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410 495-4455

BGE

January 22, 1998

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
Supplementary Responses to the April 22 and July 25, 1997, Requests for Additional Information: License Amendment Request; Change to Reactor Coolant System Flow Requirements to Allow Increased Steam Generator Tube Plugging (TAC Nos. M97855 and M97856)

By letter dated January 31, 1997 (Reference a), Baltimore Gas and Electric Company submitted a license amendment request to the Nuclear Regulatory Commission to support operation of Calvert Cliffs Units 1 and 2 with up to 2500 steam generator tubes plugged in each steam generator. The purpose of this letter is to provide supplementary responses to your April 22 and July 25, 1997, Requests for Additional Information (References c and e, respectively).

In our August 19, 1997, response (Reference b) to your April 22, 1997, request for additional information (Reference c), we committed to re-analyze the Steam Generator Tube Rupture Event quantitatively. Accordingly, Attachment (1) to this letter is a proposed revision to Section 14.15, "Steam Generator Tube Rupture Event" of the Calvert Cliffs Updated Final Safety Analysis Report containing the results of the re-analysis. The results confirm the conclusion of our qualitative evaluation (Reference a) that the acceptance criteria for the Steam Generator Tube Rupture Event would not be exceeded.

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In our September 29, 1997, response (Reference d) to your July 25, 1997, request for additional information (Reference e) concerning reactor coolant pump loop-seal clearing and break orientation, and compliance with the requirements of 10 CFR 50.46(b), we informed you that the Calvert Cliffs loop seal elevation is above the top of the core, and as a result Calvert Cliffs will not experience hydrostatically-induced core uncovery due to loop seal clearing and/or refilling behavior. During a telecon held with your staff, on October 15, 1997, we were asked to outline, in writing, our action plan to update the licensing basis for small break loss-of-coolant accident, should the configuration of the Calvert Cliffs loop seal elevation change in the future. As we informed you during the telecon, Asea Brown Bovari-Combustion Engineering (ABB-CE) is the fuel vendor for Calvert Cliffs, and has a model which is

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capable of simulating the small break loss-of-coolant accident scenario of concern. Our plan is to use the ABB-CE model to update the licensing basis should the need arise in the future.

We are currently planning to submit the analyses for Control Room Habitability for the design basis events that were revised for the subject license amendment request by March 1998. Should you have further questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,

Charles H. Cruse

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, Charles H. Cruse, being duly sworn, state that I am Vice President, Nuclear Energy Division, Baltimore Gas and Electric Company (BGE), and that I am duly authorized to execute and file this License Amendment Request on behalf of BGE. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other BGE employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Charles H. Cruse

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 22 day of January, 1998.

WITNESS my Hand and Notarial Seal: Denise D. Snuder
Notary Public

My Commission Expires: 2/1/02
Date

CHC/GT/bjd

Attachment: (1) Steam Generator Tube Rupture Event

cc: R. S. Fleishman, Esquire
J. E. Silberg, Esquire
Director, Project Directorate I-1, NRC
A. W. Dromerick, NRC
H. J. Miller, NRC
Resident Inspector, NRC
R. I. McLean, DNR
J. H. Walter, PSC

REFERENCES:

(a) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated January 31, 1997, License Amendment Request; Change to Reactor Coolant System Flow Requirements to Allow Increased Steam Generator Tube Plugging

(b) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated August 19, 1997, Response to Request for Additional Information: License Amendment Request; Change to Reactor Coolant System Flow Requirements to Allow Increased Steam Generator Tube Plugging (TAC Nos. M97855 and M97856)

(c) Letter from Mr. A. W. Dromerick (NRC) to Mr. C. H. Cruse (BGE), dated April 22, 1997, Request for Additional Information - Proposed Technical Specification Changes to Reactor Coolant System Flow Limit, Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (TAC Nos. M97855 and M97856)

(d) Letter from Mr. C. H. Cruse (BGE) to NRC Document Control Desk, dated September 29, 1997, Response to the July 25, 1997, Request for Additional Information: License Amendment Request; Change to Reactor Coolant System Flow Requirements to Allow Increased Steam Generator Tube Plugging (TAC Nos. M97855 and M97856)

(e) Letter from Mr. A. W. Dromerick (NRC) to Mr. C. H. Cruse (BGE), dated July 25, 1997, Request for Additional Information - Proposed Technical [Specification] Changes to Reactor Coolant System Flow Limit [], Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M97855 and M97856)

ATTACHMENT (1)

Steam Generator Tube Rupture Event

Calvert Cliffs Nuclear Power Plant
Units 1 & 2
January 22, 1998

14.15 STEAM GENERATOR TUBE RUPTURE EVENT

14.15.1 IDENTIFICATION OF EVENT AND CAUSES

The Steam Generator Tube Rupture (SGTR) Event is re-analyzed to account for steam generator (SG) tube plugging and also to account for an isolated atmospheric dump valve (ADV).

Tube plugging is a consequence of corrosion of the tubes and the analysis is performed for a maximum number of 2500 tubes plugged in each SG.

Isolation of an ADV may occur when an ADV begins to leak at an excessive rate and is isolated to prevent further leakage and damage to the valve. Following a SGTR, if the isolated ADV is associated with the intact SG, the ADV is unisolated after operator control of the plant is established.

Tube plugging reduces the heat transfer surface area and the flow area in the SG. Reduced Reactor Coolant System (RCS) flow rate and lower SG pressure result from tube plugging. Tube plugging appears to increase the releases somewhat during a SGTR, probably due to increased SG ΔP. Reduced cooldown rates and increased reliance on the affected SG for cooling result from ADV isolation; thus, ADV isolation also appears to increase releases somewhat.

The use of the affected ADV in this analysis is for the purpose of maximizing the radiological releases during the event since the ADVs are not required for cooldown. The ADVs do not perform a safety function; other means are available for cooldown, turbine bypass valves, main steam safety valves (MSSVs), and once-through core cooling, if ADVs are unavailable. If neither ADV were used, releases to the atmosphere would decrease.

14.15.2 SEQUENCE OF EVENTS AND SYSTEMS OPERATION

The sequence of events for a typical limiting case is presented in Table 14.15-2. Several cases were analyzed to examine the effect of time of reactor trip, initial SG pressure, auxiliary feedwater (AFW) actuation and flow, subcooling, plugged tubes, and cooldown rate on radiological dose consequences. The results, in most cases, did not differ significantly and the final results include an arbitrary margin to assure that a limiting case is presented The sequence of events for the presented case utilizes several assumptions regarding system operation that are chosen to maximize the radiological doses. The operator actions assumed in the analysis are consistent with Emergency Operating Procedures (EOPs).

The analysis assumed a loss of forced circulation following the reactor trip which results in higher hot leg temperature, higher fraction of the leak flow flashing into the affected SG, slower cooldown and RCS depressurization, and reduces the capability to cool down the plant via the unaffected SG. All of these effects result in higher doses.

No credit was taken in the analysis for operation of the steam bypass valves to the condenser. All of the steam releases are assumed to be directly to the atmosphere via the MSSVs or the ADVs.

The SG blowdown is assumed to be unavailable for level control.

The analysis assumed the lowest allowed opening setpoint for the MSSVs to maximize their releases to the atmosphere. Furthermore, minimum AFW flow was assumed based on the automatic action of the Auxiliary Feedwater Actuation System, which maximizes SG pressures and ADV releases to the atmosphere during the post-trip period prior to operator action.

The ADV of the unaffected or intact SG is isolated at the onset of the event. Therefore, initially, all of the heat removal is through the ADV of the affected SG. Also, the unblocking of the isolated ADV may comprise a one hour delay as personnel need to access the manual control station which is outside the Control Room. The use of the ADV in this analysis is for the purpose of maximizing the radiological releases; the ADVs do not perform a safety function. Other means are available for cooldown, turbine bypass valves, MSSVs, or once-through core cooling, if ADVs are unavailable. A case performed for this analysis shows that the MSSVs provide adequate steam release with less dose.

The operator actions assumed in this analysis are consistent with the Calvert Cliffs EOPs. The first operator action is assumed at 15 minutes following the reactor trip. Subsequently, a time delay of two minutes between each discrete operator action is assumed. The major post-trip EOP analysis assumptions regarding operator actions are:

1. Take manual control of the ADVs and AFW

Fifteen minutes following the trip, the operator is assumed to take manual control of the ADVs and AFW to prevent challenges to MSSVs if needed and maintain adequate SG level. The ADVs are used because of the analysis assumption that the steam bypass control system is unavailable. Both steam-driven and motor-driven AFW pumps are assumed operable, but less than half of their available capacity is assumed to be delivered to the SGs.

2. Diagnose the event and stabilize the plant

Calvert Cliffs procedures are oriented toward quickly diagnosing the event and stabilizing the RCS to a temperature which precludes a challenge to the MSSVs.

The analysis considered two cases: one for a 10 minute period of stabilization and diagnosis beyond the 15 minutes during which no operator action is assumed, and one without this additional period. The one used in this discussion includes the stabilization and diagnosis period and results in conservative radiological doses.

As a result of this diagnosis, the operator initiates action to unisolate the ADV of the intact SG, which is assumed to be isolated at this time. The actions may take up to one hour after taking control.

3. Reactor Coolant System cooldown prior to isolation of the affected SG

After the diagnosis, the operators will cool the RCS at a cooldown rate of up to about 150°F/hr (a maximum of 100°F/hr in any one hour). The range of target cooldown rates from about 80°F/hr to about 150°/hr were analyzed as limited by the postulated conditions. Since, for the bounding case of this analysis, the ADV of the intact SG is assumed blocked during the initial phase of cooldown, only a single ADV is available. The cooldown continues via the affected ADV until the hot leg temperature of the affected loop reaches the isolation temperature of 515°F. A conservatively lower temperature is assumed in the analysis which includes an appropriate hot leg temperature uncertainty, in order to delay isolation of the affected SG. Additionally, during this period, AFW is delivered to each SG as needed in order to maintain the level in both SGs per the requirements in the EOPs.

4. Isolation of the affected SG

The operator is assumed to isolate the affected SG at lea . 15 minutes after the hot leg temperature of the affected loop has reached the isolation temperature and 10 minutes after the intact SG is unisolated. This occurs following the diagnosis/stabilization and well into the cooldown period. Howe er, under the assumed conditions for this analysis, the isolation has only limited effect on the transient; the ADV is needed for level control and by the time isolation temperature is reached, the level in the affected SG is high enough to require operation of the ADV.

5. Plant cooldown following the isolation of the affected SG

The analysis assumes that following the isolation of the affected SG, the operator cools the RCS at a target rate of 35°F/hr, or as needed to control the SG level, for up to 2 hours into the event.

6. Depressurization of the RCS and required subcooling margin

The primary-to-secondary leak rate, and consequent radiological doses, are directly related to the depressurization of the RCS. In turn, the RCS depressurization behavior is constrained by the RCS cooldown rate and the required subcooling margin which must be maintained.

The analysis further assumed that the RCS temperature and pressure indications used to determine subcooling during the event included uncertainties, resulting in having an indicated RCS temperature greater than actual, and an indicated pressurizer pressure less than actual. These uncertainties would result in actual subcooling which is more than that calculated by the operator. The higher target subcooling assumed by the analysis due to inclusion of uncertainties results in significantly slower RCS depressurization, which increases the tube leak and the resultant doses.

The analysis also conservatively assumes that the auxiliary spray system is not available to the operator for depressurization of the RCS. Instead, the pressurizer vent is used. The containment air coolers remove the energy released through the vents, hence harsh conditions are not reached in the Containment.

7. Preventing the affected SG from overfill

The EOPs include instructions to the operator for minimizing leakage by equalizing the secondary and primary pressure used for preventing the overfill of the affected SG.

Although the procedure provides for SG blowdown or backflow to the RCS to limit SG level, use of the affected ADV for level control is also described, therefore, no credit is taken for the procedure instructing the operator to maintain secondary and primary pressure equal in order to minimize leakage.

The analysis assumes that, as the affected SG level reaches the high level limit of +50" above the normal water level, the operator reduces the liquid level by opening the ADV of the affected SG even after the isolation temperature has been reached. The higher leak rates caused by assuming a high subcooling margin also results in the affected SG level exceeding the high level limit and prompting ADV steaming. Opening the affected SG ADV in order to prevent overfill amounts to effectively accomplishing most of the RCS cooldown via the affected SG rather than the intact SG, resulting in significantly higher doses.

8. Maintaining adequate RCS inventory, high pressure safety injection (HPSI) throttle criteria

The operator is simultaneously charged with assuring that adequate subcooling and adequate RCS inventory is maintained. Specifically, the EOPs require the operator to retain minimum specified levels in the pressurizer and the vessel upper head prior to throttling back the HPSI flow.

Two HPSI pumps were assumed to be started on safety injection actuation signal (SIAS) and a third was conservatively assumed to come on line with operator action, thus maximizing the flow delivered to the RCS upon SIAS. These assumptions result in higher post-trip RCS pressures, and maximize the tube leakage.

The combination of the assumed cooldown rate and the high subcooling margin including instrument uncertainties result in a conservatively slow depressurization of the RCS, which maximizes the tube leakage. The increased leak rate raises the final activity level released through the affected SG. It also leads to an unacceptably high liquid level in the SG, resulting in the opening of the affected SG ADV and more frequent release of its contents to the atmosphere. The ADV steaming is increased by the assumption of a lower actual SG level to accommodate instrument uncertainties.

Together, these assumptions, in combination with the radiological assumptions presented in Section 14.15.3.2, assure that the radiological dose results from the analysis conservatively bound the expected doses for this event.

14.5.3 ANALYSIS OF EFFECTS AND CONSEQUENCES

14.15.3.1 Core and System Performance

A. Mathematical Models

The thermal hydraulic response of the Nuclear Steam Supply System to the SGTR was simulated using the CESEC-III, Mod 4 computer program (CENPD-107, "CESEC, Digital Simulation of a Combustion Engineering Nuclear Steam Supply System" April 1974 and Supplements 1-6) up to the time the operator takes control of the plant (15 minutes after trip). Operator actions to mitigate the effects of the SGTR Event and bring the plant to shutdown cooling entry conditions were simulated using a CESEC-based cooldown algorithm (COOL).

B. Input Parameters and Initial Conditions

The input parameters and initial conditions used in the analysis are listed in Table 14.15-1 for the present cycles of Unit 1 and Unit 2. The selected values of these inputs maximize the radiological releases to the atmosphere during the transient.

The maximum allowed technical specification core inlet temperature, including instrument uncertainties, results in a correspondingly high initial SG pressure. This increases the steam released through the MSSVs and the ADVs throughout the event.

The minimum core flow results in higher average coolant temperature and higher enthalpy fluid entering the SG, a resultant increase in flashing fraction, and higher activity releases through the MSSVs and ADVs.

A maximum initial pressure and a maximum initial pressurizer liquid volume delay the reactor trip. These parameters were modified to assess the impact of earlier trip times on resultant radiological doses. The late trip time was found to result in the highest radiological doses.

The SG level is maintained within a small range during operation, the limits of which would have no effect on the trip time and insignificant effect on the AFW actuation time

The analysis assumed the lowest allowed opening setpoint for the MSSVs to maximize their releases to the atmosphere. Furthermore, the initial pressurizer pressure, the AFW flow actuation time, and volume were varied to identify the most adverse combination to maximize the MSSV and ADV releases to the atmosphere during the post-trip period prior to operator action.

The selection of fuel and moderator temperature coefficients are not significant, as there is no change in the core power or temperature prior to reactor trip. The thermal margin/low pressure (TM/LP) trip uncertainty was applied to lower the setpoint to delay the trip action for the late trip case, and to raise the setpoint for the early trip case. The actual setpoint selected to delay the trip to the maximum degree was the Reactor Protective System setpoint which is lower than the lowest possible TM/LP pressure limit. Two HPSI pumps were assumed to be started on SIAS and a third was conservatively assumed to come on line with operator action, thus maximizing the flow delivered to the RCS upon SIAS. These assumptions result in higher post-trip RCS pressures, and maximize the tube leakage.

The radiological consequences of the SGTR transient are also dependent on the break size. As the break size is decreased from that of a double-ended rupture, the integral leak is reduced and the radiological consequences will be less severe. Therefore, the most adverse break size is the largest assumed break of a full double-ended rupture of a SG tube.

C. Results

Table 14.15-2 presents the sequence of events for the double-ended rupture of a SG tube event with the loss of forced circulation upon reactor trip. Figures 14.15.3-1 through 14.15.3-16 present the dynamic behavior of important Nuclear Steam Supply System parameters during this event.

The double-ended break of a SG tube results in a primary-to-secondary leak rate which exceeds the capacity of the charging pumps. As a result, pressurizer level and pressure gradually decrease from their initial values. For the case discussed here, maximum charging flow and zero letdown was assumed to delay the time of reactor trip. As the pressure decreases, the proportional heaters and then backup heaters are turned on to prevent further depressurization. All heaters are turned off automatically at 555 seconds as the pressurizer level is decreasing to levels which result in uncovery of the heaters. The depressurization of the RCS and pressurizer level decrease continue, resulting in an approach to departure from nucleate boiling (DNB) specified acceptable fuel design limits (SAFDL). The TM/LP trip is designed to trip the reactor before the DNB SAFDL is reached. The analysis of the SGTR

Event demonstrates that the action of the TM/LP trip prevents the DNB SAFDL from being exceeded, since the rate of depressurization for this event is less than the rate of depressurization for the RCS depressurization event. The analysis credits a reactor trip only when the low pressurizer pressure floor of the TM/LP trip is reached at 788.2 seconds, and the trip breakers are opened within 0.9 seconds of this time. The loss of forced circulation (reactor coolant pump pumps tripping) is assumed to occur 3 seconds after the trip breakers are opened, at 792.1 seconds, resulting in the initiation of the RCS flow coastdown.

The analysis also assumes the steam bypass system to the condenser will become unavailable and that the unaffected SG ADV is blocked for 60 minutes into cooldown. The affected SG ADV automatically opens at trip time and then modulates on a program based on RCS average temperature. The turbine valve closure due to the reactor trip causes the SG pressures to rise, and leads to the opening of the MSSVs at 794.0 seconds. Maximum SG pressure of 965.2 psia is reached at 797.9 seconds. The MSSVs close at 811.3 seconds the first time. They reopen and close several times during the period until the operator takes action to cool the plant.

The loss of forced circulation and the RCS flow coastdown result in reduction of flow into the upper head region of the reactor vessel. This region becomes thermal-hydraulically decoupled from the rest of the RCS, and due to flashing caused by the depressurization and boiloff from the metal structure to coolant heat transfer, voids begin to form in this region.

The pressurizer empties at 803.2 seconds due to the continued primary-to-secondary leak and the post-trip RCS liquid shrinkage. The continued RCS and pressurizer depressurization results in SIAS generation at 803.2 seconds and delivery of the HPSI flow to the RCS at 1001.5 seconds when the RCS pressure decreases below the HPSI pump head.

The AFW actuation setpoint is reached in the unaffected SG at 1270.5 seconds and the AFW is delivered to both SGs at 1449.6 seconds following system and piping delays. Auxiliary feedwater is delivered to both SGs at 1810 seconds by operator action.

At 1689 seconds from the start of the event, 15 minutes following the trip, the operator takes manual control of the plant, which consists of manual control of ADVs, AFW, and HPSI, including bringing the 3rd HPSI pump on line. The analysis of the limiting case assumes that at this point the operator has diagnosed the event. Other cases were analyzed for which an additional diagnosis and stabilization period was assumed and found to result in lower 2-hour doses.

Following the diagnosis at 1689 seconds, the operators begin to cool down the RCS at approximately 100°F/hr, using the ADV on the affected SG and the AFW system until the hot leg temperature of the affected loop reaches an isolation temperature of 505°F (515°F per EOPs minus 10°F uncertainty) at 3849 seconds. Since the intact SG ADV is blocked for 1 hour into cooldown, until 5289 seconds, and an additional delay of 10 minutes is allowed after the unblocking, the affected SG is isolated at 5889 seconds. The ADV of the unaffected SG is not opened because the cooldown rate is already high at this time.

At 4089 seconds, the target subcooling of 65°F is reached, but the subcooling oscillates due to ADV use and eventually exceeds 70°F at which point the operator begins to use the pressurizer vent to reduce the RCS pressure. The 65°F analytical value for target subcooling consists of 35°F, 25°F uncertainty, and 5°F modeling allowance. Because of the allowed delays and tolerances, the maximum subcooling reaches 100°F. The affected SG wide range instrumentation indicates 50" above normal water level, corresponding to an analytical level of 26" including the uncertainties for most of the cooldown after isolation temperature is reached. Therefore, despite isolation of main steam isolation valve, main feedwater isolation valve, and AFW, the ADV in the affected SG continues to steam for most of the cooldown after isolation temperature is reached.

At 6369 seconds, adequate pressurizer level is reached, allowing the operator to throttle the HPSI pumps. At 2 hours into the event, <305,000 lbm is calculated to have leaked from the primary system to the secondary system. The integrated ADV mass flow out of the affected SG ADV is <290,000 lbm. At 6600 seconds when the subcooling is decreased to the target value of 65°F, the operator terminates venting. At the same time the cooldown rate reaches a low point and the operator opens the ADV of the intact SG.

In addition to the initial 2 hour period, the analysis provided data out to 8 hours for showing the approach to shutdown cooling and for showing the filling of the affected SG. The data out to 8 hours shows that the SG will not overfill before 6-1/2 hours have elapsed (see Figure 14.15.3-11-c, overfill condition is approximately 533,000 lbm).

The affected SG mass vs. time as depicted in Figure 14.15.3-11-c is extremely conservative. In particular, due to limitations in use of the code with respect to treatment of AFW, an additional 75,000 lbm of AFW inventory is added to the affected SG during this analysis beyond what is expected. In addition, consistent with the EOP, operators would reduce the hot leg subcooling and commence "reverse flow" from the affected SG to the RCS if the affected SG approached an overfill condition. Therefore, SG overfill is not a concern.

14.15.3.2 <u>Radiological Consequences</u>

The analysis of the radiological consequences considers the most severe release of secondary as well as primary system activity leaked from the tube break. The inventory of fission product activity available for release to the environment is a function of the primary-to-secondary coolant leakage rate, the assumed increase in fission product concentration for iodine generated iodine spike (GIS) dose, and the mass of steam discharged to the environment. The pre-accident iodine spike doses are not reported since they are significantly less than the GIS. Using data from Electrical Power Research Institute Report TR-103680, "Review of Iodine Spike Data From PWR Power Plants in Relation to SGTR with MSLB", the pre-accident iodine spike is estimated to be less than 10 μCi/gm rather than 60 μCi/gm as suggested in Standard Review Plan 15.6.3.

A. Assumptions and Conditions

The assumptions and parameters employed for the evaluation of radiological releases are:

(1) Doses are calculated assuming an event GIS coincident with the initiation of the event.

(2) For this GIS case, an initial activity of 1 μCi/gm (technical specification limit) and a spiking factor of 500 is assumed.

(3) The portion of the primary fluid leaking into the SG that flashes into steam is dependent on the enthalpy of the primary liquid and the saturation enthalpy of the SG. When there is a steam release to the atmosphere, the flashed portion is released before the steam in the SG. The flashing portion has a decontamination factor of 1.0. The non-flashing portion of the primary leak flow is assumed to mix uniformly with the liquid in the SG.

(4) The SG is assumed to have a decontamination factor of 100, so that the concentration of radioactivity in the steam phase is 1/100 of the concentration in the liquid phase.

(5) A decontamination factor of 1.0 is used for releases through MSSVs and ADVs.

(6) The technical specification limit for unidentified leakage, 100 gpd in the unaffected SG, is assumed.

(7) An initial secondary activity of 0.1 μCi/gm is assumed (technical specification limit).

(8) The $\frac{\chi}{Q}$ values for the atmospheric dispersion calculations are 1.3×10^{-4} sec/m^3 for 0-2 hours exclusion area boundary (EAB). A breathing rate of 3.47×10^{-4} m^3/sec was used.

B. Calculation of RCS Activity

The initial RCS activity is assumed to be the equilibrium concentration prior to the accident.

The analysis assumed an event GIS. The iodine spiking factor is defined as the ratio of the appearance rate of I-131 in the RCS following the event, to the appearance rate required to produce a steady state equilibrium concentration.

The GIS is a direct consequence of the RCS depressurization and shutdown caused by the SGTR Event. A spiking factor of 500 is used. The analysis conservatively assumes an increase in the iodine rate of appearance at the initiation of the SGTR Event which is assumed to last for at least 2 hours to maximize the impact on the 10 CFR Part 100 Exclusion Area Boundary dose. The RCS initial radioactivity concentration was assumed to be the technical specification value of 1 μCi/gm for this analysis. However, the primary activity increases steadily due to the large spiking factor.

C. Mathematical Model

The CESEC computer code was used to determine the mass and energy releases during the first period of the event from event initiation until 15 minutes after reactor trip. This data is added to the radiological releases to the atmosphere calculated for the controlled cooldown period by the CESEC based cooldown algorithm (COOL). Table 14.15-3 provides the significant input parameters for the dose calculations.

The doses at the EAB are calculated as follows:

(1) Calculate the total activity released to the atmosphere in I-131 dose equivalent curies.

(2) Multiply the total activity released to the atmosphere by the breathing rate, atmospheric dispersion factor, and I-131 dose conversion factor using the expression:

$$DEQ\ (I - 131) = R_{total} * BR * DCF * \frac{\chi}{Q}$$

where:

R_{total} = the total activity released to the atmosphere, curies
BR = breathing rate, m^3/sec
DCF = Dose Conversion Factor in DEQ I-131, REM/Curie
$\frac{\chi}{Q}$ = atmosphere dispersion coefficient/ sec/m^3

In determining the whole body dose, the major assumption made is that all noble gases leaked through the ruptured tube will be released to the atmosphere. Therefore, the whole body dose is proportional to the total primary-to-secondary leak and is calculated using the expression:

$$WBD = K_{Gamma} * (E_{Gamma} + K_{Beta} * \frac{E_{Beta}}{K_{Gamma}}) * R * \frac{\chi}{Q}$$

where:

E_{Gamma} = the average Gamma energy (MeV/dis) for the halogen isotopes of concern

E_{Beta} = the average Beta energy (MeV/dis) for the halogen isotopes of concern

R = the activity release to the atmospheres, Ci/sec

t = time, sec

$\frac{\chi}{Q}$ = atmospheric dispersion coefficient, sec/m³

K_{Gamma} = a constant = $\frac{.25\ REM * m^3 * dis}{Mev * sec * Ci}$

K_{Beta} = a constant = $\frac{.23\ REM * m^3 * dis}{Mev * sec * Ci}$

D. Results

All doses are increased by an arbitrary margin to account for variation from case to case. The 2-hour EAB thyroid dose for the SGTR Event with GIS is less than 12 REM. The 2-hour EAB whole body dose is less than 0.55 REM.

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14.15.4 CONCLUSION

The analysis of the SGTR Event demonstrates that the action of the TM/LP trip prevents the DNB SAFDL from being exceeded. For an assumed accident with event-generated iodine spiking, the 2-hour EAB dose acceptance criteria reviewed and approved by the Nuclear Regulatory Commission in their Safety Evaluation Report for Unit 1, Cycle 6 (License Amendment No. 71) are 30.0 REM to the thyroid and 2.5 REM, whole body. The EAB doses calculated for this event are within the criteria of the Safety Evaluation Report.

TABLE 14.15-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE STEAM GENERATOR TUBE RUPTURE EVENT[a]

PARAMETER	UNITS	VALUE
Core Power	MWt	2754
T_{in}	°F	552
RCS Pressure	psia	2335
SG Tubes Plugged		2500
Core Mass Flow Rate	$x10^6$ lbm/hr	122.9
Secondary Pressure	psia	825
Tube ID	inches	0.654
Flow Constant	---	1.17
Pressurizer Liquid Level at Full Power	ft^3	975
Low Pressurizer Pressure (TM/LP Floor) Setpoint	psia	1829
Safety Injection Actuation (SIAS) Setpoint	psia	1679

[a] Unit 1 Cycle 12 and Unit 2 Cycle 11

TABLE 14.15-2

SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE EVENT

TIME	EVENT	SETPOINT OR VALUE
0.0	Tube Rupture Occurs	---
62.1	Proportional Pressurizer Heaters are Energized, psia	2275
146.1	Backup Pressurizer Heaters are Energized, psia	2200
555	Pressurizer Heaters De-energize due to Low Pressurizer Level, ft^3	270
788.2	Low Pressurizer Pressure Trip Setpoint is Reached, psia	1829
789.1	Trip Breakers Open ADVs Open, °F	--- 535
792.0	Loss of Forced Circulation, Reactor Coolant Pumps Begin to Coast Down	
794.0	MSSVs Open, psia	950
797.9	Maximum SG Pressure is Reached, psia	965
803.2	Pressurizer Empties SIAS Setpoint is Reached, psia	--- 1679
811.3	MSSVs Close, psi[illegible] The MSSVs sub[illegible]ly cycle repeatedly	892
1001.5	Safety Injection Flow Begins to Enter the RCS, psia	1203

TABLE 14.15-2 (Continued)

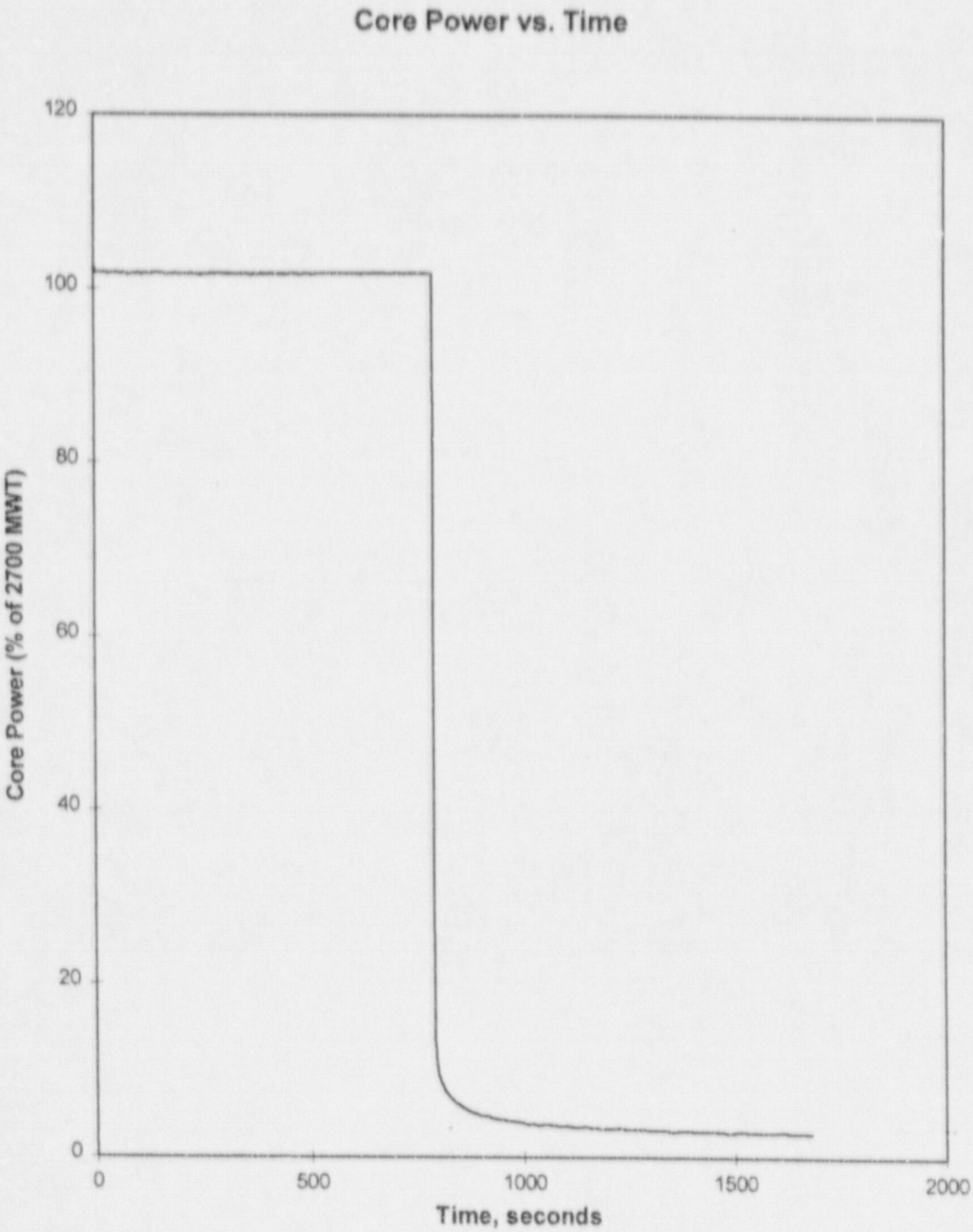
SEQUENCE OF EVENTS FOR
THE STEAM GENERATOR TUBE RUPTURE EVENT

TIME	EVENT	SETPOINT OR VALUE
1270.5	AFW Actuation Setpoint is Reached Unaffected SG, ft (% Wide Range Span)	16.3 (41.5)
1449.6	AFW is Initiated to Intact SG	180 gpm
1689.1	Operator Takes Manual Control of the Plant and Begins Cooldown at the Rate of 100°F/hr by Adjusting the ADVs on the affected SG	---
	3rd HPSI Pump is Brought on Line	---
1810	AFW Increased to Both SGs (2 minutes past takeover time)	180 gpm/SG
3849	Hot Leg Reaches Isolation Temperature, °F	505
4089	Target Subcooling is Reached, °F	65
5049	Operator Opens the Pressurizer Vent	
5289	60 minutes past takeover: Operator Unblocks ADV of Intact SG	
5889	70 minutes past takeover: Operator Isolates the Affected SG, ADV continues to steam for level control	
6389	Adequate Pressurizer Level, Inches Operator Begins to Throttle HPSIs	101 ---
6600	Target subcooling restored, °F Operator closes Pressurizer Vent, opens ADV of Intact SG	65

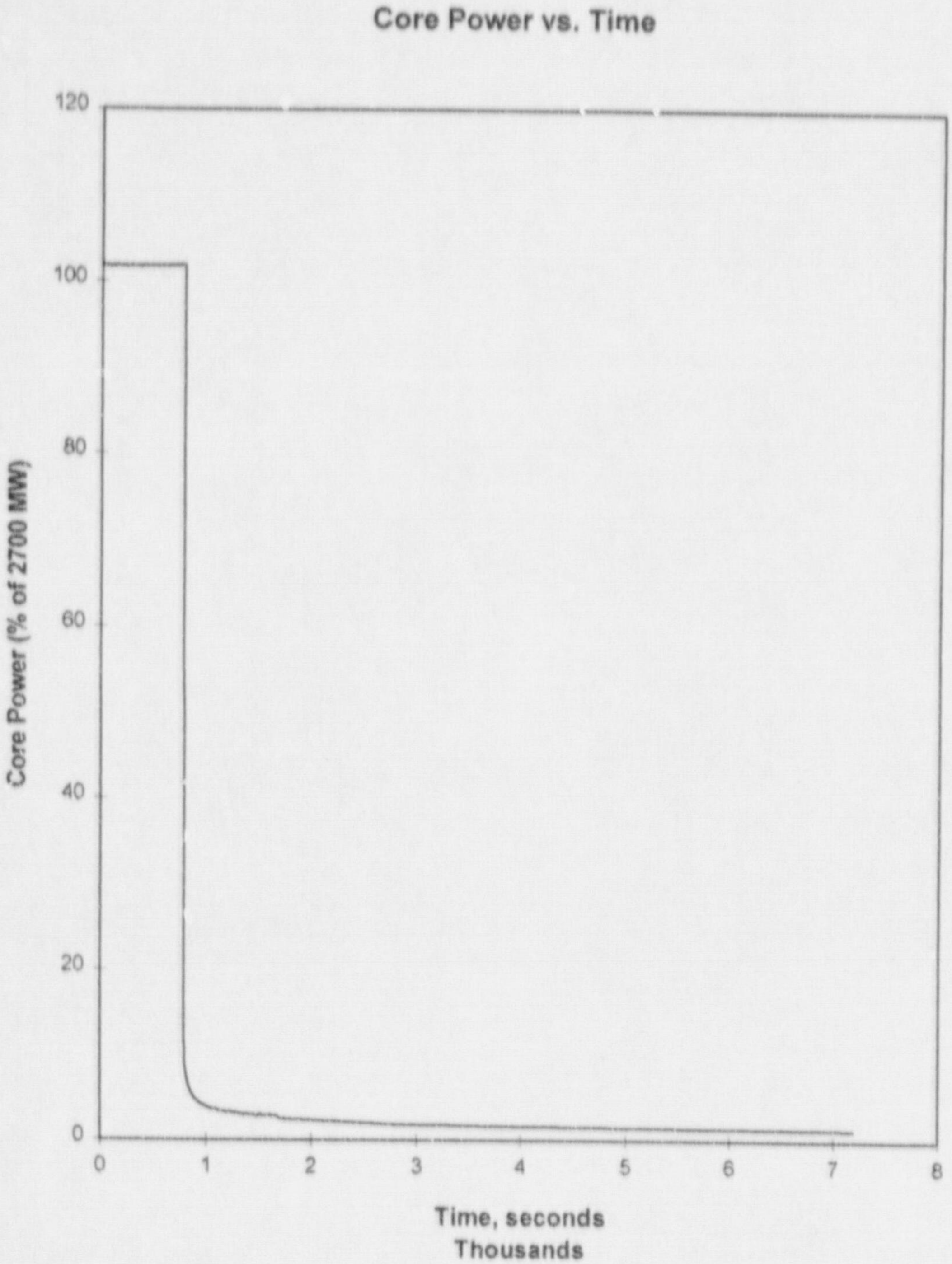
TABLE 14.15-3

ASSUMPTIONS FOR RADIOLOGICAL CONSEQUENCES OF THE STEAM GENERATOR TUBE RUPTURE EVENT

PARAMETER	DESIGN BASIS ASSUMPTION
Primary system activity:	
Event GIS, μCi/gm	1.0
Spiking factor	500
Secondary system activity, μCi/gm	0.1
Primary-to-secondary leak rate in the unaffected SG, gpm	1
0-2 hr Atmospheric Dispersion factor (X/Q) at EAB, sec/m^3	1.3×10^{-4}
Decontamination factor between the water and steam phases in the SGs	100
Breathing rate, m^3/sec	3.47×10^{-4}
I-131 dose conversion factor, REM/Ci	1.1×10^{6}

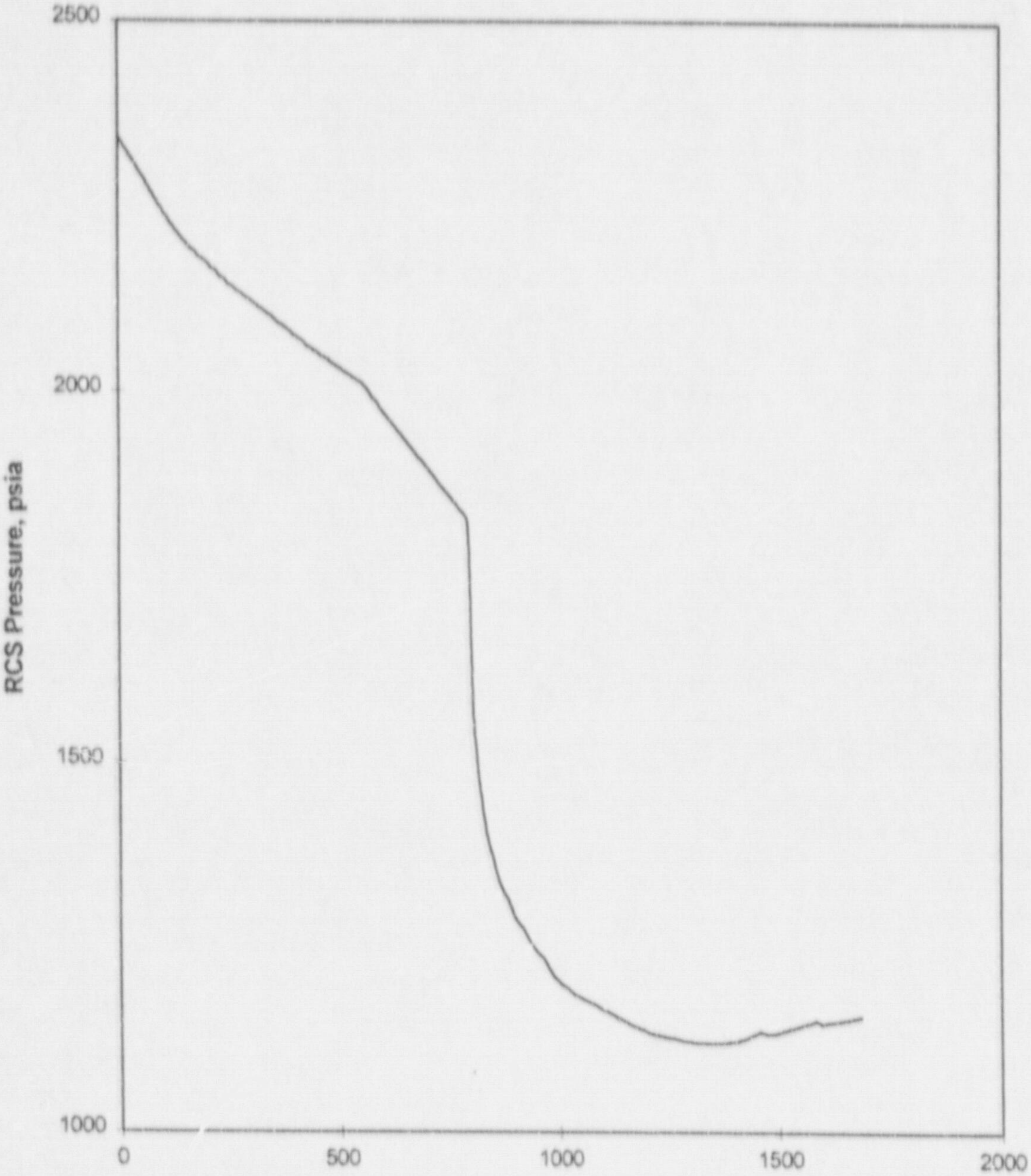


Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3- 1A



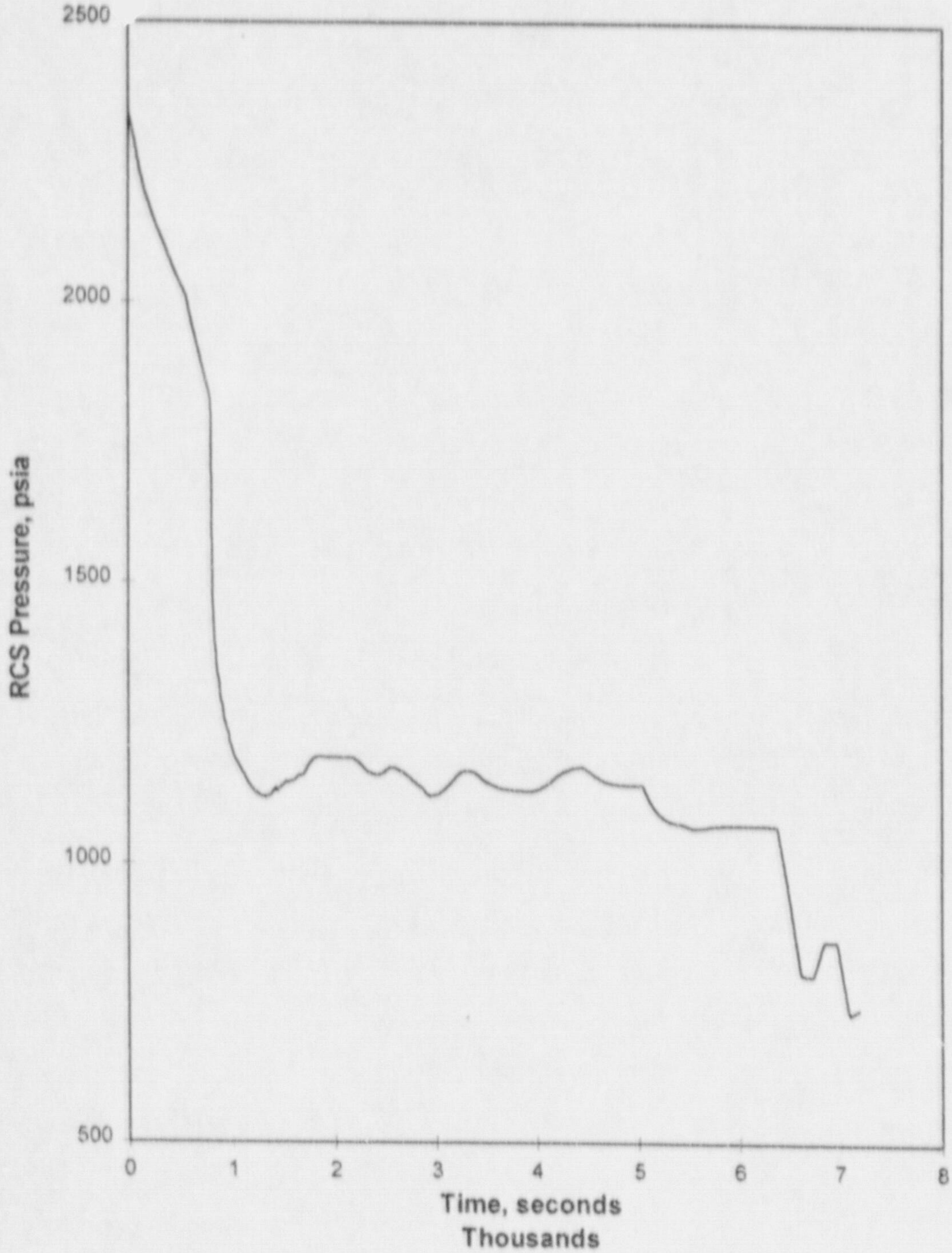
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.31-B

Reactor Coolant System Pressure vs. Time



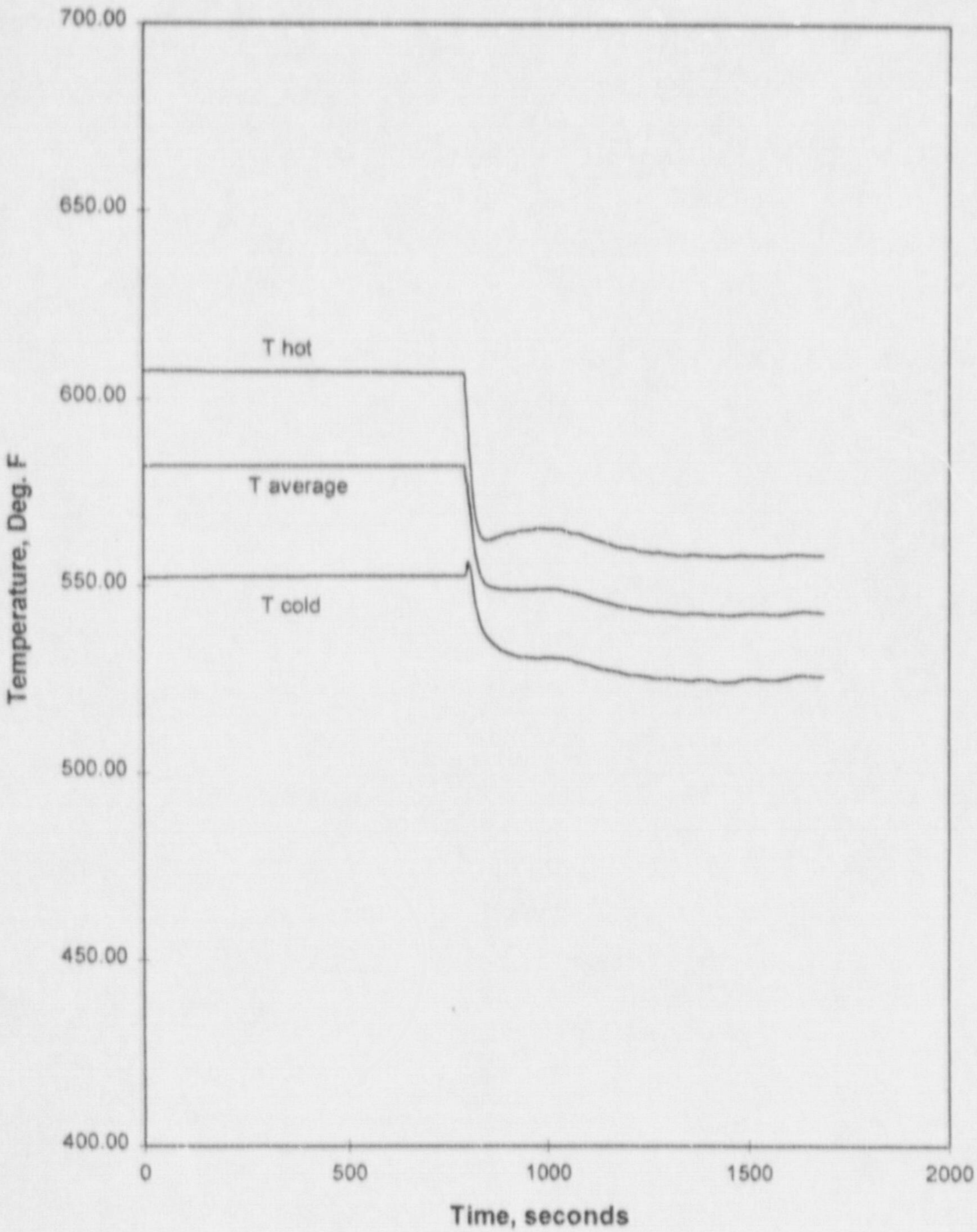
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3- 2-A

Reactor Coolant System Pressure vs. Time

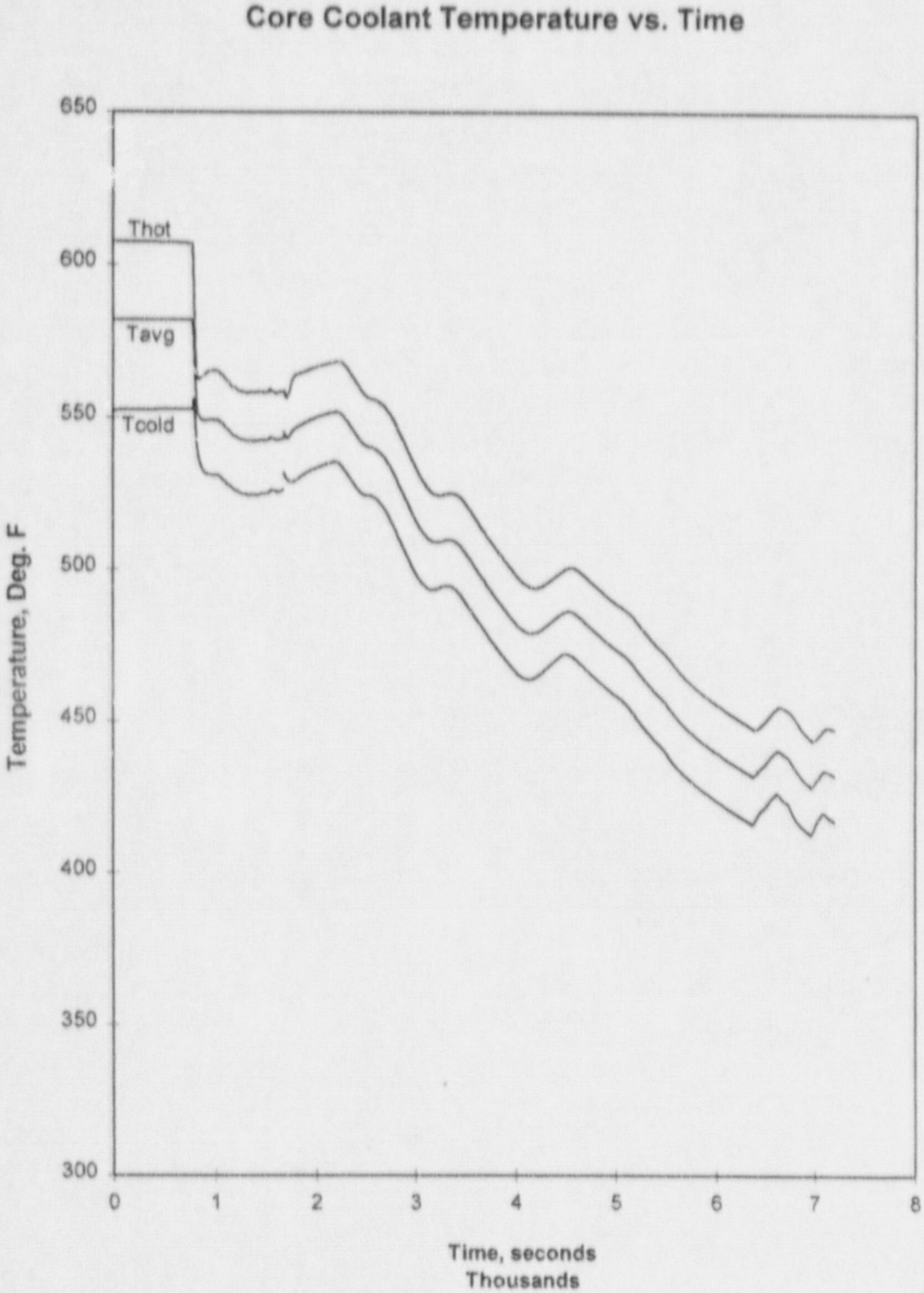


Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-2-B

Core Coolant Temperature vs. Time

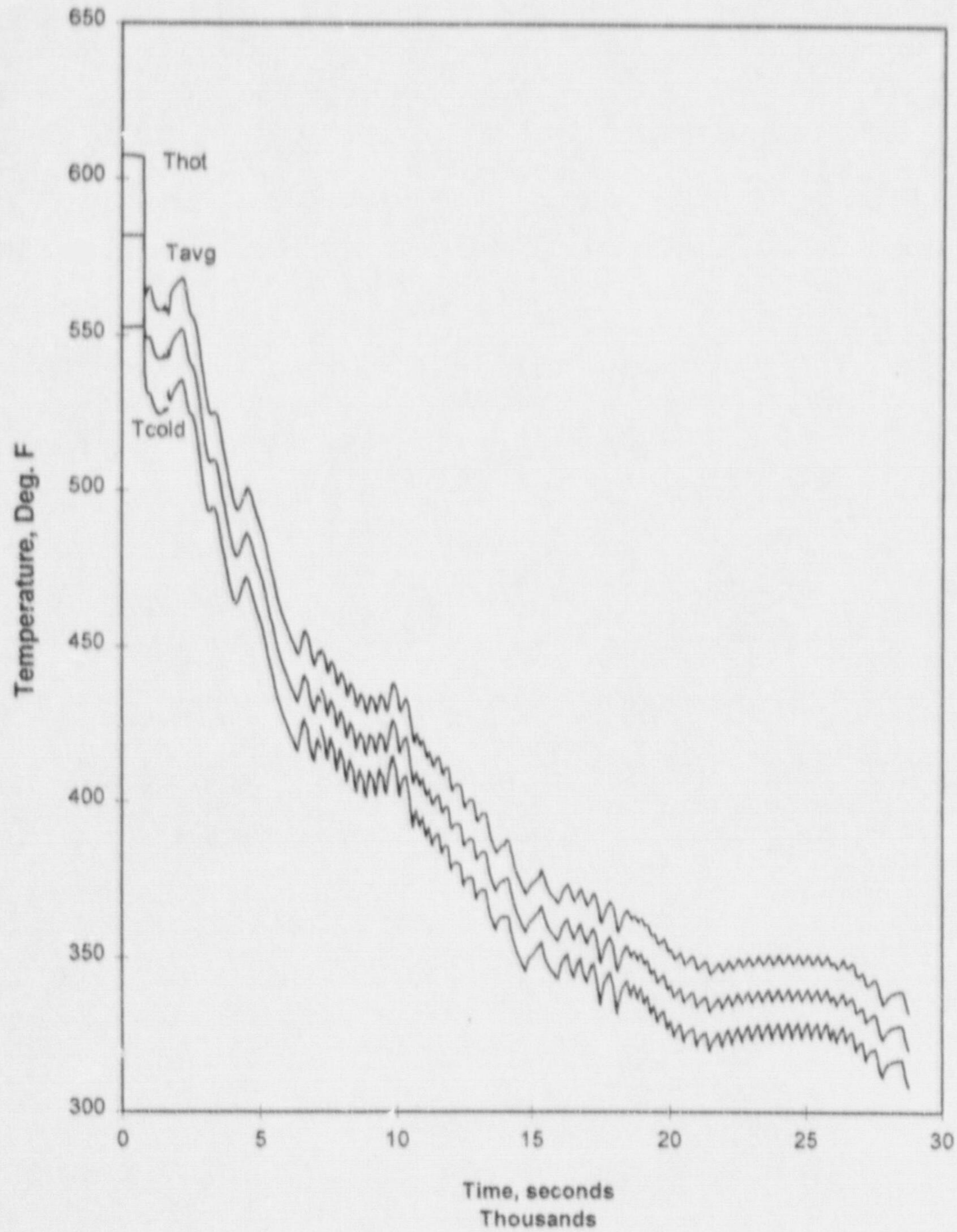


Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3- 3-A

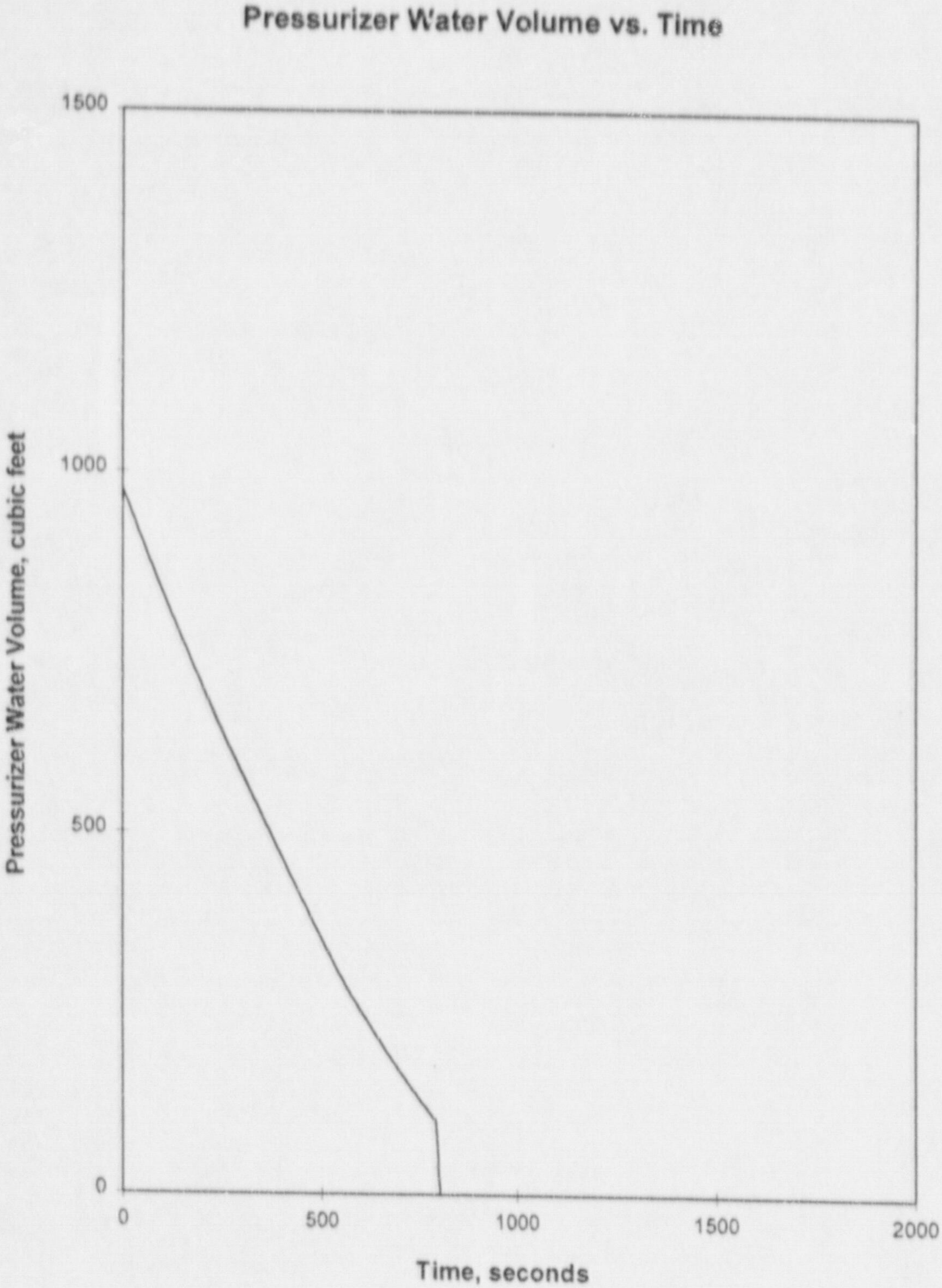


Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-3-B

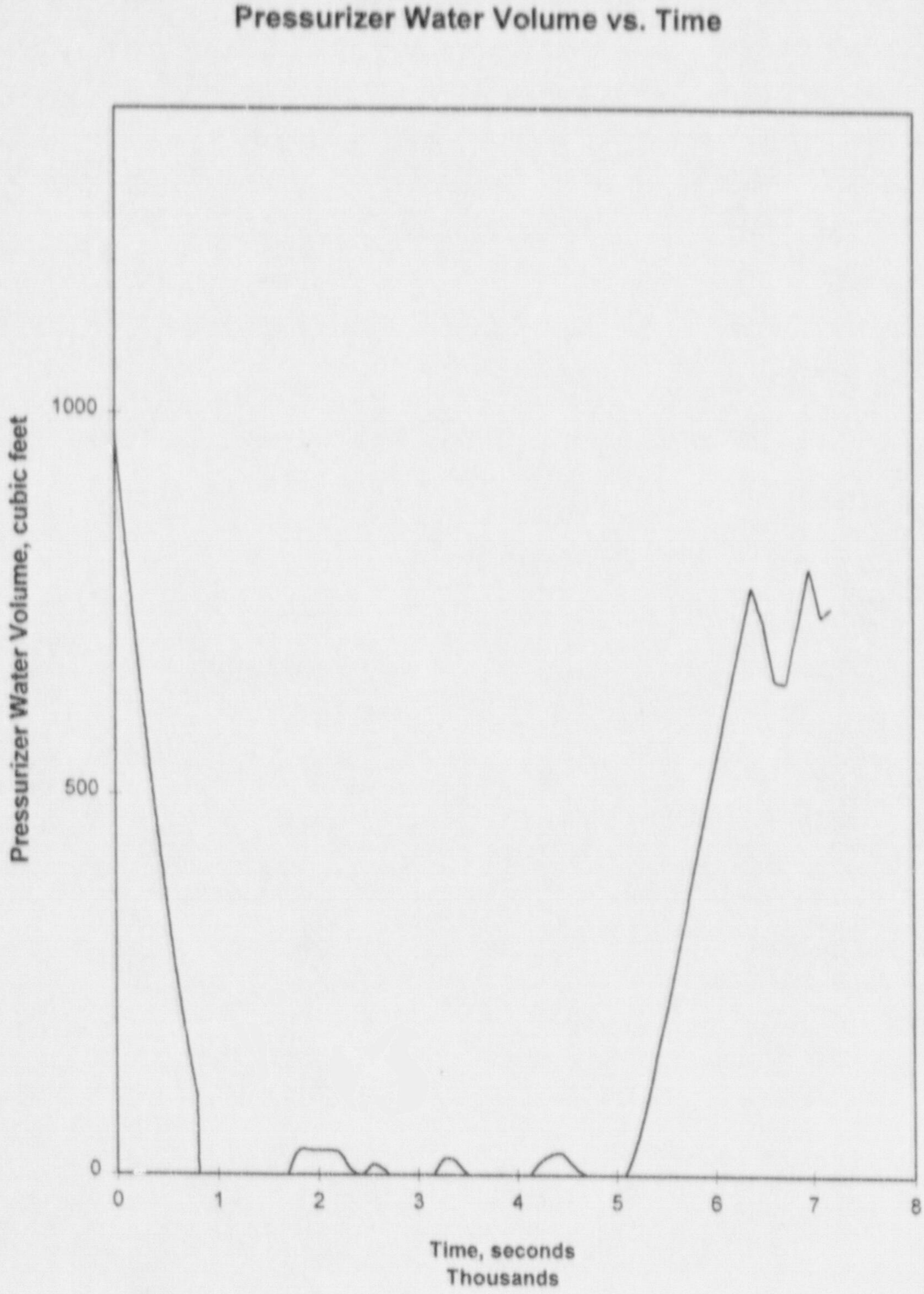
Core Coolant Temperature vs. Time



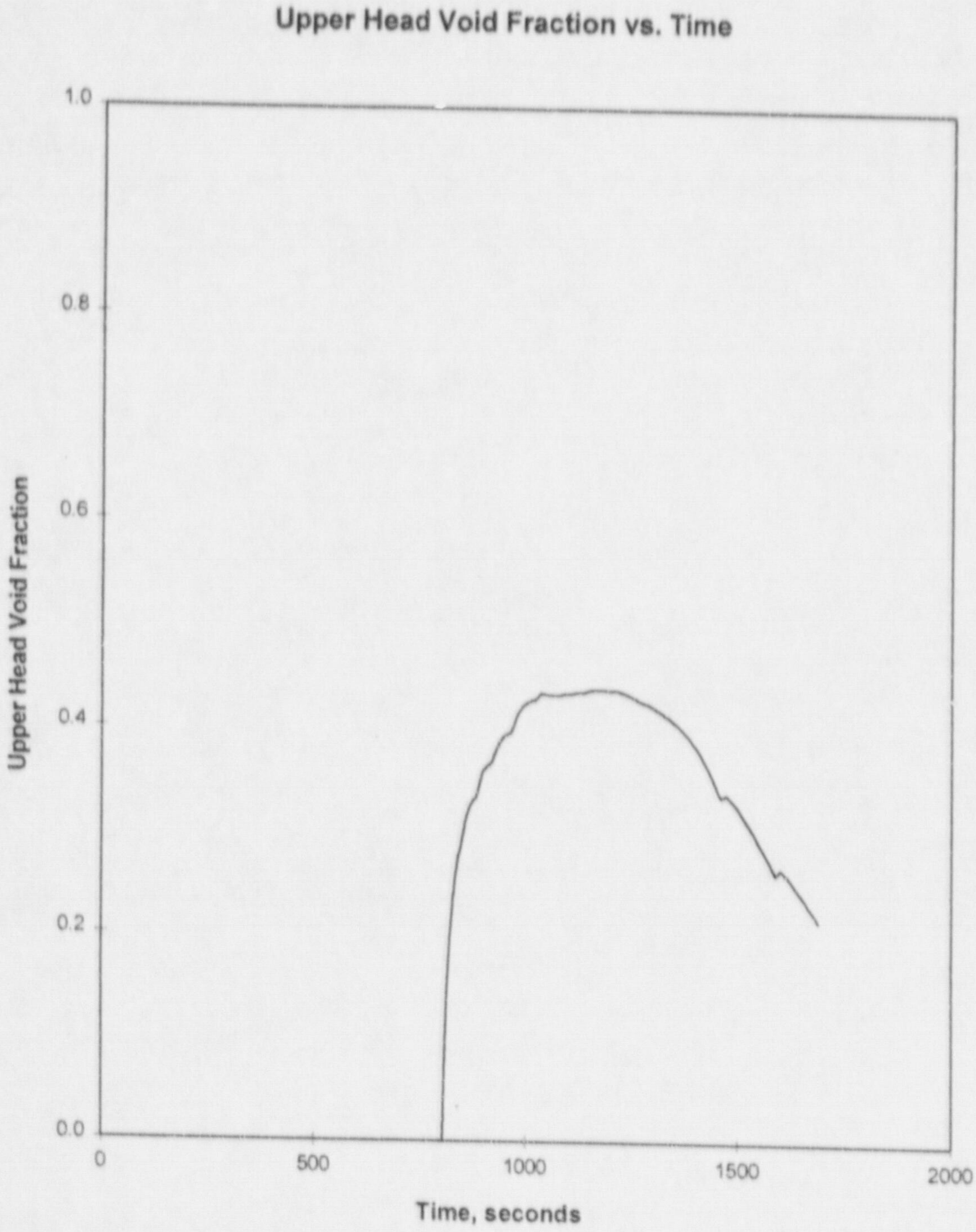
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-3-C



Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-4-A

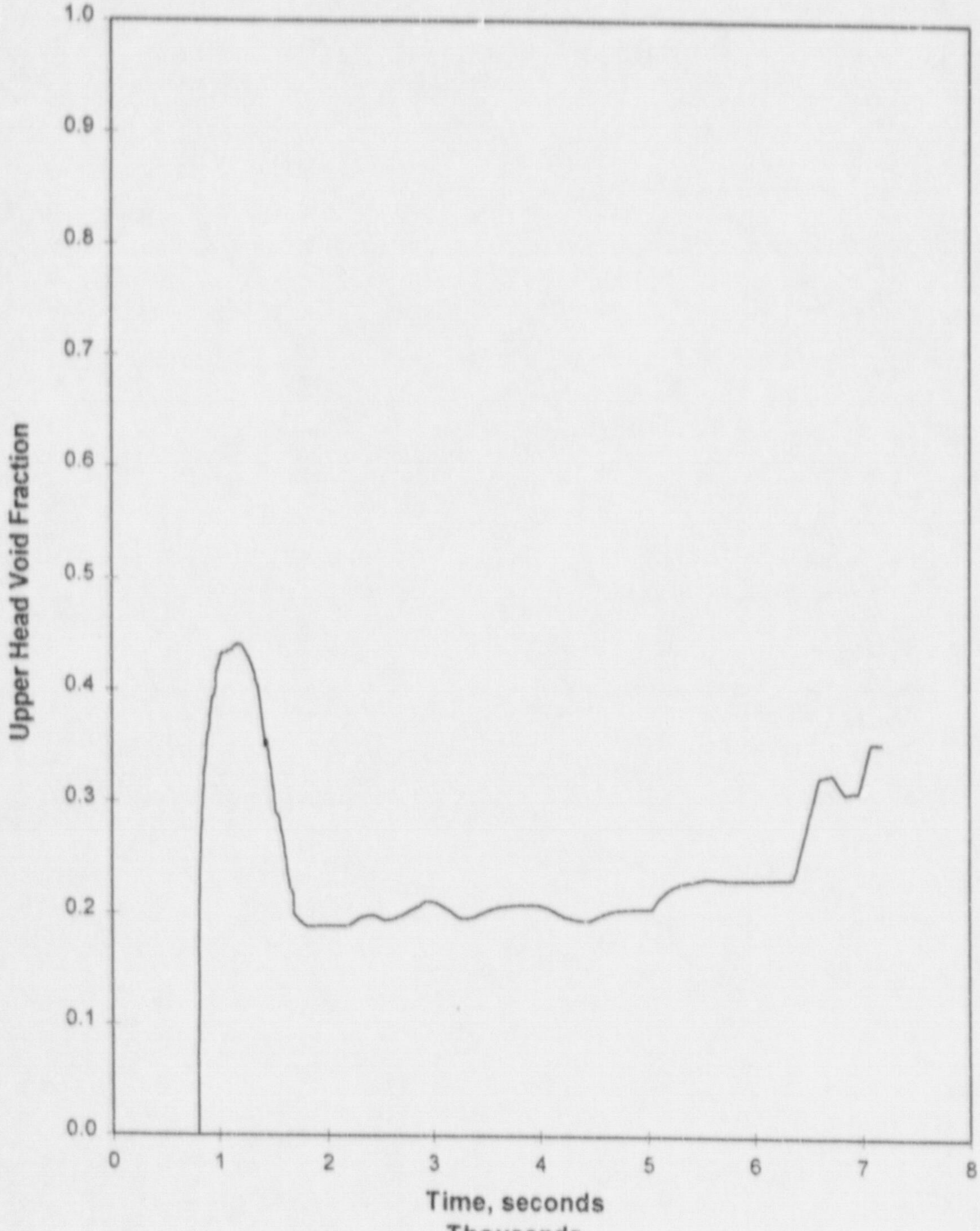


Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-4-B



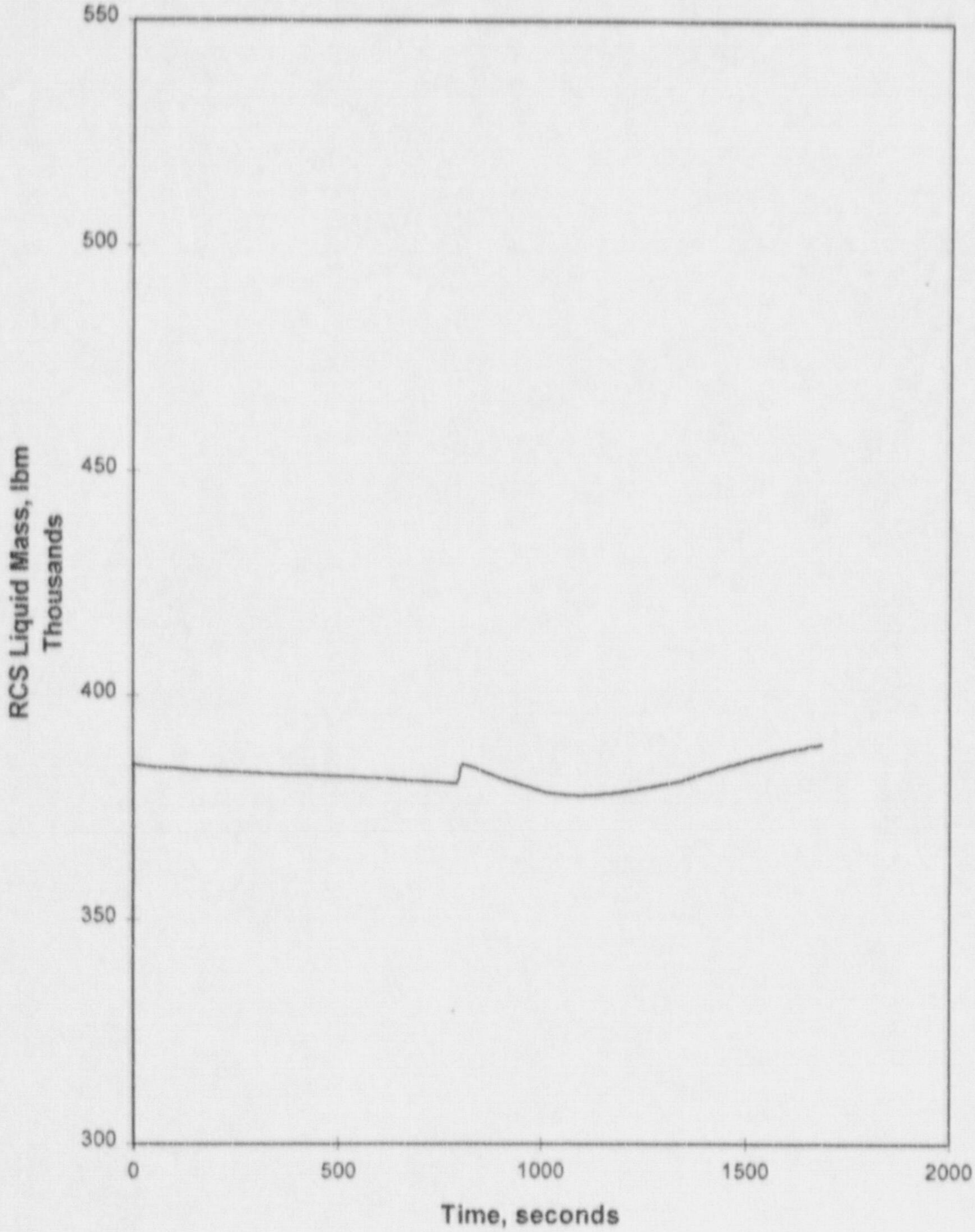
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-5-A

Upper Head Void Fraction vs. Time

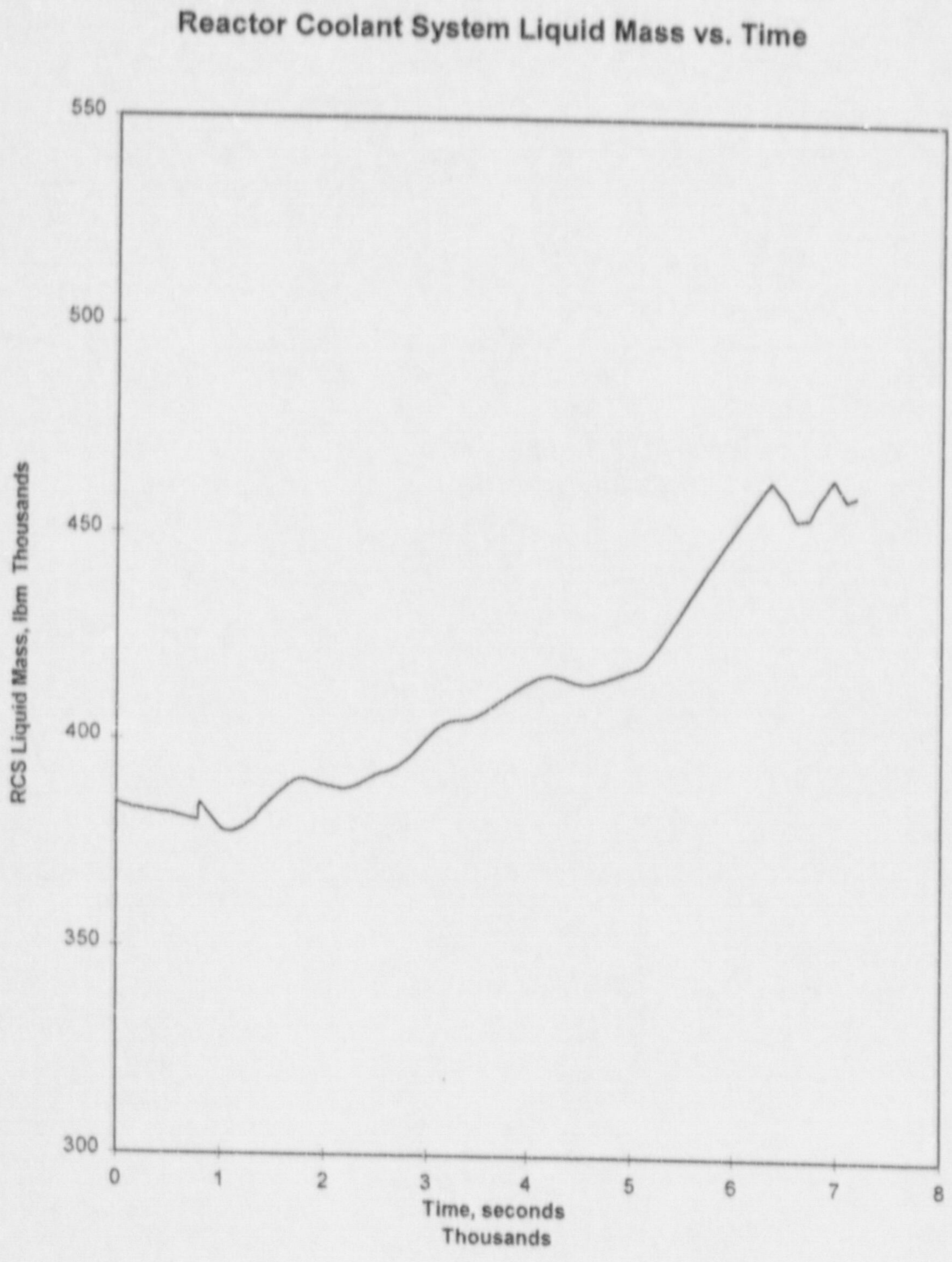


Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-5-B

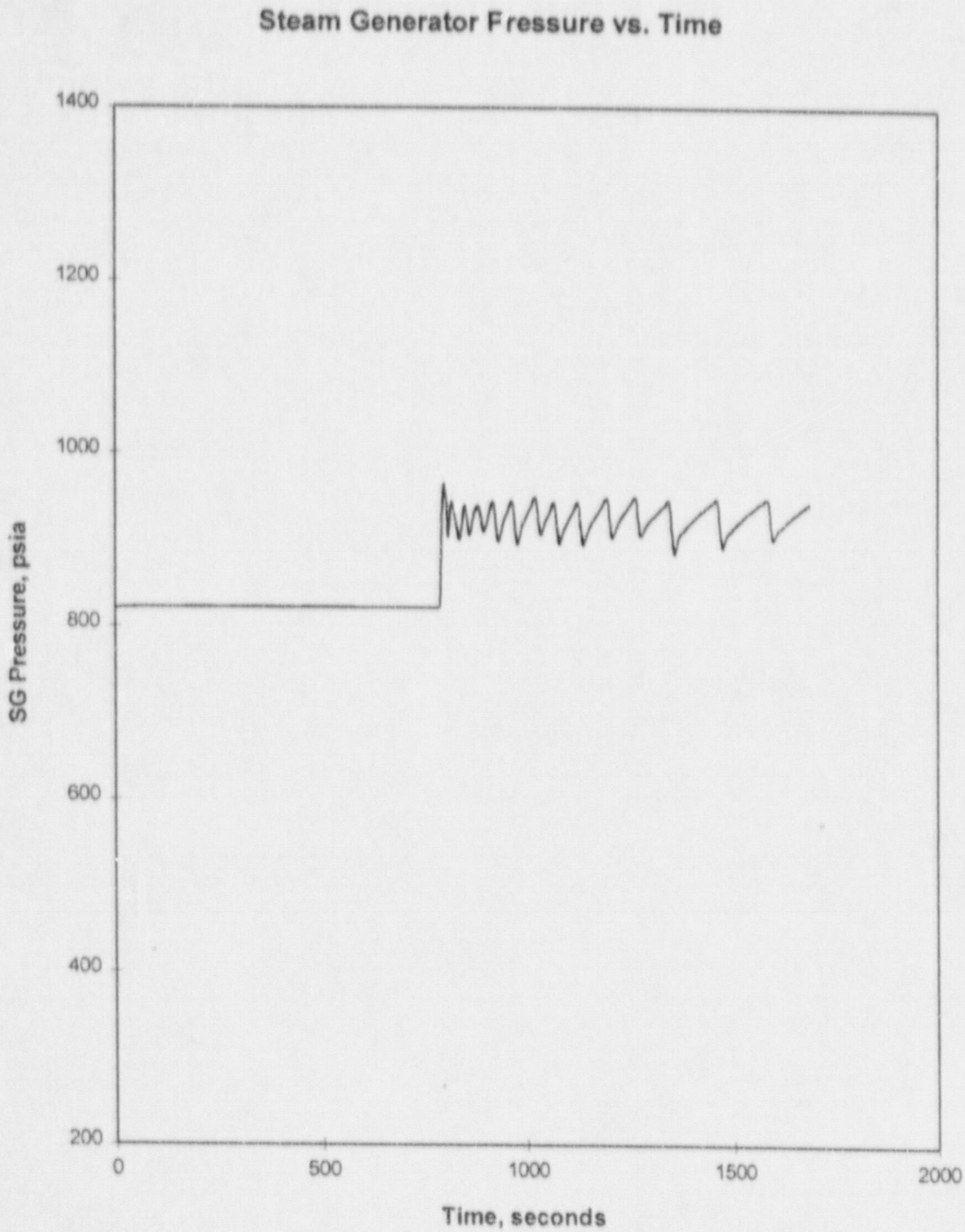
RCS Liquid Mass vs. Time



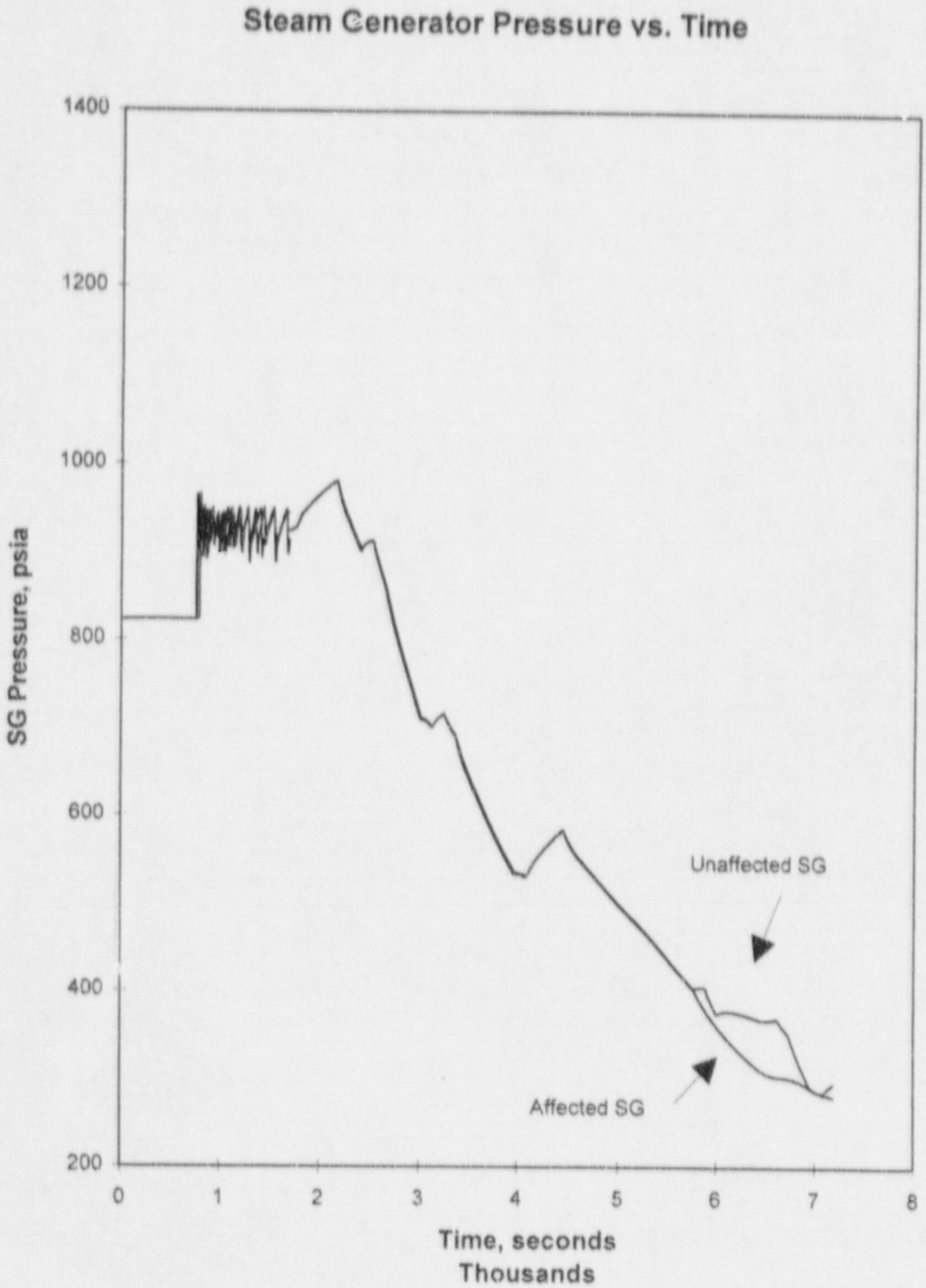
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-6A



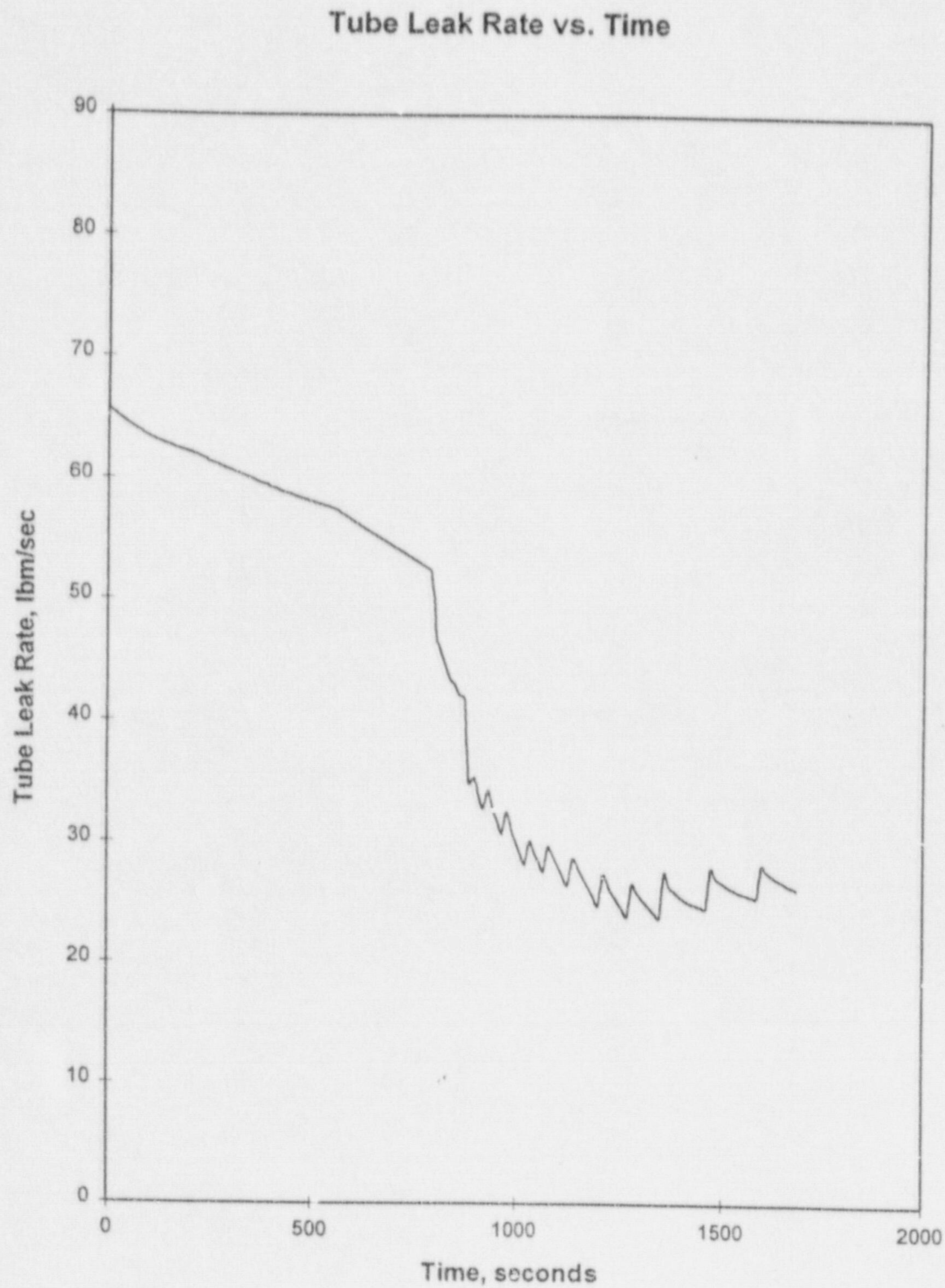
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-6-B



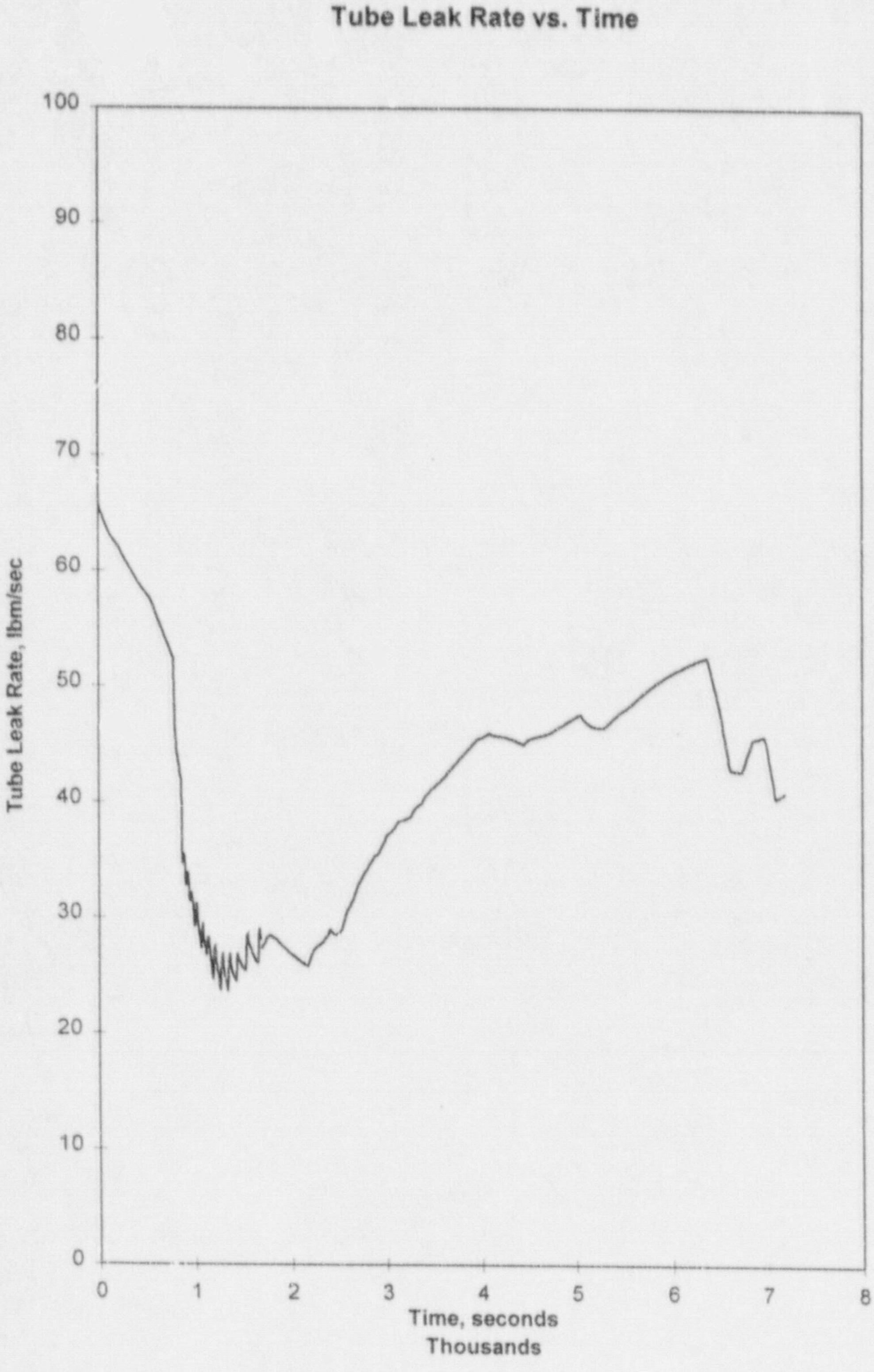
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-7A



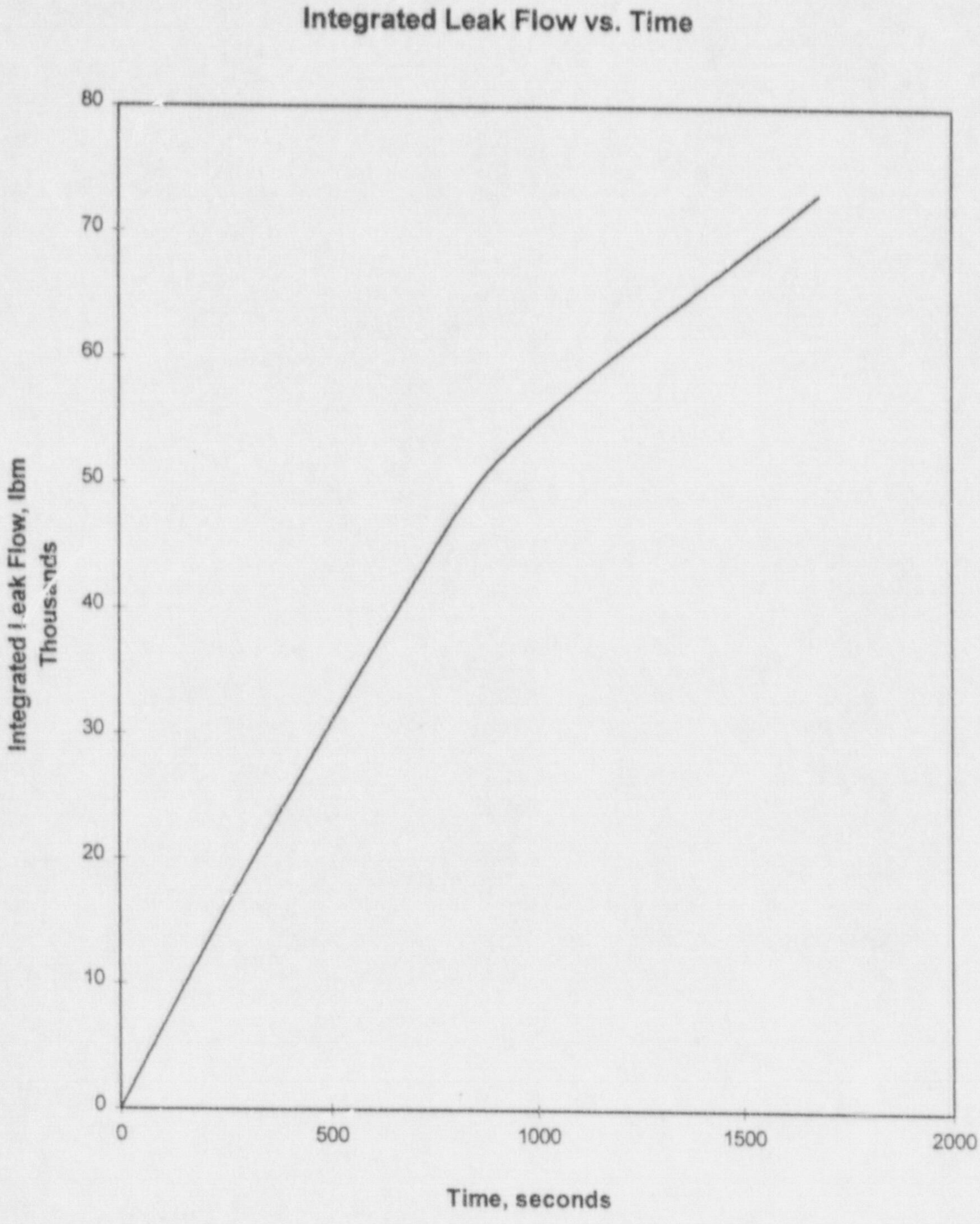
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-7-B



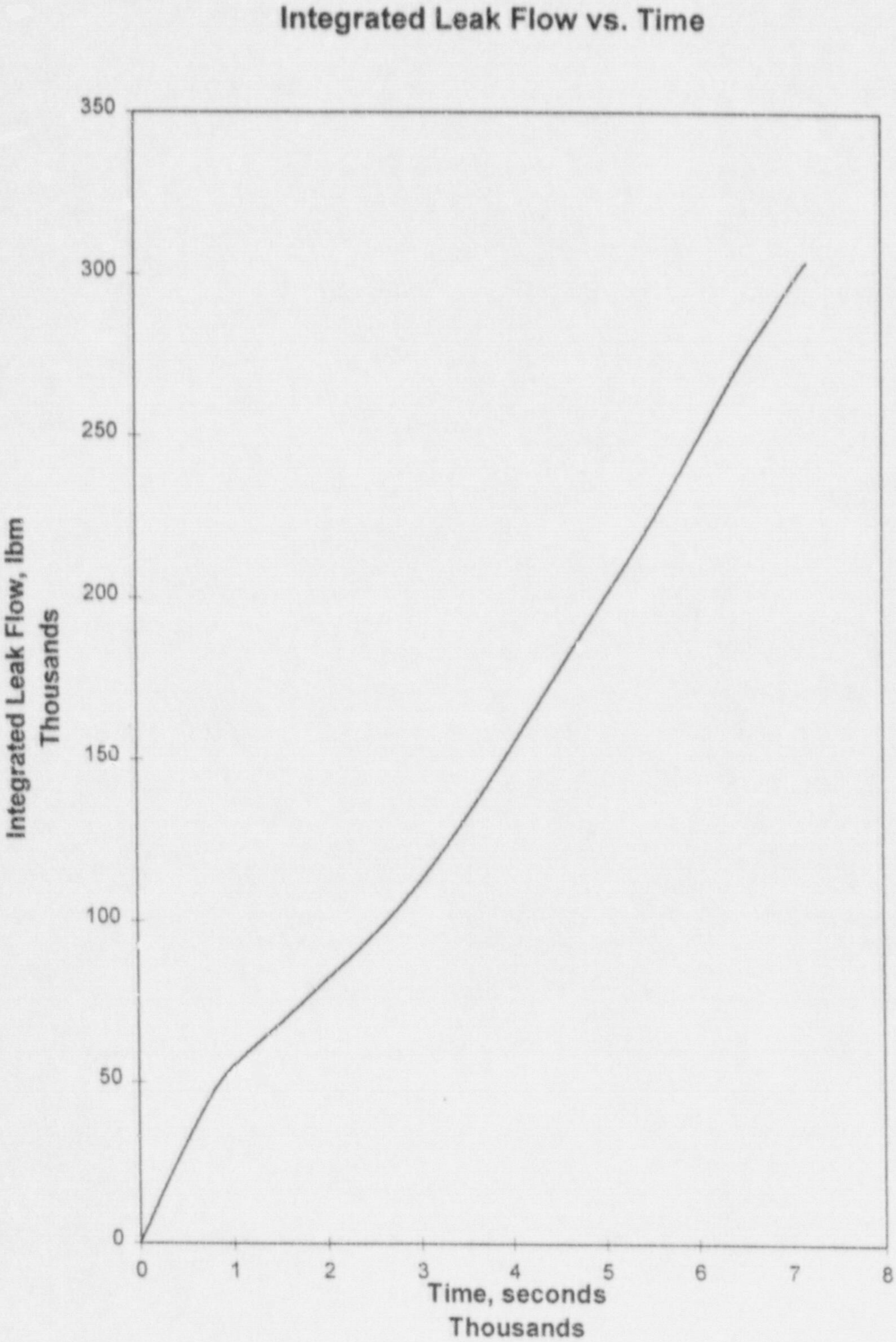
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-8-[illegible]



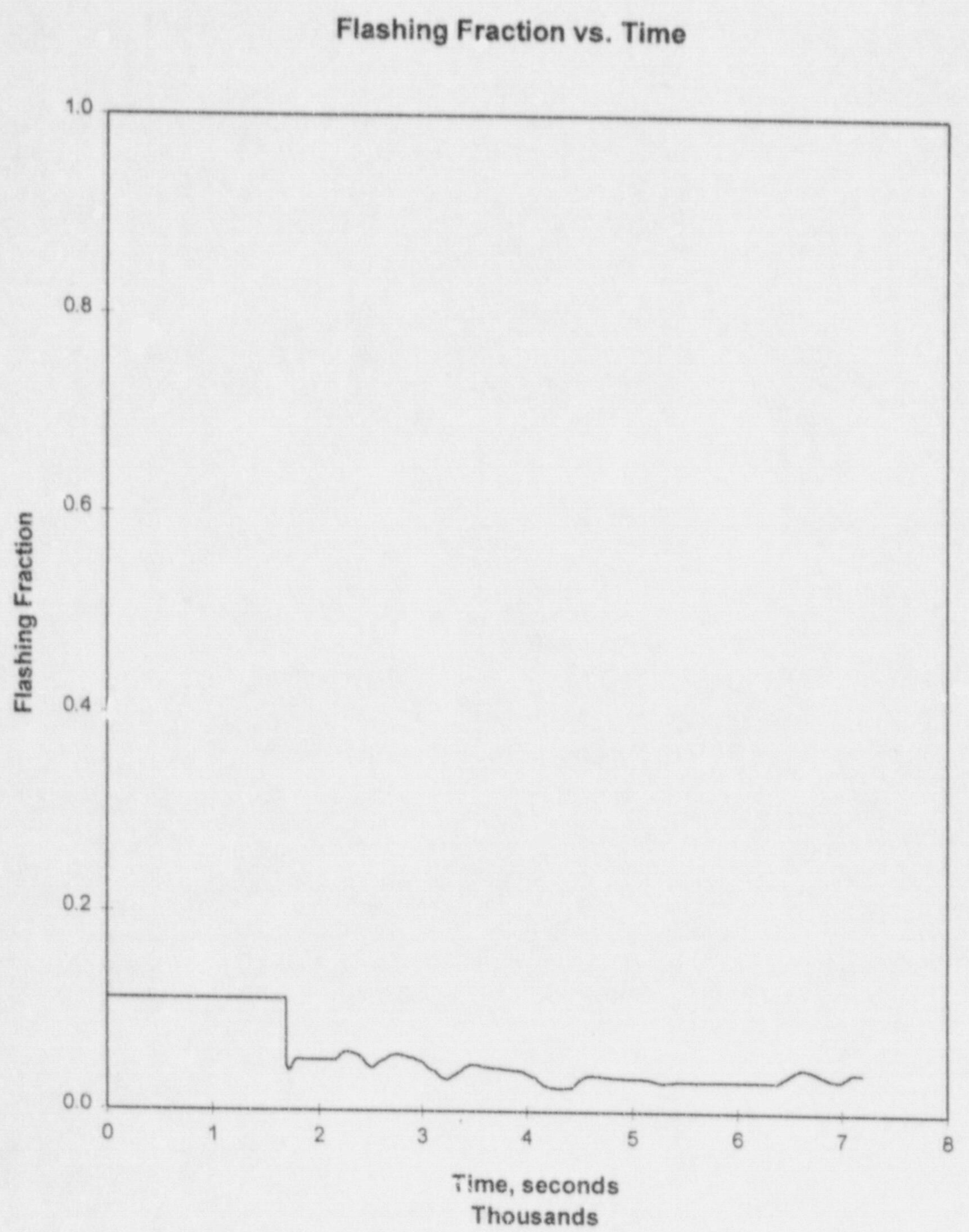
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-8-B



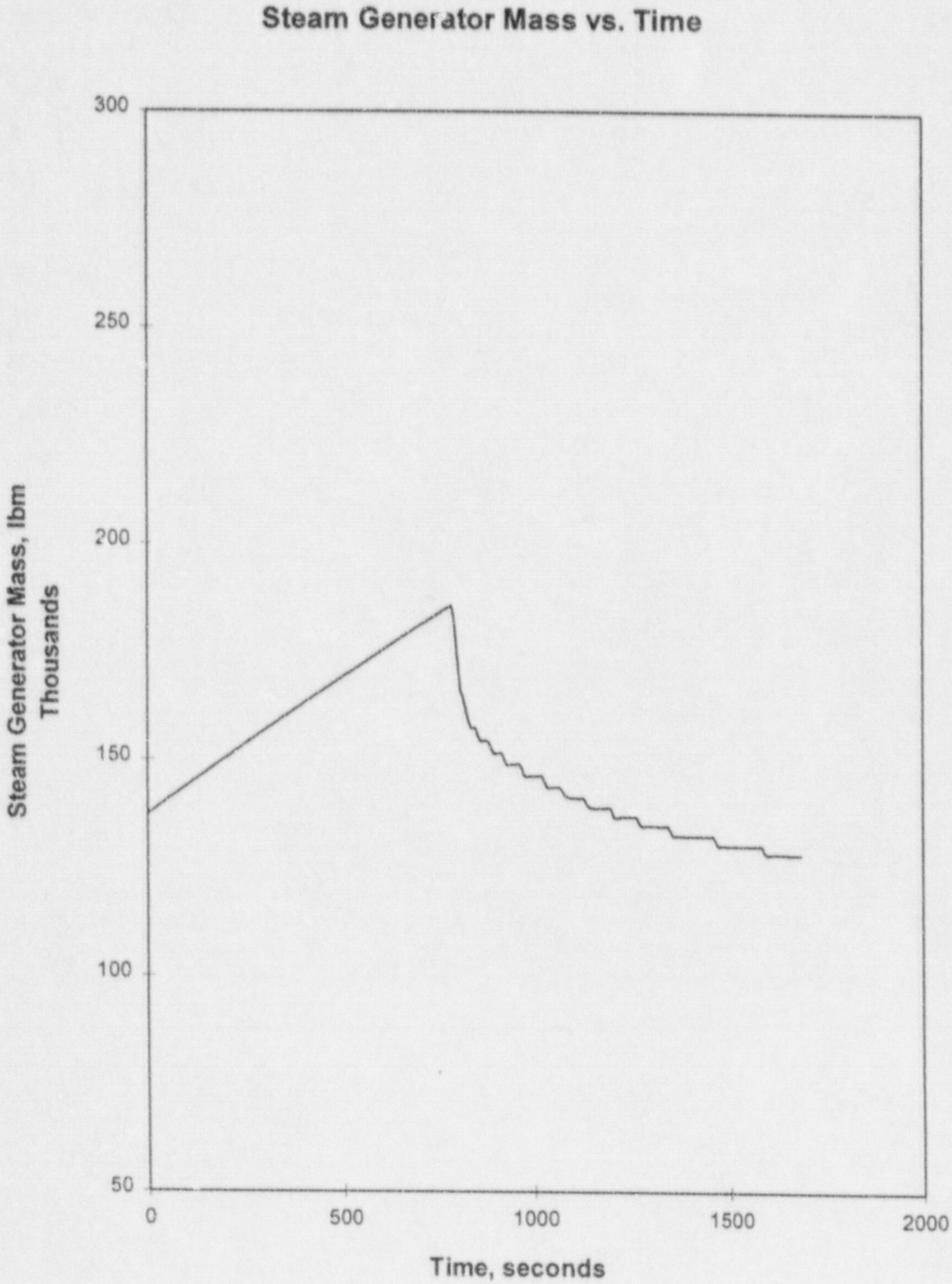
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-9A



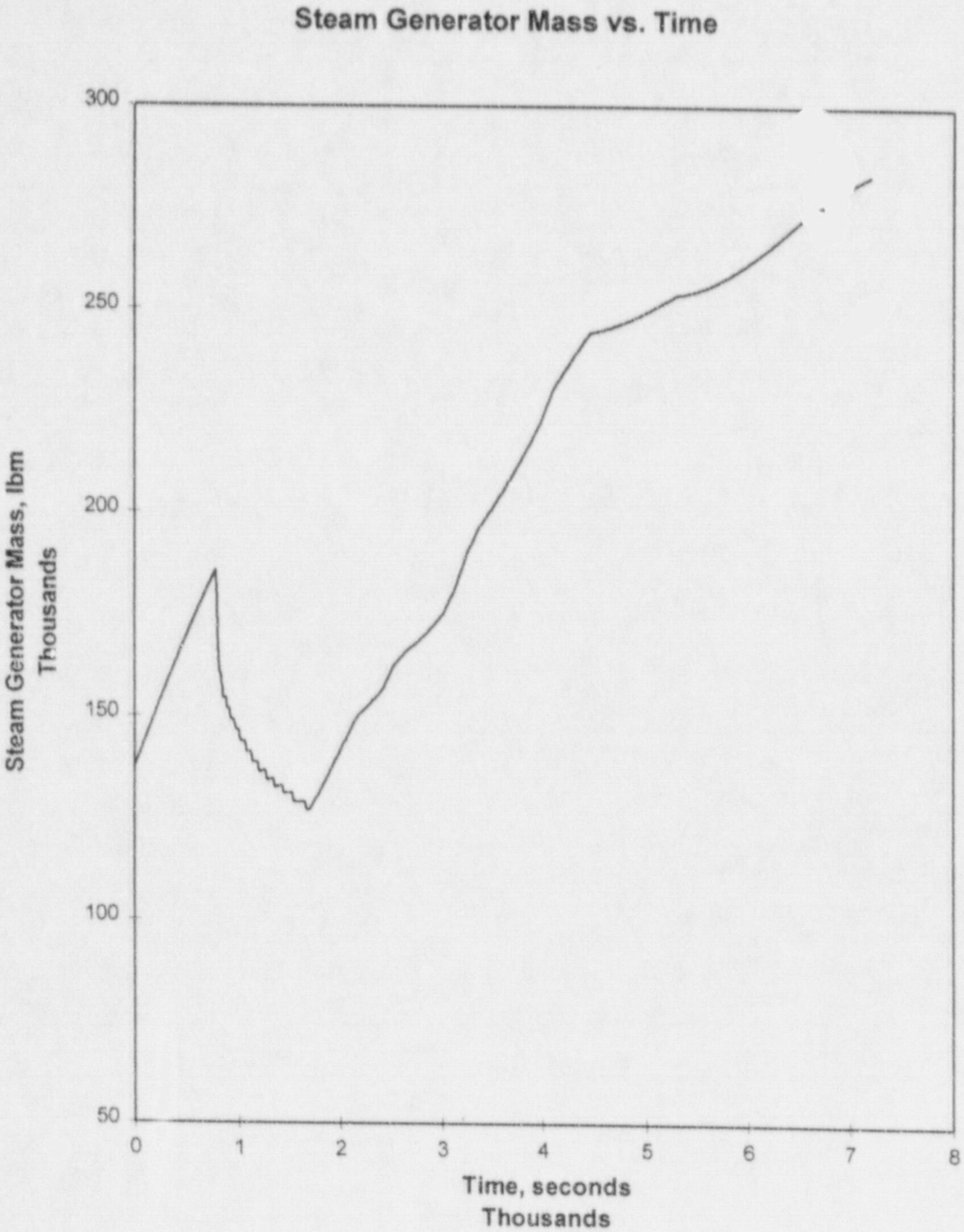
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-9-B



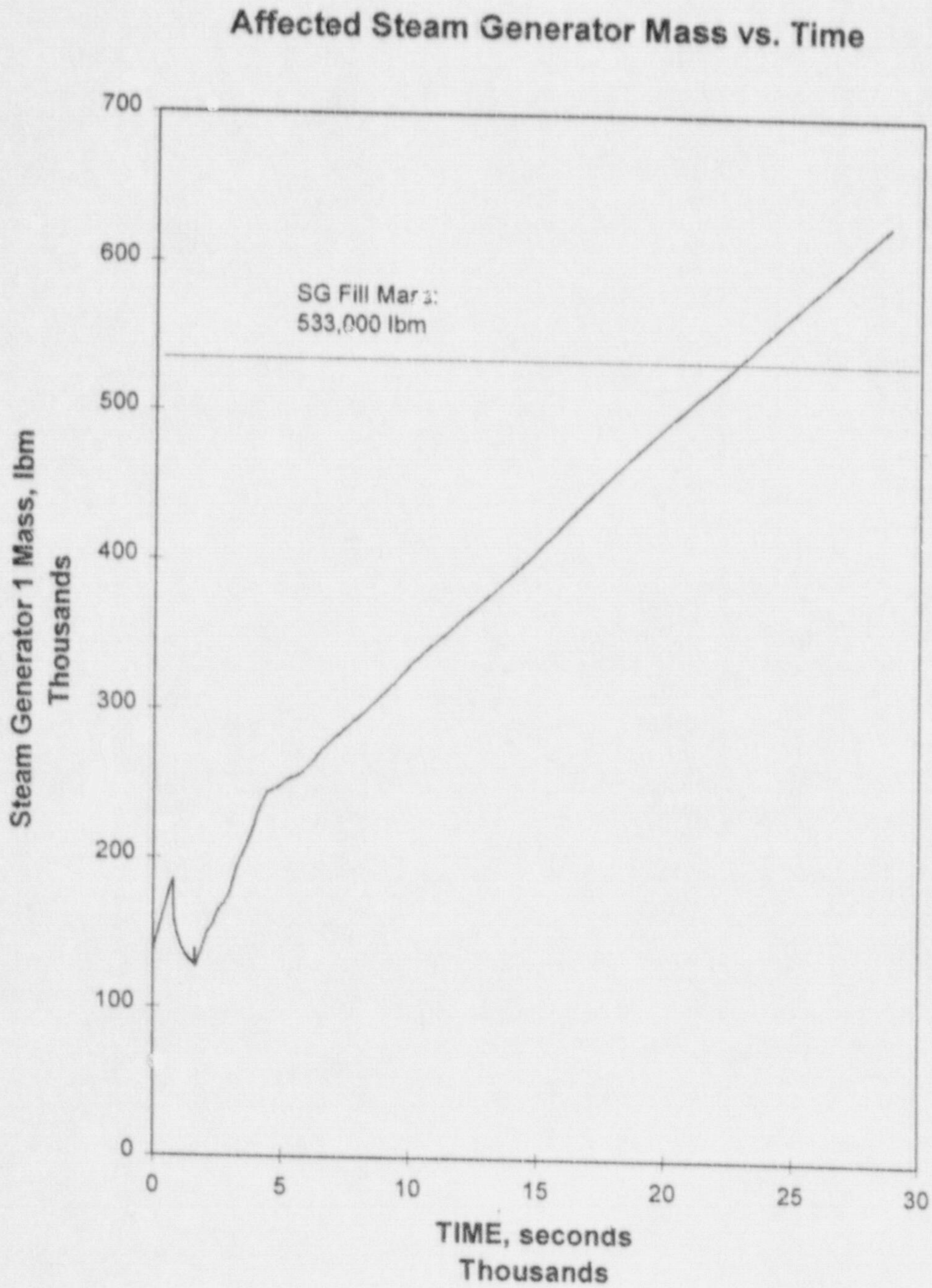
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-10



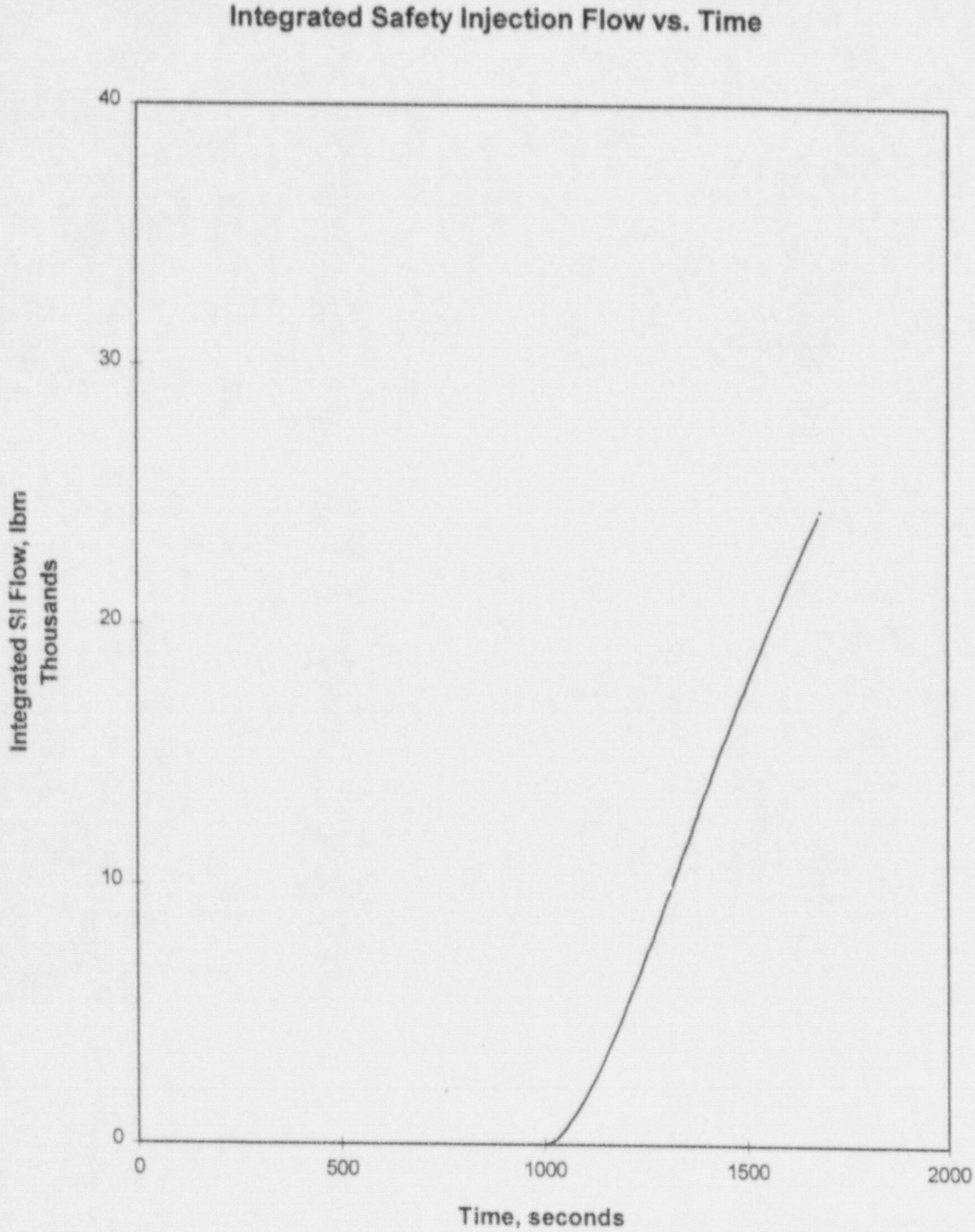
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-11-A



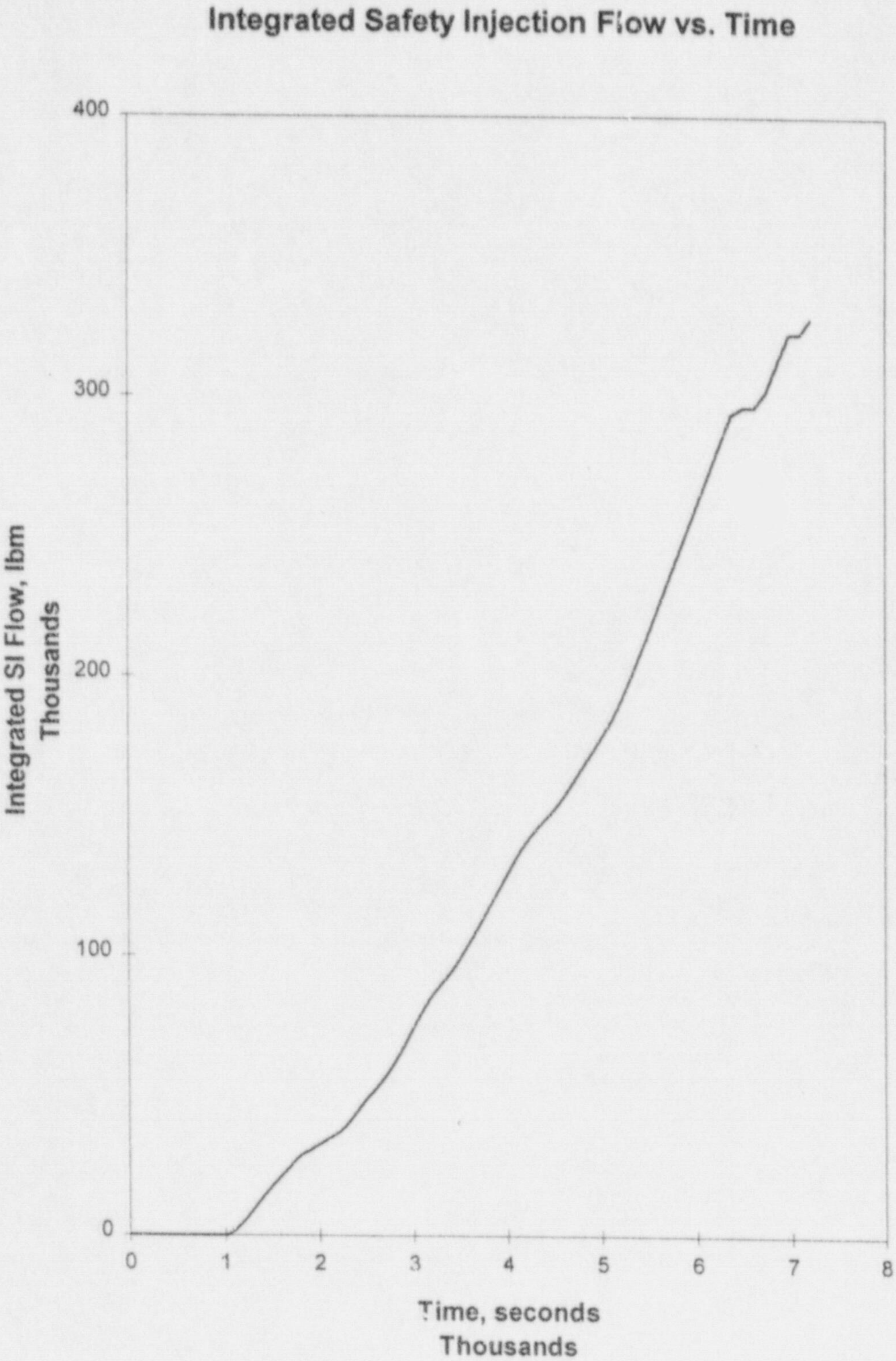
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-11-B



Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-11-C

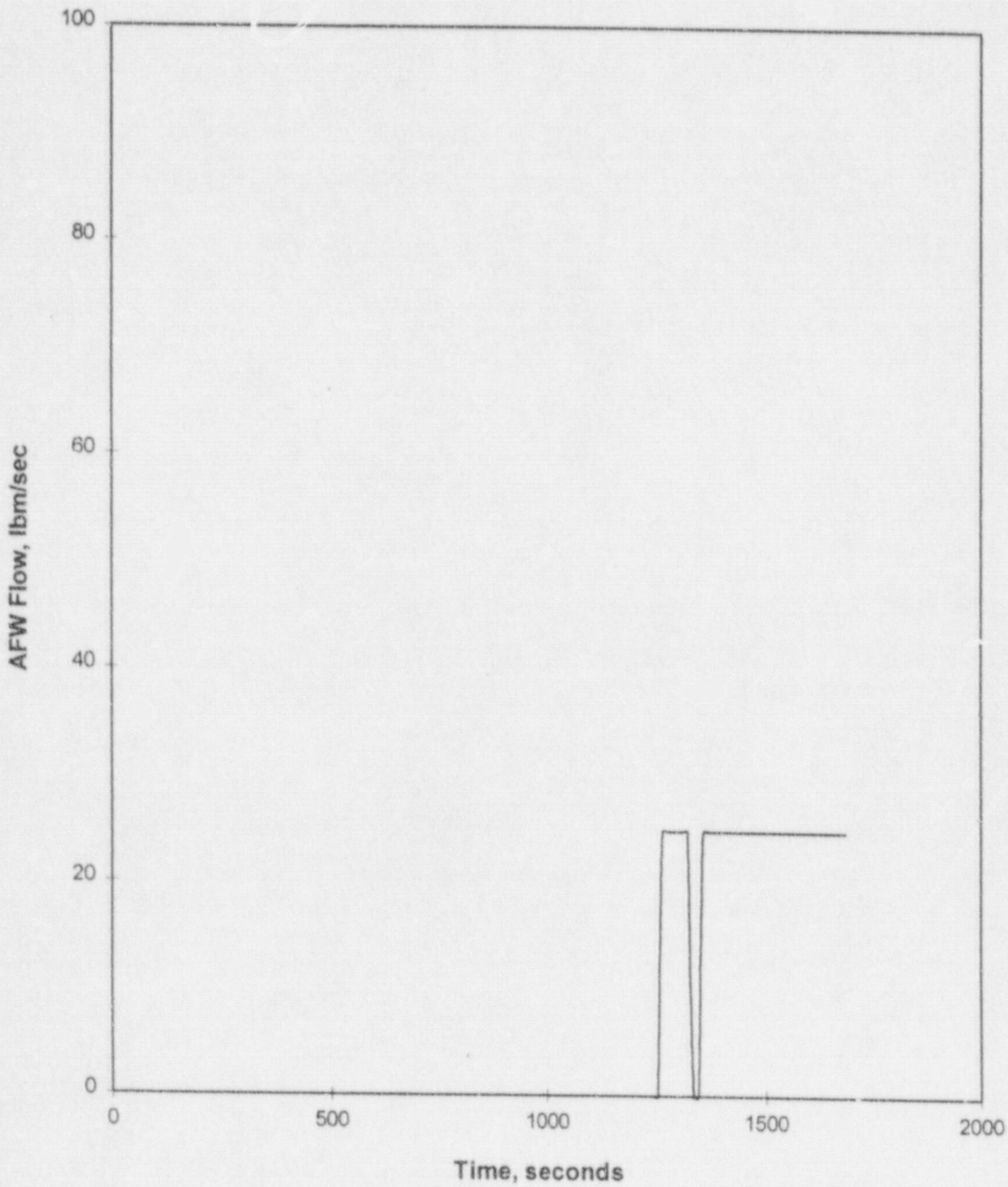


Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-12-A

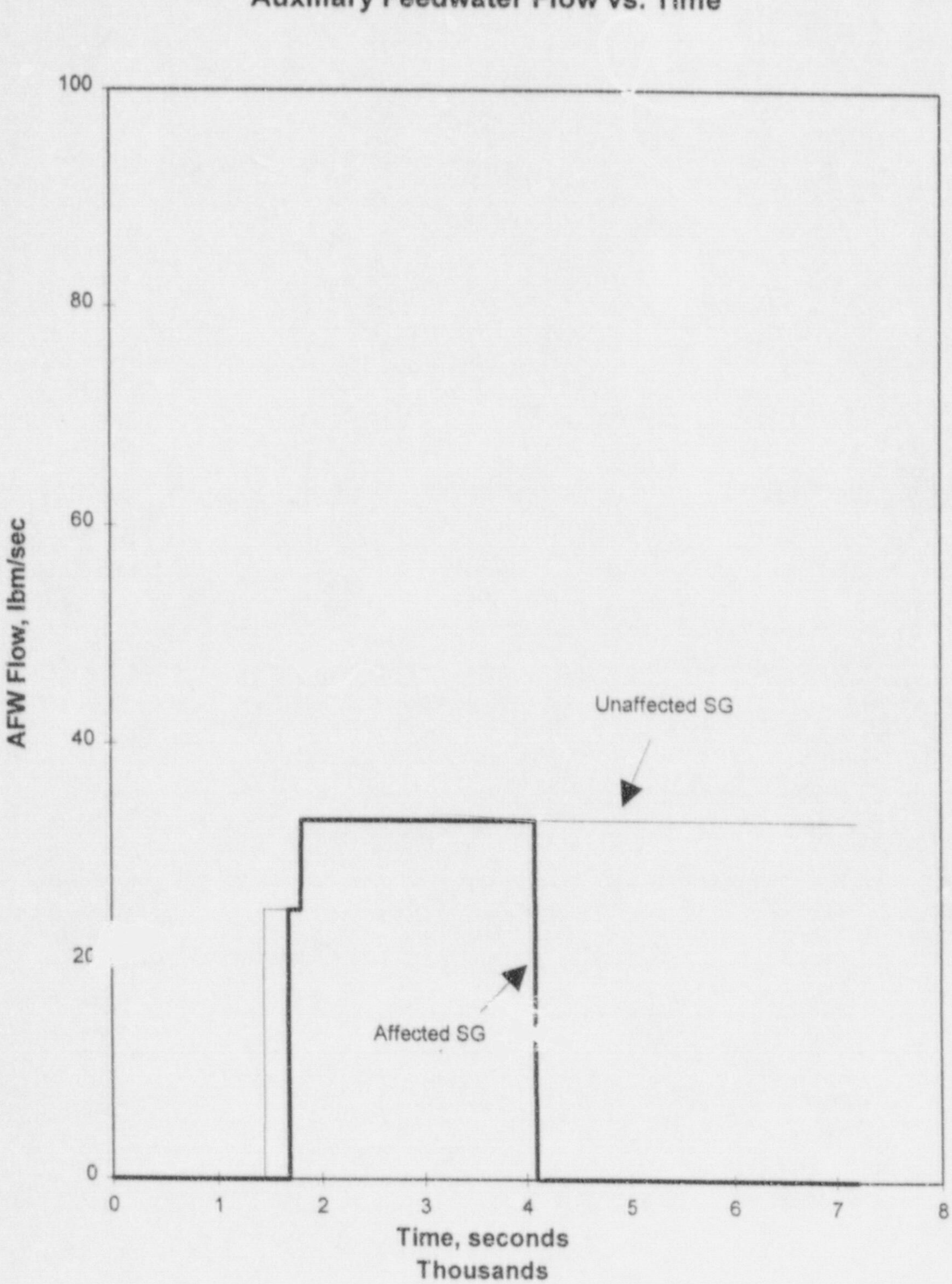


Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-12-B

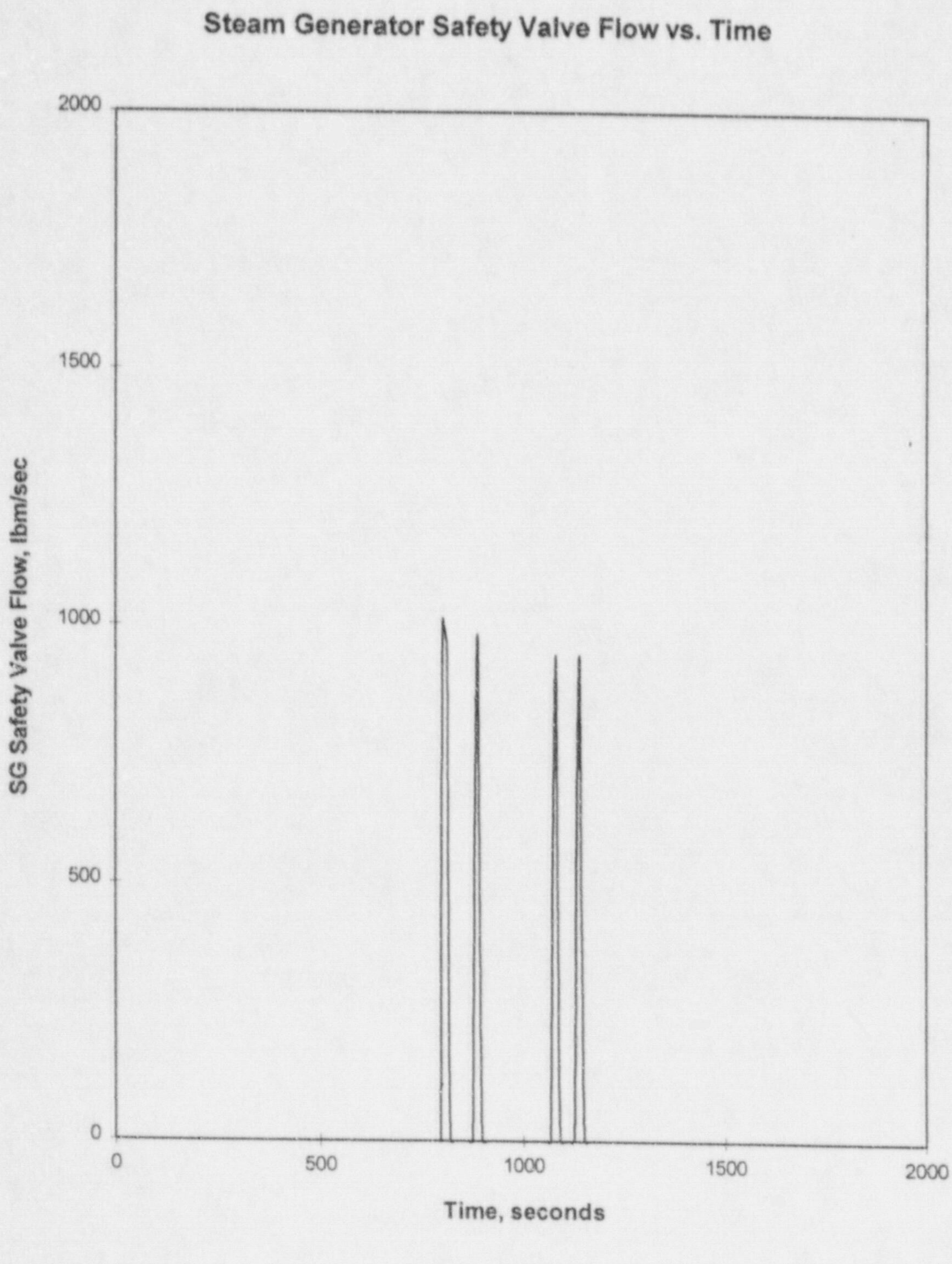
Auxiliary Feedwater Flow vs. Time



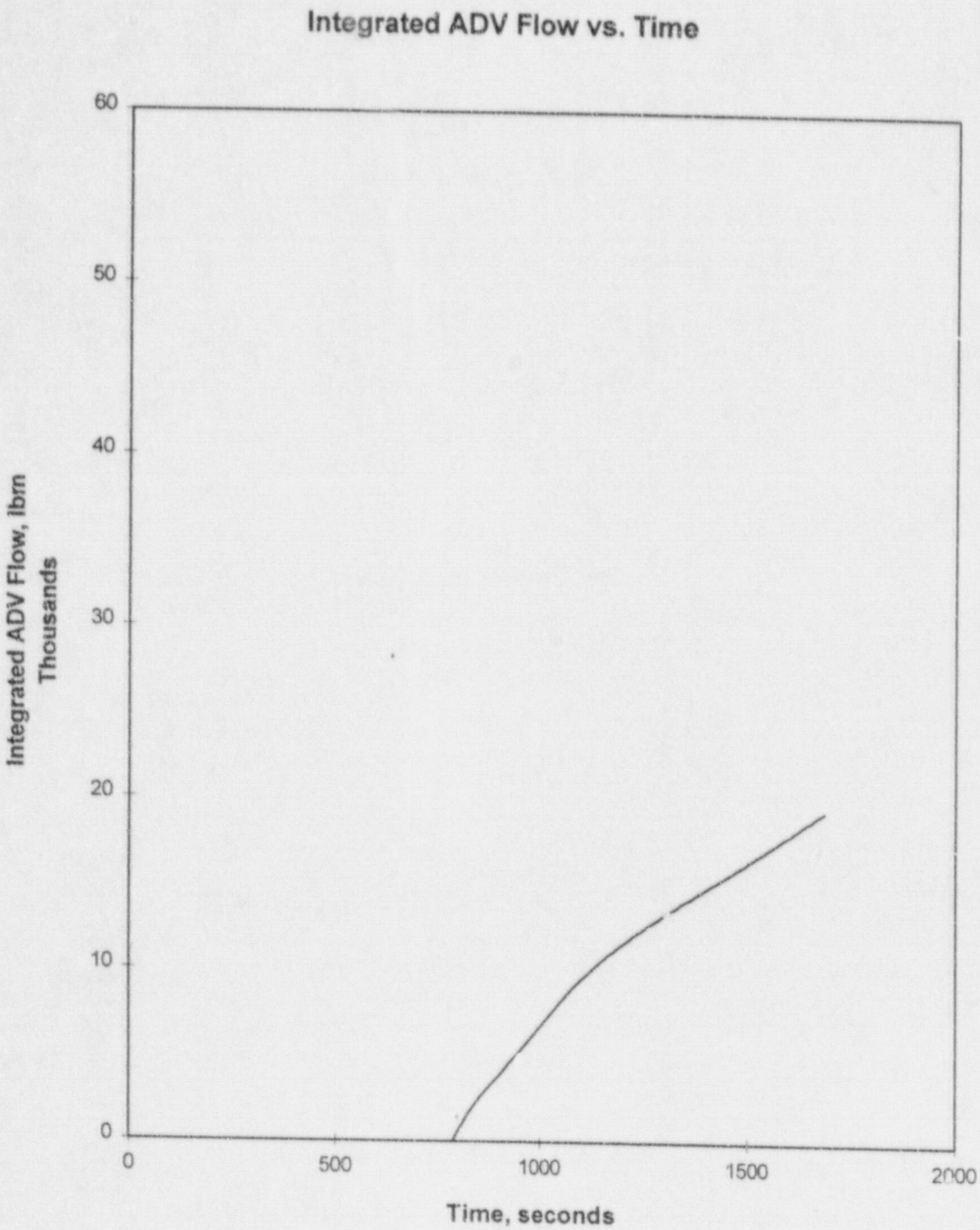
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-13-A



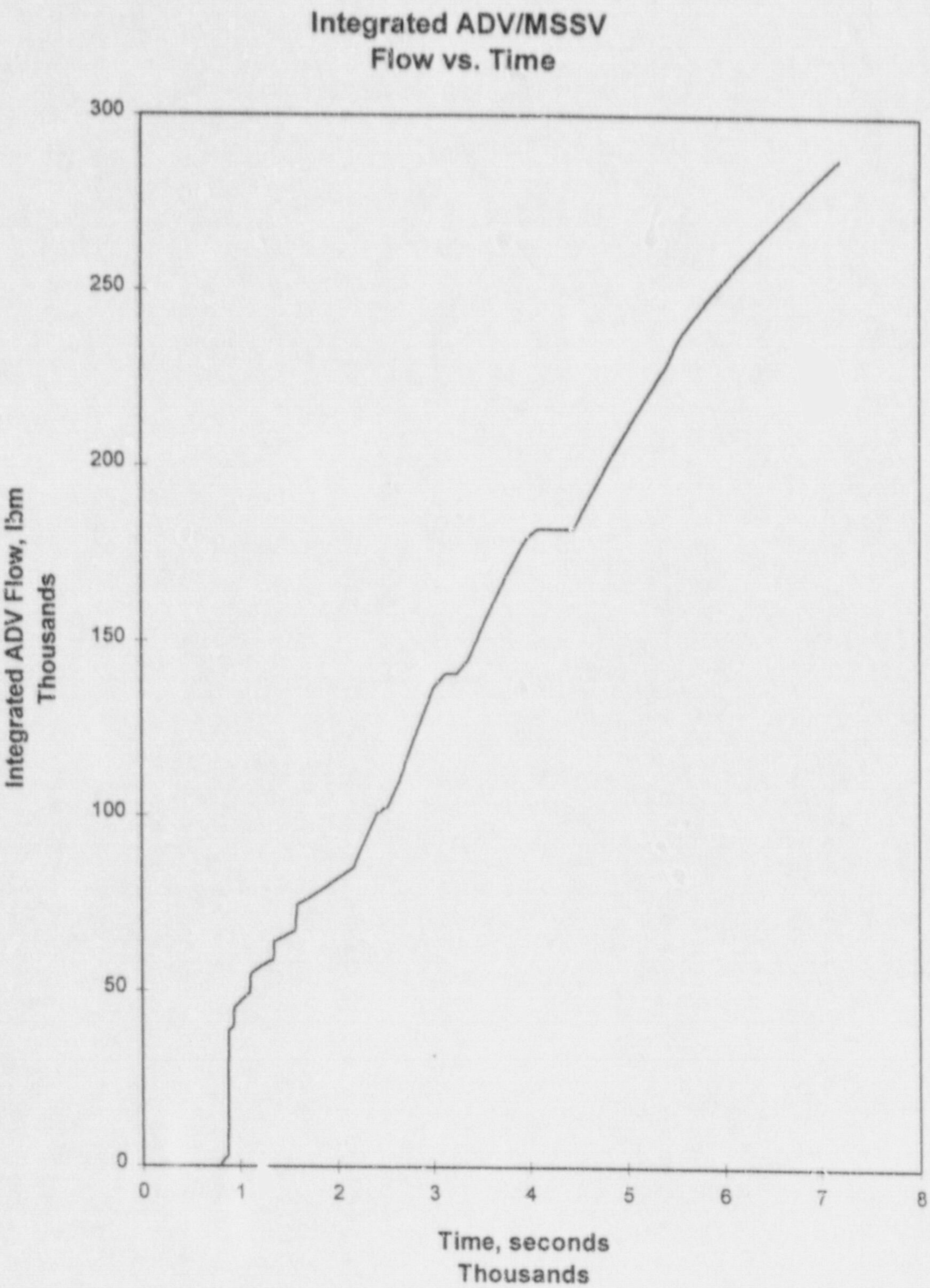
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-13-B



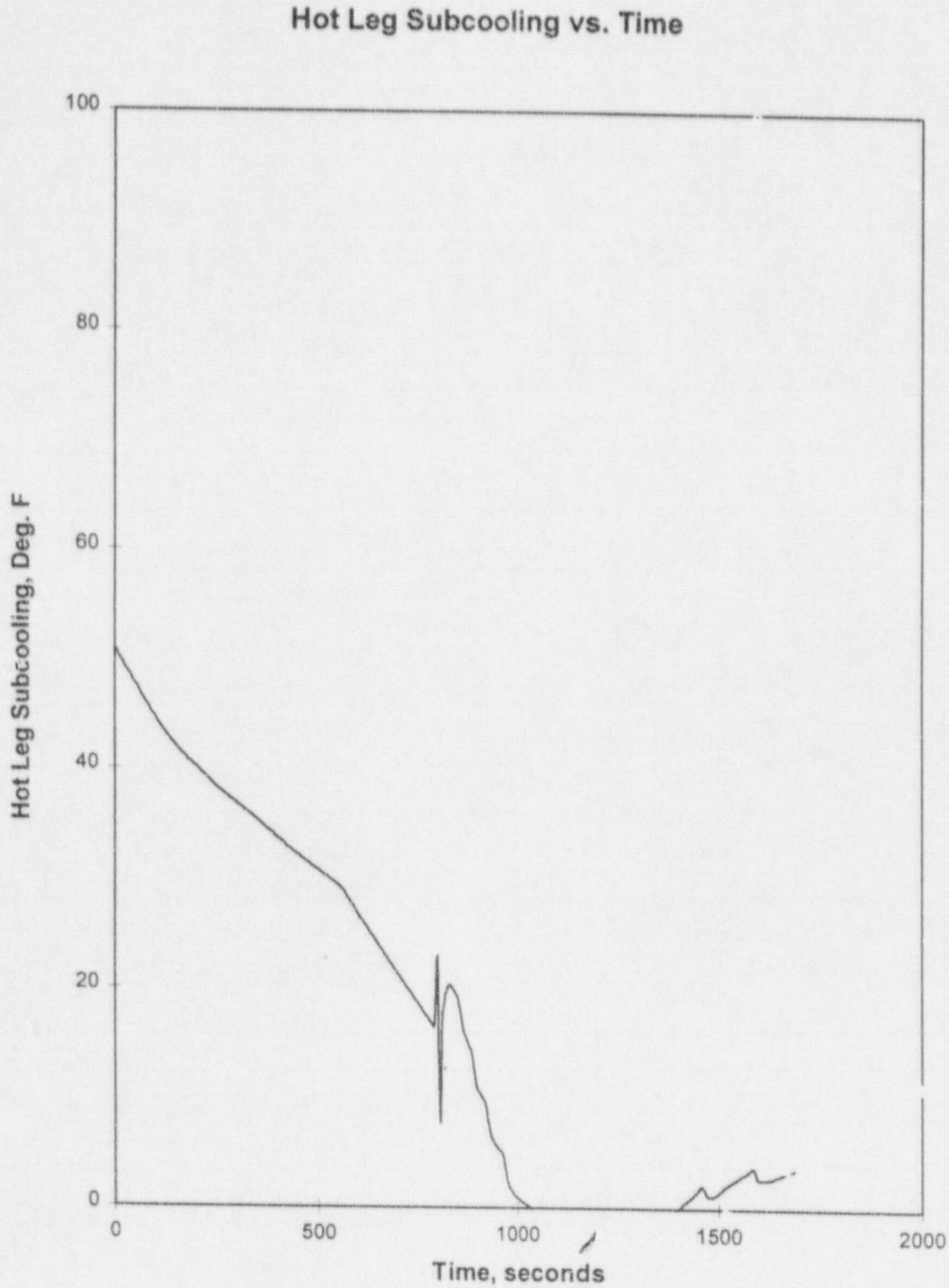
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-14



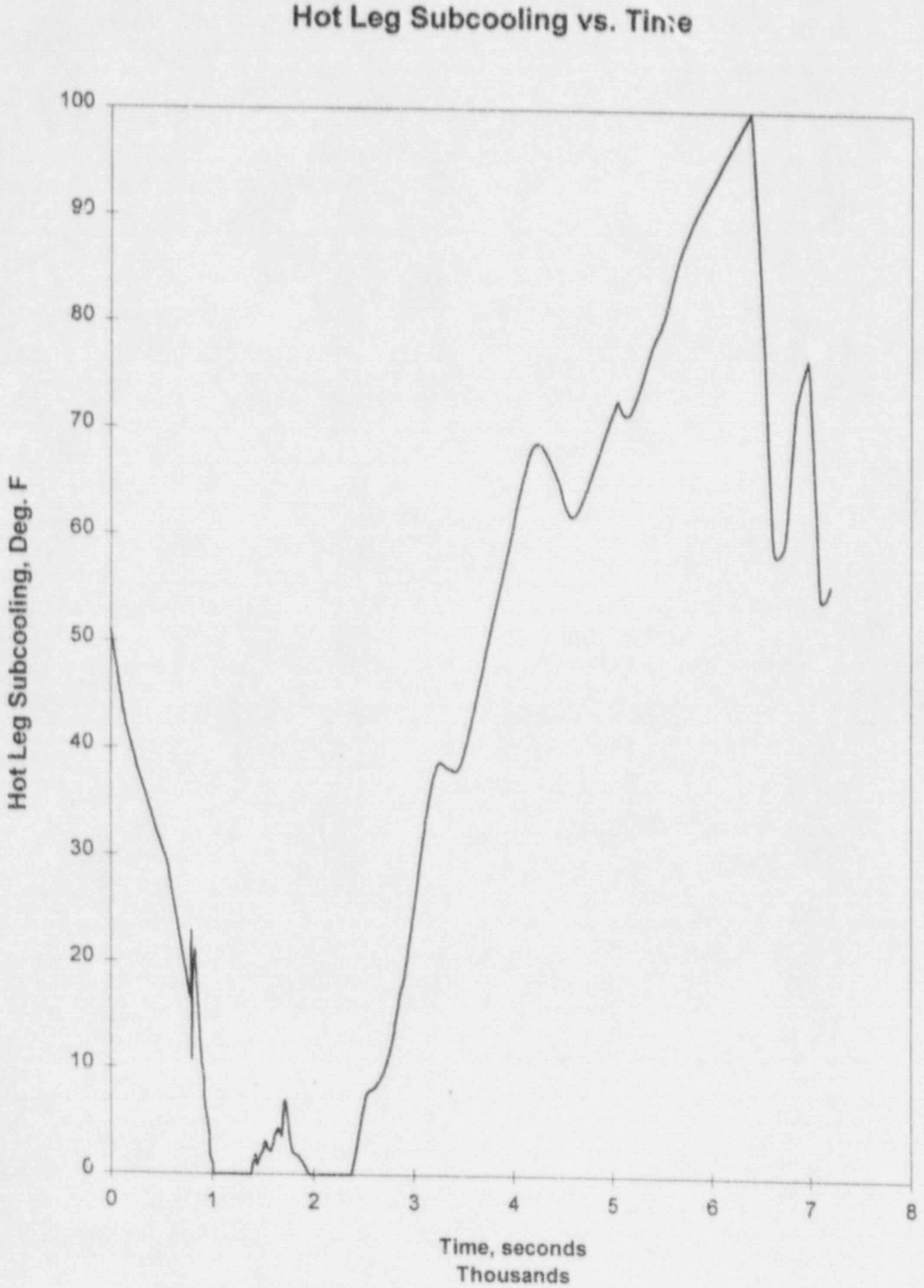
Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.345-A



Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-15-B



Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-16-A



Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant	STEAM GENERATOR TUBE RUPTURE WITH EOP BASED OPERATOR ACTIONS	Figure 14.15.3-16-B