ROBERT E. DENTON
Vice President
Nuclear Energy

Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657
410 586-2200 Ext. 4455 Local
410 260-4455 Baltimore

BGE

February 24, 1995

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Response to Request for Additional Information Regarding the Use of the Convolution Technique for Main Steam Line Break Analysis

REFERENCES:

(a) Letter from R. E. Denton (BGE) to Document Control Desk (NRC), dated November 1, 1994, Request for Approval to Use Convolution Technique in Main Steam Line Break Analysis

(b) Letter from Daniel G. McDonald (NRC) to R. E. Denton (BGE), dated December 1, 1994, Request for Additional Information Regarding the Use of the Convolution Technique for Main Steam Line Break Analysis - Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M90897 and M90898)

Baltimore Gas and Electric Company hereby responds to your request for additional information (Reference b) regarding our submittal requesting use of the convolution technique in Main Steam Line Break Analysis (Reference a). Our response to each of your questions is included in Attachment (1).

In Reference (a), we indicated that failure to apply the convolution technique to the current Unit 2 fuel cycle could result in small power reductions due to approaching the Technical Specification peaking limits. Since that submittal, the measured power peaking on Unit 2 has reached its maximum for the current cycle and is now decreasing. The measured value approached within 1.7% of the limit (1.608 measured as compared to the limit of 1.635), but no power reductions were required. However, as we point out in our attached response, the risk of power reductions still exists for future fuel cycles.

A001
1/1

Should you have questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,

RED/BDM/dlm

Attachment

cc: D. A. Brune, Esquire
J. E. Silberg, Esquire
L. B. Marsh, NRC
D. G. McDonald, Jr., NRC
T. T. Martin, NRC
P. R. Wilson, NRC
R. I. McLean, DNR
J. H. Walter, PSC

CONVOLUTION QUESTIONS

1. Q. Has the statistical convolution method been approved for other postulated events at Calvert Cliffs?

   A. The statistical convolution method has been approved for the determination of fuel failure due to Departure from Nucleate Boiling (DNB) for the Sheared Shaft/Seized Rotor event. The application of the method of convolution to this event is the example used in Appendix A of Topical Report CENPD-183-A, "Loss of Flow - CE Methods for Loss of Flow Analysis". This Topical Report was generically approved by the NRC for Seized Rotor Events in a Safety Evaluation, dated May 12, 1982. The current Calvert Cliffs Seized Rotor Event analysis, UFSAR Section 14.16, utilizes the statistical convolution method.

2. Q. What are the site boundary doses for 5.6% fuel failure?

   A. The 5.6% fuel failure results from a Pre-Trip Main Steam Line Break (MSLB) with an outside containment break location. The steam generator primary-to-secondary leak rate is limited to 100 gallons per day, per steam generator, as specified by Technical Specification 3.4.6.2.c. For this leak rate, the two-hour site boundary doses are 35 REM Thyroid and 0.2 REM Whole Body.

   Previous calculations of the site boundary doses used the previous Technical Specification limit on primary-to-secondary leak rate of one gallon per minute,. At this leak rate, the two-hour site boundary doses are 263 REM Thyroid and 0.8 REM Whole Body.

   Both these values are less than 10 CFR 100 limits for Limiting Fault accident conditions.

3. Q. What adjustments have been made to the Core Operating Limits Report (COLR) to maintain the MSLB predicted fuel damage fraction to less than 2%?

   A. It was possible to maintain fuel failure to less than 2% by restricting the allowed radial peaks in the COLR. The BASSS setpoints and the Departure from Nucleate Boiling Ratio (DNBR) excore Limiting Condition for Operation (LCO) tent for the operating cycles were based upon a radial peaking limit of 1.7. Lowering the radial peaking limit while not adjusting the DNBR LCOs has the effect of preserving the original thermal margin in the accident analysis, thereby limiting the fuel failure fraction.

   To provide the additional thermal margin, the COLR radial peaking limits (F_{xy}^T and F_r^T) have been lowered from 1.70 to 1.66 for Unit 1 Cycle 12 and 1.635 for Unit 2 Cycle 10.

   The affect of the COLR changes on the Pre-Trip MSLB event may be seen in Figures 3-1 in units of thermal margin to the Specified Acceptable Fuel Design Limits (SAFDL) and in Figure 3-2 in units of DNBR. At time 0 in the "Base Setpoints" case (without the COLR change), the margin between the most limiting pin in the core and the SAFDL is determined by the Required Overpower Margin. The Pre-Trip MSLB event steadily erodes this initial margin until the SAFDL is reached at about 8.5 seconds. The thermal margin continues to degrade, now in violation of the SAFDL, until the point of minimum DNBR occurs at about 11 seconds. The thermal margin then begins to improve as the heat flux decreases following a reactor trip.

The COLR change sets the radial peaking limit to less than the value used to determine the Required Overpower Margin. This has the effect of creating additional margin between the DNBR LCO and the actual core condition. The upper line in Figure 3-1, labeled "Base Setpoints with Reduced Radial Peak," illustrates the minimum thermal margin in the core with the lower radial peaking limit, while the lower line is the thermal margin at which the monitoring system assumes the peak pin is operating. It is seen that the peak pin is actually several percent of thermal margin further from the SAFDL than the conditions assumed by the monitoring system.

The amount of fuel predicted to fail (fuel damage fraction) is dependent upon the extent of violation of the SAFDL of the peak pin (and of the other pins in relation to the peak pin). It is seen that with the lowered radial peak, the extent of the SAFDL violation is reduced.

Figure 3-2 presents similar information in DNBR units, rather than in thermal margin units.

This request for approval for the use of convolution has been made because of a concern that these lowered radial peak limits are close to the actual (measured) radial peak behavior of the core, and measured peaks may exceed these reduced limits in future cycles. As illustrated in Figure 3-3 for Unit 2 Cycle 10, the lowered COLR peaking limit comes quite close to the predicted best estimate radial peaking at around the 3/4 point in burnup.

Should measured peaking exceed the lowered peaking limits in future cycles, a reduction in power will be necessary. The closer the predicted radial peak is to the COLR limit, the greater the likelihood that a power reduction will be necessary.

4. Q. Why is the Low Steam Generator Pressure (LSGP) trip setpoint reached so much sooner in the new MSLB analysis (15.3 sec., Updated Final Safety Analysis Report [UFSAR] Revision 17) than in the previous analysis (33.9 sec., UFSAR, Revision 16)?

   A. The worst case for a Pre-Trip MSLB is found by the consideration of many competing dynamic effects, including the relative timing of the Low Steam Generator Pressure trip and the Power Level - High trip.

   The effect of an increase in break size at a given moderator temperature coefficient (MTC) is a faster power increase and quicker depressurization of the affected steam generator.

   For a given break size (and hence rate of heat extraction) the more negative the MTC, the greater the addition of positive reactivity and the greater rate of power increase. The power at which a Power Level - High trip is reached becomes lower as the MTC becomes more negative since the rate of power increase overwhelms the trip delaying effects of temperature shadowing*.

* "Temperature Shadowing" is the effect on excore neutron detectors due to moderator density changes.

For a given break size, the peak power before a LSGP trip is seen to increase as the MTC becomes more negative. This is because the time to trip (depressurization of the steam generator) is somewhat decoupled from core power. Therefore, for a given time until trip a higher power peak occurs with a more negative MTC.

An analysis is performed which examines the effect of varying the break sizes and MTC values. Figures 4-1 and 4-2 illustrate a representative Pre-Trip MSLB response to two different break sizes, "A" and "B." For each break size, a more negative MTC results in an improved response of the Power Level - High trip seen as a lower peak power. The LSGP trip, however, occurs at higher and higher power levels because the time to depressurize the steam generator and initiate a LSGP trip is nearly constant for a given break size but the peak power increases as the MTC is more negative. Therefore, for a given break size, the limiting condition is the highest of these peak powers that occur before the action of the first of the competing Reactor Protective System trips. The limiting case for the transient as a whole is found by taking the worst predicted response for each of the individual break sizes.

As illustrated in Figures 4-1 and 4-2, the limiting power at time of trip varies only slightly for various worst case combinations of break size and MTC. Since the intersection of the worst case combinations for break size and MTC occurs in a region of the curves that is relatively flat, cycle-by-cycle differences in core parameters can significantly change the limiting break size/MTC point (although these differences have only a minor effect on the limiting power at time of trip).

The outside containment break scenario which previously resulted in the most adverse fuel failure and was included in Revision 16 of the UFSAR was found to be 0.65 Ft^2 and resulted in the LSGP value of 640 PSIA being reached in 33.9 Seconds. The current analysis demonstrates that a 1.0 Ft^2 break size is more limiting for current reloads and results in a more rapid depressurization of the steam generators. Thus the LSGP is reached in only 15.3 seconds in recent analyses. Although the trip times for these two break sizes differed by 15 seconds, the peak power reached was very similar.

5. Q. Recent experiments by the French on reactivity insertion transients indicate that fuel failures occur in high burnup fuel at very low levels of reactivity insertion. What are the peak CAL/GM levels reached when DNB occurs during Pre-Trip and Return-to-Power? What is the peak CAL/GM level reached in high burnup fuel (over 40 GWD/MTU exposure)? What are the transient amplitudes (CAL/GM)?

   A. This request for approval of the use of convolution is for application solely in the Pre-trip MSLB analysis. Convolution is not employed to predict the number of pins in DNB in the Return-to-Power MSLB scenario. For the Return-to-Power MSLB analysis, DNB is assumed when the pin experiences a MacBeth DNBR less than 1.30. The conservative assumption used in the evaluation of fuel failure for this event is that all pins which violate this SAFDL experience DNB and are predicted to fail.

Since this application for approval is limited to the Pre-trip MSLB event, only this event is addressed below.

The fuel temperatures and deposited energy values presented in Table 5-1 are based upon long term operation at the power levels indicated. For the Pre-Trip MSLB, the DNBR crisis is of a few seconds duration being brought on by the coast down of the reactor coolant pumps following reactor trip and the assumed loss of all A.C. power. As reactor power is actually decreasing during the time of minimum DNBR, the steady state assumption is in all likelihood conservative even though the brief reduction in fuel to moderator heat transfer coefficient associated with DNB in the hot pin was ignored. However, the high burnup fuel of interest does not experience DNB, so there is no reduction in the heat transfer coefficient for this fuel.

The Pre-Trip MSLB results in an increase of core power from an initial value of 100% to a peak of just under 140%. Changes in power, fuel temperature and fuel enthalpy are given in Table 5-1 for a high burnup rod (defined here as a thrice burned fuel rod with a rod average burnup of over 40 GWD/MTU, operating at a radial peak of unity) and for the hot rod. The use of a unity radial peak for the high burnup rod is conservative relative to the anticipated rod powers at end of cycle for both Unit 1 Cycle 12 and Unit 2 Cycle 10.

TABLE 5-1

	Initial Conditions			At Time of Peak Power		
	Relative Radial Peak	Fuel Temp °F	Fuel Enthalpy (Cal/Gm)	Peak Relative Power	Fuel Temp °F	Fuel Enthalpy (Cal/Gm)
Thrice Burned Fuel Rod	1.0	924	31.8	1.4	1093	38.7
Peak Pin Fuel Rod	1.7	1243	44.9	2.38	1650	62.2

As illustrated in this table, high burnup fuel normally operates with an enthalpy of about 31.8 cal/gm. The enthalpy of the high burnup fuel slowly increases during the Pre-Trip MSLB event by about 7 cal/gm to a value of 38.7 cal/gm. Note that the high burnup fuel is not predicted to experience DNB.

Although we have responded to the NRC's requests regarding the deposited energy values for the MSLB event, it should be emphasized that the Nuclear Energy Institute has established an Issues Task Force to assess the generic applicability and validity of the limited, foreign experimental data which suggests that the fuel failure criteria for reactivity insertion accidents may be lower for high burnup fuel. At present, we do not have any reason to believe that the high burnup fuel in Calvert Cliffs would fail during an MSLB event.

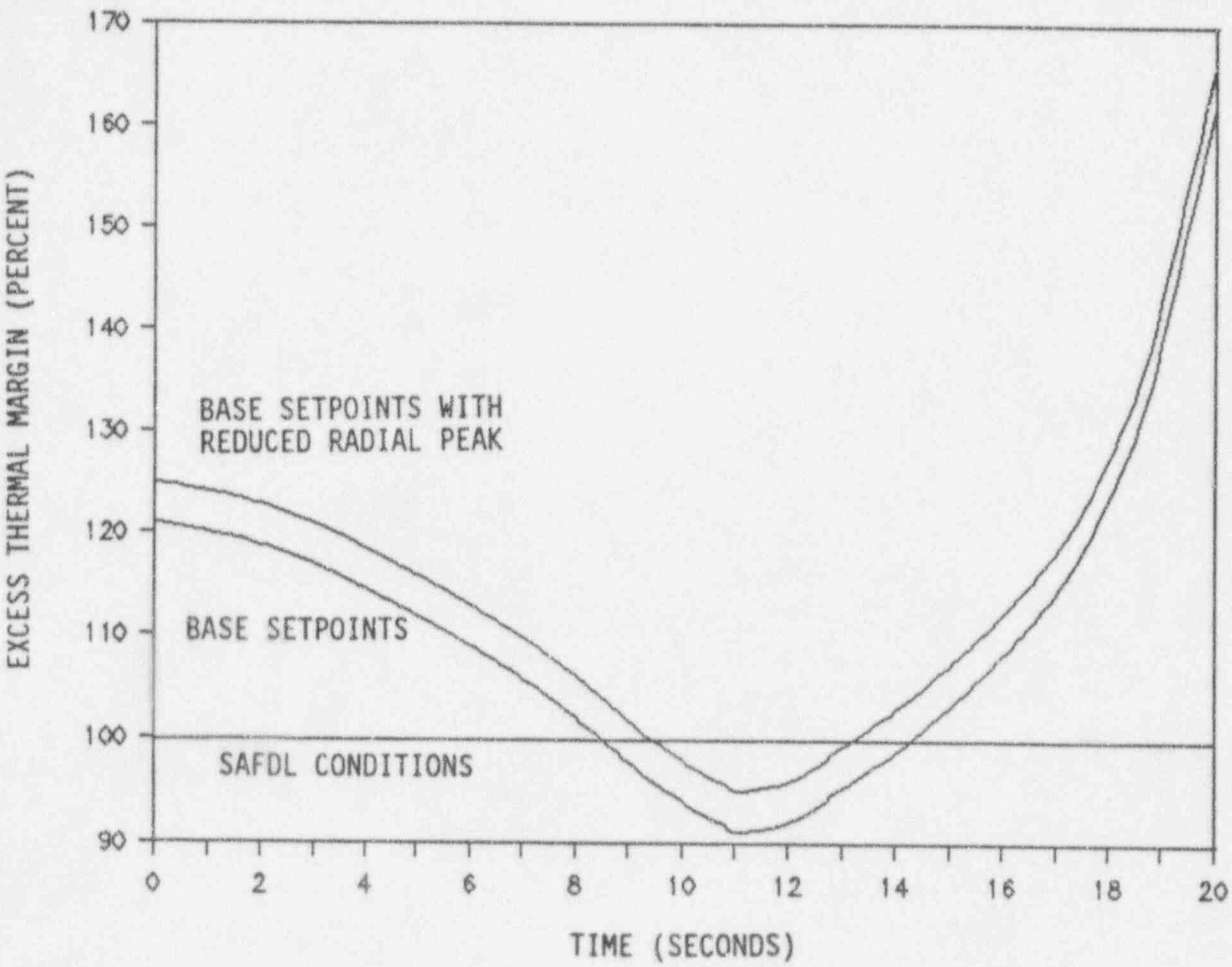
Compared to the rapid insertion of reactivity of a Control Element Assembly (CEA) Ejection, measured in a fraction of a second, the Pre-Trip MSLB goes from initial power to peak power in a comparatively long 17 seconds. The failure mechanism present in the recent French experiments is not expected for the relatively slow power excursions during the Pre-Trip MSLB event.

Specifically, it is our understanding that while the failure mechanism in the foreign test is still uncertain, it is generally believed that the speed of the reactivity insertion rate is a key determinant as to whether failure would be predicted. Design Basis Events with the fastest reactivity insertion rates would represent the most limiting case and would be potentially the most adversely affected by the recent French experiments. In Calvert Cliffs, like other pressurized water reactors, the fastest reactivity insertion event is the CEA ejection accident.

In contrast to the CEA ejection accident where enthalpy additions occur in a fraction of a second, the Pre-Trip MSLB event is characterized by a relatively slow power increase over a period of many seconds.

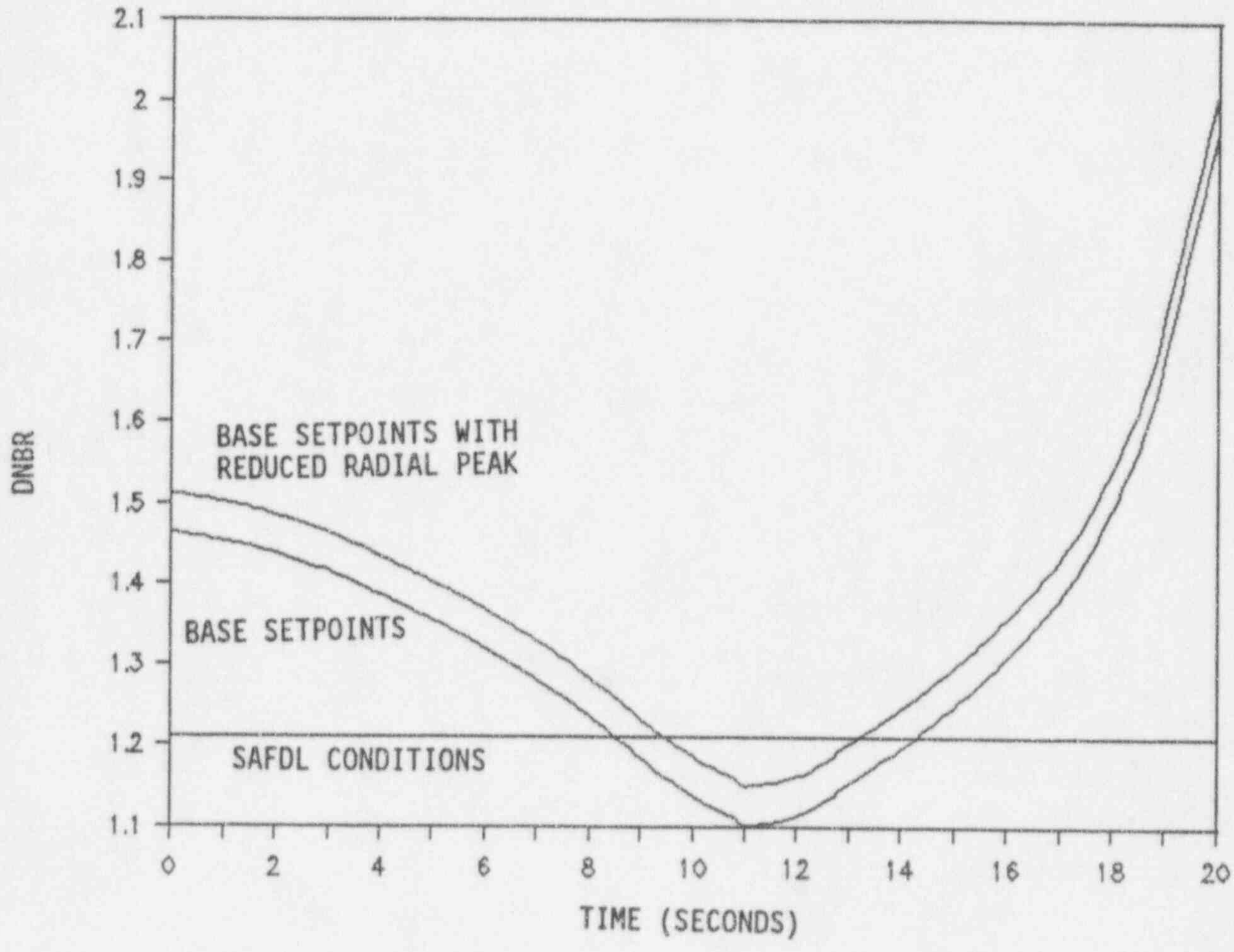
It is further our understanding that the Nuclear Energy Institute Issues Task Force evaluation of the safety significance of the high burnup fuel behavior during the most severe reactivity insertion accidents has been submitted to the NRC. This industry assessment independently came to the same conclusions reached by the NRC's preliminary safety assessment. In particular, the safety assessment concludes that this new data is not an immediate safety significance and that there will be no detrimental impact to the public health and safety.

FIGURE 3-1



170
160
150
140
130
120
110
100
90
EXCESS THERMAL MARGIN (PERCENT)
BASE SETPOINTS WITH
REDUCED RADIAL PEAK
BASE SETPOINTS
SAFDL CONDITIONS
0
2
4
6
8
10
12
14
16
18
20
TIME (SECONDS)

FIGURE 3-2

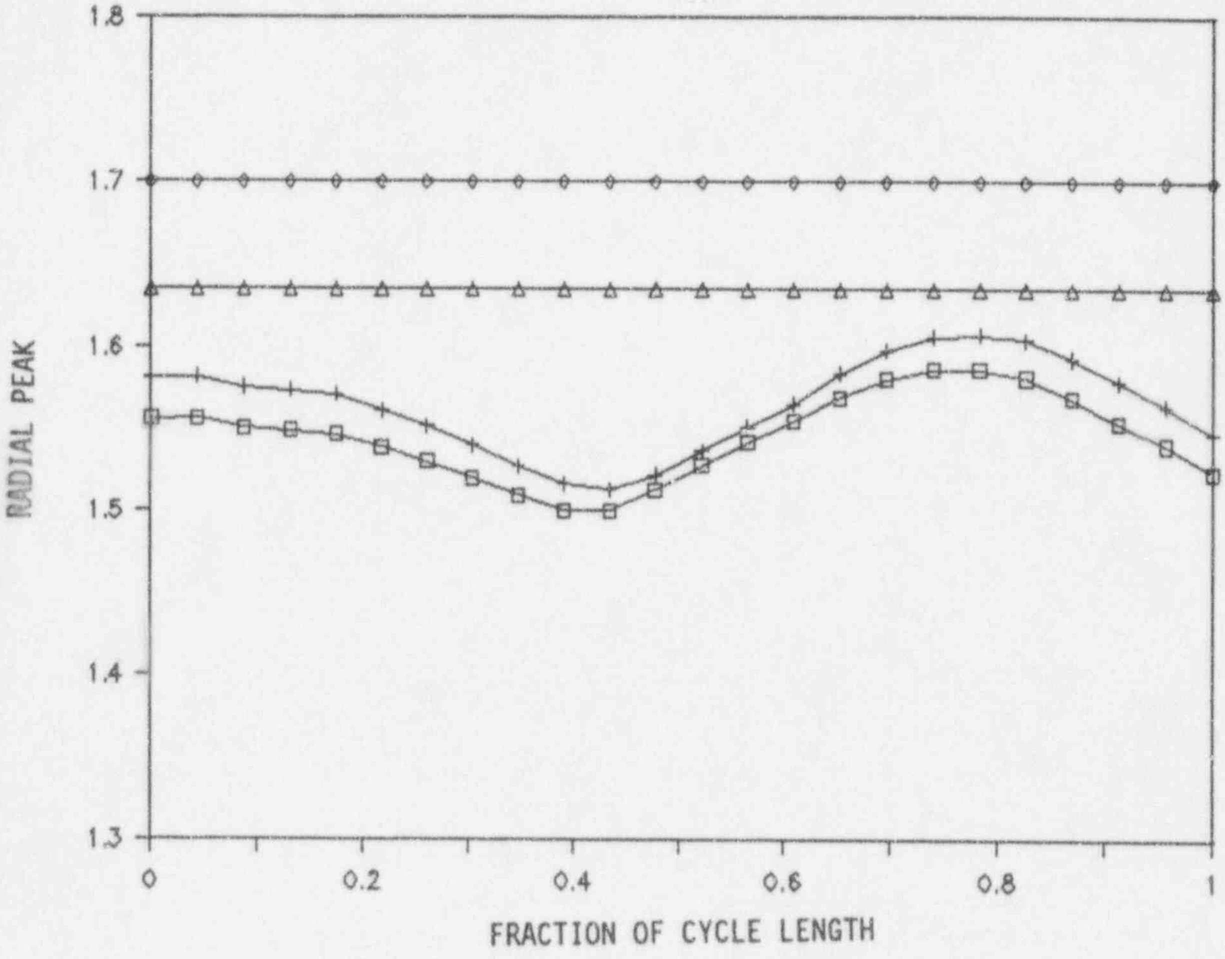


2.1
2
1.9
1.8
1.7
1.6
1.5
1.4
1.3
1.2
1.1
DNBR
BASE SETPOINTS WITH
REDUCED RADIAL PEAK
BASE SETPOINTS
SAFDL CONDITIONS
0
2
4
6
8
10
12
14
16
18
20
TIME (SECONDS)

FIGURE 3-3

UNIT 1 CYCLE 10 RADIAL PEAK VS BURNUP

<45 ACTIVE INCORE DETECTORS



◇ = ORIGINAL UNIT 2 CYCLE 10 FR LIMIT

△ = REDUCED COLR FR LIMIT

+ = PREDICTED FXY VALUES

□ = PREDICTED FR VALUES

FIGURE 4-1

SLB RESPONSE TO BREAK SIZE "A"

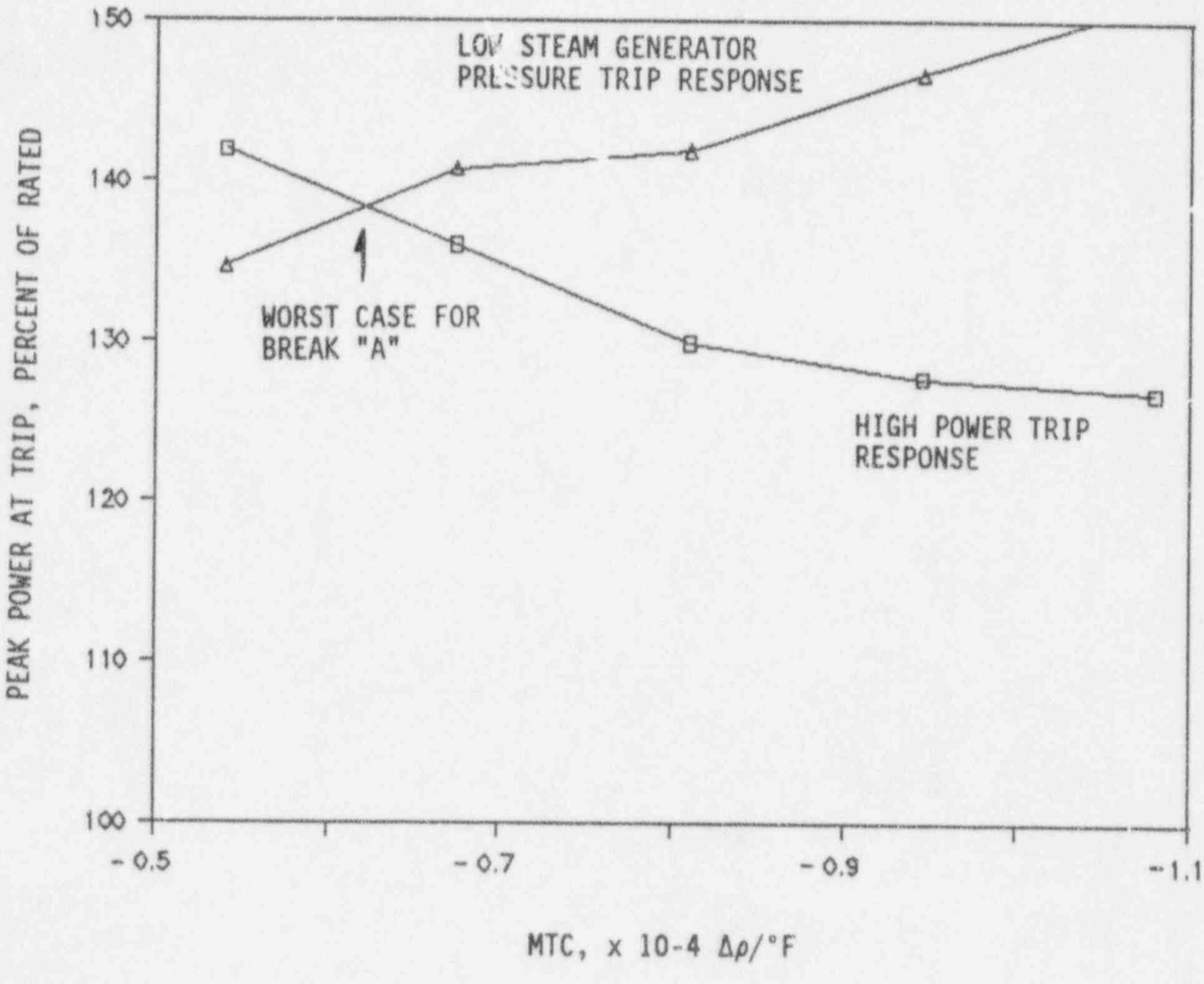


FIGURE 4-2

SLB RESPONSE TO BREAK SIZE "B"