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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR PEACH BOTTOM ATOMIC POWER STATION UNIT 2, RELOAD NO. 5

GENERAL ELECTRIC

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SUPPLEMENTAL RELOAD LICENSING SUBMITTAL
FOR
PEACH BOTTOM ATOMIC POWER STATION
UNIT 2, RELOAD NO. 5

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GENERAL ELECTRIC

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1. PLANT-UNIQUE ITEMS (1.0)*

Rotated Bundle Loading Error Analysis for P8DRB285: Appendix A
Fuel Assembly Rod Replacement: Appendix B
Lead Test Assemblies Extended Exposure: Appendix C
8x8R Fuel Extended Exposure: Appendix D
Developmental Channels: Appendix E
Transient Analysis Code Revision: Appendix F

2. RELOAD FUEL BUNDLES (1.0, 2.7, 3.3.1 and 4.0)

	Fuel Designation	Cycle Loaded	Number	Number Drilled
Irradiated	8DRB284L	4	210	210
	P8DRB284H	5	236	236
	P8DRB285	5	40	40
	LTA	2	2	2
New	P8DRB284H	6	136	136
	P8DRB285	6	16	16
	P8DRB299	6	124	124
Total			764	764

3. REFERENCE CORE LOADING PATTERN (3.3.1)

Nominal previous cycle core average exposure at end of cycle:	17.9 GWd/T
Minimum previous cycle core average exposure at end of cycle from cold shutdown considerations:	17.7 GWd/T
Assumed reload cycle core average exposure at end of cycle:	18.3 GWd/T
Core loading pattern:	Figure 1

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

BOC k_{eff}	
Uncontrolled	1.118
Fully Controlled	0.960
Strongest Control Rod Out	0.986
R, Maximum Increase in Cold Core Reactivity with Exposure Into Cycle, Δk	0.003

*( ) refers to areas of discussion in "General Electric Boiling Water Reactor Generic Reload Fuel Application," NEDE-24011-P-A-2 and NEDO-24011-A-2, July 1981.

5. STANDBY LIQUID CONTROL SYSTEM SHUTDOWN CAPABILITY (3.3.2.1.3)

ppm	Shutdown Margin (k) (20°C, Xenon Free)
660	0.04

6. RELOAD-UNIQUE TRANSIENT ANALYSIS INPUTS (3.3.2.1.5 and 5.2)[a]

	EOC	EOC-2 GWd/T
Void Coefficient N/A[b] (¢/% Rg)	-7.8/-9.7	-8.5/-10.6
Void Fraction (%)	39.8	39.8
Doppler Coefficient N/A[b] (¢/°F)	-0.23/-0.22	-0.22/-0.21
Average Fuel Temperature (°F)	1296	1296
Scram Worth N/A ($)[c]		
Scram Reactivity vs Time[c]		

7. RELOAD-UNIQUE GETAB TRANSIENT ANALYSIS INITIAL CONDITION PARAMETERS (5.2)

Fuel Design	Exposure (GWd/T)	Peaking Factors (Local, Radial, Axial)	R-Factor	Bundle Power (MWt)	Bundle Flow (10^3 lb/hr)	Initial MCPR
8x8R/ LTA	EOC	1.20, 1.47, 1.40	1.05	6.18	110	1.32
	EOC-2	1.20, 1.52, 1.40	1.05	6.42	108	1.27
P8x8R	EOC	1.20, 1.44 1.40	1.05	6.06	111	1.35
	EOC-2	1.20, 1.50 1.40	1.05	6.31	109	1.29

[a]Applies to REDY analyzed events only
[b]N = Nuclear Input Data
A = Used in Transient Analysis
[c]Generic, exposure independent values are used as given in "General Electric Boiling Water Reactor Generic Reload Fuel Application," NEDE-24011-P-A-2, July 1981

8. SELECTED MARGIN IMPROVEMENT OPTIONS (5.2.2)

Transient Recategorization:	No
Recirculation Pump Trip:	No
Rod Withdrawal Limiter:	No
Thermal Power Monitor:	No
Measured Scram Time:	No
Exposure Dependent Limits:	Yes
Exposures Analyzed (GWd/T):	EOC EOC-2

9. CORE-WIDE TRANSIENT ANALYSIS RESULTS (5.2.1)

Transient	Exposure Range (GWd/T)	$\hat{\phi}$ (% NBR)	$\widehat{Q/A}$ (%)	ΔCPR 8x8R/LTA	ΔCPR P8x8R	Figure
Load Rejection without Bypass	EOC	721	128	0.25	0.28	3a
	EOC-2	708	124	0.20	0.22	3b
Loss of 100°F Feedwater Heating	BOC to EOC	125	124	0.15	0.15	4
Feedwater Controller Failure	EOC	374	125	0.20	0.22	5a
	EOC-2	272	121	0.14	0.16	5b

10. LOCAL ROD WITHDRAWAL ERROR (WITH LIMITING INSTRUMENT FAILURE) TRANSIENT SUMMARY (5.2.1)

Limiting Rod Pattern: Figure 6
Includes 2.2% Power Spiking Penalty: Yes

Rod Block Reading	Rod Position (Feet Withdrawn)	ΔCPR 8x8R/P8x8R/LTA	MLHGR (kW/ft) 8x8R/P8x8R/LTA
104	3.5	0.09	16.4
105	4.0	0.11	17.0
106	4.5	0.12	17.3
107**	5.5	0.16	17.7
108	6.5	0.19	17.7
109	9.0	0.22	17.7
110	12.0	0.23	17.7

*Indicates set point selected

11. CYCLE MCPR VALUES (5.2)

Exposure Range (GWd/t)	Pressurization Events	Option A (8x8R&LTA/P8x8R)	Option B 8x8R&LTA/P8x8R
BOC to EOC-2			
	Load Rejection w/o Bypass	0.26/0.28	0.05/0.07
	Feedwater Controller Failure	0.19/0.21	0.13/0.15
EOC-2 to EOC			
	Load Rejection w/o Bypass	0.31/0.34	0.19/0.22
	Feedwater Controller Failure	0.26/0.28	0.19/0.21

Exposure Range	Non-Pressurization Events	8x8R&LTA/P8x8R/P8DRB285
BOC to EOC	Loss of Feedwater Heating	0.15/0.15/0.15
	Rotated Bundle Error	NA/0.14/0.22*
	Rod Withdrawal Error	0.16/0.16/0.16

12. OVERPRESSURIZATION ANALYSIS SUMMARY (5.3)

Transient	P_{sl} (psig)	P_v (psig)	Plant Response
MSIV Closure (Flux Scram)	1244	1273	Figure 7

13. STABILITY ANALYSIS RESULTS (5.4)

Rod Line Analyzed: 105% Rod Line

Decay Ratio: Figure 8

Reactor Core Stability Decay Ratio, x_2/x_0: 0.85

Channel Hydrodynamic Performance Decay Ratio, x_2/x_0
8x8R/P8x8R Channel: 0.29

14. ROTATED BUNDLE ERROR RESULTS (5.5.4)*

Variable Water Gap Misoriented Bundle Analysis: Yes
Includes 2.2% Power Spiking Penalty: Yes

Initial MCPR	Resulting MCPR	Resulting LHGR (kW/ft)
1.20	1.08	17.6

*See Appendix A

15. CONTROL ROD DROP ANALYSIS RESULTS (5.5.1)

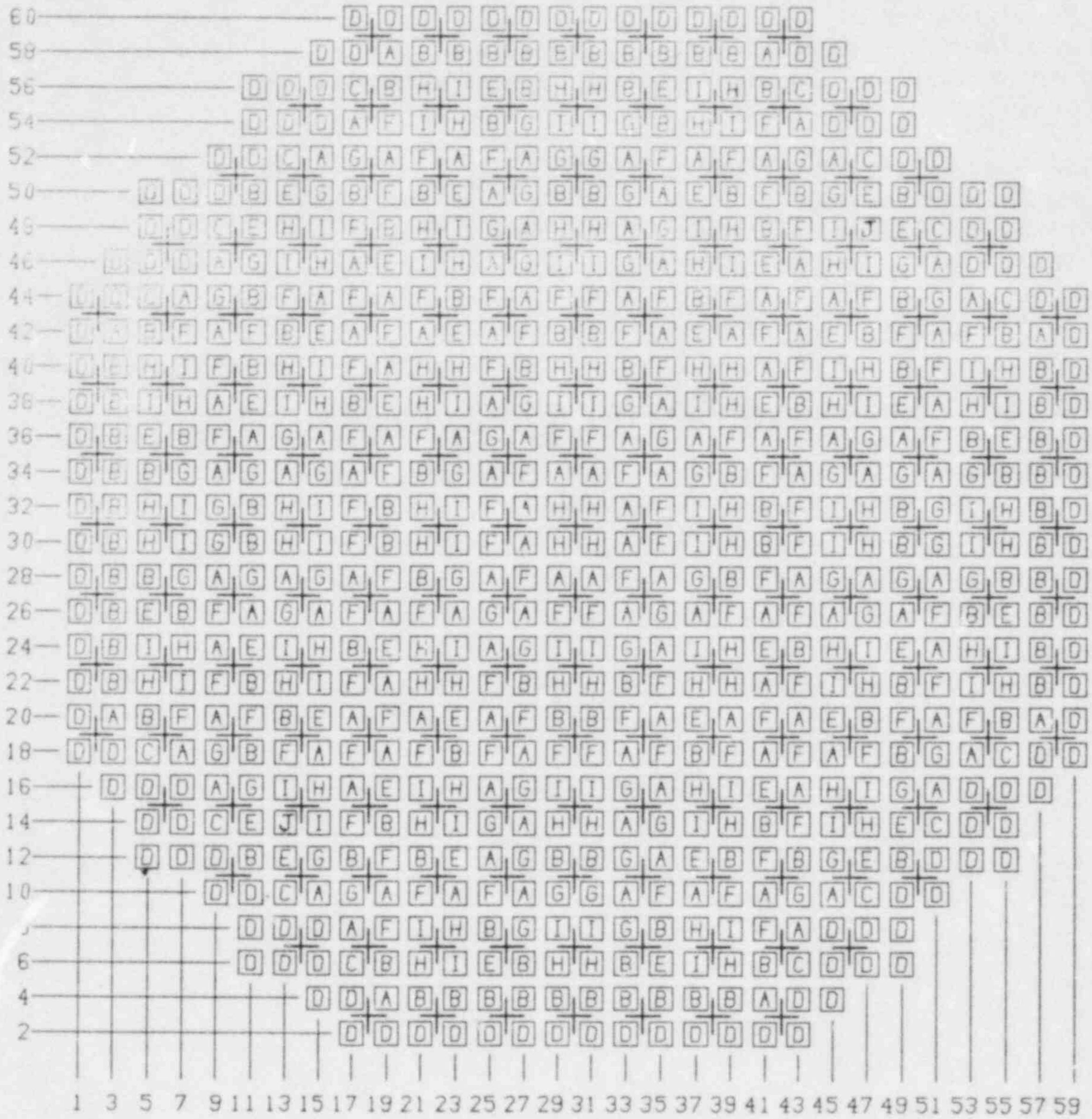
Bounding Analysis Results:
- Doppler Reactivity Coefficient: Figure 9
- Accident Reactivity Shape Functions: Figures 10 and 11
- Scram Reactivity Functions: Figures 12 and 13

Plant Specific Analysis Results:
- Parameters Not Bounded:
  - Scram Reactivity Functions: Cold and Hot Standby
- Resultant Peak Enthalpies (cal/g):

Cold	Hot Standby
165	231

16. LOSS-OF-COOLANT ACCIDENT RESULTS, NEW FUEL (5.5.2)

See "Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 2," December 1977, NEDO-24081, as amended.



FUEL TYPE

A = P8DRB284H, C6	F = P8DRB284H, C5
B = P8DRB299, C6	G = P8DRB284H, C5
C = P8DRB285, C6	H = 8DRB284L, C4
D = 8DRB284L, C4	I = P8DRB284H, C5
E = P8DRB285, C5	J = LEAD TEST ASSEMBLY, C2

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Figure 1. Reference Core Loading Pattern

DELETED

See Section 6

Figure 2. Scram Reactivity and Control Rod Drive Specifications

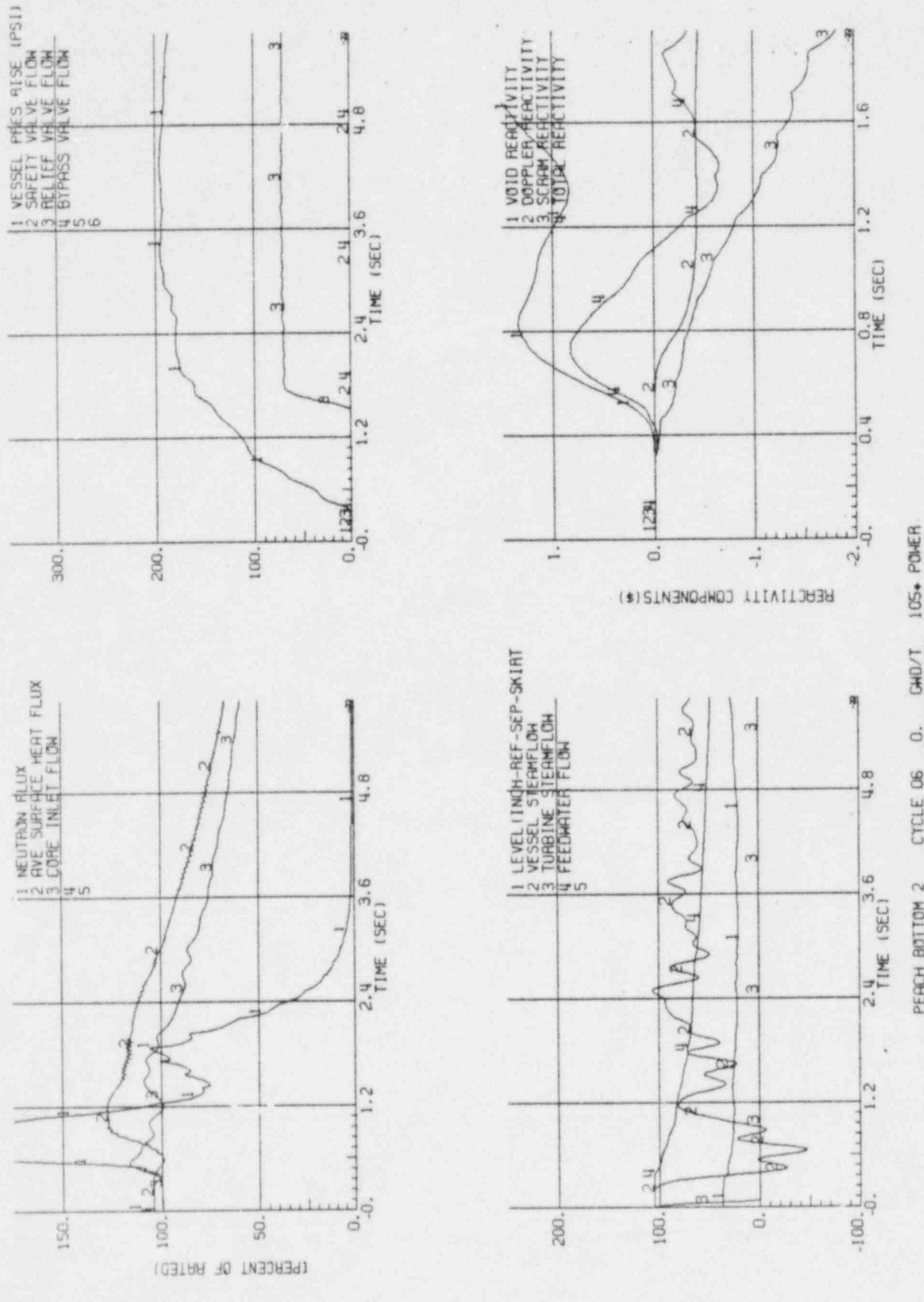

Figure 3a. Plant Response to Generator Load Rejection witho Bypass, EOC6

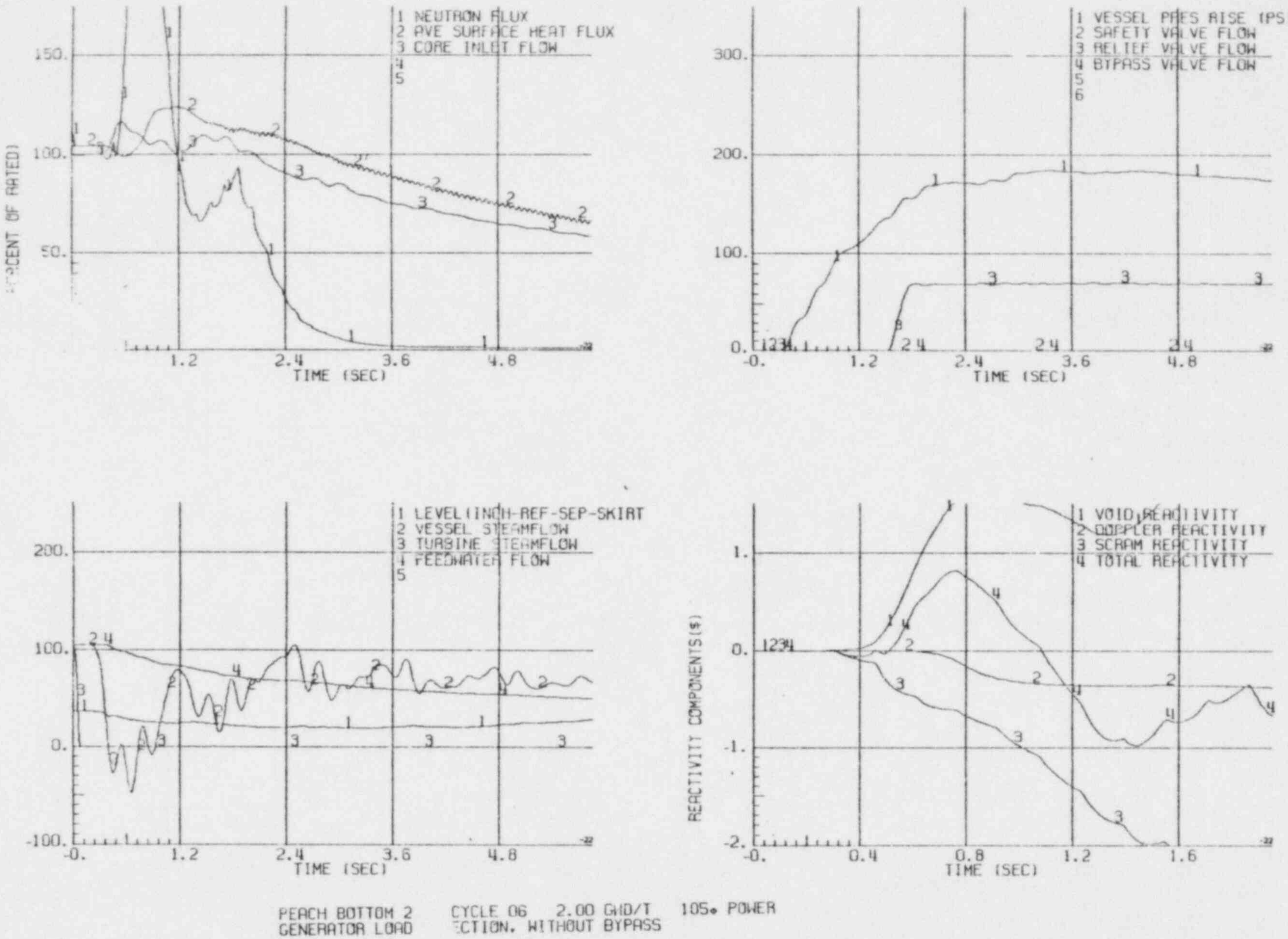

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Figure 3b. Plant Response to Generator Load Rejection without Bypass, EOC6-2 GWd/t

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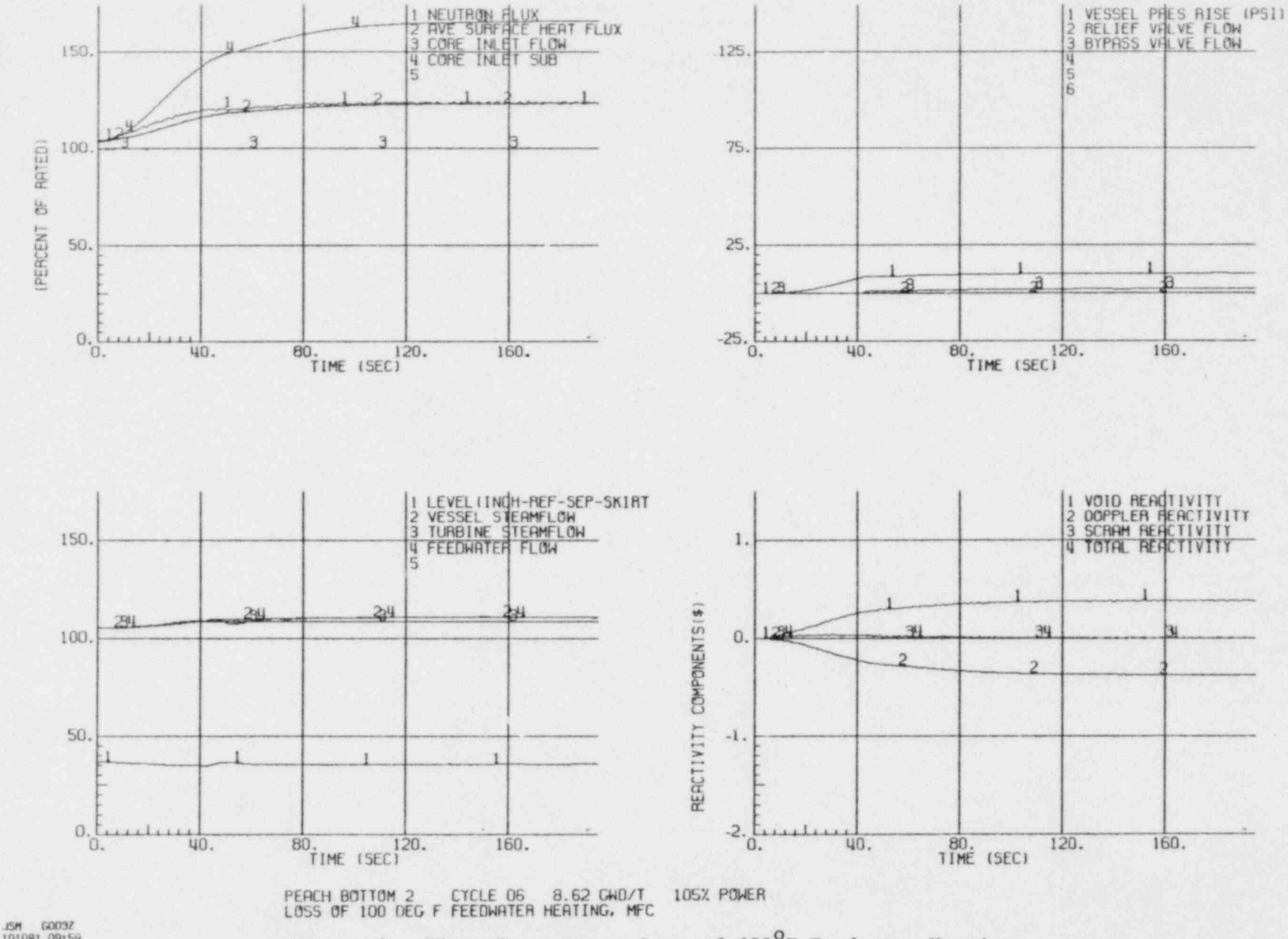

Figure 4. Plant Response to Loss of 100°F Feedwater Heating

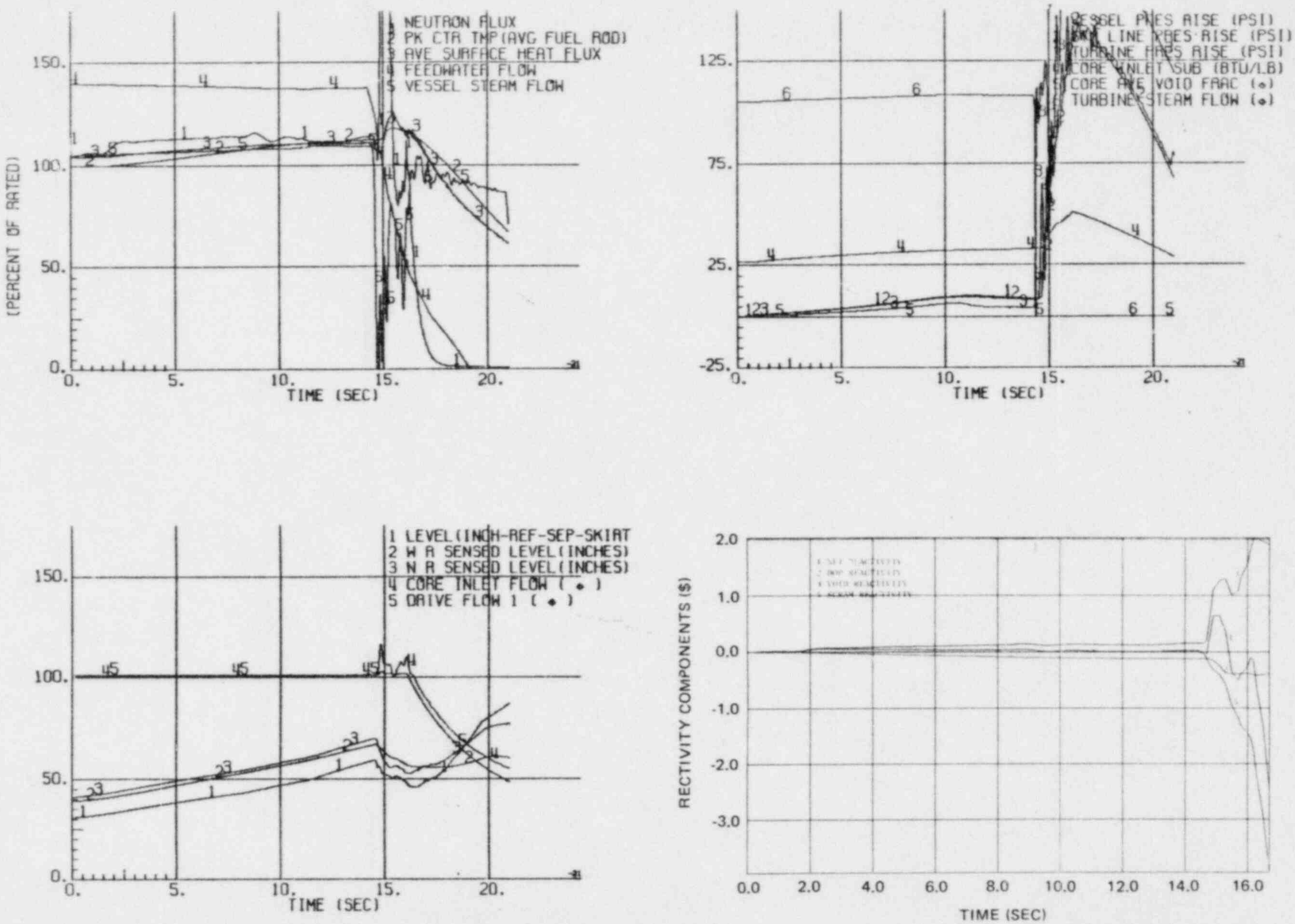

Figure 5a. Plant Response to Feedwater Controller Failure, EOC6

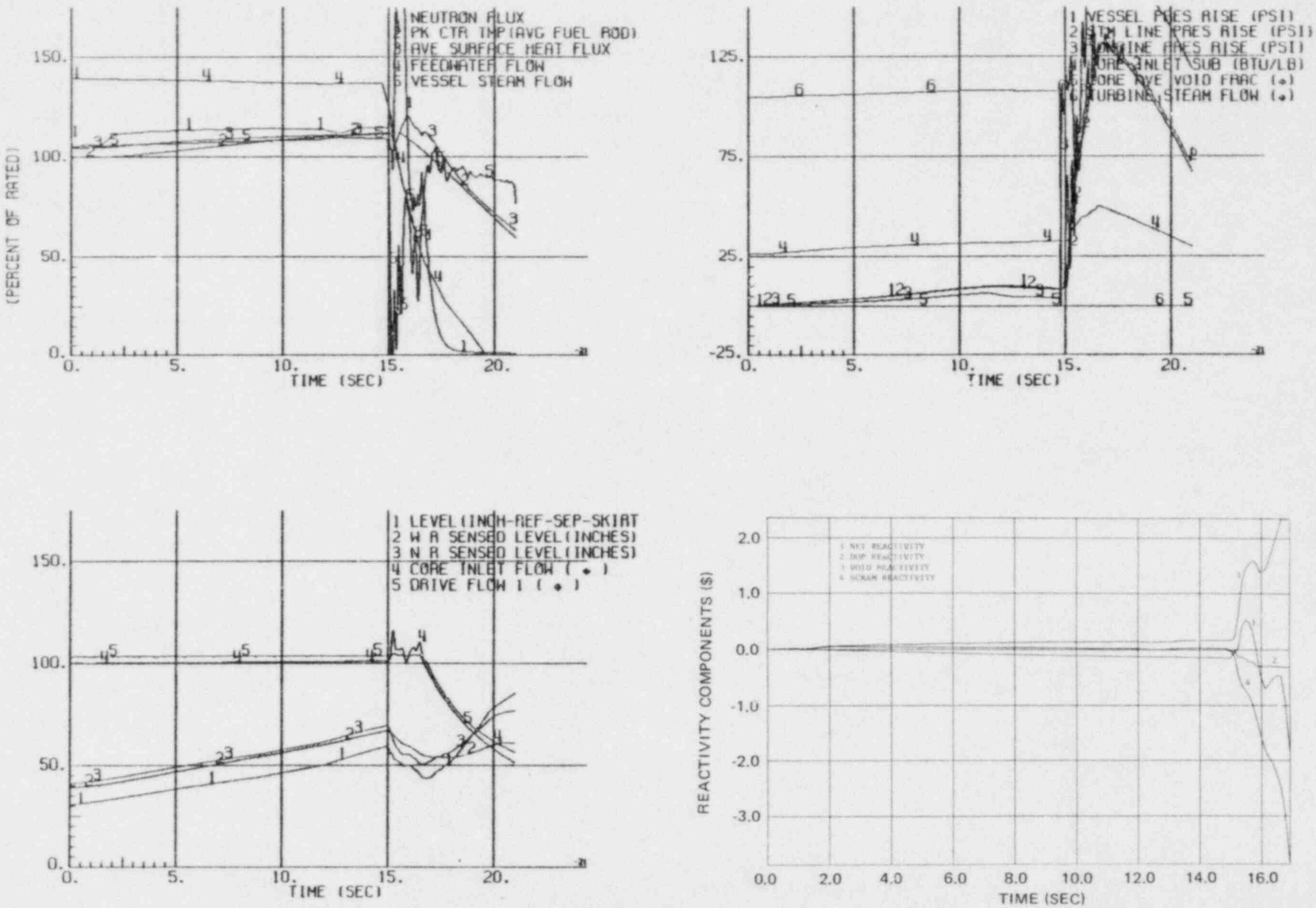

Figure 5b. Plant Response to Feedwater Controller Failure, EOC6-2 GWd/t

	02	06	10	14	18	22	26	30
59							36	
55				18		14		24
51					26		8	
47		18		10		8		8
43			26		8		28	
39		14		8				30
35	36		8		28		0	
31		24		8		30		

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Notes: 1. Rod Pattern Is 1/4 Core Mirror Symmetric, Upper Left Quadrant Shown on Map.

2. Numbers Indicate Number of Notches Withdrawn out of 48. Blank Is a Withdrawn Rod.

3. Error Rod Is Rod (26, 35).

Figure 6. Limiting RWE Rod Pattern

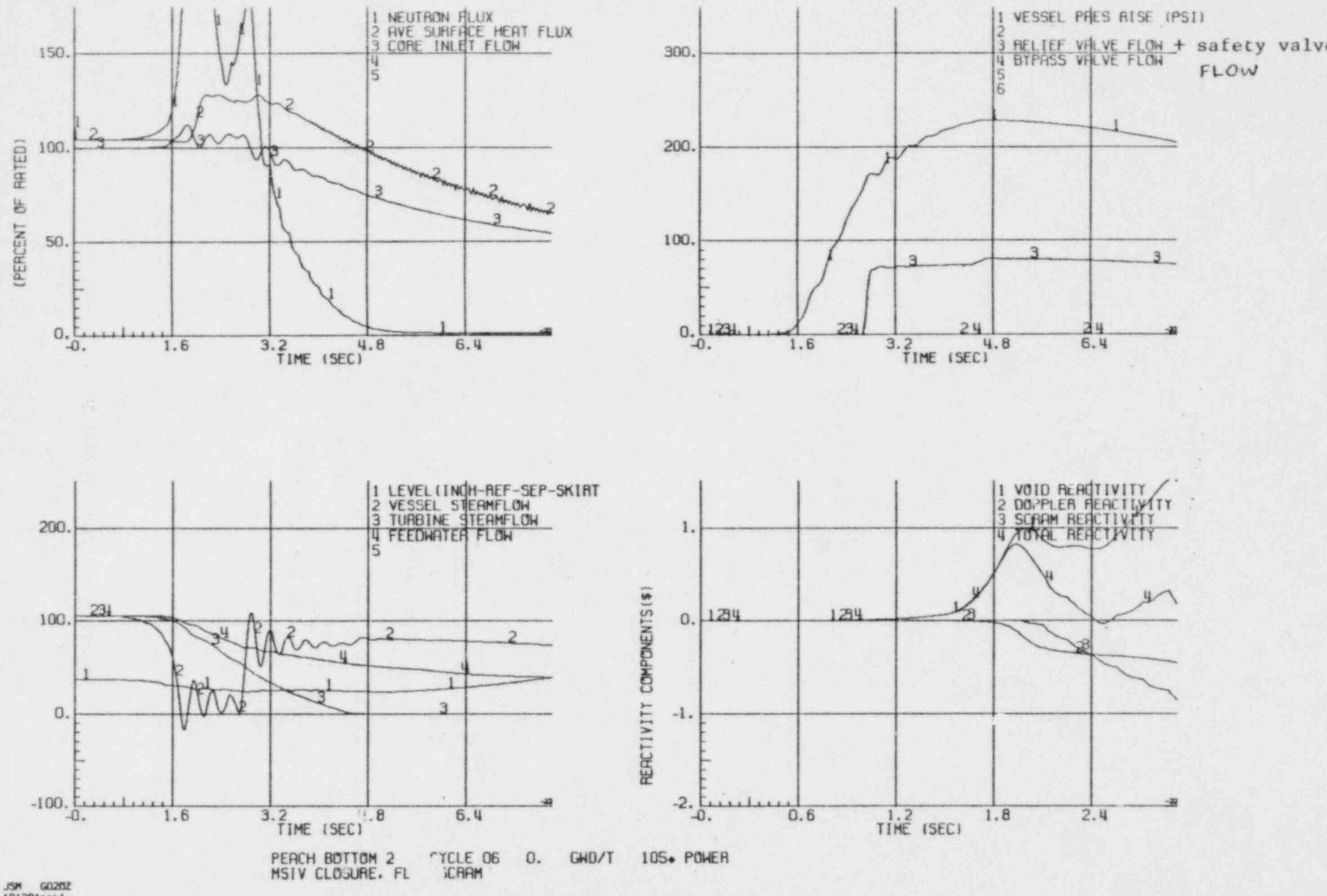

Figure 7. Plant Response to MSIV Closure (Flux Scram)

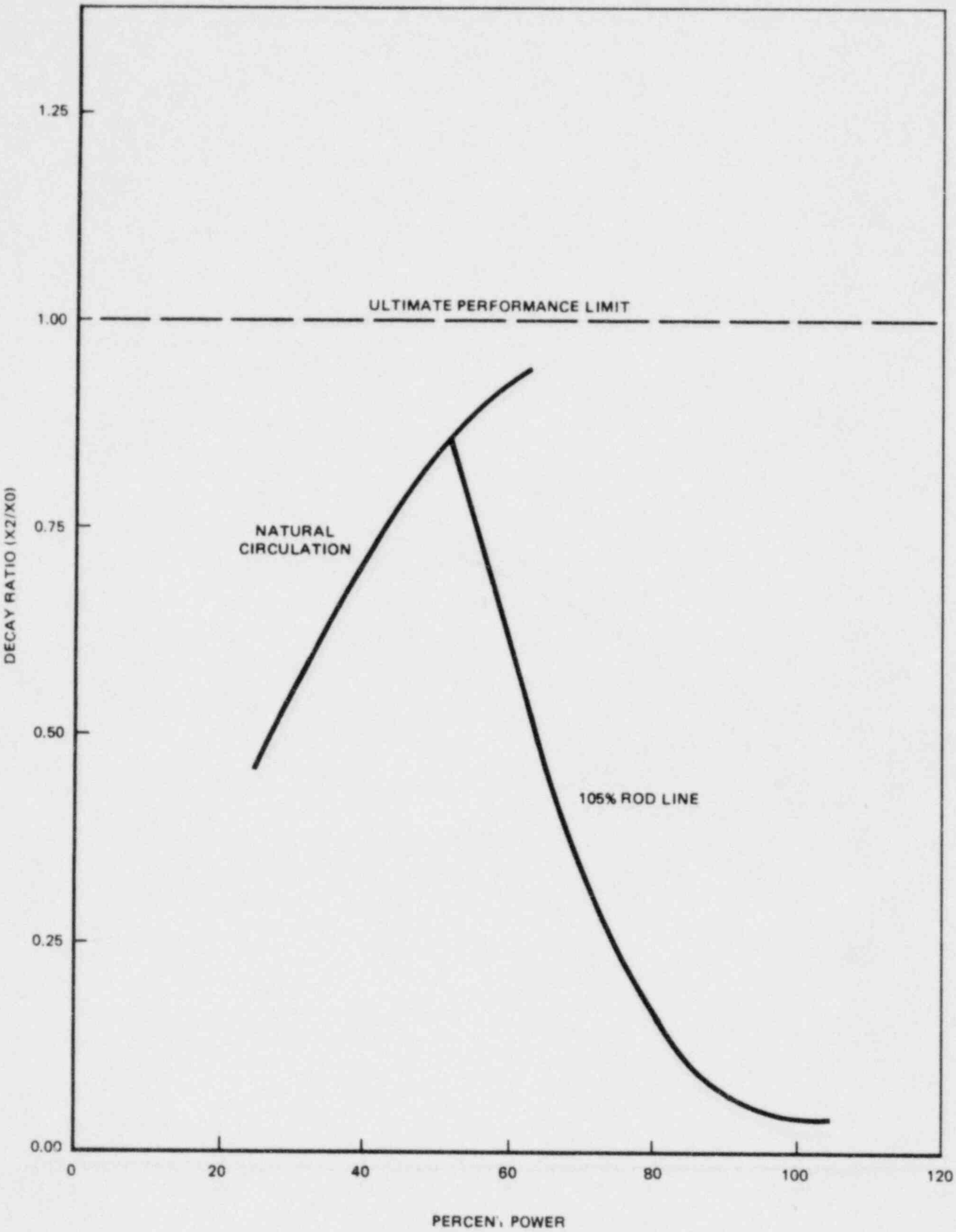

Figure 8. Reactor Core Decay Ratio versus Power

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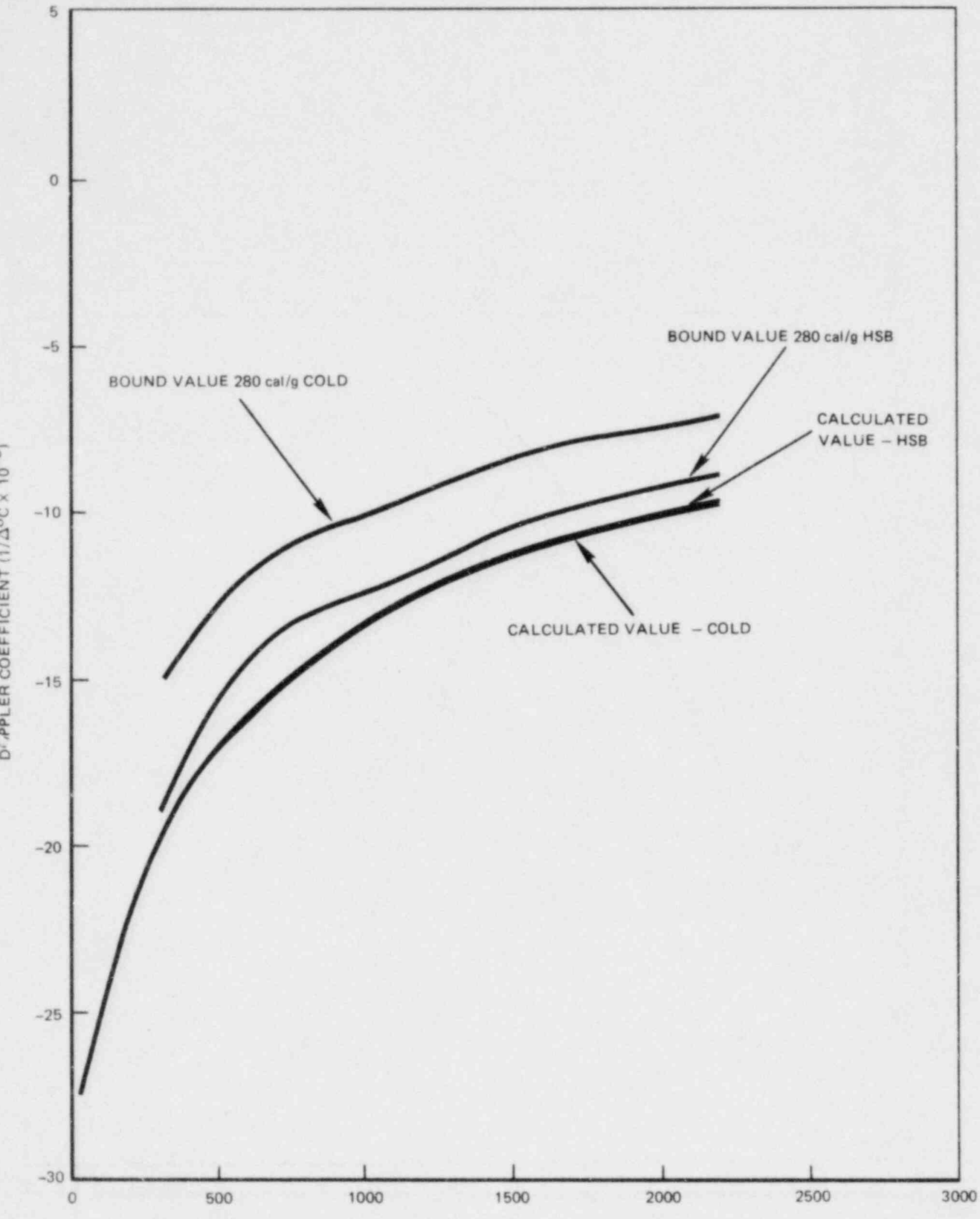

Figure 9. Doppler Reactivity Coefficient Comparison for RDA Y1003J01A34

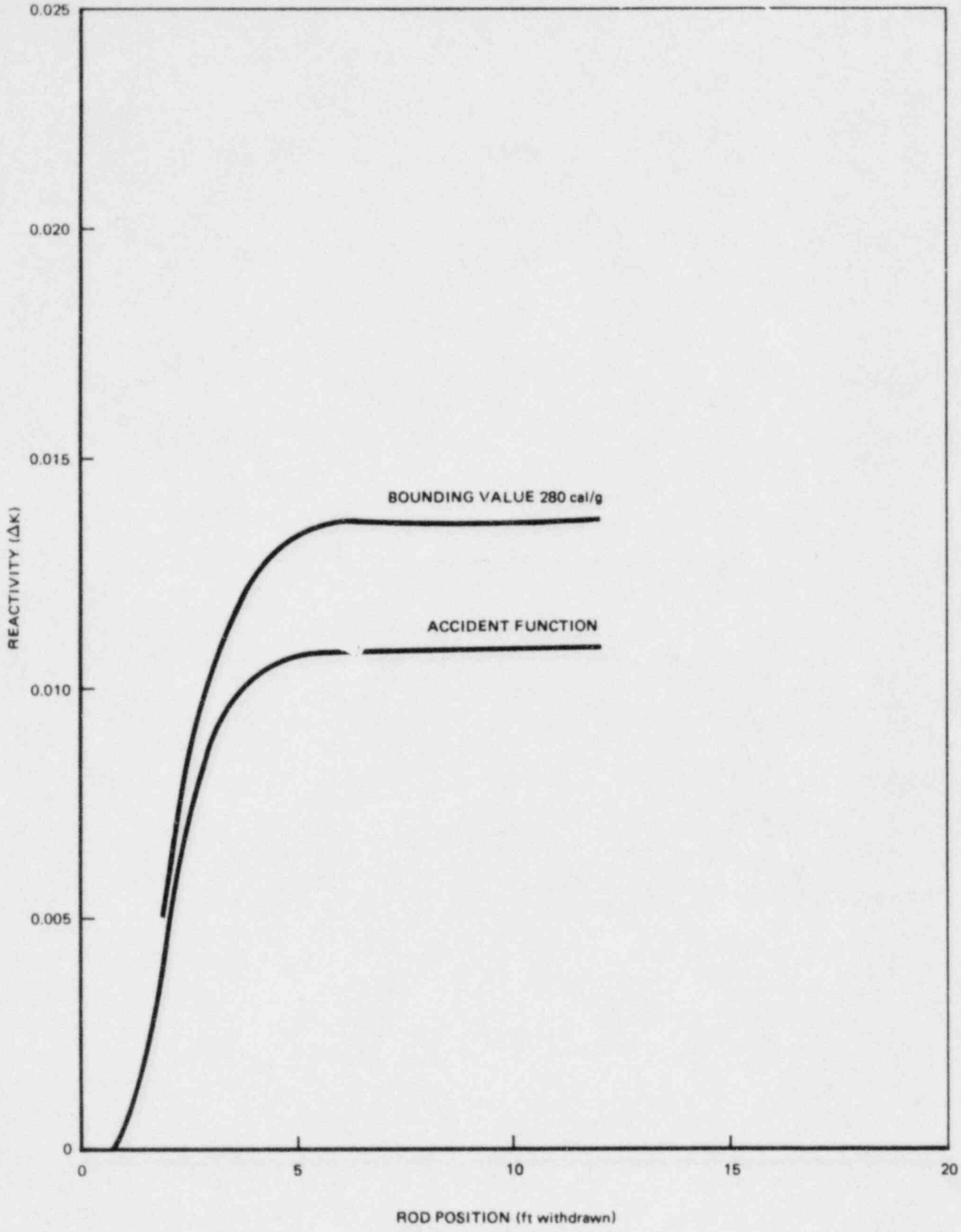

Figure 10. RDA Reactivity Shape Function, Cold

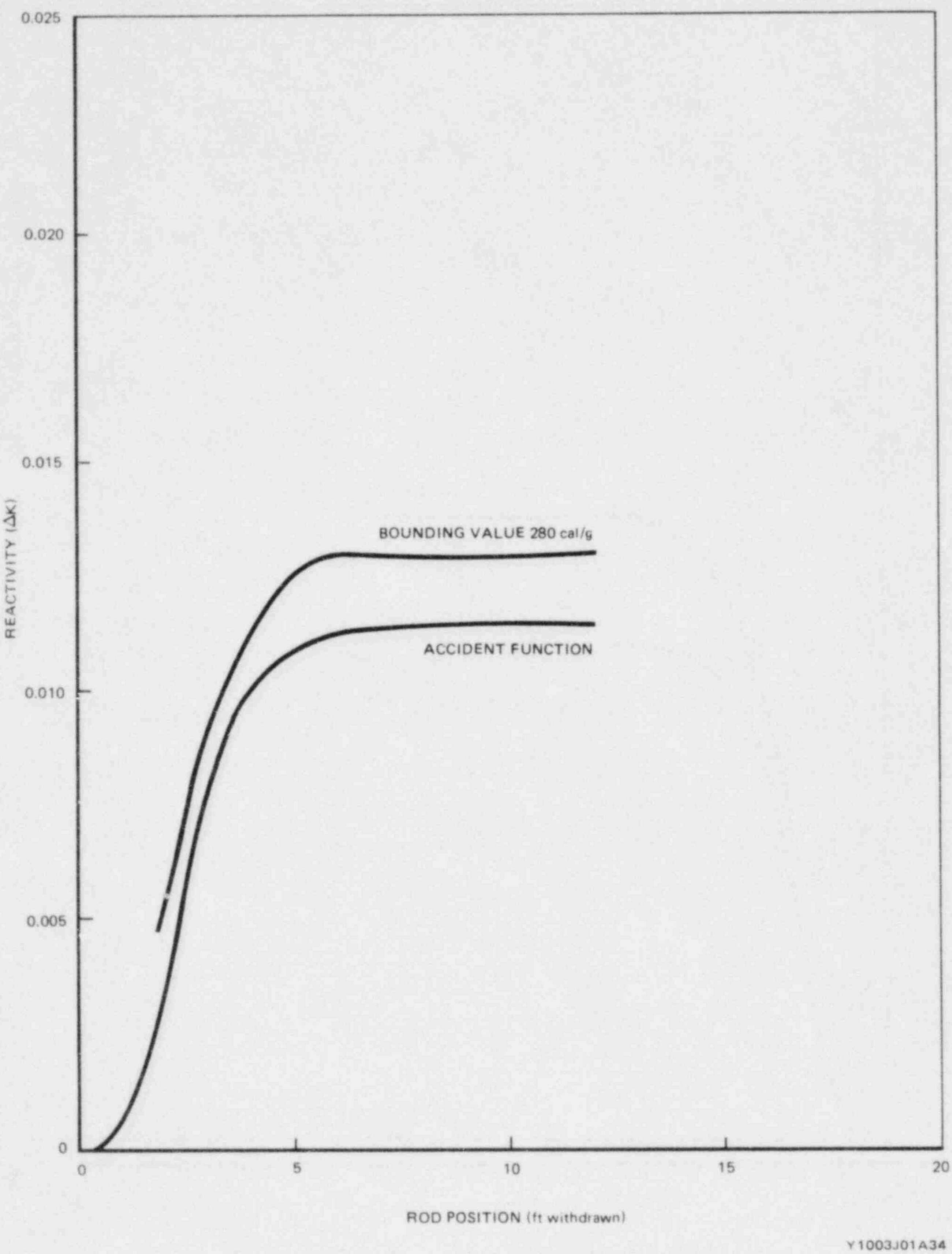

Figure 11. RDA Reactivity Shape Function, Hot Standby

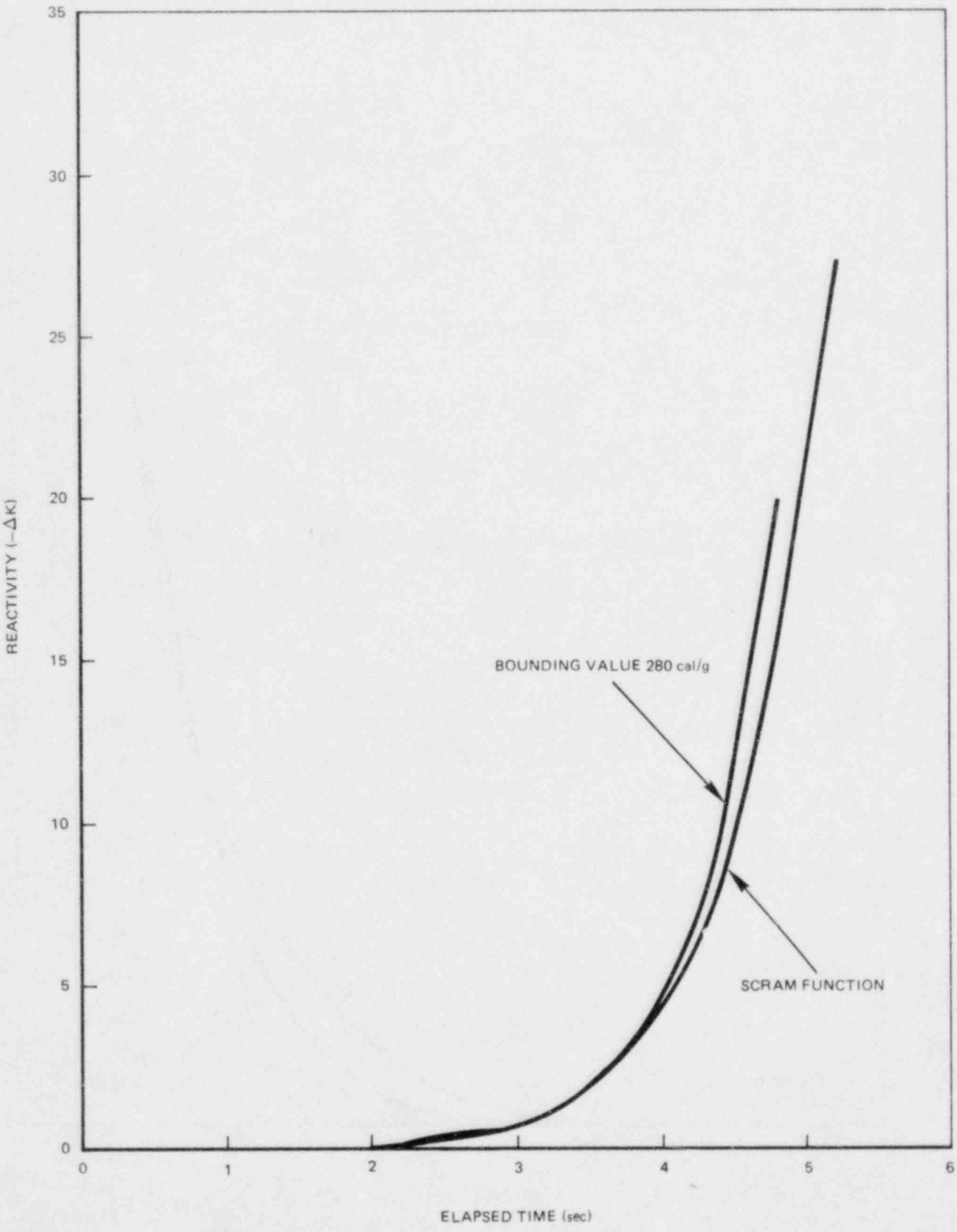

Figure 13. RDA Scram Reactivity Function, Hot Standby

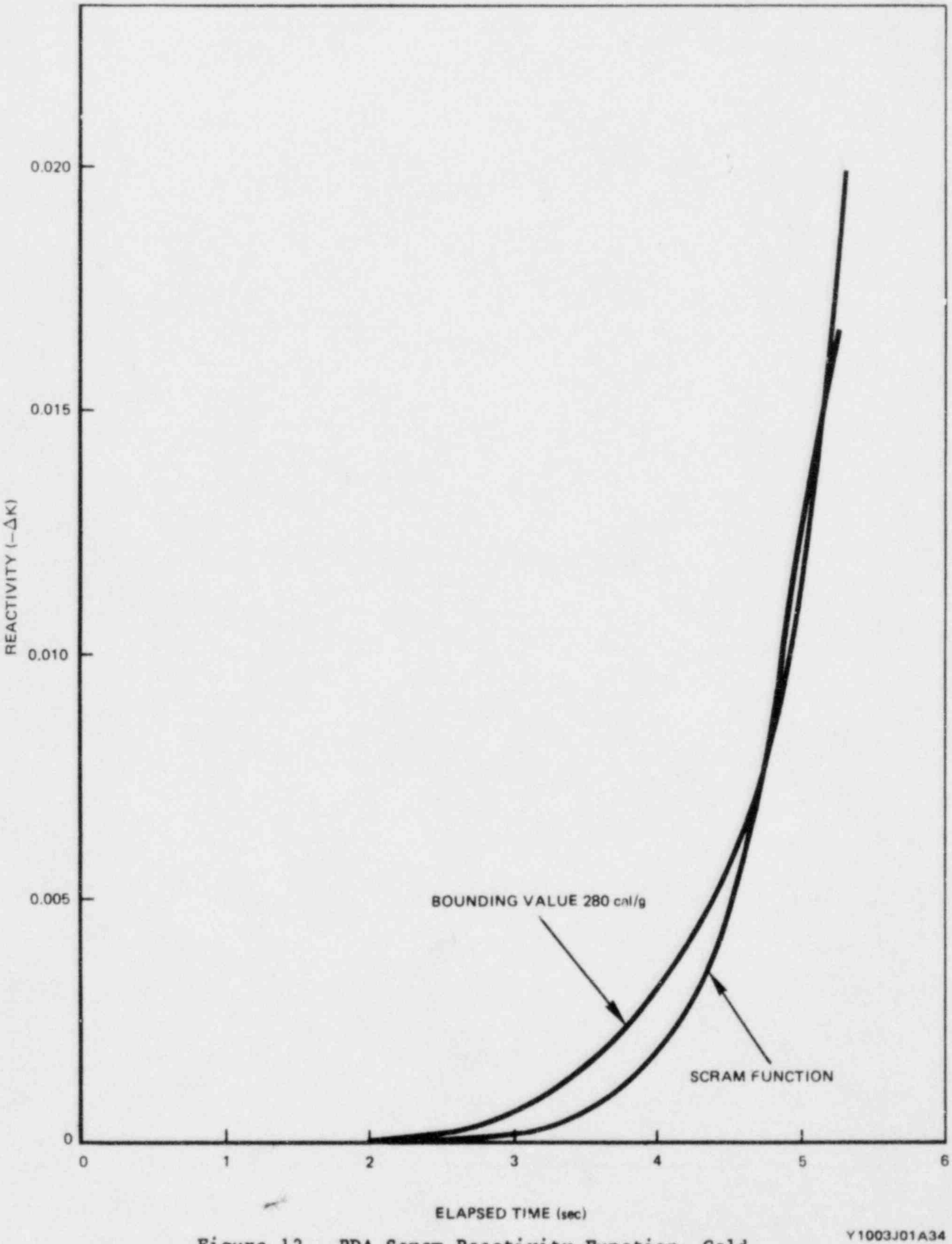

Figure 12. RDA Scram Reactivity Function, Cold

APPENDIX A

ROTATED BUNDLE LOADING ERROR ANALYSIS FOR P8DRB285

A separate rotated bundle loading error analysis was performed for the P8DRB285 bundle with the results given below. ΔCPR results for the rotated bundle loading error event are given separately for this bundle type in Section 11. The ΔCPR for other events is the same as for the P8x8R bundle type, as given in Section 11.

Bundle Type	Initial MCPR	Resulting MCPR	Resulting LHGR (kW/ft)
P8DRB285	1.28	1.08	17.7
P8x8R	1.20	1.08	17.5

APPENDIX B

FUEL ASSEMBLY ROD REPLACEMENT

B.1 INTRODUCTION

During the Reload No. 5 refueling outage, six fuel rods will be removed from each of two previously irradiated fuel assemblies and replaced with fresh rods with U-235 enrichments as shown in Table B-1. The removed rods will be examined and punctured for fission gas pressure measurement. These rods will not be used during future operation. The average enrichment of the replacement rods is less than the initial enrichment of the rods they are replacing to compensate for fuel depletion. They were selected to assure that the reactivity and power peaking of the reconstituted assemblies will be similar to that of a non-reconstituted assembly. Consequently, the nuclear characteristics of the reconstituted assemblies are essentially identical to non-reconstituted assemblies. The purpose of this appendix is to report the results of the analyses and safety evaluation for operation of the fuel assemblies after replacement of the fuel rods.

B.2 EVALUATIONS AND ANALYSES

B.2.1 Nuclear and Thermal Parameter Evaluations

Standard lattice physics calculations were made for the reconstituted assemblies, including simulation of the fresh rods. Over the exposure range of interest, the computed lattice reactivities of the reconstituted assemblies are on average within 0.03% ΔK of the non-reconstituted assembly reactivities. The maximum fuel rod power peaking values for the reconstituted assemblies are always less than 1% greater than the values for the non-reconstituted assemblies, and are lower than the power peaking for the non-reconstituted assemblies throughout most of their operation. Based on the small calculated changes to K_∞ and local peaking caused by the replacement fuel rods, there will be a negligible effect on the nuclear and thermal performance of the reconstituted fuel assemblies.

B.2.2 Mechanical Design Evaluation

The six replacement fuel rods in each of the two reconstituted assemblies are mechanically similar to the fuel rods which they are replacing and also to the standard fuel rods in the Reload No. 5 fuel bundles. The only mechanical difference is a longer upper end plug on each replacement rod to accommodate the irradiation growth of the rods in the reconstituted assemblies. An analysis of differential rod growth in the reconstituted assemblies shows that the replacement fuel rods are mechanically compatible with the irradiated rods and thus will have no adverse effect on the safety analyses for Cycle 6 or subsequent cycles for Peach Bottom 2 (PB 2). The peak linear heat generation rates of the reconstituted assemblies are still within the operating limit of 13.4 kW/ft which was used in evaluating the mechanical performance of the maximum duty fuel rod in Reload No. 5. Therefore, the results of the fuel rod thermal and mechanical design evaluations in NEDE 24011-P-A-2 are conservatively applicable to the reconstituted assemblies.

B.2.3 Evaluation of the Effect of the Fresh Fuel Rods on PCT/MAPLHGR

The effect on MAPLHGR of the replacement of six exposed fuel rods with fresh rods in two PB 2 bundles has been evaluated. The reconstitution is conservatively estimated to increase peak clad temperature (PCT) by 14°F. Since the current maximum PCT is 1958, this increase will result in a PCT of 1972°F, well below the 2200°F limit. Thus, there will be no change in MAPLHGR for the reconstituted bundles.

The estimated PCT increase is based on:

1. an increase in average stored energy of the bundle due to the introduction of fresh rods which have a higher stored energy, and

2. a slight shift in power to the center 16 rods due to the change in local peaking for the reconstituted bundles.

B.2.4 Transient Analysis for Cycle 6

Based on the analysis results described in Section B.2.1 above, the transient analysis results contained in this submittal are unaffected by fuel rod replacement of the two fuel assemblies.

B.3 SUMMARY AND CONCLUSIONS

It is concluded, based on the results of the evaluations and analysis described in Section B.2, that the accident and transient analyses of Cycle 6 are insignificantly affected and the operating limits of Cycle 6 are unaffected by the reconstitution of the two fuel assemblies. The operating MCPR limit is given in Section 11 of this submittal.

Table B-1

RECONSTITUTION ENRICHMENTS (wt % U235)

Rod Location	Original	Replacement
A-1	1.3	1.3
H-1	2.0	1.7
B-2	2.4	2.0
C-2	2.4	2.0
D-4	3.95	3.3
F-6	3.95	3.8

APPENDIX C

LEAD TEST ASSEMBLIES EXTENDED EXPOSURE

C.1 PROPOSED PROGRAM

The Peach Bottom 2 Lead Test Assembly (LTA) fuel program is one of several programs in the U.S. whereby lead burnup bundles are being extended to peak-pellet exposures greater than 44,000 MWd/MT (40,000 MWd/ST) but not to exceed 50,000 MWd/MT. Information from these programs will be used to systematically determine the impact on fuel reliability and weigh the advantages of extended exposures relative to other uranium utilization improvement methods.

The four LTAs (8DRB260) inserted into Peach Bottom 2 at the beginning of Cycle 2 (Reload 1) are currently the highest exposure 8x8R fuel assemblies in operation in any operating reactor. Previous to Cycle 5, these LTAs were licensed for operation during Cycle 5 (Reference C-1). Inspections performed during the EOC 4 refueling outage confirmed the mechanical integrity of the LTAs for continued operation in Cycle 5. Two of the four LTAs have been proposed for further extended operation in Cycle 6. The program plan again includes the inspection of these assemblies to ascertain their mechanical integrity before Cycle 6 operation.

C.2 FUEL MECHANICAL DESIGN ANALYSIS

Exposure-dependent fuel mechanical design analyses for the extended exposure LTAs have been performed for conditions which meet or exceed expected Cycle 6 operating conditions in Peach Bottom 2. Models, assumptions, and material properties used in these analyses are those documented in Reference C-2. Calculated results are given below.

C.2.1 Fuel Rod Thermal Analysis

Safety evaluations are performed and measured against established safety criteria. The consequence of calculating values which exceed such criteria is that fuel failure must be assumed to occur. For plant normal and abnormal operation, this is not permissible. Fuel failure is defined as a perforation of the cladding which would permit the release of fission products to the reactor coolant. The mechanisms which could cause fuel damage in reactor abnormal operational transients are (1) rupture of the fuel rod cladding due to strain caused by relative expansion of the UO_2 pellet and (2) severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain on the Zircaloy claddng has been established as the safety limit below which fuel damage due to overstraining of the fuel cladding is not expected to occur. The fuel cladding integrity safety limit ensures that fuel damage resulting from severe overheating of the fuel rod cladding caused by inadequate cooling is avoided. Of these criteria, only the linear heat generation rate associated with the 1% plastic strain safety limit is affected by increased fuel exposures. Analyses performed for the extended exposure fuel bundles resulted in 1% plastic strain values of 16.1 kW/ft at a peak-pellet exposure of 55,000 MWd/MT (50,000 MWd/ST) for UO_2 fuel rods and 16.8 kW/ft at 49,700 MWd/MT (45,100 MWd/ST) for urania-gadolinia rods. Both values include the 2.2% power spiking penalty documented in Reference C-2. These results assure that the same minimum margin to 1% plastic strain (175% of minimum steady-state power) reported in Reference C-2 is maintained. These linear heat generation rate values are used during specific evaluations of transients due to single operator error or equipment malfunction to ensure that the safety limit is not exceeded.

C.2.1.1 Fuel Cladding Temperatures

The cladding surface temperature is calucculated using the cladding surface heat flux at a given axial position on a fuel element in conjunction with the overall cladding-to-coolant film coefficient. The models used are noted in Reference C-2. The inside, average, and outside cladding temperature during normal operation at the end of Cycle 6 are calculated not to exceed 809°F, 773°F, and 738°F, respectively.

C.2.1.2 Fission Gas Release

The amount of fission gas released during a time increment is calculated based on the fission gas generated and fission gas release fraction. The calculated maximum fission gas release fraction in the extended exposure fuel rod with the most limiting peaking factors is less than the 25% noted in Reference C-2.

C.2.1.3 Incipient Center Melting

The fuel is designed so that fuel melting is not expected to occur during normal steady-state full power operation which remains valid even at extended exposures. Linear heat generation rates associated with incipient fuel center melting are greater than 125% of normal steady-state full power operation at EOC 6.

C.2.2 Fuel Assembly Mechanical Evaluations

The fuel assembly is evaluated by analyses, tests, and experience to demonstrate fuel assembly structural integrity. When analyses are used to demonstrate structural integrity, resulting stress and/or strain levels are compared to the associated mechanical limits documented in Reference C-2. Results of the fuel rod mechanical analyses of the normal and transient loads for the extended exposure fuel are given below. The results of the combined LOCA and seismic evaluation documented in Reference C-2 do not change.

C.2.2.1 Cladding Creep Collapse

A cladding creep collapse evaluation was performed with the models documented in Reference C-2. Results of this evaluation demonstrate that cladding creep collapse is not expected to occur in the event of a maximum overpressure transient throughout Cycle 6.

C.2.2.2 Stress Evaluations

Fuel rod stress analyses of the extended exposure LTAs were performed with the model documented in Reference C-2 for operation through Cycle 6. These analyses showed that the fuel design ratios were well below 1.0.

C.2.2.3 Deflection Evaluation

The operational fuel rod deflections considered are a result of manufacturing tolerances, flow-induced vibration, thermal effects, and axial load. Deflections of the extended exposure LTAs were evaluated and compared to the fuel rod-to-fuel rod and fuel rod-to-channel spacing deflection limits given in Reference C-2. This comparison demonstrated that the fuel rod clearance criterion was met.

C.2.2.4 Fatigue Evaluation

The cyclic loads considered in cladding fatigue analysis are coolant pressure and thermal gradients. The analysis performed for the higher exposure LTAs was based on previous and projected operating cycles through the end of Cycle 6, maximum and minimum pressures, and the stresses determined in Subsection C.2.2.2. The cumulative fatigue damage was calculated to be less than the allowable fatigue limit.

C.2.3 Fuel Rod Corrosion, Hydriding and Fretting Wear Considerations

C.2.3.1 Potential for Hydriding

The potential for hydriding is discussed in Reference C-2 and is not affected by higher fuel exposures.

C.2.3.2 Fuel Element Energy Release

Significant boiling transition is not possible at normal operating conditions or under conditions of abnormal operational transients because of the thermal margins at which the fuel is operated and the high fuel burnups. It can, therefore, be concluded that the energy release and potential for a metal-water reaction is not an important consideration during normal operation or abnormal transients. The insignificant energy released in the event of boiling transition reported in Reference C-2 does not change because of the extended fuel exposures.

C.2.3.3 Fretting Wear and Corrosion

As discussed in Reference C-2, no significant fretting wear or corrosion has been observed throughout a continuing fuel surveillance program. Increased exposures are not expected to significantly change the observed results. It is expected that the LTAs will be visually examined before loading in Cycle 6.

C.3. IMPACT ON RELOAD ANALYSES

All of the models documented in Reference C-2 are applicable for use with higher fuel exposures. However, some inputs into these models are exposure-dependent and are reflected in calculated results. A description of these exposure-dependent changes is given below.

C.3.1 Nuclear Evaluations

The nuclear evaluations are comprised of two analyses: lattice and core. Most of the lattice analysis is performed during the bundle design process. The results of these single bundle calculations are reduced to "libraries" of lattice reactivities, relative rod powers, and few group cross-sections as functions of instantaneous void, exposure, exposure-void history, control state, and fuel and moderator temperature. Because of the exposure dependence of these results, the libraries were expanded to include higher burnups as noted below. The core analysis is unique for each reload. It is performed using the above lattice "libraries" to demonstrate that the core meets all applicable safety limits. The effects of higher fuel exposures are thus reflected in the core analysis results through use of the expanded "libraries."

C.3.1.1 Reactivity

Traditionally, bundle reactivities have been expressed in terms of k_∞ (i.e., the neutron multiplication of an infinite array of like bundles). This lattice reactivity is a function of lattice average enrichment, gadolinia loading, void fraction, hydrogen-to-uranium ratio, and exposure. Hot reactivity of the extended exposure LTAs decreases by 0.05 Δk_∞ from a lattice exposure of 38,000 to 50,000 MWd/MT (35,000 to 45,000 MWd/ST).

C.3.1.2 Local Peaking Factors

For a given lattice at a given void fraction, the maximum local peaking factor will occur at different fuel rods as the exposure increases. This is due to the different depletion and generation rate of the various fissile nuclides in each fuel rod. Calculated maximum local peaking factor for the extended exposure LTAs increases by 0.06 from a lattice exposure of 38,000 to 50,000 MWd/MT (35,000 to 45,000 MWd/ST).

The local peaking factor does vary with void fraction, and this dependence is taken into account in the calculations used to assign local peaking factors to each axial segment of the fuel. The above values are for 0.40 void, as this is the typical average bundle void fraction.

C.3.1.3 Doppler Reactivity

The Doppler coefficient is of prime importance in reactor safety. The Doppler coefficient is a measure of the reactivity change associated with an increase in the absorption of resonance-energy neutrons caused by a change in the temperature of the material in question. The Doppler reactivity coefficient provides instantaneous negative reactivity feedback to any rise in fuel temperature, on either a gross or local basis. Maximum and minimum calculated Doppler coefficients at several exposures are shown in Reference C-2.

C.3.1.4 Void Effect

The most important of these effects is void reactivity. The overall void coefficient is always negative over the complete operating range, since the BWR design is undermoderated. The reactivity change due to the formation of voids results from the reduction in the number of neutrons slowing down due to the decrease in the water-to-fuel ratio. Beyond 11,000 MWd/MT (10,000 MWd/ST), the void effect is essentially constant.

C.3.2 Steady-State Hydraulic Analysis

Core steady-state thermal-hydraulic analyses are performed using a model of the reactor core, which includes hydraulic descriptions of orifices, lower tieplates, fuel rods, fuel rod spacers, upper tieplates, the fuel channel, and core bypass flow paths. Model details are documented in Reference C-2. The flow distribution to the fuel assemblies and bypass flow paths is calculated on the assumption that the pressure drop across all fuel assemblies and bypass flow paths is the same. An iteration is performed on flow through each flow path (fuel assemblies and bypass paths), which equates the total differential pressure (plenum to plenum) across each path and matches the sum of the flows through each path to the total core flow. This analysis is insignificantly affected by extended exposure fuel.

C.3.3 Reactor Limits Determination

Limits on plant operation are established to assure that the plant can be safely operated and not pose any undue risk to the health and safety of the public. This is accomplished by demonstrating that the radioactive release from plants for normal operation, abnormal operational transients, and postulated accidents meets applicable regulations in which conservative limits are documented. This conservatism is augmented by using conservative evaluation models and observing limits which are more restrictive than those documented in the applicable regulations. These observed operating limits and methods used to determine if the limits are met are documented in Reference C-2.

C.3.3.1 Fuel Cladding Integrity Safety Limit

The generation of the Minimum Critical Power Ratio (MCPR) limit requires a statistical analysis of the core near the limiting MCPR condition. Bounding statistical analyses have been performed which provide conservative safety limit MCPRs for operating BWR plants. These safety limit MCPRs conservatively apply for all reload cycles including equilibrium cycle. Insertion of low-powered extended exposure LTAs does not change the conclusions of these bounding analyses.

C.3.3.2 MCPR Operating Limits

The MCPR operating limit is established to ensure that the fuel cladding integrity safety limit is not exceeded for any moderate frequency transient. This operating requirement is obtained by addition of the absolute, maximum MCPR value for the most limiting transient from rated conditions postulated to occur at the plant to the fuel cladding integrity safety limit. Higher fuel exposures are reflected in the nuclear input data. However, due to the high exposure, these fuel assemblies will operate at significantly lower power levels than other 8x8R bundles and will not be near MCPR operating limits.

C.3.3.3 Vessel Pressure ASME Code Compliance

To assure that the peak allowable pressure of 110% of the vessel design pressure is not exceeded, the most severe isolation event with indirect scram and credit for subsequent valve operation is evaluated. The event which satisfies this specification is the closure of all Main Steam Line Isolation Valves (MSLIVs) with indirect (flux) scram, and the margin at extended exposures will not exceed the nominal end-of-cycle margin because of the reduced power levels. The model used to analyze this event is described in Reference C-2. The results of this analysis are not significantly affected by the LTA bundles.

C.3.3.4 Stability Analysis

Two types of stability are examined utilizing a linearized analytical model. First, is the hydrodynamic channel stability of one or more types of channels operating in parallel with other channels in the core. Second, is the reactivity feedback stability of the entire reactor core which also involves power oscillations. The assurance that the total plant is stable and, therefore, has significant design margin is demonstrated analytically when the acceptable performance limit of a decay ratio less than 1.0 or a damping coefficient greater than 0.0 is met for each type of stability. These criteria must be satisfied for both usual and unusual operating conditions of the reactor that may be encountered in the course of BWR plant operation.

The analysis is performed using the models documented in Reference C-2 at the most limiting condition, which usually occurs near the end of cycle, with power peaking toward the bottom of the core. The most sensitive reactor operating condition is that corresponding to natural circulation flow and a power level equal to or greater than the rated rod pattern power level. Extended exposures for the LTAs are reflected in the nuclear characteristics used in the analysis.

C.3.3.5 Accident Evaluations

Accidents are events which have a projected frequency of occurrence of less than once in every 100 years for every operating BWR. The broad spectrum of postulated accidents is covered by six categories of design basis events. These events are the control rod drop, main steam line break, loss-of-coolant, refueling, recirculation pump seizure, and fuel assembly loading accidents. Consequences of these events with the low-powered extended exposure LTAs are not as great as lower burnup bundles. However, the MAPLHGR values for the test bundles have been extended to an average planar exposure of 55,000 MWd/MT (50,000 MWd/ST). These new MAPLHGR values and associated peak cladding temperatures and oxidation fractions were incorporated into Reference C-3.

C.4 REFERENCES

C-1. **Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station Unit 2, Reload No. 4**, NEDO-24237, February 1980.

C-2. **Generic Reload Fuel Application**, NEDE-24011-P-A-2, August 1981.

C-3. **Loss of Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 2**, NEDO-24081, December 1977, including E&A sheet 6 of November 1981.

APPENDIX D

8x8R FUEL EXTENDED EXPOSURE

During Cycle 6, the 8DRB284L fuel (which was inserted at BOC4) is expected to attain a peak pellet exposure in excess of 40,000 MWd/ST (44,000 MWd/MT) but not to exceed 50,000 MWd/MT. Thermal and mechanical analyses have been performed for this fuel type in accordance with the NRC approved methods described in Reference D-1 to an exposure of 50,000 MWd/MT. Results of those analyses are within the applicable criteria of Reference D-1. Results of the analyses for the linear heat generation rates associated with 1% plastic strain and incipient center melting indicate that the LHGR values for P8x8R fuel in Table 2-3a and Table 2-4 of Reference D-1 for 50,000 MWd/MT peak pellet exposure are applicable to the 8DBR284L fuel.

REFERENCE

D-1. General Electric Boiling Water Reactor Generic Reload Fuel Application, NEDE 24011 - P-A-2, July 1981.

APPENDIX E

DEVELOPMENTAL CHANNELS

E.1 ANALYSES

The analyses given in Reference E-1 are applicable to continued use of developmental channels. The location and exposure of the developmental channels has changed. However, the thermal-hydraulic, nuclear, and safety analyses presented in the main body of this submittal are applicable to the continued use of developmental fuel channels.

E.2 REFERENCES

E-1. Developmental Channels Supplemental Information for Reload 1 Licensing Submittal for Peach Bottom Atomic Power Station Unit 2, NEDO-21172, Rev. 1, Supplement 2, March 1976.

APPENDIX F

TRANSIENT ANALYSIS CODE REVISION

F.1 CODE

The pressurization transient events reported in this submittal were analyzed with the ODYN M04 transient analysis code, which is a revision of the ODYN code described in Reference F-1. A description of this revised code and a comparison to the previous code are given in References F-2 and F-3.

F.2 REFERENCES

F-1. **General Electric Boiling Water Reactor Generic Reload Fuel Application**, NEDE-24011-P-A-2, July 1981.

F-2. Letter, J. F. Quirk (GE) to P. S. Check (NRC), **ODYN Improvements**, September 25, 1981.

F-3. Letter, J. F. Quirk (GE) to T. P. Speis (NRC), **ODYN Improvements**, October 13, 1981.

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