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# SUPPLEMENTAL RELOAD LICENSING REPORT

FOR

BRUNSWICK STEAM ELECTRIC PLANT
UNIT 1, RELOAD 6 (CYCLE 7)
(WITHOUT RECIRCULATION PUMP TRIP)

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# IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

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## ACKNOWLEDGEMENT

The engineering and reload licensing analyses which form the technical basis of this Supplemental Reload Licensing Report were performed by R. E. Polomik and T. P. Lung of the Fuel Engineering Section.

1. PLANT-UNIQUE ITEMS (1.0)*

| | |
|---|---|
| Limiting Conditions for Operation | Appendix A |
| Bases for Limiting Conditions for Operation | Appendix B |
| Plant Parameter Differences | Appendix C |
| Use of GEXL-PLUS Methods for Cycle 7 | Appendix D |
| Use of New Safety Limit MCPR for Cycle 7 | Appendix E |

2. RELOAD FUEL BUNDLES (1.0, 2.0, 3.3.1 AND 4.0)

| Fuel Type | Cycle Loaded | Number |
|---|---|---|
| Irradiated | | |
| P8DRB284H | 4 | 8 |
| P8DRB299 | 4 | 8 |
| BP8DRB299 | 5 | 184 |
| BP8DRB299 | 6 | 176 |
| New | | |
| BD339A | 7 | 60 |
| BD323B | 7 | 124 |
| Total | | 560 |

3. REFERENCE CORE LOADING PATTERN (3.3.1)

| | | |
|---|---|---|
| Nominal previous cycle core average exposure at end of cycle: | 21,072 | MWd/MT |
| Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations: | 20,481 | MWd/MT |
| Assumed reload cycle core average exposure at end of cycle: | 21,230 | MWd/MT |
| Core loading pattern: | Figure 1 | |

*( ) Refers to area of discussion in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-8, dated May 1986. A letter "S" preceding the number refers to the U.S. Supplement, NEDE-24011-P-A-8-US, May 1986.

4. CALCULATED CORE EFFECTIVE MULTIPLICATION AND CONTROL SYSTEM WORTH - NO VOIDS, 20°C (3.3.2.1.1 and 3.3.2.1.2)

Beginning of Cycle, $K_{eff}$

| | |
|---|---|
| Uncontrolled | 1.111 |
| Fully Controlled | 0.966 |
| Strongest Control Rod Out | 0.988 |
| R, Maximum Increase in Cold Core Reactivity with Exposure into Cycle, ΔK | 0.002 |

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

| ppm | Shutdown Margin (ΔK) (20°C, Xenon Free) |
|---|---|
| 600 | 0.036 |

6. RELOAD-UNIQUE TRANSIENT ANALYSIS INPUT (3.3.2.1.5 AND S.2.2)

(Cold Water Injection Events Only)

| | |
|---|---|
| Void Fraction (%) | 41.7 |
| Average Fuel Temperature (°F) | 1096 |
| Void Coefficient N/A* (¢/% Rg) | -6.710/-8.387 |
| Doppler Coefficient N/A* (¢/°F) | -0.201/-0.191 |
| Scram Worth N/A* ($) | ** |

*N = Nuclear Input Data, A = Used in Transient Analysis
**Generic exposure independent values are used as given in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-8, dated May 1986.

7. RELOAD-UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (S.2.2)

| Fuel Design | Local | Peaking Factors Radial | Axial | R-Factor | Bundle Power (MWt) | Bundle Flow (1000 lb/hr) | Initial (MCPR) |
|---|---|---|---|---|---|---|---|
| Exposure: | BOC7 to EOC7-2000 MWd/ST | | | | | | |
| BP/P8x8R | 1.20 | 1.51 | 1.40 | 1.051 | 6.409 | 109.5 | 1.25 |
| GE8X8EB | 1.20 | 1.51 | 1.40 | 1.051 | 6.418 | 112.2 | 1.26 |
| Exposure: | EOC7-2000 MWd/ST to EOC7 | | | | | | |
| BP/P8x8R | 1.20 | 1.44 | 1.40 | 1.051 | 6.127 | 111.7 | 1.31 |
| GE8x8EB | 1.20 | 1.45 | 1.40 | 1.051 | 6.139 | 114.3 | 1.32 |

8. SELECTED MARGIN IMPROVEMENT OPTIONS (S.2.2.2)

| | |
|---|---|
| Transient Recategorization | No |
| Recirculation Pump Trip: | No |
| Rod Withdrawal Limiter: | No |
| Thermal Power Monitor: | Yes |
| Measured Scram Time: | No |
| Exposure Dependent Limits: | Yes |
| Exposure Points Analyzed: | EOC7-2000 MWd/ST and EOC7 |

9. OPERATING FLEXIBILITY OPTIONS (S.2.2.3)

| | |
|---|---|
| Single-Loop Operation: | Yes |
| Load Line Limit: | Yes |
| Extended Load Line Limit: | No |
| Increased Core Flow: | No |
| Flow Point Analyzed: | N/A |
| Feedwater Temperature Reduction: | No |
| ARTS Program: | No |
| Maximum Extended Operation Domain: | No |

## 10. CORE-WIDE TRANSIENT ANALYSIS RESULTS (S.2.2.1)

Methods Used: GEMINI

Exposure Range: BOC7 to EOC7

| Transient | Flux (% NBR) | Q/A (% NBR) | ΔCPR BP/P8x8R | ΔCPR GE8x8EB | Figure |
|---|---|---|---|---|---|
| Inadvertent HPCI | 122 | 119 | 0.15 | 0.15 | 2 |

Exposure Range: BOC7 to EOC7-2000 MWd/ST

| Transient | Flux (% NBR) | Q/A (% NBR) | ΔCPR BP/P8x8R | ΔCPR GE8x8EB | Figure |
|---|---|---|---|---|---|
| Load Rejection Without Bypass | 482 | 120 | 0.18 | 0.18 | 3 |
| Feedwater Controller Failure | 285 | 116 | 0.12 | 0.13 | 4 |

Exposure Range: EOC7-2000 MWd/ST to EOC7

| Transient | Flux (% NBR) | Q/A (% NBR) | ΔCPR BP/P8x8R | ΔCPR GE8x8EB | Figure |
|---|---|---|---|---|---|
| Load Rejection Without Bypass | 592 | 125 | 0.24 | 0.25 | 5 |
| Feedwater Controller Failure | 418 | 121 | 0.18 | 0.19 | 6 |

## 11. LOCAL ROD WITHDRAWAL ERROR (WITH LIMITING INSTRUMENT FAILURE) TRANSIENT SUMMARY (S.2.2.1)

Limiting Rod Pattern: Figure 7

| Rod Block Reading (%) | Rod Position (Feet Withdrawn) | ΔCPR BP/P8x8R | ΔCPR GE8x8EB |
|---|---|---|---|
| 104 | 3.5 | 0.11 | 0.11 |
| 105 | 3.5 | 0.11 | 0.11 |
| 106 | 4.0 | 0.13 | 0.13 |
| 107 | 4.5 | 0.14 | 0.14 |
| 108 | 5.5 | 0.18 | 0.18 |
| 109 | 12.0 | 0.21 | 0.21 |
| 110 | 12.0 | 0.21 | 0.21 |

Setpoint Selected: 107

12. CYCLE MCPR VALUES (S.2.2)

Non-Pressurization Events

Exposure Range: BOC to EOC

| | BP/P8x8R | GE8x8EB |
|---|---|---|
| Inadvertent HPCI | 1.19 | 1.19 |
| Fuel Loading Error | -- | 1.25 |
| Rod Withdrawal Error | 1.18 | 1.18 |

Pressurization Events

| | Option A BP/P8x8R | Option A GE8x8EB | Option B BP/P8x8R | Option B GE8x8EB |
|---|---|---|---|---|
| Exposure Range: | | | | |
| BOC7 to EOC7-2000 MWd/ST | | | | |
| Load Rejection Without Bypass | 1.32 | 1.32 | 1.25 | 1.25 |
| Feedwater Controller Failure | 1.22 | 1.23 | 1.20 | 1.21 |
| Exposure Range: | | | | |
| EOC7-2000 MWd/ST to EOC7 | | | | |
| Load Rejection Without Bypass | 1.34 | 1.34 | 1.30 | 1.30 |
| Feedwater Controller Failure | 1.27 | 1.27 | 1.24 | 1.24 |

13. OVERPRESSURIZATION ANALYSIS SUMMARY (S.2.3)

| Transient | $P_{sl}$ (psig) | $P_v$ (psig) | Plant Response |
|---|---|---|---|
| MSIV Closure (Flux Scram) | 1235 | 1266 | Figure 8 |

14. LOADING ERROR RESULTS (S.2.5.4)

Variable Water Gap Misoriented Bundle Analysis: Yes*

| Event | ΔCPR |
|---|---|
| Misoriented | 0.19 |

15. CONTROL ROD DROP ANALYSIS RESULTS (S.2.5.1)

Bounding Analysis Results:

| | |
|---|---|
| Doppler Reactivity Coefficient: | Figure 9 |
| Accident Reactivity Shape Functions: | Figures 10 and 11 |
| Scram Reactivity Functions: | Figures 12 and 13 |

Plant-Specific Analysis Results:

| | |
|---|---|
| Resultant Peak Enthalpy, Cold: | 149.6 |
| Resultant Peak Enthalpy, HSB: | 215.6 |

16. STABILITY ANALYSIS RESULTS (S.2.4)

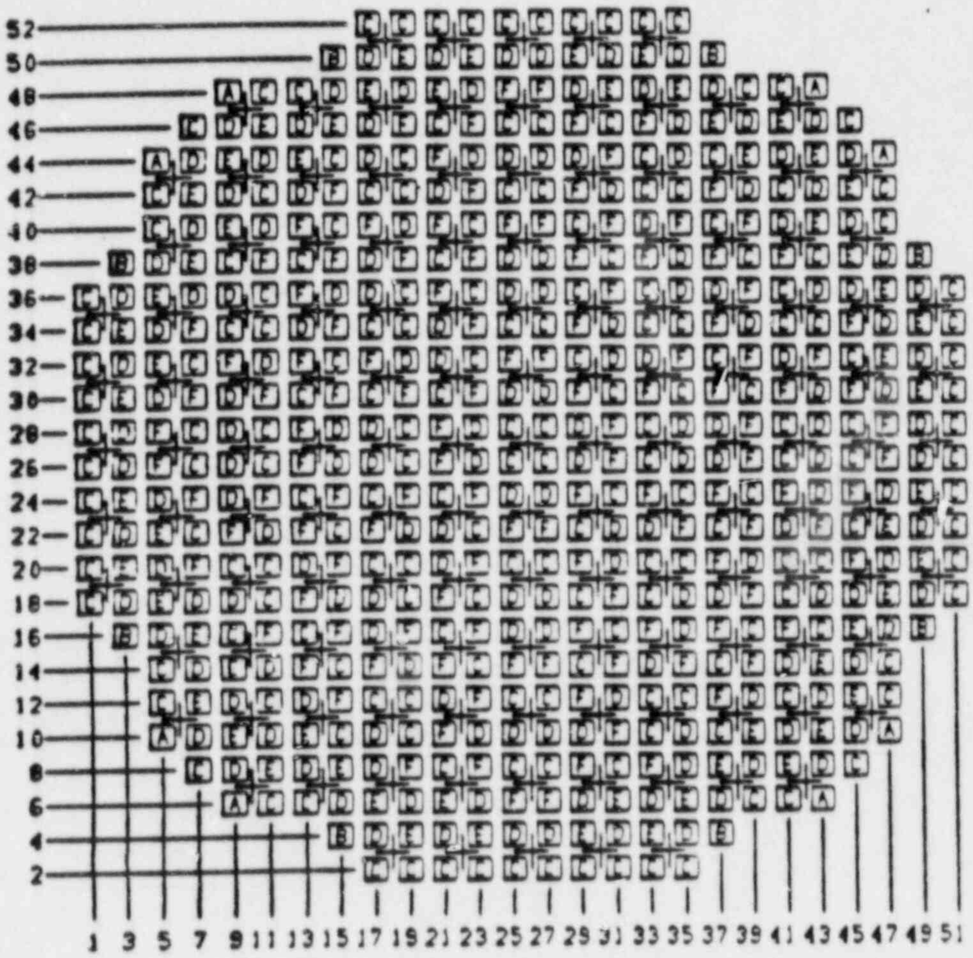
GE SIL380 recommendations have been included in the Brunswick Steam Electric Plant Unit 1 operating procedures and/or Technical Specifications and, therefore, the stability analysis is not required. NRC approval for deletion of a cycle-specific stability analysis is documented in Amendment 8 to NEDE-24011-P-A-8-US.

17. LOSS-OF-COOLANT ACCIDENT RESULTS (S.2.5.2)

LOCA Method Used: SAFE/REFLOOD/CHASTE

"Loss-of-Coolant Accident Analysis Report for Brunswick Steam Electric Plant Unit No. 1," General Electric Company (NEDO-24165, December 1978, as amended and NEDE-24165-P, April 1988.)

* ΔCPR penalty of 0.02 for the tilted misoriented bundle is applied to the cycle MCPR value reported in Section 12.



| FUEL TYPE | |
|---|---|
| A = P8DRB284H | D = BP8DRB299 (Cycle 6) |
| B = P8DRB299 | E = BD339A |
| C = BP8DRB299 (Cycle 5) | F = BD323B |

Figure 1. Reference Core Loading Pattern

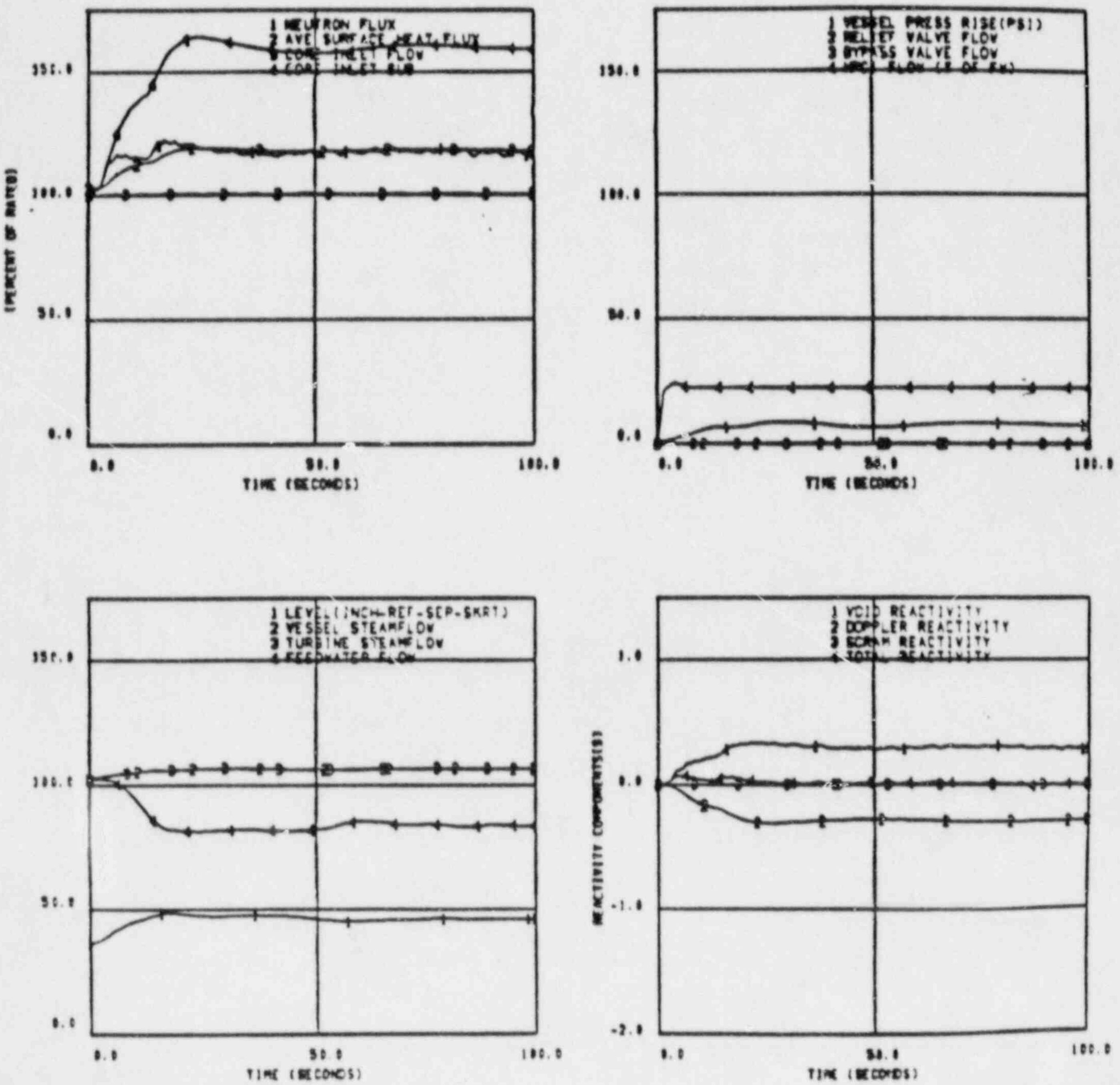



Figure 2. Plant Response to Inadvertant Activation of HPCI

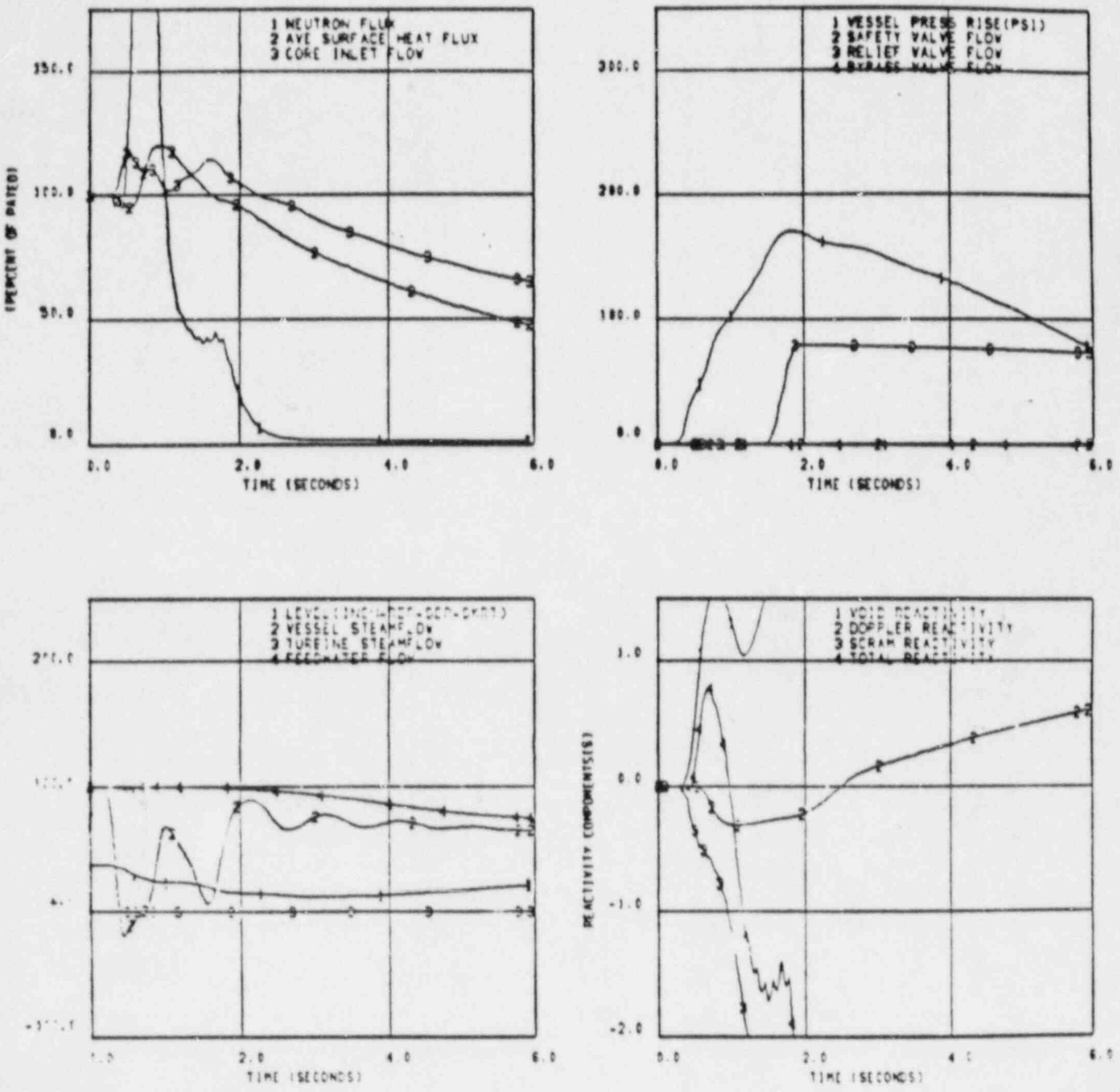



Figure 3. Plant Response to Generator Load Rejection Without Bypass (EOC7-2000 MWd/ST)

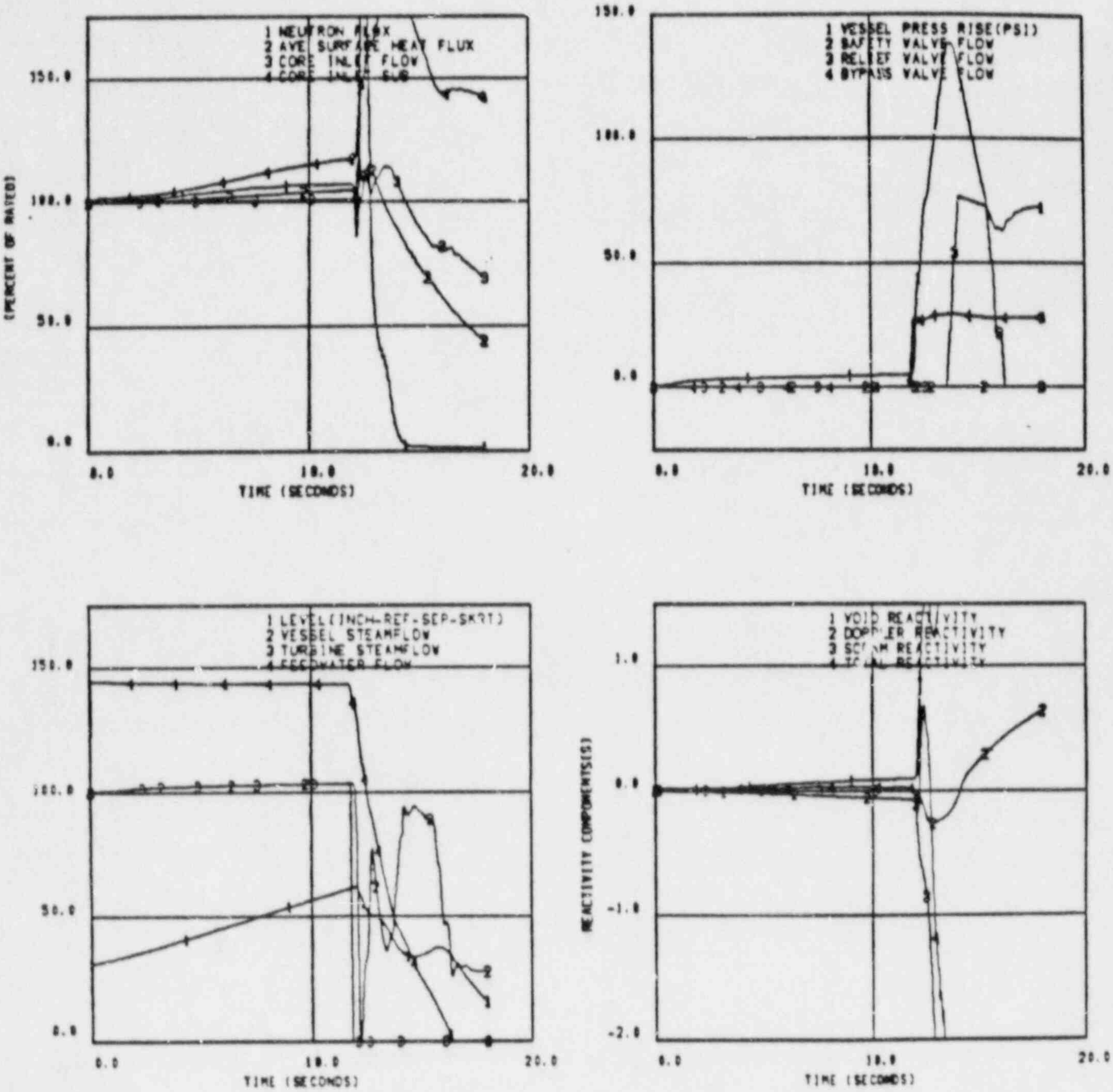



Figure 4. Plant Response to Feedwater Controller Failure (EOC7-2000 MWd/ST)

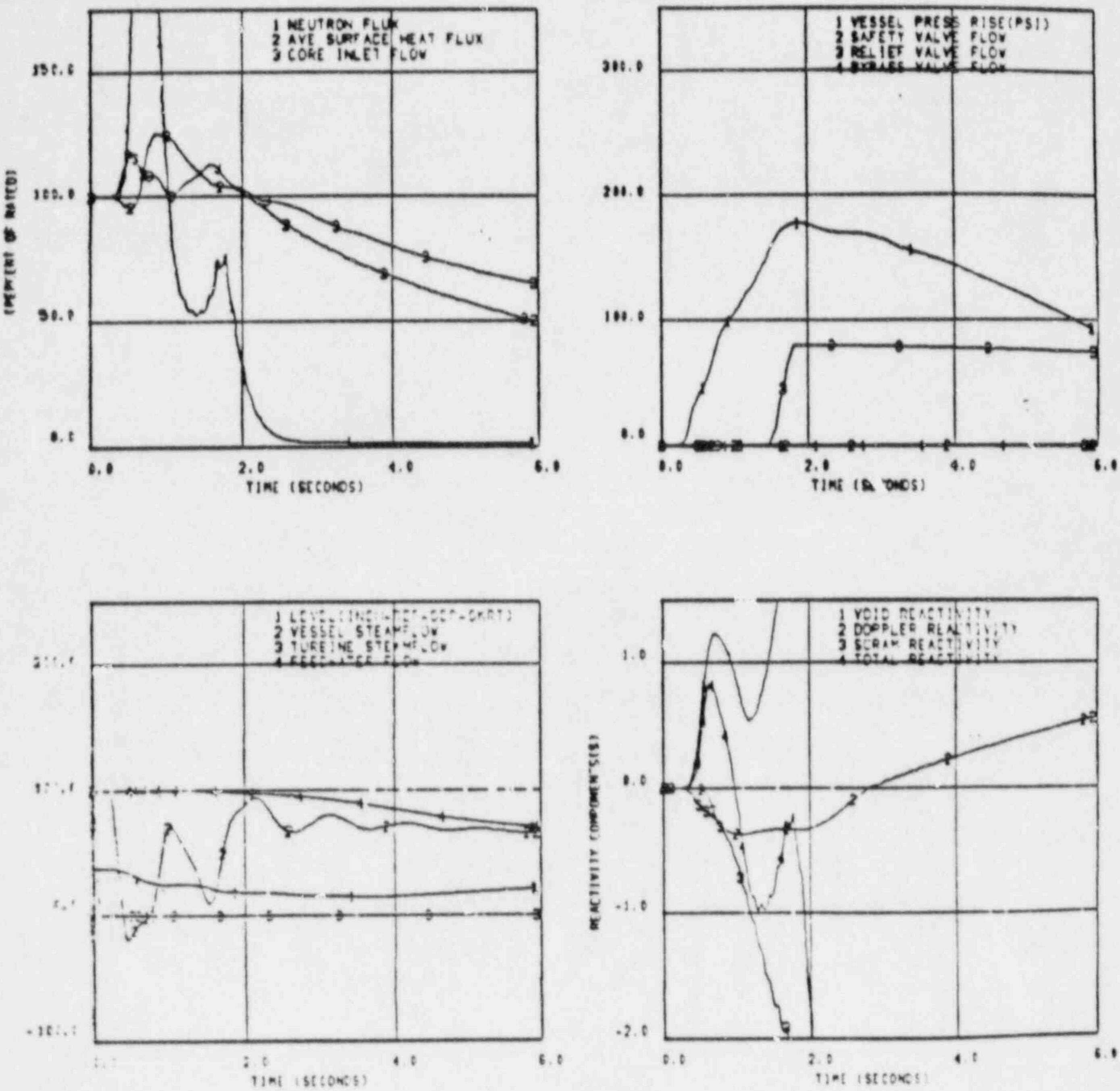



Figure 5. Plant Response to Generator Load Rejection Without Bypass (EOC7)

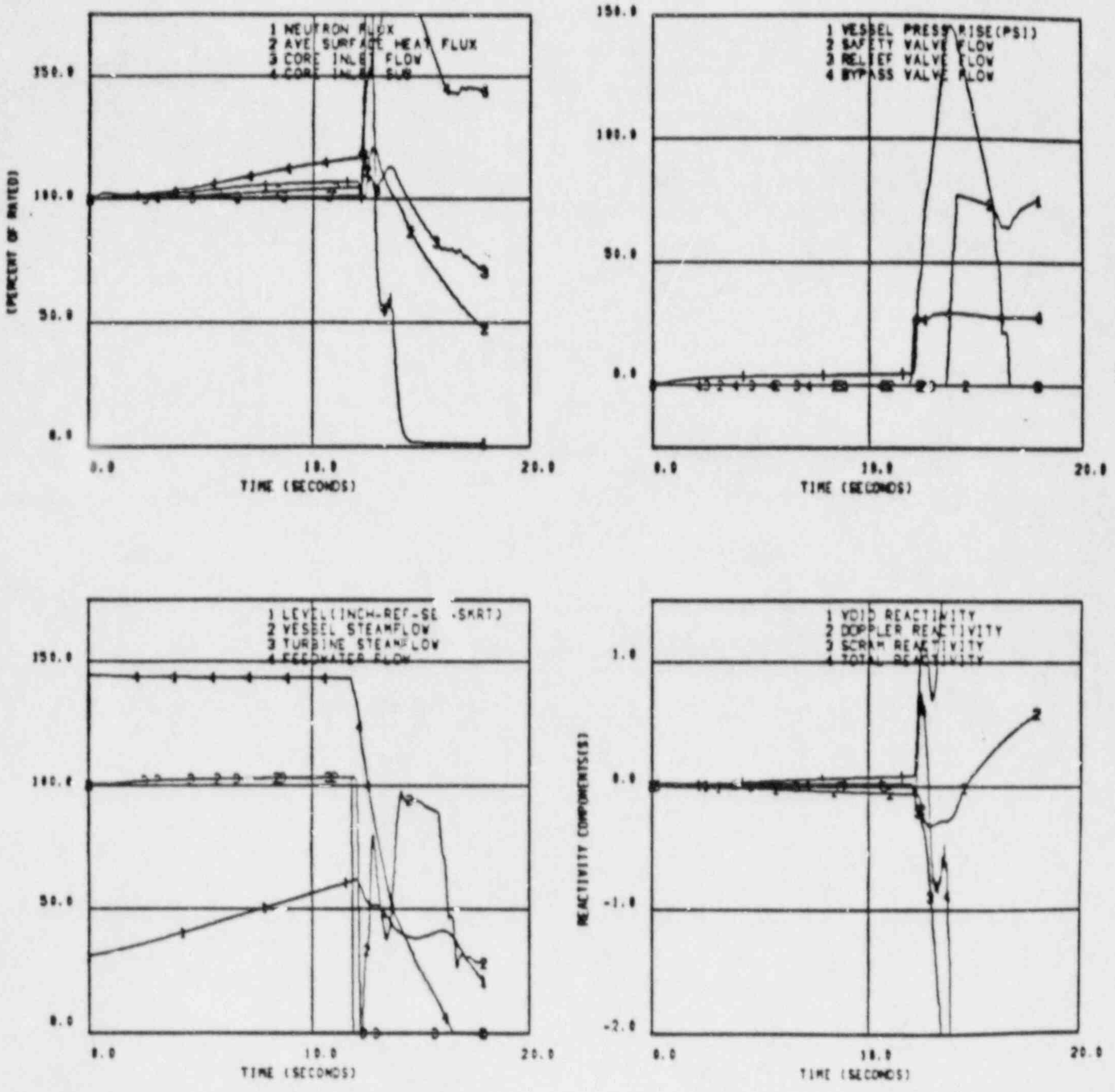



Figure 6. Plant Response to Feedwater Controller Failure (EOC7)

| | 2 | 6 | 10 | 14 | 18 | 22 | 26 | 30 | 34 | 38 | 42 | 46 | 50 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 51 | | | | | | 36 | | 36 | | | | | |
| 47 | | | 5 | | 6 | | 6 | | 6 | | 6 | | |
| 43 | | 36 | | 36 | | 36 | | 36 | | 36 | | 36 | |
| 39 | | | 6 | | 6 | | 14 | | 6 | | 6 | | |
| 35 | | 36 | | 36 | | 36 | | 36 | | 36 | | 36 | |
| 31 | 6 | | 6 | | 14 | | 0 | | 14 | | 6 | | 6 |
| 27 | | 36 | | 36 | | 44 | | 44 | | 36 | | 36 | |
| 23 | 6 | | 6 | | 14 | | 0 | | 14 | | 6 | | 6 |
| 19 | | 36 | | 36 | | 36 | | 36 | | 36 | | 36 | |
| 15 | | | 6 | | 6 | | 14 | | 6 | | 6 | | |
| 11 | | 36 | | 36 | | 36 | | 36 | | 36 | | 36 | |
| 7 | | | 6 | | 6 | | 6 | | 6 | | 6 | | |
| 3 | | | | | | 36 | | 36 | | | | | |

NOTES:

1. No. indicates number of notches withdrawn out of 48. Blank is a withdrawn rod.

2. Error rod is (26,31).

Figure 7. Limiting Rod Pattern

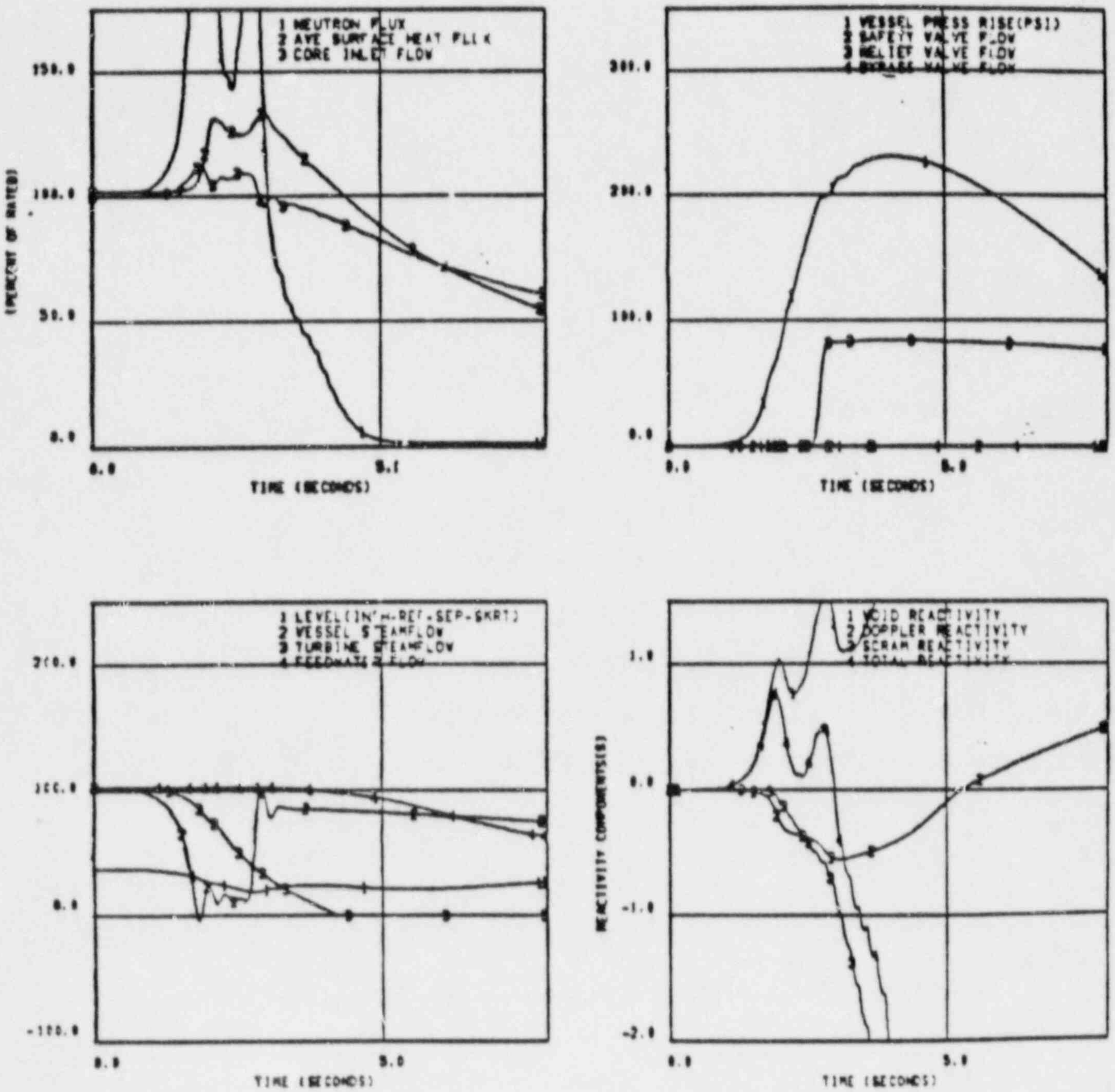



Figure 8. Plant Response to MSIV Closure (Flux Scram)

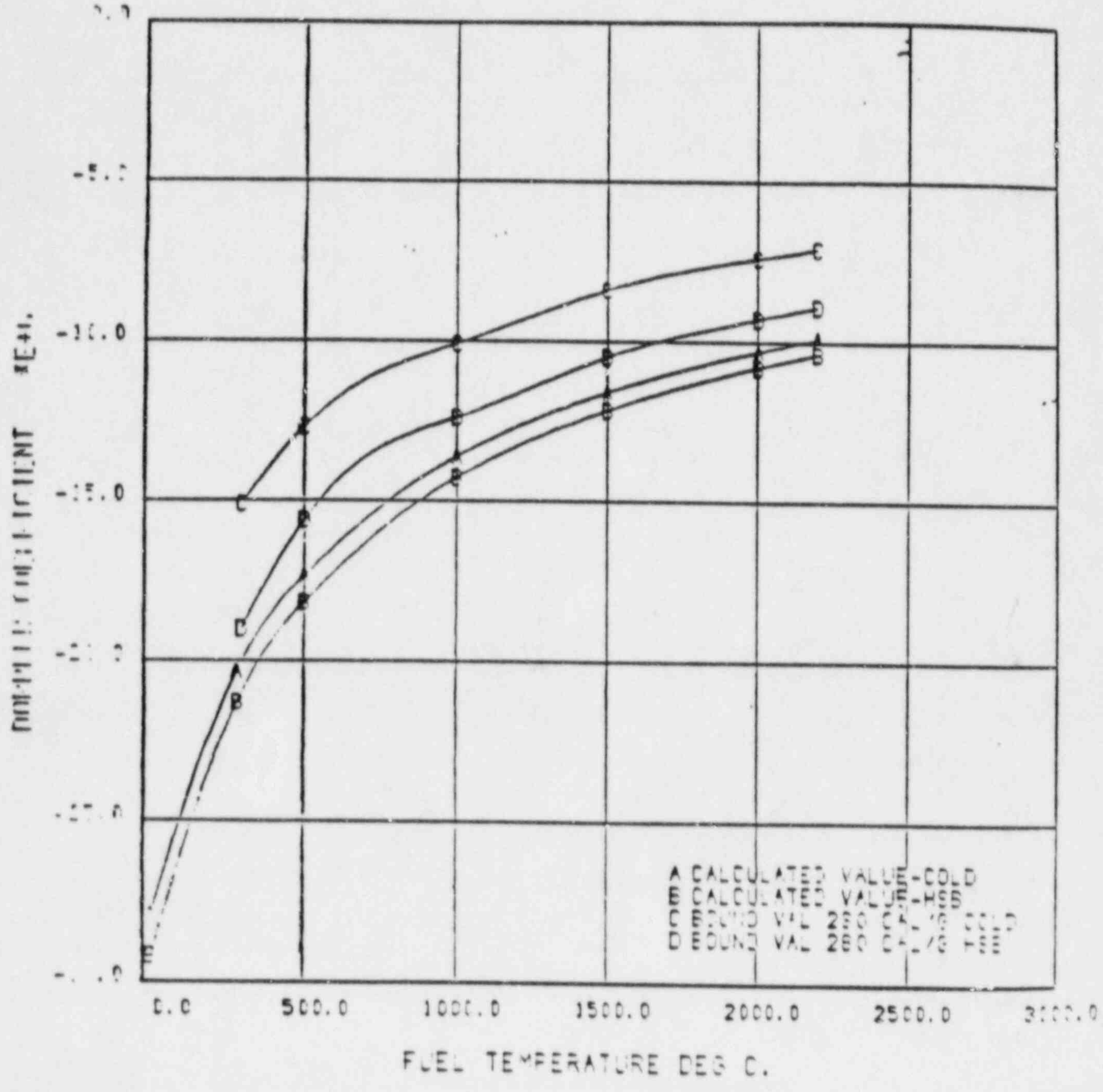



Figure 5 Fuel Doppler Coefficient in 1/Δ °C

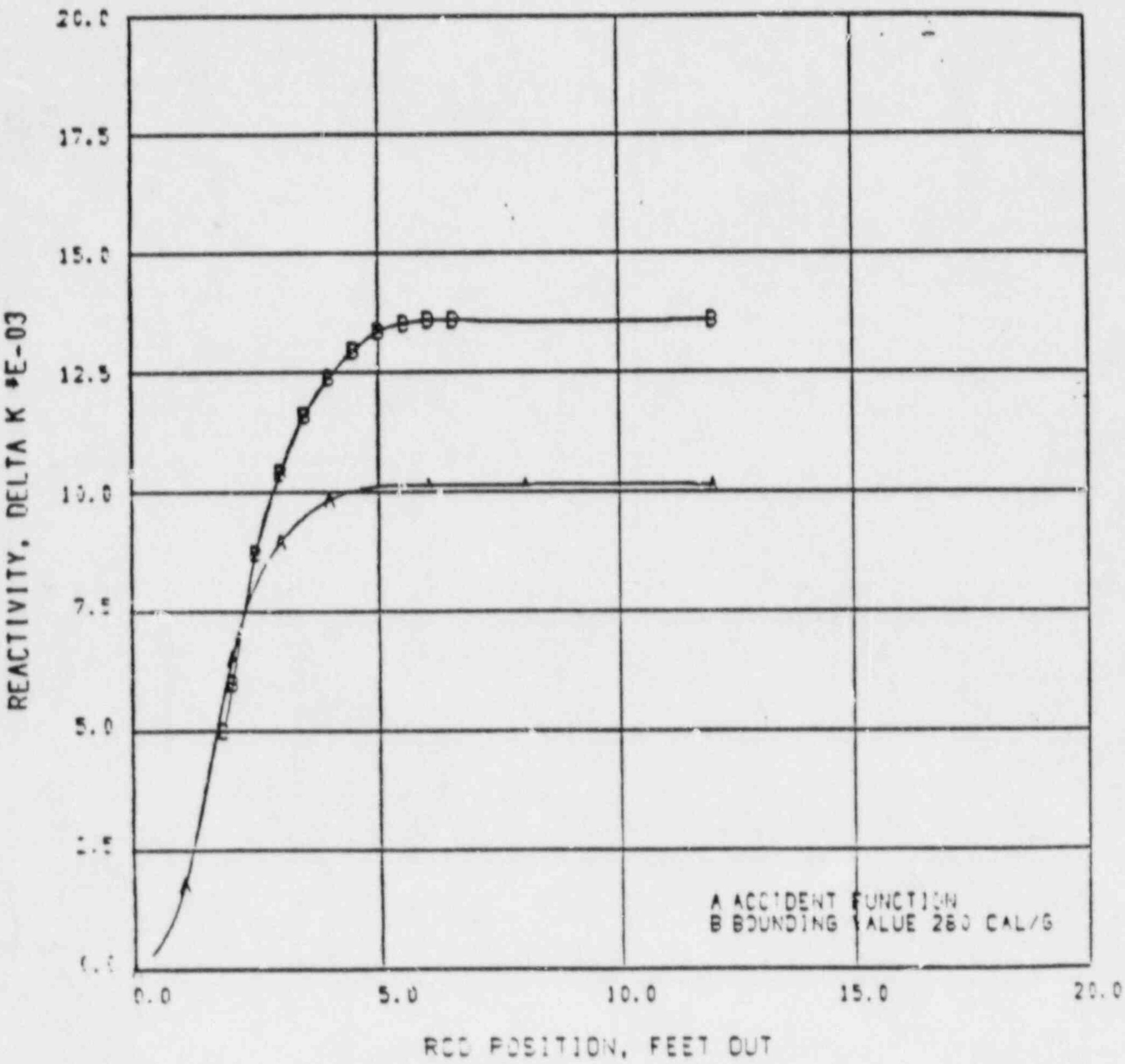



Figure 10. Accident Reactivity Shape Function, Cold Startup

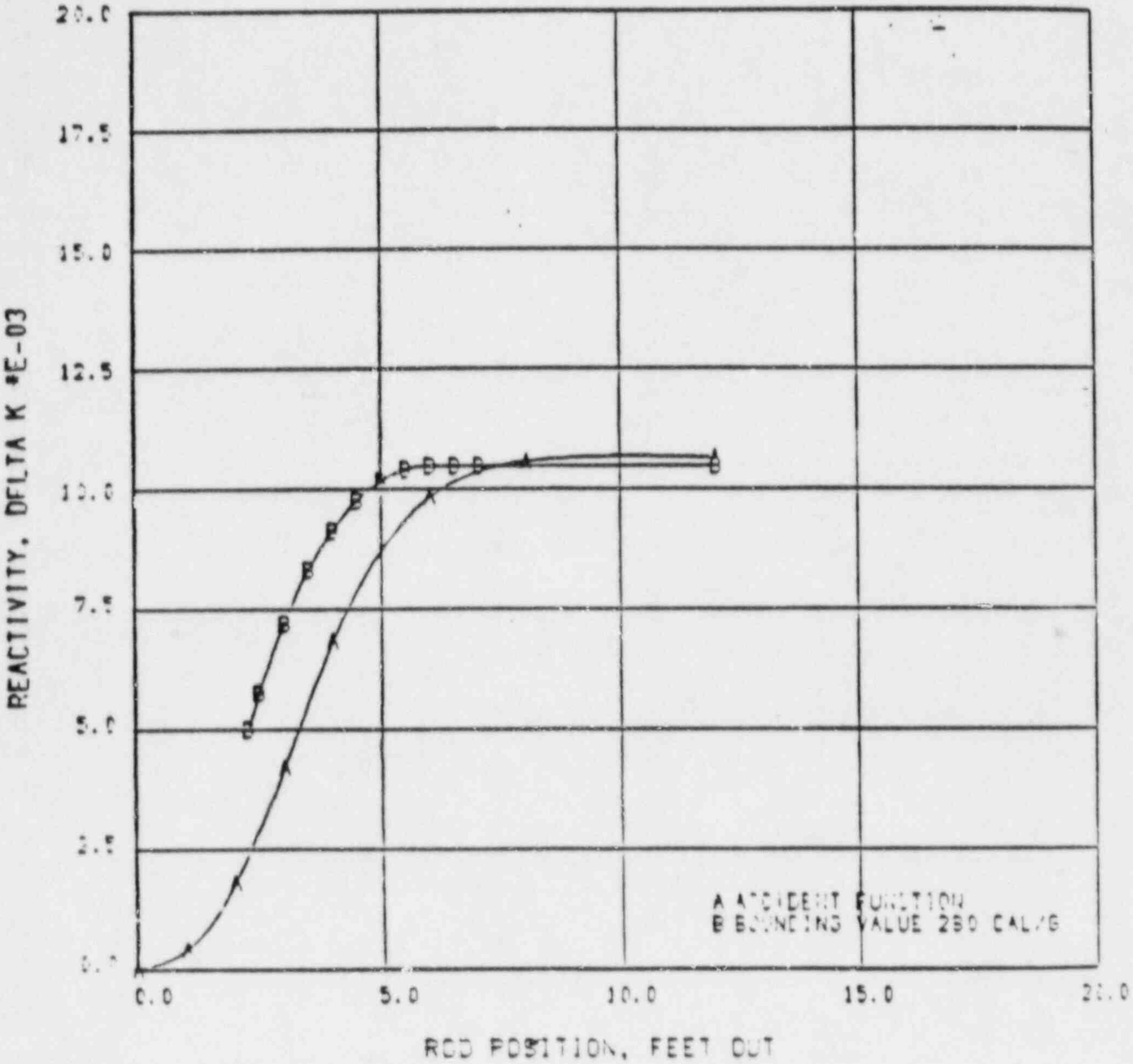



Figure 11. Accident Reactivity Shape Function, Hot Standby

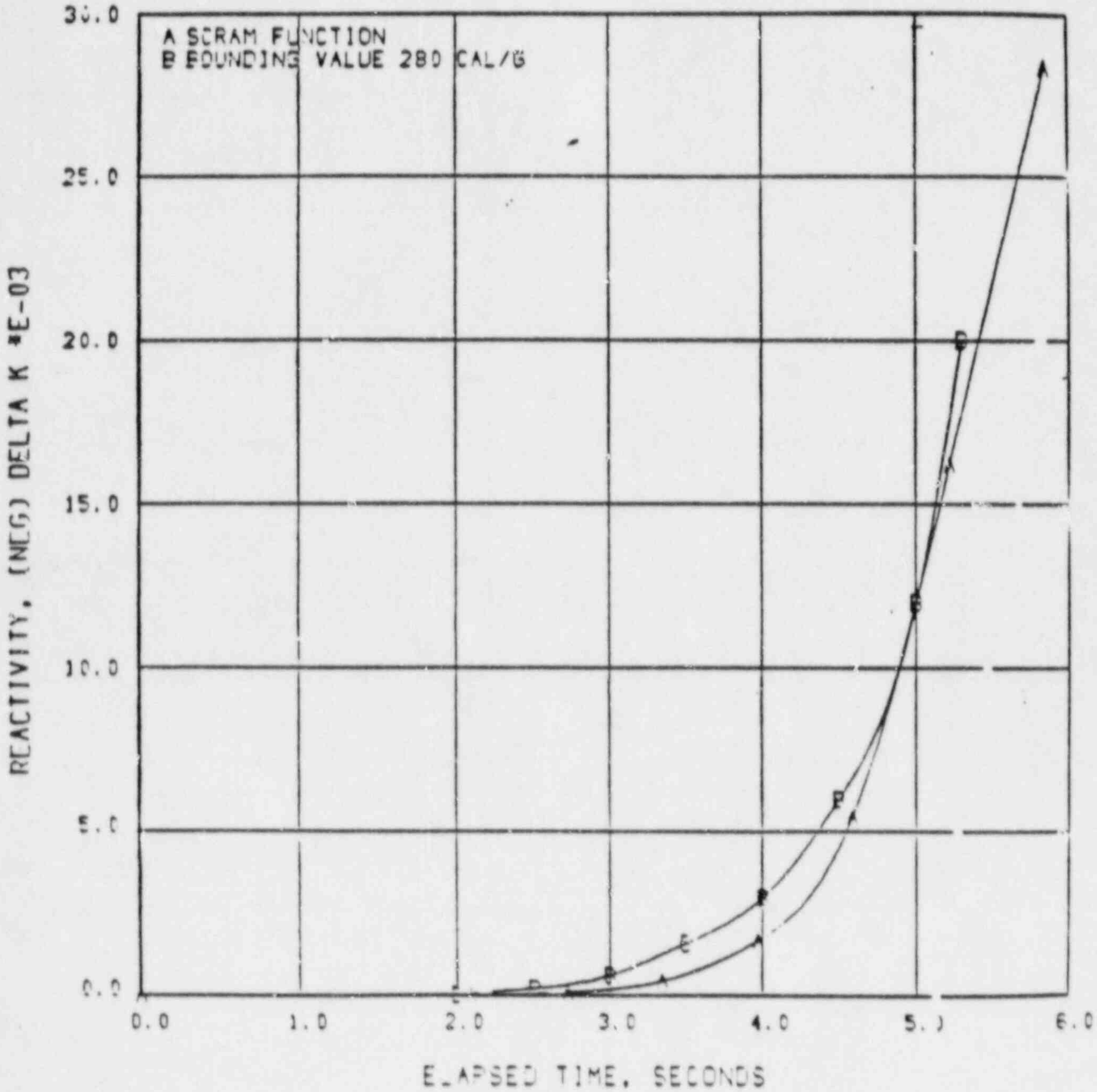



Figure 12. Scram Reactivity Function, Cold Startup

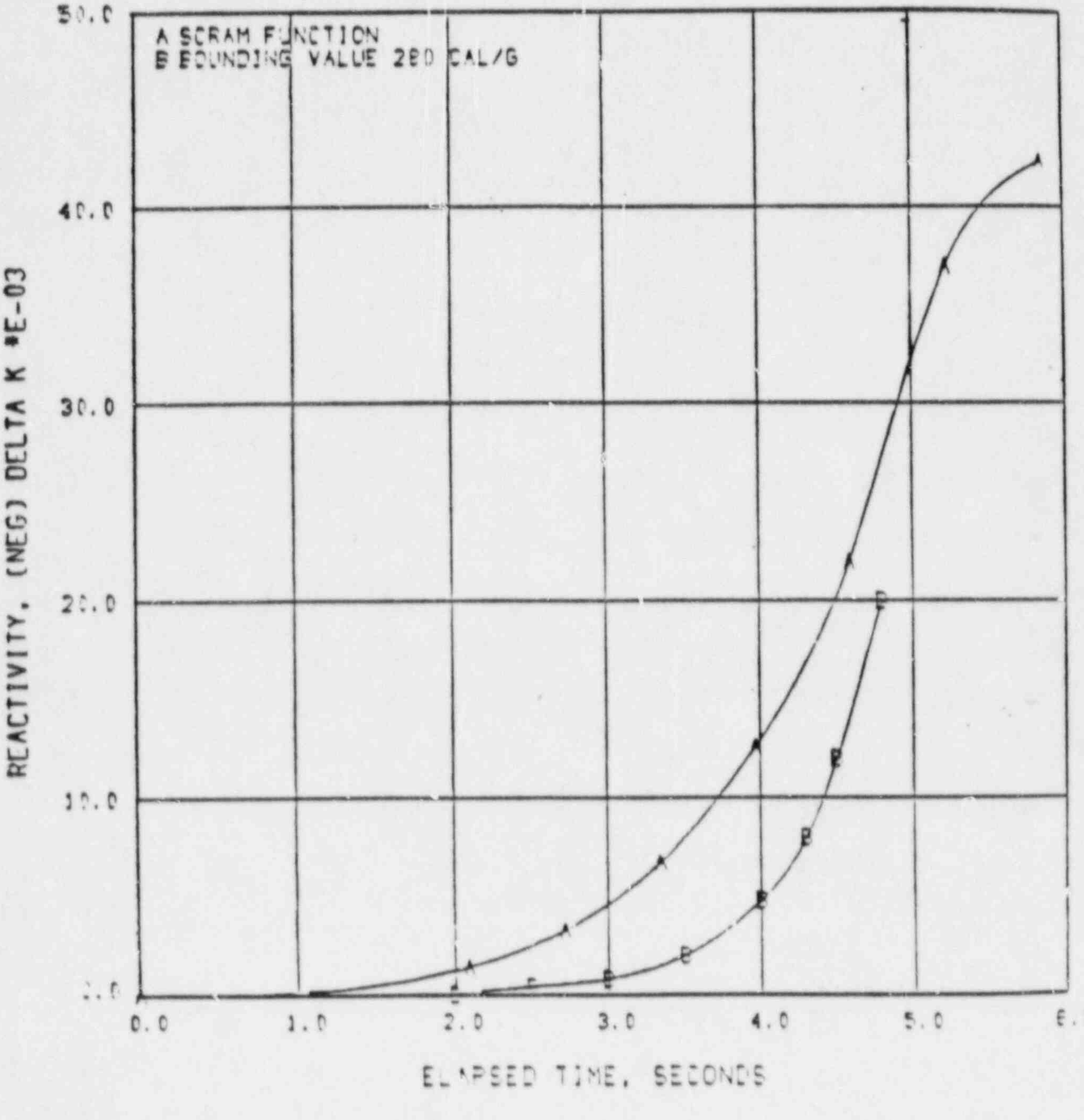



Figure 13. Scram Reactivity Function, Hot Standby

# APPENDIX A

## LIMITING CONDITIONS FOR OPERATION

This appendix provides the limiting condition for operation (LCO) for each of the power distribution limits identified below:

(1) Average Planar Linear Heat Generation Rate (APLHGR)
(2) Operating Limit MCPR
(3) APRM Setpoints

Surveillance requirements and required actions are specified in the Technical Specifications. The power distribution limit bases are given in Appendix B.

### A.1 APLHGR

During steady-state power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall not exceed limits based on applicable APLHGR limit values which have been approved for the respective fuel and lattice types determined by the approved methodology described in GESTAR-II (NEDE-24011-P-A). When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) shown in Figures A-1 through A-5, during two recirculation loop operation.

### A.2 OPERATING LIMIT MCPR

The applicable fuel cladding integrity safety limit MCPR for this cycle is 1.04. This safety limit MCPR applies to Unit 1 during this cycle because it is a D-Lattice BWR with at least two successive reloads of P8X8R, BP8X8R, GE8X8E, or GE8X8EB fuel types having high bundle R-Factors ($\geq$1.04), one of which is the fuel in its first cycle of operation. The use of this value has been approved in Amendment 14 of NEDE-24011-P-A-8. During steady-state power operation, the MCPR for each type of fuel shall not be less than the limiting value (shown in Table A-1) times the $K_f$ (shown in Figure A-6), for two recirculation loop operation.

In reference to Technical Specification 3.2.3.2, the OLMCPR for $\tau_{ave}$ less than or equal to $\tau_B$, is the greater of the non-pressurization transient or the Option B OLMCPR (Table A-1), where $\tau_{ave}$ and $\tau_B$ are given by:

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

where:

$i$ = surveillance test number.

$n$ = Number of surveillance tests performed to date in the cycle (including BOC).

$N_i$ = Number of rods tested in the $i^{th}$ surveillance test.

$\tau_i$ = Average scram time to notch 36 for surveillance test i.

and

$$\tau_B = \mu + 1.65 \left( \frac{N_1}{\sum_{i=1}^{n} N_i} \right)^{1/2} (\sigma)$$

where:

$N_1$ = Number of rods tested at BOC.

$\mu$ = 0.813 seconds (mean value for statistical scram time distribution from de-energization of scram pilot valve solenoid to pickup on notch 36).

$\sigma$ = 0.018 seconds (standard deviation of the above statistical distribution).

In reference to Technical Specification 3.2.3.2, the OLMCPR for $\tau_{ave}$ greater that $\tau_B$ shall be either:

a. The greater of the non-pressurization transient (Table A-1) or the adjusted pressurization transient MCPR ($MCPR_{adj}$) where:

$$MCPR_{adj} = MCPR_{Option\ B} + \frac{\tau_{ave} - \tau_B}{\tau_A - \tau_B} (MCPR_{Option\ A} - MCPR_{Option\ B})$$

$\tau_A$ = 1.05 seconds (control rod average scram insertion time limit to notch 36).

and

$MCPR_{Option\ A}$ as given in Table A-1

$MCPR_{Option\ B}$ as given in Table A-1

or,

b. $MCPR_{Option\ A}$ as given in Table A-1.

## A.3 APRM SETPOINTS

The flow-biased APRM scram trip setpoint (S) and rod block trip setpoint ($S_{RB}$) shall be:

$S \leq (0.66W + 54\%)\ T$, and
$S_{RB} \leq (0.66W + 42\%)\ T$;

where S and $S_{RB}$ are in percent of rated thermal power;
W = loop recirculation flow in percent of rated flow.

T is the ratio of Fraction of Rated Thermal Power (FRTP) divided by Core Maximum Average Planar Linear Heat Generation Rate Ratio (CMAPRAT):

$T = \frac{FRTP}{CMAPRAT}$ where $T \leq 1$.

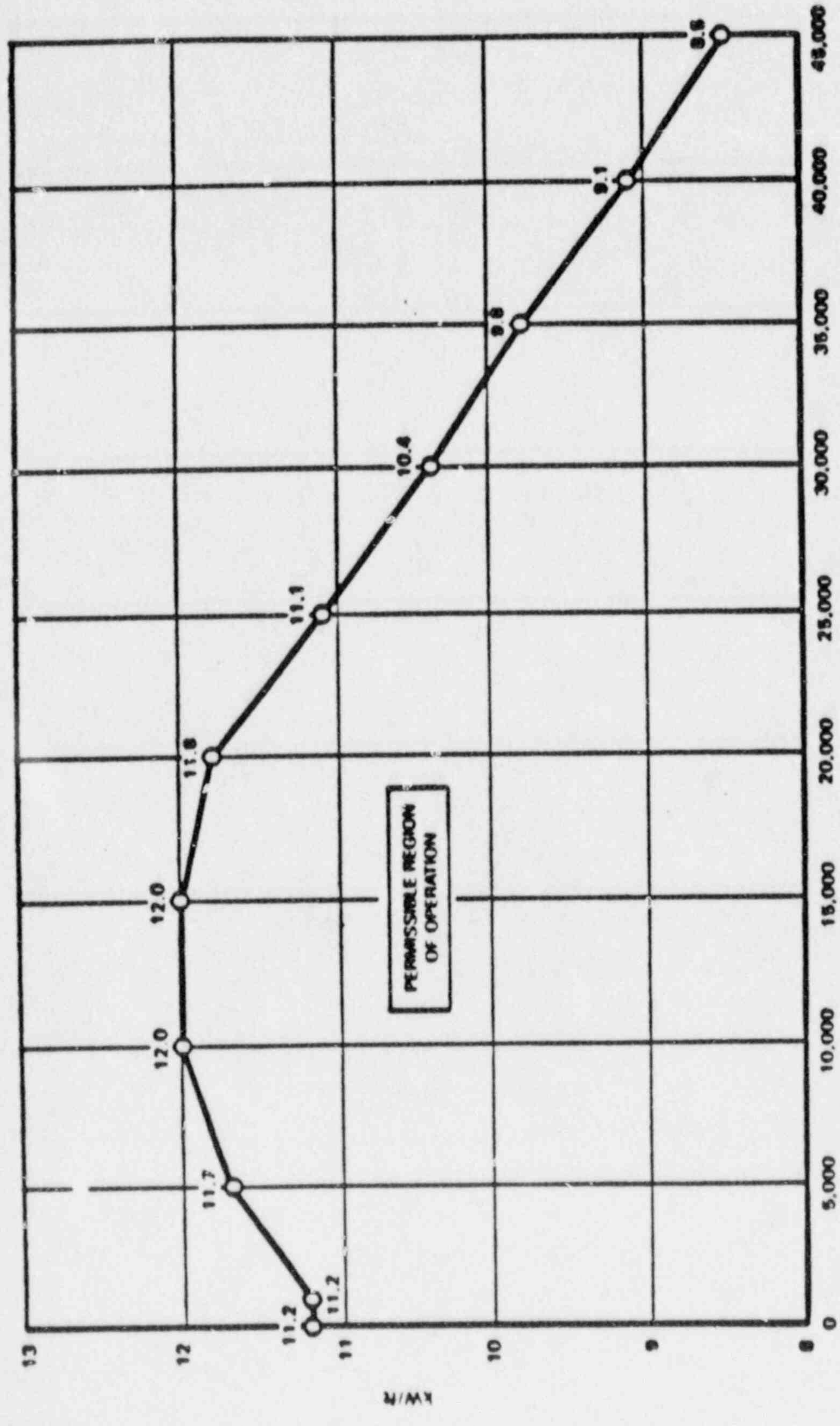



Figure A-1. Fuel Type P8DRB284H (P8x8R) Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Average Planar Exposure

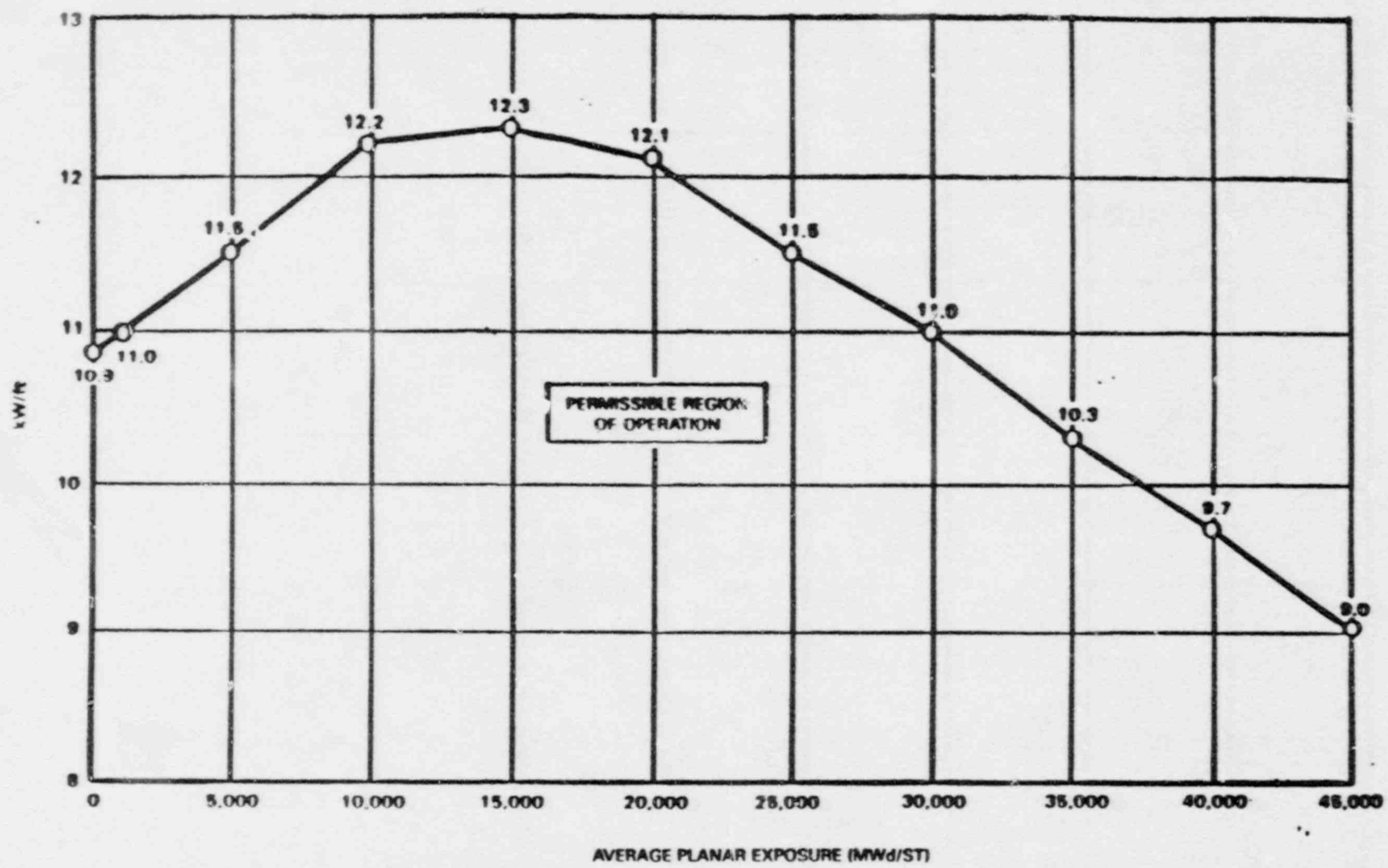



Figure A-2. Fuel Type P8DRB299 (P8x8R) Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Average Planar Exposure

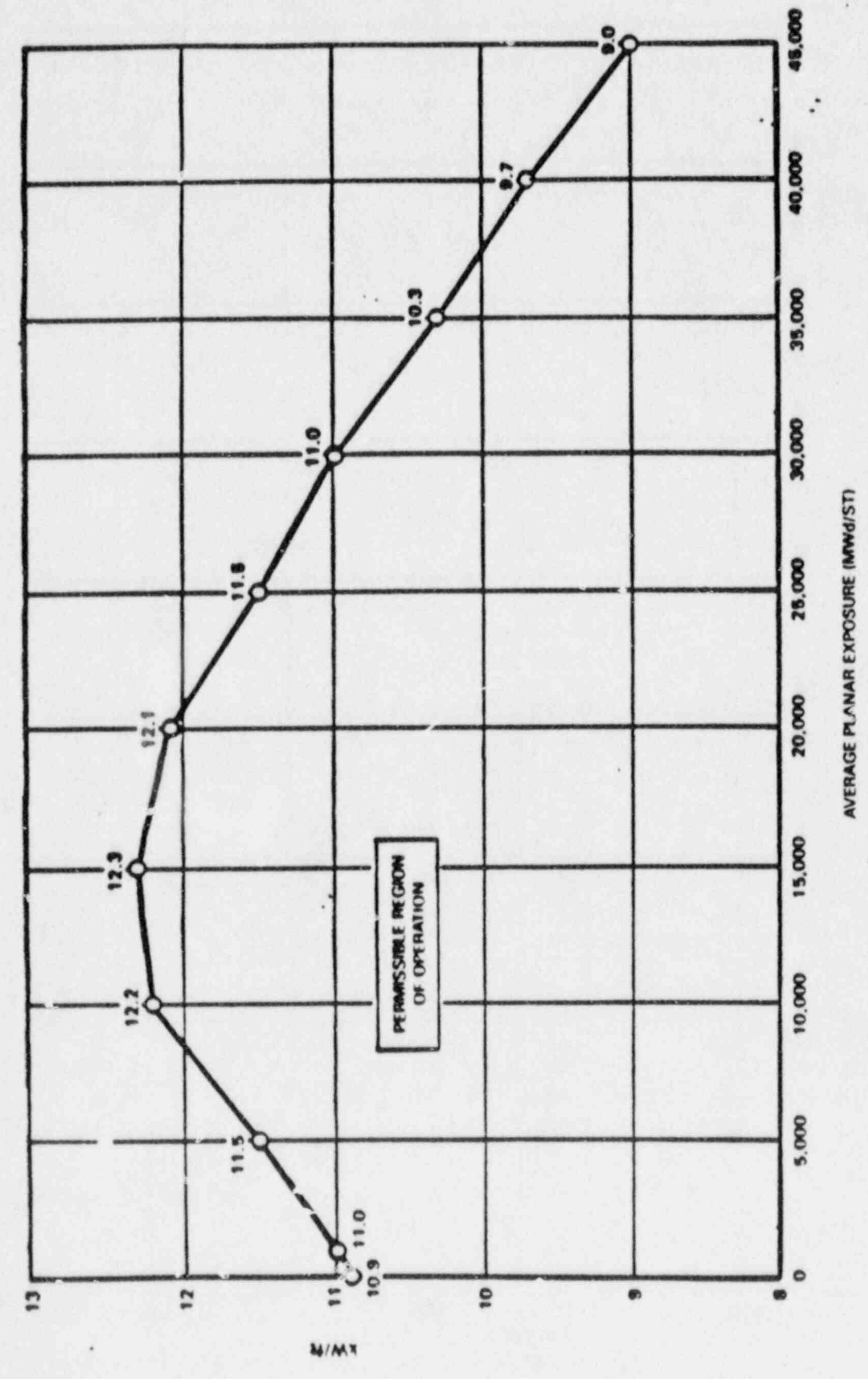



Figure A-3. Fuel Type BP8DRB299 (BP8x8R) Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Average Planar Exposure

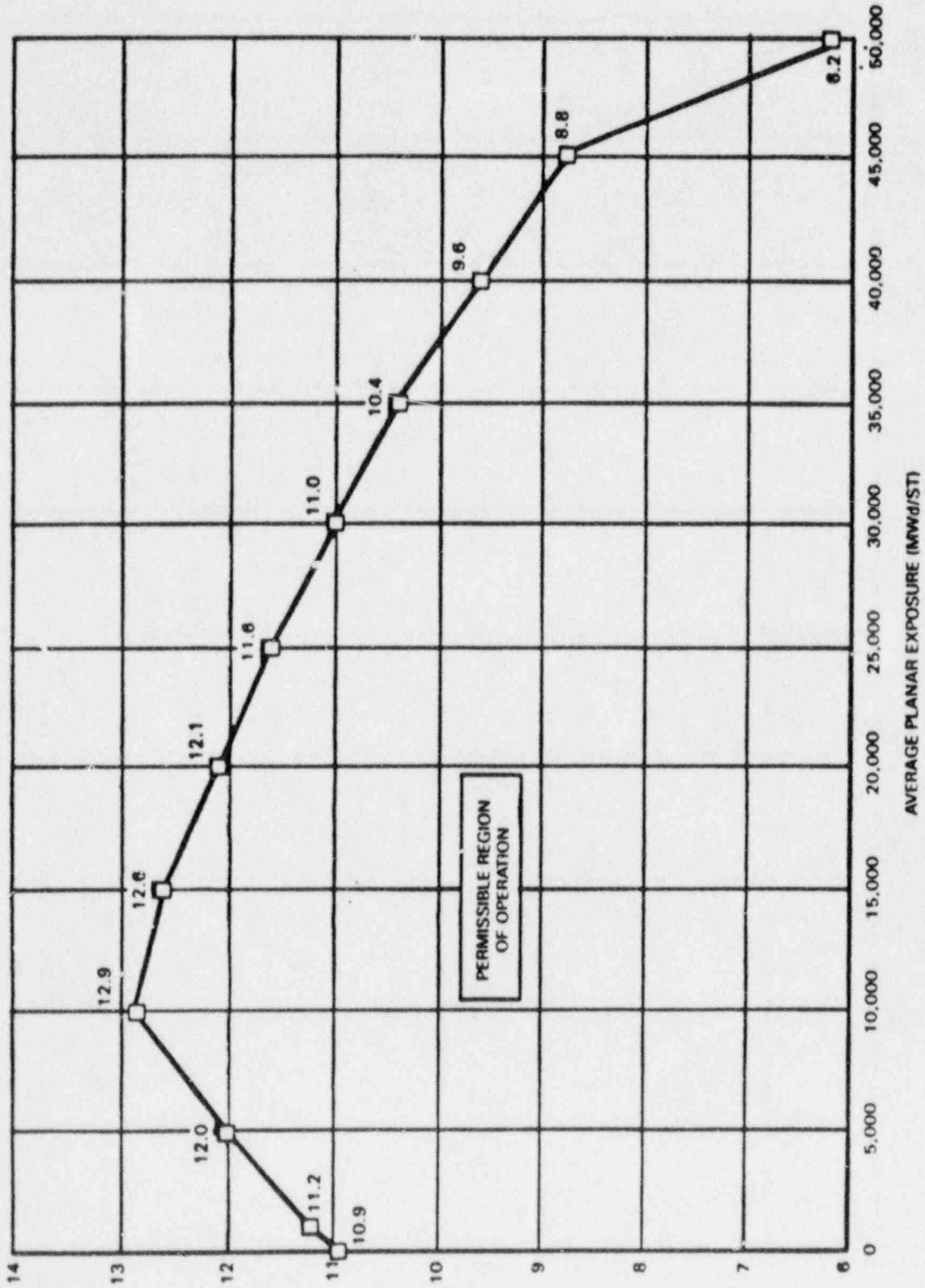



Figure A-5. Fuel Type BD339A (GE8x8EB) Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Average Planar Exposure

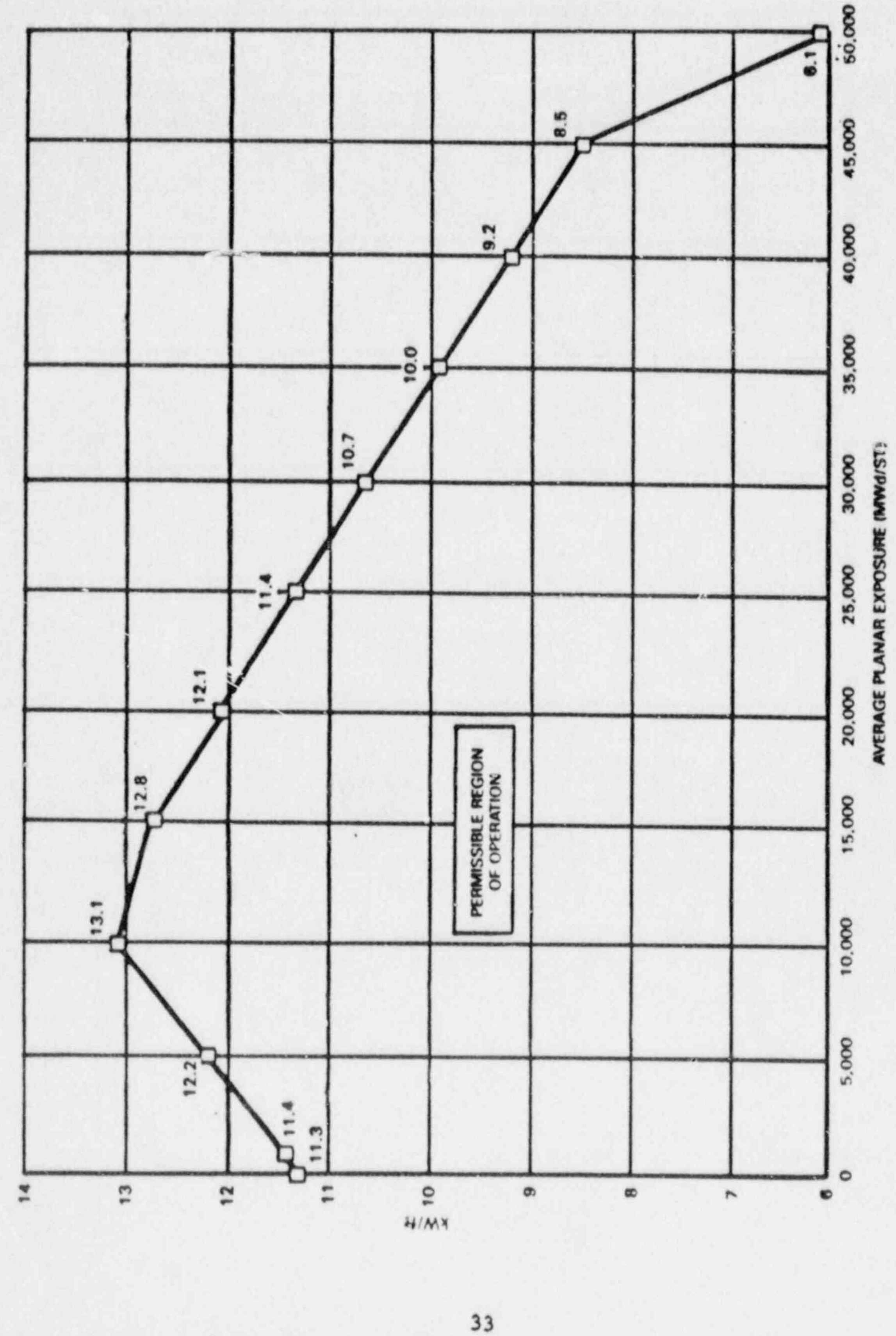



Figure A-4. Fuel Type BD323B (GE8x8EB) Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Average Planar Exposure

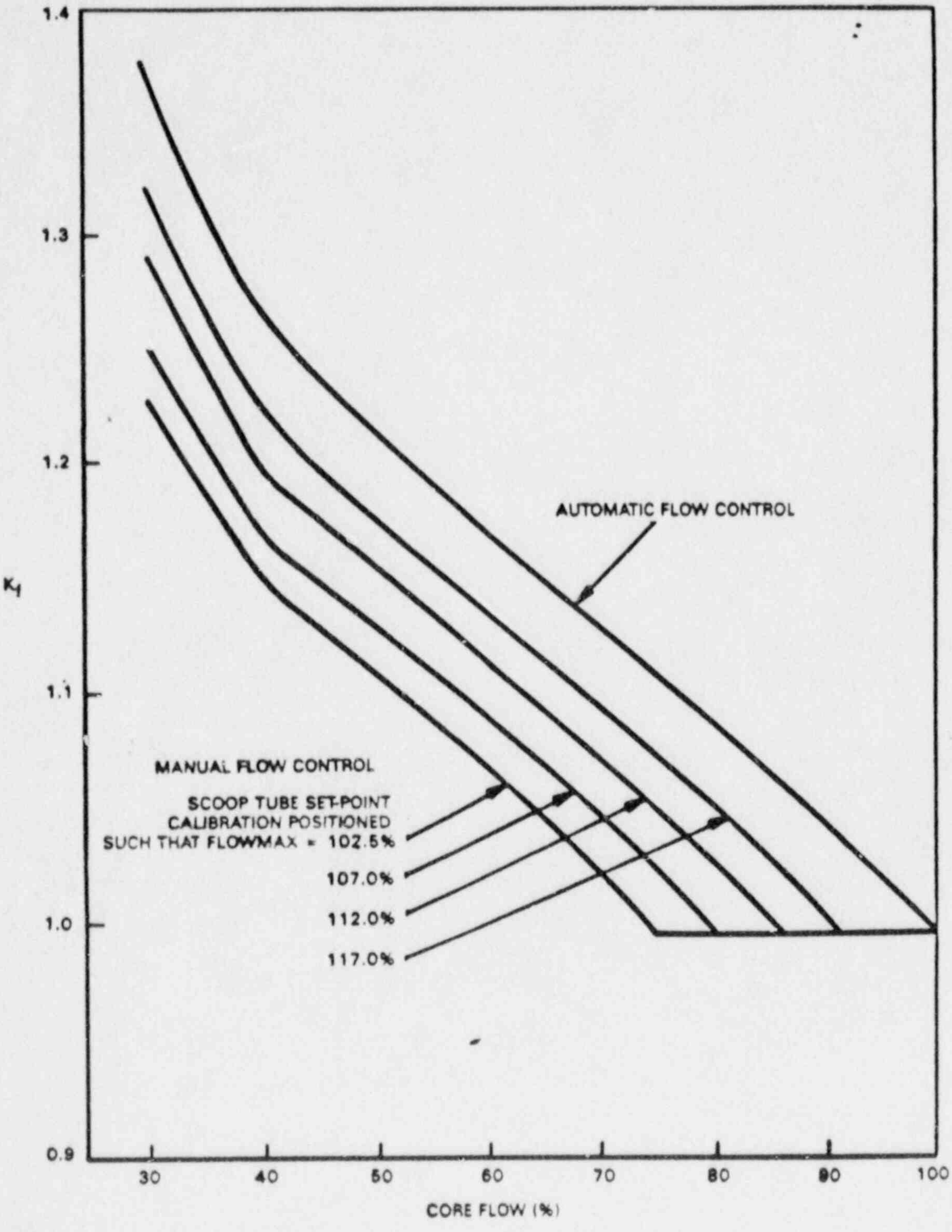



Figure A-6. $K_f$ Factor for GEXL-PLUS

Table A-1

MCPRs

Fuel Type: P8X8R, BP8X8R, and GE8X8EB

Non-Pressurized Transient MCPR = 1.25

Pressurization Transients

| Exposure Range | $MCPR_{Option\ A}$ | $MCPR_{Option\ B}$ |
|---|---|---|
| BOC7 to EOC7-2000 MWd/ST | 1.32 | 1.25 |
| EOC7-2000 MWd/ST to EOC7 | 1.34 | 1.30 |

# APPENDIX B

## BASES FOR LIMITING CONDITIONS FOR OPERATION

This appendix provides the bases for each of the power distribution limits identified in Appendix A.

### B.1 APLHGR

This specification assures that the peak cladding temperature (PCT) following the postulated design basis loss-of-coolant accident (LOCA) will not exceed the limits specified in 10CFR50.46 and that the fuel mechanical design analysis limits specified in Reference B-1 will not be exceeded.

Thermal Mechanical Design Analysis: NRC approved methods (specified in Reference B-1) are used to demonstrate that all fuel rods in a lattice operating at the bounding power history meet the fuel design limits specified in Reference B-1. No single fuel rod follows, or is capable of following, this bounding power history. This bounding power history is used as the basis for the fuel design analysis APLHGR limit.

LOCA Analysis: A LOCA analysis is performed in accordance with 10CFR50, Appendix K to demonstrate that the permissible planar power (maximum APLHGR) limits comply with the ECCS limits specified in 10CFR50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant. The methods used are discussed in Reference B-2.

The APLHGR limit is the most limiting composite of the fuel design analysis APLHGR limit and the ECCS APLHGR limit.

### B.2 OPERATING LIMIT MCPR

The required operating limit MCPRs at steady-state operating conditions as specified in Appendix A are derived from the established fuel cladding integrity safety limit MCPR specified in Appendix A and an analysis of abnormal operational transients. In the analysis of these abnormal operational transients, the GEXL-PLUS thermal correlation has been used, where applicable, to determine the appropriate initial conditions. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady-state operating limit, it is required that the resulting MCPR does not decrease below the safety limit MCPR at any time during the transient, assuming instrument trip setting as given in Specification 2.2.1 of the Technical Specifications.

To assure that the fuel cladding integrity safety limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which ones result in the largest reduction in Critical Power Ratio (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

The codes used to perform the transient analyses that serve as the basis for the operating limit MCPR are described in Reference B-1. Conditions at limiting exposures are used for nuclear data to provide conservatism relative to core exposure aspects. Plant-unique initial conditions and system parameters are used as inputs to the transient codes. The ΔCPR calculated by the transient codes is adjusted using NRC approved adjustment factors to account for code uncertainties and to provide a 95/95 licensing basis.

The limiting transient yields the largest ΔCPR. The ΔCPR for the limiting transient is added to the fuel cladding integrity safety limit to establish the minimum operating limit MCPR.

The purpose of the $K_f$ factor is to define operating limits at other than rated flow conditions. At less than 100% flow, the required MCPR is the product of the operating limit MCPR and the $K_f$ factor. Specifically, the $K_f$ factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speedup caused by a motor generator speed controller failure.

For operation in the automatic flow control mode, the $K_f$ factors assure that the operating limit MCPR in Appendix A will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the $K_f$ factors assure that the safety limit MCPR will not be violated should the most limiting transient occur at less than rated flow.

The $K_f$ factor values are generically developed as described in References B-3 and B-4.

The $K_f$ factors are conservative for the General Electric plant operation because the operating limit MCPRs in Appendix A are greater than the original 1.20 operating limit MCPR used for the generic derivation of $K_f$.

At core thermal power levels less than or equal to 25%, 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 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. During initial startup testing of the plant, a MCPR evaluation was made at 25% initial power level with minimum recirculation pump speed. The demonstrated MCPR margin was such, that future MCPR evaluations below this power level are unnecessary. The daily requirement for calculating MCPR above 25% 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 the MCPR will be known following a change in power or power shape, regardless of magnitude that could place operation at a thermal limit.

## B.3 APRM SETPOINTS

The flow-biased thermal power upscale scram setting and flow-biased neutron flux upscale control rod block functions of the APRM instruments are adjusted to ensure that fuel design and safety limits are not exceeded. The scram setting and rod block setting are adjusted in accordance with the formula in Appendix A when the combination of Thermal Power and CMAPRAT indicate a highly peaked power distribution. This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced APRM high flux scram curve by the reciprocal of the APRM gain change.

## B.4 REFERENCES


1. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A-8.

2. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K", NEDO-20566, January 1970.

3. Letter, J. S. Charnley (GE) to M. W. Hodges (NRC), "Application of GESTAR-II Amendment 15," March 22, 1988, MFN-027-88.

4. Letter, A. C. Thadani (NRC) to J. S. Charnley (GE), "Acceptance for Referencing of Application of Amendment 15 to General Electric Licensing Topical Report NEDE-24011-P-A, 'General Electric Standard Application for Reactor Fuel' (TAC No. 60903)," May 5, 1988.

# APPENDIX C

## PLANT PARAMETER DIFFERENCES

GETAB and Transient Analysis Initial Conditions

The values used in the GETAB and Transient Analysis which differ from the values reported in Tables S.2-4.1 and S.2-6 in NEDE-24011-P-A-8-US are given in Table C-1.

Table C-1

PLANT PARAMETER DIFFERNCES

| Parameter | Analysis Value | NEDE-24011-P-A-8-US Value |
|---|---|---|
| Rated Steam Flow[1] | 10.47E+06 | 10.96E+06 ± 0.2% |
| Dome Pressure[1] | 1005 | 1020 ± 2 psi |
| Turbine Pressure[1] | 950 | 960 ± 2 psi |
| Non-Fuel Power Fraction[1] | 0.039 | 0.040 |
| Number of S/RVs[2] | 10 | 11 |

[1] The indicated changes are a result of the application of the pre-approved methods outlined in Amendment 11 to NEDE-24011-P-A-8.

[2] The indicated change is a result of the simulation of a valve-out-of-service condition.

## APPENDIX D

## USE OF GEXL-PLUS METHODS FOR CYCLE 7

The analyses required for this cycle were performed with the GEXL-PLUS thermal correlation. In analyses prior to Cycle 7 (Reload 6), the GEXL thermal correlation was used. The incorporation of GEXL-PLUS into the fuel cycle analysis process is provided for in Amendment 15 to GESTAR-II (NEDE-24011-P-A-8). Any difference between this reload and the previous one are due not only to cycle differences, but also to the difference in the methods. Therefore, making direct comparisons between the two cycles may be inconclusive.

# APPENDIX E

## USE OF NEW SAFETY LIMIT MCPR FOR CYCLE 7

The analyses required for this cycle were performed with the upgraded safety limit MCPR of 1.04, instead of the previous safety limit MCPR of 1.07. The implementation of this safety limit is a result of the utilization of fuel types with high bundle R-factors, as stipulated in Amendment 14 to GESTAR-II (NEDE-24011-P-A-8). Any difference between this reload and the previous one are due not only to cycle differences, but also to the difference in the methods. Therefore, making direct comparisons between the two cycles may be inconclusive.