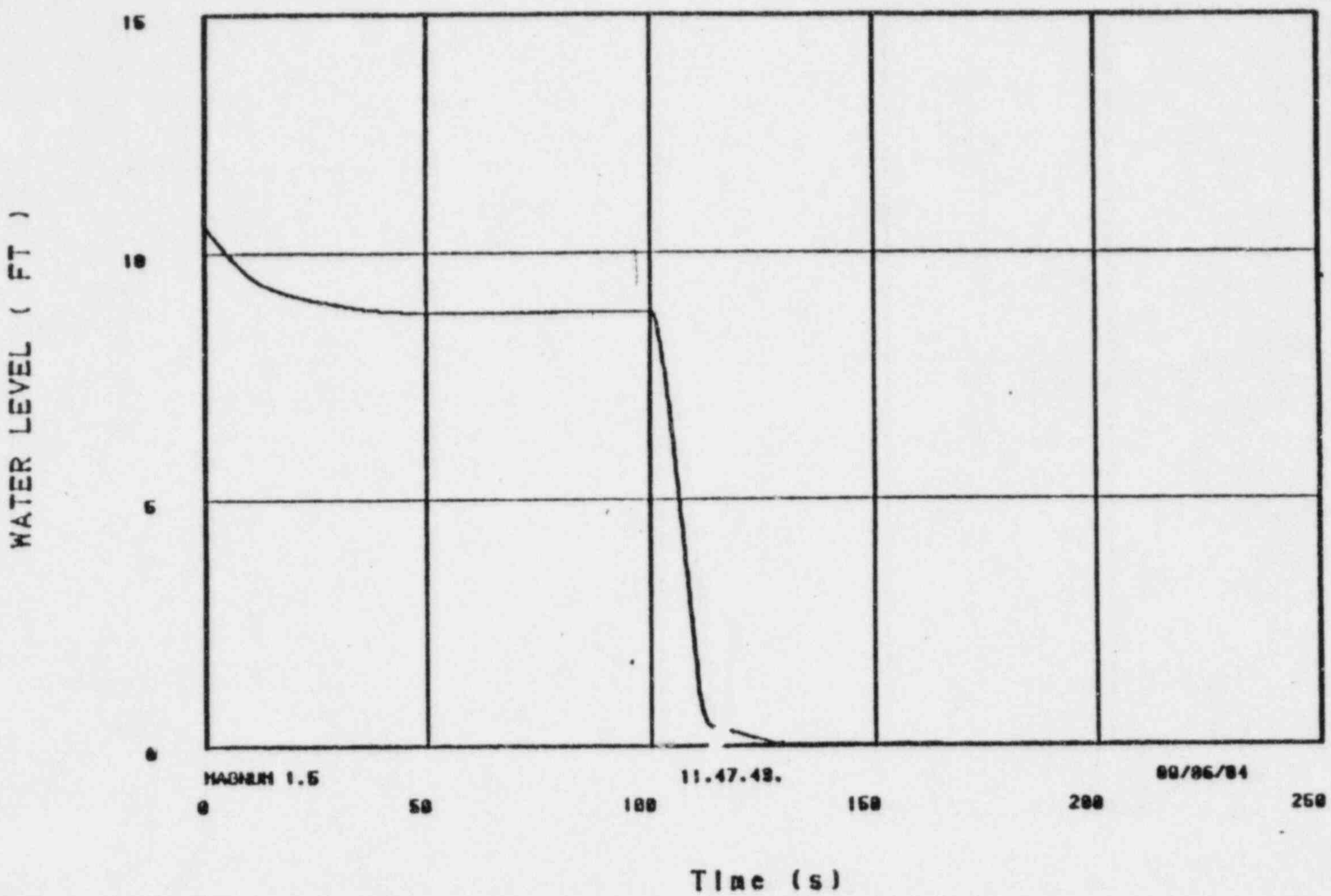
H.B.RØBINSØN CYCLE 10 RELØAD

STEAM LINE BREAK ANALYSIS

BREAK DØWNSTREAM ØF FLØW RESTR:CTØRS

FIGURE A-11

PRESSURIZER LEVEL

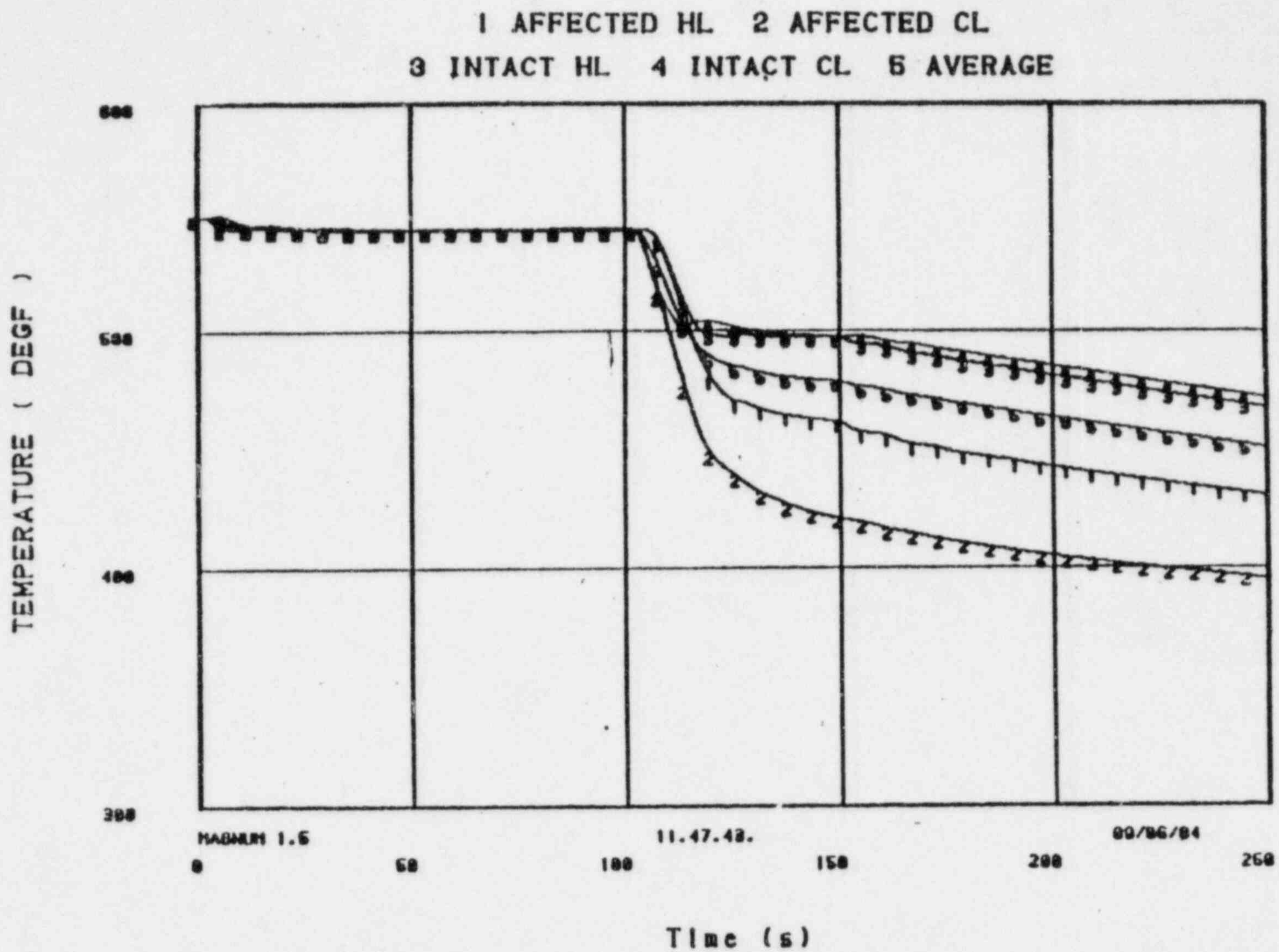


H.B.RØBINSØN CYCLE 10 RELØAD

STEAM LINE BREAK ANALYSIS

BREAK DØWNSTREAM ØF FLØW RESTRICTØRS

FIGURE A-12

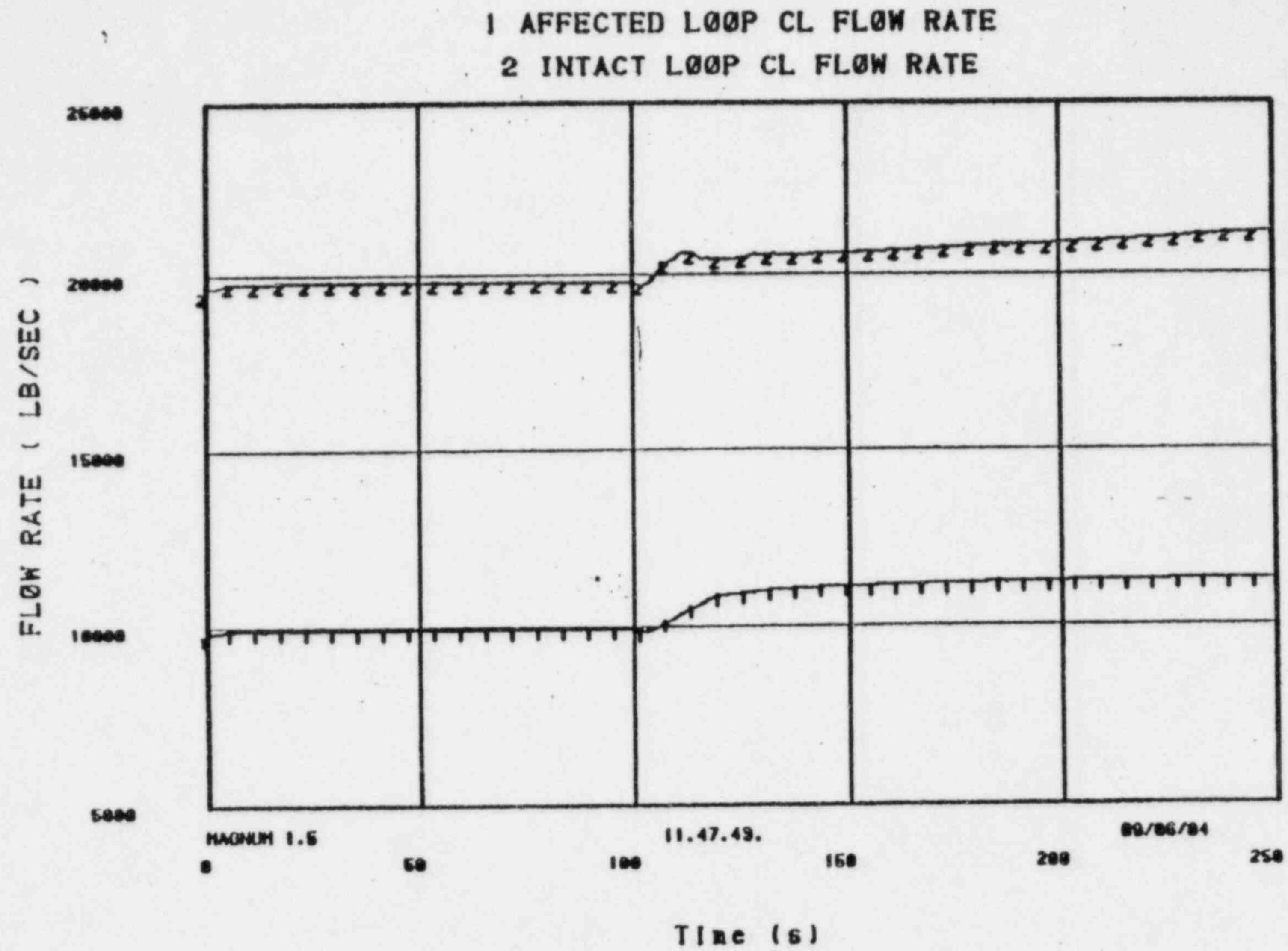


H.B.RØBINSØN CYCLE 10 RELØAD

STEAM LINE BREAK ANALYSIS

BREAK DØWNSTREAM ØF FLØW RESTRICTØRS

FIGURE A-13



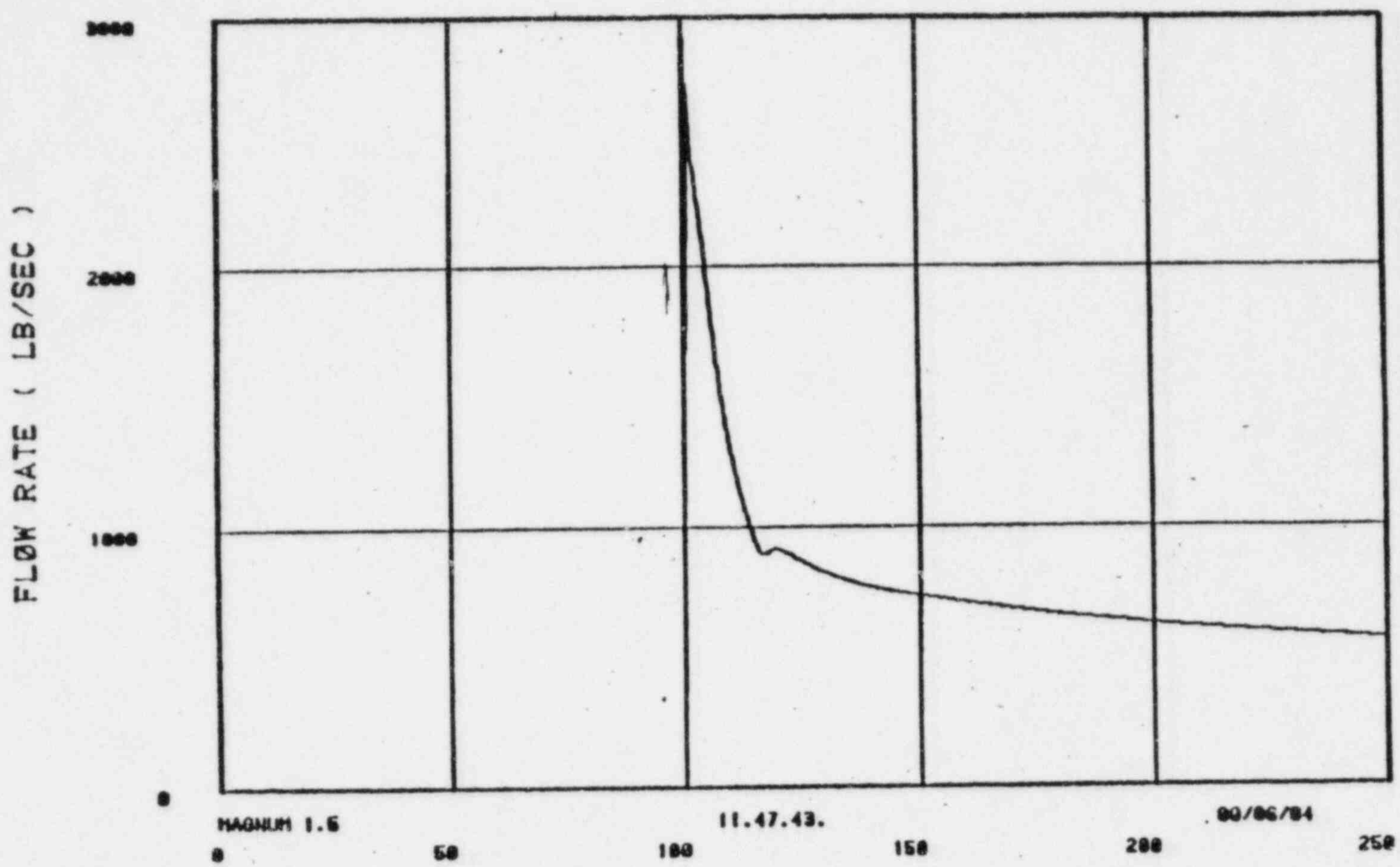
H.B.RØBINSØN CYCLE 10 RELØAD

STEAM LINE BREAK ANALYSIS

BREAK DØWNSTREAM ØF FLØW RESTRICTØRS

FIGURE A-14

BREAK FLØW RATE



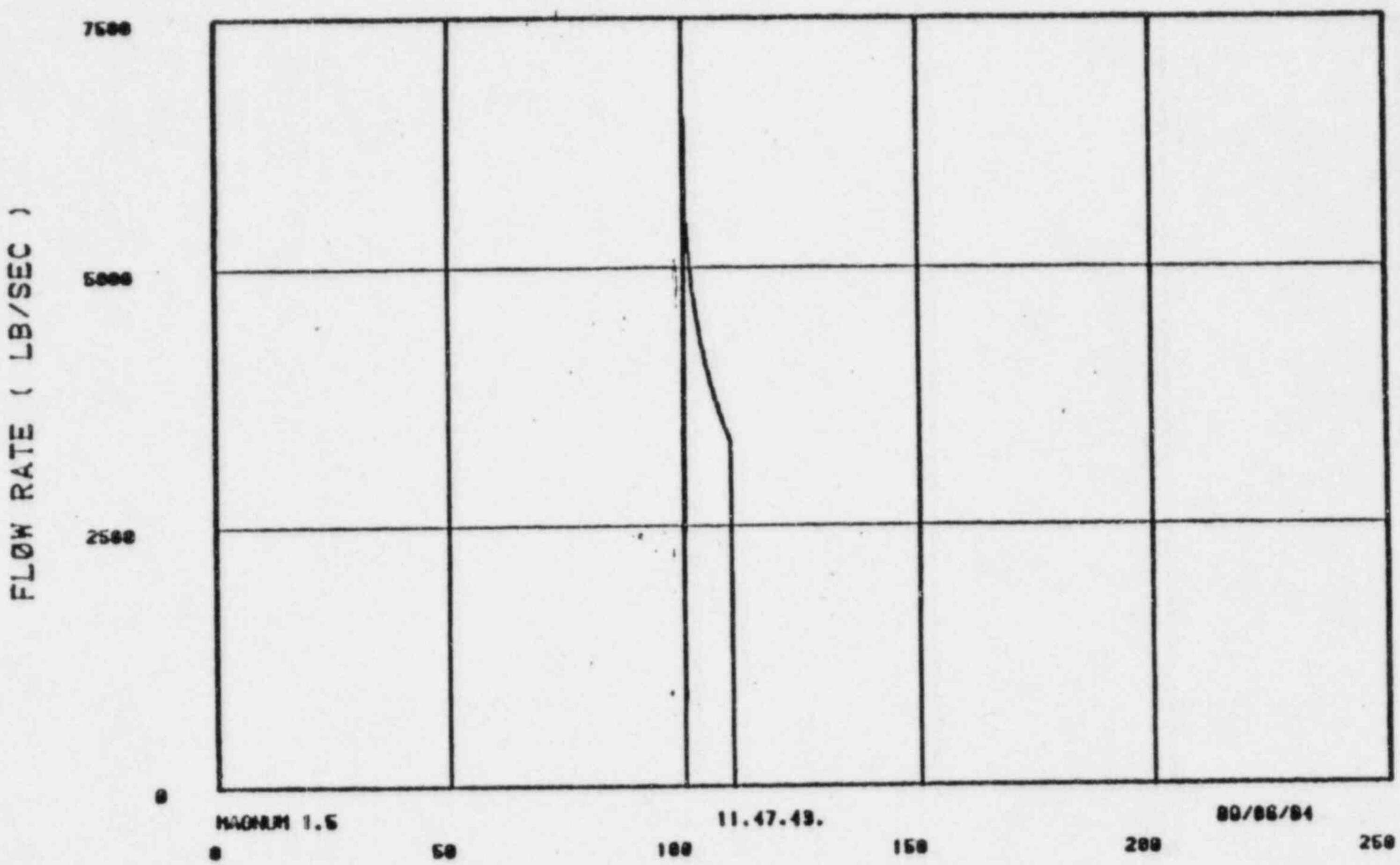
H.B.RØBINSØN CYCLE 10 RELØAD

STEAM LINE BREAK ANALYSIS

BREAK DØWNSTREAM ØF FLØW RESTRICTØRS

FIGURE A-15

INTACT STEAM LINE FLØW RATE

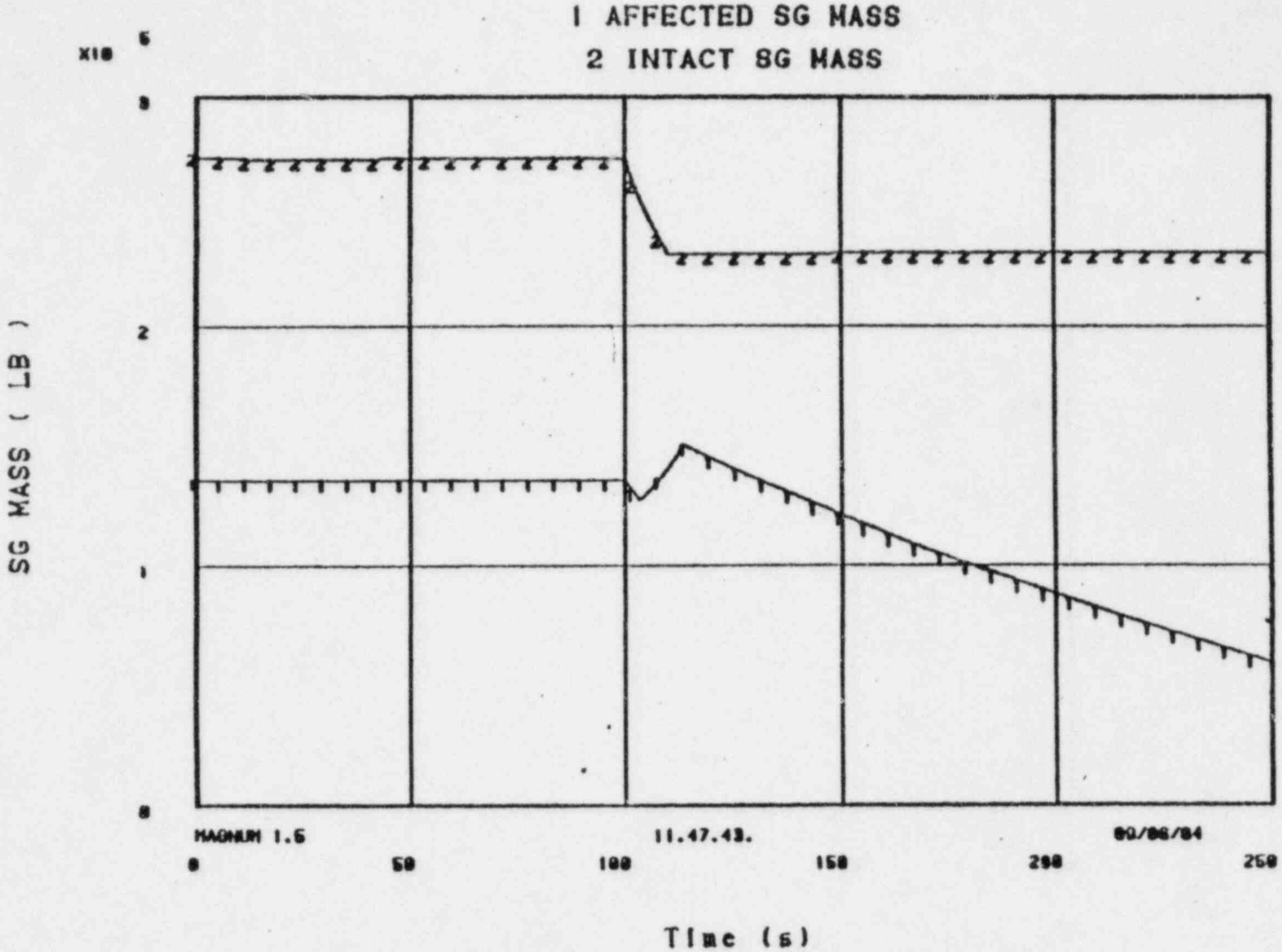


H.B.RØBINSØN CYCLE 10 RELØAD

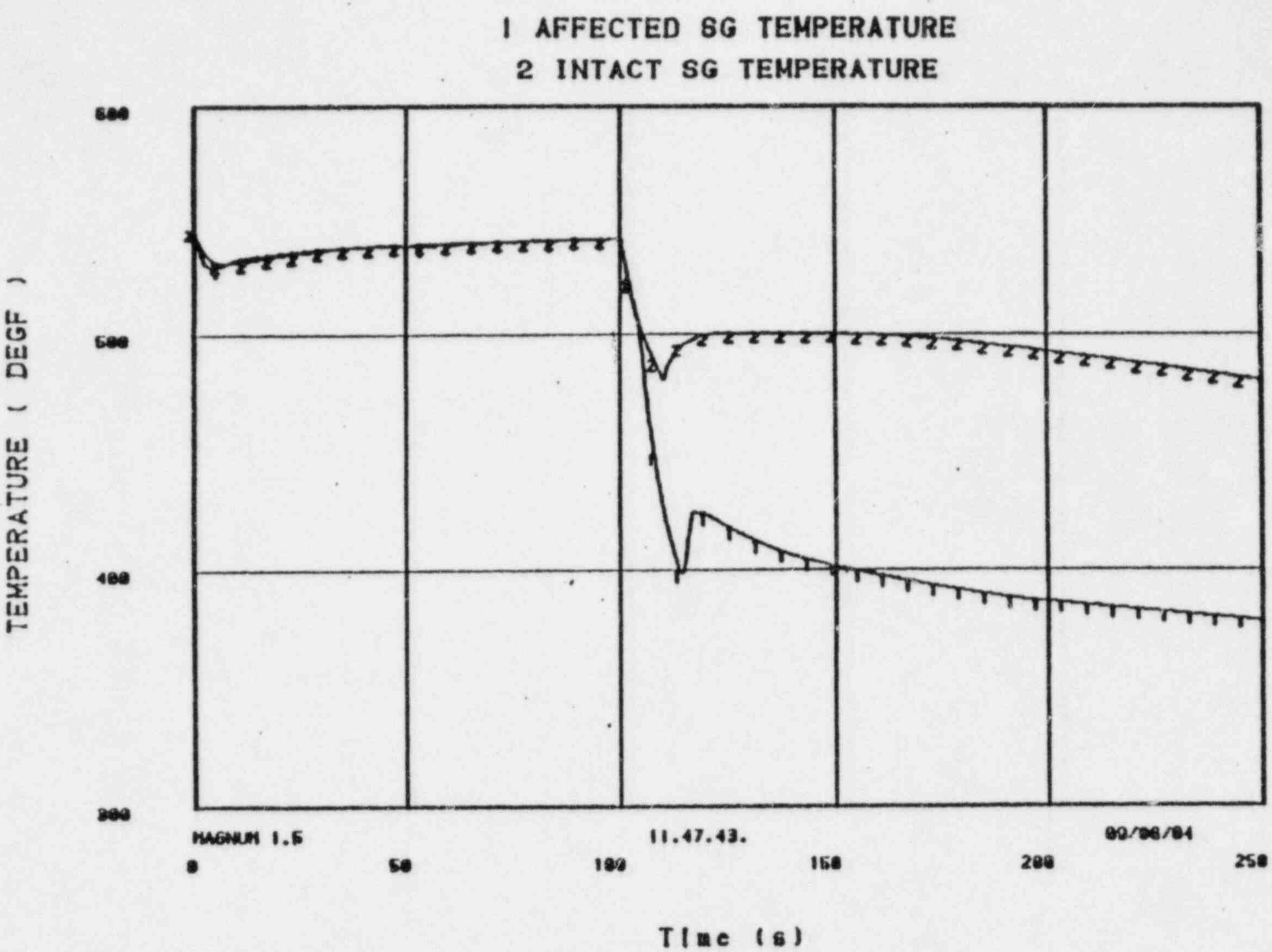
STEAM LINE BREAK ANALYSIS

BREAK DØWNSTREAM ØF FLØW RESTRICTØRS

FIGURE A-16

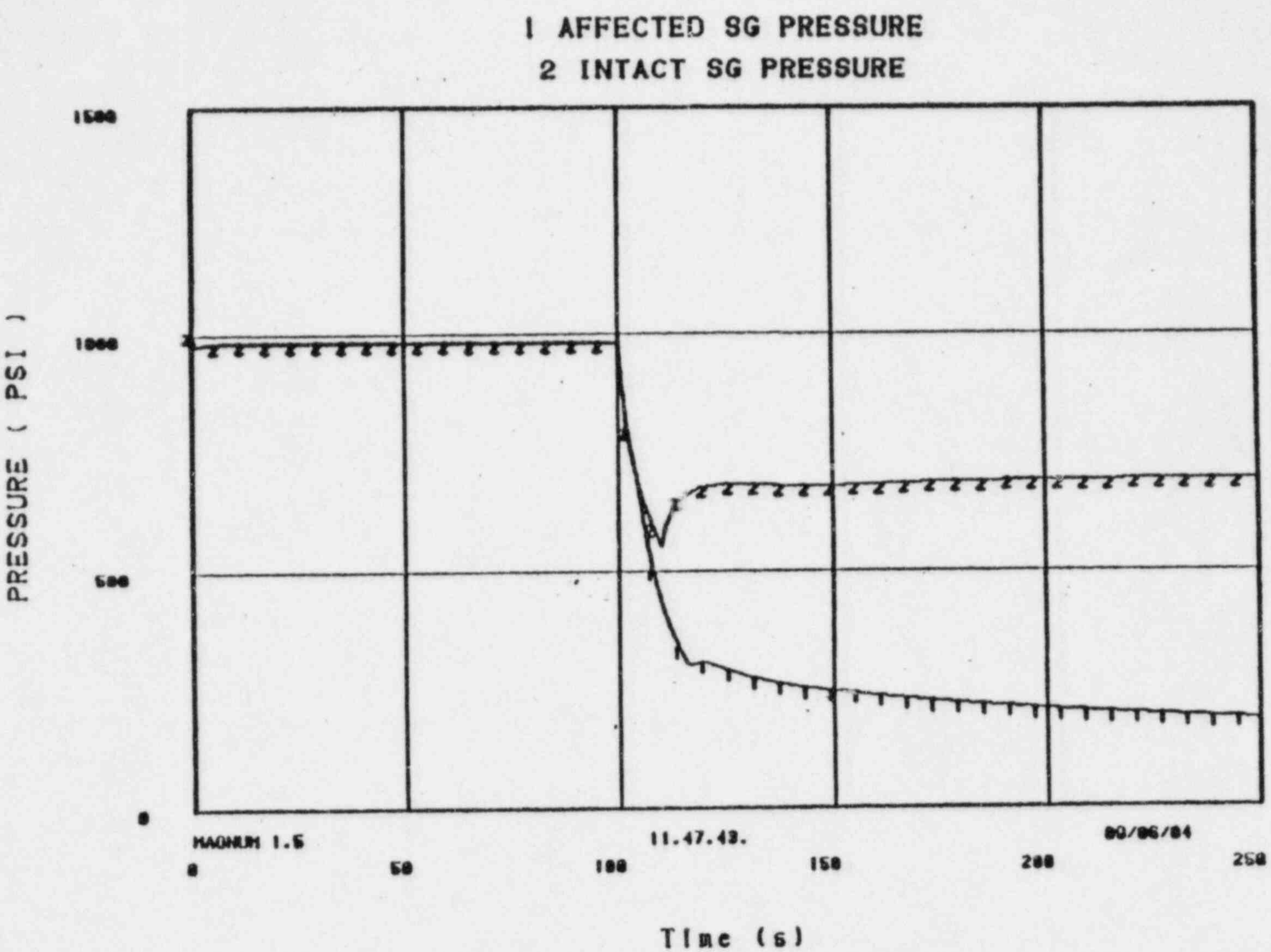


H.B.ROBINSON CYCLE 10 RELOAD
STEAM LINE BREAK ANALYSIS
BREAK DOWNSTREAM OF FLOW RESTRICTORS
FIGURE A-17



H.B.RØBINSØN CYCLE 10 RELØAD
STEAM LINE BREAK ANALYSIS
BREAK DØWNSTREAM ØF FLØW RESTRICTØRS

FIGURE A-18



H.B.RØBINSØN CYCLE 10 RELØAD
STEAM LINE BREAK ANALYSIS
BREAK DØWNSTREAM ØF FLØW RESTRICTØRS
FIGURE A-19

The reactivity for the limiting case is shown in Figure A-8 and A-9. The initial reactivity begins at zero. $3.61 of negative reactivity is inserted by the control rods between 0.2 and 2.4 seconds. As the primary system is cooled, positive reactivity is inserted by the moderator and Doppler feedbacks. Criticality occurs at 14.6 seconds. At 19 seconds the reactivity peaked at $0.58. At 46.6 seconds, boron enters the core through the emergency core cooling system (ECCS).

The reactor power response is shown in Figure A-10. The power peaked at a value of 22.4% of rated power (2300 MWt) at 47 seconds. Competing effects between the boron and Doppler reactivities led to some oscillatory power response.

Heat removal from the primary system led to a rapid decrease in primary system pressure (see Figure A-11). The depressurization rate is significantly reduced as the reactor vessel voids within the upper head. The pressurizer is depleted of liquid inventory at the same time as the depressurization rate decreased (see Figure A-12).

The hot leg and cold leg coolant temperature for both the affected and intact loops, along with the average core coolant temperature, are shown in Figure A-13. As noted by the decreasing coolant temperatures, the energy removed by the steam generators exceed the energy generated by the core. The addition of borated ECC water assures a steady decline in reactor power. As the primary system coolant temperature is decreased, the primary system flow increases. This is in response to the increasing density. Figure A-14 shows that primary system flow as a function of time.

Figure A-15 and A-16 describe the break flow characteristics for the affected and intact steam generators, respectively. The blowdown of the steam generators is the primary forcing function for the SLB transient. At 150 seconds into the event (250 seconds of plot time), the break flow from the affected steam generator decreased to 577 lbm/sec. This is equivalent to the 20% of rated steam flow. The rapid isolation of the intact steam generators result from closure of the MSIVs.

The fluid inventory, temperature and pressure for the intact and affected steam generator shells are shown in Figures A-17, A-18, and A-19. The increase in mass to the affected generators (during the initial 10 seconds of the event) comes from the addition of main feedwater. As subcooling is decreased, the secondary temperature in the downcomer begins a momentary increase. As the affected steam generator continues its blowdown, the secondary coolant temperature and pressure steadily decrease.

VI. SUMMARY

The staff analysis of a steam line break event for H. B. Robinson Unit 2, Cycle 10, confirmed that the new steam generators with integral flow restrictors decreased the severity of the event when compared with the design basis FSAR analysis. The design basis analysis resulted in a 39% return to power. This was confirmed by the benchmark analysis conducted in this review (see Section III to this Appendix).

The design basis analysis, with its 39% return to power, did not result in a calculated DNBR below the specified acceptable fuel design limit (SAFDL). Since the H. B. Robinson Unit 2 analysis for Cycle 10 resulted in a return to power of only 22.4% (approximately 50% of the design basis calculation), the margin to the SAFDL is significantly increased. Consequently, fuel integrity is maintained.