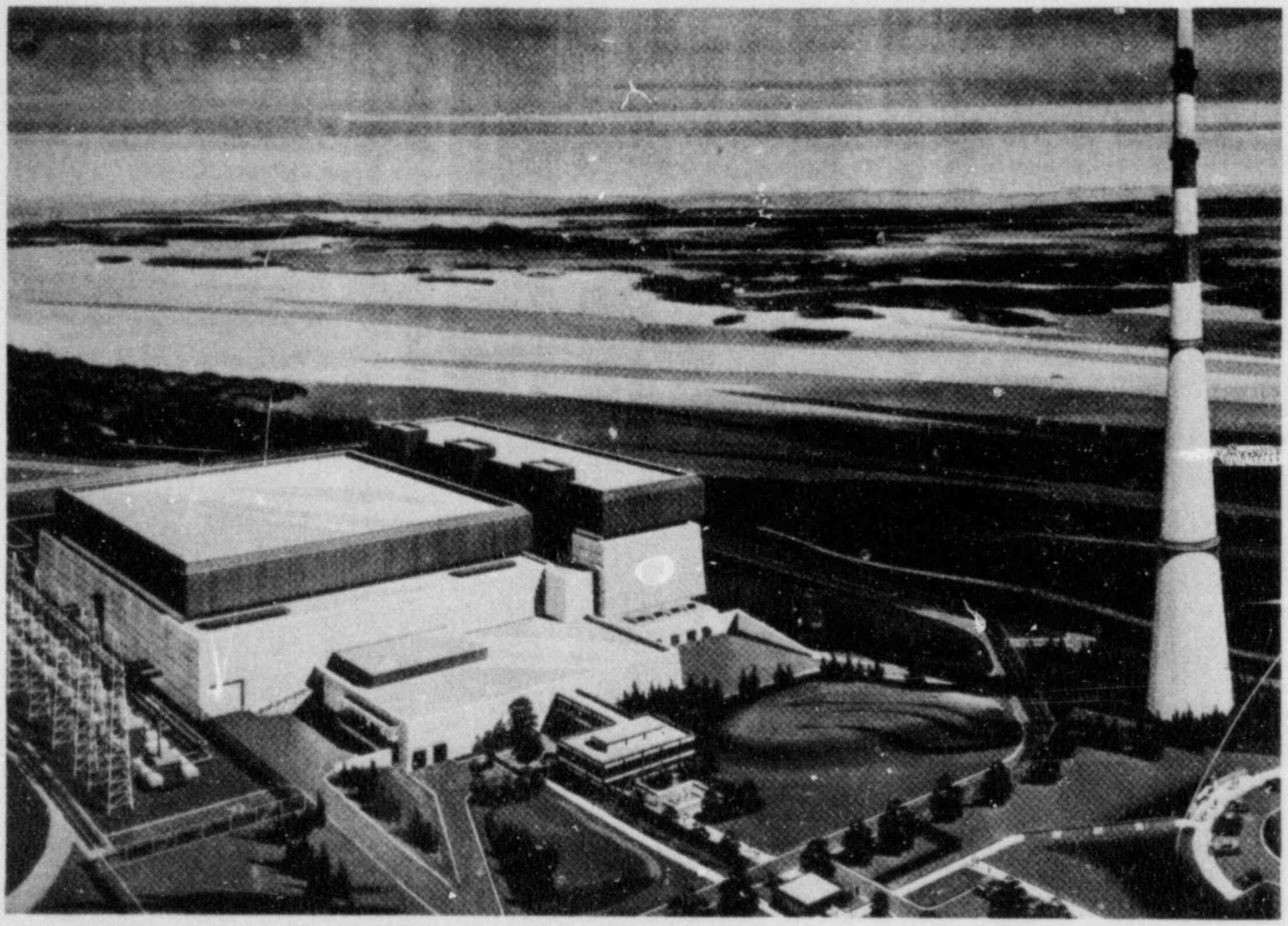
TVA-RLR-002

BROWNS FERRY NUCLEAR PLANT
RELOAD LICENSING REPORT

UNIT 2, CYCLE 6



Tennessee Valley Authority

TVA-RLR-002

RELOAD LICENSING REPORT

FOR

BROWNS FERRY UNIT 2, CYCLE 6

TENNESSEE VALLEY AUTHORITY

July 1984

I. Introduction

This reload licensing report presents the results of the core design and safety analyses performed for Browns Ferry unit 2, cycle 6 operation. The methodology and technical bases employed in the performance of these analyses are discussed in references 1-6.

Items specifically addressed here include the nuclear fuel assemblies and core loading to be used in cycle 6, the reload core nuclear design characteristics, the transient and accident safety analysis results, and the proposed operating thermal limits.

The cycle 6 reload core will include four Westinghouse QUAD+ demonstration assemblies located in nonlimiting core peripheral locations. A complete description of the demonstration assemblies is contained in Westinghouse Report WCAP-10507 (reference 8).

II. Reload Cycle Information

A. Design Basis Exposures

1. Projected cycle 5 core average exposure at end of cycle: 20.5 GWd/ST
2. Minimum cycle 5 core average exposure at end of cycle from cold shutdown considerations: 19.5 GWd/ST
3. Assumed cycle 6 core average exposure at depletion of reactivity (DOR)*: 17.4 GWd/ST

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*DOR - End of full power capability

B. Reload Fuel Assemblies

Fuel Type	Cycle Loaded	Number
Irradiated		
P8DRB284L,R2	3	15
P8DRB284L,R3	4	201
P8DRB265H,R4	5	80
P8DRB284L,R4	5	168
New		
P8DRB284L,R5	6	296
QUAD+ Demo	6	4
TOTAL		764

Descriptions of the nuclear and mechanical design of the General Electric irradiated and new fuel assemblies to be loaded in cycle 6 are contained in reference 7. The nuclear, mechanical, and thermal-hydraulic design descriptions for the Westinghouse demonstration assemblies are contained in reference 8.

C. Reference Core Loading Pattern

The reference loading pattern is the basis for all reload licensing and operational planning and is comprised of the fuel assemblies designated in item II.B of this report. It is based on the best possible prediction of the core condition at the end of the previous cycle and on the desired core energy capability for the reload cycle. The reference loading pattern is designed with the intent that it will represent, as closely as possible, the actual core loading pattern. Figure 1 shows the reference core loading pattern for cycle 6.

The reference loading pattern includes four Westinghouse QUAD+ demonstration assemblies loaded in peripheral locations. Evaluations performed by Westinghouse (reference 8) indicate that the results of licensing analyses for the lead P8x8R fuel assembly bound those for the QUAD+ demonstration assemblies. Cycle specific analyses performed by TVA confirm this conclusion. The results documented in this report are for the limiting loading pattern.

D. Special Conditions

The use of increased core flow (ICF) is planned for cycle 6 operation. Safety analyses were performed for both 100 percent and 105 percent of rated core flow with the most conservative results used for determining the operating limits. The conclusions regarding LOCA analysis, reactor internals pressure drop, and flow-induced vibration as discussed in reference 9 are applicable to cycle 6. The flow-biased instrumentation for the rod block monitor will be signal clipped for a setpoint of 106 percent since flow rates higher than rated would otherwise result in a ΔCPR higher than reported for the rod withdrawal error.

III. Nuclear Design Characteristics

A. Shutdown Margin

The reference core is analyzed in detail to ensure that adequate shutdown margin exists. This section discusses the results of core calculations for shutdown margin (including the liquid poison system).

1. Core Effective Multiplication and Control Rod Worth

Core effective multiplication and control rod worths were calculated using the TVA BWR simulator code (references 2 and 4) in conjunction with the TVA lattice physics data generation code (references 3 and 4) to determine the core reactivity with all rods withdrawn and with all rods inserted. A tabulation of the results is provided in table 1. These three eigenvalues (effective multiplication of the core, uncontrolled, fully controlled, and with the strongest rod out) were calculated at the beginning-of-cycle 6 core average exposure corresponding to the minimum expected end-of-cycle 5 core average exposure. The core was assumed to be in a xenon-free condition.

Cold k_{eff} was calculated with the strongest control rod out at various exposures through the cycle. The value R is the difference between the strongest rod out k_{eff} at BOC and the maximum calculated strongest rod out k_{eff} at any exposure point. The maximum strongest rod out k_{eff} at any exposure point is equal to or less than:

$$\text{Maximum } k_{eff}^{SRO} = k_{eff}^{SRO} \text{ (BOC)} + R$$

2. Reactor Shutdown Margin

Technical Specifications require that the refueled core must be capable of being made subcritical with 0.38-percent Δk margin in the most reactive condition throughout the subsequent operating cycle with the most reactive control rod in its full out position and all other rods fully inserted. The shutdown margin is determined by using the BWR simulator code to calculate the core multiplication at selected exposure points with the strongest rod fully withdrawn. The shutdown margin for the reloaded core is obtained by subtracting the maximum k_{eff}^{SRO} from the critical k_{eff} of 1.0, resulting in a calculated minimum cold shutdown margin of 1.1-percent Δk for Browns Ferry 2, cycle 6.

Table 1

CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL ROD WORTHS - NO VOIDS, NO XENON, 20°C

Uncontrolled, K_{eff}^{UNC} (BOC)	1.116
Fully Controlled, K_{eff}^{CON} (BOC)	0.953
Strongest Control Rod Out, K_{eff}^{SRO} (BOC)	0.980
R, Maximum Increase in Cold Core Reactivity With Exposure Into Cycle, Δk	0.009

3. Standby Liquid Control System

The standby liquid control system (SLCS) is designed to provide the capability of bringing the reactor, at any time in a cycle, from full power and a minimum control rod inventory (which is defined to be at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon-free state.

The SLCS shutdown margin is determined by using the BWR simulator code to calculate the core multiplication for the cold, xenon-free, all rods out conditions at the exposure point of maximum cold reactivity with the soluble boron concentration given in the Technical Specifications. The resulting k-effective is subtracted from the critical k-effective of 1.0 to obtain the SLCS shutdown margin. The results of the SLCS evaluation are given in table 2.

Table 2

STANDBY LIQUID CONTROL SYSTEM CAPABILITY

PPM	Shutdown Margin (Δk) (20°C, Xenon Free)
600	0.018

B. Reactivity Coefficients

The reactivity coefficients associated with the nuclear design of Browns Ferry 2, cycle 6 are implicit in the 1-D cross sections used for the safety analyses. As such, reactivity coefficients are not separately calculated for input to the transient analyses. However, a void coefficient is generated in the 3-D to 1-D cross section collapsing process and is used as a verification check. For Browns Ferry 2, cycle 6 the following results for DOR conditions were obtained:

100% core flow -	-0.0742	%Δk/%void
105% core flow -	-0.0745	%Δk/%void

C. Fuel Performance

The Browns Ferry 2, cycle 6 fuel performance is predicted by projecting the fuel burnup to the end of cycle with the 3-D simulator code. The calculated peak pellet exposures for the various fuel types are less than the limits specified in references 7 and 8. Furthermore, peak linear heat rates satisfy the assumptions made in the fuel vendors' thermal-mechanical integrity analyses (references 7 and 8). All fuel types loaded in cycle 6 are predicted to operate within these bounding assumptions. Additionally, the QUAD+ demonstration assemblies are predicted to have greater than 20-percent margin to the lead P8x8R assembly in steady state bundle power and thermal limits throughout cycle 6.

IV. Transient Analyses

A. Pressurization Events

The RETRAN computer code (reference 10) is used to analyze both the reactor system and hot channel responses during core-wide pressurization transients. The analytic models used in these analyses are described in reference 5. A description of the CPR correlation and its application to Browns Ferry is contained in reference 11. Analyses are performed for the potentially limiting events at the most adverse initial conditions expected during the cycle. Reload unique initial conditions and transient analyses results are summarized in the following tables.

NSSS Initial Conditions

Exposure	Steam Flow (%Rated)	Core Flow (% Rated)	Gap Conductance (BTU/ft^2-hr-°F)
DOR	105	105	650

Hot Channel Initial Conditions (Limiting Event)

Fuel Type	ICPR	Bundle Power (MW)	Bundle Flow (Klb/hr)	R-Factor	Gap Conductance (BTU/ft^2-hr-°F)
P8X8R	1.301	6.388	123.9	1.051	1287

Pressurization Event Analysis Results

Transient	Peak Power (% Rated)	Peak Heat Flux (% Rated)	Peak Vessel Press. (psia)	ΔCPR P8x8R	System Response
Load Rejection w/o Bypass	403.2	121.1	1235.0	0.231	Figures 2-5
Feedwater Controller Failure	234.0	115.3	1214.0	0.150	Figures 6-9

B. Nonpressurization Events

The nonpressurization events analyzed for reload licensing are either steady state events or relatively slow transients that can be analyzed in a quasi-static manner using a 3-D BWR simulator (reference 2). The methods used to analyze these events are described in reference 1. Results are summarized below.

Nonpressurization Event Analysis Results

Event	ΔCPR[3] P8x8R/QUAD+	Peak LHGR(kW/ft)[3] P8x8R/QUAD+
Loss of Feedwater Heating (100°F)	0.18	17.7
Rod Withdrawal Error	0.17[1]	17.2
Rotated Bundle Error	0.15[2]	15.3
Mislocated Bundle Error	0.18	14.6

[1] For increased core flow based on a signal clipped rod block setpoint of 106 percent.

[2] Includes 0.02 penalty required when using the varible water gap method (reference 7).

[3] Results presented were calculated for the P8x8R fuel type and conservatively bound the results calculated for the QUAD+ demonstration assemblies.

C. Overpressure Protection

The main steamline isolation valve closure with failure of direct scram is analyzed to demonstrate sufficient overpressure protection (peak vessel pressure must be less than 110 percent of design pressure - 1390 psia). The event is analyzed using the models and methods described in reference 5. Results are summarized below.

MSIV Closure (Flux Scram) Results

Peak Vessel Pressure (psia)	Peak Steamline Pressure (psia)	System Response
1283.0	1242.0	Figures 10-13

V. MCPR Operating Limit Summary

The methods used to determine the required OLMCPR values for each event analyzed are described in references 1 and 5. The application of Option A and B limits in determining the cycle OLMCPR is described in the unit Technical Specifications. Results are summarized below and in figure 14.

OLMCPR for Pressurization Events (BOC6-EOC6)

	Option A[1] P8x8R/QUAD+	Option B[1] P8x8R/QUAD+
Load Rejection Without Bypass (GLRWOB)	1.35	1.26
Feedwater Controller Failure (FWCF)	1.27	1.23

OLMCPR for Nonpressurization Events (BOC6-EOC6)

	P8x8R/QUAD+[1]
Loss of Feedwater Heaters (LFWH)	1.25
Rod Withdrawal Error (RWE)	1.24
Rotated Bundle Error (RBE)	1.22
Mislocated Bundle Error (MBE)	1.25

[1] Results presented were calculated for P8x8R fuel types. The QUAD+ demonstration assemblies will be loaded into nonlimiting core locations and monitored to the same OLMCPR.

VI. Accident Analyses

A. Loss of Coolant Accident (LOCA)

LOCA analysis results for fuel types previously loaded in unit 2 are described in reference 12. Reference 8 indicates that the MAPLHGR limits for fuel type P8DRB284L can be conservatively applied to the QUAD+ demonstration assemblies. These limits are presented below.

LOCA Limits for QUAD+ Demonstration Assemblies

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.3
5,000	11.8
10,000	12.0
15,000	12.0
20,000	11.8
25,000	11.2
30,000	10.8
35,000	10.0
40,000	9.4

B. Rod Drop Accident (RDA)

The methodology used to analyze the rod drop accident is described in appendix A of reference 6. Results for BF2, cycle 6 are summarized below.

Results for the Limiting RDA

Condition: COLD (68°F), EOC Exposure
Rod Worth: 1.33% Δk
Rod Position: 22-07
Peak Fuel Enthalpy: 245 cal/gm
Core Response: Figures 15-18

VII. Stability Analyses

The methodology used to analyze core and channel stability is described in appendix B of reference 6. The minimum stability margin occurs at the intersection of the natural circulation line and the 105-percent rod line (the flow biased scram line also passes through this point). Results for BF2, cycle 6 are summarized below and in figure 19.

Stability Analysis Results at Limiting Initial Conditions

Analysis	Maximum Decay Ratio
Core Stability	0.85
Channel Stability P8x8R/QUAD+	0.59[1]

[1] Results presented are for the P8x8R fuel type and conservatively bound the QUAD+ demonstration assemblies.

References

1. TVA-EG-047, 'TVA Reload Core Design and Analysis Methodology for the Browns Ferry Nuclear Plant,' Tennessee Valley Authority, January 1982.

2. TVA-TR78-03A, 'Three-Dimensional LWR Core Simulation Methods,' Tennessee Valley Authority, January 1979.

3. TVA-TR78-02A, 'Methods for the Lattice Physics Analysis of LWRs,' Tennessee Valley Authority, April 1978.

4. TVA-TR79-01A, 'Verification of TVA Steady-State BWR Physics Methods,' Tennessee Valley Authority, January 1979.

5. TVA-TR81-01, 'BWR Transient Analysis Model Utilizing the RETRAN Program,' Tennessee Valley Authority, December 1981.

6. TVA-RLR-001, 'Reload Licensing Report for Browns Ferry Unit 3, Cycle 6,' Tennessee Valley Authority, January 1984.

7. NEDE-24011-P-A-6, 'General Electric Standard Application for Reactor Fuel,' General Electric, April 1983.

8. WCAP-10507, 'QUAD+ Demonstration Assembly Report,' Westinghouse Electric Corporation, March 1984.

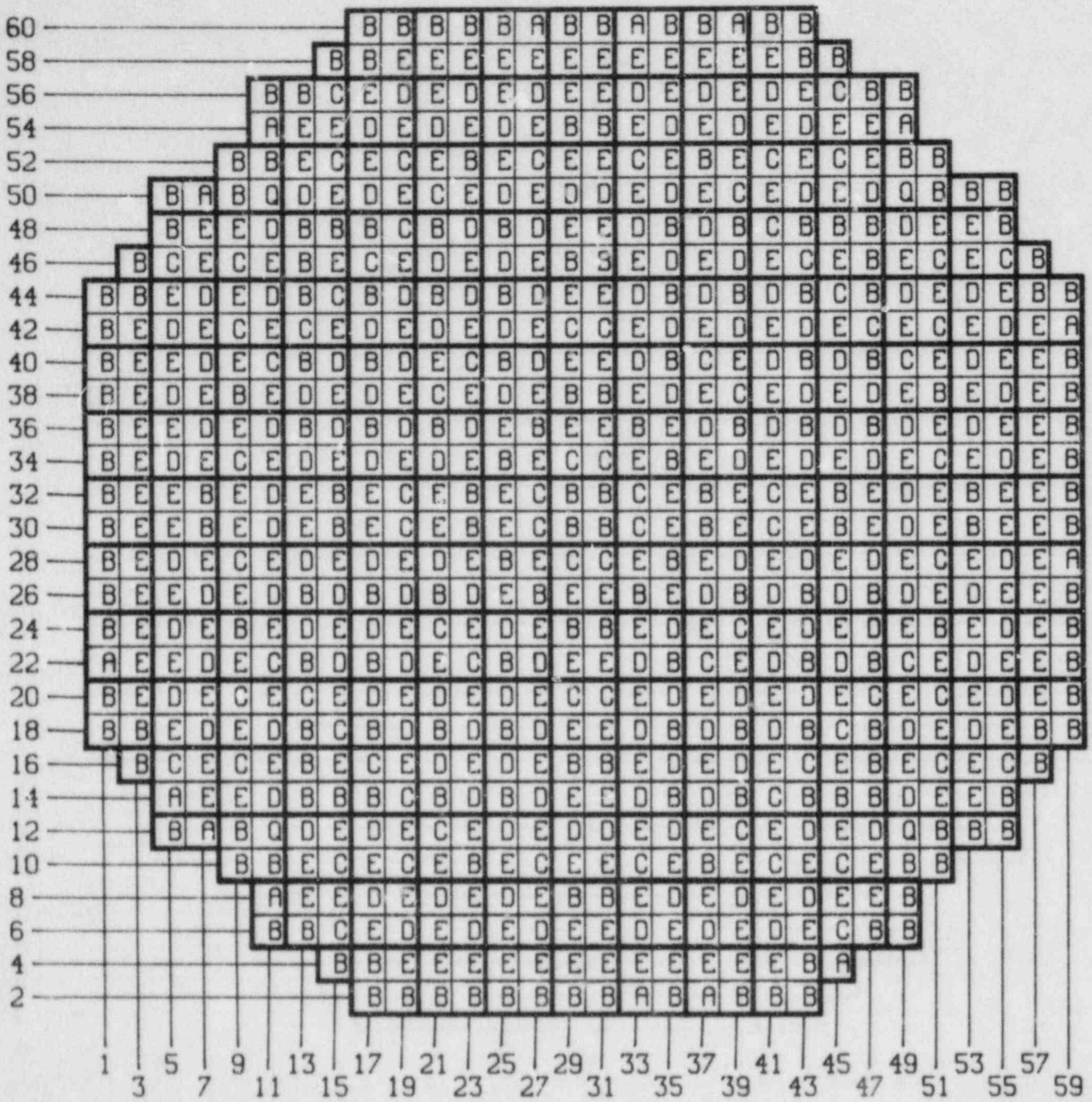
9. NEDO-22245, 'Safety Review of Browns Ferry Nuclear Plant Unit No. 2 at Core Flow Conditions Above Rated Core Flow During Cycle 5,' General Electric, October 1982.

10. EPRI NP-1850-CCM, 'RETRAN02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems,' Electric Power Research Institute, May 1981.

11. NEDE-24273, 'GEXL Correlation Application to TVA Browns Ferry Nuclear Power Station,' General Electric.

12. NEDO-24088-1 (as amended), 'Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 2,' General Electric, February 1978.

FIGURE 1
REFERENCE CORE LOADING PATTERN
BROWNS FERRY UNIT 2 -- CYCLE 6



FUEL TYPES

A-P8DRB284L,R2 (15)

B-P8DRB284L,R3 (201)

C-P8DRB265H,R4 (80)

D-P8DRB284L,R4 (168)

E-P8DRB284L,R5 (296)

Q-QUAD+ DEMO,R5 (4)

FIGURE 2 GENERATOR LOAD REJECTION W/O BYPASS

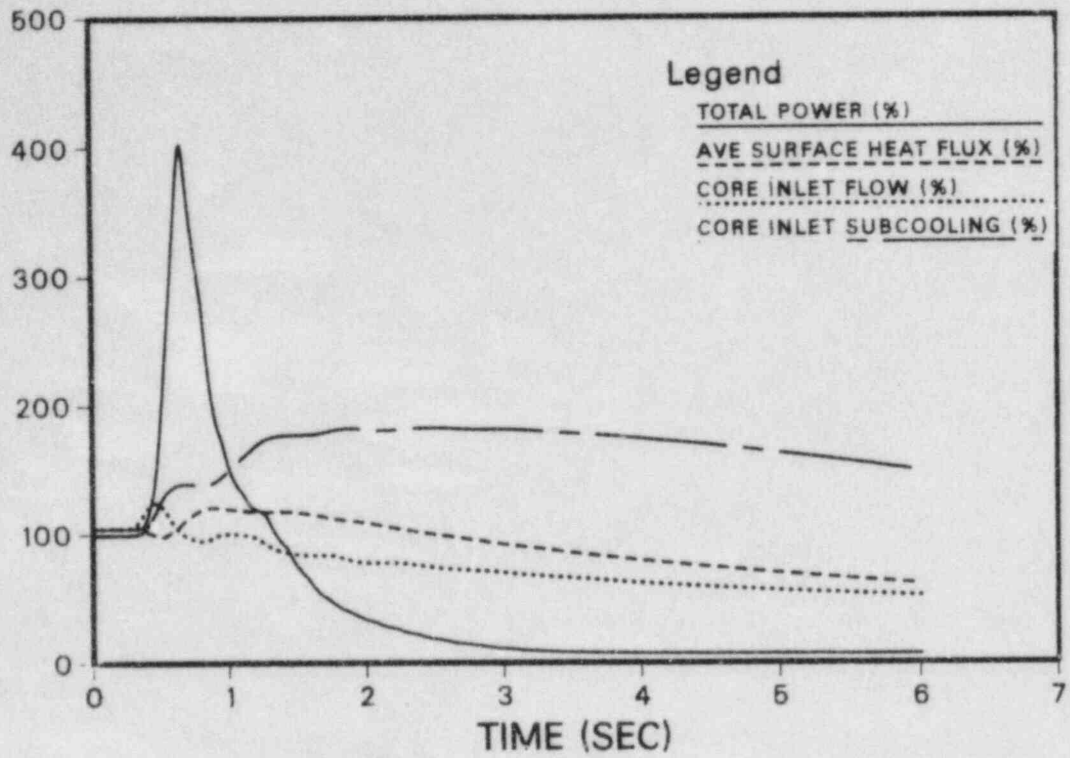


FIGURE 3 GENERATOR LOAD REJECTION W/O BYPASS

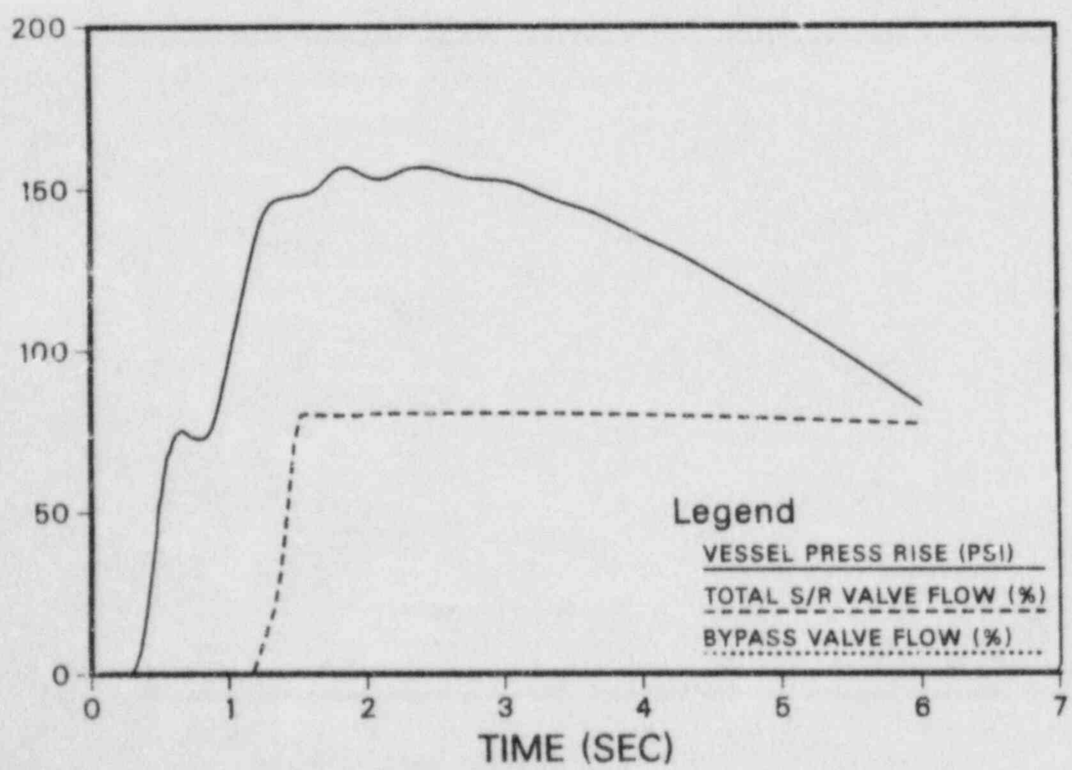


FIGURE 4 GENERATOR LOAD REJECTION W/O BYPASS

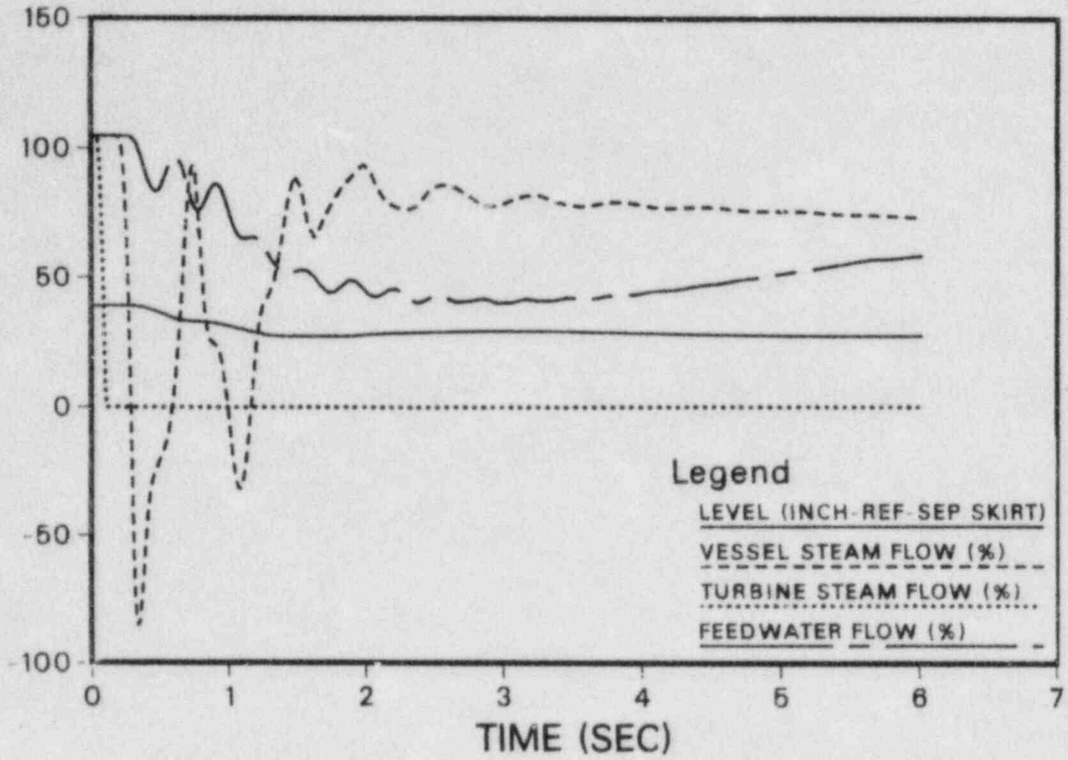


FIGURE 5 GENERATOR LOAD REJECTION W/O BYPASS

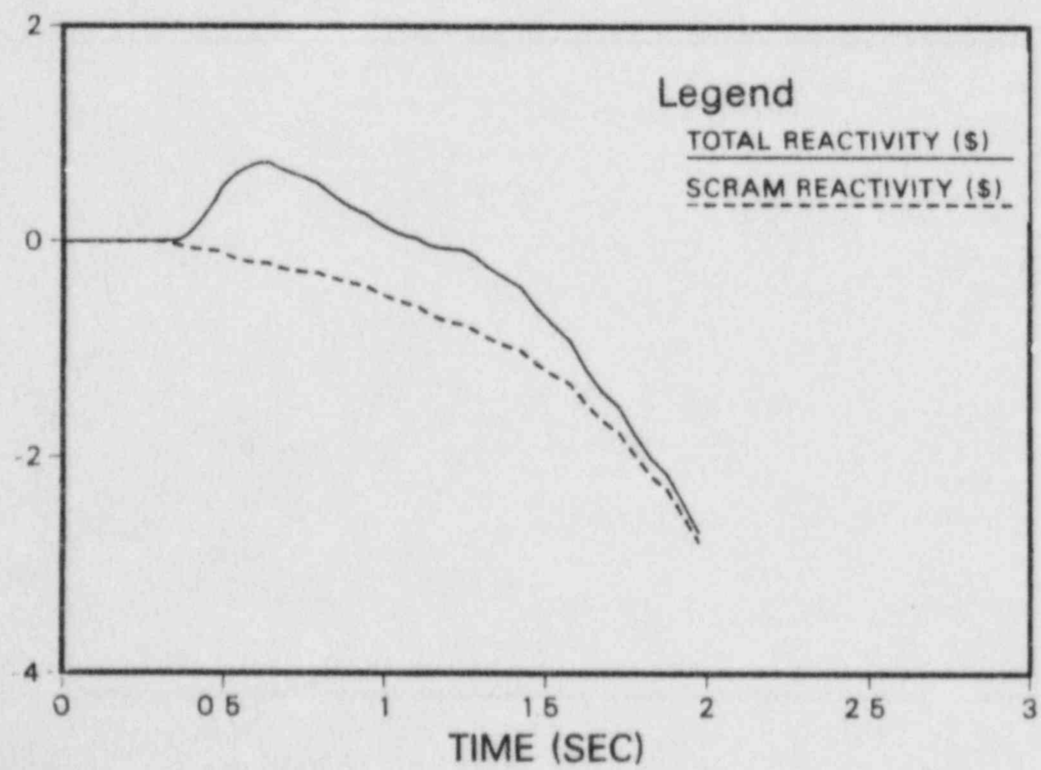


FIGURE 6 FEEDWATER CONTROLLER FAILURE

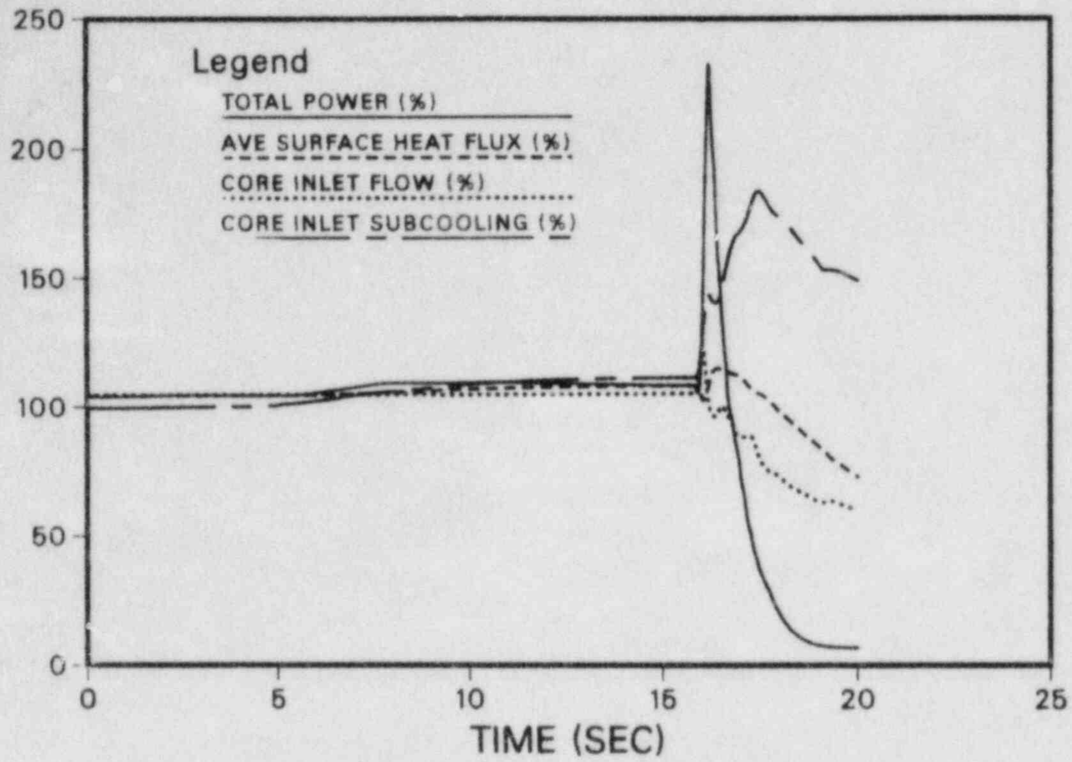


FIGURE 7 FEEDWATER CONTROLLER FAILURE

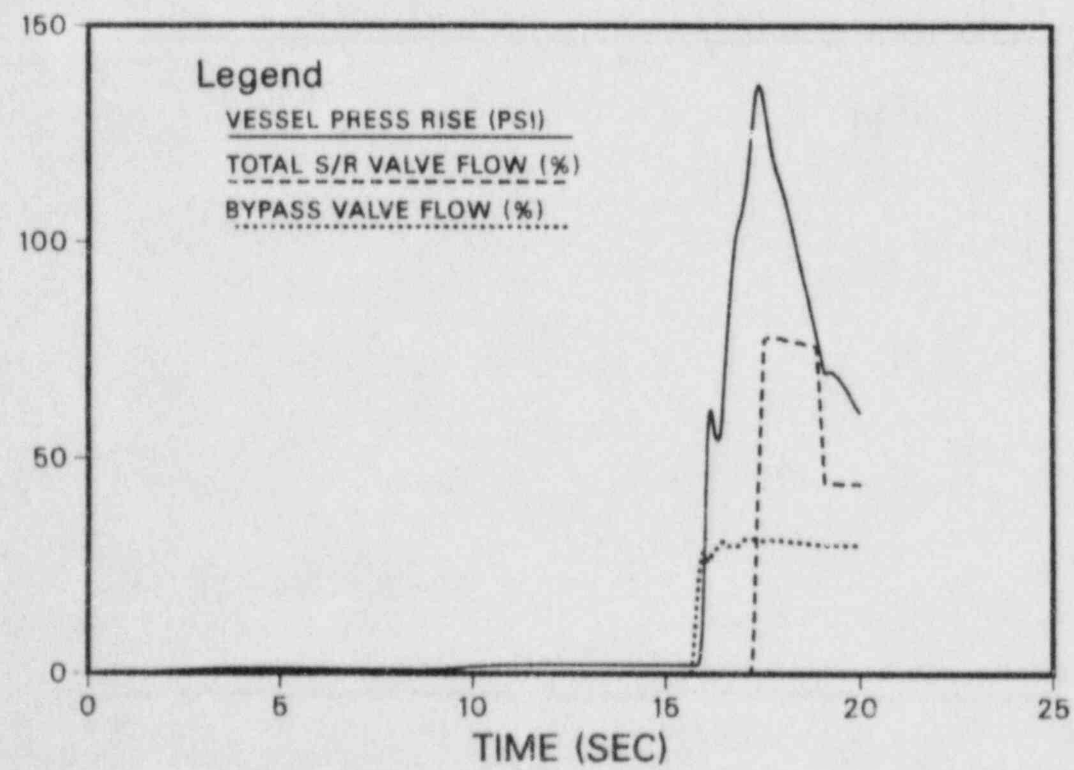


FIGURE 8 FEEDWATER CONTROLLER FAILURE

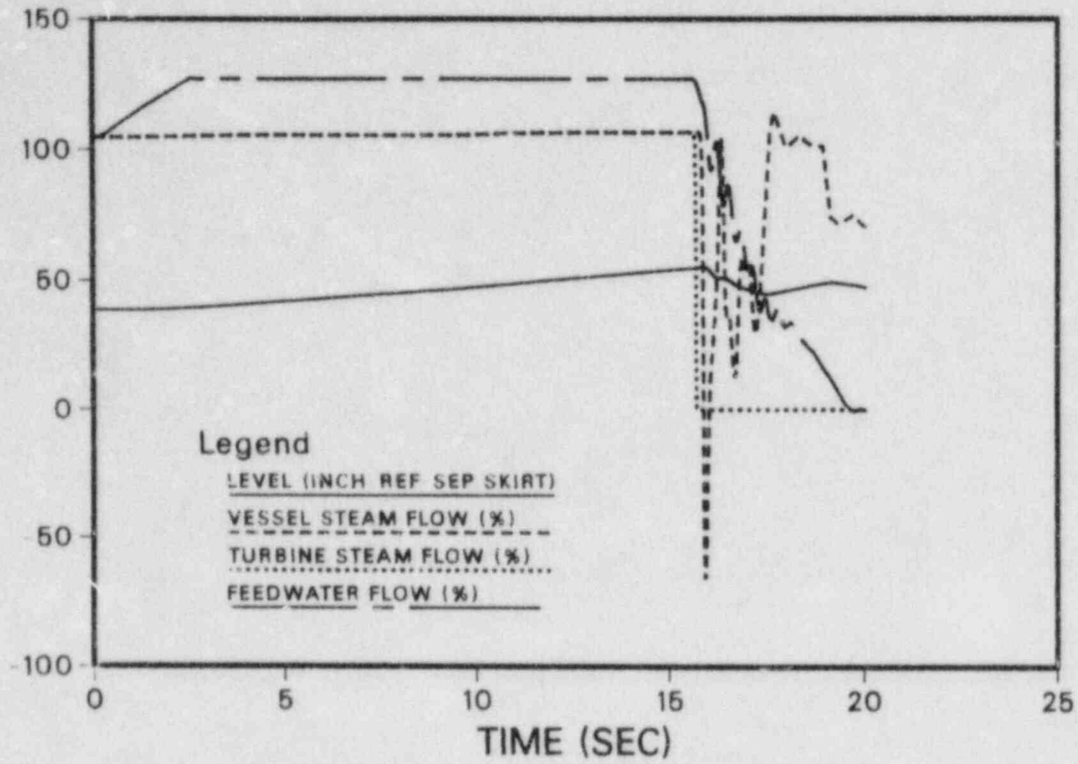


FIGURE 9 FEEDWATER CONTROLLER FAILURE

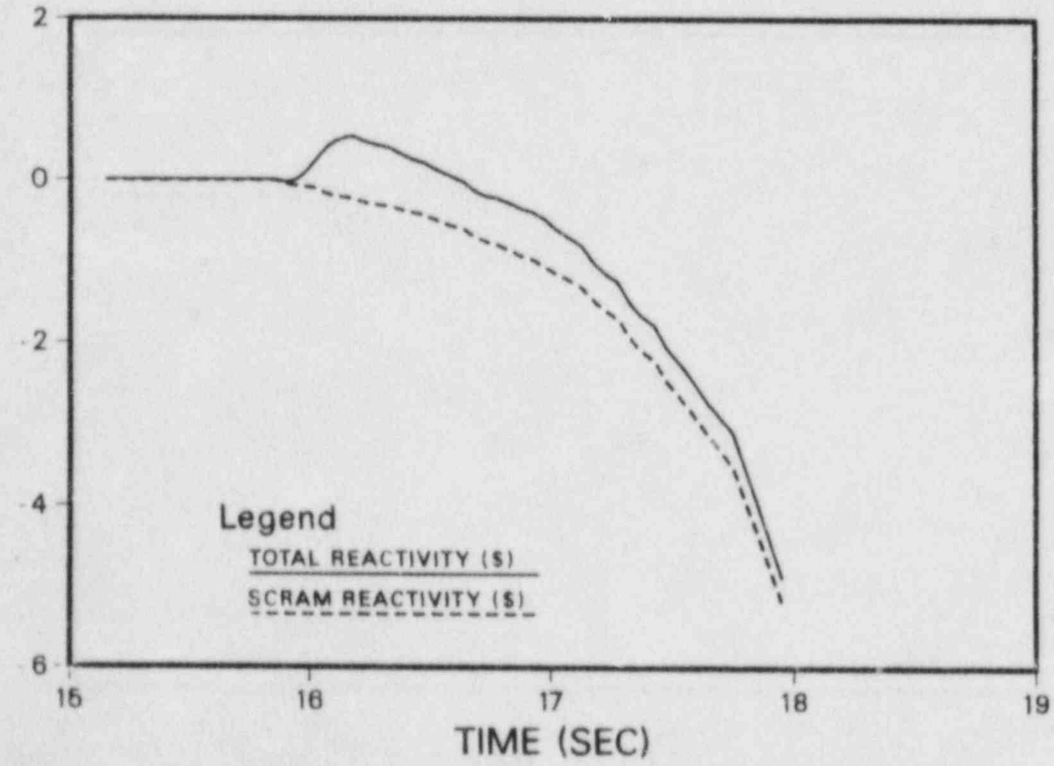


FIGURE 10 MSIV CLOSURE (FLUX SCRAM)

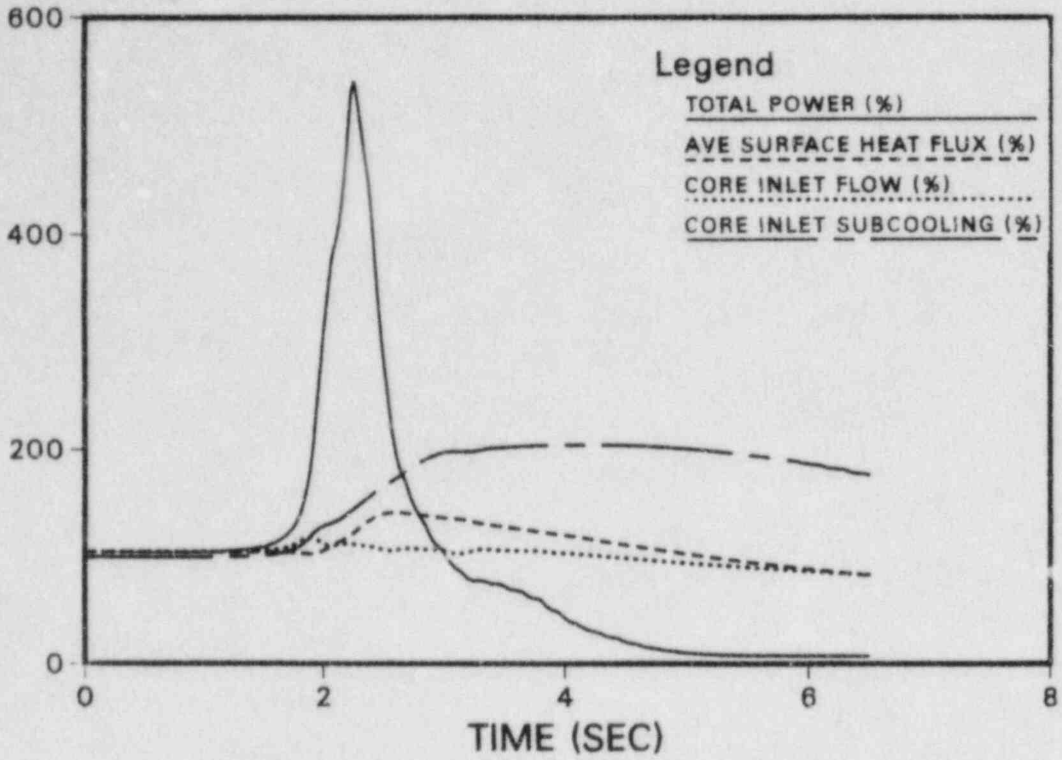


FIGURE 11 MSIV CLOSURE (FLUX SCRAM)

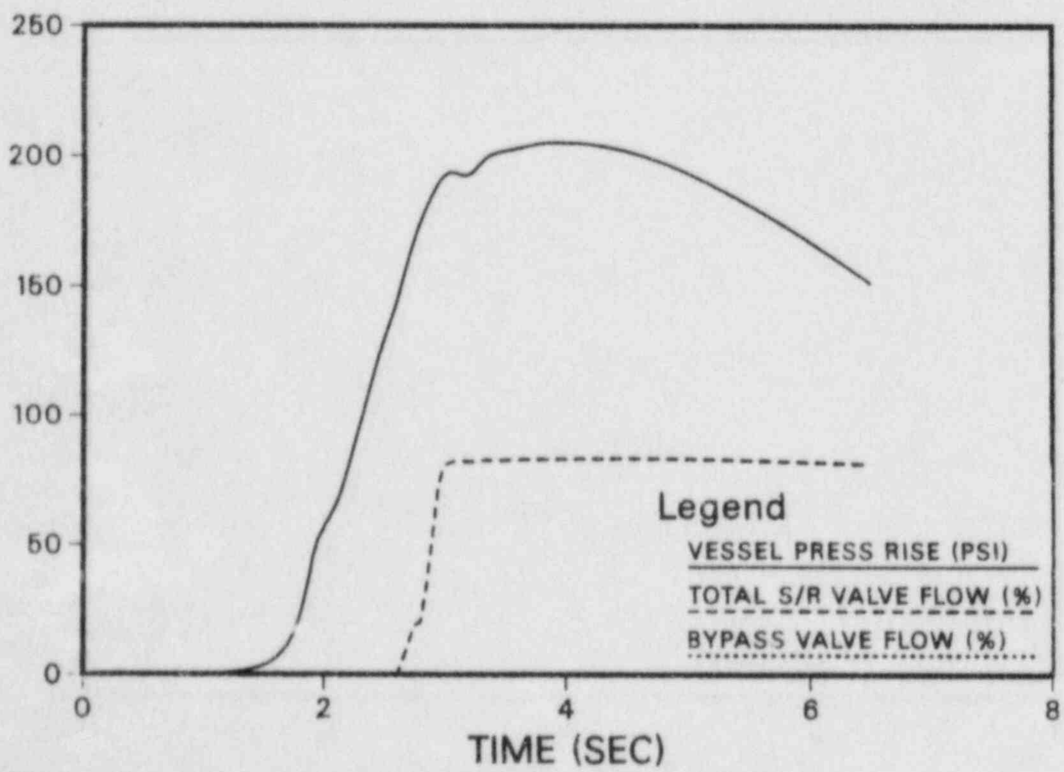


FIGURE 12 MSIV CLOSURE (FLUX SCRAM)

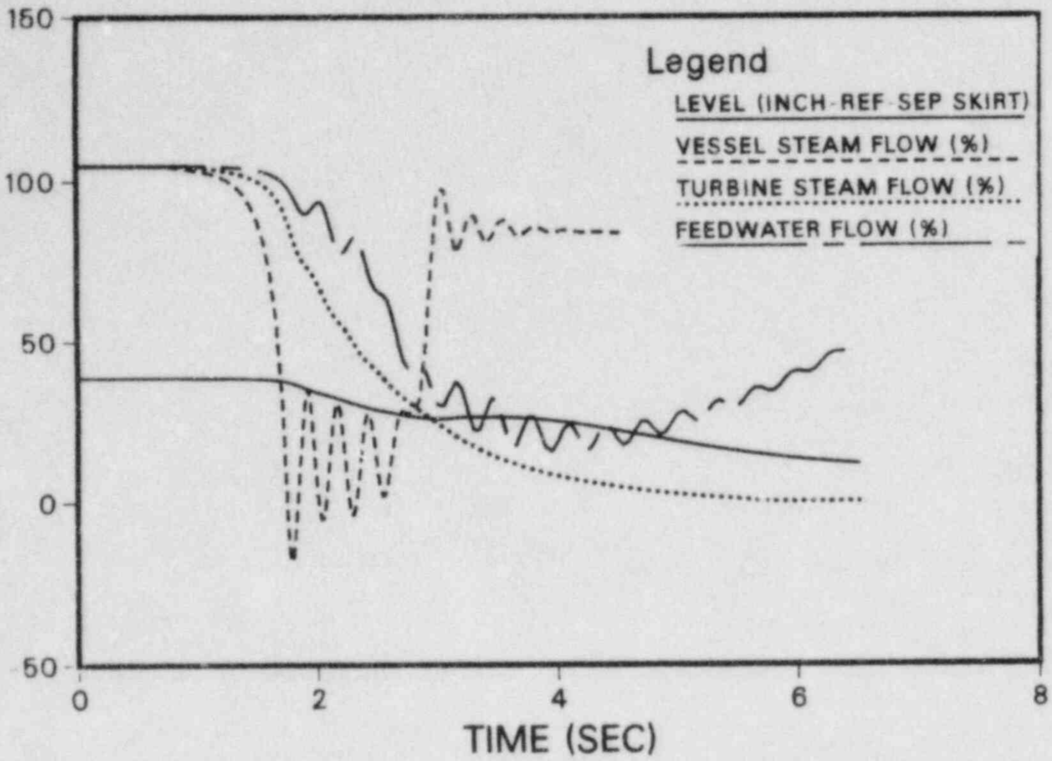


FIGURE 13 MSIV CLOSURE (FLUX SCRAM)

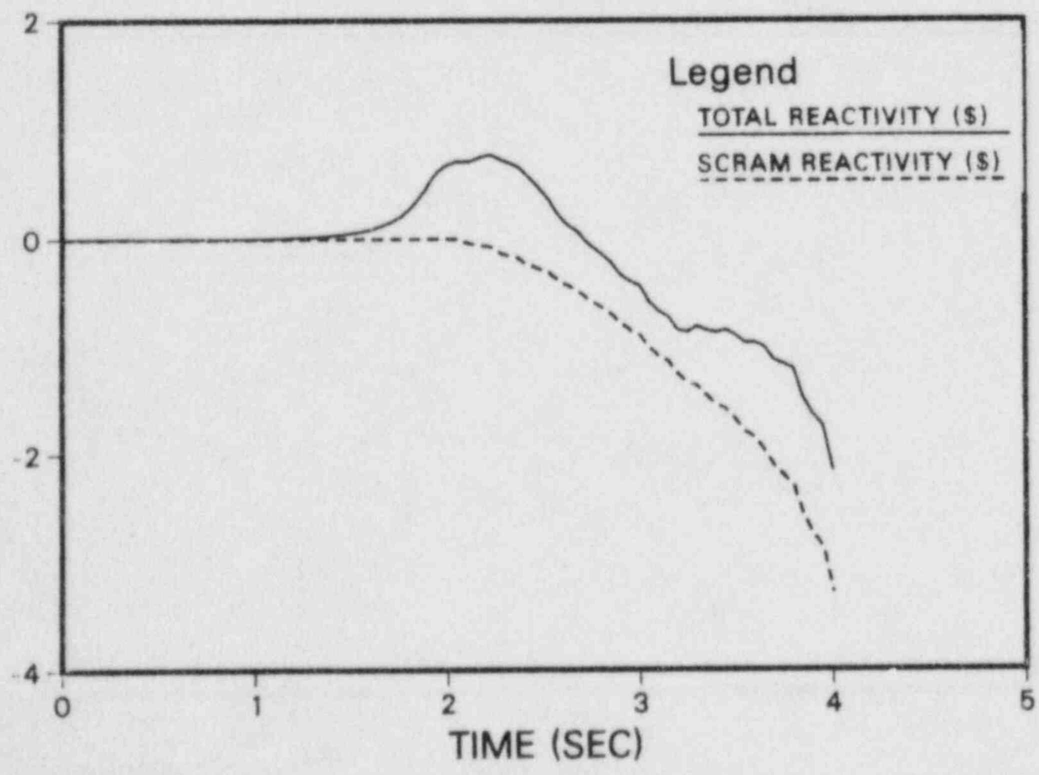
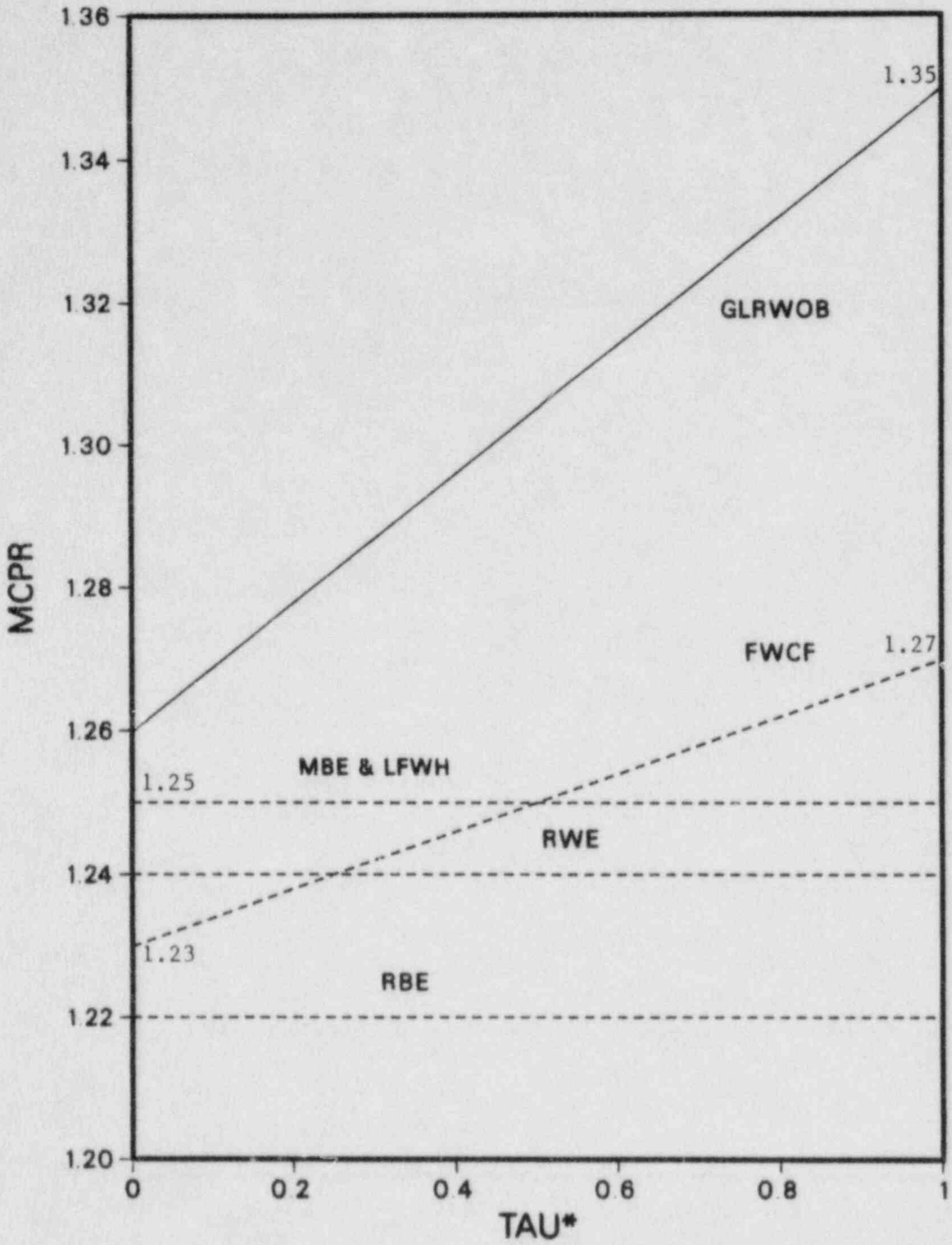


FIGURE 14
OLMCPR FOR P8X8R/QUAD+



*SCRAM SPEED INTERPOLATION PARAMETER AS DEFINED IN THE TECHNICAL SPECIFICATIONS

FIGURE 15
REACTOR POWER (MWT) VS TIME (SEC)

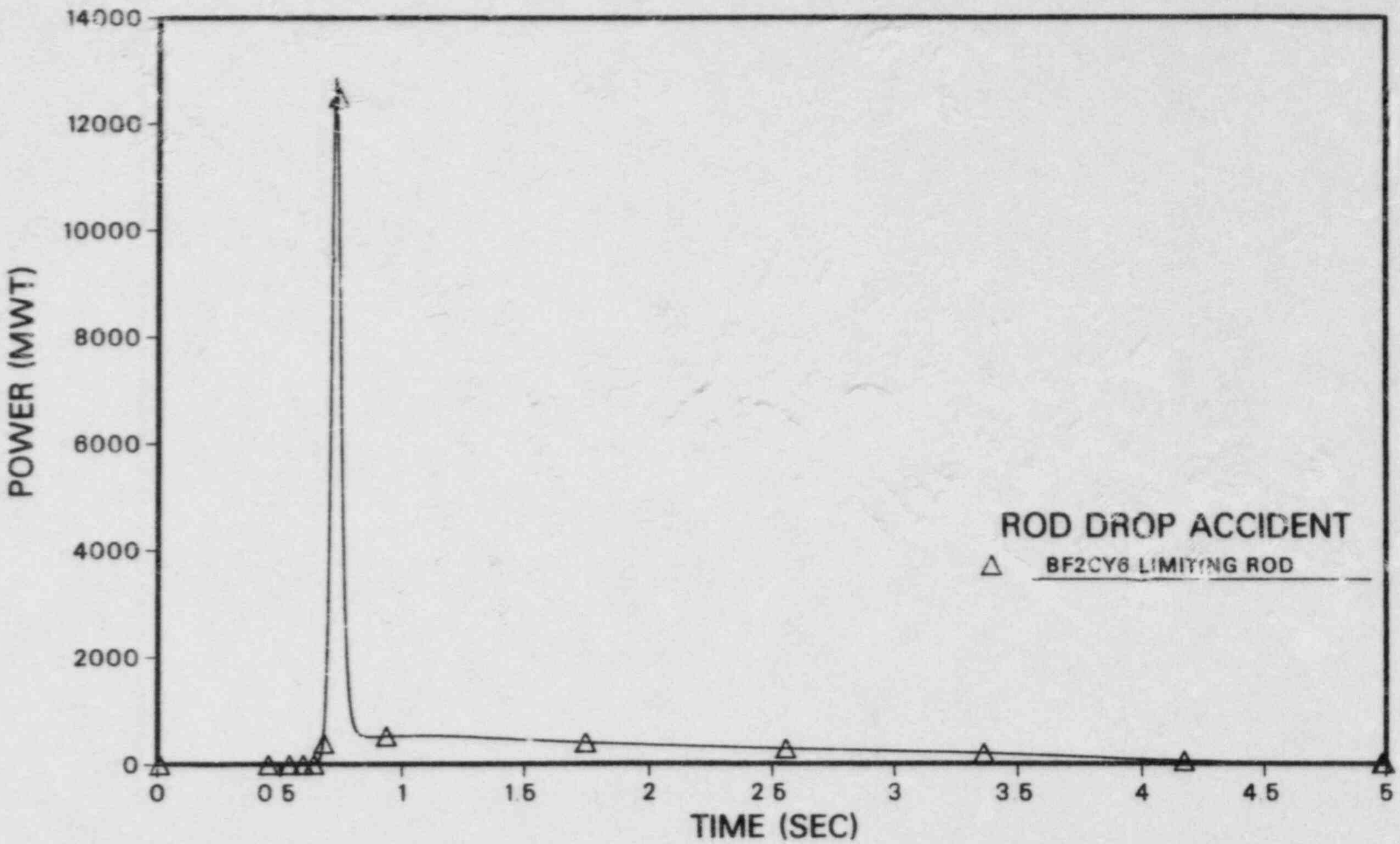


FIGURE 16
CORE REACTIVITY ($) VS TIME (SEC)

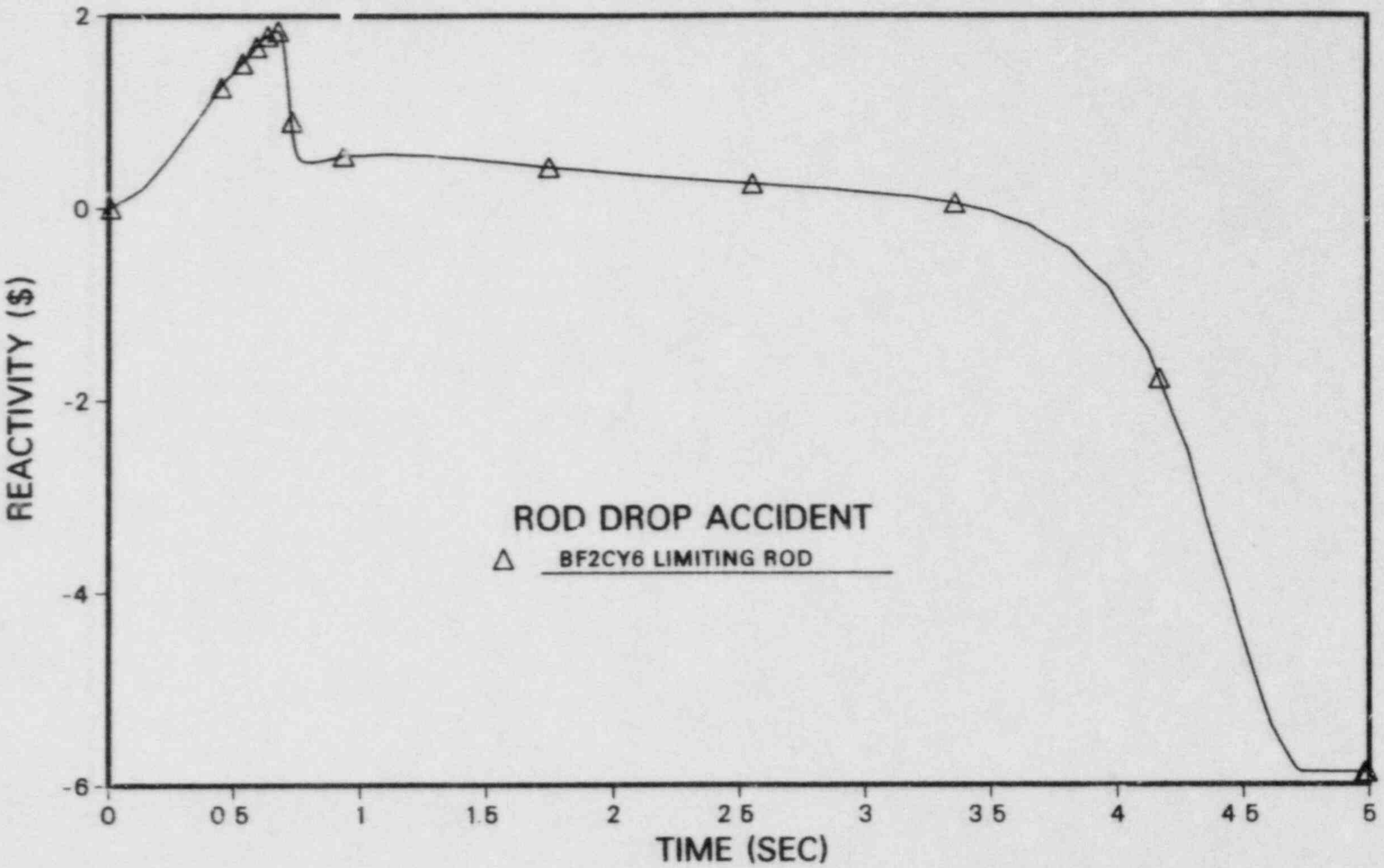


FIGURE 17
CORE AVERAGE TEMPERATURE RISE (DEG F) VS TIME (SEC)

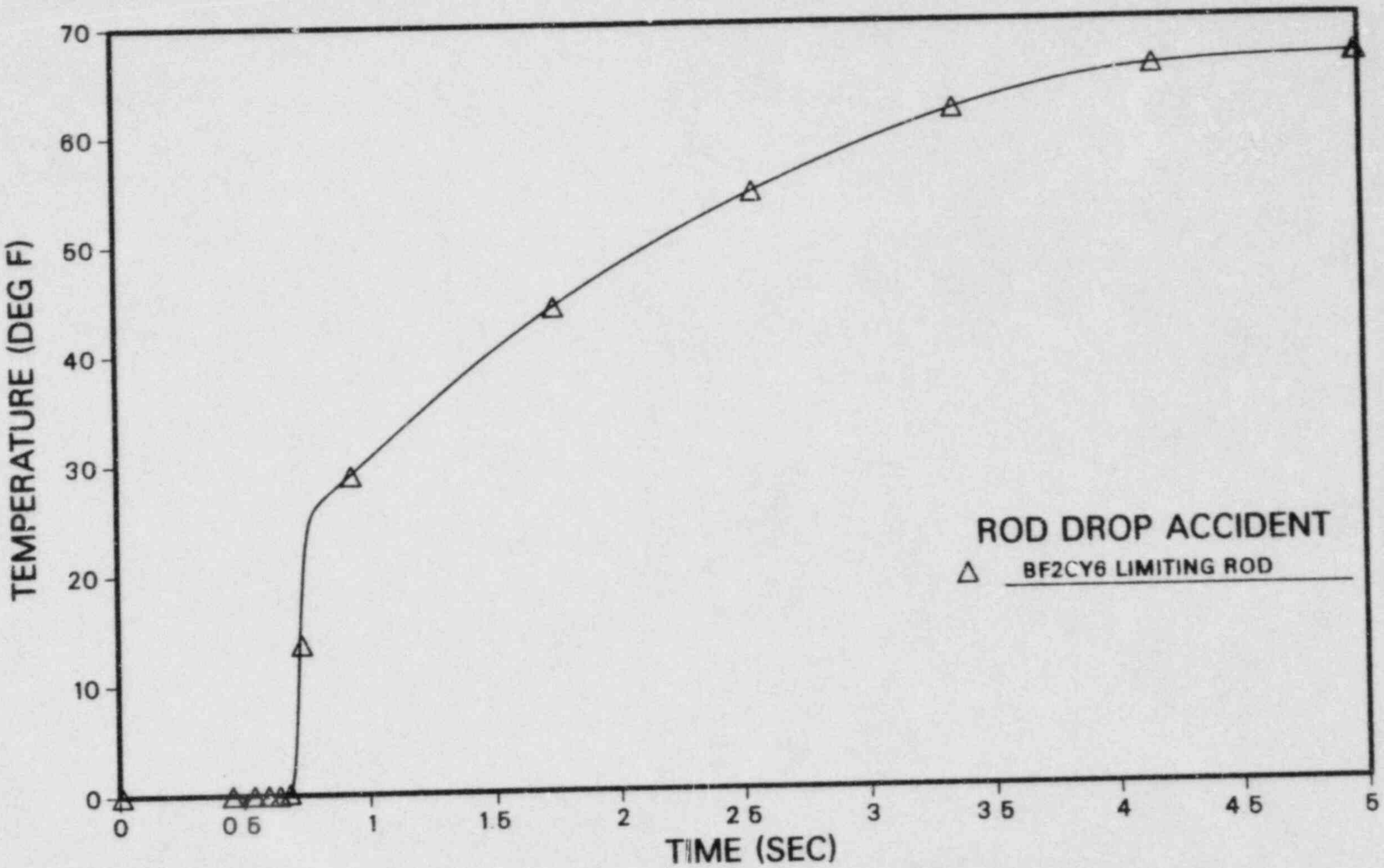


FIGURE 18
MAXIMUM PIN ENTHALPY (CAL/GM) VS TIME (SEC)

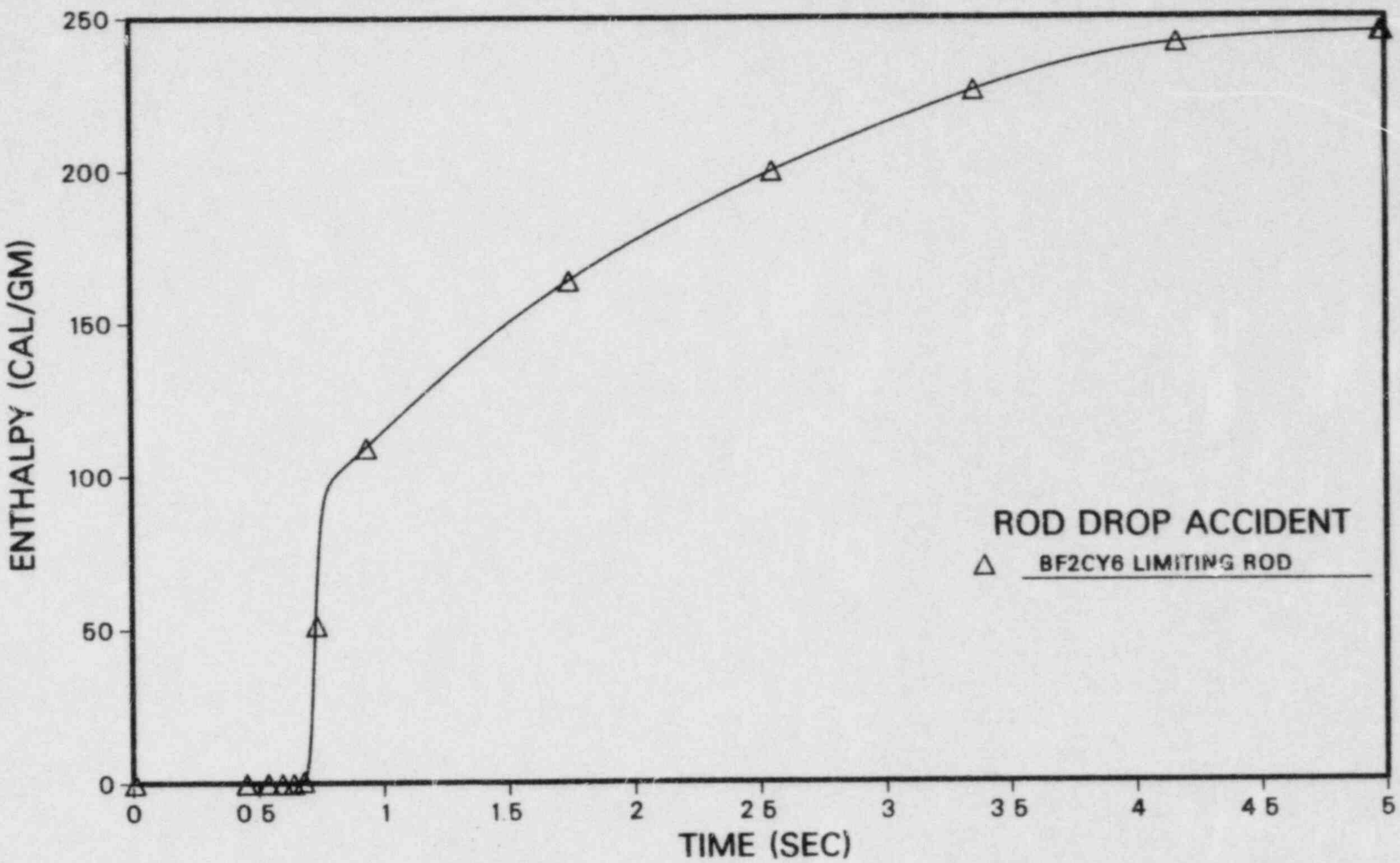
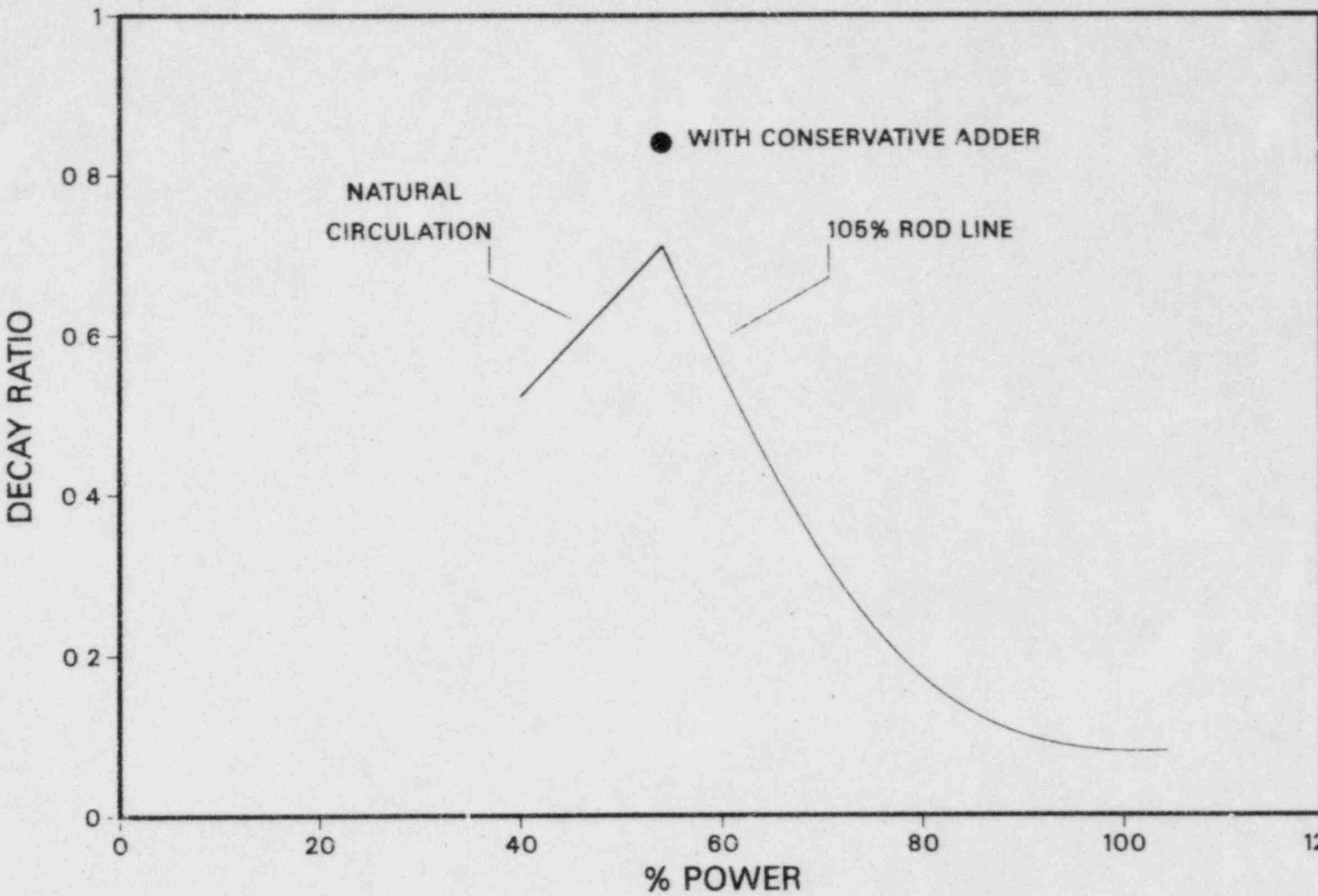


FIGURE 19 DECAY RATIO VERSUS REACTOR POWER



ENCLOSURE 3
DETERMINATION OF NO SIGNIFICANT HAZARDS
BROWNS FERRY NUCLEAR PLANT UNIT 2
(TVA BFNP TS 199)

Description of amendment request:

The amendment would revise the Technical Specifications (T.S.) of the operating license to: (1) modify the core physics, thermal and hydraulic limits to be consistent with the reanalyses associated with replacing about 1/3 of the core during the **cycle 5** refueling outage for unit 2 and (2) reflect plant modifications performed during the **cycle 5** refueling and modification outage. Specifically, the amendment would result in changes to the T.S. in the following twelve areas:

1. Changes to the license related to the Cycle 6 core reload involving removal of depleted fuel assemblies in about one-third of the nuclear reactor core and replacement with new fuel of the same type previously loaded in the core with attendant license changes in the core protection safety limits and reactor protection system setpoints. The actual changes are slight adjustment (by 0.01 in initial core life) in the Operating Limit Minimum Critical Power Ratio (OLMCPR), deleted two of four tables on maximum average planar linear heat generation rate (MAPLHGR) versus average planar exposure that will not be needed due to the fuel change and a change to the references in the bases to reflect that TVA performed the reload transient analysis.

   The loading pattern also includes four Westinghouse QUAD+ demonstration assemblies loaded in peripheral locations. Evaluations performed by Westinghouse indicate that the results of licensing analyses for the previous type assembies bound those for the QUAD+ assemblies. Cycle specific analyses performed by TVA confirm this conclusion.

2. Changes in the T.S. to reflect modifications to the torus as part of the Mark I containment program.

   This includes revising the tables listing surveillance instrumentation for suppression pool bulk temperature reflecting the installation of 16 sensors for an improved torus temperature monitoring system and a revision to the basis for the existing limits on torus water temperature.

3. Change to the T.S. to reflect modifications to the scram discharge instrument volume (SDIV); each of the SDIVs now have new, diverse level instrumentation. The changes to the T.S. are to add operability, surveillance and calibration requirements on the new level instrumentation.

4. Change to T.S. surveillance instrumentation tables to add new instrumentation for containment high-range radiation monitors and to add new instrumentation; and delete current instrumentation for drywell pressure-wide range and suppression chamber wide-range water level in response to requirements in NUREG-0737; items II.F.1.3, II.F.1.4 and II.F.1.5.

5. Changes to T.S. RPS instrumentation requirement tables to delete the bypass function if reactor pressure is less than 1055 psig and the mode switch not in the RUN mode.

6. Change to T.S. surveillance instrumentation tables to reflect new instrument numbers for the new upgraded drywell temperature and pressure instrumentation.

7. Revisions to the table of testable penetrations to reflect the new testable penetrations as a result of modifications to make the flange side of several isolation valves testable.

8. Revision to the T.S. table for containment isolation valve surveillance to add two new isolation valves that are part of a newly installed redundant discharge line from the drywell compressor into containment and to delete one isolation valve which was removed from the demineralized water system.

9. Revision to the T.S. table for containment isolation valve surveillance to delete two isolation valves for the residual heat removal head spray line which is being removed.

10. Revision of T.S. to provide limiting conditions for operation and surveillance requirements for electric power monitoring for the reactor protection system power supply.

11. Modify the T.S. to apply to the new analog (continuous measuring) instrumentation. The analog instrumentation replaces certain mechanical-type pressure and level switches with a more accurate and more stable electronic transmitter/electronic switch system and will provide improved performance of trip functions for reactor protection system actuation, and containment isolation. The changes to the T.S. include:

    a. in the tables on functional test frequencies, calibration frequencies and surveillance requirements, for each switch replaced, add the **instrument number and type of sensor beneath the parameter being monitored and/or controlled.**

    b. add notes to the above tables to specify how the functional and calibration tests are to be conducted.

    c. in addition to the above administrative changes, the calibration requirements have been changed to incorporate extended calibration intervals. However, the required setpoints, functional test frequencies and channel check frequencies for the instrumentation will not be changed. The new calibration requirements, together with the new instrumentation, are expected to provide a more reliable instrumentation system.

12. Administrative changes to the T.S. involving changes to the Table of Contents to reflect the above license changes and miscellaneous editorial changes.

Bases for proposed no significant hazards consideration determination:

The Commission has provided guidance concerning the application of the standards by providing examples of actions that are likely, and are not likely, to involve significant hazard considerations (48 FR 14870). Four examples of actions not likely to involve significant hazards considerations are:

"(i) A purely administrative change to technical specifications: for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature.

(ii) A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: for example, a more significant surveillance requirement.

(iii) For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable...

(vi) A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan; for example, a change resulting from the application of a small refinement of a previously used calculational model or design method."

Each of the twelve changes to the T.S. described previously is encompassed by one of the above example of actions not likely to involve a significant hazards consideration. The basis for this determination on each of the twelve changes is discussed below.

1. Core Reload

1.a Fuel Changes

The changes to the T.S. associated with removing depleted spent fuel from the reactor and replacing these with new fuel assemblies is encompassed by example (iii) above of those actions not likely to involve a significant hazards consideration.

The proposed reload involves fuel assemb es which have been shown to be analytically similar or which are of the same type as previously found acceptable by the staff and loaded in the core in previous cycles. The analytical methods used by the licensee to demonstrate conformance to the technical specifications have been previously approved by the staff. In addition, no changes have been made to the acceptance criteria for the technical specification changes involved.

Since the replacement fuel assemblies are analytically similar or of the same type previously added to all three Browns Ferry units and other BWRs and since the codes, models, and analytical techniques used to analyze the reload have been generically approved by the NRC, the changes to the technical specifications associated with the reload are clearly encompassed by example (iii) of the guidance provided by the Commission for an action not likely to involve a significant hazards consideration.

1.b References in the Bases

The changes in the T.S. associated with changing the references in the Bases to reflect that the reload transient analysis is now being performed by TVA is encompassed by examples (i) and (iii) above of those actions not likely to involve a significant hazards consideration.

The reload analysis, in the past, has been performed by General Electric Company. This reload analysis has been performed by TVA using analytical methods described in TVA-TR81-01-A. The analytical methods have been approved by the staff. Since NRC has previously found these methods acceptable and the T.S. changes are being made to achieve consistency between the methods used and the references in the Bases, these changes to the T.S. are clearly encompassed by examples (i) and (iii) of the guidance provided by the Commission for an action not likely to involve a significant hazards consideration.

2. Changes Related to Torus Modifications

One of the changes to the T.S. is to revise the tables that list the surveillance instrumentation associated with the suppression pool bulk temperature. This modification provides an improved torus temperature monitoring system which consists of 16 sensors. This will provide a more accurate indication of the torus water bulk temperature as required by NUREG-0661 and will replace the suppression chamber water temperature instruments presently listed in the T.S.

The change to the T.S. are necessary administrative follow up actions essential to the implementation of this improvement. The changes to the T.S. place operability and calibration requirements on the new temperature monitoring system. Since these are new instruments, the surveillance requirements are not presently in the T.S. Thus, adding those restrictions and controls is encompassed by example (ii) provided by the Commission.

3. Scram Discharge Instrument Volume

The SDVs and SDIVs are being modified to address inadequacies identified by the partial rod insertion event on Browns Ferry unit 3 in June 1980. One of the modifications includes adding electronic level switches to initiate a scram on a high level in the SDIV. Thus, the changes to the T.S. are necessary administrative follow up actions essential to the implementation of this improvement. Adding these new restrictions and controls, which otherwise would not be in the T.S., is encompassed by example (ii) of the guidance provided by the Commission.

4. Accident Monitoring Instrumentation

Item II.F.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," requires all licensees to install five new monitoring systems and to provide onsite sampling/analysis capability for a specified range of radionuclides. For all six categories, NUREG-0737 states: "Changes to technical specifications will be required." During this refueling outage, the licensee has installed: (a) a containment high-range monitoring system, (b) a drywell wide-range pressure monitoring system and (c) a suppression chamber wide-range water level monitoring system. These three items were required by NUREG-0737, items II.F.1.3, II.F.1.4 and II.F.1.5, respectively. The changes to the T.S., which track the model T.S. provided to the licensee by the staff, are to add operability and surveillance requirements on the new monitoring systems to the T.S.

The revisions also delete the present drywell pressure and suppression chamber water level instruments since they are being replaced by items b and c above. The changes to the technical specifications are necessary administrative follow up actions required by the Commission. Adding the new surveillance requirements and controls is encompassed by example (ii) of the guidance provided by the Commission.

5. Scram Permissive Pressure Switches at 1055 psig

Present configuration on unit 2 has a bypass function which allows a scram in the refuel and startup/hot standby modes of operation by the scram functions main steamline isolation valve closure and turbine condenser low vacuum when the reactor pressure is greater than 1055 psig.

The reactor high-pressure scram is set at 1055 psig and is operable in these two modes of operation. If reactor pressure exceeds 1055 psig, the reactor scrams due to the reactor high-pressure scram function, and the main steamline isolation valve closure and the turbine condenser low vacuum functions become operable. The bypass circuit therefore serves no real purpose. When the two scram functions become available, the reactor is already scrammed. Since the reactor is protected by the high-pressure scram function, the proposed change does not result in any reduction in the margin of safety. The T.S. changes therefore are encompassed by example (vi) of the guidance provided by the Commission.

6. Drywell Temperature and Pressure

The drywell temperature and pressure surveillance instrumentation is being upgraded this outage to provide qualified, more reliable instrumentation. The T.S. are being revised to reflect new instrument numbers. The surveillance requirements remain unchanged. The changes to the technical specifications are necessary administrative follow up actions required by the Commission and are clearly encompassed by example (i) of the guidance provided by the Commission.

7. Testable Penetrations

Modifications are being made to the flange side of fourteen containment isolation valves which cannot be isolated from primary containment to be tested. This modification will provide two gaskets with a pressure tap between the gaskets to allow the flange to be leak tested. Operability of the valve will not be affected by this modification. Fourteen new testable penetrations resulted and they were added to the table of testable penetrations with double o-ring seals. New surveillance requirements are being added. The change is encompassed by example (ii) of the guidance provided by the Commission.

Several editorial changes were also made to this table. They include revising the identification name on several penetrations, adding a penetration that was tested but was inadvertently left out of the table and removing penetration X-213A which no longer exists. These changes are purely administrative and are encompassed by example (i) of the guidance provided by the Commission.

8. Redundant Air Supply to Drywell

During the current outage, TVA has installed a second discharge line from the drywell compressor into containment. This line was added to provide the capability for isolation of approximately one-half of the drywell suppression equipment in the case of a drywell line leak. This air supply will be used to supply two inboard main steam isolation valves (MSIVs), approximately one-half of the main steam relief valves (MSRVs), and approximately one-half of all other air-operated equipment in the drywell. This will significantly reduce the possibility of any one control air pipe break inside containment from requiring immediate shutdown and isolation **as a result of MSIVs, MSRVs, and drywell coolers being inoperable, Since any line** penetrating containment requires two isolation valves, the table in the Technical Specifications listing the isolation valves that must be periodically tested is being revised to add these two new isolation valves. TVA has concluded that this modification will increase the margin of safety. The changes to the technical specifications are necessary administrative follow up actions essential to the implementation of this improvement. The two isolation valves being added to the T.S. are new valves not presently listed in the T.S. If they are not added to the table of valves to be periodically tested, there would be no T.S. requirement to test these valves. Adding these additional controls is encompassed by example (ii) of the guidance provided by the Commission.

One isolation valve on the demineralized water system was removed from unit 2. The demineralized water system is no longer used. The isolation valve was removed and the line capped. The T.S. are being revised to remove this valve from the table of valves to be tested. The changes to the technical specifications are necessary administrative follow up actions essential to the implementation of the improvement. The changes are clearly encompassed by example (i) provided by the Commission.

9. Residual Heat Removal Head Spray Line

Two isolation valves on the residual heat removal head spray line were removed from unit 2. The head spray line was removed and the penetration capped. The T.S. are being revised to remove these valves from the table of valves to be tested. The changes to the T.S. are necessary administrative follow up actions essential to the implementation of the improvement. The changes are clearly encompassed by example (i) provided by the commission.

10. Monitoring of RPS Power Supply

By letter dated August 7, 1978, the Commission advised TVA that during review of Hatch unit 2, the staff had identified certain deficiencies in the design of the voltage regulator system of the motor generator sets which supply power to the reactor protection system (RPS). Pursuant to 10 CFR 50.54(f), TVA **was** required to evaluate the RPS power supply for Browns Ferry 1, 2, and 3 in light of the information set forth in our letter. **By** letter dated September 24, 1980, the staff informed TVA (and most other BWRs) that "we have determined that modifications should be performed to provide fully redundant Class IE protection at the interface of non-Class IE power supplies and the RPS." The staff also advised TVA that "we have found that the conceptual design proposed by the General Electric Company and the installed modification on Hatch are acceptable solutions to our concern." By letter dated December 4, 1980, TVA committed to install the required modifications. By letters dated October 30, 1981 and July 28, 1982, NRC sent TVA model Technical Specifications for electric power monitoring of the RPS design and modification. During the current outage of unit 2, the RPS is being modified to provide a fully redundant Class IE protection at the interface of the non-Class IE power supplies and the RPS. This will ensure that failure of a non-Class IE reactor protection power supply will not cause adverse interaction to the class IE reactor protection system.

The Technical Specifications are being revised similar to the model T.S. provided to TVA to reflect the limiting conditions for operation and surveillance requirements associated with the RPS modifications. Page 42 is being modified to add a description of these sections in the bases. The changes to the T.S. are necessary administrative follow up actions essential to the implementation of these improvements. The additional limitations and controls, which are presently not in the T.S., are encompassed by example (ii) of the guidance provided by the Commission.

11. Analog Trip System

The RPS, the primary containment isolation system (PCIS), and the core standby cooling systems (CSCS) use mechanical-type switches in the sensors that monitor plant process parameters. These mechanical-type switches are very subject to drift in the setpoint as is evident from the many licensee event reports (LERs) that have been submitted reporting calibration drifts in these switches.

Advances in technology make it possible to replace the mechanical-type switches with a more accurate and more stable electronic transmitter/electronic switch system. For several years, TVA has been planning to replace existing pressure switches that sense drywell and reactor pressures with analog loops and modify the reactor water level indication loops to improve the reliability, accuracy and response time of this instrumentation. The modification involves removing one device and substituting other devices to perform the same function. Changes in design bases, protective function, redundancy, trip point and logic are not involved. Similar modifications have been approved for other BWRs.

As described previously, most of the changes to the T.S. are administrative in nature (i.e., adding the specific number and types of sensor and adding notes to describe how testing is conducted). As such, they are encompassed by example (i) of the guidance provided by the Commission. The changes in surveillance requirements relates to example (ii) of the guidance provided by the Commission. Some of the surveillance intervals have been decreased as appropriate for each new instrument. However, the overall effect of the changes in technical specifications will be to increase the total surveillance requiremens in support of a more reliable instrumentation system.

12. Administrative Changes

Several administrative changes are being made to the Technical Specifications. These include revising the Table of Contents to reflect the change discussed above, and miscellaneous editorial changes. The surveillance requirement for the personnel air lock is being changed to be consistent with the surveillance for units 1 and 3. These changes are editorial in nature and have no safety significance. These changes are encompassed by example (i) cited by the Commission as an action not likely to pose a significant hazards consideration.

Since all of the changes to the T.S. given in the twelve areas above are encompassed by an example in the guidance provided by the Commission of actions not likely to involve a significant hazards consideration, the staff has made a proposed determination that the application for amendment involves no significant hazards consideration.