See Correction letter of 11/20/84 + Correction letter of 3/15/85

November 7, 1984

Docket No. 50-261

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Mr. E. E. Utley, Executive Vice President
Power Supply and Engineering & Construction
Carolina Power and Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

The Commission has issued the enclosed Amendment No. 87 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your request dated July 23 and August 1, 1984, as supplemented by letters dated August 8, 17, 20(2), and 23, 1984; September 7(2) and 17, 1984; and October 4, 12, and 22, 1984.

The amendment:

1. Authorizes Cycle 10 operation at full power (2300 Mwt) with new steam generators.

2. Revises the Appendix A Technical Specification to:

    a. Incorporate changes resulting from Cycle 10 core reload analysis,

    b. Add provisions for certain control rod evolutions while the containment is not intact and,

    c. Incorporate administrative changes for consistency within the Technical Specifications and with the FSAR.

By letters dated October 12, 1984, and November 7, 1984, CP&L has committed to the following confirmatory documentation:

1. Submit consequences of postulated steam line break events, including documentation of the methodology and a copy of the RELAP 5 input deck by January 31, 1985 (see SER Attachment I, 15.0 and 15.1.5).

2. Submit sensitivity studies not included in document XN-NF-84-73(P) as described in SER page 17, Chapter 15 transient and Accident Events as expanded in Attachment I Sections 15.2.2.1, 15.3.1, 15.3.1.1 and 15.3.3.1, by December 31, 1984.

3. Provide code validation of SLOTRAX by November 30, 1984 (see page 3 of Attachment I).

4. Provide confirmation of the scram shutdown margin analysis during CY 1985.

5. Submit a fuel misloading analysis during Cycle 10 operations.

We have not completed our review of the Exxon Topical Report "Mechanical Design Report Supplement for H. B. Robinson Extended Burnup Fuel Assemblies", XN-NF-83-55, submitted by your letter dated October 5, 1984. This report is almost identical to the Exxon generic report XN-NF-82-06. The peak assembly burnup at the end of Cycle 10 is about 35,000 MWD/MTU which is still in the range of normal burnup. Therefore, we will postpone our review of XN-NF-83-55 until the generic report XN-NF-82-06 can be approved. Our current target for completing the generic report is December 1984.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

Glode Requa, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. to DPR-23
2. Safety Evaluation

cc: w/enclosures
See next page

ORB#1:DL	ORB#1:DL	~~C-ORB#1:DL~~	OELD	AD:OR:DL
CParrish	GRequa;ps	SVarga	H. Karman	GLainas
10/24/84	10/24/84	10/ /84	10/25/84	10/ /84

11/2/84

Mr. E. E. Utley
Carolina Power and Light Company

H. B. Robinson Steam Electric
Plant 2

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County Courthouse
Darlington, South Carolina 29535

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
License No. DPR-23

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Carolina Power and Light Company (the licensee) dated July 23, 1984 and August 1, 1984, as supplemented by letters dated August 8, 17, and 20(2), and 23, 1984; September 7(2) and 17, 1984; and October 4, 12, and 22, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-23 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 87 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel G. McDonald

for Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 7, 1984

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 87 FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.1-1 thru 2.1-8	2.1-1 thru 2.1-4
2.3-1 thru 2.3-6	2.3-1 thru 2.3-6
3.1-1 thru 3.1-3a	3.1-1 thru 3.1-3b
3.1-11 thru 3.1-12	3.1-11 thru 3.1-12
3.5-7 and 3.5-7a	3.5-7 and 3.5-7a
3.5-10a	3.5-10a
3.6-1 thru 3.6-2	3.6-1 thru 3.6-2
3.6-2a	3.6-3
3.8-6	3.8-6
3.10-2 thru 3.10-7	3.10-2 thru 3.10-7b
3.10-12	3.10-12
3.10-14 thru 3.10-20	3.10-14 thru 3.10-20
3.10-22	3.10-22 thru 3.10-24
3.11-1 thru 3.11-2	3.11-1 thru 3.11-2
4.11-1 thru 4.11-3	4.11-1 thru 4.11-3
5.3-1 thru 5.3-2	5.3-1 thru 5.3-2

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, coolant temperature, and flow when the reactor is critical.

Objective

To maintain the integrity of the fuel cladding.

Specification

a. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1 when full flow from three reactor coolant pumps exists.

b. When full flow from one reactor coolant pump exists, the thermal power level shall not exceed 20%, the coolant pressure shall remain between 1820 psig and 2400 psig, and the Reactor Coolant System average temperature shall not exceed 590°F.

c. When natural circulation exists, the thermal power level shall not exceed 12%, the coolant pressure shall remain between 2135 psig and 2400 psig, and the Reactor Coolant System average temperature shall not exceed 620°F.

d. The safety limit is exceeded if the combination of Reactor Vessel inlet temperature and thermal power level is at any time above the appropriate pressure line in Figure 2.1-1 or if the thermal power level, coolant pressure, or Reactor Vessel inlet temperature violates the limits specified above.

Amendment No. 87

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by maintaining the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters, thermal power, reactor coolant temperature and pressure, have been related to DNB through the XNB DNB correlation. The XNB DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNB ratio, DNBR, during normal operational transients and anticipated transients is limited to 1.17. A DNB ratio of 1.17 corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.[1] The DNB ratio limit of 1.17 is a conservative design limit which is used as a basis for setting core safety limits. Based on rod bundle DNB tests, no fuel rod damage is expected at this DNB ratio or greater.

The curves of Figure 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (three loop operation) represent the loci of points of thermal power, reactor vessel inlet temperature, and coolant system pressure for which the DNB ratio is not less than 1.17. The area where clad integrity is assured is below these lines. The temperature limits at low power are considerably more conservative than would be required if they were based upon a minimum DNB ratio of 1.17 but

Amendment No. 87

are set to preclude bulk boiling at the vessel exit. An arbitrary upper safety limit of 118% thermal power is shown. This limit is based on the high flux trip including all uncertainties.

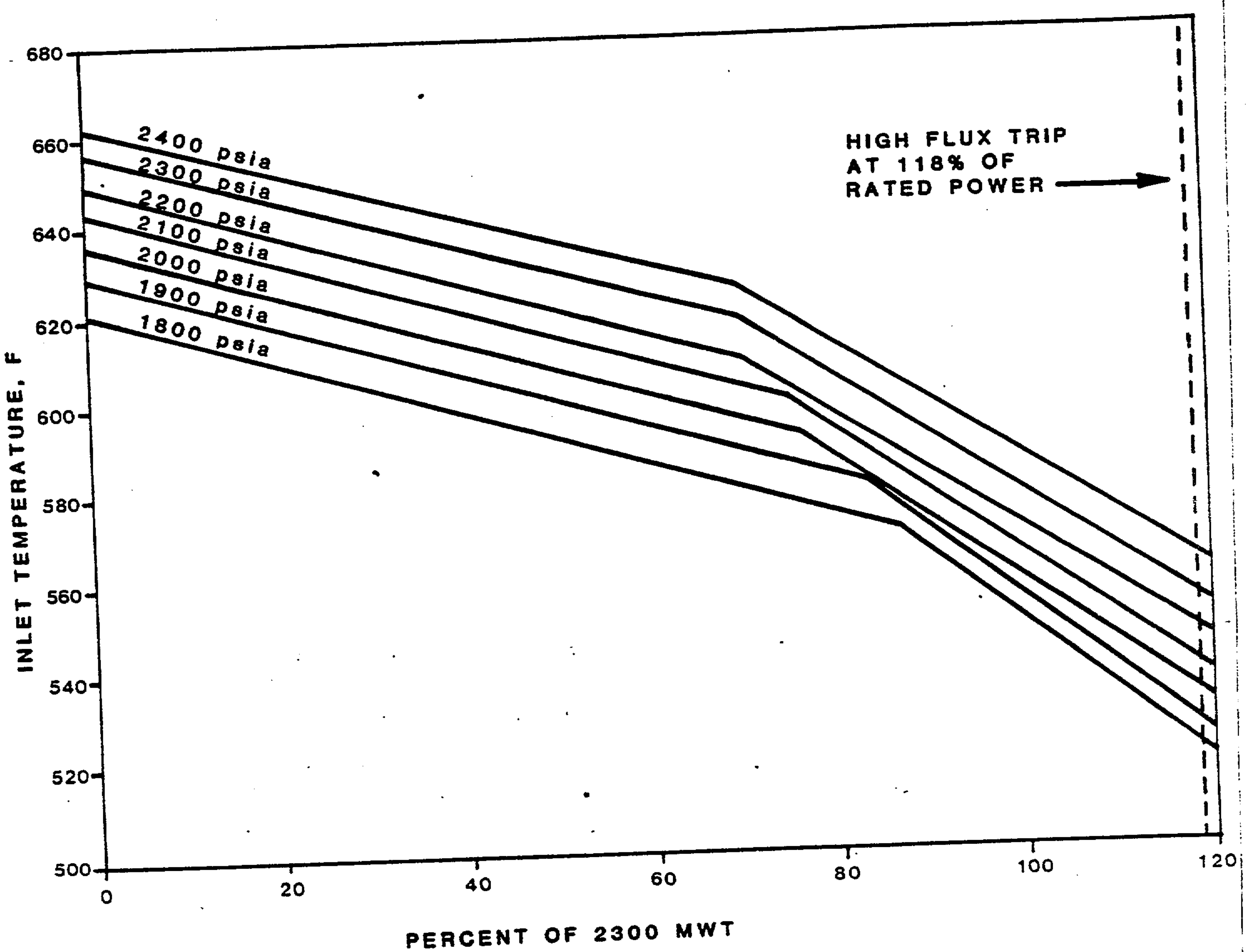
Radial power peaking factors consistent with the limit on $F_{\Delta H}$ given in Specification 3.10.2.1 have been employed in the generation of the curves in Figure 2.1-1. An additional heat flux factor of 1.03 has been included to account for fuel manufacturing tolerances and in-reactor densification of the fuel.

The safety limit curves given in Figure 2.1-1 are for constant flow conditions. These curves would not be applicable in the case of a loss of flow transient. The evaluation of such an event would be based upon the analysis presented in Section 15.3 of the FSAR.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure, and thermal power level that would result in a DNB ratio of less than 1.17[2] based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 575.4°F, and a steady state nominal operating pressure of 2235 psig. Allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature, and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions.

References

(1) XN-NF-711(P) Rev. 0, "XNB Addendum for 26 Inch Spacer."
(2) FSAR Section 15.



CORE PROTECTION BOUNDARIES FOR 3-LOOP OPERATION
Figure 2.1-1

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, and flow and pressurizer level.

Objective

To provide for automatic protection action in the event that the principal process variables approach a safety limit.

Specification

2.3.1 Protective instrumentation settings for reactor trip shall be as follows:

2.3.1.1 Start-up protection

a. High flux, power range (low setpoint) $\leq$ 25% of rated power.

2.3.1.2 Core protection

a. High flux, power range (high setpoint) $\leq$ 109% of rated power

b. High pressurizer pressure $\leq$ 2385 psig.

c. Low pressurizer pressure $\geq$ 1835 psig.

d. Overtemperature ΔT

$$\leq \Delta T_o \{K_1 - K_2 \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} (T - T') + K_3 (P - P') - f(\Delta I) \}$$

where:

ΔT_o = Indicated ΔT at rated thermal power;
T = Average temperature, °F;
P = Pressurizer pressure, psig;
K_1 < 1.1565;
K_2 = 0.01228;
K_3 = 0.00089;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation;
τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg}, τ_1 = 20 seconds, τ_2 = 3 seconds;
T' = 575.4°F Reference T_{avg} at rated thermal power;
P' = 2235 psig (Nominal RCS Operating Pressure);
S = Laplace transform operator, sec^{-1};

and f(ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant start-up tests such that:

(1) For (q_t - q_b) within +12% and -17%, where q_t and q_b are percent power in the top and bottom halves of the core, respectively, and q_t + q_b is total core power in percent of rated power (2300 Mwt), f(ΔI) = 0. For every 2.4% below rated power (2300 Mwt) level, the permissible positive flux difference range is extended by +1 percent. For every 2.4% below rated power (2300 Mwt) level, the permissible negative flux difference range is extended by -1 percent.

(2) For each percent that the magnitude of (q_t - q_b) exceeds +12% in a positive direction, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power (2300 Mwt).

(3) For each percent that the magnitude of $(q_t - q_b)$ exceeds -17% in the negative direction, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power (2300 Mwt).

e. Overpower Δ T

$$\leq \Delta T_o \{ K_4 - K_5 (\frac{\tau_3 S}{1 + \tau_3 S}) T - K_6 (T - T') - f(\Delta I) \}$$

where:

ΔT_o = Indicated ΔT at rated thermal power, °F;

T = Average temperature, °F;

T' = 575.4°F Reference T_{avg} rated thermal power;

K_4 < 1.07;

K_5 = 0.0 for decreasing average temperature, 0.02 sec/°F for increasing average temperature;

K_6 = 0.00277 for $T > T'$ and 0 for $T \leq T'$;

S = Laplace transform operator, sec^{-1};

$\frac{\tau_3 S}{1 + \tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation;

τ_3 = Time constant utilized in the rate-lag controller for T_{avg}, τ_3 = 10 seconds;

$f(\Delta I)$ = As defined in d. above

f. Low reactor coolant loop flow $\geq$ 90% of normal indicated flow.

g. Low reactor coolant pump frequency $\geq$ 57.5 Hz.

h. Undervoltage $\geq$ 70% of normal voltage.

2.3.1.3 Other Reactor Trips

a. High pressurizer water level $\leq$ 92% of span.

b. Low-low steam generator water level $\geq$ 14% of narrow range instrument span.

2.3.2 Protective instrumentation settings for reactor trip interlocks shall be as follows:

2.3.2.1 The low pressurizer pressure trip, high pressurizer level trip, and the low reactor coolant flow trip (for two or more loops) may be bypassed below 10% of rated power.

2.3.2.2 The single-loop-loss-of-flow trip may be bypassed below 45% of rated power.

Basis

The power range reactor trip low setpoint provides protection in the power range for a power excursion beginning from lower power. This trip value was used in the safety analysis.(1)

In the power range of operation, the overpower nuclear flux reactor trip protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed set point, with allowance for errors, is consistent with the trip point assumed in the accident analysis.(2)

The source and intermediate range reactor trips do not appear in the specification, as these settings are not used in the transient and accident analysis (FSAR Section 15). Both trips provide protection during reactor startup. The former is set at about 10^{+5} counts/sec and the latter at a current proportional to approximately 25% of full power.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is also a backup to the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss-of-coolant accident.(3)

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds)[4], and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors,[2] is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to Specification 2.3.1.2.d.

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 118% of design power density as discussed in Section 7.2.2 of the FSAR and includes corrections for axial power distribution, change in density, and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[2]

The setpoints in the Technical Specifications ensure the combination of power, temperature, and pressure will not exceed the core safety limits as shown in Figure 2.1-1.

The low flow reactor trip protects the core against DNB in the event of a sudden loss of power to one or more reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.[5] The undervoltage and underfrequency reactor trips protect against a decrease in flow caused by low electrical voltage or frequency. The specified setpoints assure a reactor trip signal before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1150 ft^3 of water corresponds to 92% of span. The specified setpoint allows margin for instrument error[2] and transient level overshoot beyond this trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified set point assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.[6]

The specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point above which these trips are unblocked assures their availability in the power range where needed.

Above 10% power, an automatic reactor trip will occur if two reactor coolant pumps are lost during operation. Above 45% power, an automatic reactor trip will occur if any pump is lost. This latter trip will prevent the minimum value of the DNB ratio, DNBR, from going below 1.17 during normal operational transients and anticipated transients when only two loops are in operation and the overtemperature ΔT trip setpoint is adjusted to the value specified for three loop operation.

The turbine and steam-feedwater flow mismatch trips do not appear in the specification, as these settings are not used in the transient and accident analysis. (FSAR Section 15)

References

(1) FSAR Section 15.4
(2) FSAR Section 15.0
(3) FSAR Section 15.6
(4) FSAR Section 15.4.2
(5) FSAR Section 15.3
(6) FSAR Section 15.2

3.0 LIMITING CONDITIONS FOR OPERATION

Except as otherwise provided for in each specification, if a Limiting Condition for Operation cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in hot shutdown within eight hours and in COLD SHUTDOWN within the next 30 hours unless corrective measures are taken that permit operation under the permissible Limiting Condition for Operation statements for the specified time interval as measured from initial discovery or until the reactor is placed in a condition in which the specification is not applicable.

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those Reactor Coolant System conditions which must be met to assure safe reactor operation.

Specification

3.1.1 Operational Components

3.1.1.1 Coolant Pumps

a. With reactor power less than 2% of rated thermal power and less than two reactor coolant pumps in operation, one of the following actions shall be taken:

   1. maintain a shutdown margin of at least 4% $\Delta k/k$, or
   2. open the lift disconnect switches for all control rods not fully withdrawn, or
   3. open reactor trip breakers.

b. Power operation with less than three loops in service is prohibited.

c. At least one reactor coolant pump or residual heat removal pump shall be in operation when T_{avg} > 200°F and reactor power is less than 2% of rated thermal power. In the event this condition cannot be satisfied, the following actions shall be taken:

1. Proceed to establish a boron concentration in the reactor coolant equal to or greater than that concentration needed to maintain a shutdown margin of 1% Δk/k at 200°F, and
2. Restore at least one reactor coolant pump or residual heat removal pump to operation within one hour, or prepare and submit a Special Report to the NRC within 30 days.

d. A reactor coolant pump may be started (or jogged) only if there is a steam bubble in the pressurizer or the steam generator temperature is no higher than 50°F higher than the temperature of the reactor coolant system.

Basis

Specification 3.1.1.1.a contains requirements designed to limit the consequences of the uncontrolled bank withdrawal at low or subcritical power conditions as analyzed in the safety analysis. The requirement of two reactor coolant pumps in operation below 2% power is consistent with the assumptions utilized in the bounding transient that was analyzed. The specification makes allowance for less than two pumps in operation by specifying either of three actions that must be taken. Either maintaining the specified shutdown margin, opening the lift disconnect switches on the control rods or opening the reactor trip breakers will prevent the occurrence of the postulated uncontrolled bank withdrawal transient, therefore allowing the two pump requirement to be lifted.

Maintaining a shutdown margin of 4% Δk/k is sufficient to prevent a return to criticality if the worth of the two most reactive control rod banks are simultaneously withdrawn as is the assumption of the postulated transient.

Specification 3.1.1.1.b requires that all three reactor coolant pumps be operating during power operation to provide core cooling in the event that a loss of flow occurs. The flow provided will keep DNB well above 1.17. Therefore, cladding damage and release of fission products to the reactor coolant will not occur.

Specification 3.1.1.1.c is designed to allow for adequate mixing of the reactor coolant to maintain a uniform boron concentration during dilution, and to provide a means of boron injection. Should no residual heat removal pump or reactor coolant pump be available, boration via natural circulation shall be initiated. A boron concentration corresponding to 1% $\Delta k/k$ at 200°F (which assumes most reactive rod stuck out) would prevent a return to criticality during the cooldown phase of the postulated steam line break event. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

The purpose of Specification 3.1.1.1.d is to limit pressure surges exhibited in the RCS during a RCP startup. These pressure surges can be controlled in one of two ways. One method would be to require a steam bubble in the pressurizer and thus control pressure using pressurizer controls. The other method would be to limit the temperature difference (< 50°F) between the RCS average temperature and the idle pump's cold leg water temperature.

3.1.1.2 Steam Generator

At least two steam generators shall be operable whenever the average primary coolant temperature is above 350°F.

Basis

One steam generator capable of performing its heat transfer function will provide sufficient heat removal capability to remove core decay heat after a normal reactor shutdown. The reactor cannot be made critical without water in all three steam generators, since the low-low steam generator water level trip prevents this mode of operation. Two operable steam generators are therefore adequate.

3.1.1.3 Pressurizer (Pzr)

a. At least one Pzr code safety valve shall be operable whenever the Reactor Head is on the vessel and the RCS is not open for maintenance.

b. The Pzr, including necessary spray and heater control systems, shall be operable before the reactor is made critical.

c. Whenever the RCS temperature is above 350°F or the reactor is critical:

1. All three pressurizer code safety valves shall be operable. Their lift settings shall be maintained between 2485 psig and 2560 psig.

2. At least 125 kw of pressurizer heaters capable of being powered from an emergency power source shall be operable.

d. If the requirements of 3.1.1.3.c.2 are not met and at least 125 kw or Pzr heaters capable of being powered from an emergency source cannot be provided within 72 hrs., commence a normal plant shutdown and cooldown to an RCS average temperature of less than or equal to 350°F.

Basis

The pressurizer is necessary to maintain acceptable system pressure during normal plant operation, including surges that may result following anticipated transients.

Each of the pressurizer code safety valves is designed to relieve 288,000 lbs. per hr. of saturated steam at the valve setpoint.[1] Below 350°F and 450 psig in the Reactor Coolant System (RCS), the Residual Heat Removal System can remove decay heat and thereby control system temperature. The pressurizer safety valves are sized to protect the RCS against overpressure without taking credit for the steam bypass system.[2] If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than the capacity of a single valve. One valve therefore provides adequate defense against overpressurization of the RCS for primary coolant temperatures less than 350°F and two valves provide protection for any temperature.

ASME Section III of the Code allows a maximum variation in the setpoint of 3 percent above the design set pressure.

The requirement that 125 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency power source provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

References

(1) FSAR Table 5.4.6-1
(2) FSAR Section 15.2

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature, above which the moderator temperature coefficient is greater than:

a) +5.0 pcm/°F less than 50% of rated power, or
b) +5.0 pcm/°F at 50% of rated power and linearly decreasing to 0 pcm/°F at rated power.

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1.

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is greater than as specified in 3.1.3.1 above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

Basis

During the early part of fuel cycle, the moderator temperature coefficient may be slightly positive at low power levels. The moderator coefficient at low temperatures or powers will be most positive at the beginning of the fuel cycle, when the boron concentration in the coolant is the greatest. At all times, the moderator coefficient is calculated to be negative in the high power operating range, and after a very brief period of power operation, the coefficient will be negative in all circumstances due to the reduced boron concentration as Xenon and fission products build into the core. The requirement that the reactor is not to be made critical when the moderator coefficient is more positive than as specified in 3.1.3.1 above has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[1] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The heatup curve of Figure 3.1-1 includes criticality limits which are required by 10 CFR Part 50, Appendix G, Paragraph IV.A.2.c. Whenever the core is critical, additional safety margins above those specified by the ASME Code Appendix G methods are imposed. The core may be critical at temperatures equal to or above the minimum temperature for the inservice hydrostatic pressure tests as calculated by ASME Code Appendix G methods, and an additional safety margin of 40°F must be maintained above the applicable heatup curve at all times.

If the specified shutdown margin is maintained (Section 3.10), there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.[1]

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of one percent subcriticality will assure that the Reactor Coolant System will not be solid when criticality is achieved.

References

(1) FSAR Section 4.3

TABLE 3.5-1

ENGINEERED SAFETY FEATURE SYSTEM INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
1.	High Containment Pressure (HI Level)	Safety Injection*	≤ 5 psig
2.	High Containment Pressure (HI-HI Level)	a. Containment Spray** b. Steam Line Isolation	≤ 25 psig
3.	Pressurizer Low Pressure	Safety Injection*	≥ 1700 psig
4.	High Differential Pressure Between any Steam Line and the Steam Line Header	Safety Injection*	≤ 150 psi
5.	High Steam Flow in 2/3 Steam Lines***	a. Safety Injection* b. Steam Line Isolation	≤ 40% (at zero load) of full steam flow ≤ 40% (at 20% load) of full steam flow ≤ 110% (at full load) of full steam flow
	Coincident with Low T_{avg} or Low Steam Line Pressure		≥ 541°F T_{avg} ≥ 600 psig steam line pressure
6.	Loss of Power		
	a. 480V Emerg. Bus Undervoltage (Loss of Voltage) Time Delay	Trip Normal Supply Breaker	328 Volts ± 1 Volt .75 ± .25 sec.

TABLE 3.5-1 (Continued)

ENGINEERED SAFETY FEATURE SYSTEM INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
6. (Cont'd)	b. 480V Emerg. Bus Undervoltage (Degraded Voltage) Time Delay	Trip Normal Supply Breaker	412 Volts $\pm$ 1 Volt 10.0 Second Delay $\pm$ 0.5 sec.
7.	Containment Radioactivity High	Ventilation Isolation	$\leq$ 2 X Reading at the Time the Alarm is Set with Known Plant Conditions

* Initiates also containment isolation (Phase A), feedwater line isolation and starting of all containment fans.

** Initiates also containment isolation (Phase B).

*** Derived from equivalent ΔP measurements.

TABLE 3.5-3 (Continued)

INSTRUMENTATION OPERATING CONDITIONS FOR ENGINEERED SAFETY FEATURES

NO.	FUNCTIONAL UNIT	1 MINIMUM CHANNELS OPERABLE	2 MINIMUM DEGREE OF REDUNDANCY	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
2.	CONTAINMENT SPRAY			
	a. Manual*	2	0**	Cold Shutdown
	b. High Containment Pressure* (Hi-Hi Level)	2/set	1/set	Cold Shutdown
3.	LOSS OF POWER			
	a. 480V Emerg. Bus Undervoltage (Loss of Voltage)	2/bus[(a)]	1/bus[(b)]	Main Hot Shutdown
	b. 480V Emerg. Bus Undervoltage (Degraded Voltage)	2/bus	1/bus	Maintain Hot Shutdown[(c)]

* Also initiates a Phase B containment isolation.
** Must actuate two switches simultaneously.
*** When primary pressure is less than 2000 psig, channels may be blocked.
**** When primary temperature is less than 547°F, channels may be blocked.
***** In this case the 2/3 high steam flow is already in the trip mode.
(a) During testing and maintenance of one channel, may be reduced to 1/bus.
(b) During testing and maintenance of one channel, may be reduced to 0/bus.
(c) The reactor may remain critical below the power operating conditions with this feature inhibited for the purpose of starting reactor coolant pumps.

3.6 CONTAINMENT SYSTEM

Applicability

Applies to the integrity of reactor containment.

Objective

To define the operating status of the reactor containment for plant operation.

Specification

3.6.1 Containment Integrity

a. The containment integrity (as defined in 1.7) shall not be violated unless the reactor is in the cold shutdown condition.

b. The containment integrity shall not be violated when the reactor vessel head is removed unless a shutdown margin greater than 10% $\Delta k/k$ is constantly maintained.

c. Positive reactivity changes shall not be made by rod drive motion when the containment integrity is not intact except during any one of the following evolutions:

1. rod drop timing test

2. rod drive mechanism timing test

3. control rod exercise test

4. shutdown banks fully withdrawn and control banks withdrawn to $\leq$ 5 steps.

During any of the aforementioned evolutions the shutdown margin shall be maintained $\geq$ 1% $\Delta k/k$.

d. Positive reactivity changes shall not be made by boron dilution when the containment integrity is not intact unless the shutdown margin is maintained $\geq$ 1% $\Delta k/k$.

3.6.2 Internal Pressure

If the internal pressure exceeds 1 psig or the internal vacuum exceeds 1.0 psig, the condition shall be corrected within eight (8) hours or the operator shall start to place the reactor in the hot shutdown condition utilizing normal operating procedures.

3.6.3 Containment Automatic Isolation Trip Valves

The following exceptions apply only to automatic containment isolation valves required to be closed during accident conditions and which are either redundant or installed in a line which is part of a closed system within containment.

With one or more of the automatic containment isolation trip valves inoperable, either:

a. Restore the inoperable valve(s) to operable status within 4 hours, or

b. Isolate the affected penetration(s) within 4 hours by use of a deactivated automatic valve(s) secured in the isolation position(s), or

c. Isolate the affected penetration(s) within 4 hours by use of a closed manual valve(s) or blind flange(s), or

d. Be in cold shutdown within the next 36 hours.

Basis

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence there would be no pressure buildup in the containment if the Reactor Coolant System ruptures.

The shutdown margins are selected based on the type of activities that are being carried out. The 10% $\Delta k/k$ shutdown margin during refueling precludes criticality, even though fuel is being moved. When the reactor head is not to be removed, the specified cold shutdown margin of 1% $\Delta k/k$ precludes criticality.

Regarding internal pressure limitations, the containment design pressure of 42 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 2 psig.[(1)] The containment is designed to withstand an internal vacuum of 2.0 psig.[(2)]

References

(1) FSAR Section 6.2.1
(2) FSAR Section 3.8.1.3

The relative humidity (R.H.) of the air processed by the refueling filter systems should be less than the R.H. used during the testing of the charcoal adsorbers in order to assure that the adsorbers will perform under accident conditions as predicted by the test results. Heaters have been installed upstream of the Spent Fuel Building filters to assure of an R.H. of less than 70 percent for the air processed by the Spent Fuel Building filter system. If an R.H. in the containment atmosphere exceeds 70 percent, operation of the containment purge system will be terminated until this specification can be met. If the Spent Fuel Building filter system is found to be inoperable, all fuel handling and fuel movement operations in the Spent Fuel Building will be terminated until the system is made operable.

The temperature limit specified for the fuel cask handling crane is based on the recorded ambient temperature at the time of the 125% load test. The limit is imposed to assure adequate toughness properties of the crane structural materials.

References

(1) FSAR Section 9.4.1
(2) FSAR Section 4.3
(3) FSAR Section 9.4.1
(4) H. B. Robinson Unit 2 Radiological Assessment of Postulated Accidents, XN-NF-84-68(P), July 1984.

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows:

- at positions $\geq$ 200 steps and is > 15 inches out of alignment with its bank position, or
- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

then within two hours, perform the following:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power

3.10.1.6 Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted.

3.10.2 Power Distribution Limits

3.10.2.1 At all times except during low power physics tests, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the basis, must meet the following limits:

$F_Q(Z) \leq (2.32/P) \times K(Z)$ for $P > 0.5$

$F_Q(Z) < 4.64 \times K(Z)$ for $P \leq 0.5$

$F_{\Delta H} < 1.65 \; (1 + 0.2(1-P))$

where P is the fraction of rated power (2300 Mwt) at which the core is operating. $F_Q(Z)$ is the measured $F_Q(Z)$ including the measurement uncertainty factor $F_u^N = 1.05$ and the engineering factor $F_Q^E = 1.03$. $F_{\Delta H}$ is the measured $F_{\Delta H}$ including a 1.04 measurement uncertainty factor. K(Z) is based on the function given in Figure 3.10-3, and Z is the axial location of F_Q.

3.10.2.1.1 Following initial loading, or upon achieving equilibrium conditions after exceeding by 10% or more of rated power, the power $F_Q(Z)$ was last determined, and at least once per effective full power month, power distribution maps using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.1 are satisfied and to establish the target axial flux difference as a function of power level (called the target flux difference).*

If either measured hot channel factor exceeds the specified limit, the reactor power shall be reduced so as not to exceed a fraction equal to the ratio of the $F_Q(Z)$ or $F_{\Delta H}$ limit to the measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio.

If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

3.10.2.2 $F_Q(Z)$ shall be determined to be within the limit given in 3.10.2.1 by satisfying the following relationship for the middle axial 80% of the core at the time of the target flux determination:

$$F_Q(Z) \leq \left(\frac{2.32}{P}\right) \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P > 0.5$$

$$F_Q(Z) < 4.64 \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P \leq 0.5$$

* During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

where V(Z) is defined in Figure 3.10-4 which corresponds to the target band and $P > 0.5$.

3.10.2.2.1 If the relationship specified in 3.10.2.2 cannot be satisfied, one of the following actions shall be taken:

a) Place the core in an equilibrium condition where the limit in 3.10.2.2 is satisfied and re-establish the target axial flux difference

b) Reduce the reactor power by the maximum percent calculated with the following expression for the middle axial 80% of the core:

$$\left[\left[\text{max. over Z of } \frac{F_Q(Z) \times V(Z)}{\frac{2.32}{P} \times K(Z)} \right] -1 \right] \times 100\%$$

c) Comply with the requirements of Specification 3.10.2.2.2.

3.10.2.2.2 The Allowable Power Level above which initiation of the Axial Power Distribution Monitoring System (APDMS) is required is given by the relation:

$$\text{APL} = \text{minimum over Z of } \frac{2.32 \times K(Z)}{F_Q(Z) \times V(Z)} \times 100\%$$

where $F_Q(Z)$ is the measured $F_Q(Z)$, including the engineering factor $F_Q^E = 1.03$ and the measurement uncertainty factor $F_u^N = 1.05$ at the time of target flux determination from a power distribution map using the movable incore detectors. V(Z) is the variation function defined in Figure 3.10-4 which corresponds to the target band. K(Z) is the function defined in Figure 3.10-3.

The above limit is not applicable in the following core plane regions.

1) Lower core region 0% to 10% inclusive.
2) Upper core region 90% to 100% inclusive.

At power levels in excess of APL of rated power, the APDMS will be employed to monitor $F_Q(Z)$. The limiting value is expressed as:

$$[F_j(Z)\ S(Z)]_{max} \leq \frac{2.103/P}{\overline{R}_j\ (1 + \sigma_j)}$$

where:

a. P is the fraction of rated power (2300 Mwt) at which the core is operating ($P \leq 1.0$).

b. $\overline{R}_j$, for thimble j, is determined from core power maps and is by definition:

$$\overline{R}_j = \frac{1}{6} \sum_{i=1}^{6} \frac{F_{Qj}}{[F(Z)_{ij}\ S(Z)]_{max}}$$

F_{Qj} is the value obtained from a full core map including S(Z), but without the measurement uncertainty factor F_u^N or the engineering uncertainty factor, F_Q^E. The quantity $F(Z)_{ij}$ S(Z) is the measured value without inclusion of the instrument uncertainty factors F_Q^a. Those uncertainty factors, $F_u^N = 1.05$, $F_Q^a = 1.02$, as well as the engineering factor $F_Q^E = 1.03$, have been included in the limiting value of 2.103/P.

c. σ_j is the standard deviation associated with the determination of $\overline{R}_j$.

d. S(Z) is the inverse of the K(Z) function given in Figure 3.10-3.

This limit is not applicable during physics tests and excore detector calibrations.

3.10.2.2.3 With successive measurements indicating the enthalpy rise hot channel factor, $F_{\Delta H}^N$, to be increasing with exposure, the total peaking factor, $F_Q(Z)$, shall be further increased by two percent over that specified in Specifications 3.10.2.2, 3.10.2.2.1, and

3.10.2.2.2 or $F_Q(Z)$ shall be measured and a target axial flux difference re-established at least once every seven (7) effective full power days until two successive measurements indicate enthalpy rise hot channel factor, $F_{\Delta H}^{N}$, is not increasing.

3.10.2.3 The reference equilibrium-indicated axial flux difference as a function of power level (called the target flux difference) shall be determined in conjunction with the measurement of $F_Q(Z)$ as defined in Specification 3.10.2.1.1.*

3.10.2.4 The indicated axial flux difference shall be considered outside of the limits of Sections 3.10.2.5 through 3.10.2.9 when more than one of the operable excore channels are indicating the axial flux difference to be outside a limit.

3.10.2.5 Except during physics tests, and except as modified by 3.10.2.6 through 3.10.2.9 below, the indicated axial flux difference shall be maintained within the applicable target band about the target flux difference (defines the target band on axial flux difference).

3.10.2.6 At a power level greater than 90 percent or 0.9 x APL** (whichever is less) of rated power, if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band immediately or reactor power shall be reduced to a level no greater than 90 percent or 0.9 x APL (whichever is less) of rated power.

3.10.2.7 At a power level between 50 percent and 90 percent or 0.9 x APL (whichever is less) of rated power,

* During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

** APL is the Allowable Power Level defined in Specification 3.10.2.2.2.

a. The indicated axial flux difference may deviate from its target band for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference does not exceed the limits shown in Figure 3.10-5. If the cumulative time exceeds one hour, then the reactor power shall be reduced immediately to no greater than 50 percent of rated power and the high neutron flux setpoint reduced to no greater than 55 percent of rated power.

b. A power increase to a level greater than 90 percent or 0.9 x APL (whichever is less) of rated power is contingent upon the indicated axial flux difference being within its target band.

3.10.2.8 At a power level no greater than 50 percent of rated power

a. The indicated axial flux difference may deviate from its target band.

b. A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24-hour period. One-half of the time the indicated axial flux difference is out of its target band up to 50 percent of rated power is to be counted as contributing to the one-hour cumulative maximum the flux difference may deviate from its target band at a power level less than or equal to 90 percent or 0.9 x APL (whichever is less) of rated power.

3.10.2.9 Calibration of the excore detectors will be performed at a power level no greater than 90% or 0.9 x APL (whichever is less) of rated power. The indicated axial flux difference may deviate from its target band during the calibration provided the flux difference does not exceed the limits shown in Figure 3.10-5.

3.10.2.10 Alarms shall normally be used to indicate non-conformance with the flux difference requirement of 3.10.2.6 or the flux difference-time requirement of 3.10.2.7.a. If the alarms are temporarily out of service, the axial flux difference shall be logged, and conformance with the limits assessed, every hour for the first 24 hours, and half-hourly thereafter.

3.10.2.11 The axial flux difference target band about the target axial flux difference shall be determined in conjunction with the measurement of $F_Q(Z)$ as specified in 3.10.2.1.1. The allowable values of the target band are shown in Figure 3.10-4. Redefinition of the target band from more restrictive to less restrictive ranges between determinations of the target axial flux difference is allowed when appropriate redefinitions of APL are made. Redefinition of the target band from less restrictive to more restrictive ranges is allowed only in conjunction with the determination of a new target axial flux difference.

3.10.3 Quadrant Power Tilt Limits

3.10.3.1 Except for physics tests and during power increases below 50 percent of rated power, whenever the indicated quadrant power tilt ratio exceeds 1.02, the tilt condition shall be eliminated within two hours or the following actions shall be taken:

a. Restrict core power level and reset the power range high flux setpoint to be less two percent of rated values for every percent of indicated power tilt ratio exceeding 1.0, and

b. If the tilt condition is not eliminated after 24 hours, the power range high flux setpoint shall be reset to 55 percent of rated power. Subsequent reactor operation would be permitted up to 50 percent of rated power for the purpose of measurement and testing to identify the cause of the tilt condition.

3.10.3.2 Except for low power physics tests, if the indicated quadrant tilt exceeds 1.09 and there is simultaneous indication of a misaligned rod:

a. The core power level shall be reduced by 2 percent of rated values for every 1 percent of indicated power tilt exceeding 1.0, and

b. If the tilt condition is not eliminated within two hours, the reactor shall be brought to a hot shutdown condition.

c. After correction of the misaligned rod, reactor operation will be permitted to 50 percent of rated power until the indicated quadrant tilt falls below 1.09.

3.10.3.3 If the indicated quadrant tilt exceeds 1.09 and there is not a simultaneous indication of rod misalignment, except as stated in Specification 3.10.3.2.c, the reactor shall immediately be brought to a hot shutdown condition.

equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

Two criteria have been chosen as a design basis for fuel performance related to fission gas release, pellet temperature, and cladding mechanical properties. First, the peak value of linear power density must not exceed 21.1 kW/ft. Second, the minimum DNBR in the core must not be less than 1.17 in normal operation or in short term transients.

In addition to the above, the initial steady state conditions for the peak linear power for a Loss-of-Coolant Accident must not exceed the values assumed in the accident evaluation. This limit is required in order for the maximum clad temperature to remain below that established by the ECCS Acceptance Criteria. To aid in specifying the limits on power distribution the following hot channel factors are defined.

a. F_Q, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

b. F_Q^N, Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

c. F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface

e. Axial power distribution control procedures, which are given in terms of axial flux difference within the target band about the target flux difference, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of axial offset which is defined as the difference in power between the top and bottom halves of the core.

For operation at a fraction P of full power, the design limits are met, provided the limits of Specification 3.10.2.1 are not exceeded.

The permitted relaxation in $F^N_{\Delta H}$ with reduced power allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions a through e are observed, these hot channel factors limits are met.

The procedures for axial power distribution control referred to above include operator control of flux difference to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a value which corresponds to the full power target flux difference established in conjunction with incore power distribution measurements. The target flux difference varies with power level.

The target value of the flux difference is determined at equilibrium xenon conditions. The control rods must be positioned in accordance with their insertion limits. This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and the specified deviation of ΔI is permitted from the indicated reference value. The periodic updating of the target flux difference is necessary to reflect the impacts of core burnup on power distribution.

Strict control of the flux difference is not possible during certain physics tests, control rod exercises, or during the required periodic excore calibration which require larger flux differences than permitted. Therefore, the specifications on power distribution are not applicable during physics tests, control rod exercises, or excore calibrations; this is acceptable due to the extremely low probability of a significant accident occurring during these operations. Excore calibration includes that period of time necessary to return to equilibrium operating conditions. In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band; however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequence of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the allowable range shown in Figure 3.10-5 for 90 percent or 0.9 x APL (whichever is less). Therefore, while the deviation exists, the power level is limited to 90 percent or 0.9 x APL (whichever is less) of rated power or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled with the target band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent of rated power is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the limits is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the chemical volume control system to position the full length control rods to produce the required indication flux difference.

An upper bound envelope of peaking factors has been determined from extensive analysis considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10.2 The specifications on power distribution control ensure that xenon distributions are not developed, which at a later time, could cause greater local power peaking even though the flux difference is then within limits. The results of a Loss-of-Coolant Accident analysis based on this upper bound envelope indicate that a peak clad temperature would not exceed the 2200°F limit. The nuclear analyses of credible power shapes consistent with the power distribution control procedures have shown that the F_Q^T limit is not exceeded.

For transient events, the core is protected from exceeding 21.1 kw/ft locally, and from going below a minimum of DNBR of 1.17 by automatic protection on power, flux difference, pressure and temperature.

Measurements of the hot channel factors are required as part of startup physics tests and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors.

In the specified limit of F_Q^N there is a 5 percent allowance for uncertainties(5) which means that normal operation of the core within the defined conditions and procedures is expected to result in a measured F_Q^N 5 percent less than the limit, for example, at rated power even on a worst case basis. When a measurement is taken, experimental error must be allowed for, and 5 percent is the appropriate allowance for a full core representative map taken with the movable incore detector flux mapping system.

In the specified limit of $F_{\Delta H}^N$, there is an 8 percent allowance for design prediction uncertainties, which means that normal operation of the core is expected to result in $F_{\Delta H}^N$ at least 8 percent less than the limit at rated power. The uncertainty to be associated with a measurement of $F_{\Delta H}^N$ by the movable incore system, on the other hand, is 4 percent, which means that the normal operation of the core shall result in a measured $F_{\Delta H}^N$ at least 4 percent less than the value at rated power. The logic behind the larger design uncertainty in the case is that (a) abnormal perturbation in the radial power shape (e.g., rod misalignment) affects $F_{\Delta H}^N$ in most cases without necessarily affecting F_Q^N, and can limit it to the desired value; (b) while the operator has some control over F_Q^N through F_z^N by motion of control rods, he has no direct control over $F_{\Delta H}^N$, and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests, can be compensated for in F_Q^N by tighter axial control, but compensation for $F_{\Delta H}^N$ is less readily available.

Quadrant power tilts are based upon the following considerations. The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions, measured as part of the startup physics testing, are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions is consistent with the assumptions used in power capability analyses. It is not intended that extended reactor operation would continue with a power tilt condition which exceeds the radial power asymmetry considered in the power capability analysis.

During normal plant startup, quadrant power tilt ratio may exceed 1.02 due to instrumentation instabilities as a result of rodded configurations and low excore detector signal levels below 50 percent of full power. Sustained power operation below 50 percent of full power would require a renormalization of the calculational methods for determining power tilt to compensate for change in signal levels once equilibrium conditions are met.

The two-hour time interval in this specification is considered ample to identify a dropped or misaligned rod and complete realignment procedures to eliminate the tilt. In the event that the tilt conditions cannot be eliminated within the two-hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full core physics map utilizing the movable detector system. For a tilt condition $\leq$ 1.09 an additional 22 hours' time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of two percent for each one percent of indicated tilt is required. Physics measurements have indicated that the core radial power peaking would not exceed a two-to-one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment.

In the event the tilt condition of 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to the range required for low power physics testing. To avoid reset of a large number of protection setpoints, the power range nuclear instrumentation would be reset to cause an automatic reactor trip at 55 percent of allowed power. A reactor trip at this power has been selected to prevent, with margin, exceeding core safety limits even with a nine percent tilt condition. If a tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor power shall be brought to a hot shutdown condition for investigation.

However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (2 percent for each one percent the tilt ratio exceeds 1.0) for the two-hour period necessary to correct the rod misalignment.

The specified rod drop time is consistent with safety analyses that have been performed.(1)

An inoperable rod imposes additional demands on the operator. The permissible number of inoperable control rods is limited to one in order to limit the magnitude of the operating burden, but such a failure would not prevent dropping of the operable rods upon reactor trip.

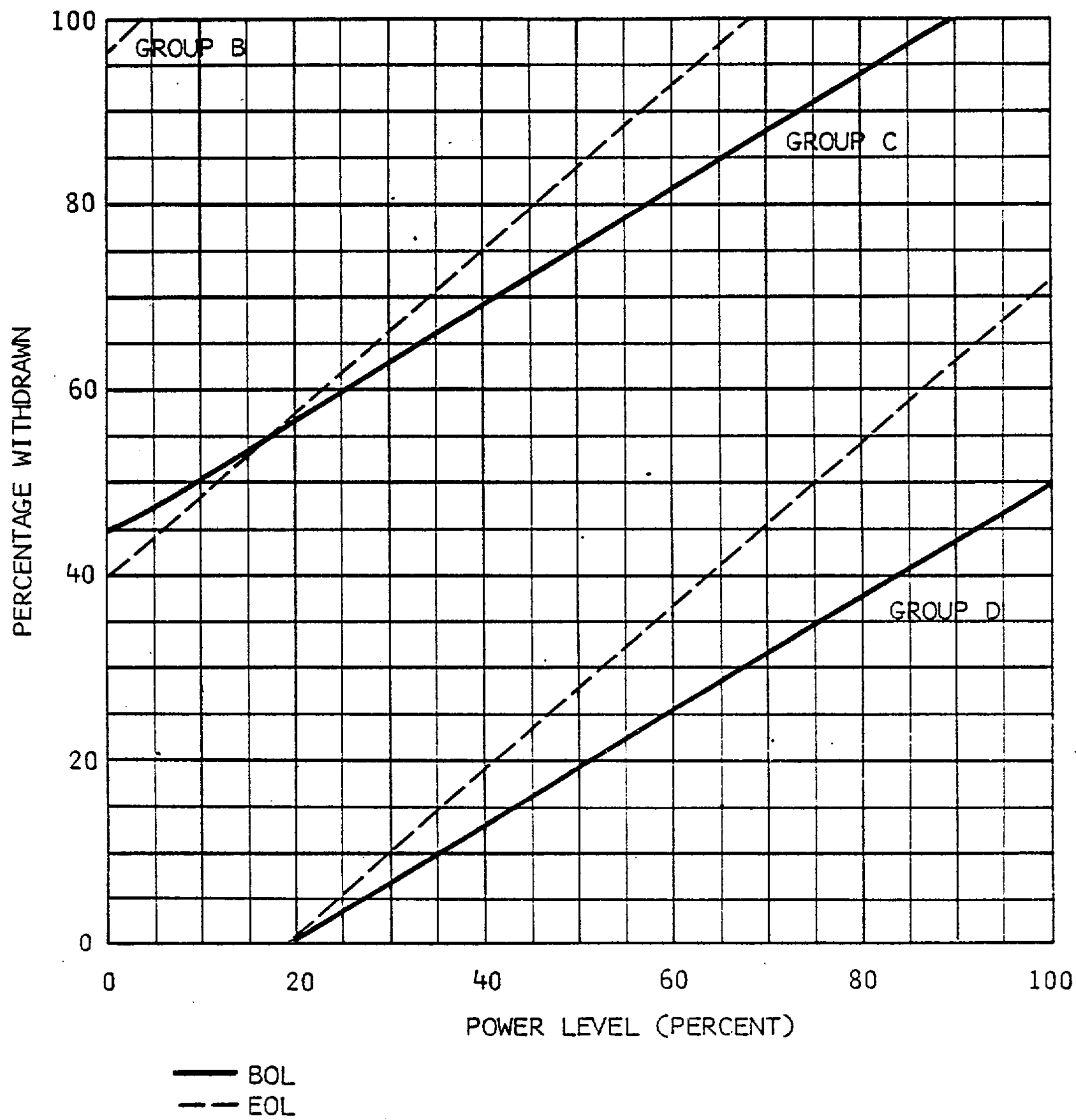
Normal reactor operation causes significant pellet cracking and fragmentation. Consequently, handling of irradiated fuel assemblies can result in relocation of these fragments against the cladding. Calculations show that high cladding stresses can occur if the reactor power increase is rapid during the subsequent startup.

The 72-hour period allows for stress relaxation of the clad before the ramp rate requirement is removed, thereby reducing the potential harmful effects of possible pellet or fragment relocation.

The 3 percent limit is imposed to minimize the effects of adverse cladding stresses resulting from part power operation for extended periods of time. The time period of 30 days is based upon the successful power ramp demonstrations performed on Zircaloy clad fuel in operating reactors, resulting in no cladding failures.

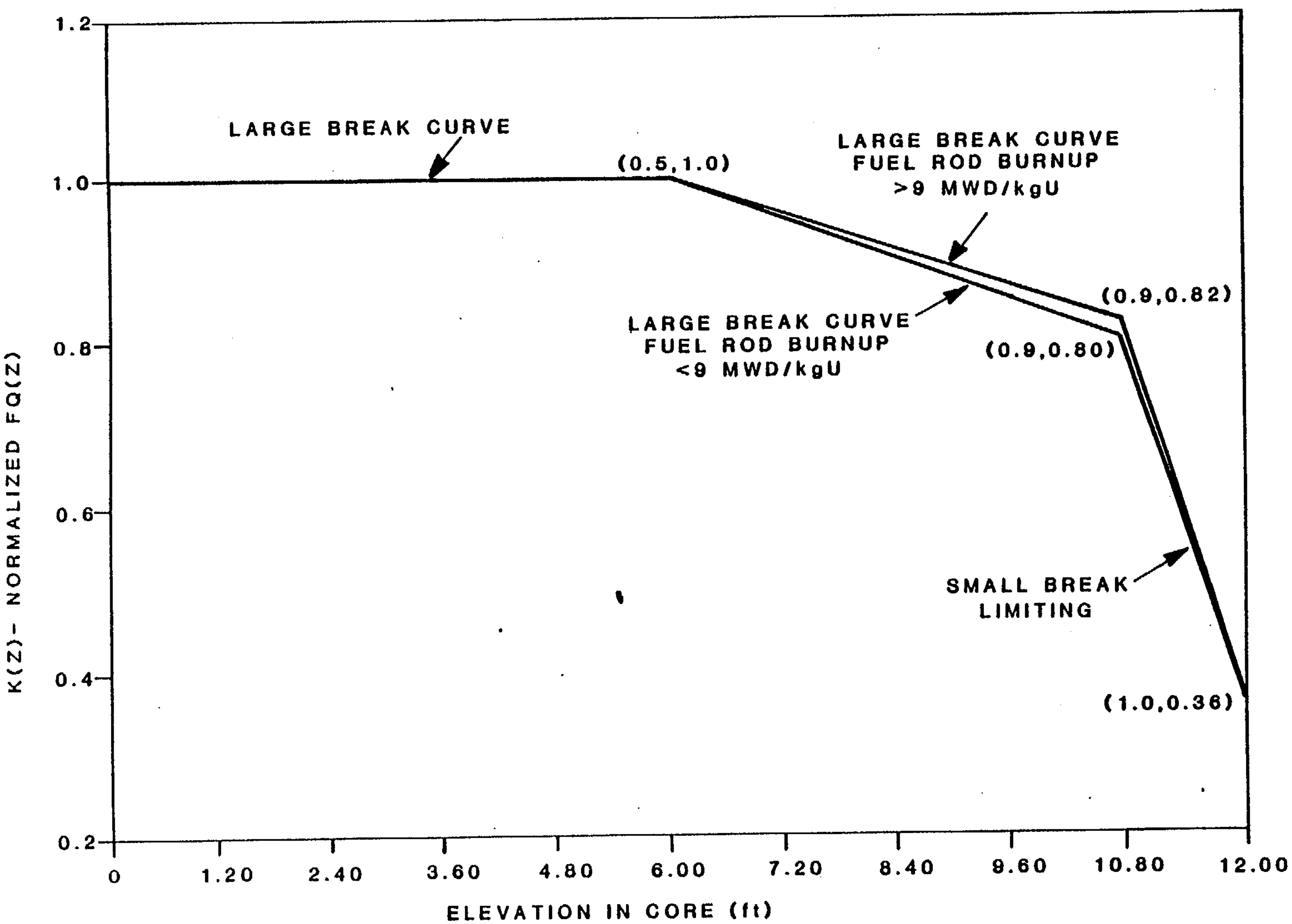
References

(1) FSAR Section 15.0
(2) FSAR Section 7.7
(3) FSAR Section 15.4
(4) FSAR Section 15.4
(5) FSAR Section 15



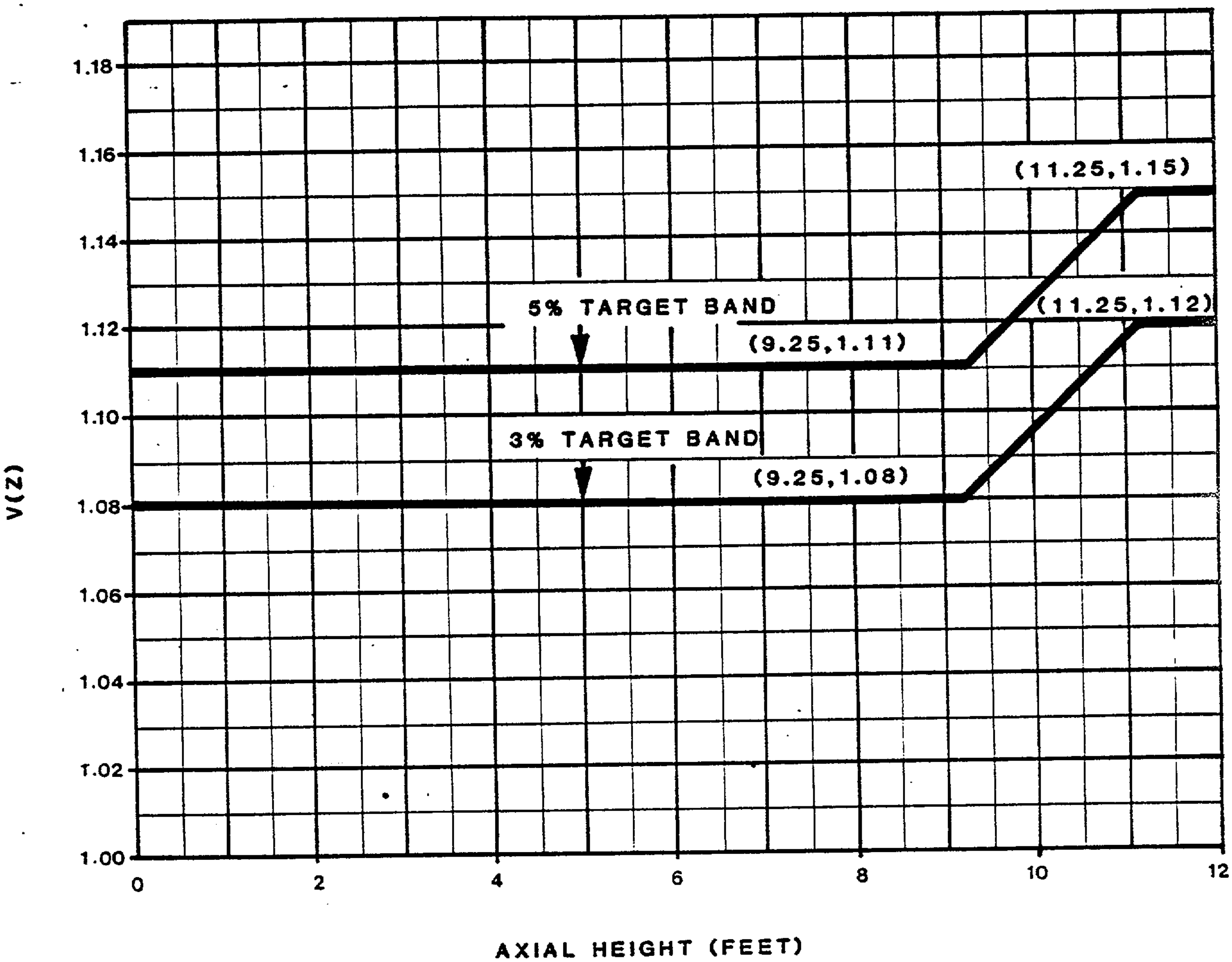
Amendment No.

H.B. ROBINSON Unit #2 Carolina Power & Light Company **Technical Specifications**	**Control Group Insertion Limits For Three Loop Operation**	FIGURE **3.10-1**



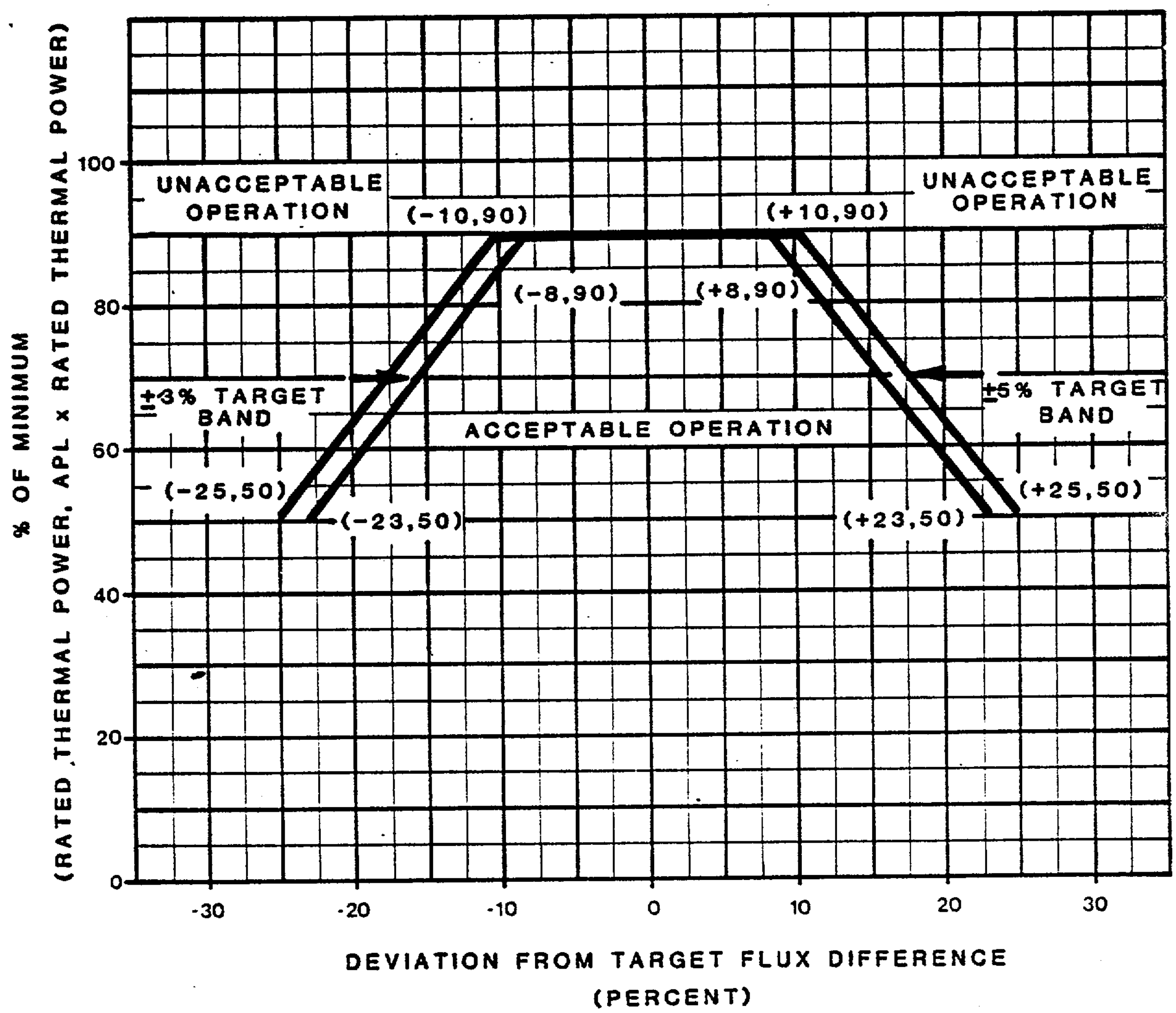
NORMALIZED AXIAL DEPENDENCE FACTOR FOR $F_Q^T = 2.32$
VERSUS ELEVATION FOR $F_{\Delta H} = 1.65$

Figure 3.10-3



V(Z) AS A FUNCTION OF CORE HEIGHT

Figure 3.10-4



ALLOWABLE DEVIATION FROM TARGET FLUX DIFFERENCE

Figure 3.10-5

3.11 MOVABLE IN-CORE INSTRUMENTATION

Applicability

Applies to the operability of the movable detector instrumentation system.

Objective

To specify functional requirements on the use of the in-core instrumentation systems, for the calibration of the excore symmetrical offset detection system.

Specification

3.11.1 A minimum of 16 total accessible thimbles and at least 2 per quadrant and sufficient movable in-core detectors shall be operable during recalibration of the excore symmetrical offset detection system.

3.11.2 Power shall be limited to 90% of rated power if recalibration requirements for the excore symmetrical offset detection system identified in Table 4.1-1 are not met.

Basis

The Movable In-Core Instrumentation System[1] has five drives, five detectors, and 48 thimbles in the core. Each detector can be routed to twenty or more thimbles. Consequently, the full system has a great deal more capability than would be needed for the calibration of the excore detectors.

To calibrate the excore detector system, it is only necessary that the Movable In-Core System be used to determine the gross power distribution in the core as indicated by the power balance between the top and bottom halves of the core.

After the excore system is calibrated initially, recalibration is needed only infrequently to compensate for changes in the core, due, for example, to fuel depletion, and for changes in the detectors.

If the recalibration is not performed, the mandated power reduction assures safe operation of the reactor since it sill compensate for an error of 10% in the excore protection system. Experience at the Beznau No. 1 and R. E. Ginna plants has shown that drift due to the core on instrument channels is very slight. Thus, limiting the operating levels to 90% of the rated power is very conservative.

Reference

(1) FSAR Section 7.7.1.5

4.11 REACTOR CORE

Applicability

Applies to surveillance of the reactor core.

Objective

To ensure the integrity of the fuel cladding.

Specification

4.11.1 APDMS Operation

4.11.1.1 Prior to establishing normal operation with APDMS, at least six maps will be taken to determine applicable values of $\overline{R}$ and σ for surveillance thimbles.

4.11.1.2 Plant operation up to rated power shall be permitted for the purposes of obtaining the initial maps of Specification 4.11.1.1, provided the APDMS is operational and hot channel factors are shown to be below the limiting values set forth in Specification 3.10.2. Suitably conservative values of $\overline{R}$ and σ shall be derived from maps previously run during the current fuel cycle for use in the APDMS system during this initial period.

4.11.1.3 Subsequent updates of $\overline{R}$ and σ shall employ the last six maps in accordance with Specification 4.11.1.1.

4.11.1.4 Each power distribution map will be based on flux traverses obtained from 36 or more of the 48 monitoring channels.

4.11.2 Except during physics tests and EXCORE calibrations, axial surveillance of F(Z)S(Z) shall consist of traverses with the movable incore detectors in appropriate pairs of detector paths, taken every eight hours, or a frequency of approximately 0, 10, 30, 60, 120, 180, 240, 360, and 480 minutes following accumulated control rod motion in any one direction of five steps or more, exclusive of control rod movement within 15 steps from the top of the core. From the traverses, determination of F(Z)S(Z) shall be made and shown to result in a value less than the limiting value specified in 3.10.2. If the APDMS is out of service, reactor operation above APL of rated power can be continued for fourteen equivalent full power days provided that traverses are taken manually at equivalent frequencies, and a log of accumulated rod motion and time of manual traverses is kept.

4.11.3 The following criteria will be used for selecting the channels for measuring F(Z)S(Z):

a. The channel is not acceptable if it contains a control rod allowed by the insertion limits at power levels requiring APDMS.

b. For the latest full core power map, i, channels, j, are acceptable if:

$$\frac{R_{ij} - \overline{R}_j}{\overline{R}_j} \leq 2\sigma_j$$

<u>Basis</u>

The $\overline{R}$ technique provides a means for using many of the monitoring thimbles to determine $F_Q(Z)$ without fully mapping the core. Frequent core maps assure that appropriate values of $\overline{R}$ are being used for each thimble.

Upon return to power following a refueling outage or other situation where establishment of normal APDMS operation is required, power operation above APL of rated power is desirable to establish hot channel factors at full power.

By using maps that have been previously obtained during the power ascension and deriving conservative values of $\overline{R}$ and σ from these maps for use in the APDMS, operation of the plant within the peaking factor limitations can be ensured.

If the APDMS is out of service, adequate monitoring of the core power distribution can be maintained for a limited period of time by manual actuation of the flux mapping system and calculation of the values of F(Z)S(Z).

5.3 REACTOR

5.3.1 Reactor Core

5.3.1.1 The reactor core contains approximately 68 metric tons of uranium in the form of natural or slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods which are all pre-pressurized. The reactor core is made up of 157 fuel assemblies. Each fuel assembly contains 204 fuel rod locations occupied by rods consisting of natural or slightly enriched uranium pellets, solid inert materials, or a combination of the aforementioned.(1)

5.3.1.2 Deleted.

5.3.1.3 Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 3.9 weight percent of U-235.

5.3.1.4 Deleted.

5.3.1.5 There are 45 full-length RCC assemblies in the reactor core. The full-length RCC assemblies contain 144 inch segments of silver-indium-cadmium alloy clad with the stainless steel.(2)

5.3.1.6 Up to 10 grams of enriched fissionable material may be used either in the core, or available on the plant site, in the form of fabricated neutron flux detectors for the purposes of monitoring core neutron flux.

5.3.2 Reactor Coolant System

5.3.2.1 The design of the Reactor Coolant System complies with the Code requirements.(3)

5.3.2.2 All piping, components and supporting structures of the Reactor Coolant System are designed to Class I requirements.

5.3.2.3 The nominal liquid volume of the Reactor Coolant System, at rated operating conditions, is 9343 cubic feet.[4]

<u>References</u>

(1) FSAR Section 4.2.1
(2) FSAR Section 4.2.2
(3) FSAR Table 3.2.2-1
(4) FSAR Table 5.1.0-1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

1.0 INTRODUCTION

By applications dated July 23, 1984 and August 1, 1984 and supplemental information dated August 8, and 20(2), 1984; September 7(2), 1984; and October 4, 12, and 22, 1984, Carolina Power and Light Company (the licensee) requested amendment to Facility Operating License No. DPR-23 for the H. B. Robinson steam Electric Plant, Unit No. 2 (the facility) to permit operation for Cycle 10 at full power (2300 Mwt). Prior to the July 23, 1984 amendment request, documents in support of the forthcoming core reload were submitted by letter dated October 5, 1983. The supplementing letters provided information as follows:

1. August 8 and 20(2), 1984 provided confirmatory analysis in accordance with the July 23, 1984 application letter.
2. September 7, 1984 (84-366) provided Technical Specification (TS) changes resulting from our review due to clarifications, error corrections, and consistency within the TS. No significant changes were made.

3. September 7, 1984 (84-400) resubmitted a K(z) curve for the Technical Specifications. Confirmatory analysis, since the original (July 23, 1984) submittal, necessitated minor revisions to the curve. This was a minor revision to an already submitted document resulting from a standard analytical process, therefore, it was not a significant change or a new submittal.

The amendment consists of:

a. Appendix A Technical Specification (TS) changes resulting from the analysis required for the Cycle 10 core reload,
b. Appendix A Technical Specification (TS) changes to allow the performance of certain control rod drive evolutions when containment integrity in not intact, and
c. Appendix A Technical Specification (TS) change of an administrative nature such as deleting references to N-1 loop operation and clarifications within the TS and the FSAR.

The H. B. Robinson Unit 2 (HBR-2) plant has operated at reduced power since Cycle 8 in order to minimize degradation of the steam generators. The steam generators have been replaced during the current outage and HBR-2 intends to operate at full power (2300 Mwt) during Cycle 10 and subsequent cycles. HBR-2 will install, during Cycle 10 refueling, Part Length Shielding Assemblies (PSLA's). The PSLA's are designed to reduce the fast neutron flux to the pressure vessel weld seams by a factor greater than 7, thus, preventing the vessel from reaching the pressurized thermal shock screening criteria prior to expiration of the current operating license.

In order to accommodate the rated power level, power distribution and the concurrent use of PLSAs the licensee requested thermal margin relief for Cycle 10 (and subsequent cycles) i.e., $F_Q = 2.32$ and $F^T_{\Delta H} = 1.65$ and corresponding revision of certain reactor protection system setpoints. The peak discharge fuel assembly exposure is estimated at 44,000 MWD/MTU.

In support of these changes for Cycle 10 operation, the licensee submitted:

1. Document XN-NF-84-74, "Plant Transient Analysis For H. B. Robinson Unit 2 At 2300 MWt With Increased $F_{\Delta H}$." The document presents the analysis of the SRP Chapter 15 transient and accident events and,
2. A revised LOCA Analysis,
3. A Cycle 10 core reload report including Technical Specifications modifications.

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 Chapter 15 analysis methodology. As a consequence, Exxon Nuclear Company (ENC) developed a stand-alone methodology which is at present under staff review.

The licensee has requested and provided justification to defer submittal of the steam line break event until January 31, 1985. The request was made to provide Exxon Nuclear Company time to develop an acceptable methodology for analyzing steam line break events. Further details concerning this request and the staff's independent analysis is contained in Attachment I of this SER.

2.0 EVALUATION

Reload Fuel Design

The H. B. Robinson-2 core consists of 157 fuel assemblies, each having a 15x15 fuel rod array. There are 204 fuel rods, 20 control rod guide tubes and one instrument guide tube. Each fuel assembly has seven zircaloy spacers, inconel springs and zircaloy cladding. There are 65 fresh assemblies supplied by ENC including 12 PLSAs located at the core flats and especially designed to reduce the fast neutron flux to the lower girth weld seam of the pressure vessel. The lower 42 inches of the PLSAs contain a 304 stainless steel column instead of fuel but otherwise are identical in design with all other assemblies. The new assemblies of this reload are axially blanketed, except for the lower part of the PLSAs.

Fuel exposure for Cycle 10 has been based on a Cycle 9 exposure of 10,637 MWD/MTU and is estimated to be 10,820 MWD/MTU (312 EFPDs) with an estimated peak exposure of 34,705 MWD/MTU. The basic Exxon Nuclear design and design methods and the extended burnup mechanical design are described in Reference 2 and 3, respectively. The mechanical design for the PLSAs is covered in Reference 4.

The 65 new assemblies (designated XN-7) include 36 which contain gadolinia bearing pins. The Cycle 10 fuel assembly design parameters are listed in Table 4.1 of Ref. 10.

Fuel Mechanical Design

The mechanical design of the new reload assemblies is identical with that of previous reload assemblies with the following exceptions: (a) they contain a natural uranium blanket (6 inches in length) at the top and bottom, and (b) the column insulator discs are no longer used. The PLSA design is similarly identical to previous reload designs with the exception as noted that the lower 42 inches of fuel is replaced with 304 SS. Because of the presence of the stainless steel the following aspects of the mechanical -thermal design needed confirmation: (a) thermal expansion effects for the stainless steel, (b) loss in assembly hold down capability due to the lower weight of the rods, (c) sensitivity to irradiation induced bowing and (d) the seismic stability of the lower weight assemblies. These issues have been discussed and the PLSA mechanical design has been approved. (See topical report XN-NF-83-71, Reference 4).

Fuel Thermal-Hydraulic Design

The thermal-hydraulic design of the Cycle 10 assemblies is identical to that of previous reloads, assuring compatibility. The original Exxon analysis was documented in Ref. 7. The philosophy followed in the analyses was to choose a bounding fuel assembly power distribution at an exposure which has the worst radial peaking. Similarly to assure that the lowest value of the DNBR was accounted for, a bounding local power distribution was used with the maximum radially peaked assembly.

The analysis used a 5% lower plenum factor, 4.5% core flow bypass, a low estimate of the total reactor coolant system (RCS) flow rate, 6% steam generator tube plugging and an additional 3% reduction to account for flow measurement uncertainty. The results of the thermal-hydraulic analysis (which are given in ref. 7) are used as bases for the analyses of the anticipated operational occurrences.

Thermal-Hydraulic Analysis

A bounding fuel assembly power distribution with the limiting values of $F_{\Delta H} = 1.65$ and $F_Q = 2.32$ were utilized in order to assure the computation of the lowest DNBR results. The primary flow used in the safety analysis is based on an assumed 6% steam generator tube plugging and a 3% plant calorimetric flow measurement uncertainty. This minimum primary flow will be assured by adjusting the low flow trip set points after the full power plant calorimetric is performed. A Technical Specification change will be effected if required. Core flow is based on a 4.5% core flow bypass and a 5% lower plenum inlet flow maldistribution factor In this manner, the analysis considered all fuel types in a bounding manner and is acceptable.

Fuel Rod Bowing

The core reload for Cycle 10 consists of Exxon fuel assemblies for which the hydraulic design is similar to the existing fuel. According to the approved topical report XN-NF-75-32 (reference 19) Exxon fuel must be reviewed for possible rod bowing penalty as a function of burnup. For the H. B. Robinson Cycle 10 fuel design, burnup to 47,000 MWD/MTU are acceptable without penalty. The calculated maximum design assembly burnup is 44,000 MWD/MTU and, therefore, no rod bow penalty needs to be applied.

NUCLEAR DESIGN

Core Characteristics

The H. B. Robinson-2 Cycle 10 is neutronically similar to Cycle 8; however, it differs in major respects from the previous reloads in that it will operate at a power level of 2,300 MWt and it incorporates 12 PLSAs to minimize fast flux irradiation of the lower girth weld seam of the pressure vessel. In addition, the 65 new assemblies have a natural uranium axial blanket and utilize 4.0 w/o gadolinia. The average loading enrichment is 3.08 w/o in U-235 and the maximum is 3.34 w/o U-235. The estimated exposure of Cycle 10 has been based on a Cycle 9 exposure of 10,637 MWD/MTU and it is estimated to be 10,820 MWD/MTU or the equivalent of 312 EFPD. The peak assembly exposure is estimated to be 34,705 MWD/MTU.

Power Distribution

At full power (2,300 MWt) and equilibrium xenon conditions (100 MWD/MTU), the calculated $F_{\Delta H}$ = 1.48 and the peak F_Q = 2.18, including a 4% and 5% measurement uncertainty respectively. In addition F_Q includes a 3% engineering factor and an 11% allowance for operation with the Power Distribution Control-II (PDC-II) within ±5% target bands. For both, i.e., $F_{\Delta H}$ and F_Q, the maximum value will occur at 5,000 MWD/MTU and their respective calculated values are 1.56 and 2.18. These values were calculated with approved methods (Ref. 5) and are within the limits of the Technical Specifications.

Reactivity Coefficients and Control Requirements

The H. B. Robinson-2, isothermal temperature coefficient at Hot Zero Power (HZP) and Hot Full Power (HFP) conditions at beginning of cycle (BOC) and end of cycle (EOC) are shown in Table 6.1 of Reference 10. The same table also compares the corresponding values of cycle 8 and cycle 10 for the boron worth (ppm/10^3 pcm) at HZP and HFP, the prompt neutron lifetime (μsec) and the delayed neturon fraction. The values for the two cycles are nearly identical. The isothermal temperature coefficient at HFP and BOC10 is estimated to be -5.1 pcm/°F at a critical boron concentration of 1,002 ppm, and at HZP, BOC10 is -.7 pcm/°F at a critical boron concentration of 1,134 ppm. At both extremes the value is negative. The moderator temperature coefficient at HZP condition for the BOC10 is estimated at +1.0 pcm/°F, the value used in the transient analysis and required in the Technical Specifications is +5.0 pcm/°F (Ref. 8, 10). Similarly at HFP the estimated value of the moderator temperature coefficients is -3.8 pcm/°F, well below the required value of 0.0 pcm/°F.

Control rod worths and shutdown margins have been calculated for cycle 10 and are summarized in Table 6.2 of Ref. 7. Control rod worths for cycle-10 are slightly higher than the corresponding values for cycle 8 as expected. For cycle 10 the power distribution is higher in the core interior due to the PLSAs in the core flats. The required shutdown margin in the two cycles is assumed the same, i.e., 1,770 pcm, thus the excess shutdown margin is 461 pcm for cycle 10 vs 565 for cycle 8 at EOC. The corresponding values for BOC are 1,911 vs 1,554. Hence, the shutdown margins for both cycles are comparable. The control rod groupings shall remain the same.

Power distribution control is to be effected following the Exxon procedures known as Power Distribution Control, Phase II (PDC-II, Ref. 7). The topical report and two supplements have been approved by the staff. The analytical methodology for the neutronic calculations (Ref. 5) has also been approved. The reference neutronic analysis was performed using XTG (Ref. 11), a two-group, three dimensional coarse mesh code. The cycle power distribution is calculated using PDQ/HARMONY. (Ref. 12, 13)

Because the results discussed above have been obtained using approved methods, and were used in the safety analyses with appropriate calculational uncertainties and are included in the Technical Specifications, we find these results to be acceptable.

SAFETY ANALYSES

For the new set of parameters, namely the increased power level, $F_{\Delta H}$ and F_Q the values have been reviewed to determine which ones influence the results of the transient and accident analyses. It was determined that the following events needed to be reanalyzed:

- Excess Load
- Scram shutdown margin
- Steam generator tube rupture
- Loss of load
- Loss of normal feedwater
- 3-pump coastdown
- Locked rotor
- Uncontrolled rod withdrawal (subcritical or low power)
- Uncontrolled rod withdrawal (at power)
- RCCA misalignment
- CVCS malfunctions with decreasing boron concentration
    - Refueling
    - Startup
- RCCA ejection
- LOCA fuel damage limits

The nominal plant rated operating conditions and the nominal core and fuel design parameters used in the accident analyses are listed in tables 15.0.2-1 and 15.0.2-2 of Ref. 8. The axial power distribution used for transients which do not require power redistribution is shown in Figure 15.0.3-1 of the same reference. The nuclear enthalpy rise factor is 1.65, the axial peaking factor is 1.65, the total heat flux peaking factor is 2.32, and the fraction of power generated in the fuel is .974. Operating parameter ranges and reactivity coefficients used in the analyses are shown in Tables 15.0.4-1 and 15.0.5-1 of ref. 8.

A discussion of the event analyses vs the acceptance criteria will follow.

Excess Load

Excess load event manifests itself whenever there is a rapid increase in the heat removal from the reactor coolant without a corresponding increase in the reactor power. This power-energy removal mismatch results in a decrease of the reactor coolant temperture and pressure. Hence, when the moderator temperature reactivity coefficient is negative an increase in power may occur. If there is a positive temperature coefficient the power will decrease and will not produce a challenge to the acceptance criteria.

This event constitutes a challenge to the Specified Acceptable Fuel Design Limits (SAFDL). The conditions for the minimum SAFDL margin are full power and maximum feedback at EOC. The event initiator was a 10% step increase in turbine steam flow. The reactor control is assumed to be in automatic. The secondary and primary pressures initially fall and the primary temperature will fall resulting in reactivity insertion from the moderator coefficient and a decrease of the pressurizer level. Additional reactivity will be inserted from the control rods which initiate withdrawal to increase power responding to the load demand. In about 80 sec a new steady state will be reached. The minimum DNBR computed was 1.33 which is above the limit of 1.17. Hence, this transient is acceptable. For the analysis the PTSPWR2 code was used to provide input to XCOBRA-IIIC. This methodology is under review by the staff. However, the review has progressed to the point where the portions pertinent to this application have been found to be acceptable.

Scram Shutdown Margin

The particular quantity of interest here is the shutdown margin after trip. This is part of the inadvertent opening of a steam generator or reload safety valve. This event is most limiting at the end of cycle. There is adequate shutdown margin at BOC10. The required margin is 1,000 pcm and the excess margin is 1,911 pcm. (Reference 10). The required analysis will be performed and be submitted for review during CY85 i.e., before the EOC10. This is acceptable.

Steam Generator Tube Rupture

A rupture of a steam generator tube will release primary coolant into the lower pressure secondary. This event is similar (and bounded by) the inadvertent opening of a pressu izer PORV and was reported in reference 10. In that analysis a single valve was assumed to have failed open at full power. The maximum relief capacity was set at 288,000 lb/hr at the PORV pressure setting. The initial conditions included 102% of rated power (2,436 psia), a primary pressure of 2,220 psia, $F_{\Delta H}$ = 1.65 and F_Q = 2.32. No credit was taken for lowering of the primary coolant pressure. The result of this calculation was a MDNBR value greater than the limit of 1.17 allowed in the XNB DNB correlation. Thus, an assumption of no fuel failure is acceptable for the evaluation of the radiological consequences of the transient, which were reviewed and were also found to be acceptable.

Loss of Load

The loss of load event is an undercooling transient that results from station disconnection from the grid, turbine trip or electrical generator malfunctions. Following the loss of load the main steam stop valve closes causing a large mismatch between reactor power output and heat removal which in turn causes a secondary temperature and pressure increase. As the primary to secondary ΔT decreases the primary to secondary heat transfer decreases and the primary temperature and pressure will rise. Assuming that the reactor does not trip after the turbine trip the high primary pressure will trip the reactor and open the primary safety valves. Energy from the system will also be removed through the steam generator relief valves. The primary challenge of this transient is to the primary system overpressurization acceptance criterion (peak pressure $\leq$ 110% of the design value) and the secondary challenge is to the SAFDL because of the increasing primary temperature.

The purposes of the analyses for this transient were to maximize the overpressurization and the SAFDL challenge, hence, the input parameters were biased to maximize the overpressure and minimize DNBR respectively.

The analyses were carried out with the PTSPWR2 and the XCOBRA-IIIC methodology. The results indicate that for the maximum pressure case (from 102% of full power) a 27°F rise in the cold leg temperature occurs at 14.5 sec into the transient. The reactor trips at 6.80 sec (on high pressure) and the safety valves open at 6.83 sec. The maximum pressure at pump discharge was computed at 2,661 psia which is below the 110% of the 2,500 psia (i.e., 2,750 psia) of the design pressure. For the MDNBR case the value computed is 1.19 which is above the 1.17 limit. Because these values are within the NRC acceptance criteria of SRP section 15.2.1 we find these analyses acceptable.

Loss of Normal Feedwater

Loss of feedwater could result from loss of feedwater pumps, feed control valve malfunction, loss of offsite power etc. Loss of feedwater flow results in a decrease of steam generator water level, decrease in secondary system heat removal capability and increase in primary pressure and temperature. The reactor trips due to the steam generator low-low water level signal. The objective of the analysis is to demonstrate the adequacy of the steam generator inventory and relief capacity and proper setpoint of the safety valves, to prevent primary pressure from exceeding 110% of the design pressure of 2,500 psia. In the analyses, the single failure event is the failure of the steam driven auxiliary feedwater pump to start and it is assumed that the primary pumps are on or tripped. (two different cases) In addition the turbine trips with simultaneous closure of the turbine stop valve, the main feedwater valves are ramped closed and the diesel generator is initiated with specified delay. The analyses were carried out with the Exxon code SLOTRAX. (Ref. 14)

The results indicate that the primary pump off case challenges the primary pressure limit and the pumps on case the minimum steam generator inventory criterion. The pressure was estimated to reach the primary relief setpoint in 1.5 sec with a required relief valve rate of 215 lb/sec. The rated rate is 240 lb/sec, therefore, pressure will not rise above the valve setpoint. For the pumps-off case the minimum steam generator inventory is 75% of the initial amount. The SAFDLs are bounded by the loss of flow event. We find that the results of the analyses meet the SRP Section 15.2.7 criteria and they are acceptable.

Three Pump Coastdown

Loss of all three primary coolant pumps can result from loss of electric power to the pump motors. Following loss of power the pumps will coast down and that is governed by the pump flywheel inertia. Loss of primary coolant when the reactor is at power will result in rapid coolant temperature rise and corresponding reduction in DNB margin or even DNB if the reactor is not tripped promptly. Loss of primary flow will result in temperature and pressure rise, which will be mitigated by the primary system safety valves. This event results in a heatup transient and challenges the 110% of design pressure criteria and the MDNBR criteria. Reactor trip signals are based on signals from pump motor power supply undervoltage or underfrequency and reactor coolant low flow. In the analysis only the low flow trip is taken into account. The transient is governed by the initial overpower DNB margin, rate of flow degradation, low reactor coolant flow trip setpoint, available shutdown reactivity and the moderator temperature coefficient. The objectives of the analyses are to investigate the overpresssurization limit and the MDNBR value. The analysis methodology is based on the PTSPWR2 code providing the input to the XCOBRA-III code. For both cases analyzed it is assumed that reactor control is in manual, the power level is at 2,346 MWt, i.e., rated + 2%, reactor trip setpoint of low flow-3%, moderator temperature coefficient at 1.2 BOC and the pellet to clad heat transfer coefficient at the maximum value.

The results indicate a maximum primary bounding pressure of 2,574 psia which is lower than the 2,750 psia (110% of 2,500 psia) and the MDNBR at 1.21 which is greater than the allowed minimum value of 1.17. These values meet the criteria of SRP Section 15.3.1 and 2 and, therefore, we conclude that the results are acceptable.

Locked Rotor for a Reactor Coolant Pump

This event is caused by an instantaneous seizure of a primary reactor coolant pump rotor, when flow through the affected loop diminishes rapidly. A reactor trip signal on low primary flow will be initiated and the reactor will trip. However, because of the reduced flow the primary temperature and pressure will rise. Following turbine trip the secondary temperature (and pressure) will rise, further reducing primary to secondary heat transfer and further increasing primary temperature and pressure. Primary pressure will be relieved by opening of the safety valves. The rapid rise of the primary temperature will cause a rapid decrease of the DNB and the primary pressure rise will challenge the primary boundary pressure limit. The analysis method is based on PTSPWR2 and the XCOBRA-IIIC codes. The objectives of the analyses were to estimate the peak primary pressure, the extend of fuel failures and that the possible radiological consequences are bounded by the limits of 10 CFR 100. The initial conditions assumed (among others): power at rated +2% (i.e., 2,346 MWt), maximum pellet to clad heat transfer coefficient and manual reactor control. The characteristics of this transient are unique in the sense that the flow will be reversed in the affected loop due to the higher pressure in the reactor vessel and the effective core coolant flow will be about 60% of normal at 40 sec into the transient. Increased temperature will cause increased power to 107.6% of rated due to the assumed positive moderator temperature coefficient. The maximum pressure is reached at 3.51 sec of 2,516 psia which is less than the allowable value 2,750 psia. However the MDNBR will be .90 i.e., less than the allowed by the XNB correlation of 1.17. The estimated fuel failures used in the evaluation of the radiological consequences are 55% of the total number of assemblies. The radiological consequences are bounded by those of the LOCA, and are within the Standard Review Plan guideline limits of 10 CFR 100. The results of the analysis are acceptable.

Uncontrolled Rod Cluster Withdrawal

This event results from an uncontrolled rod bank withdrawal and it can result in a rapid and large reactivity insertion. The maximum insertion rate is determined from the worth of the rod bank. We distinguish three sub-events depending on initial power, i.e., subcritical, low power and full power. The analysis is performed using the PTSPWR2 and XCOBRA-IIIC codes.

(a) Subcritical and Low Power

In this case the malfunction will result in a rapid and large reactivity insertion which will be terminated by the low range setting of the power range flux trip. The reactivity insertion is countered initially (promptly) by the Doppler effect followed by rod insertion. Other trips for this event are: source and intermediate range flux trips, intermediate range rod block and low and high power range trip settings. In the subcritical case the objective of the analysis is to determine the fraction of fuel exceeding the MDNBR limit for input to the radiological consequences evaluation. The conditions of the analysis maximized power and minimized core flow. A peak surface heat flux equivalent to 69% of full power was reached at 16 sec into the transient. The MDNBR value (for the XNB correlation) was 1.26, which is above the 1.17 limit. Since there is no core damage, these results are acceptable.

(b) Power Operation

Power operation for the purposes of this analysis is defined between 10-100% of rated power. In this transient the overpower ΔT is set to protect against MDNBR. The power range reactor trip protects against high power levels. The objective of the analysis is to examine the broad range of reactivity insertion possible and assure of the adequacy of the trip setpoints to meet the acceptance criteria i.e., on primary pressure and MDNBR. The cases analyzed included 10%, 60% and 102% of rated power and negative and positive reactivity feedbacks. The rated power reactivity insertions were found to be bounding with respect to MDNBR for the lower power ratings. From full power with positive moderator temperature coefficient the nuclear high power trip (118% of rated power) is reached in 1.84 sec. and the MDNBR reaches 1.32 at 3.02 sec. The pressure increases to a maximum of 2260 psia. The results of the analysis indicate that neither of the acceptance criteria is violated, hence, the results of this transient are acceptable.

Rod Control Cluster Assembly Misalignment

Analyses for this event (with its subdivisions) have been submitted in Ref. 15, (which is Supplement 1 to Ref. 8).

Rod Control Cluster Assembly (RCCA) misoperation includes, withdrawal of a single full length RCCA, static misalignment of a RCCA, a dropped full length RCCA and a dropped RCCA bank. The conditions to be satisfied for the first are primary pressure, core coolability and radiological consequences for 10 CFR 100. The other three should satisfy primary pressure limit and the MDNBR limit.

The withdrawal of a RCCA at power will insert positive reactivity and cause increased power generation. If the secondary does not respond to the increased power production the temperature and pressure of the primary will increase. A single RCCA withdrawal will cause in addition severe power redistribution in its immediate neighborhood, hence, locally it is possible that some assemblies may experience boiling transition and some fuel failure. The overtemperature ΔT trip will afford the primary protection. This transient is evaluated with the PTSPWR2, XCOBRA-IIIC codes and: power at 102% of nominal, radial peaking factor of 1.27 and conservative values for the moderator temperature and Doppler coefficients. The system pressure is estimated to reach a peak value of 2,275 psia, however, the MDNBR will (locally) reach .64 i.e., below the limiting value of 1.17. The estimated fuel failures to be used for the evaluation of the radiological consequences are estimated to be 7.8% of the total number of the fuel assemblies. The radiological consequences of this evaluation are acceptable.

For the other three cases i.e., static misalignment, dropped full length RCCA and RCCA bank the PTSPWR2, XCOBRA-IIIC codes were used for the analyses.

Rated full power and the design values of the radial peaking factors and conservative values of the remaining parameters were utilized. The results indicate that neither the pressure, MDNBR nor the peak pellet linear heat generation rate violate the coresponding criteria of: 110% of design pressure, 1.17 and above 21 kw/ft, respectively and, therfore, the results are acceptable.

CVCS Malfunctions with Decreasing Boron Concentration

Reactivity can be added to the reactor by feeding reactor makeup water to the RCS. The normal dilution procedure is subject to administrative procedures to prevent inadvertent dilution. The method for the resolution of such event is to estimate the time from the initiation of the dilution to the time the adverse effect manifests itself. This is not exactly what the SRP recommends but it is consistent with past and current practice. The events which were examined were dilution during refueling, cold shutdown, hot standby, startup and at power. The minimum time to loss of shutdown margin was 16.4 min. which is greater than 15 min and considered adequate for operator action to stop the dilution process. The results are considered acceptable.

Inadvertent Loading of Fuel Assembly into an Improper Location

This event could result from a misplacement of a fuel assembly (Ref. 10). A mismatch of fuel assembly and fuel assembly location could result in higher power production in an assembly and cause it to exceed the maximum values of the peaking factors. Administrative procedures have been established to avoid such misplacement. These procedures require core power distribution monitoring at several power levels to assure that no technical specification limits have been violated.

However, the licensee did not perform an analysis to evaluate the effects of such fuel assembly misplacement. We approve of the Cycle 10 operation under the administrative procedures, however, we will require that the licensee submit such an analysis before the next fuel reload.

RCCA Ejection

This transient is described in Ref. 10, and could result from the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of a RCCA. The result of such failure is very rapid reactivity insertion coupled with severe distortion of the power distribution. The transient has been analyzed using the XTG an approved Exxon computer code (Ref. 16).

No credit was taken for the power peak flattening effects of the Doppler feedback or the moderator feedback. (where the value was negative). The fuel pellet energy deposition resulting from the ejected RCCA was estimated for BOC and EOC conditions. Under hot full power conditions the maximum fuel pellet energy was estimated to be 165 cal/gm at BOC and 172 cal/gm at EOC. This results in less than the maximum allowed of 280 cal/gm as stated in Regulatory Guide 1.77 and is acceptable.

Chapter 15 Transient and Accident Events

To support the Cycle 10 core reload with a return to full power (2300 Mwt) and increase $F^N_{\Delta H}$, the licensee submitted, "NX-NF-84-74, Plant Transient Analysis for H. B. Robinson Unit 2 At 2300 Mwt With Increased $F^N_{\Delta H}$." Document XN-NF-84-74 presents the analysis of the SRP Chapter 15 transient and accident events. The staff's SER is contained in Attachment I.

The licensee has requested and provided justification to defer submittal of the steam line break events until January 31, 1985 in order to provide ENC time to develop an acceptable methodology for analysing the consequences of postulated steamline break events, (see additional details in Attachment I to this SER).

Attachment A of Attachment I provides the staff's independent analysis of the steamline break event. Our analysis confirmed that the new steam generators with integral flow restrictors decreased the severity of the event when compared with the FSAR analysis. The design basis analysis did not result in a calculated DNBR below the specified acceptable fuel design limit (SAFDL). The analysis for Cycle 10 showed the margin the SAFDL is significantly increased. Consequently, fuel integrity is maintained.

Based on the above discussion as expanded in Attachment I and the licensee's justifications, we find the licensee's request to defer the submittal of the steamline break event until January 1985 acceptable. Based on our review of the Chapter 15 transient and accident events contained in Attachment I, we find them acceptable with the licensee's commitments contained therein and reiterated in the forwarding letter of SER.

Loss of Coolant Accident (LOCA) Analysis

To support the changes described in the introduction (Section 1.0) for Cycle 10 operations, the licensee provided a revised LOCA analysis. The ECCS evaluation model utilized to perform the LOCA analysis for HBR-2 is the revised Exxon Nuclear Company (ENC) evaluation model EXEM/PWR also submitted for review. The staff's review of the ENC revised model as utilized for the Cycle 10 LOCA and the staff's evaluation for the LOCA analysis including codes and models is combined as a separate SER and contained in Attachment II to this SER.

Based on the Attachment II SER, we conclude that the LOCA analysis satisfies the requirements of 10 CFR 50.46 and that the evaluation model utilized satisfies the requirements of Appendix R to 10 CFR 50 and, therefore, is acceptable.

TECHNICAL SPECIFICATION CHANGES

The licensee has proposed (References 1, 17 and 18) a number of changes to the Technical Specifications for the cycle 10 reload. The changes to the Technical Specifications can be associated with the major changes for the cycle 10 loading, i.e., return to the 2,300 MWt power level, the new DNB correlation, revision of the power distribution control and the deletion of N-1 loop operation. A listing of all the proposed Technical Specification changes is given in Attachment I of References 1, 17 and 18. Our review and evaluation of each proposed change follows, with the numbering corresponding to that presented in References 1, 17 and 18.

Technical Specification 2.1, Figure 2.1-1

This figure shows the limits and the allowable combinations of thermal power, coolant pressure and coolant inlet temperature, under full flow conditions. The Technical Specification limits shown incorporate the new DNB correlation, the new thermal power level of 2,300 MWt the new hot channel factors and the combined steady state uncertainties. The changes indicated in Technical Specifications 2.1 (a) and 2.1 (d) and in the basis of 2.1 (TS 2.1 pp 1-5, attachment 8. Ref. 1 and the editorial changes of reference 18) have been reviewed and found to be in agreement with approved changes and, therefore, are acceptable.

Specification 2.3

This specification deals with the overtemperature and overpower ΔT setting changes shown in pp 2.3-1 to 2.3-6 of attachment 8 ref. 1 and the editorial changes of ref. 18. The reasons for the changes are: return to the 2,300 MWt power level, the new DNB correlation and deletion of N-1 loop operation. The overtemperature and the overpressure limit settings form the largest part of the changes, the rest are either reference or editorial changes. We have reviewed the proposed changes and we find that they consist of utilization of previously approved (Westinghouse) overtemperature ΔT and overpower ΔT analytical expressions with parameter values adjusted for the use of the new DNB correlation and the new power level of 2,300 MWt. We find these acceptable.

Specification 3.1.1.1, LCO for Coolant Pumps

This specification deals with the limiting conditions of operation for the reactor coolant pumps. The changes are indicated in pp 3.1-1 to 3.1-3b of attachment 8 Reference 1 and in the editorial changes listed on pp 3.1-1 and 3.1-2 of Ref. 18. The most important aspect of this specification is that no power operation is allowed without all primary coolant pumps being in operation. This condition and the conditions and actions specified for operation at or less than 2% of thermal power, have been reviewed and are acceptable.

Specification 3.1.1.2 Steam Generator

This specification requires that at least two steam generators shall be operable whenever the average primary coolant temperature is above 350°F. This specification is listed in pp 3.1-3 to pp 3.1-3b of attachment 8, reference 1. The specification itself has not changed. Some editioral and minor changes were made in the basis. These changes were reviewed and found to be acceptable.

Specification 3.1.3.1 Minimum Conditions for Criticality

This specification defines +5.0 pcm/°F as the upper limit of the moderator temperature coefficient for the reactor to be critical at less than 50% of rated power and linearly decreasing to 0 pcm/°F at full power. This Specification and its basis are listed on pp 3.1-11 and 3.1-12 of attachment 8, reference 1. The value of the moderator temperature coefficient has been changed from +2.0 pcm/°F. However, the transient analyses which involve heatup of the primary coolant supported the value of +5pcm/°F. On this basis we find this specification acceptable.

Table 3.5-1. Engineered Safety Feature System Initiation Instrument-Setting Limits.

The table and the proposed changes are listed in pp 3.5-7 and 3.5-7a. The changes are either of editorial nature or a consequence of returning to 2,300 MWt of rated power operation. On this basis the changes to Table 3.5-1 are acceptable.

Table 3.5-3, Instrumentation Operating Conditions for Engineered Safety Features.

This table is partially listed on p. 3.5-10a and the changes refer to the footnotes and they are of editorial nature. On this basis the changes to Table 3.5-3 are acceptable.

Specification 3.6.1, Containment Integrity

This specification deals with containment integrity and the circumstances under which operations can be performed in the reactor regarding core reactivity changes. The proposed changes are listed on pp. 3.6-1 and 3.6-2 of attachment 8 ref. 1 and pp. 3.6-1 to 3.6-3 of ref. 17 and are due to the elimination of the part length rods. The proposed changes define the conditions and the operations (tests) to be performed when the containment is not intact. However, the shutdown margin during any of these tests will always be maintained at a level greater than 1% of $\Delta k/k$. The boron dilution will be monitored during any such change. Under these conditions the proposed changes are acceptable.

Specification 3.6.2, Internal Pressure

The proposed changes in this specification are listed on pp 3.6-1 and 3.6-3 of attachment 8 ref. 1 and on p 3.6-2 of ref. 17. The changes are editorial and reference updating to the FSAR. The proposed changes are acceptable.

Specification 3.6-3, Containment, Automatic Isolation Trip Valves

The proposed changes are listed on p.3.6-3 of attachment 8, ref. 1 and on pp 3.6-2 and 3.6-3 of ref. 17 and are of editorial nature and reference updating. The proposed changes are acceptable.

Page 3.86 , on Containment

The proposed changer are reference updating and, therefore, are acceptable.

Specification 3.10.1.5 Control Rod Operation

The proposed changes are listed on p. 3.102 of attachment 8 reference 1. The changes are due to returning to 2,300 MWt power level and are acceptable.

Specification 3.10.2 Power Distribution Limits

The proposed changes in this specification are listed on pp 3.102 to 3.107a of attachment 8 in ref. 1. The proposed changes emanate from the return to 2,300 MWt rated power, the revised power distribution control and the deletion of N1 loop operation. The determination of F_Q (Z) and $F_{\Delta H}$ under different conditions is specified in detail. In addition alternate actions are specified regarding reactor power assuming that the required values of F_Q (Z) or $F_{\Delta H}$ cannot be satisfied. Conversely, the allowable power level is estimated for given values of the peaking factors with the required uncertainty factors for engineering (F_Q^E = 1.03) measurement (Fu^N = 1.05) and the instruments (F_Q^a = 1.02) and their proper usage is specified. The required frequencies for core power distribution maps and for the target axial flux difference are specified. Alternate actions for the power level depending on the axial flux difference are also given. Finally the calibration of the ex-core detectors is specified. We have reviewed the proposed changes and found them acceptable for the operation of cycle 10 because they have been taken into account in the transient and accident analysis.

Specification 3.10.3, Quadrant Power Tilt Limits

The proposed changes to this specification are listed on pp 3.10-7a and 3.10-7b of attachment 8, reference 1. The proposed changes reflect the return to 2,300 MWt rated power and the revised power distribution control.

The new specification is in compliance with the provisions of the standard review plan and therefore is acceptable.

Specification 3.10.8, Required Shutdown Margin

The proposed changes are indicated on p 3.10-12, pp 3.10-14 to 3.10-20 and pp 3.10-22 to 3.10-24 of attachment 8 reference 1. The changes are due to: return to 2,300 MWt, the new DNB correlation, the new power distribution control, deletion of references to part length rods, updating of references and repagination. Figures 3.10-4 and 3.10-5 have been added. We have reviewed the changes and found the values of MDNBR, peak value of the linear power density, $F_{\Delta H}$ and F_Q to be within their approved ranges. The method and the specified limits of the allowable deviation from the target flux difference are acceptable. Therefore the proposed specification is acceptable.

Specification 3.11.2, Movable In-Core Instrumentation

The proposed changes are indicated on pp 3.11-1 and 3.11-2 of attachment 8 of reference 1. The changes are due to returning to 2,300 MWt rated power, change of the number of the in-core instrumentation thimbles, and deletion of N-1 loop operation. The proposed change in the number of thimbles is an increase and is acceptable, likewise the changes due to the power level and N-1 loop operation are also acceptable.

Specification 4.11.1 APDMS Operation

The proposed changes are listed on pp 4.11.1 to 4.11.3 of attachment 8 in reference 1. The changes are due to returning to the 2,300 MWt level of operation, the change in the number of thimbles (see specification 3.11.1 above) and the improved power distribution control. The changes have been reviewed and found to be acceptable.

Specification 5.3.1 Reactor Core

The proposed changes are listed on p 5.3-1 of attachment 8, reference 1 and p 5.3-1 of reference 18. The proposed changes allow the reconstitution of the partial length shield assemblies, specify the new total core fuel loading, specify the maximum fuel enrichment and allow axial natural uranium blanket. We have reviewed the proposed changes and find them necessary and acceptable for the operation of Cycle 10.

Specification 5.3.2 Reactor Coolant System

The proposed changes are listed on pp 5.3-1 and 5.3-2 of attachement 8, reference 1. The changes are updating of references and editorial. The proposed changes are acceptable.

3.0 SUMMARY

We have reviewed the information submitted on Cycle 10 for the H. B. Robinson Unit 2. This included the original and subsequent submittals which were additions and clarifications in response to questions generated by the review. We find the Cycle 10 operation acceptable for the fuel system mechanical design, nuclear design, thermal-hydraulic design and analysis, transients and accidents, the radiological consequence analysis for the locked rotor, the rod cluster control assembly withdrawal and the steam generator tube rupture; and the Technical Specifications proposed.

This approval is subject to the following conditions:

(a) confirmation of the scram shutdown margin analysis (to be submitted during CY85) and

(b) submittal of a fuel misloading analysis (during Cycle 10).

4.0 FINAL DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

4.1 On August 24, 1984, the Commission published in the Federal Register (49 FR 33764) a Notice of Consideration of Issuance of Amendment To Facility Operating License And Proposed No Significant Hazards Consideration Determination And Opportunity for Hearing. That notice addressed a change requested by the licensee in their letter dated July 23, 1984. The change requested involved changes to the Technical Specifications in support of their Cycle 10 core reload and return to full power (2300 MWt) with new steam generators. The reload application included special peripheral part length shielded fuel assemblies, which will be installed to accommodate the pressurized thermal shock program. To accommodate these assemblies, a low leakage core and return to full power (2300 MWt) with new steam generators, hot channel factor (F_Q and $F_{\Delta H}$) limits and BOC moderator temperature coefficients are being increased and over temperature and overpower ΔT setpoints are being reduced. In support of these changes the licensee provided a safety evaluation for the Cycle 10 core reload, reanalyzed the Chapter 15 events and provided a LOCA analysis.

Because the Commission determined there was insufficient time for its usual 30-day notice of the proposed action for public comment, that notice established a period until September 14, 1984 for comment, state that a final determination on no significant hazards would be made before issuance of the license amendment, and provided that if no significant hazards are involved, a subsequent notice of opportunity for a hearing would be published.

In our evaluation of the LOCA and fuel performance analysis we determined that the revised analysis for Cycle 10 reload of HBR-2 satisfies the requirements of 10 CFR 50.46 and that the evaluation model utilized satisfies the requirements of Appendix K to 10 CFR 50.

The application for Cycle 10 reload incorporated plant design changes resulting from steam generator replacement and justification of return to full power. In support of this the licensee submitted analyses of the Chapter 15 transient and accident events.

The Safety Evaluation Reports (SER) for Cycle 8 and Cycle 9 required the licensee to develop a stand-alone analysis methodology which does not infringe upon other vendor's methods. The methodology is under our staff's final review and have not yet been approved. We have evaluated and discussed these items in detail in our SER Attachment I and our review has progressed sufficiently to conclude that analyzed events would not be significantly altered upon completion of review. This conclusion is based upon code validation results, limiting boundary conditions applied to each event and benchmarking the computer codes with known experimental transients conducted at the LOFT facility.

The licensee's contractor has not finalized its methodology for evaluating the consequences of postulated steam line break events. The licensee will reanalyze the SLB event for Cycle 10 by January 31, 1985. Our bases for accepting the late submittal 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) The staff analysis of the steam line break event (guillotine break) showed ample margin to the acceptance criteria for H. B. Robinson Unit 2. (See Appendix A to Attachment I of our SER).

4.2 On October 5, 1984, the Commission published in the Federal Register (49 FR 39396) a Notice of Consideration of Issuance of Amendment to Facility Operating License And Proposed No Significant Hazards Consideration Determination And Opportunity for Hearing. That notice specifically addressed a change requested by the licensee in their letter dated August 1, 1984. Because the Commission determined there was insufficient time for its usual 30-day notice of the proposed action for public comment, that notice established a period until October 17, 1984 for comment, stated that a final determination on no significant hazards would be made before issuance of the license amendment, and provided that if no significant hazards are involved, a subsequent notice of opportunity for a hearing would be published. The proposed change as requested by letter dated August 1, 1984, involves changes to the Technical Specification to allow additional control rod evaluations while containment integrity is not intact but only while maintaining a shutdown margin of $\geq$ 1% $\Delta k/k$.

In our evaluation we note that during the additional conditions and operations (tests) to be performed when the containment is not intact, the shutdown margin must be maintained at $\geq$ 1% $\Delta k/k$ as already required by Technical Specification 3.6.1c. The boron dilution will be monitored during any such changes.

4.3 We have determined that the proposed change does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility or a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

5.0 Environmental Conclusions

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. A portion of the amendment proposed was subsequently changed; the Commission has also made a final no significant hazards consideration finding with respect to the changed portion of this amendment. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.2 Safety Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 7, 1984

PRINCIPAL CONTRIBUTORS:

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References

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2. XN-75-39, "Generic Fuel Design for 15x15 Reload Assemblies for Westinghouse Plants" Exxon, September 1975.

3. XN-NF-83-55, "Mechanical Design Report Supplement for H. B. Robinson Extended Burnup Fuel Assemblies" Exxon, August, 1983.

4. XN-NF-83-71, "Mechanical Design Report Supplement for H. B. Robinson Part Length Sheild Assemblies" Exxon, September, 1983.

5. XN-NF-75-27(A) "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors: Exxon, June 1975. Supplement 1, September 1976, Supplement 2, December 1977 and Supplement 3, November 1980.

6. XN-NF-77-57(A) "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors - Phase II" Exxon, January 1978. Supplement 1, June 1979 and Supplement 2, September 1981.

7. XN-75-38, "H. B. Robinson Unit 2, Cycle 4 Reload Fuel Licensing Data Submittal" Exxon, August, 1975.

8. XN-NF-84-74, "Plant Transient Analysis for H. B. Robinson Unit 2 at 2300 MWt with Increased $F_{\Delta H}$" Exxon, August 1984.

9. XN-NF-84-72, "H. B. Robinson Unit Large Break LOCA-ECCS Analysis with Increased Enthalpy Rise Factor: Break Spectrum" Exxon, August 1984.

10. XN-NF-83-72, "H. B. Robinson Unit 2, Cycle 10 Safety Analysis Report" Revision 2, Exxon, July 1984. Supplement 1, Exxon July, 1984.

11. XN-CC-28, Rev. 5, "XTG - A Two Group Three-Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing (PWR-Version)" Exxon, July 1979.

12. WAPD-TM-678, "PDQ7 Reference Manual" Westinghouse Electric Corporation, January 1967.

13. WAPD-TM-478, "HARMONY: System for Nuclear Reactor Depletion Computation" Westinghouse Electric Corporation, January 1965.

14. XN-NF-82-91(P), "H. B. Robinson-2, Loss of Feedwater Transient at 1955 MWt: Model Description and Results" Exxon, November, 1982.

15. XN-NF-84-74, Supplement 1. "Plant Transient Analysis for H. B. Robinson Unit 2 at 2,300 MWt with Increased $F_{\Delta H}$. Supplement 1, Analysis of Control Rod Misoperation Events" Exxon Nuclear Company, September 1984.

16. XN-CC-28 (NP) (A), Revision 5, " XTG - A Two Group 3 - Dimensional Reactor Simulator Utilizing Course Mesh Spacing (PWR Version)" Exxon Nuclear Company, July 1979.

17. A. B. Cutter, letter to S. A. Varge, "H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 50-261/License No. DPR-23. Request for License Amendment Control Rod Drive Evolution" Serial NSL-84-186, dated August 1, 1984.

18. A. B. Cutter, letter to S. A. Varga, "H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 50-261/License No. DPR-23, Cycle 10 Operation", Serial: NLS-84-366, dated September 7, 1984.

19. XN-NF-75-32(P)(A) "Computational Procedures for Evaluating Fuel Rod Bowing", Exxon October 1983, and Supplements 1, 2, 3 and 4.

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 <u>Inadvertent Opening of a Steam Generator Relief or Power Operated Relief Valve</u>

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 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 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

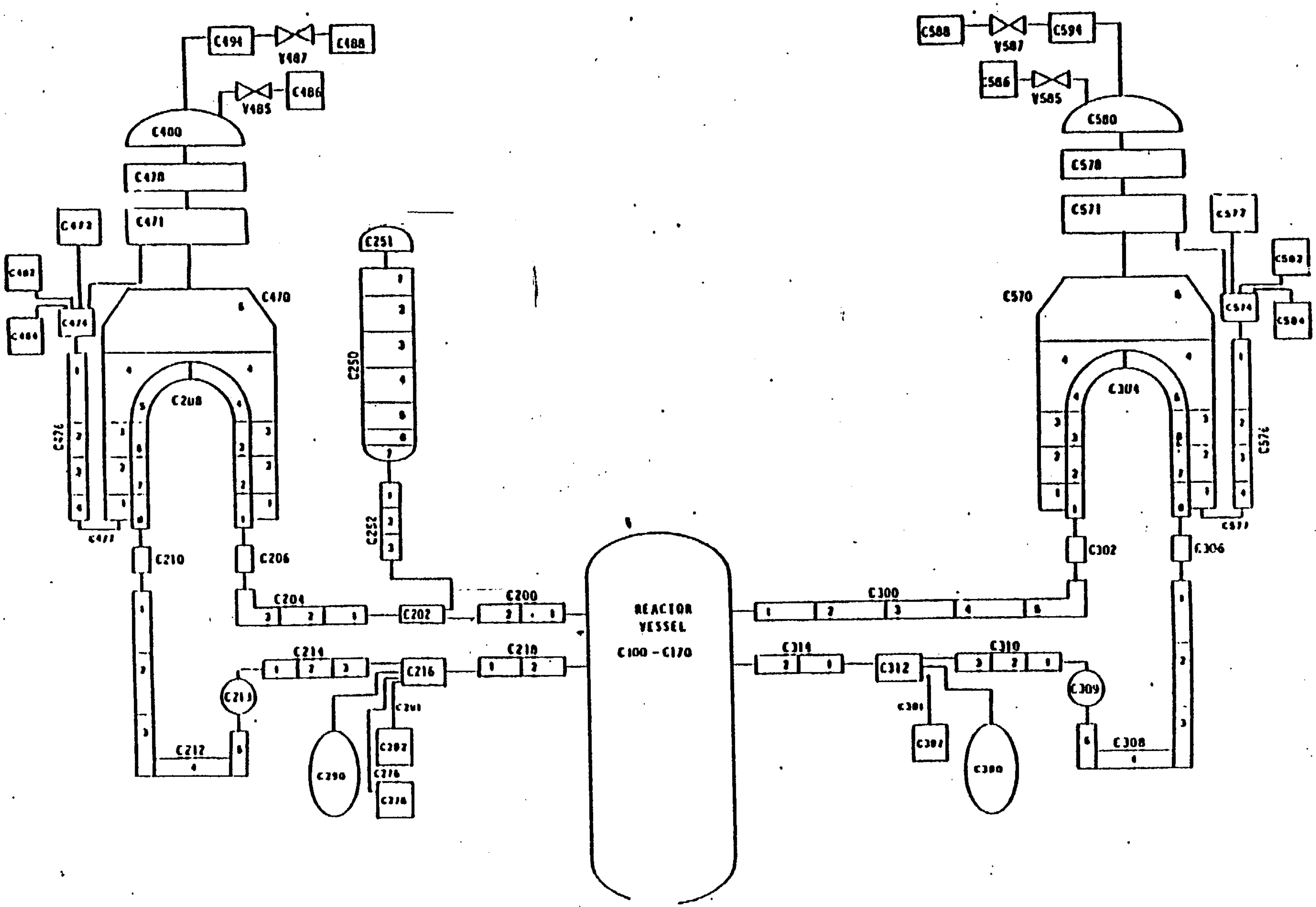
The computer code used for analyzing the steam line break (SLB) event was RELAP5/MOD1.5 Cycle 39. An input deck of a generic 3-loop Westinghouse plant was modified by data supplied by CP&L and ENC to model the H. B. Robinson plant.

II.1 NODALIZATION

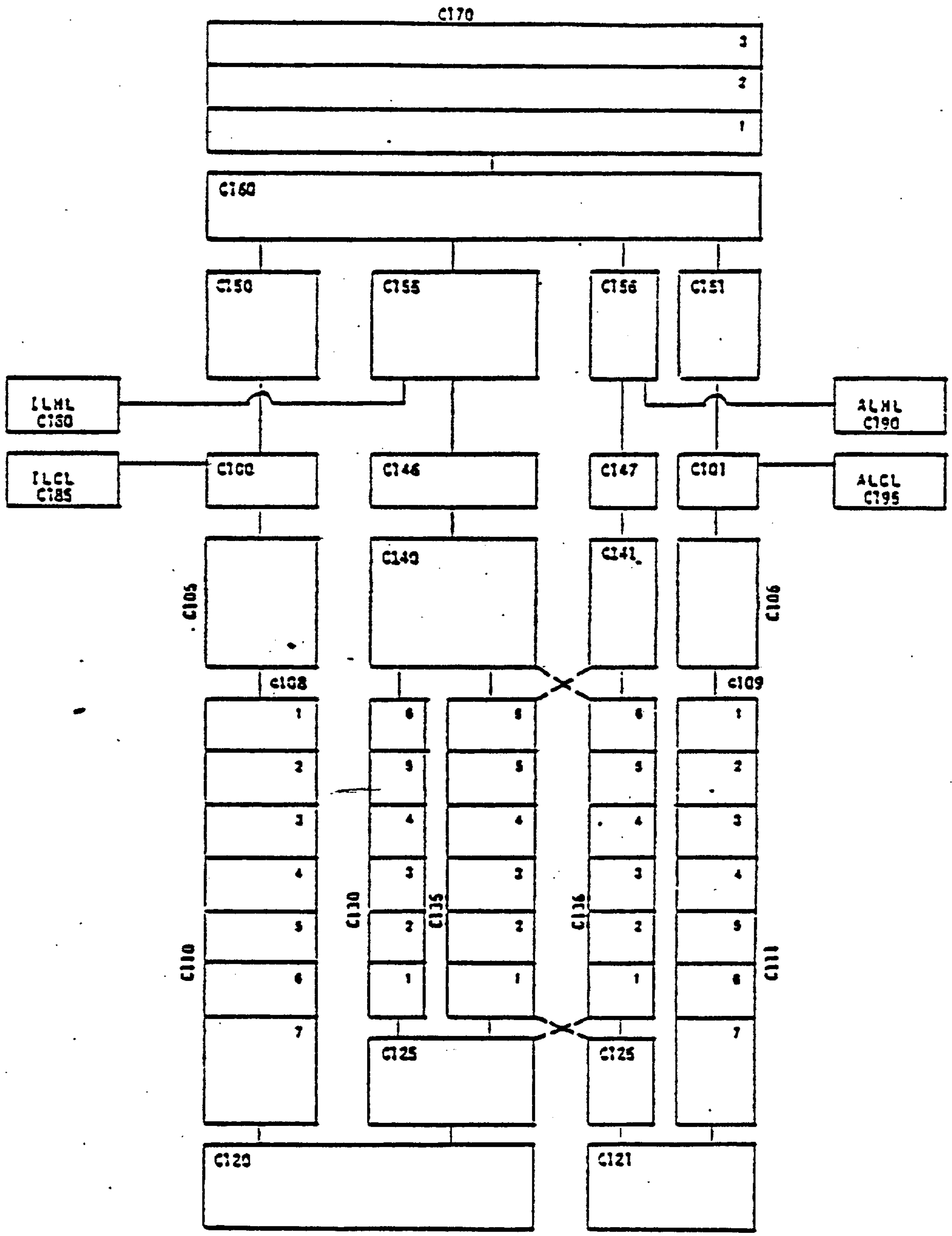
The model nodalization used for the H. B. Robinson SLB calculations is shown in Figs. A-1 and A-2. This nodalization represents the major components and flow paths of the H. B. Robinson 3-loop nuclear steam supply system.

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.

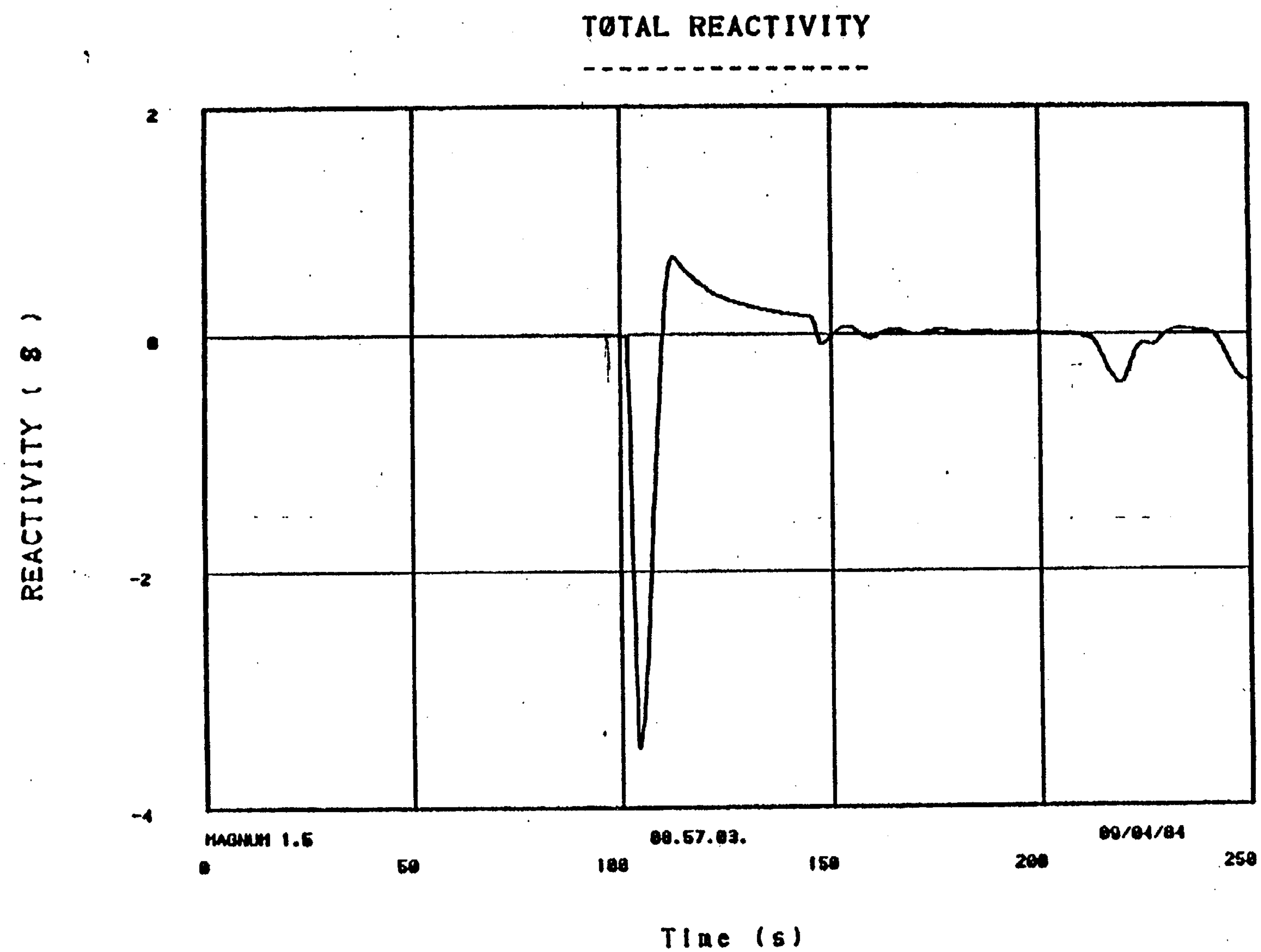
The results from the original design basis FSAR analysis are shown in Fig. A-7. Comparisons between the staff calculation and the FSAR design basis analysis are in good agreement.

IV. DETERMINATION OF THE LIMITING BREAK

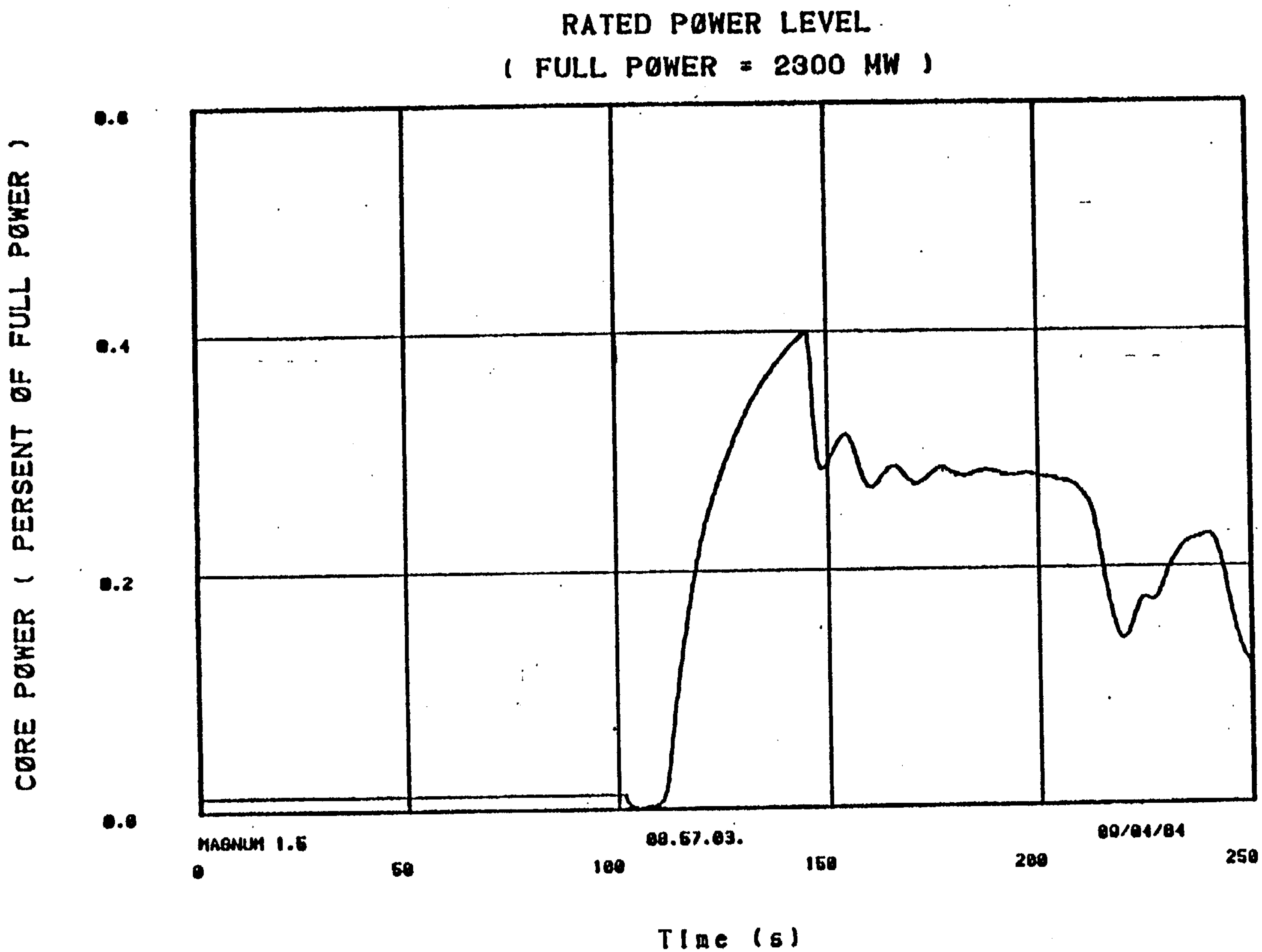
The following cases were analyzed to determine the limiting break location and conditions for H. B. Robinson Unit 2 Cycle 10:

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).

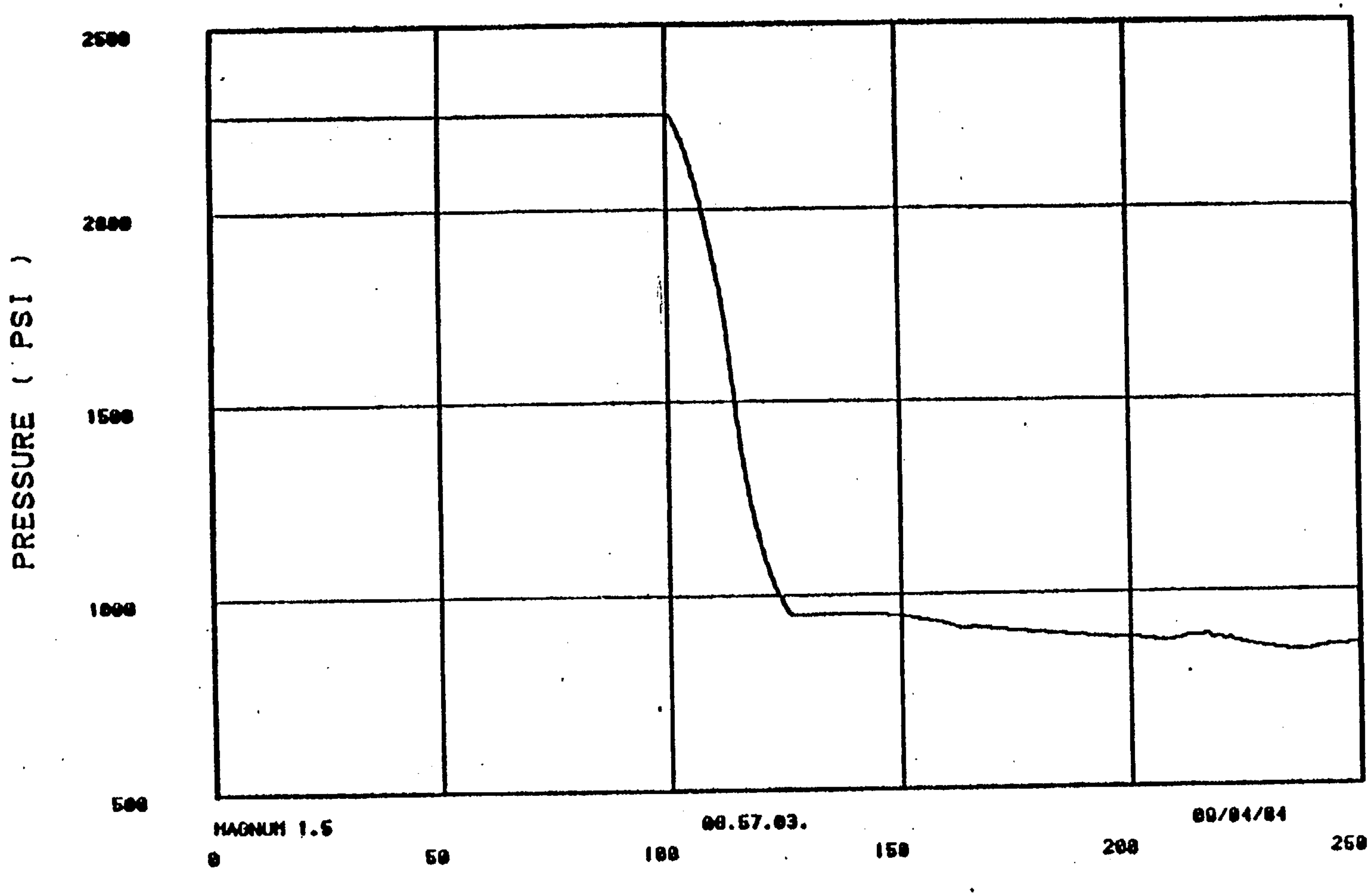


H.B.RØBINSØN CYCLE 10 RELØAD
STEAM LINE BREAK ANALYSIS
BREAK UPSTREAM ØF FLØW RESTRICTØRS
FIGURE A-3

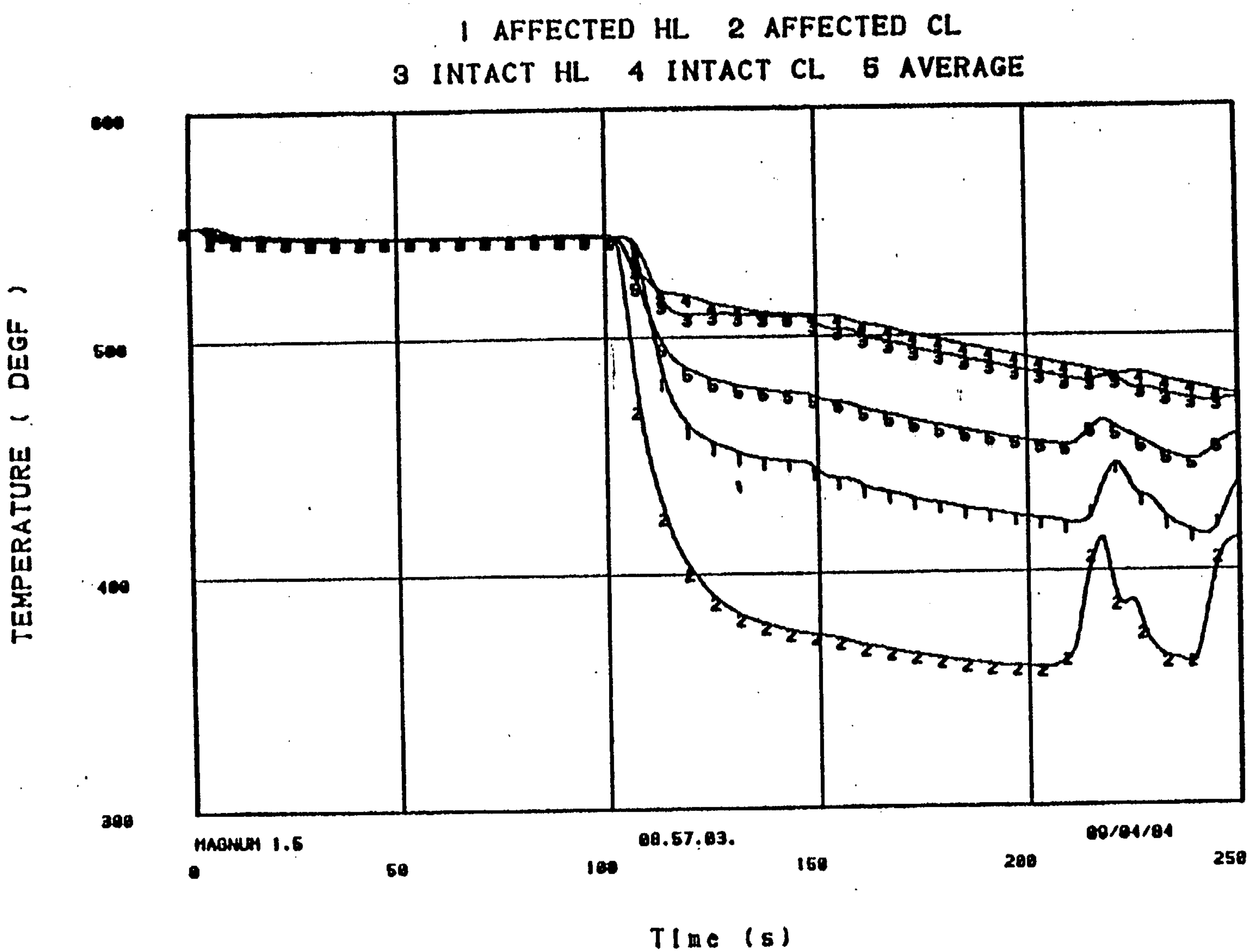


H.B.RØBINSØN CYCLE 10 RELØAD
STEAM LINE BREAK ANALYSIS
BREAK UPSTREAM ØF FLØW RESTRICTØRS
FIGURE A-4

PRESSURISER PRESSURE



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

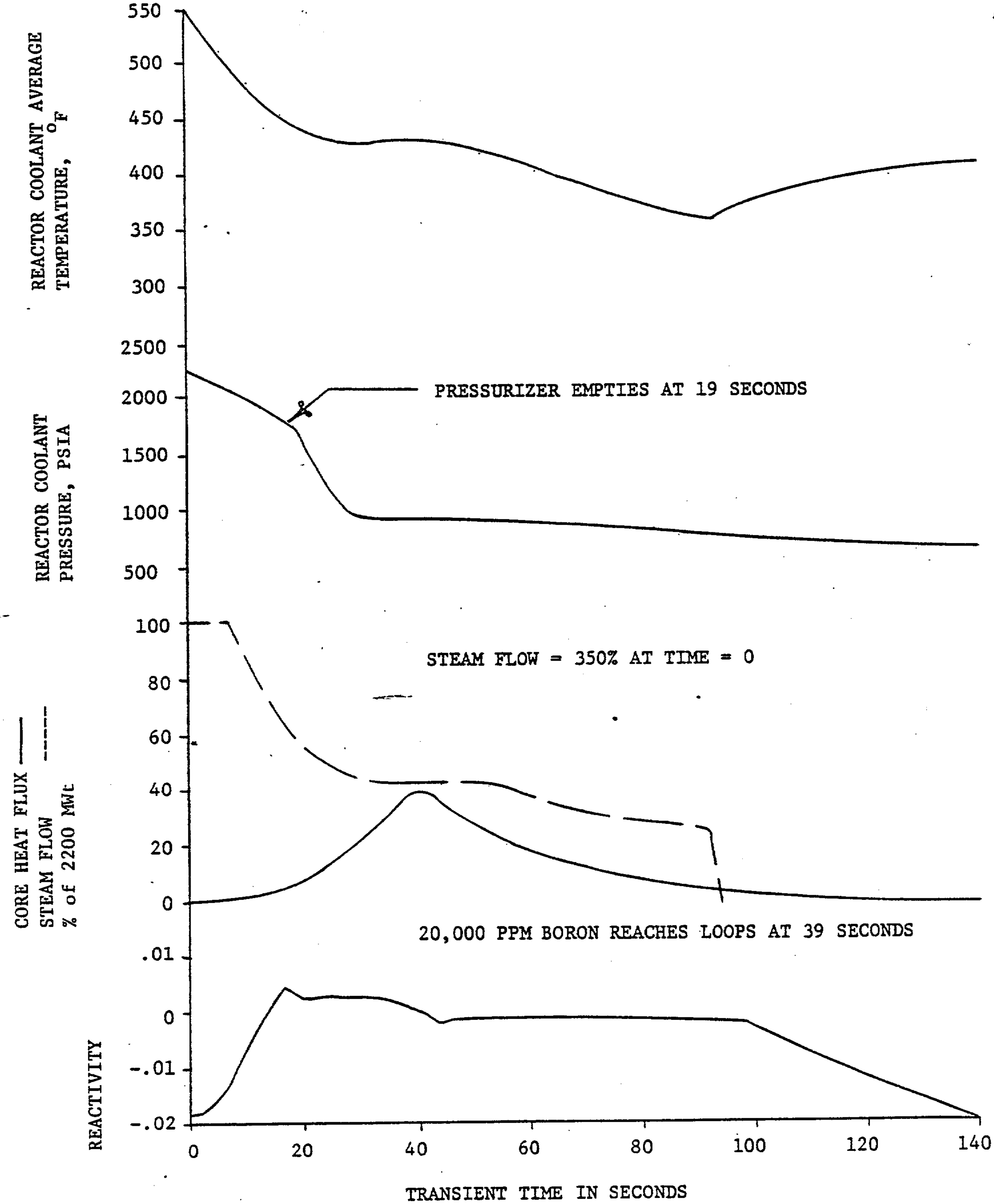


FIGURE A-7

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

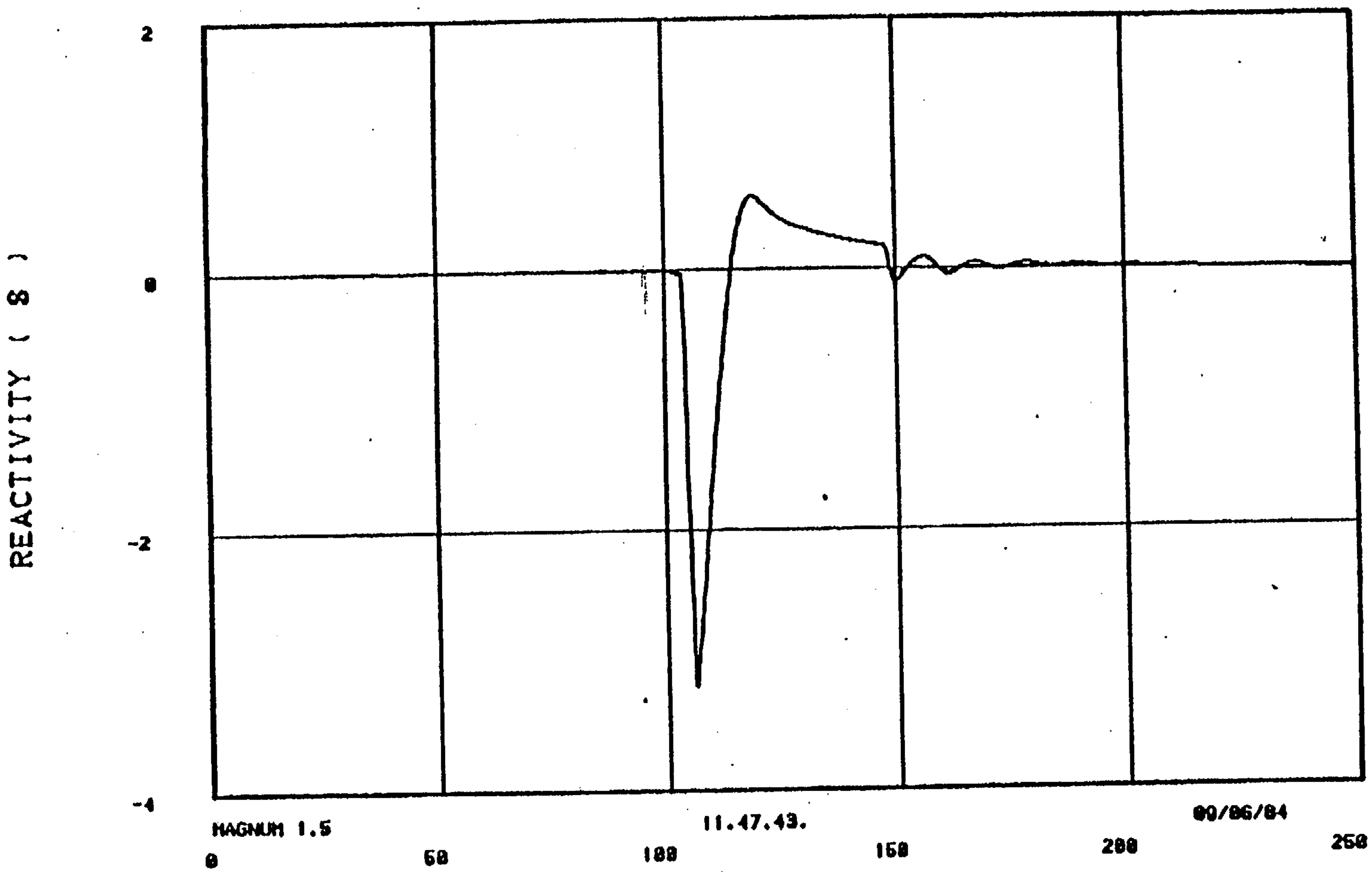
	Maximum Return to Power (% of 2300 MWt)	Time (sec)
Case 1	19.2	48.0
Case 2	22.4	47.4
Case 3	13.1	51.2
Case 4	13.6	50.6

TABLE A-7

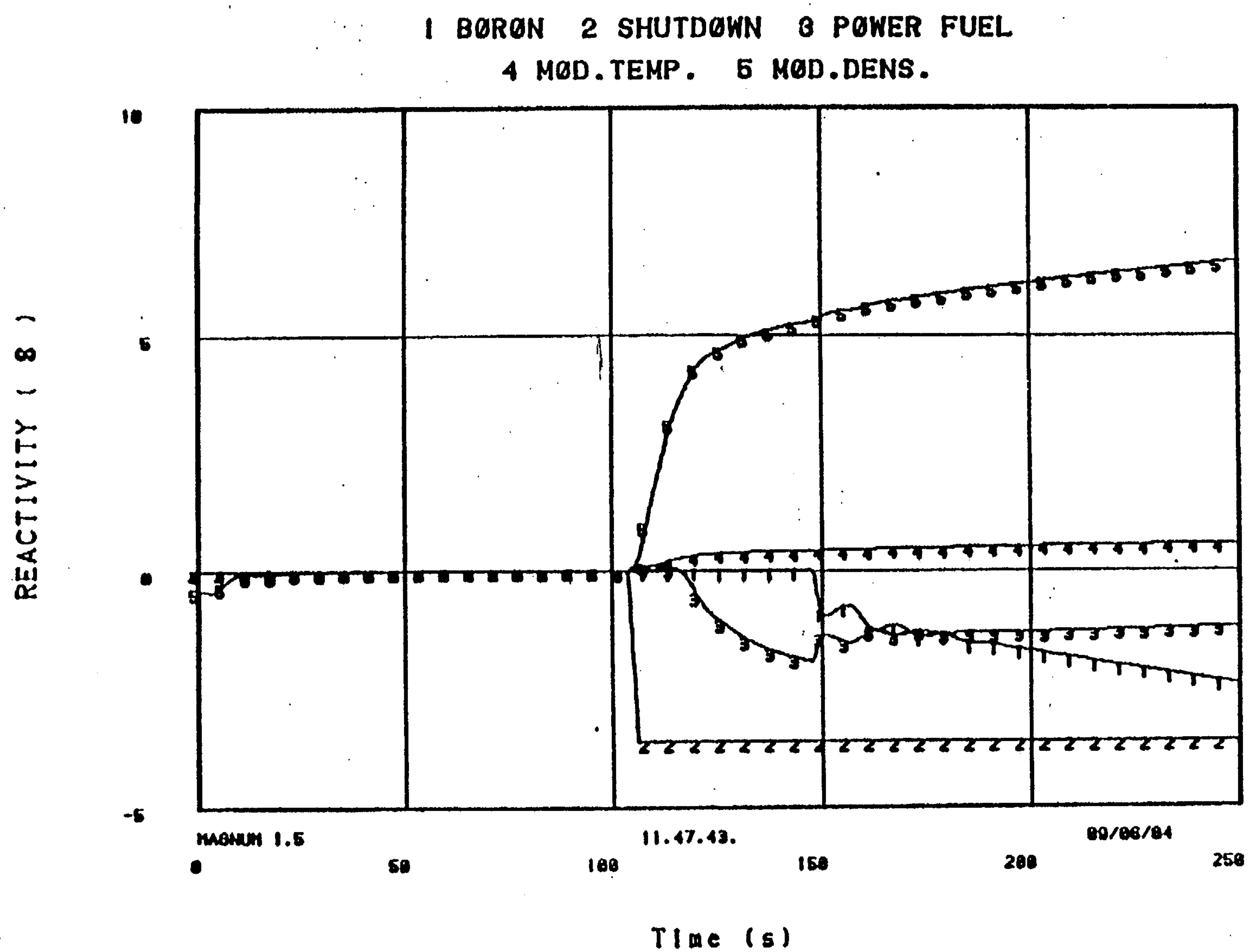
SEQUENCE OF EVENTS FOR
THE LIMITING SLB

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

TØTAL REACTIVITY

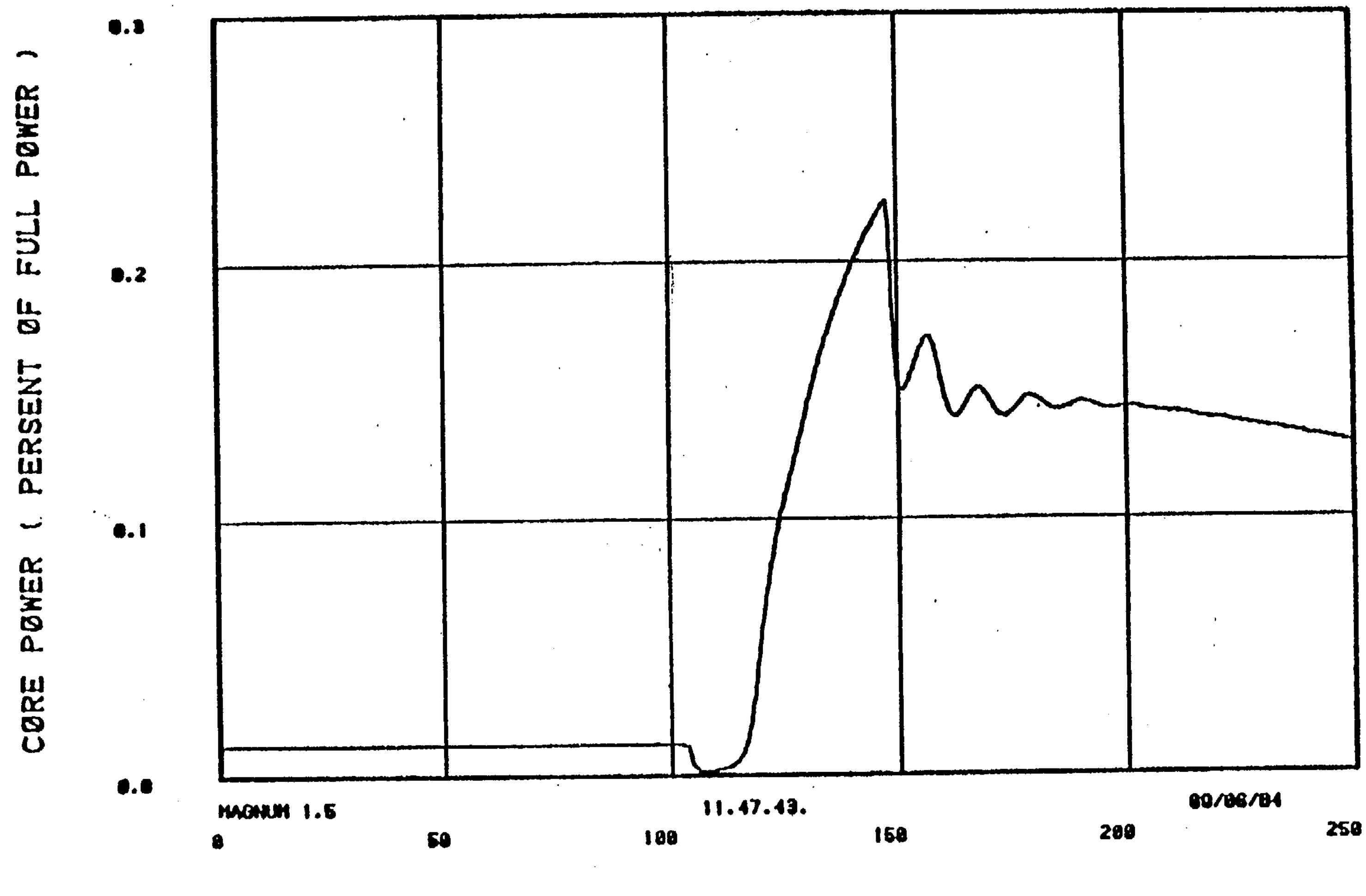


H.B.RØBINSØN CYCLE 10 RELØAD
STEAM LINE BREAK ANALYSIS
BREAK DØWNSTREAM ØF FLØW RESTRICTØRS
FIGURE A-8



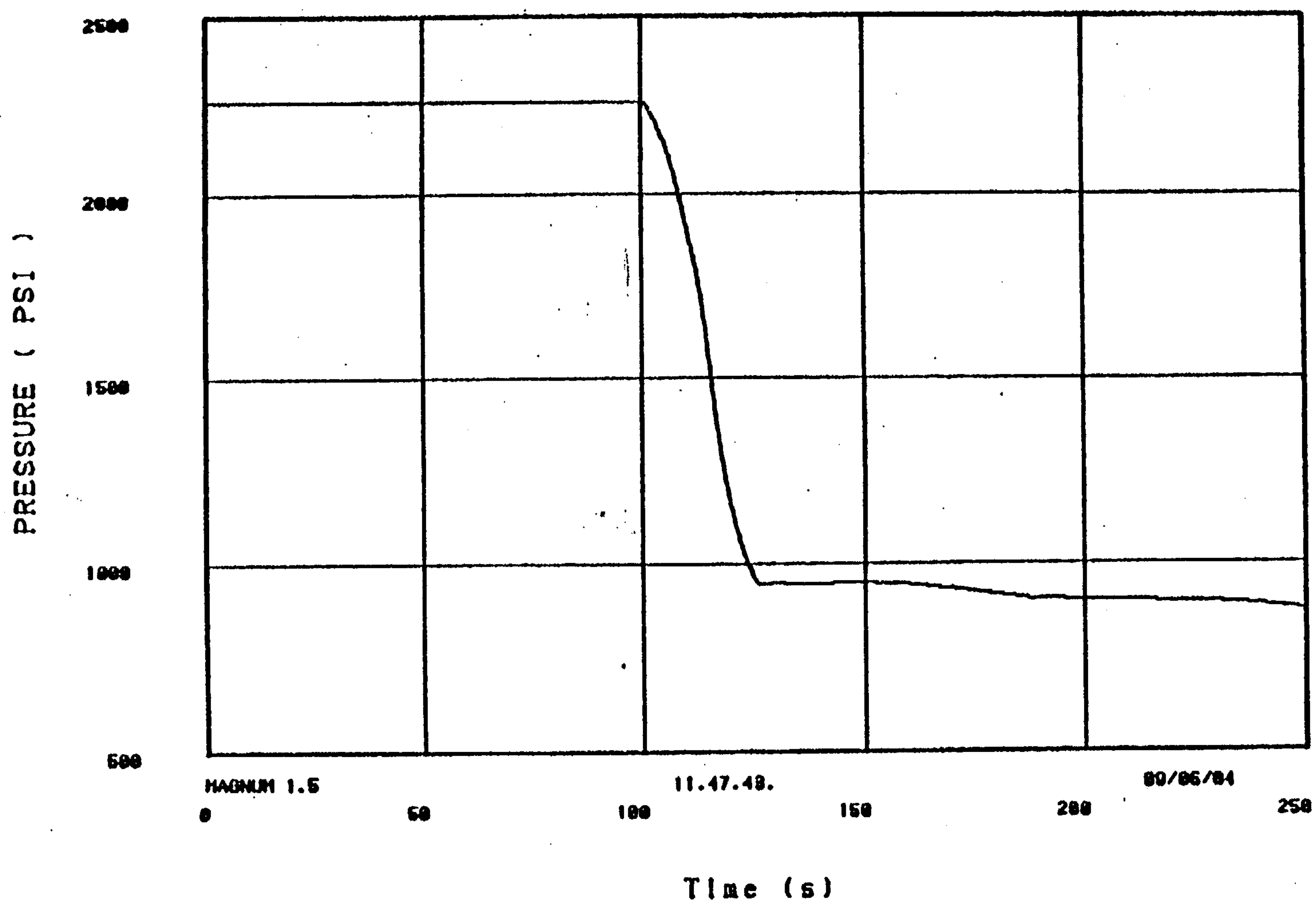
H.B.RØBINSØN CYCLE 10 RELØAD
STEAM LINE BREAK ANALYSIS
BREAK DØWNSTREAM ØF FLØW RESTRICTØRS
FIGURE A-9

RATED PØWER LEVEL
(FULL PØWER = 2300 MW)



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

PRESSURISER PRESSURE

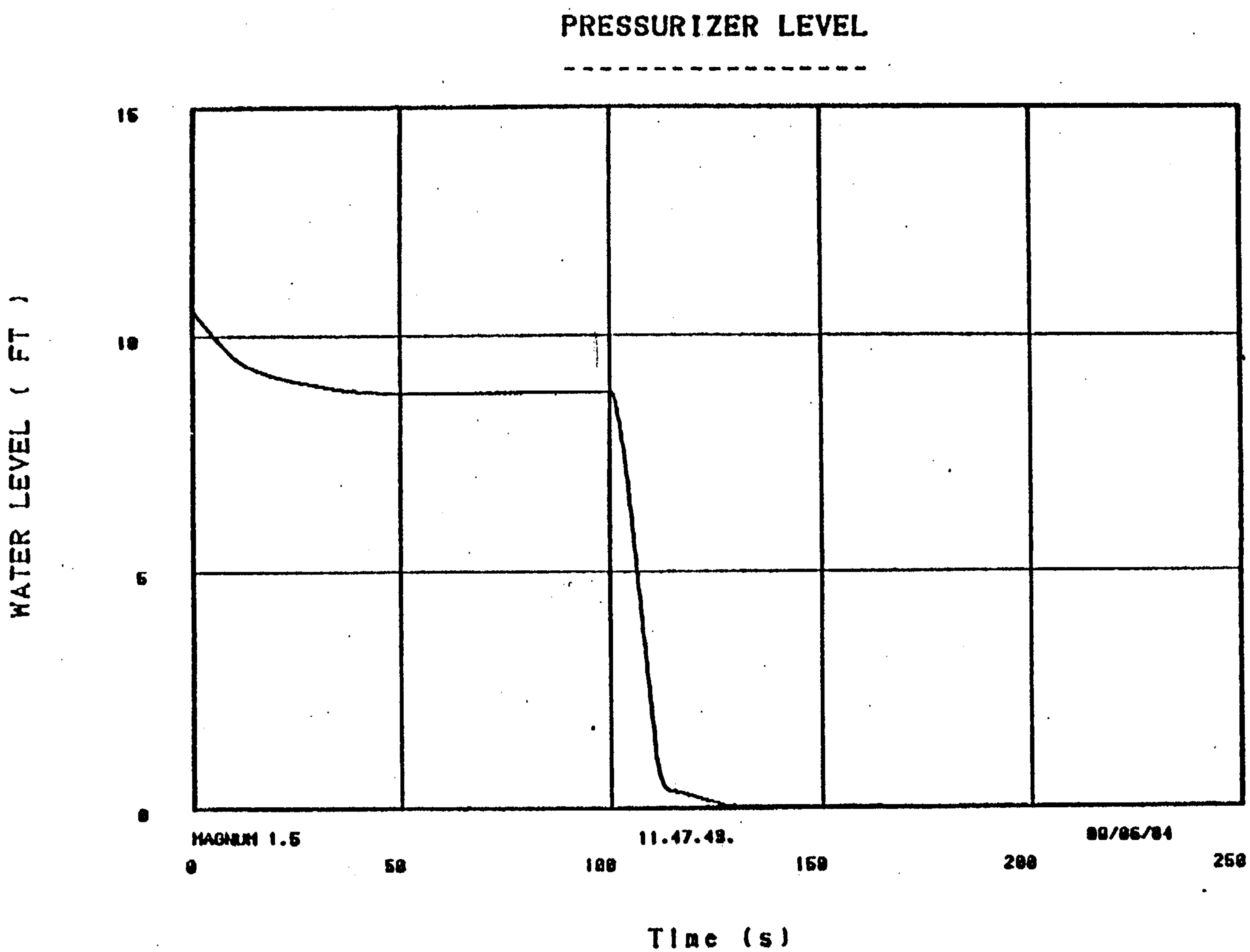


H.B.RØBINSØN CYCLE 10 RELØAD

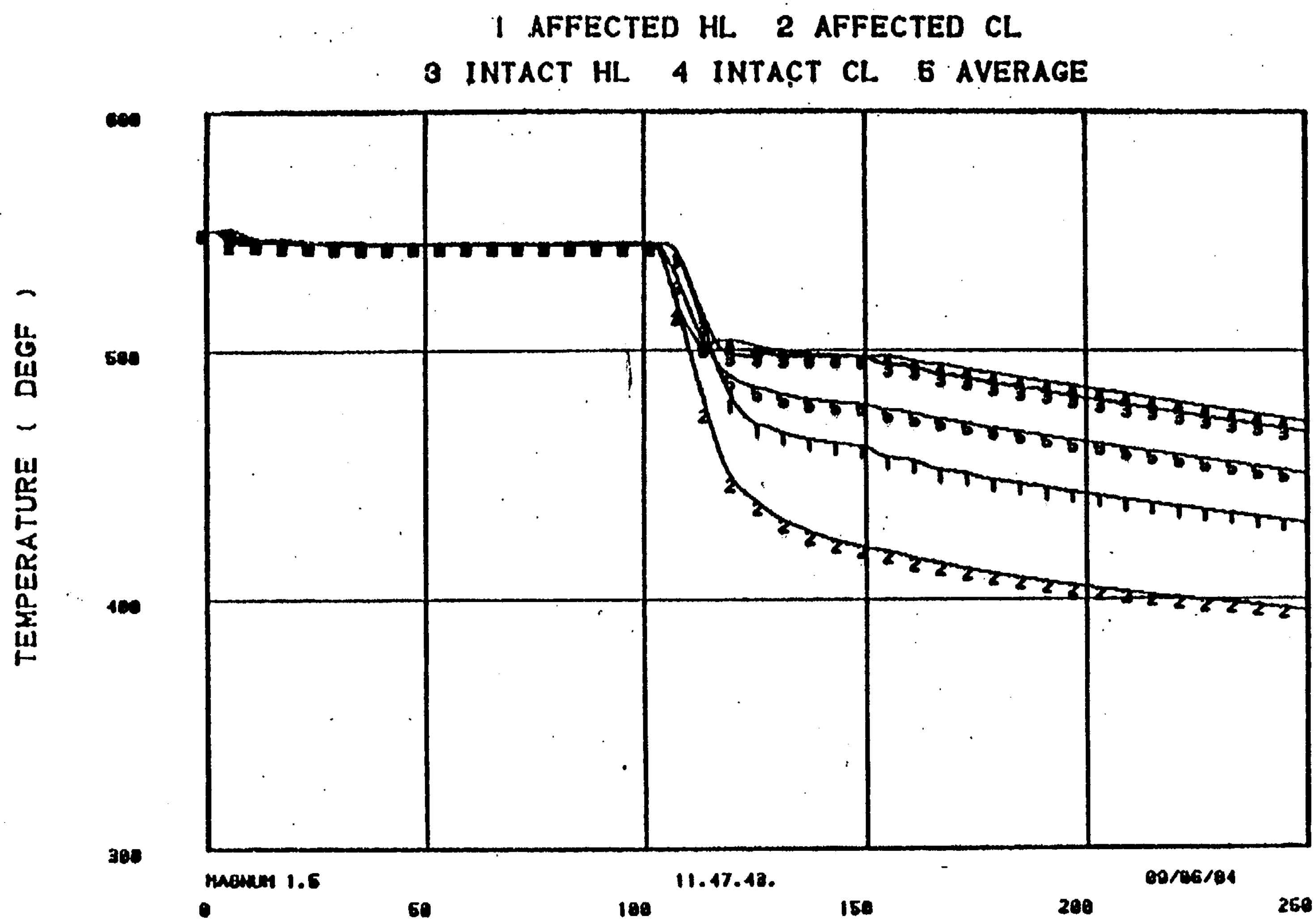
STEAM LINE BREAK ANALYSIS

BREAK DØWNSTREAM ØF FLØW RESTRICTØRS

FIGURE A-11



H.B.RØBINSØN CYCLE 10 RELØAD
STEAM LINE BREAK ANALYSIS
BREAK DØWNSTREAM ØF FLØW RESTRICTØRS
FIGURE A-12



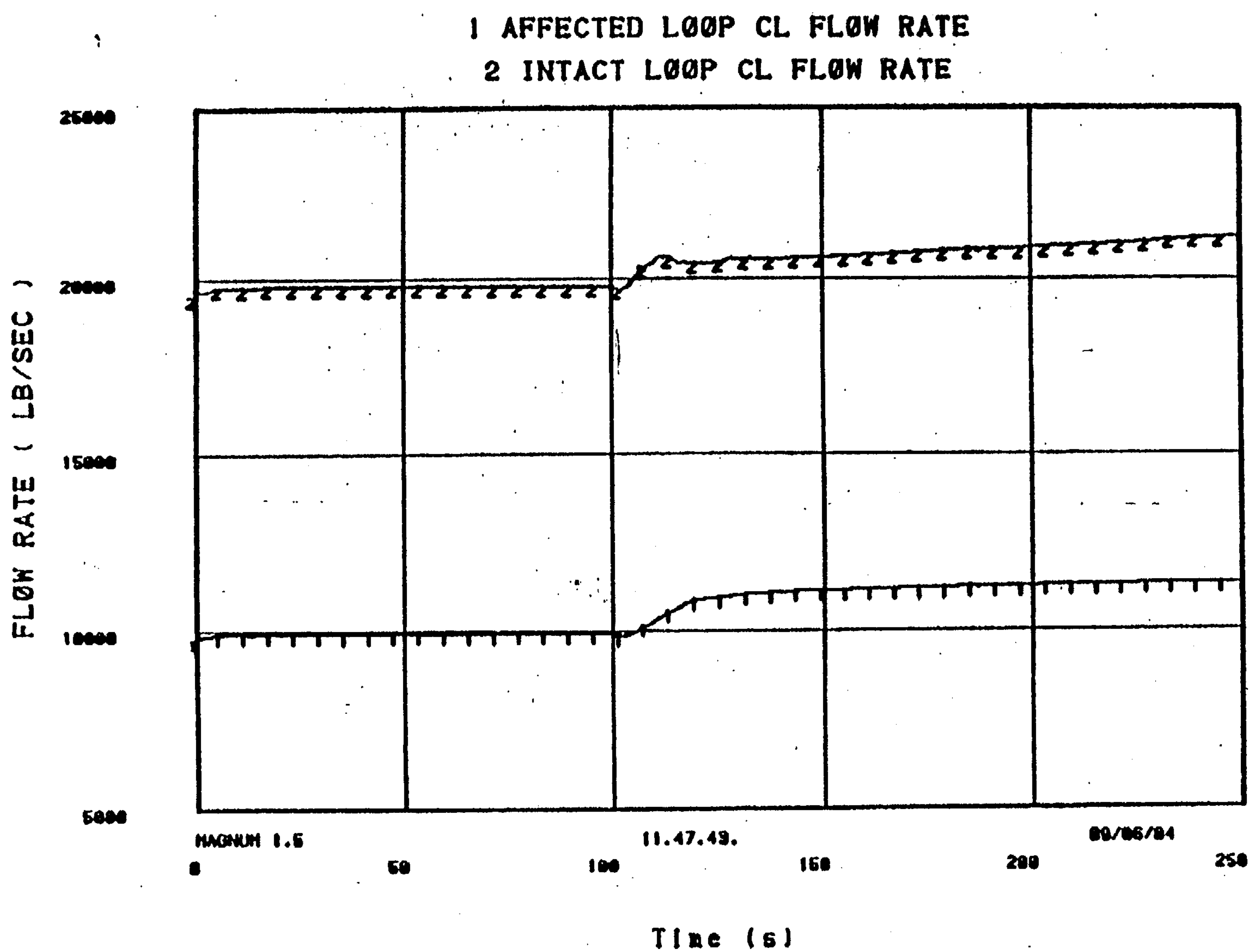
Time (s)

H.B.RØBINSØN CYCLE 10 RELØAD

STEAM LINE BREAK ANALYSIS

BREAK DØWNSTREAM ØF FLØW RESTRICTØRS

FIGURE A-13



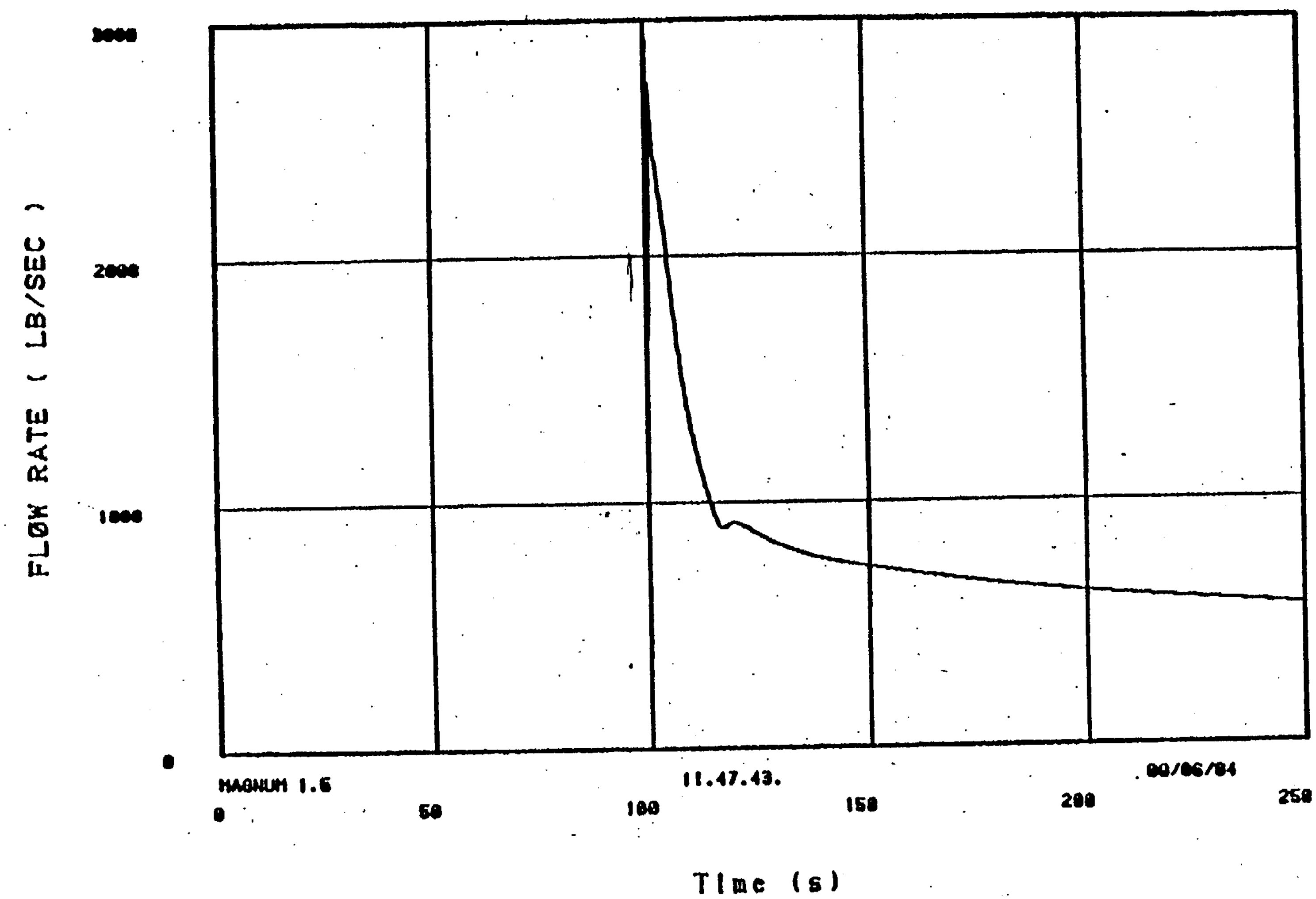
H.B.RØBINSØN CYCLE 10 RELØAD

STEAM LINE BREAK ANALYSIS

BREAK DØWNSTREAM ØF FLØW RESTRICTØRS

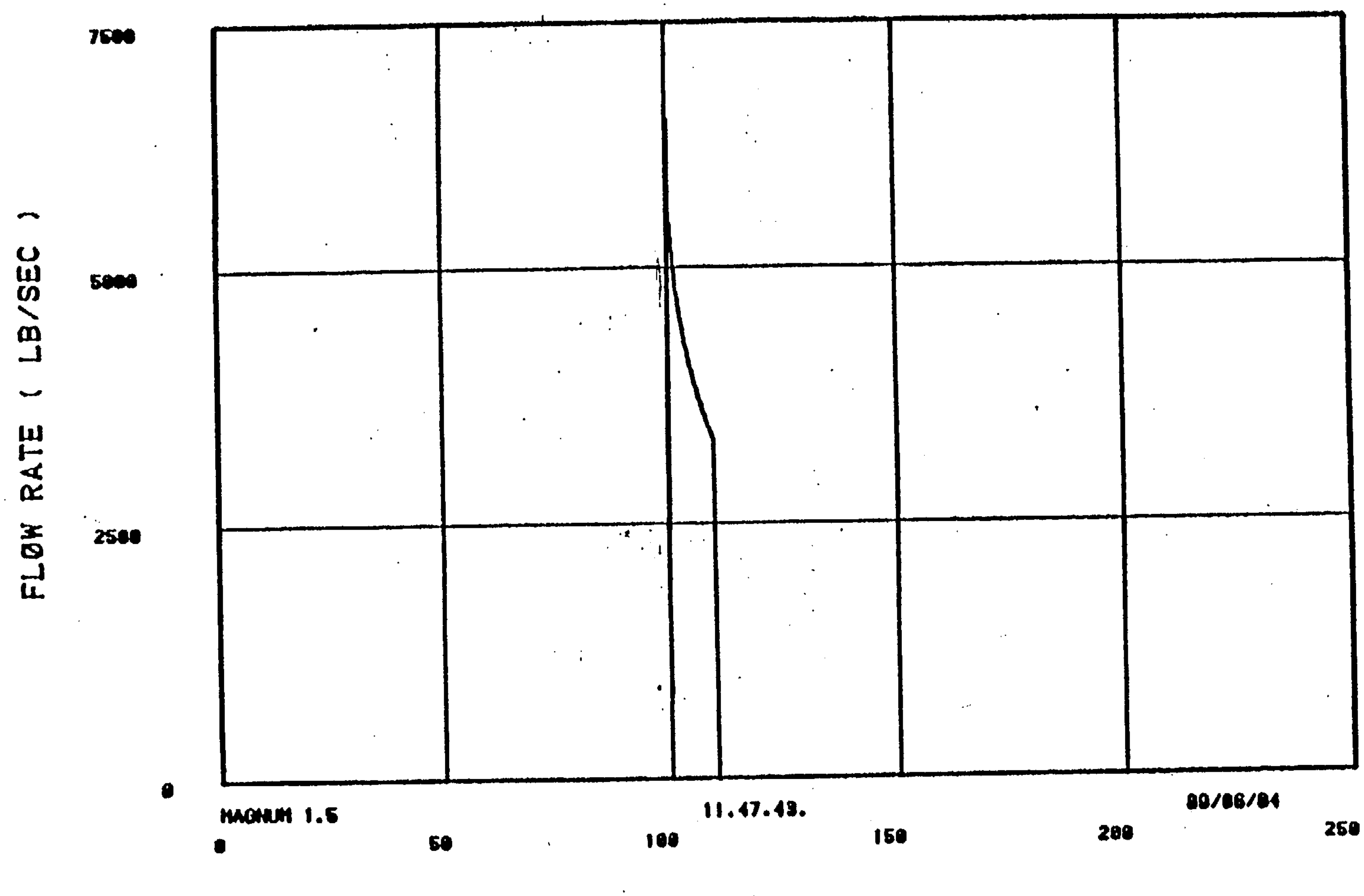
FIGURE A-14

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

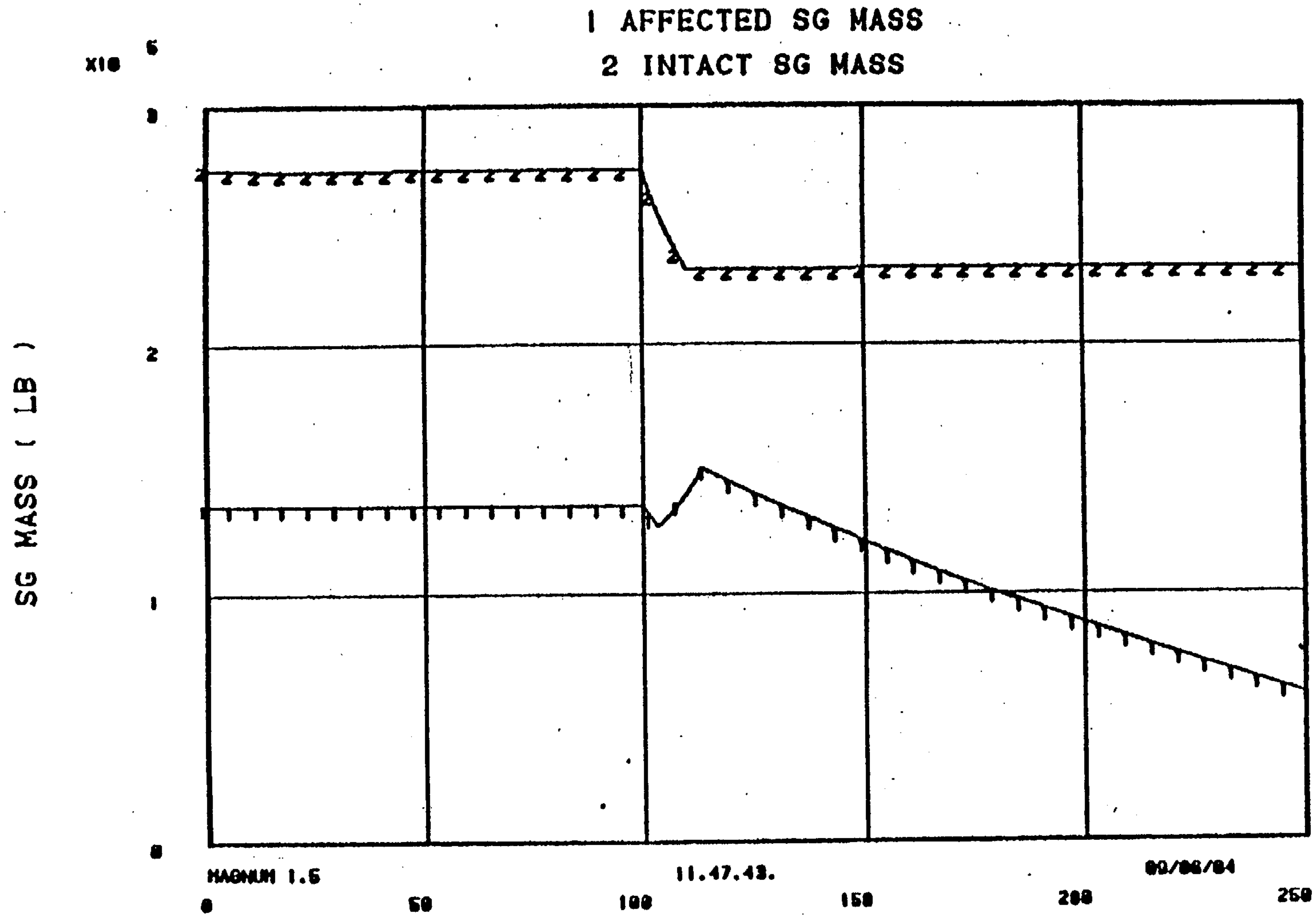


H.B.RØBINSØN CYCLE 10 RELØAD

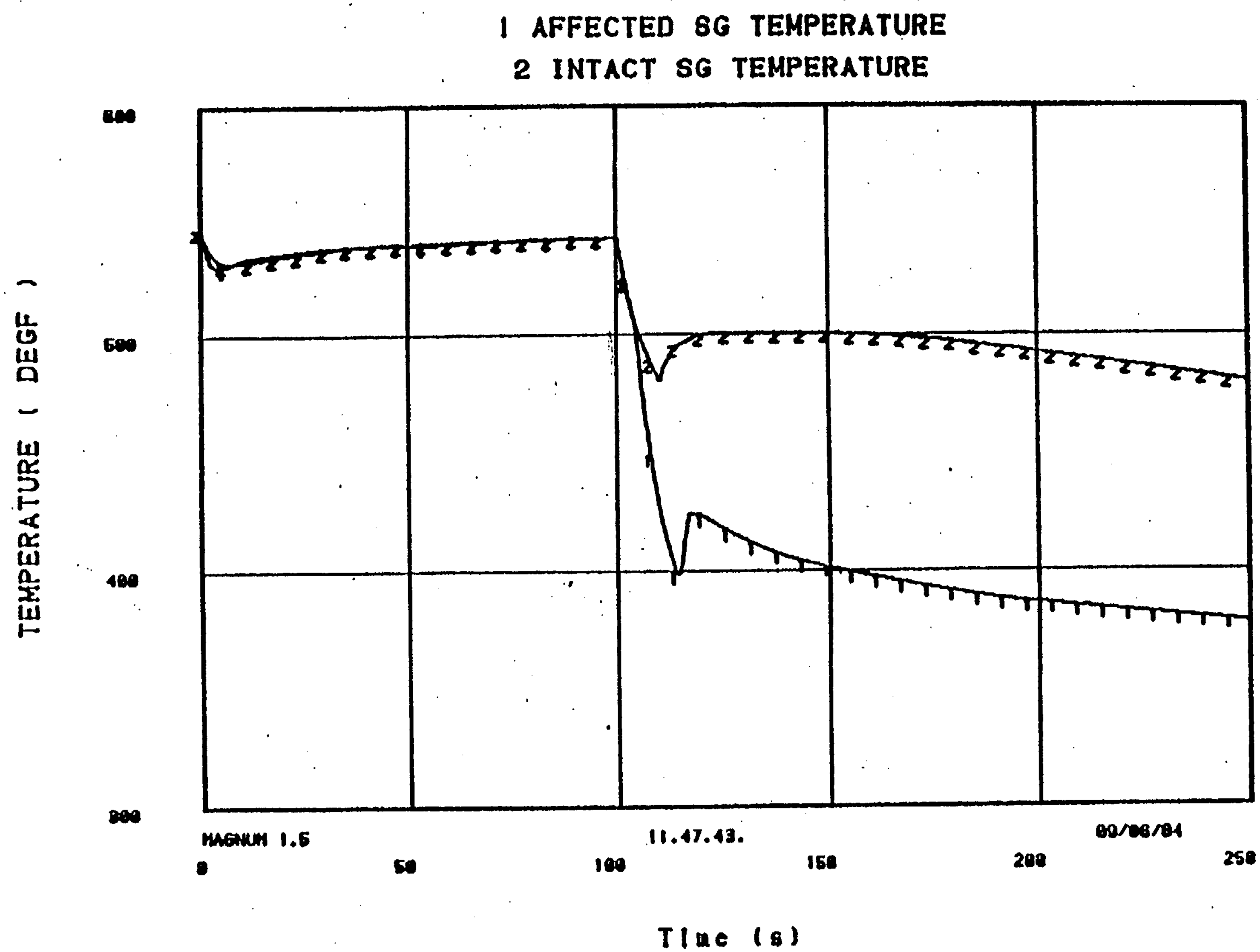
STEAM LINE BREAK ANALYSIS

BREAK DØWNSTREAM ØF FLØW RESTRICTØRS

FIGURE A-16



H.B.RØBINSØN CYCLE 10 RELØAD
STEAM LINE BREAK ANALYSIS
BREAK DØWNSTREAM ØF FLØW RESTRICTØRS
FIGURE A-17

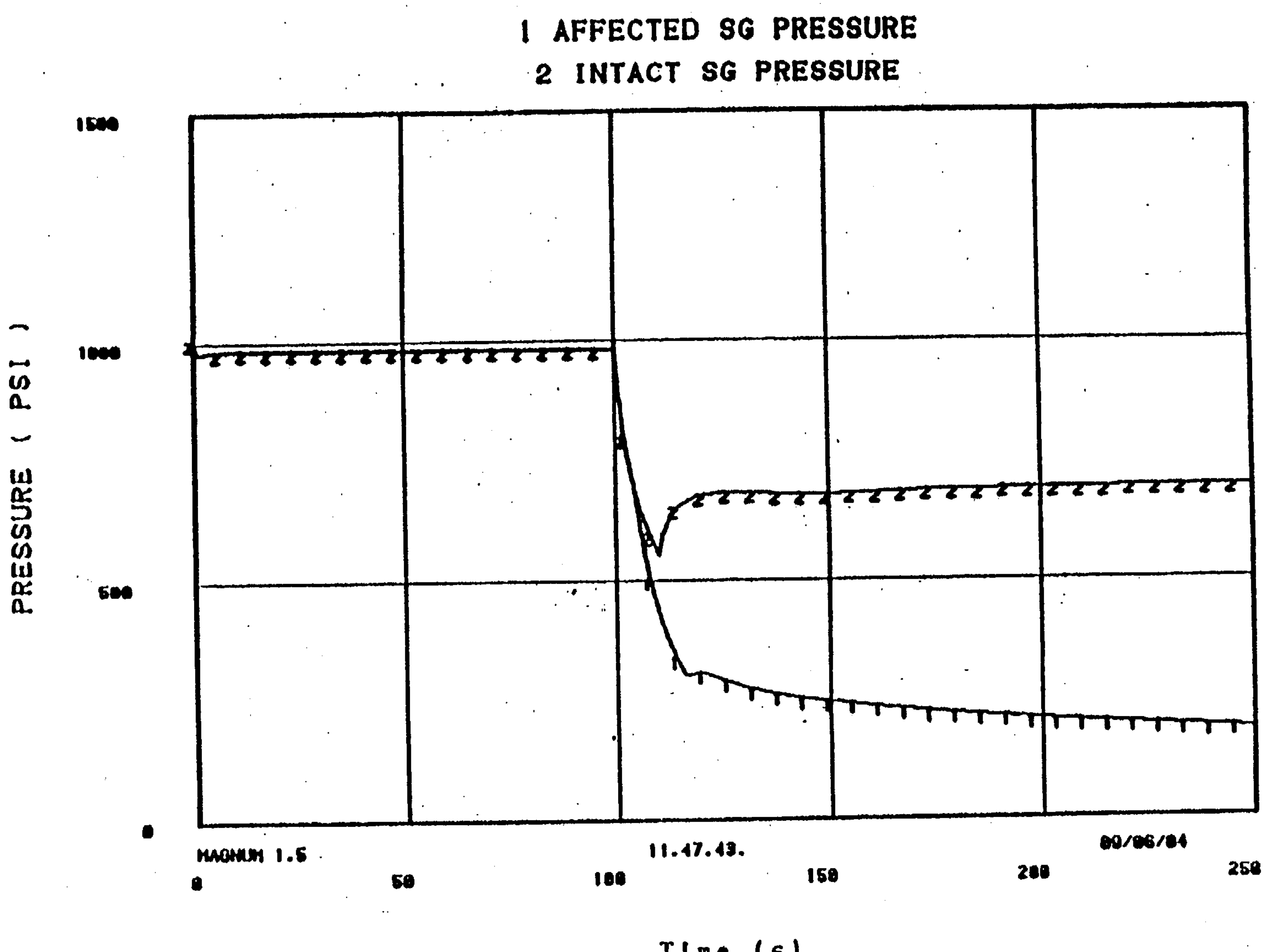


H.B.RØBINSØN CYCLE 10 RELØAD

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.

ATTACHMENT II

SAFETY EVALUATION REPORT

H. B. ROBINSON, UNIT 2, CYCLE 10

RELOAD ANALYSIS - LOCA ANALYSES

1.0 INTRODUCTION

During Cycle 9, H. B. Robinson Unit 2 (HBR-2) operated at reduced power and system temperature in order to improve operating conditions for the steam generators. For Cycle 10, the licensee, Carolina Power and Light (CP&L), has replaced the steam generators in order to allow a return to full power operation at 2300 MWT. In addition, for Cycle 10, the licensee has implemented a low radial leakage fuel management scheme in order to reduce vessel fluence. Peak assembly discharges are also being increased for HBR-2 to 44,000 MWD/MTU. As a result of the latter two changes, the total nuclear enthalpy rise factor ($F_{\Delta H}^{T}$) has been increased to 1.65.

To support these changes for Cycle 10 operation at HBR-2, the licensee has provided revised LOCA analyses in References 1 through 3. This SER presents our evaluation of these submittals. We first address the compliance of the ECCS evaluation model, utilized for these analyses, to the requirements of Appendix K to 10 CFR 50. We then evaluate the adequacy of the LOCA analyses performed to demonstrate compliance with 10 CFR 50.46.

2.0 Evaluation Model

The ECCS evaluation model utilized to perform the LOCA analysis for HBR-2 is the revised Exxon Nuclear Company (ENC) evaluation model. This model is called EXEM/PWR and is documented in references 4, 5 and 6. This model is currently under staff review and a more detailed SER on EXEM/PWR will be issued separately. This section documents our review of EXEM/PWR, as utilized for the

HBR-2 Cycle 10 LOCA analysis, and evaluates its conformance to the required features of Appendix K to 10 CFR 50.

EXEM/PWR contains several models updates to the currently approved ENC-WREM IIA PWR ECCS evaluation model, reference 7. The model updates for EXEM/PWR are shown in Table 1., Each of these changes is discussed separately below.

2.1 Fuel Rod Model-RODEX2 Code

The RODEX2 Code is documented in reference 8. The RODEX2 code is based upon the previously approved GAPEX code, reference 9. As part of the EXEM/PWR model, ENC uses the RODEX2 code to provide the initial fuel stored energy and fuel rod internal pressures utilized as inputs to various portions of the evaluation model.

The staff has previously reviewed and approved the RODEX2 code for LOCA applications. Our evaluation of this code is contained in reference 10. Specifically, we found that the RODEX2 code satisfies the requirements of Appendix K, section I.A.I.

2.2 Clad Swelling and Rupture Model

In reference 11, ENC proposed a revised clad swelling and rupture model. This model, which includes the data base of NUREG-0630, reference 12, is used in the RELAP4 and TOODEE2 codes.

The staff has previously reviewed this model for compliance with section I.B of Appendix K. As documented in reference 13, we found this model meets those requirements.

2.3 Revised Fuel Rod Model - RELAP4-EM Code

The RELAP4-EM code, used as part of the EXEM/PWR ECCS evaluation model, has been updated to make its fuel models consistent with the approved RODEX2 fuel performance code. These updates include gap conductance, internal rod pressure, fuel conductivity and radial power distribution and are described in reference 5.

We have reviewed the modifications to the RELAP4-EM fuel model updates and find them acceptable.

2.4 Split Break Model

Currently the REFLEX code only simulates a guillotine break configuration with a discharge coefficient of 1.0. This assumption is conservative for split breaks and guillotine breaks with discharge coefficients less than 1.0. As part of EXEM/PWR, the REFLEX code has been modified to allow modeling of split breaks and guillotine breaks with smaller discharge coefficients.

For modeling of split breaks, the REFLEX code has been modified to allow the fluid streams from the downcomer and steam generators to mix before leaving the break. A junction is then used to simulate the break path to containment.

Double-ended guillotine breaks with smaller discharge coefficients are simulated with the current REFLEX noding scheme. However, to account for the smaller discharge coefficient, an equivalent K-factor is used to simulate the increased break resistance.

We have reviewed these model changes and find them acceptable.

2.5 REFLEX Core Outlet Enthalpy Model

The currently approved REFLEX model uses a constant value for the core exit enthalpy. The core exit enthalpy used is determined at the upper plenum pressure and the fluid temperature corresponding to the steam generator secondary side saturation temperature. The core exit enthalpy model has been upgraded such that fluid enthalpy is calculated based upon an energy balance performed for the core.

The revised core outlet enthalpy model accounts for energy added to the fluid below the quench front, stored energy release as the quench front progresses, and energy added to the fluid above the quench front. To demonstrate the appropriateness of the model, ENC performed benchmarks of FLECHT tests 34711, 34610, and 31922, reference 14. These benchmarks showed good agreement to the data.

Based upon the benchmarks performed, and a detailed review of the equations utilized, we have concluded that this model is acceptable.

2.6 Steam Cooling Model

Section I.D.5 of Appendix K to 10 CFR 50 requires that a steam cooling model be utilized to predict heat transfer coefficients when flooding rates fall below one inch per second. In addition, the steam cooling model must take into account the effect of flow blockage relative to both local steam flow and heat transfer. EXXON developed, as part of their currently approved ENC WREM-IIA PWR ECCS evaluation model, a steam cooling model which fully complied with these requirements. However, recent experimental data in references 15 and 16 have shown that the currently approved Exxon steam cooling model is overly conservative. As a result, Exxon develop, and submitted as part of EXEM/PWR, a revised steam cooling model.

The revised steam cooling model calculates an equivalent steam flow for use in the TOODEE-2 (reference 17) energy solution which assures that superheated steam exits the core. This flow rate includes the effect of blockage based upon the currently approved flow divergence model of the ENC WREM-IIA PWR ECCS evaluation model.

The rod surface heat transfer coefficients are calculated by the following method. First, local unblocked heat transfer coefficients are calculated using an appropriate reflood heat transfer correlation for the fuel modeled. The local heat transfer coefficients are then modified to account for the effect of blockage on mass flux and hydraulic diameter. In addition, the heat transfer coefficients are adjusted to account for the effects of increased turbulence and breakup of entrained liquid droplets downstream of the blockage. The net effect of these modifications is a decrease in heat transfer downsteam of the flow blockage relative to that which would be obtained in an unblocked core. Calculations performed by Exxon with the revised steam cooling model indicate that peak cladding temperatures are approximately 15°F higher relative to that which would be obtained using the unblocked ENC-2 FLECHT coefficients.

The staff has reviewed the revised steam cooling model and finds it acceptable. Recent experimental data in reference 15 and 16, obtained with flooding rates below one inch per second, indicate that the effect of blockage is to enhance heat transfer, relative to an unblocked fuel assembly, downsteam of the blockage plane. Since the revised Exxon steam cooling model predicts decreased heat transfer, we find that the effect of flow blockage on local steam flow and heat transfer has been treated conservatively. Thus, the revised steam cooling model fully meets the requirements of section I.D.5 of Appendix K to 10 CFR 50.

2.7 FLECHT Heat Transfer Coefficients

As part of the EXEM/PWR ECCS evaluation model, revised FLECHT-based reflood heat transfer coefficients were proposed. These heat transfer coefficients were not used in the LOCA analyses performed for HBR-2 Cycle 10 operation. Rather, the currently approved WREM-IIA reflood heat transfer coefficient were utilized. We find this approach acceptable.

In performing the analyses, documented in reference 3, to verify the allowable linear heat generation rates versus axial location proposed for Cycle 10, the WREM-IIA reflood heat transfer coefficients were modified to account for axial power distribution effects. To account for the effects of axial power distribution, adjustments are made to both the REFLEX and TOODEE2 codes. These adjustments are made based upon conserving the integral power between the fuel rod and the FLECHT rod. The specific methodology employed is detailed in reference 6.

To demonstrate the appropriateness of their model, ENC benchmarked data for the FLECHT skewed profile low flooding rate heat transfer tests 11428, 14331 and 16110. These data were obtained from reference 21. The benchmarks showed that the method for adjusting for axial power distribution yielded higher cladding temperatures, and hence lower heat transfer coefficients, than observed in the FLECHT experiments. Thus, the method is conservative.

In addition to evaluating the information provided by ENC, we have reviewed some of the FLECHT data to assure that the ENC methodology is conservative. Comparisons were made between the FLECHT cosine tests 02114 and 03113 and the skewed power shape tests 15305 and 11003 using the proposed ENC method. These comparisons further showed that the ENC method is conservative. Thus, we find the adjustment to the FLECHT heat transfer coefficients to be acceptable.

2.8 Summary of EXEM/PWR Model Compliance

Based on the foregoing, we find that the EXEM/PWR evaluation model, as utilized to support Cycle 10 operation for HBR-2, is wholly in conformance with Appendix K to 10 CFR 50.

3.0 LOCA Analyses

To support Cycle 10 operation of HBR-2, the licensee submitted several LOCA analyses. In reference 1, the limiting break, based on previous HBR-2 LOCA analyses, was analyzed to demonstrate conformance to 10 CFR 50.46 for a peak rod burnup range up to 49,000 MWD/MTU. Since a new ECCS evaluation model, EXEM/PWR, was utilized for the analyses, the licensee provided, via reference 2, a break spectrum analysis to confirm that the limiting break remained the same. Finally, reference 3 provides verification that the allowable linear heat generation rates as a function of axial elevation satisfies the requirements of 10 CFR 50.46. Our evaluation of these submittals follow.

3.1 <u>Limiting Break Analysis</u>

An analysis of the limiting break, a double-ended guillotine cold leg break with a discharge coefficient of 0.8, was performed using the EXEM/PWR ECCS evaluation model. The analysis was performed using the following assumptions:

- -102% of the rated power level of 2300 MWT,
- -Steam generator tube plugging of 6%,
- -Peak linear heat generation rate of 14.16 KW/FT,
- -Total peaking factor, F_Q^T, of 2.36,
- -Enthalpy rise factor, $F_{\Delta H}^T$ of 1.65,
- -Peak assembly discharge exposure of 44,000 MWD/MTU,
- -Peak rod exposure of 49,000 MWD/MTU,
- -Single failure assumption of loss of one HPSI and one LPSI pump.

The results of the limiting break analysis are summarized in Table 2. As shown, the peak cladding temperature is 2042°F, local zirconium metal-water reaction is 4.65%, and whole core metal-water reaction is less than 1% for the worst case analyzed. Thus, the analysis demonstrates conformance with the requirements of 10 CFR 50.46.

We have reviewed the assumptions utilized within the licensee's analyses. The use of 102% of the rated power level satisfies the requirement of Appendix K, section I.A. The peaking factors utilized are consistent with HBR-2 Technical Specification 3.10.2.1. The single failure assumption utilized satisfies Appendix K, section D.1. To assure that the LOCA analysis covers fuel conditions for a burnup range up to 49,000 MWD/MTU peak rod exposure, a burnup sensitivity study was performed. Values analyzed were 2,000 MWD/MTU (highest stored energy), an EOL burnup of 49,000 MWD/MTU (highest internal rod pressure), and an intermediate burnup of 9,000 MWD/MTU. We find the burnups analyzed are sufficient to demonstrate conformance to 10 CFR 50.46 for rod exposure up to 49,000 MWD/MTU in HBR-2.

Based on the foregoing, we find that the limiting break for HBR-2 complies with the requirements of 10 CFR 50.46

3.2 Break Spectrum Analysis

The LOCA analyses performed for Cycle 10 operation of HBR-2 utilized the EXEM/PWR ECCS evaluation model. As this was the first application of EXEM/PWR ECCS evaluation model for HBR-2, the licensee provided, in reference 2, a break spectrum analysis to confirm that the limiting break had not been changed. The analysis was performed using the same assumptions employed in the limiting break analysis described above except that only the worst case burnup, 2,000 MWD/MTU, was analyzed. The results of the analysis are summarized on Table 3 and demonstrates comformance to 10 CFR 50.46. As shown, the analysis demonstrated that the limiting break remained the double-ended guillotine cold leg break with a discharge coefficient of 0.8.

We find that the break spectrum analysis performed for Cycle 10 operation of HBR-2 satisfies Appendix K, Section C.1. Thus, conformance to 10 CFR 50.46 has been demonstrated for the entire break specture.

3.3 K(z) Curve

To define allowable linear heat generation rates as a function of core elevation, HBR-2 utilizes the K(z) curve. This curve, which is given in Figure 3.10-3 of the HBR-2 Technical Specifications, defines the normalized peaking factor, relative to the total peaking factor, F_Q^T of 2.32, as a function of elevation. To confirm that the linear heat generation rates allowed by the K(z) curve satisfies the requirements of 10 CFR 50.46, the licensee submitted additional LOCA analyses in reference 3.

The K(z) curve analyses were as performed for the limiting break and utilized the same input assumptions and model described above except for two areas. First, in order to examine linear generation rates in the upper portion of the core, the axial power shape was modified from a chopped cosine to a shape which peaked at 9 feet. The peaking factors utilized at and above the 9 foot were chosen to give the same peaking factor as that allowed by the K(z) curve. Secondly, the model utilized included the EXEM/PWR methodology; documented in reference 6, which adjusts the reflood heat transfer coefficients for axial power distribution effects.

A summary of the analyses provided in reference 3 is given on Table 4. Since the proposed K(z) curve is burnup dependent, two evaluations were performed using the axial power shape peaked at 9 feet in order to cover the different burnup ranges. The inputs utilized for each of the burnup ranges were chosen to maximize peak cladding temperature. As shown in the table, both cases yielded peak cladding temperatures less than the 2200°F criteria of 10 CFR 50.46. In addition both the local zirconium metal-water reaction and whole core metal water reaction are less than the criteria specified by 10 CFR 50.46.

We have reviewed the licensee's analysis and have concluded that the K(z) curve limits the allowed linear heat generation rates such that the requirements of 10 CFR 50.46 are met.

4.0 SUMMARY

Based upon the analyses provided by the licensee in references 1 through 3, we have concluded that the LOCA analyses performed for Cycle 10 of HBR-2 satisfies the requirements of 10 CFR 50.46 and that the evaluation model utilized satisfies the requirements of Appendix K to 10 CFR 50.

TABLE 1

ECCS Model Updates of EXEM/PWR

- ° Fuel Rod Model - RODEX 2
  - ° Stored Energy
  - ° Fission Gas Release

- ° Blowdown Model - RELAP4-EM Code
  - ° NUREG-0630 Clad Rupture Blockage Model
  - ° Modified Fuel Rod Model

- ° Reflood Model - REFLEX Code
  - ° Leakage Flow From Upper Plenum to Downcomer*
  - ° Split Break Model
  - ° Core Outlet Enthalpy Model
  - ° Revised Carryout Rate Fraction Correlation*

- ° Heatup Model - TOODEE2 Code
  - ° 17 x 17 FLECHT Heat Transfer Correlation*
  - ° Revised Steam Cooling Model
  - ° NUREG-0630 Clad Rupture Blockage Model
  - ° Adjustments to FLECHT Heat Transfer Coefficient

*Not used in HBR-2 Cycle 10 LOCA Analysis

Table 2

HBR-2 Limiting Case LOCA Analyses

(Double-Ended Guillotine Cold Leg Break, Discharge Coefficient = 0.8)

Analysis Results	2 MWD/MTU Peak Rod Exposure	9 MWD/MTU Peak Rod Exposure	49 MWD/MTU Peak Rod Exposure
Peak Clad Temperature (PCT), °F	2042	1815	1785
Peak Clad Temperature Reached, sec	60	139	139
Peak Clad Temperature Elevation, ft	6	8.5	8.5
Local Zr/H_2O Reaction (max.), %	4.65	1.93	1.72
Total H_2 Generation, % of Total Zr Reacted	<1	<1	<1

TABLE 3

H. B. Robinson Unit 2

Large Break Spectrum Results

<u>Calculational Basis</u>

License Core Power, MWt	2300
Power Used for Analysis, MWt	2346
Peak Linear Power for Analysis, kW/ft	14.16
Total Peaking Factor, F_Q^T	2.32
Enthalpy Rise, Nuclear $F_{\Delta H}^T$	1.65
Steam Generator Tube Plugging (%)	6.00

	*(CD = 1.0) ** DECLG	(CD = 0.8) DECLG	(CD = 0.6) DECLG
Peak Cladding Temperature °F	1885	2042	1808
Peak Temperature Location, ft	6.0	6.0	8.5
Local Zr/H_2O Reaction (Max.), %	2.70	4.65	2.18
Local Zr/H_2O Location, ft	6.0	6.0	6.0
Local Zr/H_2O	<1%	<1%	<1%
Hot Rod Burst Time, sec	39.66	39.9	46.40
Hot Rod Burst Location, ft	6.0	6.0	6.0

*CD = Discharge Coefficient
**DECLG = Double-Ended Cold Leg Guillotine

Table 4

H. B. Robinson Unit 2 K(Z) Determination Results

<u>Calculational Basis</u>

License Core Power, MWt	2300
Power Used for Analysis, MWt	2346
Break Size, DECLG	0.8
Enthalpy Rise, Nuclear, $F^T H$	1.65
Steam Generator Tube Plugging, %	6.00

	Peaked at 6 feet	Peaked at 9 feet	Peaked at 9 feet
Hot Rod Exposure Range, MWD/kgU	0 - EOL	0 - 9	9 - EOL
Peak Linear Heat Generation Rate (LHGR)	14.16	12.39	12.57
Total Peaking Factor, $F_Q T$	2.32	2.03	2.06
Peak Cladding Temperature, °F	2042	2197	2183
Peak Temperature Location, ft	6.0	10.75	10.75
Local Zr/H_2O Location, ft	6.0	10.75	10.75
Local Zr/H_2O Reaction (Max.), %	4.65	6.19	5.89
Total Zr/H_2O	<1%	<1%	<1%
Hot Rod Burst Time, sec	39.9	49.37	51.57
Hot Rod Burst Location, ft	6.0	9.0	9.0

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