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TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT
UNIT 2, CYCLE 6

RELOAD LICENSING REPORT

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Tennessee Valley Authority

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I. Introduction

This reload licensing report presents the results of the core redesign and safety analyses performed for Browns Ferry Nuclear Plant (BFN) unit 2, cycle 6 operation. The current licensed design is documented in references 1 and 2. The methodology and technical bases employed in the performance of these analyses are discussed in references 3-8.

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 9).

The cycle 6 core loading has been changed based on results of inspection and reconstitution of the fuel available for use in cycle 6. The unit 1 once-burned fuel will replace the unit 2 once-burned for unit 2, cycle 6. Also, 212 twice- and thrice-burned bundles to be loaded were inspected and reconstituted as needed.

II. Reload Cycle Information

A. Design Basis Exposures

1. Actual cycle 5 core average exposure at end of cycle: 20.8 GWd/ST

2. Minimum cycle 5 core average exposure at end of cycle from cold shutdown considerations: 20.8 GWd/ST

3. Assumed cycle 6 core average exposure at depletion of reactivity (DOR)*: 17.9 GWd/ST

B. Reload Fuel Assemblies

Fuel Type	Cycle Loaded	Number
Irradiated		
8DRB284L,U2R2	U2CY3	53
P8DRB284L,U2R3	U2CY4	159
P8DRB265H,U1R5	U1CY6	160
P8DRB284L,U1R5	U1CY6	80
P8DRB284Z,U1R5	U1CY6	8
New		
P8DRB284L,U2R5	U2CY6	300
QUAD+ Demo	U2CY6	4
TOTAL		764

*DOR - End of full power capability

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 10. The nuclear, mechanical, and thermal-hydraulic design descriptions for the Westinghouse demonstration assemblies are contained in reference 9.

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 core condition at the end of the previous cycle, the number and type of fuel assemblies suitable for use, 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. These locations satisfy the criteria specified in references 2 and 9. Evaluations performed by Westinghouse (reference 9) show 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.

A total of 212 twice- and thrice-burned assemblies were inspected and reconstituted for use in cycle 6. Prior to the reconstitution project, guidelines were implemented to ensure that performance of the reconstituted assemblies would not differ significantly from the original assemblies. Consequently, the safety analysis results reported in this document were generated with the reconstituted assemblies modeled as original assemblies. Following completion of the reconstitution work, this modeling assumption was verified by individually analyzing each reconstituted assembly and by performing core-wide analyses to specifically address the effects of reconstitution. These analyses confirmed that all design criteria are satisfied and that operating limits reported in this document remain valid.

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 11 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 standby liquid control 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 4 and 6) in conjunction with the TVA lattice physics data generation code (references 5 and 6) 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 actual end-of-previous-cycle core average exposure. The core was assumed to be in a xenon-free condition.

Cold keff 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 keff at BOC and the maximum calculated strongest rod out keff at any exposure point. The maximum strongest rod out keff at any exposure point is equal to or less than:

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

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 keff of 1.0, resulting in a calculated minimum cold shutdown margin of 1.0-percent Δk for BFN unit 2, cycle 6.

Table 1

CALCULATED CORE EFFECTIVE MULTIPLICATION - NO VOIDS, NO XENON, 20°C

Uncontrolled, K_{eff}^{UNC} (BOC)	1.120
Fully Controlled, K_{eff}^{CON} (BOC)	0.956
Strongest Control Rod Out, K_{eff}^{SRO} (BOC)	0.985
R, Maximum Increase in Cold Core Reactivity With Exposure Into Cycle, Δk	0.005

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 condition 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)
660	0.029

B. Reactivity Coefficients

The reactivity coefficients associated with the nuclear design of BFN unit 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 BFN unit 2, cycle 6 the following results were obtained:

100% core flow, DOR	-0.0734	%Δk/%void
105% core flow, EDOR[1]	-0.0745	%Δk/%void

[1] EDOR - extended depletion of reactivity resulting from increased core flow.

C. Fuel Performance

The BFN unit 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 9 and 10. Furthermore, peak linear heat rates satisfy the assumptions made in the fuel vendors' thermal-mechanical integrity analyses (references 9 and 10). All fuel types loaded in cycle 6 are predicted to operate within these bounding assumptions. Additionally, the QUAD+ demonstration assemblies are predicted to have substantial margin to the lead P8x8R assembly in steady-state bundle power and thermal limits throughout cycle 6 (figures 20-22). The minimum margin for bundle power is 27 percent which satisfies the requirement for at least a 20-percent margin specified by NRC (reference 2). For MCPR the minimum margin is 43 percent and for LHGR it is 32 percent.

IV. Transient Analyses

A. Pressurization Events

The RETRAN computer code (reference 12) 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 7. A description of the CPR correlation and its application to Browns Ferry is contained in reference 13. 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)
EDOR	105	105	674

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	1.295	6.416	123.7	1.051	1287

Pressurization Event Analysis Results

Transient	Peak Power (% Rated)	Peak Heat Flux (% Rated)	Peak Vessel Press. (psia)	ΔCPR[1] P8x8R	System Response
Load Rejection w/o Bypass	403.4	121.6	1235.3	0.225	Figures 2-5
Feedwater Controller Failure	234.8	115.5	1215.1	0.149	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 4). The methods used to analyze these events are described in reference 3. Results are summarized below.

Nonpressurization Event Analysis Results

Event	ΔCPR[4] P8x8R/8x8R/QUAD+	Peak LHGR(kW/ft)[4] P8x8R/8x8R/QUAD+
Loss of Feedwater Heating (100°F)	0.18	17.5
Rod Withdrawal Error	0.20[2]	20.8
Rotated Bundle Error	0.19[3]	15.3
Mislocated Bundle Error	0.13	14.4

[1] Results presented were calculated for P8x8R fuel and will be conservatively applied to 8x8R.

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

[3] Includes 0.02 penalty required when using the variable water gap method (reference 10).

[4] Results presented were calculated for the P8x8R fuel type and conservatively bound the results calculated for the 8x8R fuel type. The results are also bounding for the QUAD+ demonstration assemblies which will be loaded into nonlimiting, peripheral core locations.

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 7. Results are summarized below.

MSIV Closure (Flux Scram) Results

Peak Vessel Pressure (psia)	Peak Steamline Pressure (psia)	System Response
1281.0	1242.5	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 3 and 7. The application of Options 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/8x8R/QUAD+	Option B[1] P8x8R/8x8R/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/8x8R/QUAD+[1]
Loss of Feedwater Heaters (LFWH)	1.25
Rod Withdrawal Error (RWE)	1.27
Rotated Bundle Error (RBE)	1.26
Mislocated Bundle Error (MBE)	1.20

[1] Results presented were calculated for the P8x8R fuel type and conservatively bound the results calculated for the 8x8R fuel type. 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)

MAPLHGR limits for the unit 1 P8DRB284Z fuel type (from reference 14) still apply for fuel being transferred to unit 2 since the LOCA responses for the two units are identical (reference 15). The limits for remaining fuel types are taken from reference 16. Reference 9 indicates that the MAPLHGR limits for fuel type P8DRB284L can be conservatively applied to QUAD+ demonstration assemblies. Tables of MAPLHGR limits for all fuel types in unit 2, cycle 6 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	10.2
40,000	9.5
45,000	8.8

LOCA Limits for GE Fuel Type P8DRB284Z

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.2
5,000	11.7
10,000	12.0
15,000	12.0
20,000	11.8
25,000	11.1
30,000	10.4
35,000	9.8
40,000	9.1
45,000	8.5

LOCA Limits for GE Fuel Type P8DRB265H

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.5
1,000	11.6
5,000	11.9
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7
35,000	10.2
40,000	9.6

LOCA Limits for GE Fuel Type 8DRB284L

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	10.2
40,000	9.5

LOCA Limits for GE Fuel Type P8DRB284L

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	10.2
40,000	9.5
45,000	8.8

B. Rod Drop Accident (RDA)

The methodology used to analyze the rod drop accident is described in appendix A of reference 8. Results for unit 2, cycle 6 are summarized below.

Results for the Limiting RDA

Condition: 375°F, MOC Exposure
Rod Worth: 1.05 percent Δk
Rod Position: 38-15
Peak Fuel Enthalpy: 194.5 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 8. 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 BFN unit 2, cycle 6 are summarized below and in figure 19.

Stability Analysis Results at Limiting Initial Conditions

Analysis	Maximum Decay Ratio
Core Stability	0.84[1]
Channel Stability P8x8R/8x8R/QUAD+	0.59[2]

[1] Includes 0.14 uncertainty adder as described in appendix B of reference 8.

[2] Results presented are for the P8x8R fuel type and conservatively bound the 8x8R fuel type and the QUAD+ demonstration assemblies.

References

1. TVA-RLR-002, Rev. 1 dated April 1985, "Reload Licensing Report for Browns Ferry Unit 2, Cycle 6," TVA.

2. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 125 to Facility Operating License No. DPR-52, Tennessee Valley Authority, Browns Ferry Nuclear Power Plant, Unit 2, Docket No. 50-260.

3. TVA-EG-047 dated January 1982, "TVA Reload Core Design and Analysis Methodology for the Browns Ferry Nuclear Plant," TVA.

4. TVA-TR78-03A dated January 1979, "Three-Dimensional LWR Core Simulation Methods," TVA.

5. TVA-TR78-02A dated April 1978, "Methods for the Lattice Physics Analysis of LWRs," TVA.

6. TVA-TR79-01A dated January 1979, "Verification of TVA Steady-State BWR Physics Methods," TVA.

7. TVA-TR81-01A dated December 1981, "BWR Transient Analysis Model Utilizing the RETRAN Program," TVA.

8. TVA-RLR-001 dated January 1984, "Reload Licensing Report for Browns Ferry Unit 3, Cycle 6," TVA.

9. WCAP-10507 dated March 1984, "QUAD+ Demonstration Assembly Report," Westinghouse Electric Corporation.

10. NEDE-24011-P-A-8 dated May 1986, "General Electric Standard Application for Reactor Fuel," General Electric.

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

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

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

14. NEDO-24056, Rev. 1, dated May 1983, "Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 1," General Electric.

15. DGC:88-146, Letter from D. G. Churlik to J. D. Robertson dated July 13, 1988, "Telecon of 7/13/88," General Electric.

16. NEDO-24088-2 (as amended) dated May 1985, "Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 2," General Electric.

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

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	35	37	39	41	43	45	47	49	51	53	55	57	59
60									B	B	B	B	B	B	B	B	B	B	B	B	B	B								
58								B	A	F	A	F	A	F	F	F	F	A	F	A	F	B	B							
56						B	B	B	F	A	F	C	F	C	F	F	C	F	C	F	A	F	B	B	B					
54						B	B	F	C	F	C	F	C	F	C	C	F	C	F	C	F	C	F	B	B					
52					B	B	F	D	F	C	F	C	F	B	F	F	B	F	C	F	C	F	D	F	B	B				
50			B	B	A	Q	C	F	C	F	C	F	C	F	E	E	F	C	F	C	F	C	F	C	Q	A	B	B		
48			B	B	F	C	F	C	F	C	F	C	F	D	F	F	D	F	C	F	C	F	C	F	C	F	B	B		
46		B	B	F	D	F	C	F	D	F	D	F	D	F	B	B	F	D	F	D	F	D	F	C	F	D	F	B	B	
44	B	B	F	C	F	C	F	D	B	D	B	C	B	D	F	F	D	B	C	B	D	B	D	F	C	F	C	F	B	B
42	B	F	A	F	C	F	C	F	D	F	D	F	C	F	C	C	F	C	F	D	F	D	F	C	F	C	F	A	F	B
40	B	A	F	C	F	C	F	D	A	D	F	D	B	C	F	F	C	B	D	F	D	A	D	F	C	F	C	F	A	B
38	B	F	C	F	C	F	C	F	C	F	D	F	D	F	B	A	F	D	F	D	F	C	F	C	F	C	F	C	F	B
36	B	A	F	C	F	C	F	D	B	C	B	D	F	A	F	F	A	F	D	B	C	B	D	F	C	F	C	F	A	B
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32	B	F	F	C	F	E	F	B	F	C	F	B	F	C	A	A	C	F	B	F	C	F	B	F	E	F	C	F	F	B
30	A	F	F	C	F	E	F	B	F	C	F	B	F	C	A	A	C	F	B	F	C	F	B	F	E	F	C	F	F	A
28	B	F	C	F	B	F	D	F	D	F	C	F	A	F	C	C	F	A	F	C	F	D	F	D	F	B	F	C	F	B
26	B	A	F	C	F	C	F	D	B	C	B	D	F	A	F	F	A	F	D	B	C	B	D	F	C	F	C	F	A	B
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16		B	B	F	D	F	C	F	D	F	D	F	D	F	B	B	F	D	F	D	F	D	F	C	F	D	F	B	B	
14			B	B	F	C	F	C	F	C	F	C	F	D	F	F	D	F	C	F	C	F	C	F	C	F	B	B		
12			B	B	A	Q	C	F	C	F	C	F	C	F	E	E	F	C	F	C	F	C	F	C	Q	A	B	B		
10					B	B	F	D	F	C	F	C	F	B	F	F	B	F	C	F	C	F	D	F	B	B				
8						B	B	F	C	F	C	F	C	F	C	C	F	C	F	C	F	C	F	B	B					
6						B	B	B	F	A	F	C	F	C	F	F	C	F	C	F	A	F	A	B	B					
4								B	A	F	A	F	A	F	F	F	F	A	F	A	F	B	B							
2									B	B	B	B	B	B	A	B	B	B	B	B	B	B								

A= 8DRB284L,U2R2

B= P8DRB284L,U2R3

C= P8DRB265H,U1R5

D= P8DRB284L,U1R5

E= P8DRB284Z,U1R5

F= P8DRB284L,U2R5

Q= QUAD+DEMO,U2R5

FIGURE 2 BF2CY6, GLRWOB – ICF

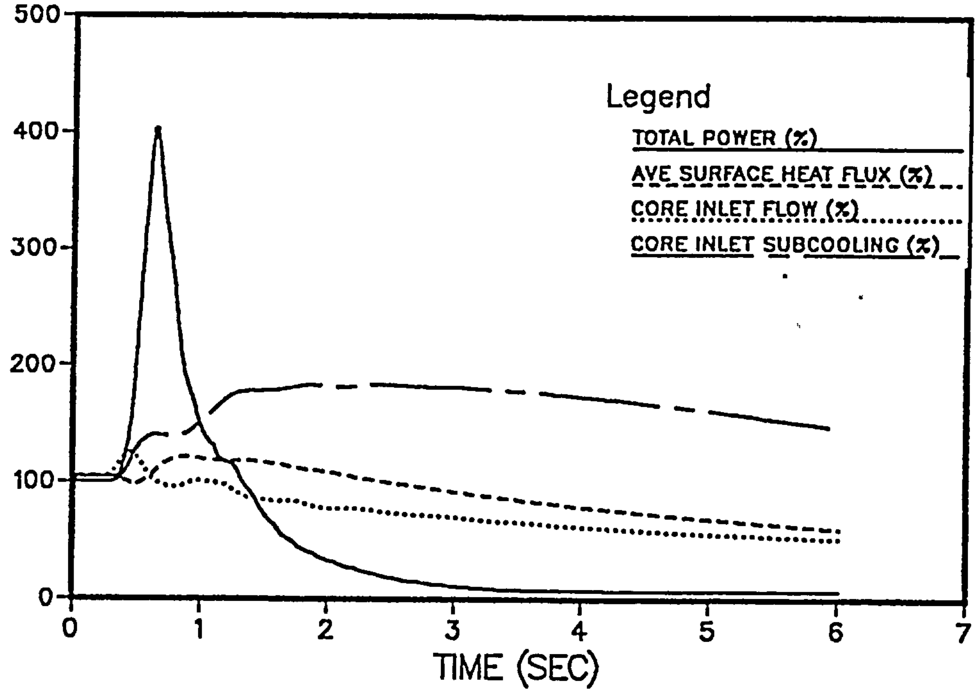


FIGURE 3 BF2CY6, GLRWOB – ICF

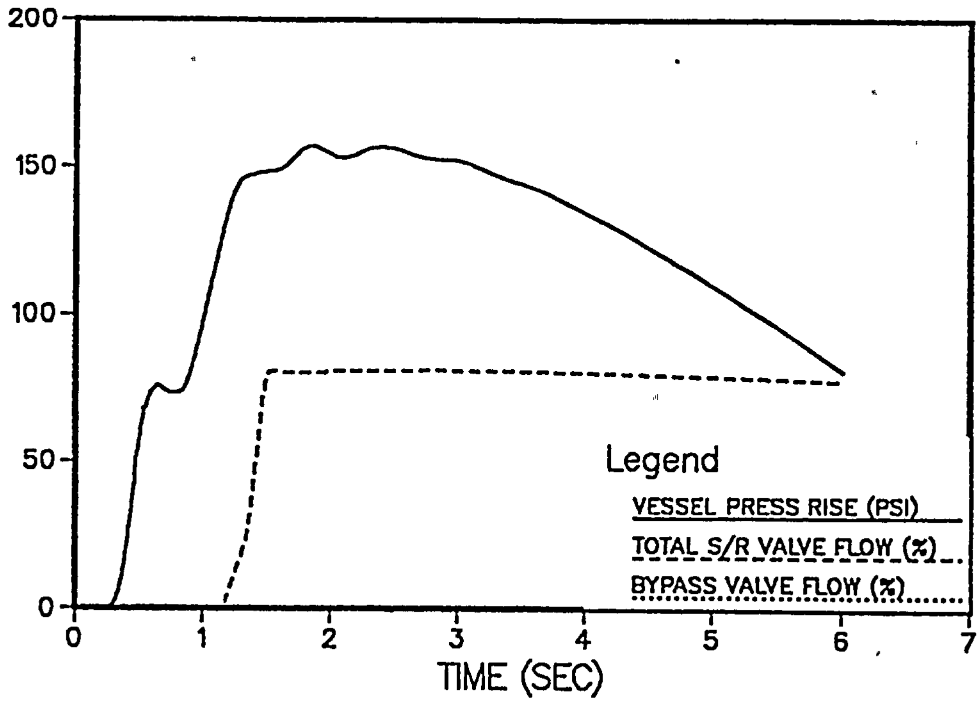


FIGURE 4 BF2CY6, GLRWOB – ICF

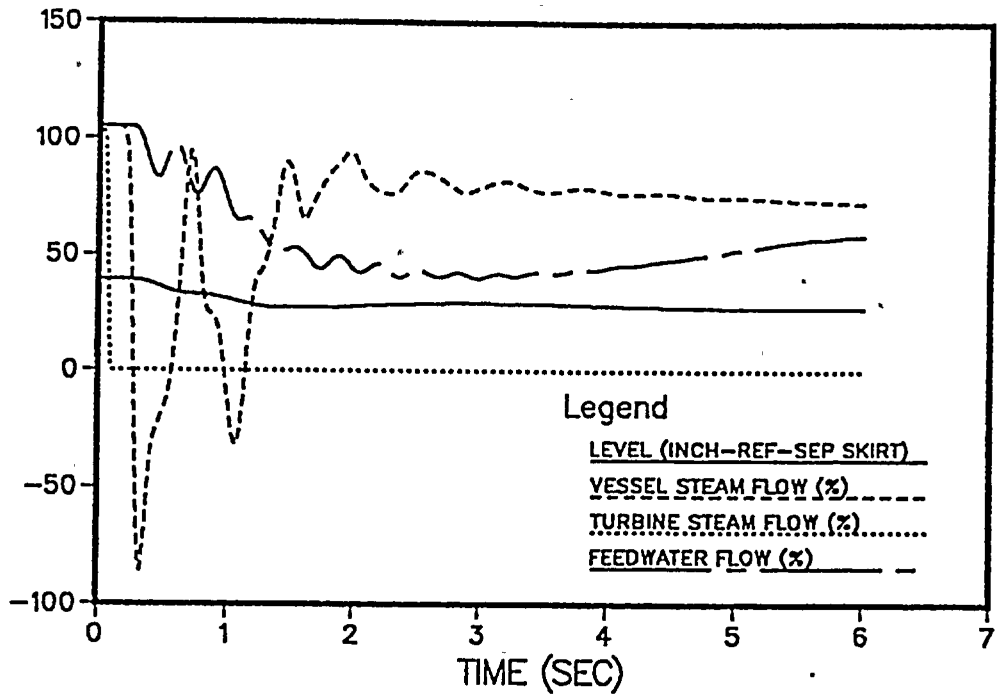


FIGURE 5 BF2CY6, GLRWOB – ICF

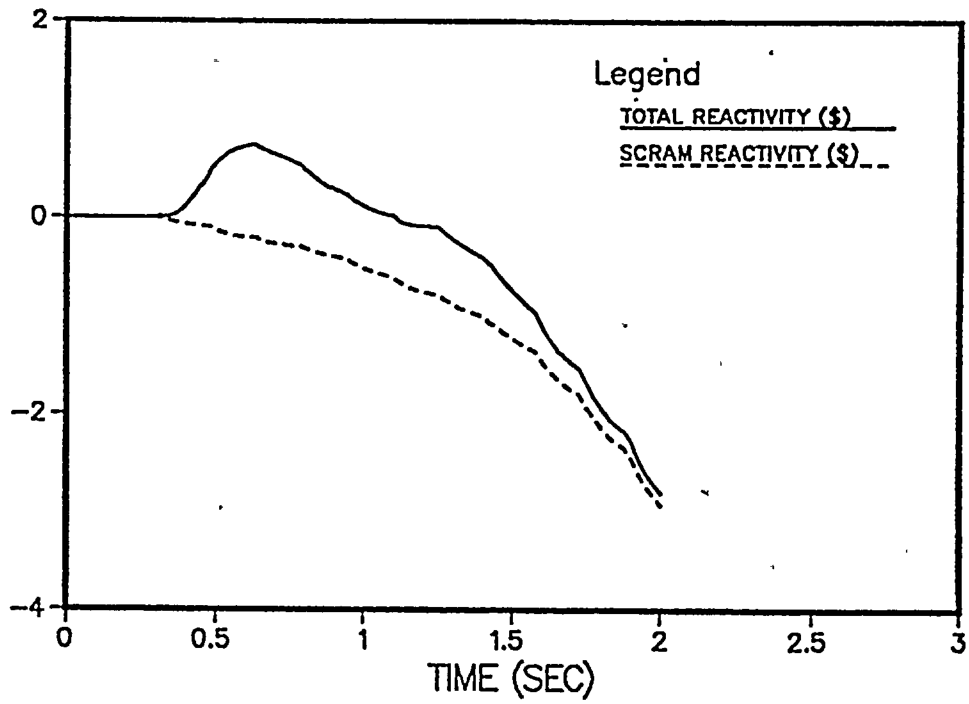


FIGURE 6 BF2CY6, FWCF – ICF

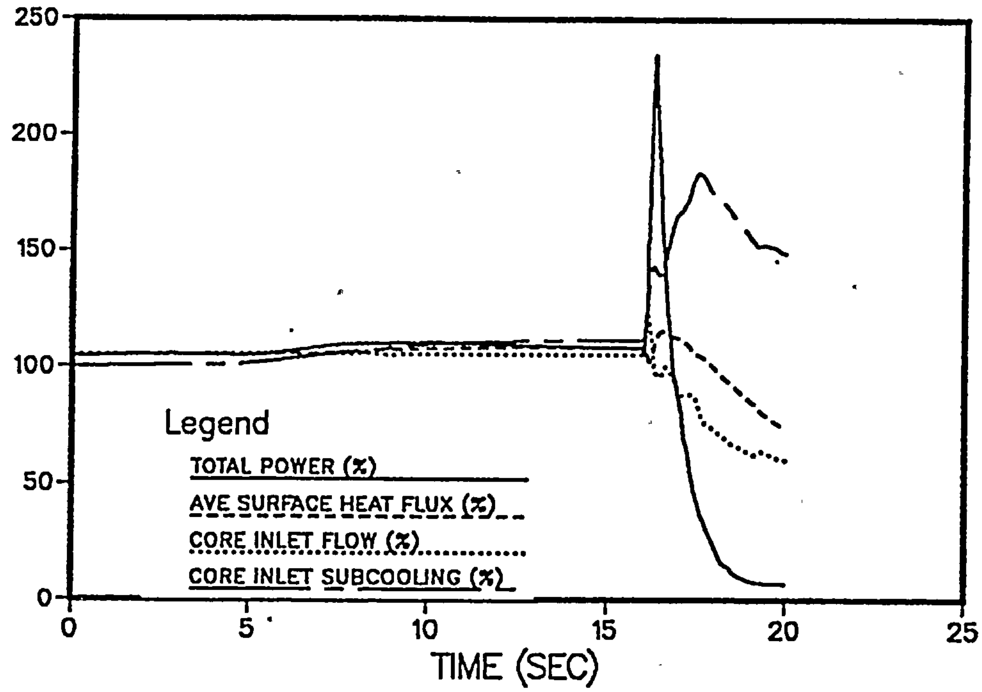


FIGURE 7 BF2CY6, FWCF – ICF

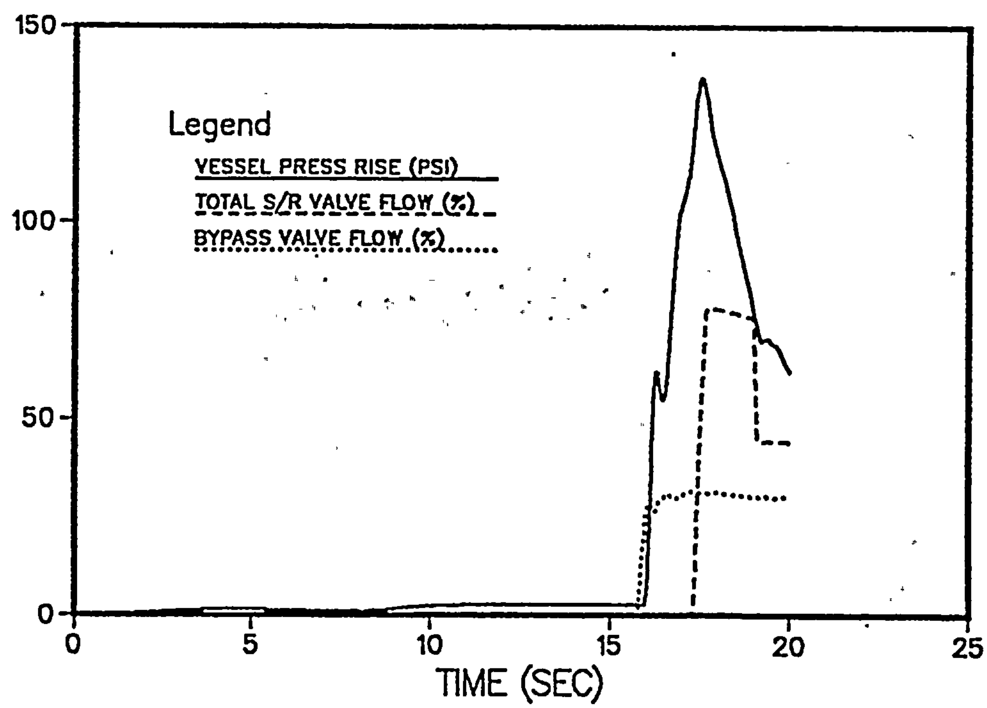


FIGURE 8 BF2CY6, FWCF – ICF

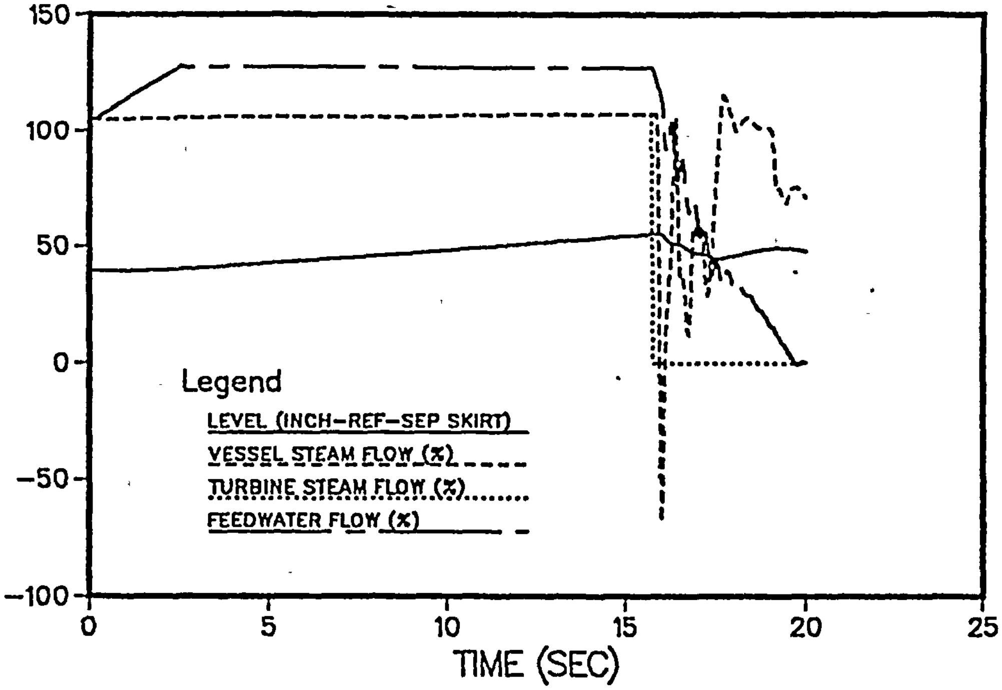


FIGURE 9 BF2CY6, FWCF – ICF

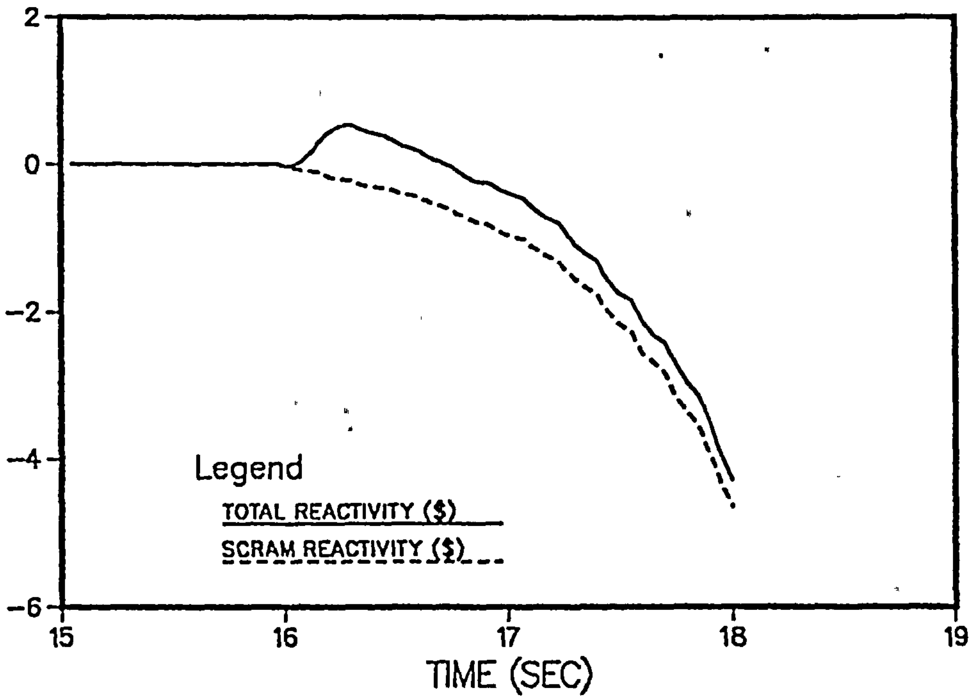


FIGURE 10 BF2CY6, MSIVC – ICF

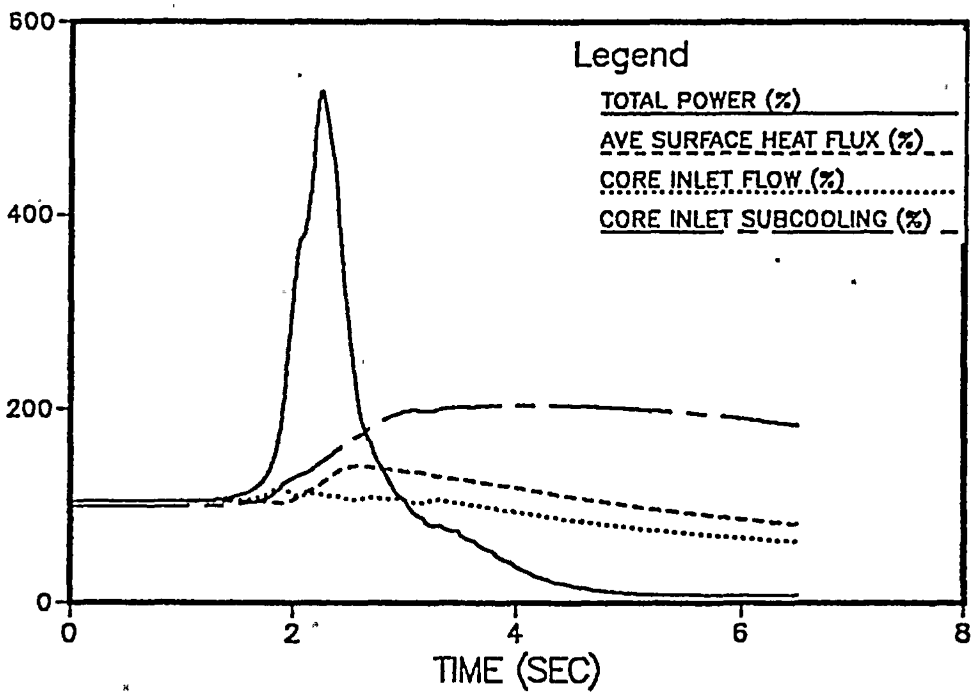


FIGURE 11 BF2CY6, MSIVC – ICF

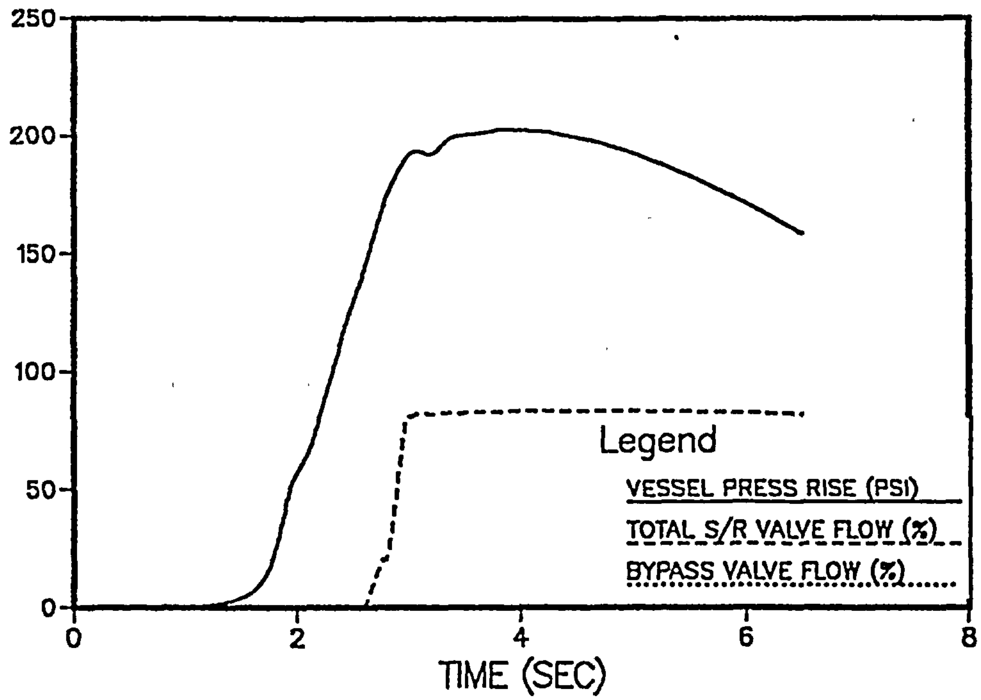


FIGURE 12 BF2CY6, MSIVC – ICF

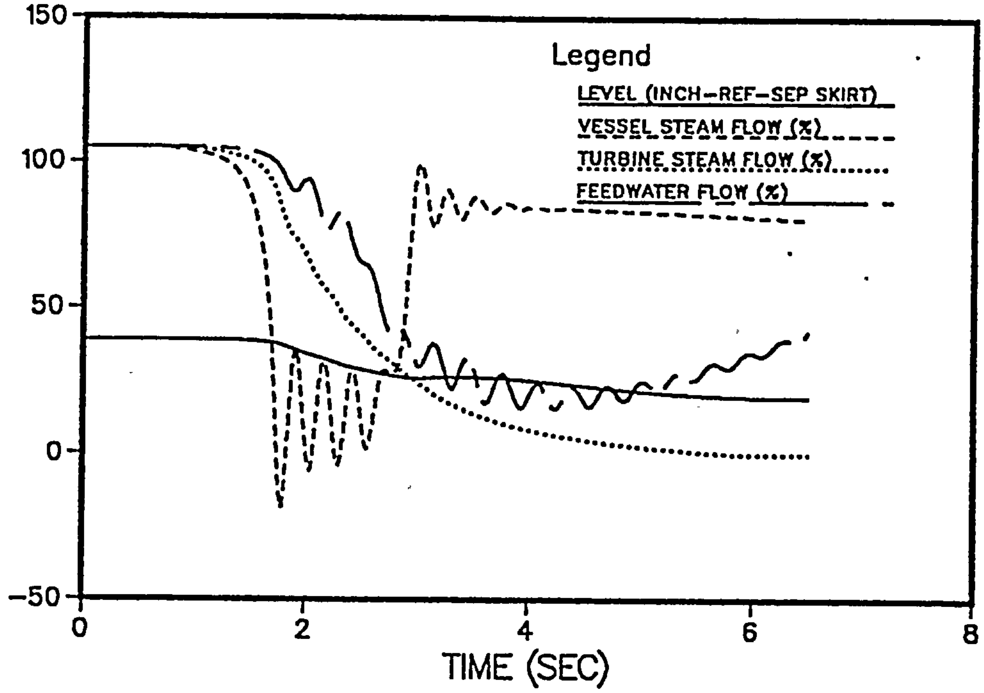


FIGURE 13 BF2CY6, MSIVC – ICF

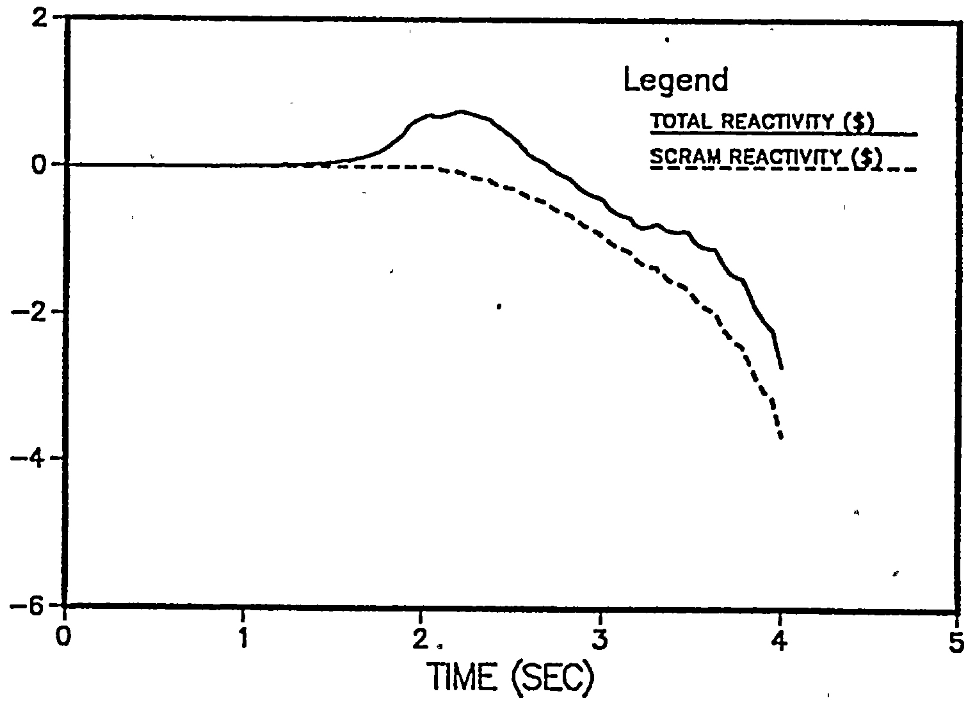
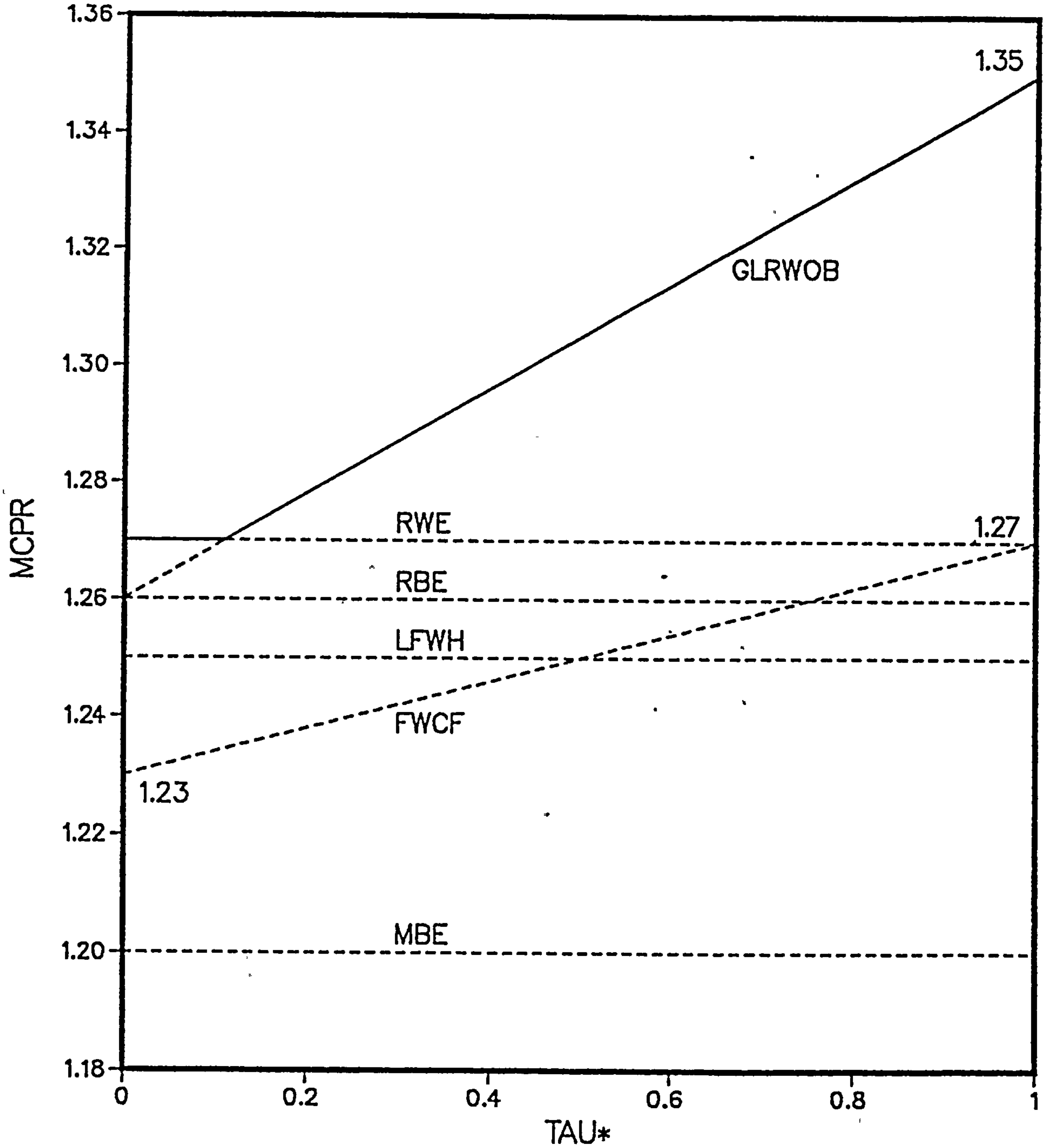


Figure 14
OLMCPR for P8X8R/8X8R/QUAD+



*SCRAM Speed Interpolation Parameter as Defined in the Technical Specifications

Figure 15
BF2CY6 Rod Drop Accident

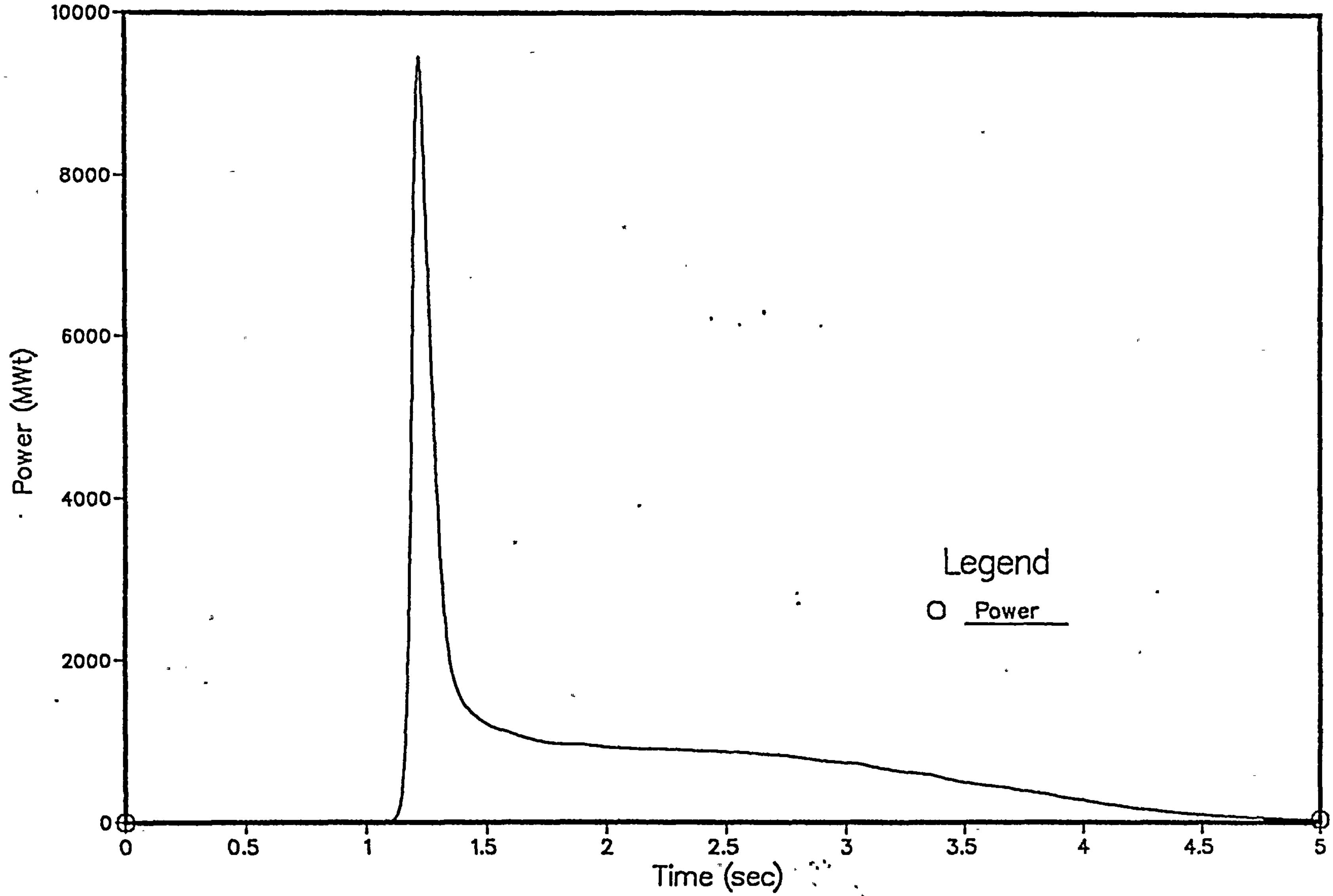


Figure 16
BF2CY6 Rod Drop Accident

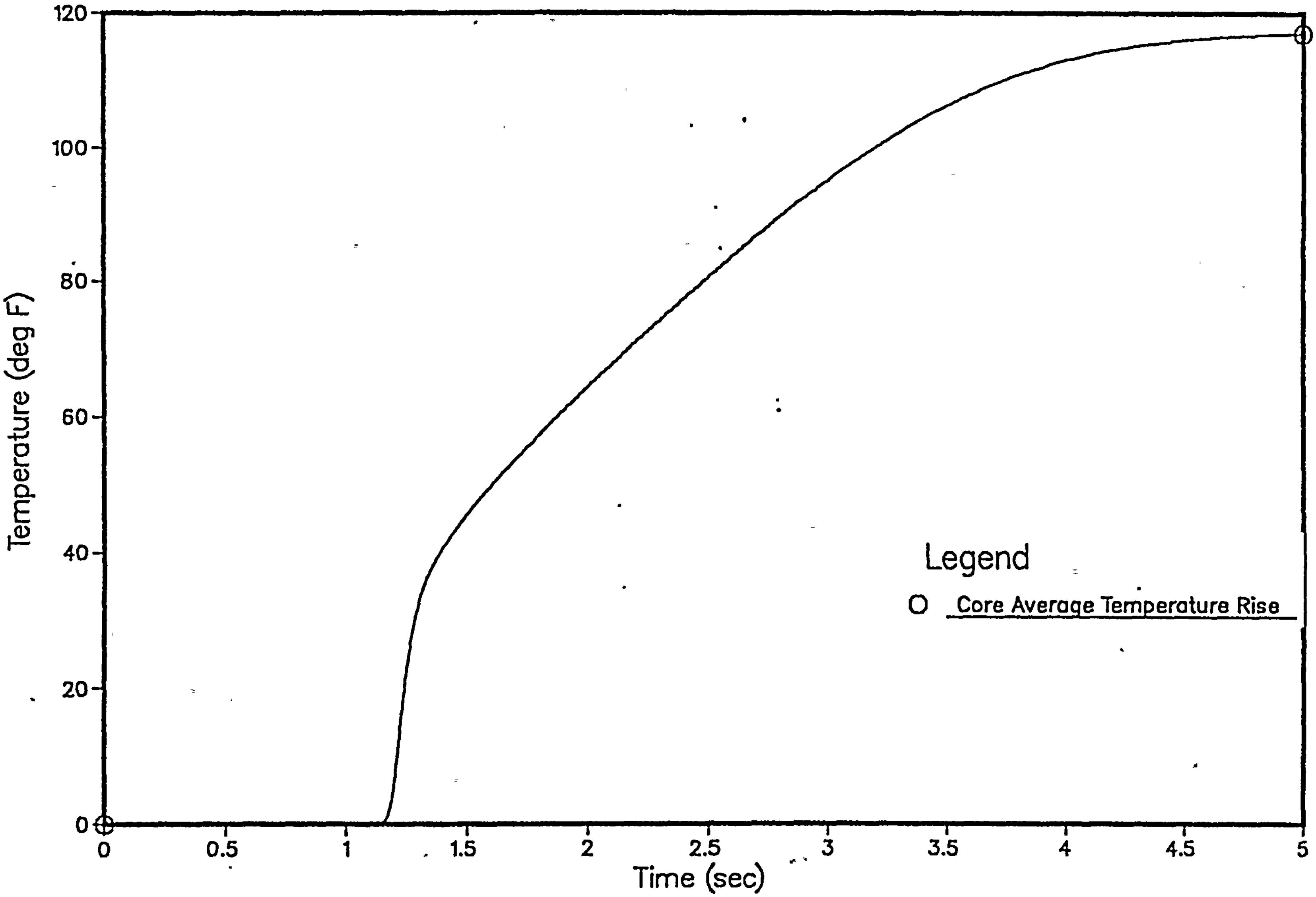


Figure 17
BF2CY6 Rod Drop Accident

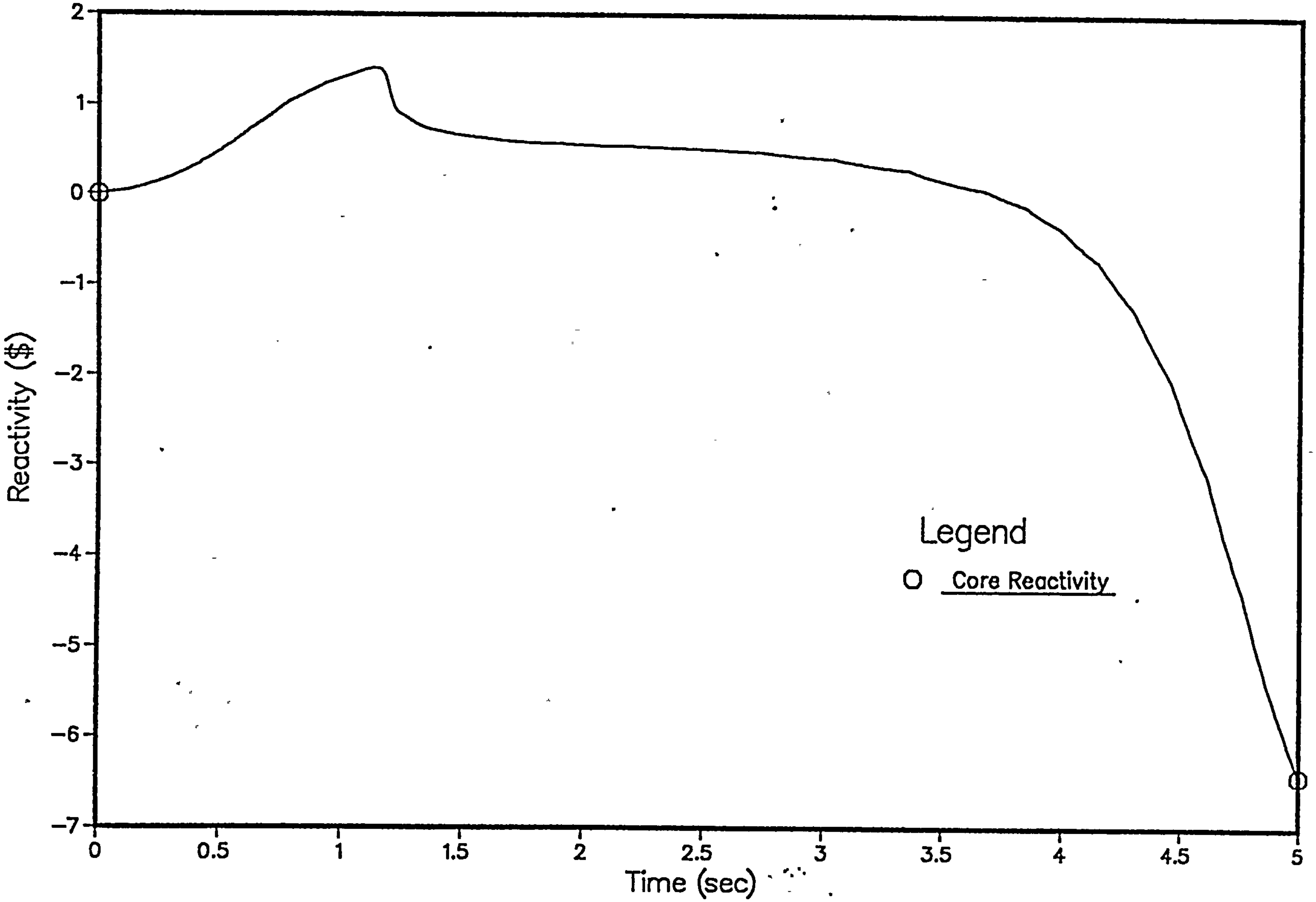


Figure 18
BF2CY6 Rod Drop Accident

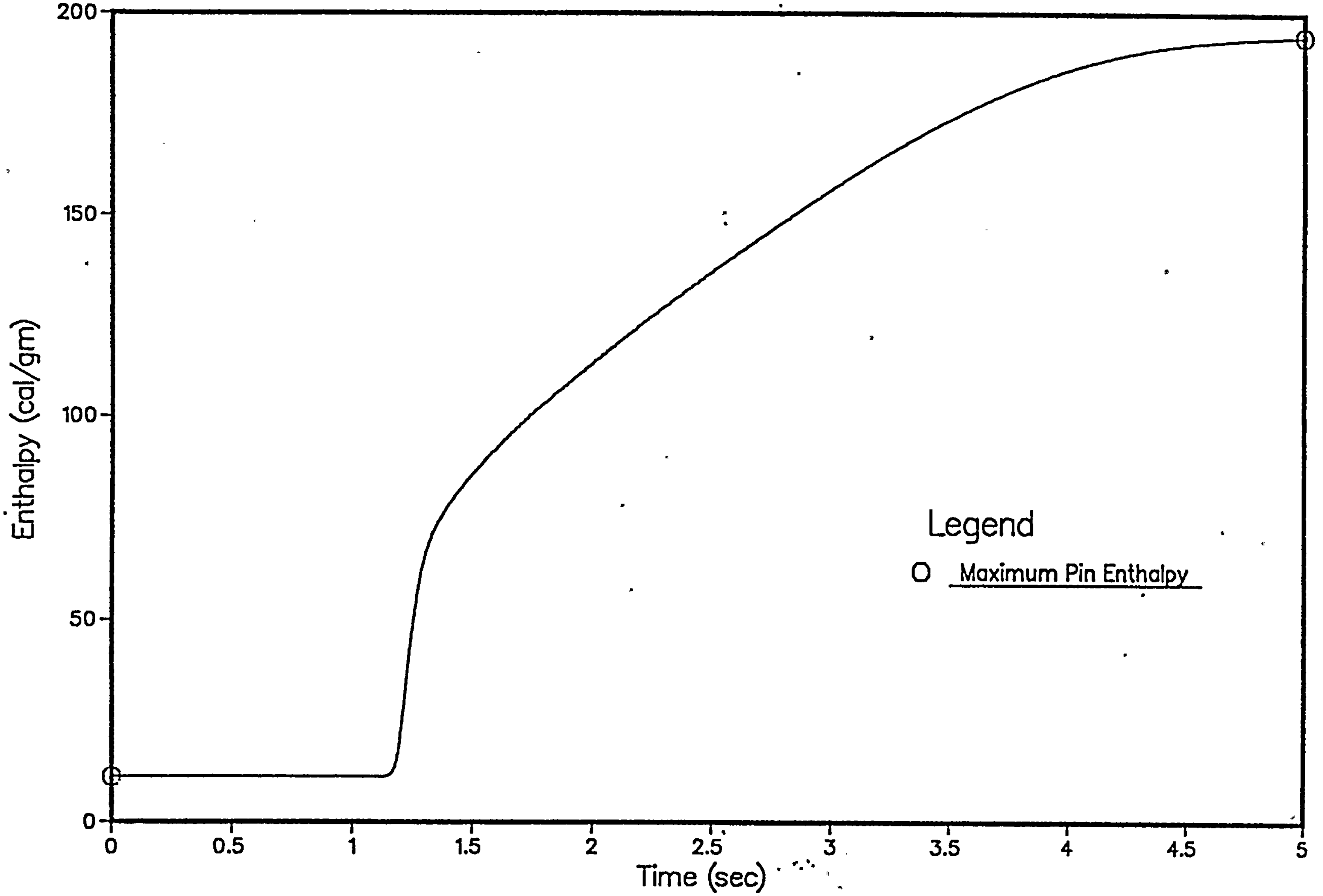


Figure 19
Decay Ratio vs. Power

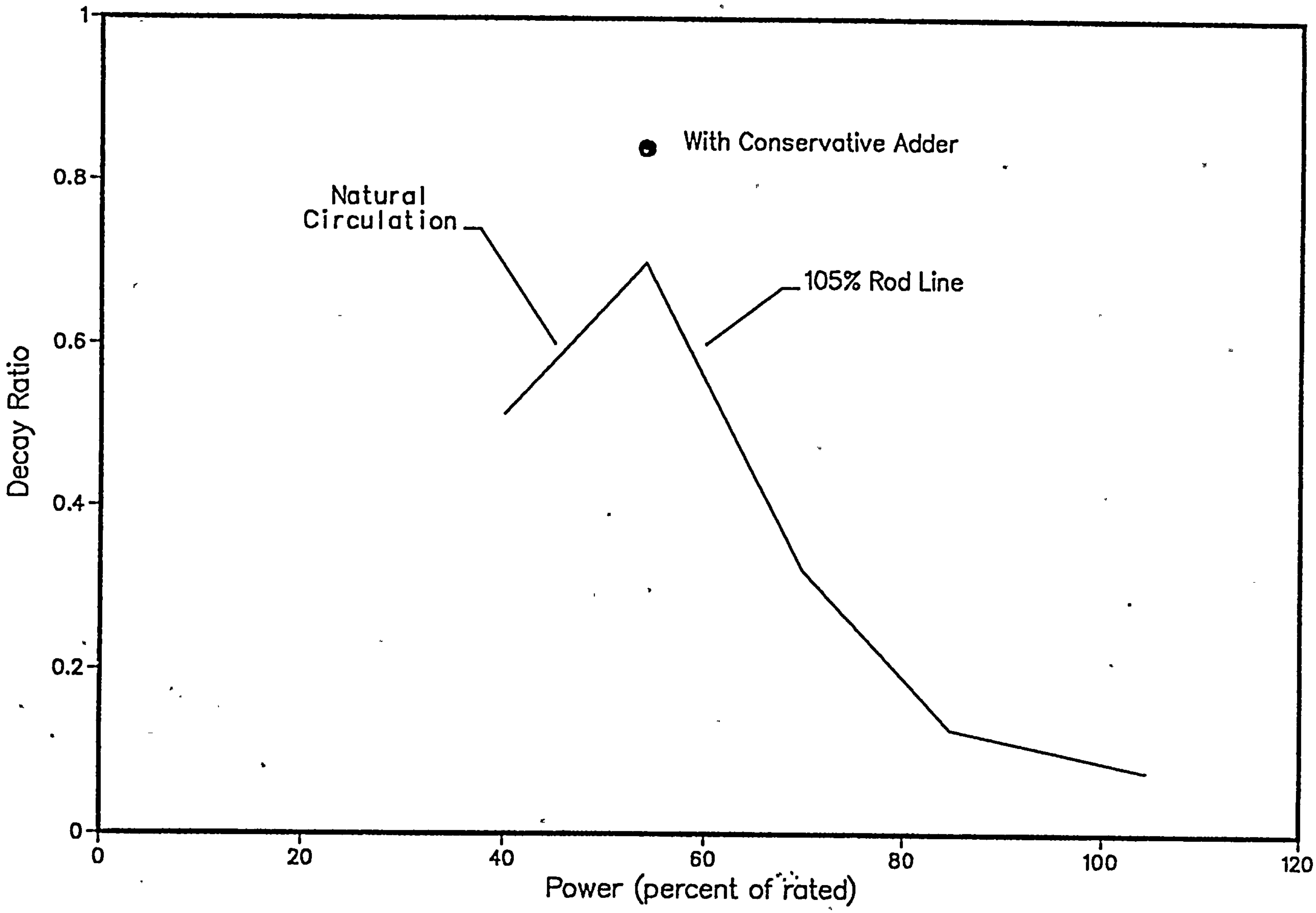


Figure 20
Bundle Power Comparison: QUAD+ vs Lead Bundle

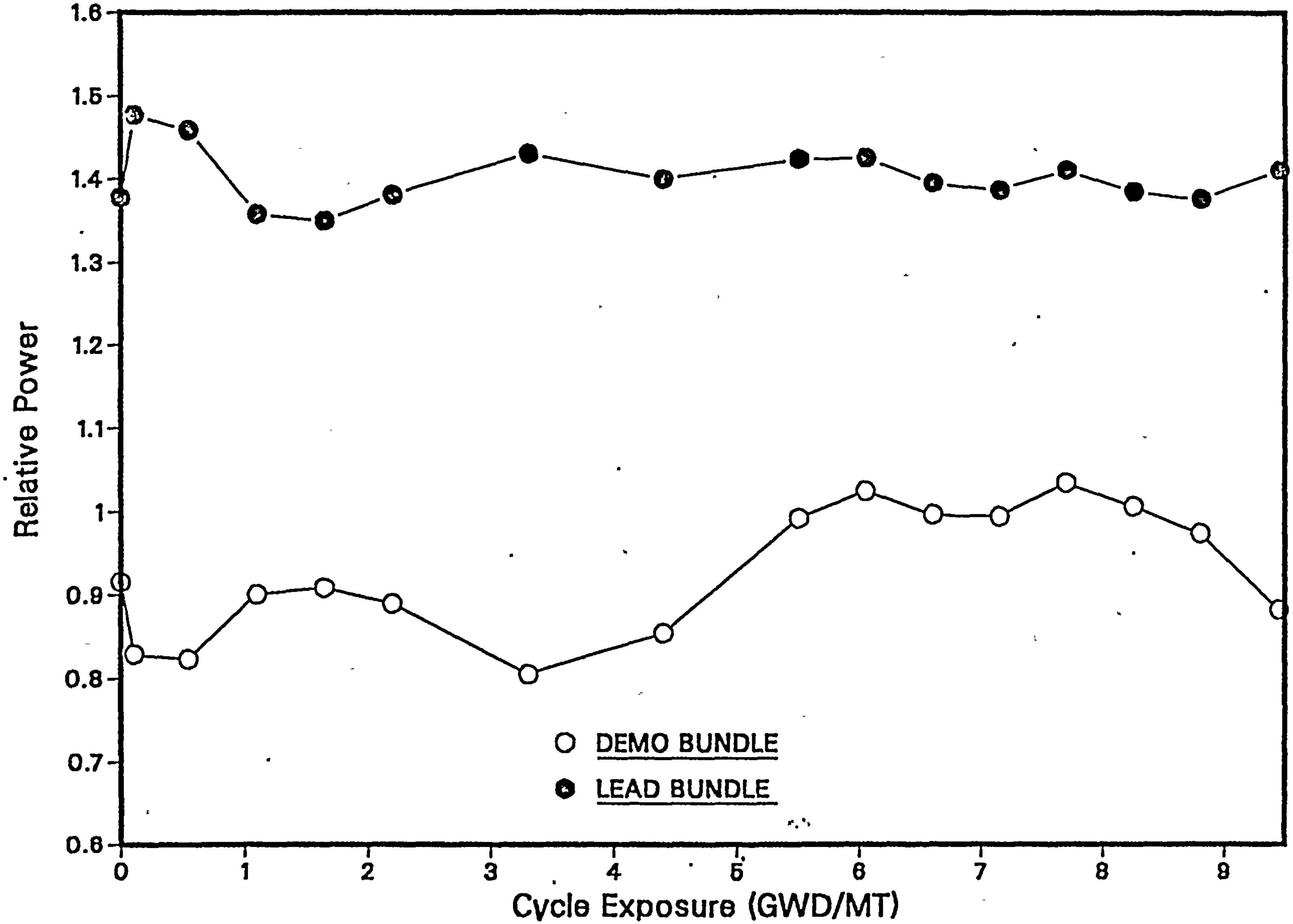


Figure 21
MCPR Comparison: QUAD+ vs Lead Bundle

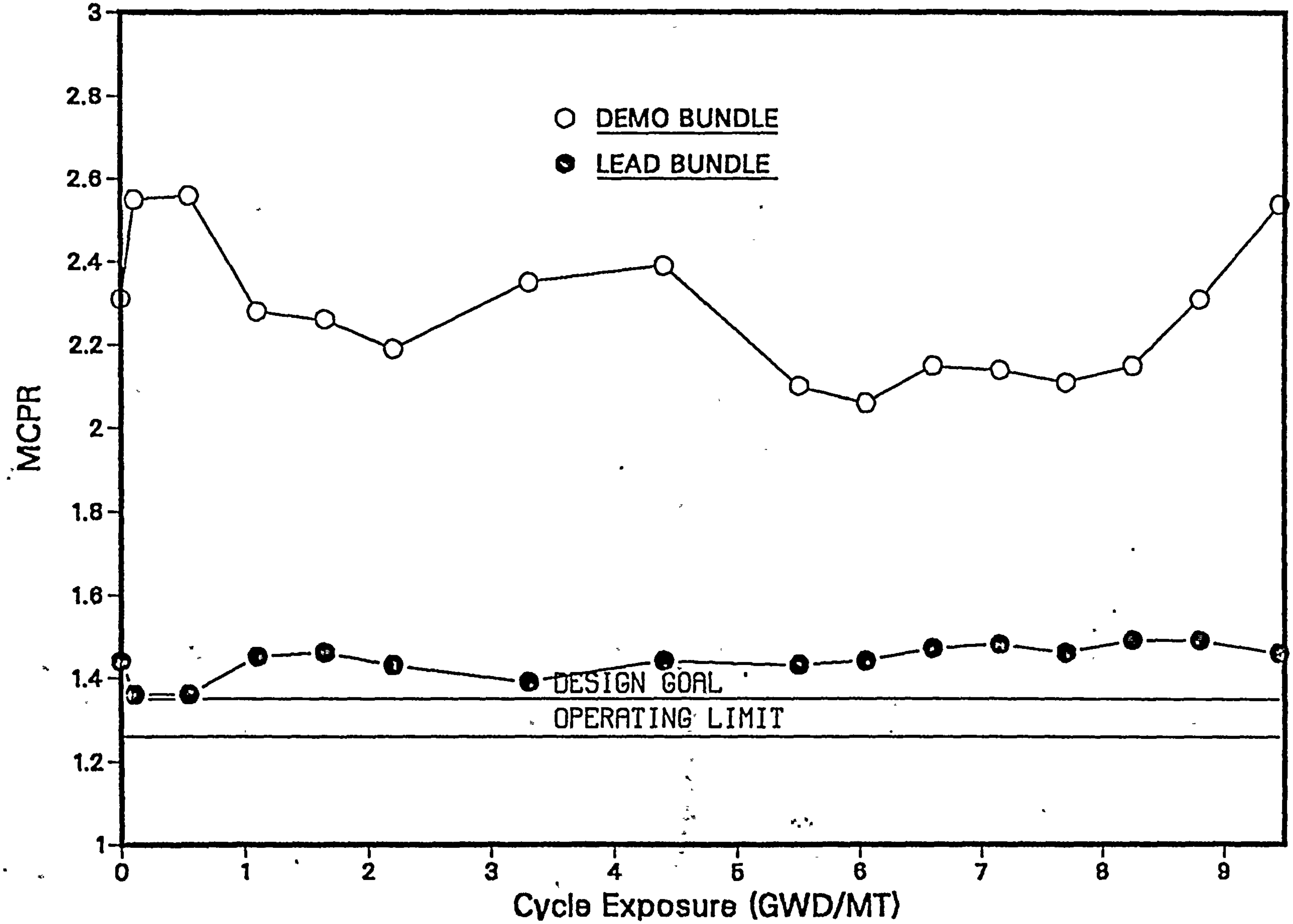
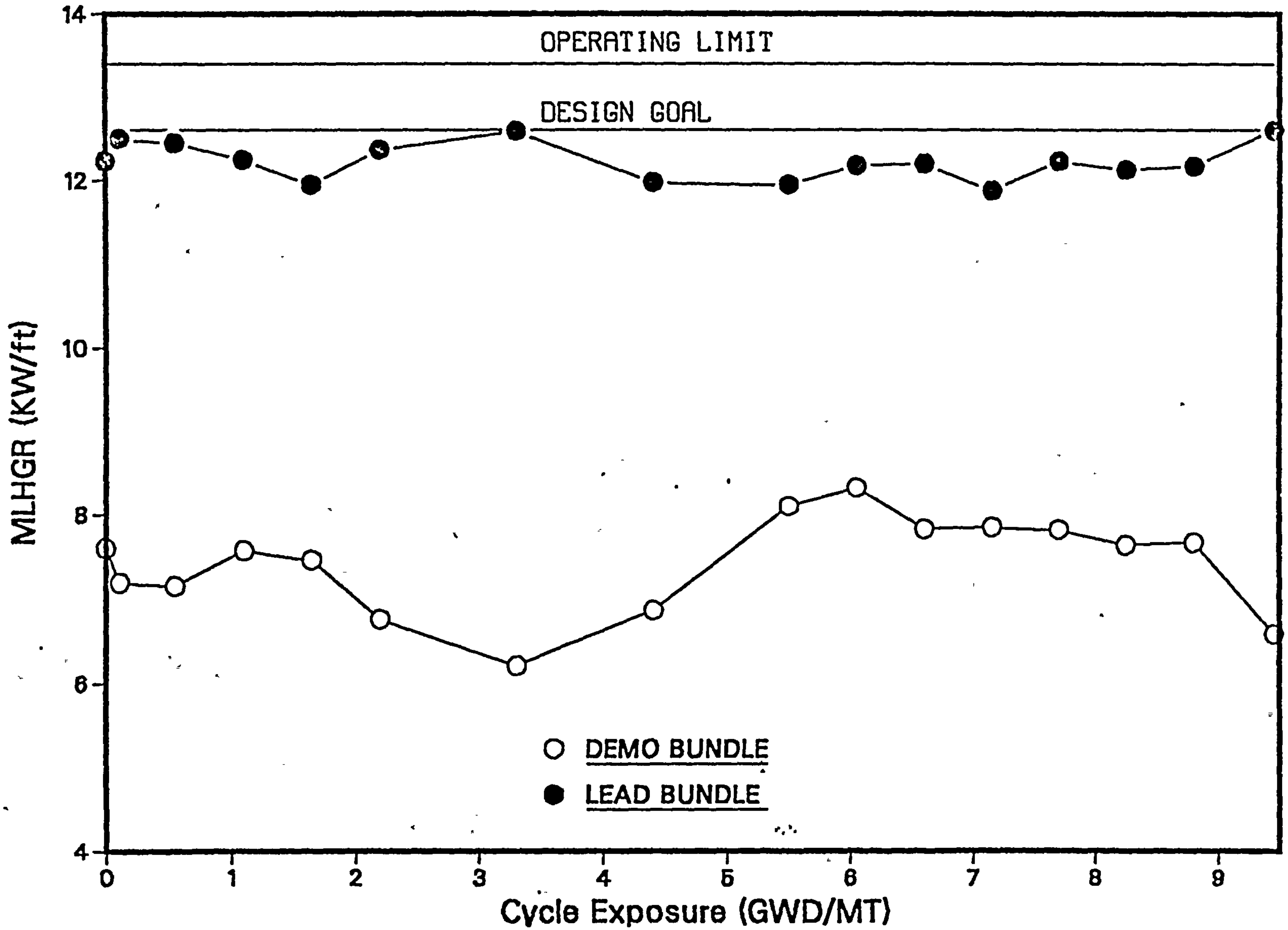


Figure 22
MLHGR Comparison: QUAD+ vs Lead Bundle



ENCLOSURE 1

PROPOSED
TECHNICAL SPECIFICATION CHANGES

BROWNS FERRY UNIT 2, CYCLE 6

BASED ON
BROWNS FERRY NUCLEAR PLANT RELOAD LICENSING REPORT
UNIT 2, CYCLE 6
TVA-RLR-002 REVISION 2

LIMITING CONDITIONS FOR OPERATION

3.5.I Average Planar Linear Heat Generation Rate

During steady-state power operation, the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Tables 3.5.I-1, 2, 3, and 4. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

J. Linear Heat Generation Rate (LHGR)

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed 13.4 kW/ft. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.5.I Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

J. Linear Heat Generation Rate (LHGR)

The LHGR shall be checked daily during reactor fuel operation at $\geq$ 25% rated thermal power.

Table 3.5.I-1

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE
Fuel Type: P8DRB284L/QUAD+

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	10.2
40,000	9.5
45,000	8.8

Table 3.5.I-2

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE
Fuel Type: P8DRB265H

Average Planar Exposure (MWd/T)	MAPLHGR (kW/ft)
200	11.5
1,000	11.6
5,000	11.9
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7
35,000	10.2
40,000	9.6

Table 3.5.I-3

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE
Fuel Type: P8DRB284Z

Average Planar Exposure (MWd/T)	MAPLHGR (kW/ft)
200	11.2
1,000	11.2
5,000	11.7
10,000	12.0
15,000	12.0
20,000	11.8
25,000	11.1
30,000	10.4
35,000	9.8
40,000	9.1
45,000	8.5

Table 3.5.I-4

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE
Fuel Type: 8DRB284L

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	10.2
40,000	9.5

Table 3.5.I-1

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE
Fuel Type: P8DRB284L/QUAD+

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	10.2
40,000	9.5
45,000	8.8

Table 3.5.I-2

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE
Fuel Type: P8DRB265H

Average Planar Exposure (MWd/T)	MAPLHGR (kW/ft)
200	11.5
1,000	11.6
5,000	11.9
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7
35,000	10.2
40,000	9.6

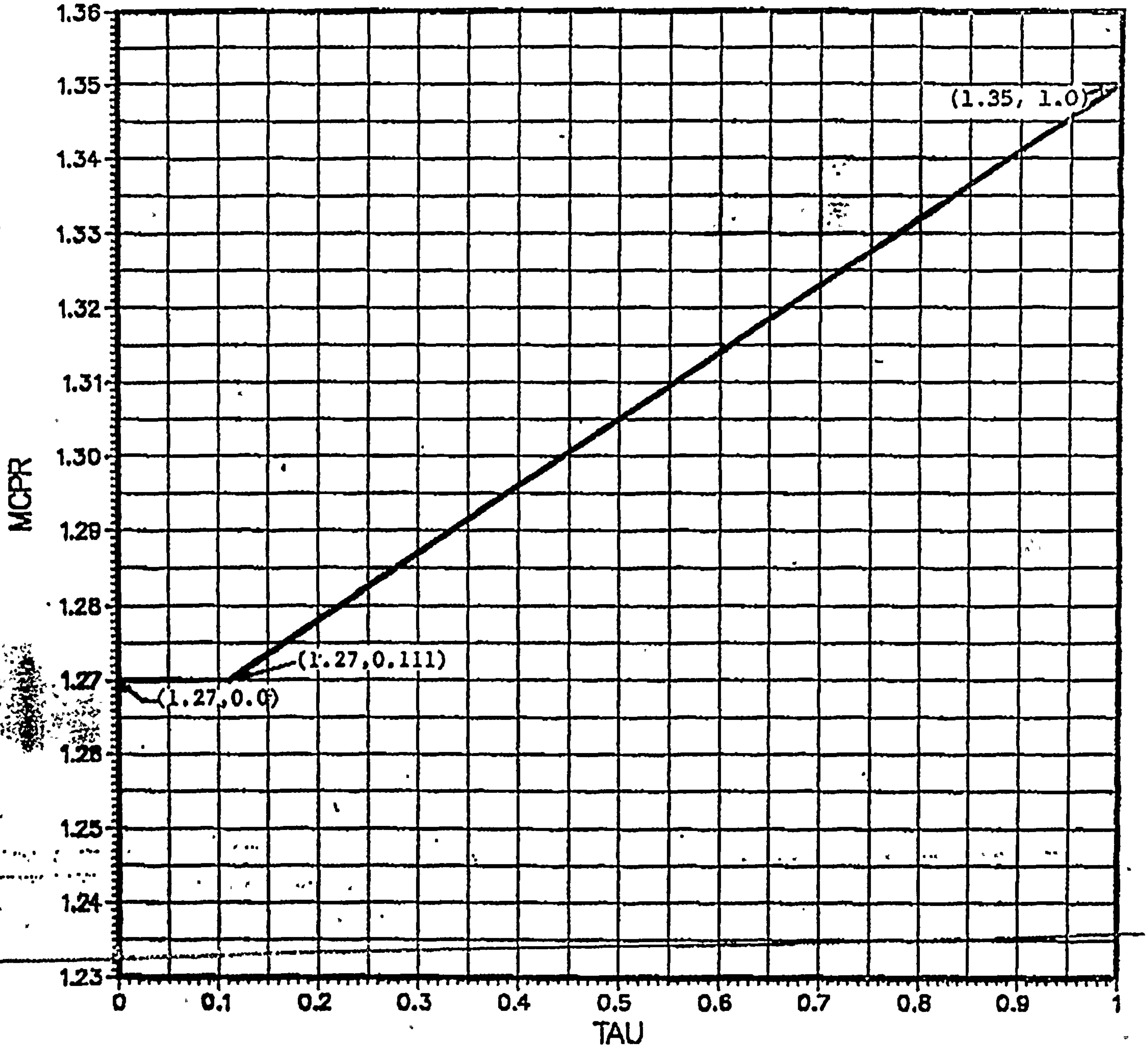


Figure 3.5.K–1
MCPR Limits for P8 X 8R/8 X 8R/ QUAD+

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting value for MAPLHGR is shown in Tables 3.5.I-1, 2, 3, and 4. The analyses supporting these limiting values are presented in Reference 1.

3.5.J. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated.

The LHGR shall be checked daily during reactor operation at ≥ 25 percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent rated thermal power, the R factor would have to be less than 0.241 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

3.5.L. APRM Setpoints

Operation is constrained to a maximum LHGR of 13.4 kW/ft for 8x8 fuel. This limit is reached when core maximum fraction of limiting power density (CMFLPD) equals 1.0. For the case where CMFLPD exceeds the fraction of rated thermal power, operation is permitted only at less than 100-percent rated power and only with APRM scram settings as required by Specification 3.5.L.1. The scram trip setting and rod block trip setting are adjusted to ensure that no combination of CMFLPD and FRP will increase the LHGR transient peak beyond that allowed by the 1-percent plastic strain limit. A 6-hour time period to achieve this condition is justified since the additional margin gained by the setdown adjustment is above and beyond that ensured by the safety analysis.

ENCLOSURE 2
DESCRIPTION, REASON AND JUSTIFICATION FOR CHANGE
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 2

Description Of Change

The BFN Unit 2 Technical Specifications are being updated to reflect the limits for cycle 6 operations. The cycle 6 core loading has been changed because of the results of inspection and reconstitution of the fuel completed in July 1988.

The actual changes are a slight adjustment in the Minimum Critical Power Ratio (MCPR) and the addition of two Tables of Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus average planar exposure.

Reason For Change

The Minimum Critical Power Ratio as a function of scram time (figure 3.5.K-1) has changed because of the reanalysis performed to include BFN Unit 1 fuel in the Unit 2 reload.

The MAPLHGR for each type of fuel as a function of average planar exposure is presented in tables 3.5.I-1, 2, 3, and 4. These tables have changed because of the inclusion of a different fuel type from BFN Unit 1, and the pressurized and unpressurized MAPLHGR have been separated into two tables. Technical specification 3.5.I and the bases are changed to reflect the addition of the two MAPLHGR Tables (Table 3.5.I-3 and 4).

Justification For Change

The initial cycle 6 reload was submitted to NRC by letter dated August 23, 1984, and was approved by the issurance of BFN Technical Specification 199 dated August 19, 1986. The cycle 6 core loading has changed as a result of the fuel inspection and reconstitution program completed in July 1988. The justification and safety analysis results for the changes are presented in TVA-RLR-002 Revision 2, July 1988, "Reload Licensing Report for Browns Ferry Unit 2 Cycle 6." A summary is presented below.

Figure 3.5.K-1 MCPR vs TAU is changed because of the reanalysis. The reanalysis indicated the bounding accidents are rod withdrawal error and generator load reject without bypass. All of the accidents and the bounding envelope are shown in figure 14 of the Reload Licensing Report.

Four MAPLHGR figures are required to define the limits for all fuel to be loaded in cycle 6. The current technical specification have two of these figures. Table 3.5.I-3 is specific to fuel type P8DRB284Z. This fuel type was not in the initial cycle 6 fuel load but was added as a result of the fuel inspection and reconstitution program. Table 3.5.I-4 was added to separate the pressurized (P8DRB284L/QUAD + shown in Table 3.5.I-1) and the non-pressurized fuel (8DRB284L). The pressurized fuel allows higher exposures.

The changes to specification 3.5.I and the Bases are administrative in nature to reference the additional MAPLHGR tables.

ENCLOSURE 3

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 2

Description of Proposed Amendment

The BFN Unit 2 Technical Specifications are being updated to reflect the limits for cycle 6 operations. The changes consist of a slight revision to the Minimum Critical Power Ratio (MCPR) and the addition of two Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus average planar exposure tables.

Basis for Proposed No Significant Hazards Consideration Determination

NRC has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92 (c). A proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from an accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. Operational transients analyzed in the Final Safety Analysis Report have been reevaluated in detail. The Reload Licensing Report for Browns Ferry Unit 2, Cycle 6, Revision 2, provides a summary of the limiting operating transient, stability, and selected accident analyses for the proposed core arrangement. The 8x8 fuel assemblies to be installed in the core are not significantly different from the 8x8 fuel assemblies they are replacing. The NRC staff has previously approved the design of the GE P8x8R assemblies as described in the GESTAR document (NEDO-24011-P-A-8). The NRC staff has previously evaluated and approved the use of four Westinghouse designed QUAD + demonstration assemblies in the low power region of the core. The NRC staff has also approved the analysis methods used by TVA.

2. The proposed amendment does not create the possibility of a new or different accident. This reload changes the initial conditions and/or final condition used in the existing analyses and does not create any new accident mode.

3. The proposed amendment does not involve a significant reduction in a margin of safety because the plant will be operated under the same safety limits with MCPR and MAPLHGR operating limits comparable to those currently established. The Reload Licensing Report provides a summary of the limiting operating transient, stability, and selected accident analyses for the proposed core arrangement. The MCPR and MAPLHGR limits have been revised to assure the margin of safety is maintained as demonstrated in the Reload Licensing Report for Browns Ferry Unit 2, Cycle 6, Revision 2.

Based on the above reasoning, TVA has determined that the proposed amendment does not involve a significant hazards consideration.