Enclosure 1

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ARTHUR E. LUNDVALL, JR.
VICE PRESIDENT
SUPPLY

May 17, 1983

Director of Nuclear Reactor Regulation
Attention: Mr. R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Calvert Cliffs Nuclear Power Plant
Units Nos. 1 & 2; Dockets Nos. 50-317 and 50-318
Main Steam Line Break Inside Containment

References:

(a) NRC I&E Bulletin 80-04, Main Steam Line Break With Continued Feedwater Addition, dated February 8, 1980.

(b) NRC Letter from R. A. Clark to A. E. Lundvall, dated January 20, 1982.

(c) BG&E Letter from A. E. Lundvall to R. A. Clark, dated November 16, 1982.

(d) BG&E Letter from A. E. Lundvall to R. A. Clark, Sixth Cycle License Application, dated February 17, 1982.

Gentlemen:

On April 30, 1983 BG&E contacted the NRC and reported that a deficiency had been identified in the assumptions used to evaluate the potential effects of a main steam line break inside containment. This letter describes the circumstances that resulted in this licensee event report and constitutes our justification for continued operation.

1. **Description of Analyses**

In response to the concerns raised by I&E Bulletin 80-04 (see References (a), (b) and (c)) BG&E recently performed several main steam line break (MSLB) analyses to evaluate the effects of a failed-open main feedwater regulating valve (MFRV) on peak containment pressure and core reactivity response. The purpose of these analyses was to identify the potential for exceeding containment design pressure or experiencing a return-to-power event, and to provide data that could be used to support any corrective actions that might be deemed necessary.

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The results of these engineering-oriented analyses indicated that with the current feedwater system design, the consequences of a MSLB would be significantly worsened by the assumption of a failed-open MFRV.

On April 29, 1983 the Off-Site Safety Review Committee (OSSRC) reviewed these analytical results. The OSSRC determined that although a failure of the MFRV was not considered in the as-licensed design basis for Calvert Cliffs, an appropriate treatment of this non-safety grade component would have been to disallow any credit for its function. On the basis of this determination, the OSSRC concluded that the main steam line break analysis contained in the FSAR was erroneous and that this issue constituted an unreviewed safety question.

On April 30, 1983 a licensee event report was initiated pursuant to paragraph 6.9.1.8.h of the Technical Specifications to inform the NRC of our conclusions. Efforts were immediately begun to quantify the impact of a MFRV failure on the core reactivity and containment pressure responses to a MSLB for the existing plant configuration. This information was required to support any decision with regard to the safety of continued operations.

An additional MSLB analysis was performed using FSAR methodology to determine the maximum peak containment pressure that would result from runout main feedwater flow to the affected steam generator. This case assumes that a full-size MSLB (guillotine rupture) occurs during full power operations. Other assumptions used in this analysis include:

a. The reactor coolant pumps are not manually tripped upon SIAS as required by the operating procedures;

b. Only half of the containment cooling and containment spray system capacity is available; and

c. The steam release contains 20% moisture carryover.

This analysis yielded a peak containment pressure of approximately 80 psig and a peak temperature of 306°F.

The analysis that was performed to bound the core reactivity response used a methodology similar to that described in Reference (d) with the exception that a 1300gpm auxiliary feedwater flow was assumed to initiate at 180 seconds (and was not isolated), a six-second MSIV closure time was assumed, and a 60-second MFIV closure time was used. Two hot-full-power cases were examined, one assuming a single stuck-out CEA and one assuming that all CEAs scram and both safety injection trains operate.

For the case with the stuck CEA, negative reactivity credit was assumed during return-to-power due to the local heating of the inlet fluid in the hot channel which occurs near the stuck CEA. This credit is based on three dimensional coupled neutronic thermal-hydraulic calculations performed with the HERMITE/PORC Code. As a result of the continued excess auxiliary feedwater flow, this analysis resulted in a peak return-to-power of about 10% at 400 seconds and would show some fuel failures.

For the case where all CEAs scram, the resulting core reactivity is about -0.5% $\Delta\rho$. There is no return to power, and no fuel failures are predicted.

2. **Discussion of Conservatisms**

A significant amount of conservatism is inherent in the analyses discussed above. The principal contributors of this conservatism are summarized below:

Containment Pressure Response

a. The reactor coolant pumps are assumed to continue running throughout the event. A more realistic assumption would be to assume that the pumps are manually tripped on SIAS in accordance with the operating procedures. Continued operation of the pumps results in higher heat transfer in the steam generators and consequently results in a higher peak containment pressure than if the pumps were tripped.

b. The analyses only assume credit for half the containment cooling capacity (coolers and sprays).

c. A delay time of 60 seconds is assumed for the delivery of water to the containment spray header. A more realistic assumption would be a delay time of 30 seconds. Earlier spray delivery would reduce peak containment pressure.

d. The main feedwater isolation valve is assumed to close in 80 seconds upon receipt of a steam generator isolation signal. Experience with surveillance testing of this valve indicates that it will close in 60 seconds, thus reducing the total amount of feedwater introduced to the affected steam generator.

e. Feedwater flow is assumed to continue at runout conditions during the period when the MFIV was closing. A more realistic treatment of feedwater flow would be to include the throttling effect of this valve as it closes. Consideration of this effect would decrease the total flow to the affected steam generator and would yield a lower peak containment pressure.

Core Reactivity Response

a. Item (e) above also applies to the core reactivity response in that any reduction in the amount of feedwater delivered to the affected steam generator will reduce the magnitude of RCS cooldown;

b. A 1300 gpm auxiliary feedwater flow (AFW) far exceeds the current setpoint;

c. The analyses assume a conservative end-of-cycle moderator temperature coefficient;

d. Partial failure of safety injection is assumed;

e. No credit is taken for concentrated boric acid addition from the charging pumps after SIAS;

f. The analyses assume a conservatively low value for boron reactivity worth (-1.0% $\Delta\rho$ per 95 ppm);

g. Auxiliary feedwater temperature is assumed to be $40^{o}F$;

h. No mixing in the reactor vessel inlet plenum is credited (the coldest cold leg temperature is used);

i. The cool-down assumes a 6.3 ft^2 pure steam break;

j. The steam generator is assumed to blow down to atmospheric pressure;

k. Primary-to-secondary heat transfer is not reduced as steam generator level decreases;

l. No operator action is credited to terminate AFW flow; and

m. Less HERMITE credit is taken than is expected to be justifiable.

There are also additional conservatisms of less significance.

3. **Discussion of Potential Consequences**

We have reviewed the potential consequences of a MSLB break inside containment with a failed-open MFRV and have determined that the possible offsite doses would be a small fraction of those allowed by 10 CFR Part 100 for design basis events.

A return-to-power in the core is only predicted to occur if the most reactive CEA is stuck in the withdrawn position. The return-to-power transient would have to be of sufficient magnitude to cause departure from nucleate boiling in order for fuel failures to occur, and those failures would be localized to the vicinity of the stuck-out CEA. Any fission products released from failed fuel pins would be contained within the reactor coolant system. Even assuming that a leakage pathway existed to the containment through the steam generators (as the result of increased steam generator tube leakage) releases to the environment would be insignificant unless a leakage pathway had already been established through the containment. Other minor release pathways may exist through auxiliary systems (e.g. via the auxiliary feedwater turbine steam exhaust) or through the secondary system if it were assumed that the tube leakage occured in the intact steam generator.

It should be noted that the assumptions which were used for the return-to-power analysis, in some cases, are opposite from those used for the containment pressure response analysis and would tend to lessen the severity of the containment response, and vice-versa. For example, in order to achieve a return-to-power condition in the core, it is necessary to assume that the reactor operator trips the reactor coolant pumps upon SIAS actuation in accordance with the emergency procedures or that the pumps are tripped as a result of a loss of off-site power. This assumption results in the most severe cold-edge temperatures following a MSLB. In the case of the containment analysis, however, a trip of the reactor coolant pumps upon SIAS reduces the peak pressure by up to 10 psig as a result of the reduced heat transfer rate in the affected steam generator.

As previously noted, the peak containment temperature was calculated to be 306°F. Although this exceeds the existing design limit of 276°F, the duration of the temperature peak is too short to adversely affect the operation of safety-related equipment located inside containment or to degrade containment structural integrity.

The containments for Calvert Cliffs were designed to an internal accident pressure of 50 psig. The containments were tested to 57.5 psig during structural integrity tests. The containments were further evaluated for a load combination which includes a 1.5 load factor on the design pressure. Therefore, it can be concluded that the containments are adequate for an internal pressure of at least 75 psig with sufficient margin as required by the code.

In further studying the capacity of the containments beyond the design conditions, it is evident that additional margins are provided in the code allowables and the actual material strengths. Although actual material strengths have not been tabulated accurately for Calvert Cliffs, it is reasonable to assume at least 20% higher material strengths were provided based on records and past experience. Together with a margin of 10% provided in the code allowable, a total of 30% margin over the pressure of 75 psig can be expected. This amounts to a pressure of 97.5 psig. The highest stressed section under this condition is at the cylinder-base junction.

Studies indicate that, with regard to containment structural integrity at elevated pressures, electrical penetrations are the limiting components. The Amphenol canister electrical penetrations used at Calvert Cliffs have been tested at 62 psig. Conax penetrations have been tested for as high as 100 psig.

To bound all possible scenarios, we have considered the potential consequences that would be associated with the following concurrent events:

a. Localized fuel failures due to a return-to-power transient;

b. Primary-to-secondary leakage as the result of increased steam generator tube leakage; and

c. Loss of containment integrity due to a failed penetration.

The radiological source term for this scenario would be limited by the fact that fuel failures in the core would be restricted to the vicinity of the stuck-out CEA. This source term would be less than that assumed for the maximum hypothetical event. Transport of these radionuclides to the containment would be dependent upon the size of the primary-to-secondary leak and would be hindered by the absence of forced reactor coolant flow (RCPs tripped upon SIAS). Consequently, only a small fraction of the total radioactivity released from the fuel would be introduced into the containment. Of this, it is expected that only a small fraction would ultimately be released to the environment. This is due to the fact that, unlike in the case of a LOCA, elevated pressures in the containment would exist for a relatively short period of time. By the time any appreciable amounts of radioactivity had been introduced into the containment, the containment sprays and coolers would have reduced the internal containment pressure, and little driving force would remain to effect leakage across the failed penetration.

Consequently, the maximum offsite doses associated with the scenario described above would be well below 10 CFR Part 100 limits and are bounded by the accident analyses described in the FSAR.

4. **Evaluation of Probability of Occurrence**

To evaluate the likelihood of the event sequence in question, we reviewed the probabilistic data base that has been assembled to support the Interim Reliability Evaluation Program (IREP). In that data base (adapted from EGG-EA-5887), the probability of a pipe rupture for pipes larger than three inches in diameter is identified as 1E-10 per hr per pipe section. If we conservatively assume 60 sections of main steam pipe inside each containment, this yields an overall probability of about 5E-5 per reactor year of operation. The probability assigned to the failure of an air-operated valve to close upon demand is 3E-4. Thus, the overall probability of occurrence for a major pipe rupture such as a MSLB concurrent with a failed-open valve (such as the MFRV) can be placed at approximately 2E-8 (plus or minus one or two orders of magitude). This is substantially less than the 3E-4 value that is normally assigned to the probability of a large break loss-of-coolant accident (LOCA).

It should be noted that the overall probability discussed above applies only to a scenario that could lead to exceeding the containment design pressure and does not consider the magnitude of the pressure response, nor does it include the likelihood of a subsequent loss of containment integrity.

To assess the overall probability of a sequence of events that could result in a return-to-power in the core, the likelihood of a failure of a single CEA to insert must be included. The IREP data base provides a value of 3E-5 per demand for this failure probability. Thus, the overall probability of a MSLB that results in a rapid RCS cooldown and subsequent return-to-power can be estimated to be on the order of 1E-12 per reactor year of operation.

5. **Corrective Measures**

Based on the results of engineering analyses performed in response to Bulletin 80-04 concerns, BG&E is proceeding with feedwater system modifications that will provide at least two barriers to the continued addition of feedwater to the affected steam generator after a MSLB. Two modifications are currently being pursued, either of which should result in an acceptable peak containment pressure and core reactivity response.

The first modification includes an automatic trip of the feedwater system (main feed pumps, heater drain pumps, and condensate booster pumps) on high containment pressure.

The second modification involves decreasing the closure time for the MFIV to about 15 seconds. This modification requires installation of a new valve actuator or possibly replacement of the entire valve.

Current analyses indicate that the calculated peak containment pressure is very sensitive to the amount of feedwater that is introduced to the affected steam generator within the first minute of the event. Under calculated runout conditions, complete isolation must occur in less than 30 seconds. Under loss-of-forced-flow conditions; i.e., feedwater system pumps tripped, two-phase expansion (flashing) of water in the feed system piping will result in some continued flow to the affected steam generator.

The feedwater expansion phenomenon was conservatively modeled and included in the analysis. The results of the analysis indicate that, without rapid MFIV closure, the feedwater system must be tripped promptly to limit the total flow delivered to the affected steam generator. Although an automatic feature will be required to ensure that this function is properly performed in the event of a MSLB, we have temporarily modified the plant operating procedures to require a manual trip of the feedwater system upon receipt of a Steam Generator Isolation Signal (SGIS).

Our schedule for implementation of the feedwater system modifications discussed above is as follows:

- Complete MSLB analyses to support engineering validity of both proposed modifications (for submittal to NRC).	May 17, 1983
- Complete licensing grade analyses of MSLB to support each of the proposed (tentative) modifications (for submittal to NRC).	June 1, 1983
- Complete installation of automatic feedwater system trip.	Unit 1 - Prior to start-up after the fall 1983 refueling outage Unit 2 - Nov. 17, 1983 (tentative, subject to equipment delivery)
- Complete MFIV modification	Unit 1 - Spring 1985 refueling outage Unit 2 - Fall 1985 refueling outage (tentative, subject to equipment delivery)

6. **Conclusions**

The POSRC and OSSRC have reviewed the information provided above and have concluded that the continued operation of Calvert Cliffs Units 1 & 2 does not constitute a threat to the health and safety of the public. This conclusion is supported by the following considerations:

a. The analyses that were performed to evaluate the effects of a MSLB concurrent with a failed-open MFRV were conservative;

b. The containment structure should be capable of withstanding peak pressures higher than the maximum calculated value of 80 psig by virtue of the safety margins incorporated into its design. Taking credit for load factors and code allowables, the Calvert Cliffs containment is designed to withstand 97.5 psig. The electrical penetrations have been tested to 62 psig; however, other penetrations of similar design have been successfully tested to pressures as high as 100 psig. Assuming a leak developed in an electrical penetration, the leak would discharge into the electrical penetration room. Air exhausted from this room would pass through particulate and charcoal filters before being discharged to the environment.

c. In the event of a return-to-power transient which ultimately resulted in fuel failures, a pathway would not likely exist for the subsequent release of radionuclides to the environment;

d. The overall probability for a MSLB which could cause fuel failures is very low (about 1E-12 per year); and finally

e. Analytical and engineering work is proceeding on an expedited basis to implement appropriate corrective design measures.

If you should have any questions, please do not hesitate to contact us.

Sincerely,

CH Poindexter

for A. E. Lundvall, Jr.

AEL/BSM/pdy

cc: Messrs. J. A. Biddison, Jr., Esq.
G. F. Trowbridge, Esq.
D. H. Jaffe, NRC
R. E. Architzel, NRC
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RESPONSE TO NRC QUESTIONS*

1. Under normal operating conditions, what is the closure time for the main feedwater regulating valve on reactor scram? Describe the experience at Calvert Cliffs concerning the reliability of the feedwater regulating valves. Can the reliability of the main feedwater regulating valves be improved by inspection or testing?

   Response

   The MFRV will close within twenty seconds. As the MFRV closes, the bypass valve opens to allow 5% flow. The result is a ramp-down of feedwater flow to the five percent value. Although MFRV closure time is not documented by test, the bypass valve has been shown to open within 4 to 5 seconds following receipt of a reactor trip signal. The test procedure for the bypass valve includes a step to verify closure of the MFRV. According to maintenance personnel familiar with this test procedure, the MFRV closes in roughly the same time frame required for bypass valve opening.

   The MFRVs must perform their isolation function on demand for each reactor trip. In addition, each MFRV is tested again before the unit is restarted after a trip. Given that there are two MFRVs per unit (one for each steam generator), and that there have been a total of 241 trips between Units 1 and 2, the total number of MFRV challenges recorded to date is 2 x 2 x 241 = 964.

   There have been a total of six MFRV failures. Four of these failures involved the MFRV going shut as the result of controller problems. In the remaining two cases, the MFRV failed as is (open) due to a loss of instrument air.

   This information suggests a failure rate (MFRV failed open) of approximately $2x10^{-3}$ per demand based upon actual operating experience at Calvert Cliffs. Given this low failure rate, it is not expected that MFRV reliability can be enhanced by further inspection or testing.

2. Is the 20% moisture carryover a standard assumption?

   Response

   A 20% moisture carryover through the ruptured main steam line was used in the analyses which are presented in the Calvert Cliffs FSAR. It is our understanding that this assumption was typical for Combustion Engineering plants of the Calvert Cliffs vintage.

3. What are the Main Feedwater Isolation Valve (MFIV) flow characteristics?

*Forwarded by NRC letter from R. A. Clark to A. E. Lundvall, Jr., dated May 10, 1983.

Response

According to the valve manufacturer (Velan), the flow characteristics of the MFIV are closely represented by the attached Cv versus lift curve.

4. Provide the assumptions used to determine the primary system temperature response and the assumptions used to model heat transfer from the primary system to the liquid and vapor regions of the steam generators.

   Provide the energy balance for the containment atmosphere based on the energy available from a main steam line break.

   Response

   The SGN III computer code was used (with modifications) to calculate the mass/energy release from the steam generator for use in the containment pressure analysis. The primary side thermal hydraulic models used by SGN III are based on the governing mass-energy-volume conservation equations. Core power is determined in a point-kinetics model augmented with a safety injection system model for reactivity control during the core cooldown which follows a steam line break event.

   The SGN III code is NRC-approved and is described in detail in Appendix 6B of CESSAR, dated December 19, 1973 (as amended).

   The Combustion Engineering CONTRANS code was used to calculate the containment response. This code is described in the Combustion Engineering report CENPD-140, "Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis," April 1974.

5. A 20% increase in material strengths was assumed in the May 4, 1983 submittal. Provide a statistical evaluation of the actual strengths of materials used in the construction of the Calvert Cliffs containment and utilized in the evaluation of the design basis capability of the containment. In addition, a 10% margin in the code allowables was assumed. Provide the justification for use of this margin to increase the containment pressure, since the resulting stress should meet the design basis criteria.

   Response

   The certified material test reports (CMTRs) for the rebar and concrete used in the construction of the Calvert Cliffs containments have been retrieved from our quality control records and are now undergoing statistical evaluation by a contractor. We have been informed that the results of this evaluation will be available in time for inclusion in our followup submittal scheduled for June 1, 1983.

In our letter of May 4, 1983 we stated that margins in containment capacity are available beyond the design conditions. The intent of that letter was to identify in a simple but admittedly conservative manner, a more realistic capacity of the containment structure due to internal pressure loads. The "10% margin in code allowable" is taken from the design basis criteria, which limits the stress to no more than 90% of the minimum specified yield stress. This is an arbitrarily established limit that, when reached, does not represent actual containment capacity. Rather, the yield stress is a more critical parameter. Adjustments for this artificial limit, which provides approximately a 10% margin, constitute a portion of actual containment capacity beyond the design condition.

The controlling design condition at the cylinder base juncture is tension in the reinforcing steel.

Attachment to Enclosure 2

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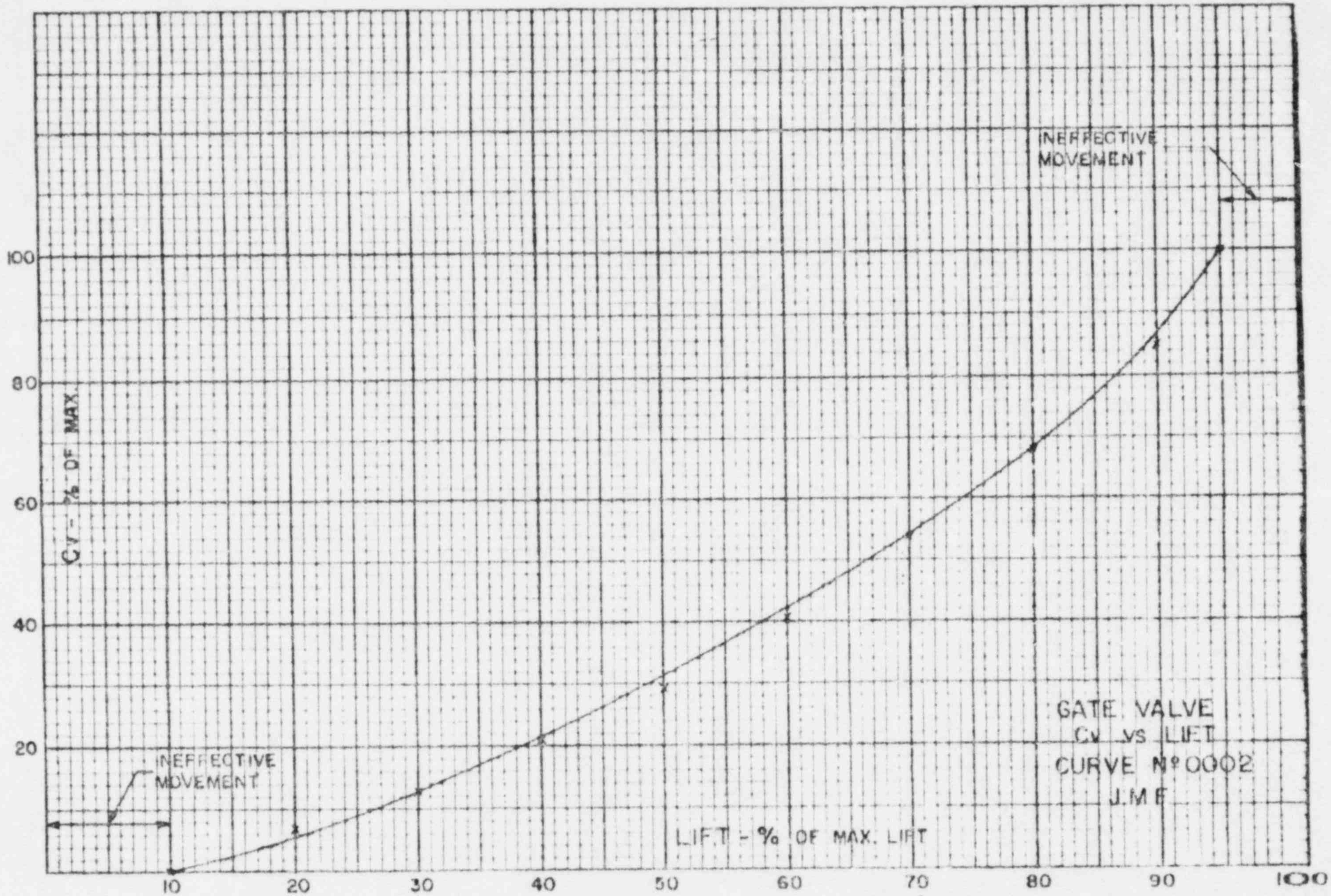


MAIN STEAM LINE BREAK - CONTAINMENT RESPONSE

Methodology

The MSLB containment response was calculated using methods similar to the circa 1972 methodology of the licensing analysis reported in the Calvert Cliffs FSAR. This analytical methodology assumes a large ($6.3 ft^2$) guillotine MSLB, with the mass/energy release modeled as pure steam (less credit for 20% moisture carryover) throughout the blowdown.

The analysis is based upon full load initial conditions, whereas the 1972 FSAR analysis predicted no load as the worst case (due to the higher initial steam generator water inventory). The full load case is limiting because the effects of runout MFW flow due to a failed-open MFRV are most adverse from full load conditions. At no load conditions, initial main feedwater (MFW) flow is negligible.

The SGN III computer code was used to calculate the mass/energy release. A steam bubble rise factor of 100 was used to obtain a pure steam release, and the blowdown from the pure steam release was reduced by 20% to credit moisture carryover. The blowdown results were coupled to the CONTRANS computer code to calculate the containment response.

Assumptions

The containment response analysis assumptions are listed in the Attachment to this enclosure. A simplistic first approximation of 105% of full MFW flowrate, for 6.0 seconds, was assumed. The flashing of MFW ($1,771 ft^3$ at $436^{\circ}F$) which would result from the depressurizing steam generator was included in a conservative fashion assuming isentropic expansion. In addition, 4000 lbm of steam was included to account for the steam line inventory between the ruptured steam generator and its isolation valve. A reactor trip on high containment pressure actuation (at 4.75 psig) was credited. Credit was taken for both containment spray trains and all four containment atmosphere coolers. It should be noted that the assumption of full containment sprays does not have a significant impact on peak containment pressure due to the assumed delay time of 60 seconds for spray actuation. The analyses that will be provided in our follow-up submittal scheduled for June 1 will assume the worst single failure in the containment spray/cooler systems. It is expected that the earlier spray actuation discussed inthe transmittal letter will more than compensate for the effects of the worst single failure. It was assumed that the reactor coolant pumps remain running throughout the event, which is consistent with the assumption that off-site power remains available. A loss of off-site power would trip the MFW train and the reactor coolant pumps, and would therefore minimize both the mass/energy added by MFW and the primary to secondary energy transfer. Such an assumption would have resulted in a less adverse containment response.

Results

The calculated peak containment pressure and temperature was 53 psig and $278^{\circ}F$, respectively, at 60 seconds into the event.

MSLB CONTAINMENT RESPONSE

ASSUMPTIONS

INITIAL POWER LEVEL -	100%
INITIAL CONTAINMENT PRESSURE -	1.8 psig
INITIAL CONTAINMENT TEMPERATURE -	120°F
INITIAL CONTAINMENT HUMIDITY -	50%
INITIAL STEAM GENERATOR LEVEL -	35.06' above tubesheet (Normal level)
INITIAL STEAM GENERATOR INVENTORY -	141,425 LBM (Adjusted 4.0%)
BREAK TYPE -	6.3 Ft.^{2} Gillotine at SG nozzle
PARTITION MODEL -	Instantaneous Mixing
MASS/ENERGY RELEASE -	20% Moisture Carryover - Ruptured SG
STEAM SEPARATION RATE MULTIPLIER -	100
REATOR TRIP -	On high containment pressure @ 4.75 psig
DECAY HEAT -	Decay heat curve after reactor trip
MFIV -	0.9 sec. delay + 80 sec. step closure
MSIV -	0.9 sec. delay + 6.0 sec. linear closure
SG ISOLATION LOGIC -	On high containment pressure (4.75 psig), or low SG pressure (548 psia), whichever occurs first
STEAM HEADER K FACTOR -	17. (Based on 5.585 ft^2 pipe)
REACTOR COOLANT PUMP -	Remain on
MAIN FEEDWATER FLOW -	105% for 6.0 seconds @ 436°F
AUXILIARY FEEDWATER -	0
WATER IN FEED PIPE (FLASHING) -	1771 Ft^3 @ 436°F
STEAM IN HEADER PIPE -	4,000 LBM
CONTAINMENT FAN COOLERS -	4 coolers on @ 11 sec (26900 BTU/sec each)
CONTAINMENT SPRAYS -	2 spray trains (1350 gpm ea., 110°F at 4.75 psig + 60 sec)
CONTAINMENT HEAT SINKS -	Condensation per Uchida Correlation

RESULTS

TIME OF REACTOR TRIP	0.88 sec. on high containment pressure
TIME OF SGIS -	0.88 sec. on high containment pressure
PEAK PRESSURE/TIME -	53 psig @ 60 sec.
PEAK TEMPERATURE/TIME - (NO 2ND PEAK)	278 °F @ 60 sec.

MAIN STEAM LINE BREAK - CORE RESPONSE

Introduction

The purpose of this analysis is to provide additional information to the NRC concerning the reactivity increase and potential return to power following a main steam line break event in the Calvert Cliffs plants, as requested in I&E Bulletin 80-04. The steam line break event was analyzed assuming the worst single failure of a main feedwater (MFW) component as well as the assumption of the most restrictive single active failure in the safety injection system. The event was initiated from hot power (HFP) and assumed Loss of AC power (LOAC) on reactor-turbine trip. The analysis incorporates the safety grade Auxiliary Feedwater Actuation System (AFAS) which has been installed in Unit 2.

Analysis Assumptions and Initial Conditions

The steam line break event was analyzed to determine the impact of runout main feedwater flow (due to the failure of the MFW regulating valve to the affected steam generator to close on a reactor trip signal). The failure of the feedwater regulating valve on the affected steam generator is determined to be the worst single active failure of a MFW component since the regulating valves closure time are shorter than closure times for the main feed isolation valves (MFIVs).

The MFW flow data used in this analysis were adapted from earlier MSLB cases which included a conservative treatment of feedwater flashing after coastdown of the MFW system pumps. Essentially all of the water contained in the feedwater lines between the main feed pumps and the steam generators is assumed to be available for flashing into the damaged steam generator. This results in an additional 112, 000 lbm which was assumed to enter the damaged steam generator at a rate of 3270 lbm/sec for 34.25 seconds following coastdown of the MFW system pumps. The delay time used for the MFW train trip was conservative in that the trip was assumed to occur at about 1.8 seconds. With the planned modifications the trip would actually occur at about 0.88 seconds on containment high pressure.

The analysis conservatively assumed that on a safety injection actuation signal (SIAS), only one high pressure safety injection (HPSI) pump starts. This assumption is based on the Technical Specification requirements on ECCS subsystems. It requires that two independent ECCS subsystems be operable with each subsystem comprised of one operable HPSI, one operable LPSI and an operable flow path capable of taking suction from the refueling water tank on a SIAS. Crediting only one operable HPSI pump is based on the assumption of a most restrictive single active failure which would lead to unavailability of one independent ECCS subsystem. It should be noted that the above assumption is consistent with the previous steam line break analyses performed for Calvert Cliffs.

A safety injection actuation setpoint of 1645.0 psia was assumed in the analysis. This represents a Technical Specification setpoint of 1725.0 psia and an uncertainty of 80.0 psia. In addition, a maximum time delay of 30 seconds for HPSI pumps to accelerate to full speed was assumed in the analysis. Because of LOAC power, additional time delays are included in the analysis. It includes 10.0 seconds for the diesel generators to start and reach speed following the LOAC and 5.0 seconds for the HPSI pump to be loaded on line regardless of which sequencer (i.e., shutdown or LOCA) is initiated.

The HFP SLB event was initiated from the conditions in Table 1. The Moderator Temperature Coefficient (MTC) of reactivity assumed in the analysis corresponds to end of life, since this MTC results in the greatest positive reactivity insertion during the RCS cooldown caused by the Steam Line Rupture. Since the reactivity change associated with moderator feedback varies significantly over the moderator density covered in the analysis, a curve of reactivity insertion versus density rather than a single valve of MTC is assumed in the analysis. The moderator cooldown curve asssumed in the analysis is given in Figure 1.

The reactivity defect associated with the fuel temperature decrease is also based on an end of life Doppler defect. The Doppler defect based on an end of life Fuel Temperature Coefficient (FTC), in conjunction with the decreasing fuel temperatures, caused the greatest positive reactivity insertion after the MSLB event. The uncertainly on the FTC assumed in the analysis is given in Table 1. The β fraction assumed is the maximum absolute value including uncertainties for end of life conditions. This too is conservative since it maximizes subcritical multiplication and thus, enhances the potential for Return-To-Power (R-T-P). The analysis also assumed a conservatively low value of boron reactivity worth of -1.0%$\Delta\rho$ per 95 PPM for safety injection flow from the High and Low Pressure Safety Injection pumps.

The maximum CEA scram worth available following the trip is 6.89%$\Delta\rho$. This available scram worth was calculated for the stuck rod which produced the moderator cooldown curve in Figure 1. A negative reativity credit was assumed during a return to power in the analysis. This negative reactivity credit is due to the local heatup of the inlet fluid in the hot channel, which occurs near the location of the stuck CEA. This credit is based on the three dimensional coupled neutronic-thermal hydraulic calculations performed with the HERMITE/TORC code for St. Lucie Unit 2 Cycle 1. A conservative comparison of Calvert Cliffs and St. Lucie 2 demonstrated that Calvert Cliffs HERMITE credits are larger than the reactivity credits obtained for St. Lucie 2.

The analysis only credited the low steam generator pressure trip. An analysis trip setpoint of 600.0 psia was assumed. This represents the Technical Specification setpoint of 685.0 psia and an uncertainty of 85.0 psia. The analysis also assumed that a Stea Generator Isolation Signal (SGIS) is generated when secondary pressure reaches 600.0 ps and an uncertainly of 85.0 psia. A Main Steam Isolation Valve (MSIV) closure time of [illegible] seconds (includes valve closure time and signal processing delay time) consistent with the Technical Specification value was assumed in the analysis.

The analysis assumptions regarding the auxiliary feedwater (AFW) actuation analysis setpoint, the associated time delays, and the AFW flow through each leg are given below. They were conservatively chosen to initiate AFW flow sooner and deliver the maximum AFW flow to the ruptured steam generator, which maximizes the primary cooldown and enhances the potential R-T-P.

The AFW actuation setpoint was assumed to be reached at 7.2 seconds. This is the time that a setpoint of 78.1% of steam generator wide range span is reached. An AFAS setpoint of 78.1% represents a Technical Specification actuation setpoint of 54.4% and includes a 23.7% uncertainty. This large uncertainty is assumed to account for the large differences, at full power, between steam generator wide range and narrow range level indication. The actuation signal activates a motor driven AFW pump and a steam driven AFW pump which deliver AFW to both steam generators. The delay time involved for the motor driven pump includes 10 seconds for the diesel generators to start and reach speed following the LOAC and 15.0 seconds for the motor driven AFW pump to be loaded on line if the shutdown sequencer is initiated. A 30.0 second time delay is assumed for the motor driven AFW pump to be loaded on line if the LOCA sequencer is initiated. The LOCA sequencer is initiated when SIAS is generated. The flow from the motor driven pump to each steam generator is controlled by a flow control valve installed in the "leg" connecting the pump to the steam generator. A maximum flow of 217 gpm through each leg is conservatively assumed in the analysis. It represents the Technical Specification limit on AFW flow rate of 160 gpm through the flow control valve and an uncertainty of 57 gpm. The anlaysis conservatively assumed that the AFW flow legs are filled with water and, thus, no time delay associated with AFW flow through the piping was included in the analysis. It should be noted that the amount of time which AFW can reach the ruptured steam generator is so short that assuming higher feedwater flows (e.g. 1300 gpm) would not impact the results of the transient. The steam driven pump's AFW reaches the steam generator 9.5 seconds after the AFW actuation setpoint is reached. This includes a minimum time delay of 5.0 seconds required to open steam admission valves to the AFW pump, and 4.5 seconds for the pump to accelerate to speed. The analysis conservatively assumed that the AFW flow legs are filled with water and, thus, no time delay associated with AFW flow through the piping was included in the analysis. The flow from the steam driven pump to each steam generator is also controlled by a flow control valve installed in the flow "leg" connecting the pump to the steam generator. A minimum flow of 217 gpm through each leg is assumed in the analysis. It represents the Technical Specification limit on AFW flow rate of 160 gpm through the flow control valve and an uncertainty of 57 gpm.

The analysis also included isolation of the ruptured steam generator when the steam generator differential pressure reaches the analysis setpoint of 250.0 psid. The repsents a Technical Specification setpoint of 130.0 psid and an uncertainty of 120.0 psid. In addition, a 20.0 second time delay was assumed in the analysis to close the AFW isolation (i.e., block) valves. These assumptions are conservative since it delays the isolation of AFW to the ruptured steam generator.

Results

The SLB event with LOAC power on turbine trip is presented here. This case maximizes the moderator reactivity insertion, therefore maximizing the potential for the post trip return to power and consequent lower DNBRs. This occurs because LOAC power causes the Reactor Coolant Pumps (RCPs) to coastdown. The decreasing coolant flow is assumed to result in no flow mixing at the core inlet plenum. Thus, cold edge temperatures were used to calculate the moderator reactivity insertion.

The sequence of events for the 6.305 ft^2 SLB with LOAC on turbine trip initiated from HFP conditions is given in Table 2. The reactivity insertion as a function of time is presented in Figure 2. The NSSS responses during the transient are given in Figures 3 through 8.

The results of the analysis show that the HFP SLB causes the secondary pressure to rapidly decrease until a reactor trip on low steam generator pressure is initiated at 1.8 seconds. The CEAs drop into the core at 3.2 seconds and terminate the power and heat flux increases.

The Steam Generator Isolation Analysis Setpoint is reached at 1.8 seconds. At 2.7 seconds, the MSIVs begin to close and are completely closed at 8.7 seconds. The blowdown from the intact steam generator is terminated at this time.

A LOAC power on turbine trip is assumed to occur at 3.2 seconds. At this time, RCPs start coasting down and the diesel generators start coming on line. At 13.2 seconds, the diesel generators reach full speed and shutdown sequencer is initiated to load emergency systems. At 21.8 seconds the safety injection actuation signal is on and diesel generators switch from shutdown sequencer to LOCA sequencer to load emergency systems. At 26.8 seconds HPSI pump is loaded on line and at 56.8 seconds the HPSI pump reaches full speed.

An AFW isolation signal based on steam generator differential pressure is initiated at 2.9 seconds. At 22.9 seconds, the AFW block valve associated with the steam generator with lowest pressure (i.e., ruptured steam generator) is completely closed.

At 7.2 seconds, an AFAS is assumed based on low steam generator level. The steam admission valve to the AFW pump is opened at 12.2 seconds and the steam driven AFW pump reaches full speed and delivers AFW flow to both steam generators at 16.7 seconds. At 22.9 seconds, AFW to the affected steam generator is terminated due to closing of its AFW block valve. The motor driven AFW pump is loaded on line by diesel generators at 51.8 seconds and is assumed to reach full speed and deliver AFW flow to the intact steam generator instantaneously.

The continued blowdown from the ruptured steam generator causes the core reactivity to increase. The ruptured steam generator blows dry at 265.4 seconds, which terminates the cooldown of the RCS. A peak reactivity of .41% $\Delta\rho$ at 312.0 seconds is obtained. A return to power of 7.09% of 2700 MWt occurs. Mcbeth DNBR values calculated remain above the 1.30 limit and therefore, it is concluded that critical heat fluxes are not exceeded. The decrease in moderator reactivity as well as negative reactivity inserted due to boron injection via the HPSI pump terminate the approach to criticality and the core becomes more subcritical.

Conclusions

The results of the steam line break event analyzed above show that the critical heat fluxes are not exceeded. It is therefore concluded that the consequences of a steam line break event with the worst single failure of a main feedwater component as well as the most restrictive single active failure in the safety injection system are within the criteria set for the event.

TABLE 1

KEY PARAMETERS ASSUMED IN THE
STEAM LINE BREAK EVENT INITIATED FROM HFP

Parameter	Units	Value
Initial Core Power	MWt	2754.0
Initial Core Inlet Temperature	^oF	550.0
Initial RCS Pressure	psia	2300.0
Initial Steam Generator Pressure Analysis Trip Setpoint	psia	860.0
Auxiliary Feedwater Actuation Analysis Setpoint	% Wide Range Steam Generator Level Indication	78.1
Steam Generator Differential Pressure Analysis Setpoint	psid	250.0
Safety Injection Actuation Signal	psia	1645.0
Minimum CEA Worth Available at Trip	$\% \Delta\rho$	-6.89
Doppler Multilier		1.15
Moderator Cooldown Curve	% vs. density	See Fig. 1
Inverse Boron Work	$PPM/\% \Delta\rho$	95.0
Effective MTC	$x10^{-4}/^oF$	-2.2
β Fraction (including uncertainty)		.0060

TABLE 2

SEQUENCE OF EVENTS FOR STEAM LINE BRAK EVENT WITH LOSS OF AC POWER ON TURBINE TRIP INITIATED FROM HFP

Time (sec)	Event	Setpoint or Value
0.0	Steam Line Break Runout Main Feedwater Flow is Initiated to Steam Generators	6.305 ft^2
1.8	Low Steam Generator Pressure Analysis Trip Setpoint is Reached; Steam Generator Isolation Analysis Setpoint is Reached	600.0 psia
2.7	Trip Breakers Open; Main Steam Isolation Valves Begin to Close; Main Feedwater Pump Coastdown Begins	—
2.9	Steam Generator Differential Pressure Analysis Setpoint is Reached	P = 250.0 psid
3.2	CEAs Enter Core; Loss of AC Power on Turbine Trip; RCPs Coastdown Begins; Diesel Generator Start Coming On Line	—
6.1	Main Feedwater Pumps Coatdown Completed	—
	Main Feedwater Contained in Piping Begins to Enter Affected Steam Generator	3270 lbm/sec
7.2	Auxiliary Feedwater Actuation Analysis Setpoint is Reached	78.1**
8.7	Main Steam Isolation Valves Completely Closed	—

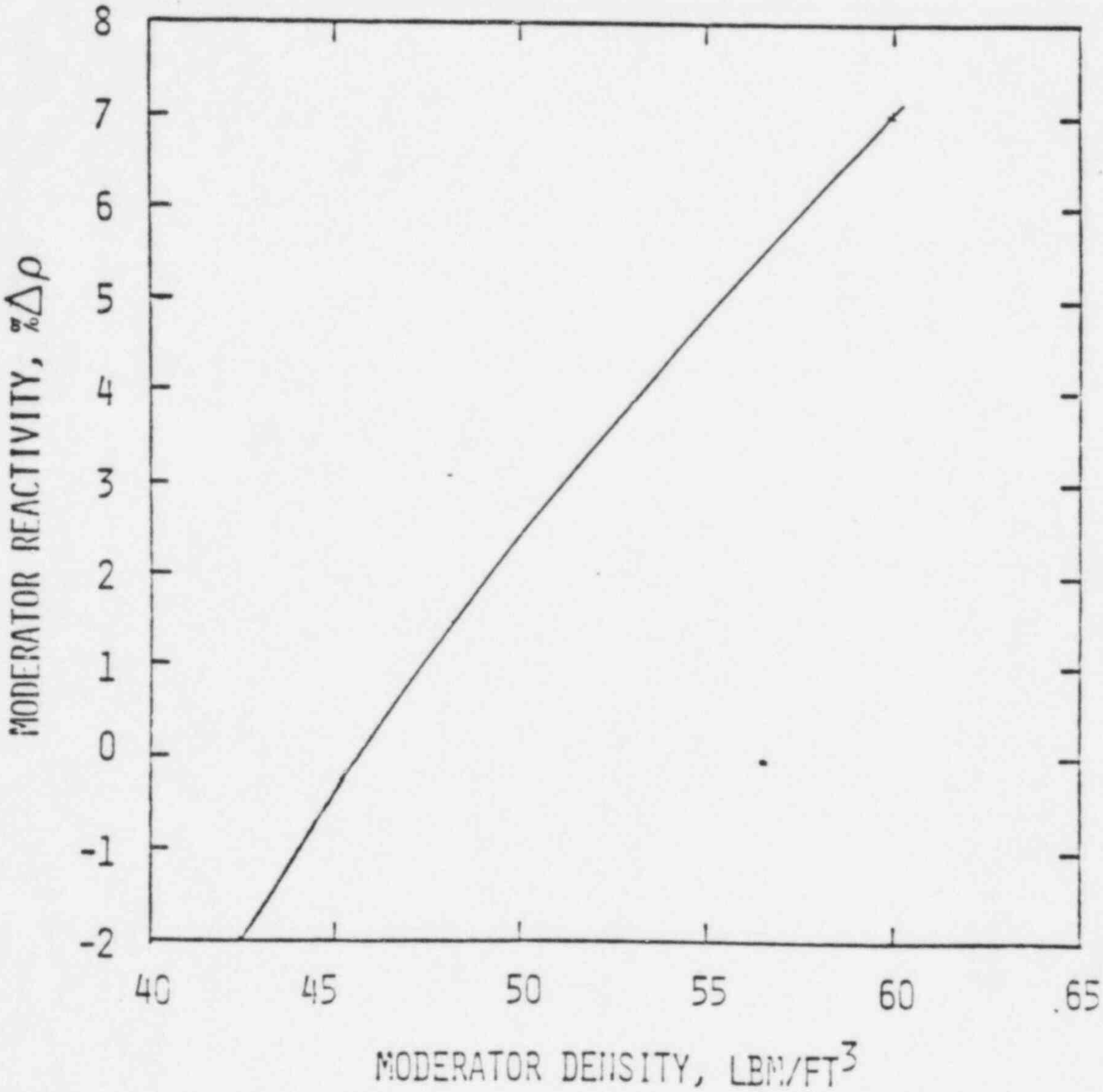
TABLE 2
(continued)

Time (sec)	Event	Setpoint or Value
12.2	Steam Admission Valves to Steam Driven AFW Pump Completely Open	— —
13.2	Diesel Generator Reach Rated Speed Following LOAC Power; Shutdown Sequencer Initiated	—
16.7	Steam Driven AFW Pump at Full Speed and Delivers AFW Flow to Both Steam Generators	217.0 gpm/SG
21.8	Safety Injection Actuation Signal is On; LOCA Sequencer Initiated	1645.0 psia
22.9	AFW Block Valve Completely Closed; AFW Flow to Affected Steam Generators is Terminated.	
24.9	Pressurizer Empties	—
26.3	Power Provided to High Pressure Safety Injection Pumps	
40.4	Delivery of Main Feedwater Contained in the Pipes to Affected Steam Generator is Completed	—
51.8	Power Provided to Motor Driven AFW Pump	—
51.8	Motor Driven AFW Pump at Full Speed and Delivers AFW Flow to Intact Steam Generator	217.0 gpm
56.8	High Pressure Safety Injection Pump at Full Speed	—

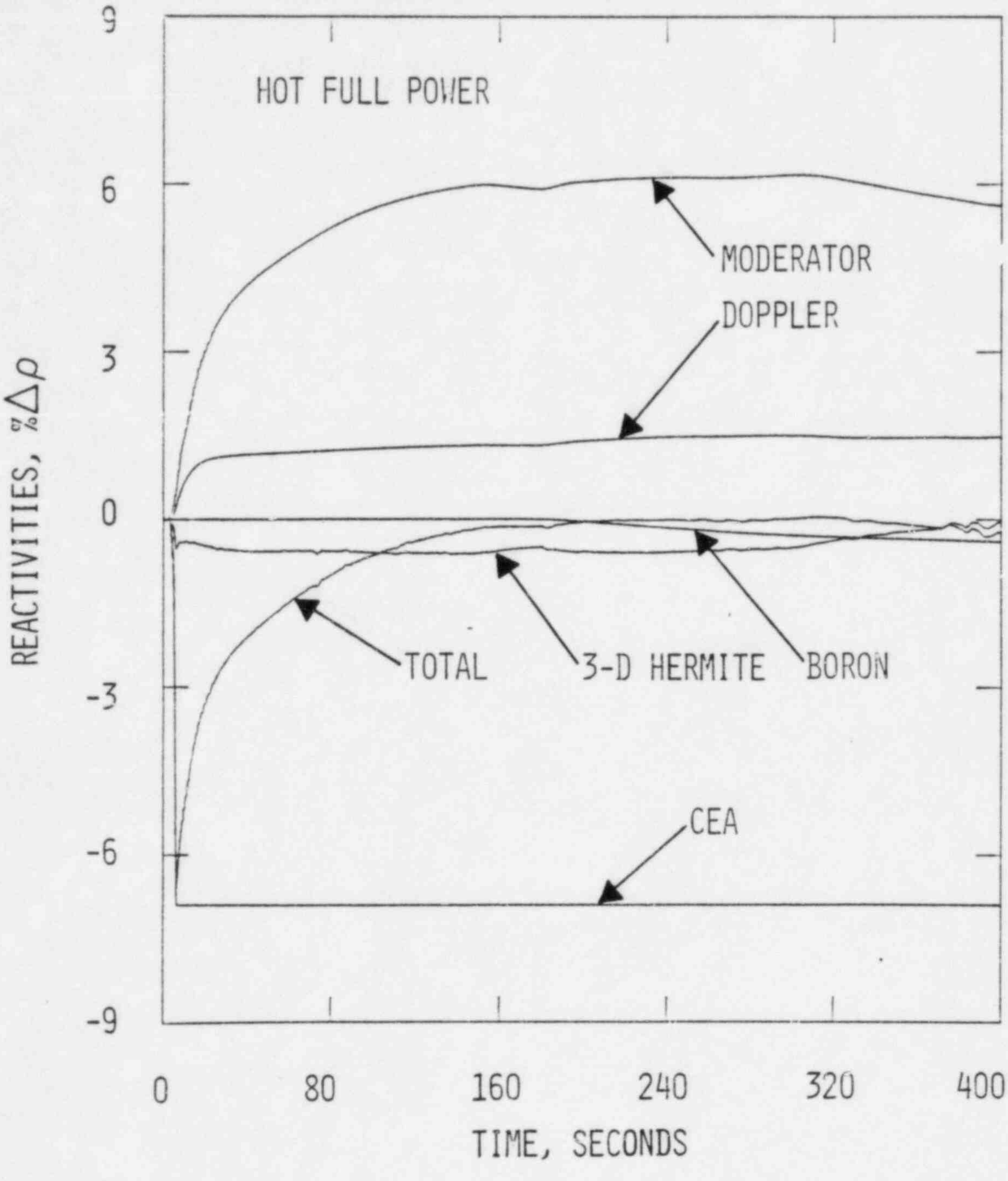
TABLE 2
(continued)

Time (sec)	Event	Setpoint or Value
82.7	Main Feedwater Isolation Valve Complete Closed	—
265.4	Affected Steam Generator Blows Dry	—
312.0	Peak Reactivity	+.41% $\Delta\rho$
319.5	Peak Return to Power	7.09% of 2700 MWt

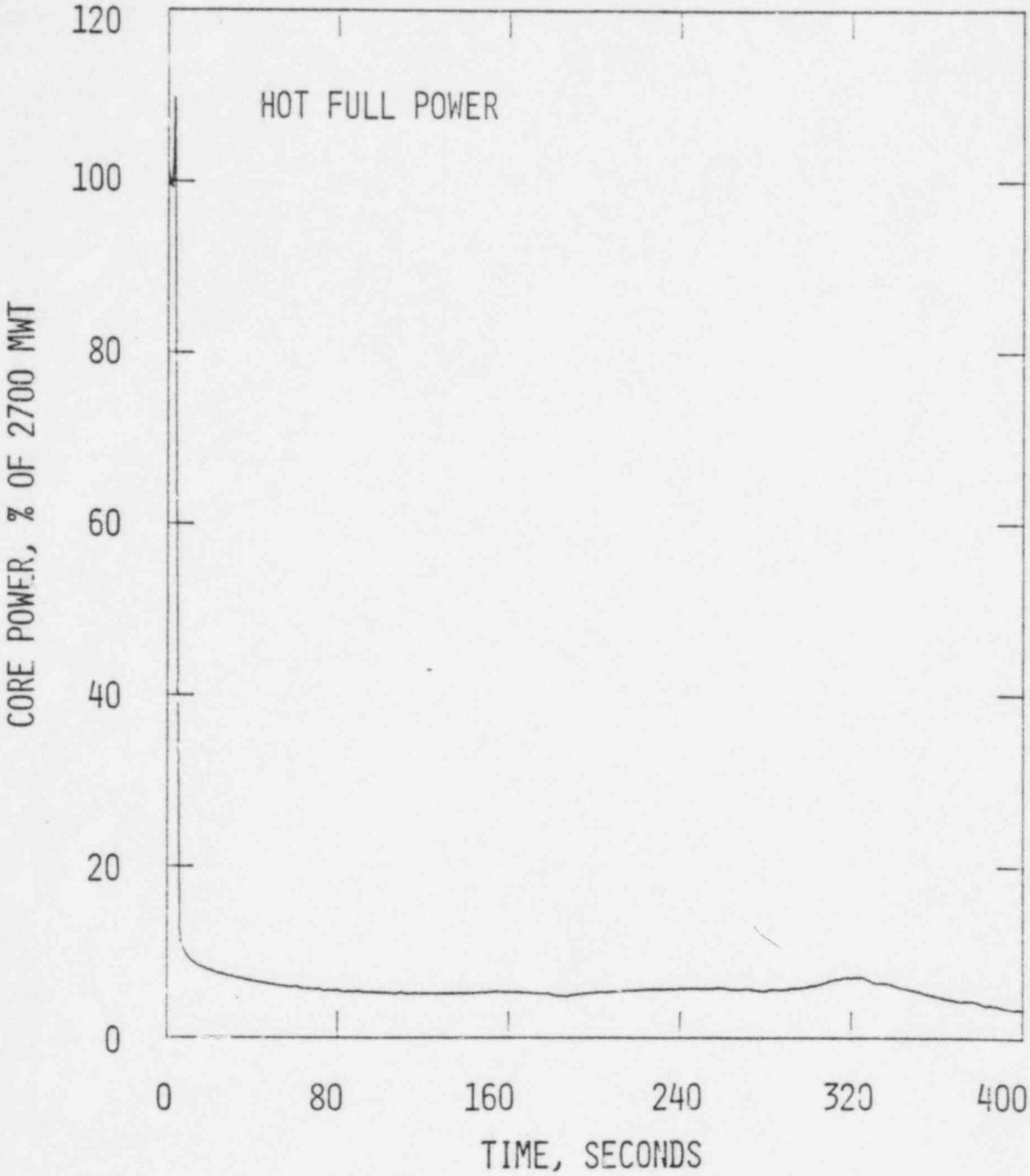
**% of distance between steam generator wide range upper and lower level instrument taps.



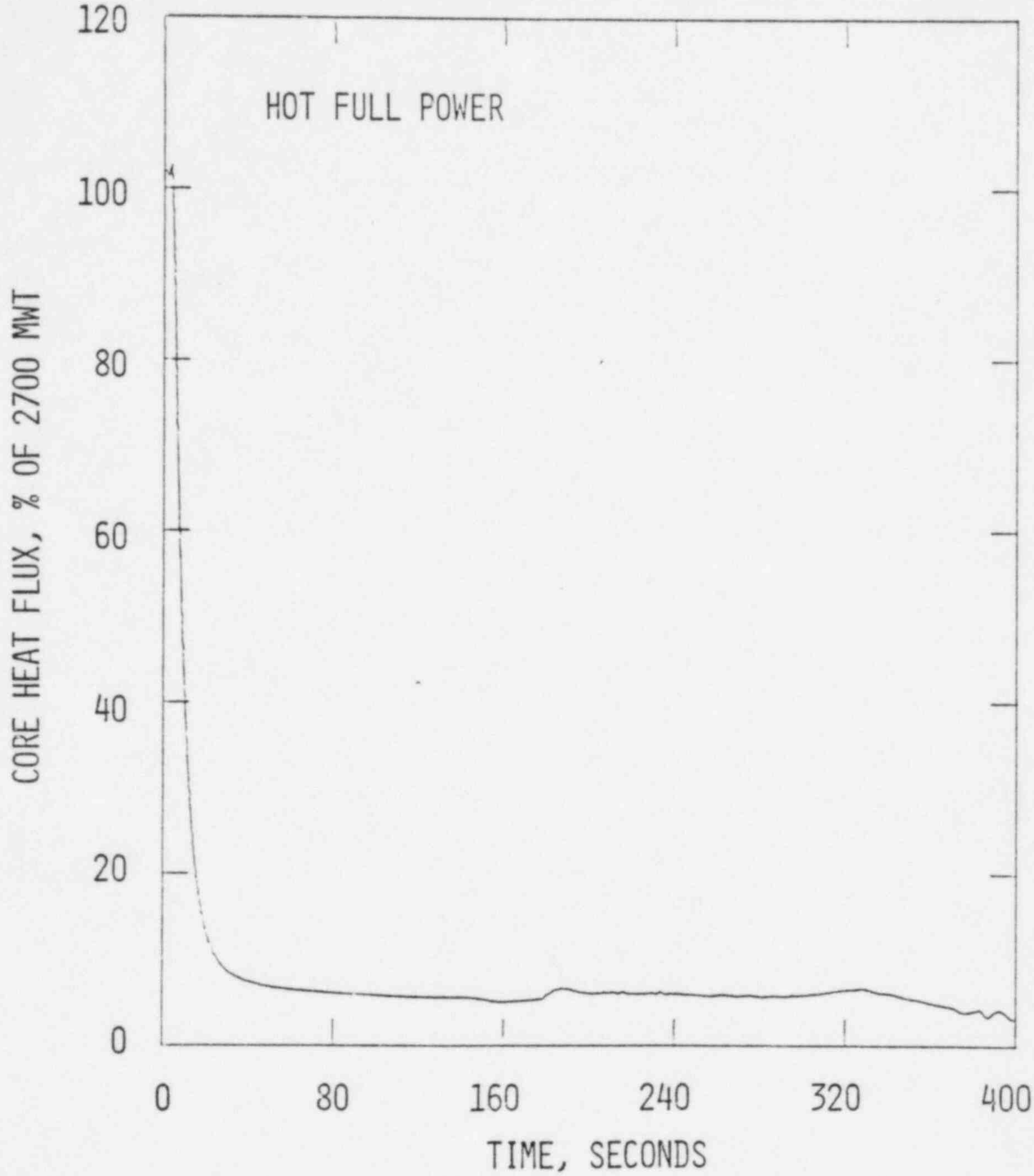
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT MODERATOR REACTIVITY VS MODERATOR DENSITY	FIGURE 1



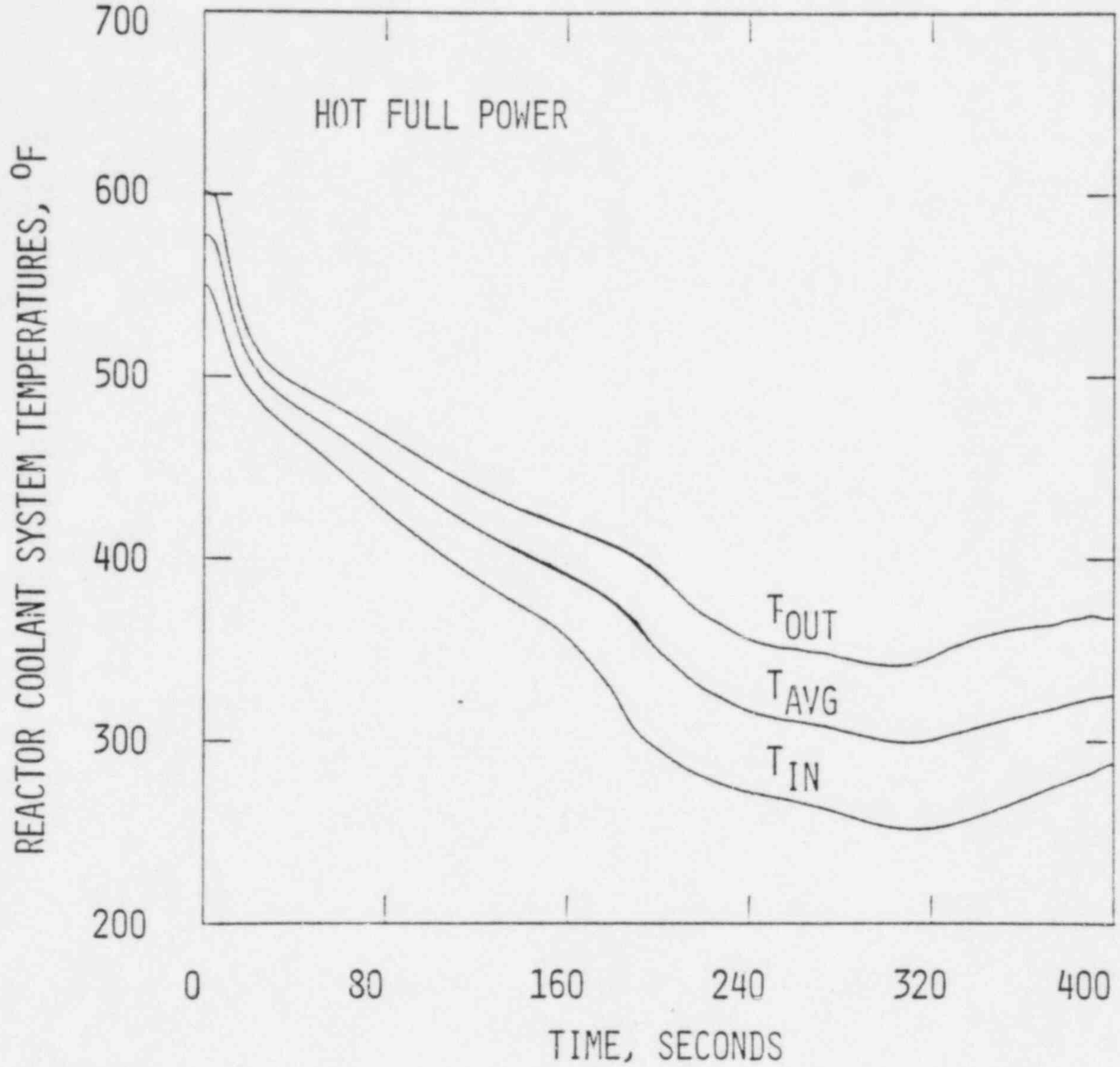
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT REACTIVITIES VS TIME	FIGURE 2



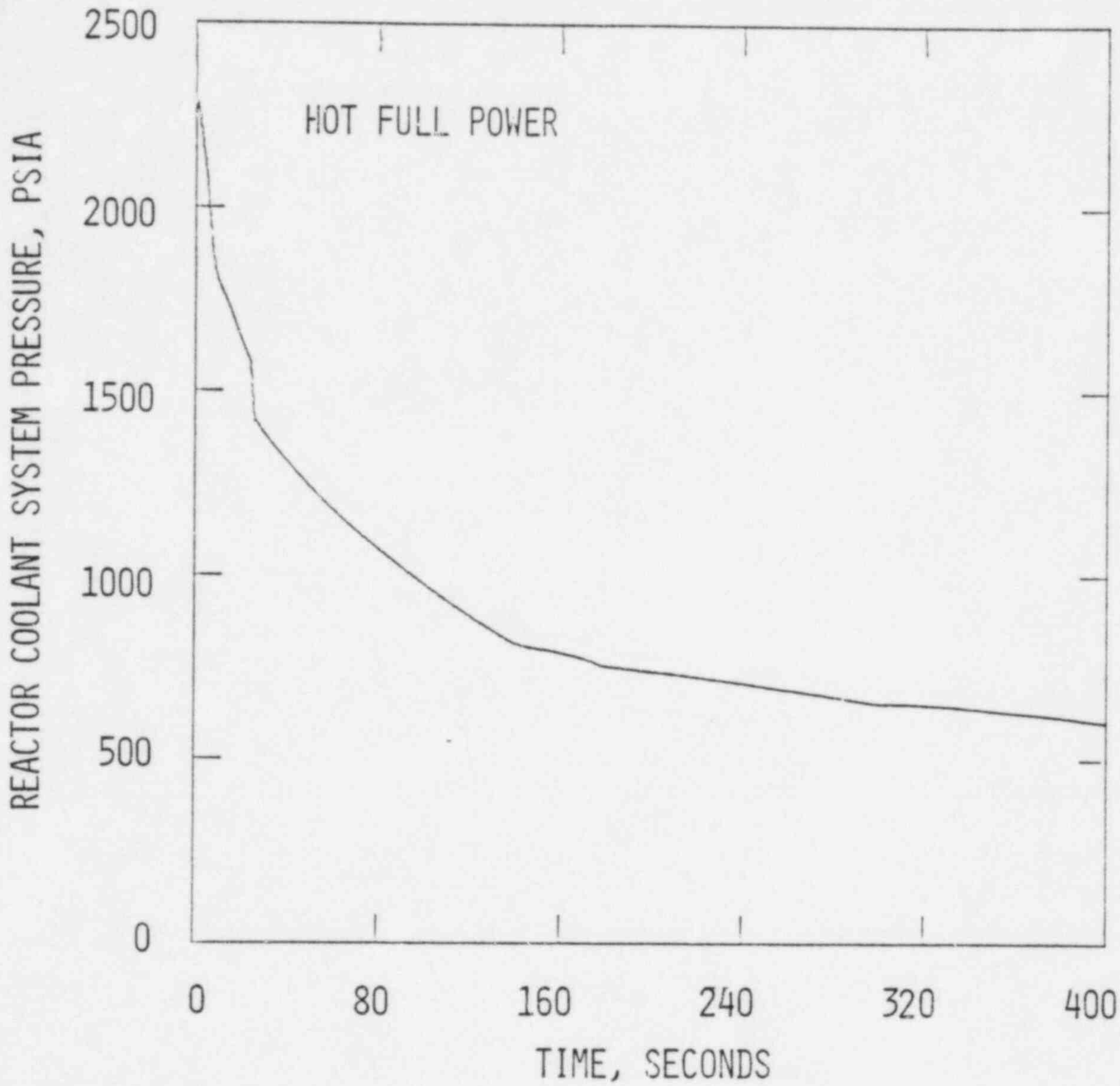
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT CORE POWER VS TIME	FIGURE 3



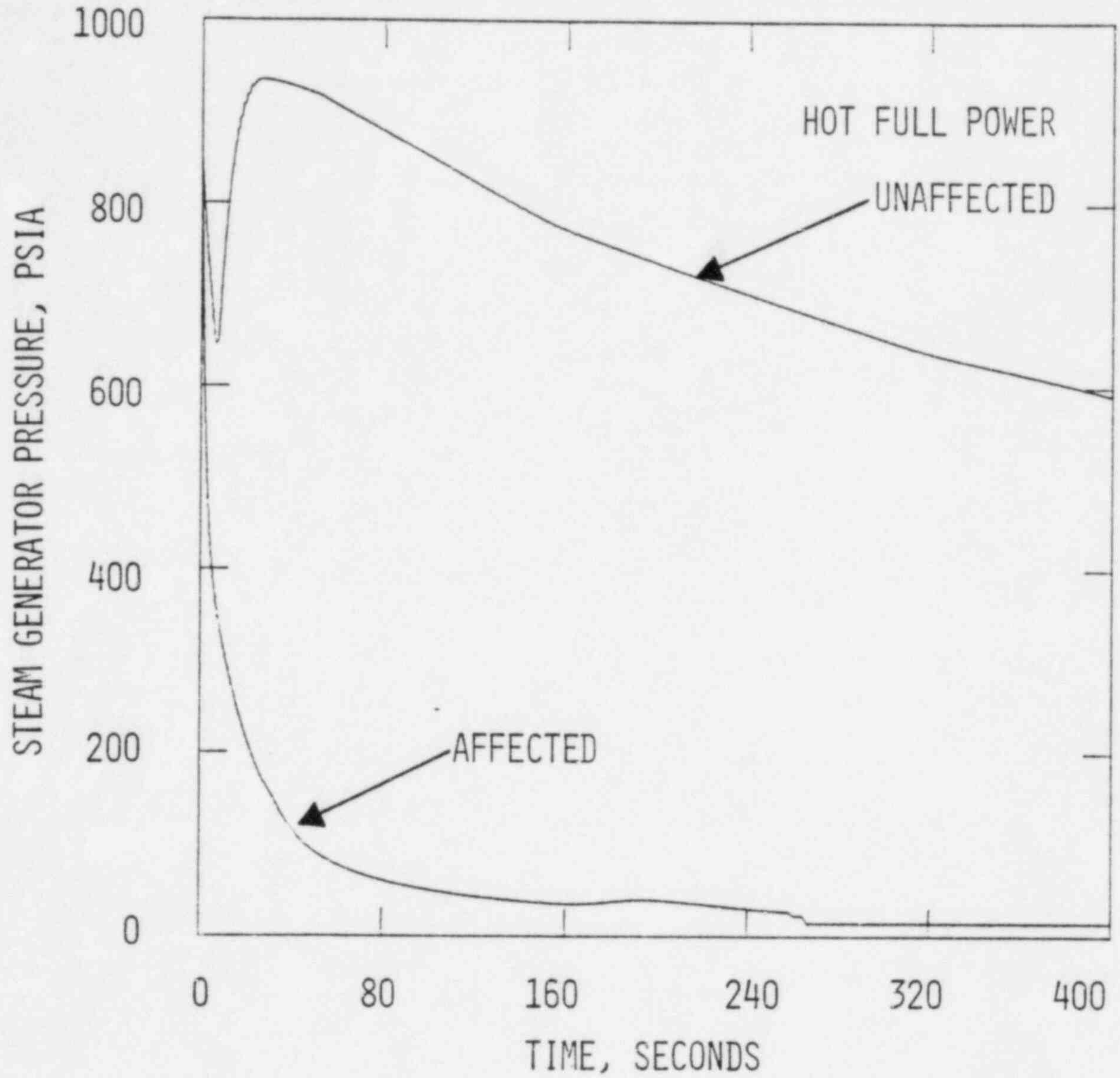
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT CORE HEAT FLUX VS TIME	FIGURE 4



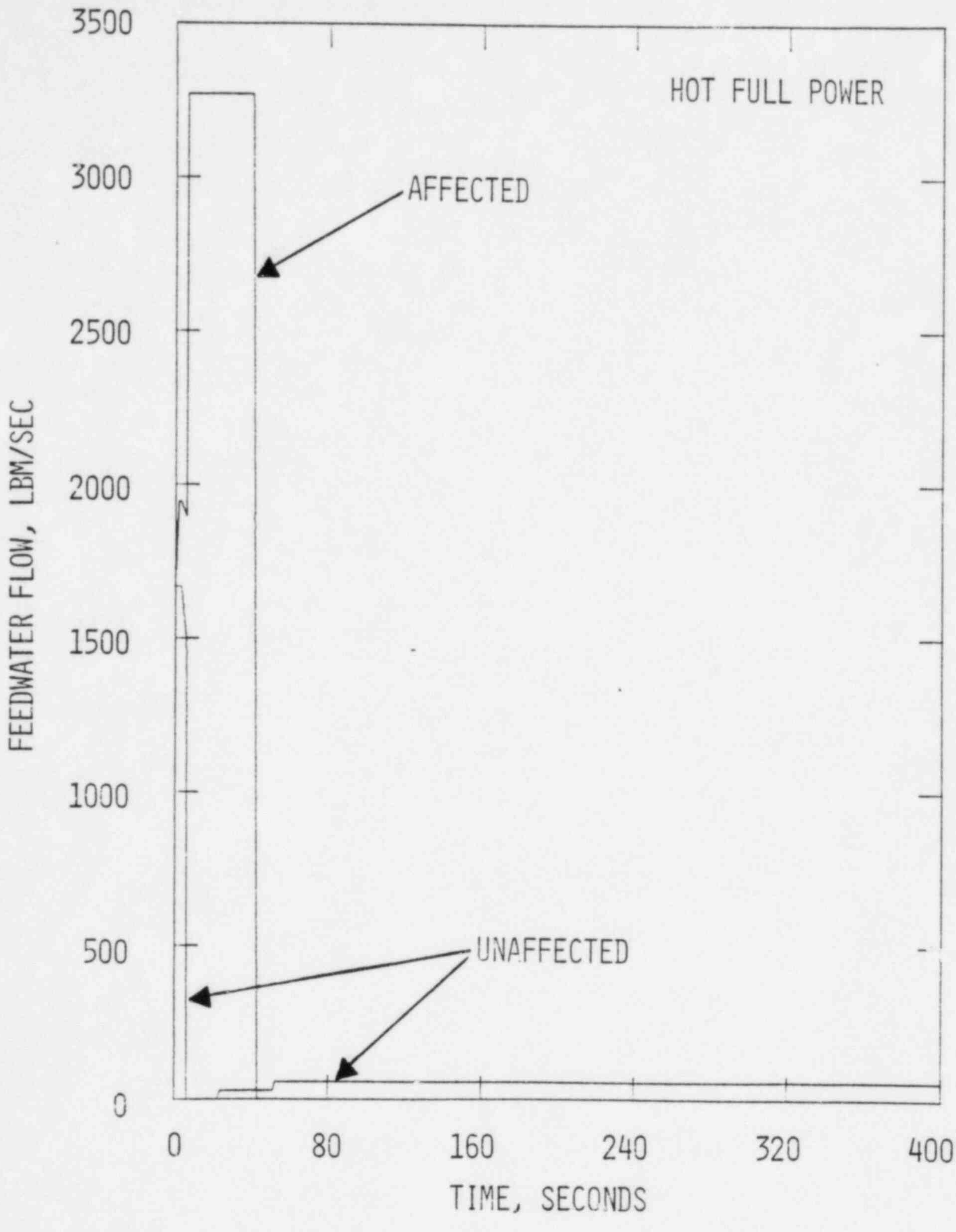
BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT REACTOR COOLANT SYSTEM TEMPERATURES VS TIME	FIGURE 5



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT REACTOR COOLANT SYSTEM PRESSURE VS TIME	FIGURE 6



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT STEAM GENERATOR PRESSURE VS TIME	FIGURE 7



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	STEAM LINE BREAK EVENT FEEDWATER FLOW VS TIME	FIGURE 8