ASG-94-060

ABB

ASEA BROWN BOVERI

ABB COMBUSTION ENGINEERING NUCLEAR SYSTEMS

Design Analysis

QPF 0304-1

ANALYSIS NUMBER: CENC-1965	REVISION NUMBER: 0

TITLE: AMENDED ANALYSIS OF MAINE YANKEE STEAM GENERATOR CIRCUMFERENTIALLY FLAWED TUBES AT TUBESHEET

SUMMARY OF CONTENTS:

Calculation	11	Pages
Appendix	0	Pages
Microfiche	0	Sheets

PURPOSE:

To establish the allowable tube wall degradation for the steam generator based on the requirements of NRC Regulatory Guide 1.121.

COGNIZANT ENGINEER:

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Name, Title, Signature, Date

DESIGN VERIFICATION:

VERIFICATION STATUS: COMPLETE

The design information contained in this document has been verified to be correct by means of:

x Design Review

____ Alternate Calculation -- Copy Attached.

____ Qualification Testing -- Test Report No. ____.

D. G. Slack D. G. Slack 10/18/94

Independent Reviewer (Name, Signature, Date)

MANAGEMENT APPROVAL:

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1.0 SUMMARY AND CONCLUSIONS

This analysis establishes the allowable tube wall degradation for the Maine Yankee steam generator based on the requirements of Regulatory Guide 1.121 (Reference 1). The analysis revises the previous evaluation (Reference 2) by considering a circumferential defect without axial extent. The analysis considers the tube loading due to normal operation, Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB) and Safe Shutdown Earthquake (SSE). Allowable tube wall degradation is established in accordance with the ASME Code Section III allowables (or Margins) consistent with the provisions of Reg. Guide 1.121.

The following summarizes the results of this analysis:

- The Reg. Guide 1.121 structural integrity margin which stipulates "The factor of safety against failure by bursting under normal operating conditions is not less than **three** at any tube location where defects have been detected," is the controlling criteria for the Maine Yankee circumferential I.D. cracks at the tube expansion transition region (top of the tubesheet). The allowable degradation is **79%**.

- The maximum stress intensity at the flaw location due to the combined effects of a postulated pipe break accident, earthquake, maximum differential pressure and flow induced vibration is:

  S.I. = 52.8 ksi < allowable 56.0 ksi
  (for a 79% wall degraded tube)

  The 79% degradation is supported by the leak rate and pressure test program documented in Reference 7.

2.0 INTRODUCTION

The analysis presented herein is performed to establish the maximum allowable tube wall degradation for the Maine Yankee steam generator tubes per the requirement of NRC Reg. Guide 1.121. The results of this analytical study are used in conjunction with the results of the previous tube leak rate and pressure test program to assess the steam generator tube integrity when subjected to I.D. circumferential cracking at the tube expansion transition region. The results apply to circumferential defects which have no axial extent.

This report addresses the requirements of NRC Reg. Guide 1.121 for maximum allowable differential pressures during normal operation and accident conditions as well as the ASME Code Section III Appendix F requirements for faulted load conditions of Loss of Coolant Accidents (LOCA) plus Safe Shutdown Earthquake (SSE) loads, Main Steam Line Break (MSLB) plus SSE loads and flow-induced vibration loads. Most of the loadings were determined previously in Reference 2.

The report will also evaluate the tube structural integrity margins required by the NRC as stated in Regulatory Guide 1.121.

3.0 GEOMETRY

The Maine Yankee steam generator tube bundle is constructed with .75 inch x .048 inch wall, SB-163 Inconel 600 material tubes placed in a triangular pattern with a pitch of 1.0 inch. The tubes are welded to the primary face of the tubesheet and explanded through the thickness of the tubesheet. The tubes are supported in the axial flow region by grid type (eggcrate) supports. The first support is 47 inches above the secondary face of the tubesheet with five other eggcrate supports equally spaced at 42 inches. There is one partial eggcrate support 32 inches above the uppermost full eggcrate support. There are two floating partial drilled support plates at two elevations. These drilled support plates have been modified to remove the rim and detach them from the shroud.

4.0 DESCRIPTION OF TUBE FLAWS

Circumferential cracking is located on the tube inside surface in the transition zone between the explanded and unexplanded portions. This zone is located at the top surface of the tubesheet. Reference 2 considered that a flaw had longitudinal dimensions and therefore resulted in a conservative acceptable crack depth based on the hoop stress in the tube. This evaluation considers the allowable degradation assuming a purely circumferential crack with axial loads on the tube.

5.0 LOADING TO BE CONSIDERED

Reference 2 considered the following loads:

LOCA Rarefaction Wave
Pipe Break Impulse Response
MSLB Secondary Side Blowdown
Differential Tube Pressure
Safe Shutdown Earthquake
Flow Induced Vibration

This analysis considers critical axial loads, the worst being the longitudinal stress resulting from the Main Steam Line Break pressure of 2520 psi (Reference 3).

LOCA rarefaction loads are negligible near the tubesheet as indicated in Reference 2.

The effects of locked tubes during normal operation were considered in Reference 4 and were found to be negligible.

Reference 5 determined that the axial loads resulting from blowdown flow on tube supports during a Steam Line Break were negligible.

6.0 NRC REQUIRED STRUCTURAL INTEGRITY MARGINS

The loadings identified are conditions which in combination must satisfy appropriate ASME Code, Section III allowable stresses. In addition to those requirements, Regulatory Guide 1.121 requires that certain structural integrity margins be satisfied for flawed tubes which have not been removed from service.

The governing criteria is that a factor of safety against failure by bursting under normal operating conditions is not less than three at any tube location where defects have been detected.

The rationale for this criteria is that it represents a margin of safety which is inherent in the design rules of Section III of the ASME Code. It is possible for flawed tubes to meet the above requirements because the steam generator original tube thickness is provided with margins much larger than minimum ASME Code requirements.

7.0 ALLOWABLE DEGRADATION

The allowable degradation for a tube having a circumferential crack near the tubesheet is controlled by the pressure stress in the axial direction normal to the crack.

The required tubewall thickness to maintain a safety factor of 3 on burst is:

Given:

$$P_1 = 2.25 \text{ ksi} \qquad P_2 = .815 \text{ ksi}$$

$$R_i = .365 \text{ in.} \qquad S_u = 80 \text{ ksi}$$

$$t_r = \frac{3(P_1 - P_2)\, R_i}{2S_u - (P_1 + P_2)}$$

$$t_r = .010 \; in.$$

The allowable degradation based on the original tube thickness of .048 inches is:

$$\frac{t - t_r}{t} = \frac{.048 - .010}{.048} = 79\%$$

From Reference 3 the Main Steam Line Break maximum pressure differential is 2520 psi. The MSLB stress intensity is:

$$\sigma_x - \sigma_r = \frac{\Delta P R_i}{2t_r} + \frac{\Delta P}{2} = \frac{(2.52)\,(.365)}{2(.010)} + \frac{2.52}{2} = 47.25 \; ksi$$

The MSLB stress intensity is combined with the stresses from Reference 2 which are increased by the are^ ratio of the 79% wall reduction and the original tube wall.

Pipe Break Impulse	2.3 ksi
SSE	2.3 ksi
FIV	.9 ksi

Therefore the combined stress intensity is:

$$S.I. = 47.25 + 5.5 = 52.75 \text{ ksi}$$

The ultimate strength for the SB-163 Inconel tubing is 80.0 ksi at a maximum operating temperature of 600°F. Appendix F of the ASME Code Section III gives the allowable stress intensity for faulted conditions as S = .7Su.

Therefore:

$$S.I. = 52.75 < .7Su = 56.0 \text{ ksi}$$

Test Results

In Reference 7 the results of the Main Yankee Pressure Test Program support the 79% allowable degradation. In this test, tubes were tested with artificially induced I.D. circumferential cracks which simulated the flawed tubes in the Maine Yankee Steam Generator.

All the representative defected tubes were subjected to differential pressures of approximately 4800 psi without sustaining tube rupture. This result indicates that a pressure differential of 4350 psi (at least 3 times the operating differential pressure) would not cause "guillotine" tube failure at a circumferential defect until the tube wall penetration exceeded 80%.

8.0 REFERENCES

1. U.S. Nuclear Regulatory Commission Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes", August 1976.

2. ABB-CE Report CENC-1934, "Maine Yankee Steam Generator Analysis of Circumferentially Flawed Tubes at Tubesheet", Jan. 1991.

3. Letter TAG-MY-94-050, Kenneth R. Rousseau of Yankee Atomic Electric Co. to Paul L. Anderson, MSLR Peak Differential Pressure, October 17, 1994.

4. ABB-CE Calculation MY-SS-900, Rev. 0, "Steam Generator Tube Loads for Tube Lock-up", 10/4/94.

5. ABB-CE Calculation MY-SS-901, Rev. 0, "Steam Generator Tube Loads Due to Main Steam Line Break", 10/4/94.

6. ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, 1992 Edition.

7. ABB-CE Report CENC-1930, "Test Report for Leak Tests of Flawed Steam Generator Tubes Simulating Circumferential I.D. Cracking of Tubes Found in the Maine Yankee Steam Generators", Jan. 1991.

ENCLOSURE 5

MAINE YANKEE NOVEMBER 14, 1994 LETTER ENCLOSURE'S REFERENCE 8

ABB C-E Technical Report M-PENG-TR-002, "Maine Yankee Run Duration Limit Evaluation for Circumferential Cracking," October 1994

ABB C-E's Technical Report M-PENG-TR-002 has previously been transmitted to the NRC in Attachment A of Maine Yankee's November 14, 1994 letter to the NRC.

ENCLOSURE 6

MAINE YANKEE NOVEMBER 14, 1994 LETTER ENCLOSURE'S REFERENCE 9

Yankee Atomic Electric Co. Memo TAG-MY-94-51, "Maine Yankee S/G Tube Leakage Assessment - Summary of Results," October 27, 1994

MEMORANDUM

YANKEE ATOMIC - BOLTON

To	P. L. Anderson	Date	October 27, 1994
		Group #	TAG-MY-94-051
From	K. R. Rousseau/V. M. Esquillo	W.O.#	5037
Subject	Maine Yankee Steam Generator Tube Leakage Assessment - Summary of Results	LMS.#	K/02/01/03
		File #	MY94-051.TAG

EXECUTIVE SUMMARY

The purpose of this memo is to document a summary of the results of the Maine Yankee steam generator tube leakage assessment performed at YNSD (Reference 1) in response to Service Request M-94-131 (Reference 2). The assessment entailed a realistic look at past operation at Maine Yankee assuming that the steam generator tubes were in a degraded condition. The objective was to support the development of a report which presents a safety assessment of operation with steam generator tube leakage measured during the 1994 steam generator outage. The report was to address steam generator tube leakage induced during a Main Steam Line Rupture (MSLR) event as well as multiple Steam Generator Tube Rupture (SGTR) events.

Table 1 provides a summary of the calculated offsite consequences for the MSLR with induced leakage cases. Table 2 provides the offsite consequences for the multiple SGTR events. The results presented in Tables 1 and 2 cover a variety of assumptions including realistic (i.e., best-estimate) as well as conservative (i.e., FSAR) radiological assumptions.

Reference 3 provided a conservative assessment of the dose consequences versus effective leakage during a MSLR event with induced tube leakage. Figures 1 through 8 in Reference 3 provide a set of working figures to conservatively establish allowable leakage limits to meet 10CFR100 limits. Reference 3 is attached for completeness.

The focus of these analyses has been on effective leakage in the range from 100 to 3,500 gpm (up to 4 complete tube severances). Recent data from ABB-CE puts the effective leakage at less than 10 gpm. Since the expected leakage is so small, the results from the majority of this assessment are not required. To support the report, for the purposes of evaluating the offsite consequences for the leakage observed, the curves provided in Figures 1 through 8 of Reference 3 are appropriate to evaluate the dose consequences for both realistic and conservative radiological assumptions.

Note that the results provided in this memo are based on scoping analyses and are not intended to be construed as replacements for any of the licensing analyses presented in the FSAR. All of the analyses supporting these assessments have been documented and have been reviewed for reasonableness. This assessment has not been prepared or reviewed under the YNSD Quality Assurance Program. Use of this information to support safety related activities or licensing correspondence should be reviewed by YNSD prior to its implementation.

This memo closes out Service Request M-94-131.

DISCUSSION

The purpose of this memo is to document a summary of the results of the Maine Yankee steam generator tube leakage assessment performed at YNSD (Reference 1) in response to Service Request M-94-131 (Reference 2). The intent of the original program was to look at the past operation at Maine Yankee to determine the impact on the safety analysis of potentially higher primary to secondary leakages during a Main Steam Line Rupture (MSLR) event as well as multiple Steam Generator Tube Rupture (SGTR) events. YNSD evaluated the impact on the safety analysis for MSLR events with induced steam generator tube leakages ranging from small leakages up to 4 double-ended guillotine ruptures of steam generator tubes. In addition, YNSD evaluated the impact of multiple steam generator tube ruptures (without MSLR).

In an independent but parallel effort, ABB-CE evaluated the specific conditions of the Maine Yankee steam generator tubes, and for the limiting MSLR condition, determined the potential leakage which could have occurred, as well as the potential for multiple SGTR events. At the beginning of the program, the leakage rates were not known. Since the efforts for determining leakage rates and evaluating the consequences were being performed in parallel, the original program intent was to have YNSD iterate on the maximum allowable leakage while still meeting safety analysis acceptance criteria using realistic assumptions, while ABB-CE determined the actual expected leakage.

The original program plan was intended to assess past operation only, crediting realistic assumptions and actual observed reactor coolant activities over the past several cycles. During the course of the program, Maine Yankee requested the scope be expanded to cover the use of licensing assumptions for both the radiological and thermal-hydraulic events to determine the maximum allowable leakage while still meeting the safety analysis acceptance criteria.

In all, three key areas were addressed in this assessment:

1. MSLR with induced tube leakage - realistic assessment. Credit was taken for the closure of Loop Isolation Valves (LIVs). An assessment using conservative (e.g., licensing) radiological assumptions was also performed.

2. Multiple Steam Generator Tube Rupture Event - realistic assessment was performed.

3. MSLR with induced tube leakage, with no credit for LIVs. A scoping assessment of RWST inventory capacity was performed.

In addition, several supporting analyses were also performed. These included:

4. Bounding radiological assessment of the MSLR event with induced leakage.

5. MSLR thermal-hydraulic analysis with no tube leakage (maximum steam generator tube differential pressure evaluation).

6. Thermal-hydraulic assessment of the LIV dynamic pressure differential response (dynamic loads on the valve disk).

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6. Thermal-hydraulic assessment of the LIV dynamic pressure differential response (dynamic loads on the valve disk).

MSLR With Induced SGTR Leakage Assessment - With LIVs (Current Procedures)

A series of MSLR cases was run using the approved MSLR RETRAN model updated to reflect the modelling changes necessary to evaluate these events. Several cases were run covering a range of induced tube leakages. The modelling was performed by assuming fractions and multiples of complete tube ruptures with an equivalent leakage determined from the output. For these events, credit was taken for commencing closure of the LIVs via current EOP guidance. Table 4 provides a summary of the assumed operator actions and times. Table 5 provides a typical sequence of events for these cases, with Figure 1 (Sheets 1 through 7) containing the plotted output.

Table 1 provides a matrix of the final radiological results for each of the cases run and the effective leakages for each of the events analyzed. Figure 3 provides a comparison of the difference in offsite consequences between the conservative radiological analyses performed in Reference 3 and the cases presented here crediting the actual plant response.

Multiple SGTR Event - Realistic Assessment

The results from the ABB-CE analysis support a position that the potential for multiple SGTRs was very low, and was not considered credible. However, this assessment did address a series of multiple SGTR events. These events were modelled with leak isolation assumptions ranging from crediting the normal operator actions (cooldown and depressurize), crediting the LIVs (with SG overfill) and no credit for LIVs (with SG overfill). Steam generator overfill cases were assumed to cause a MSSV to fail an MSSV in the fully open position resulting in a release of unpartitioned coolant directly to atmosphere. Radiological analyses were performed for these events using a realistic assumption that the RCS coolant activity was no higher than the maximum coolant activity observed since Cycle 4. For liquid relieved out the MSSV, a conservative assumption of 100% flashing was made in lieu of detailed topographical analysis to determine locations for "puddling" around the relief point. Table 2 provides a summary of the results of these cases.

The assessment demonstrates that, for a three SGTR event, using best-estimate assumptions on initial conditions, operator action times and equipment availability (cooldown with steam dump and bypass, depressurize with normal sprays), operators can terminate the leakage without overfilling the steam generator resulting in minimal offsite release.

For more limiting conditions, such as loss of offsite power, the results indicate that, with or without credit for LIVs, overfill will most likely occur in the faulted steam generator for leakages corresponding to between one and two SGTRs. However, the offsite consequences, based on the realistic radiological assumptions described above, would still be below 10 CFR 100 limits.

MSLR With Induced Leakage, No LIVs - RWST Depletion Scenario - Potential Additional EOPs

One of the concerns raised in draft NUREG-1477 is the potential for core damage due to RWST depletion during this postulated event. The continuous leakage and boiloff into the faulted steam generator can eventually deplete the RWST with a bypass of the containment if the steam line rupture is outside containment. Without inventory in the containment sump to recirculate, the concern is that continued leakage will cause the core to eventually uncover causing core damage.

The Maine Yankee EOP actions to close the LIVs precludes this problem since the leakage outside containment is directly isolated. Thus, under best-estimate conditions, RWST depletion is not a concern for Maine Yankee.

The current Maine Yankee EOP set does not include procedures in the unlikely event that the MSLR with induced tube leakage occurs with a concurrent failure of the LIVs. The Westinghouse ERGs ECA-3.1 and ECA-3.2 do contain directions for operators to deal with this issue for plants without loop isolation valves. For the purposes of illustration, a scoping assessment was performed to assess the ability of Maine Yankee to respond to this event without crediting the LIVs. The recommended actions for ECA-3.1 and ECA-3.2 are straightforward. From the Westinghouse ERGs, the following key actions are associated with these procedures:

1. Prepare for and initiate RCS cooldown to cold shutdown

2. Depressurize RCS to refill pressurizer

3. Reduce RCS injection flow

4. Depressurize RCS to minimize RCS subcooling

5. Makeup to the RWST

A RETRAN analysis of the MSLR with multiple SGTRs was performed without credit for the LIVs, but crediting the operator actions described above. This analysis is intended to be a scoping assessment of the feasibility of implementing the aforementioned EOPs at Maine Yankee if so desired. The analysis was performed for 1, 2, 3 and 4 double ended SGTR cases. RCPs were assumed to continue operating to maximize the heat load in the system. To illustrate the characteristics of this event, a MSLR with 3-SGTRs is presented. Table 6 provides a chronological sequence of events. Figure 2 (Sheets 1 through 9) provides the plotted output for this case.

It is assumed in each analysis that operators will initiate RHR and cool and depressurize the RCS to cold shutdown conditions within the 8 hour period. For simplicity and conservatism, the analysis credited only cooldown with the Atmospheric Dump Valve and depressurization of the RCS to maintain 100°F of subcooling. Following the RCS depressurization to atmospheric conditions, leakage out of the RCS can be terminated by back filling the faulted steam generator. This will create a static head at the leak location in excess of the RCS pressure.

The results of this scoping evaluation showed that a successful recovery from this event without depletion of the RWST is possible. The RWST has a Technical Specification minimum usable volume of 300,000 gallons (without credit for makeup). From the 3-SGTR case, the total RWST inventory used was 257,000 gallons. Somewhat higher (an additional 40,000 gallons) RWST requirements were calculated from sensitivity analyses where the RCPs were tripped. Table 7 provides a summary of the RWST inventory required versus number of SGTRs assumed.

Assuming borated makeup at 200 gpm is established to the RWST within 2 hours, an additional 72,000 gallons is available to allow additional injection until the RCS can be cooled and depressurized to atmospheric conditions.

The results of these scoping analyses appear to indicate that the RWST inventory requirements plateau in the range of 3 to 4 SGTRs. The reason for this is related to the assumed operator action to depressurize the RCS to 100°F of subcooling and the final subcooling maintained during the event. The cooldown and leakage from the four SGTR case was more severe than the three SGTR case and did not allow for maintaining the 100°F subcooling in the RCS. Hence, the long term equilibrium RCS pressure remained below the pressure from the 3 SGTR case.

The key variables that affect the total RWST requirement is the throttling of the HPSI and the final pressure to which operators depressurize and maintain the RCS for the long term. Provided that the leakage is controlled (by RCS depressurization) to minimize leakage, then it is postulated that larger leakage areas would not yield significantly higher RWST requirements. Thus, the 300,000 gallons in the RWST, plus establishment of RWST makeup, should be sufficient to cool the RCS and establish cold shutdown conditions for four SGTRs and potentially more.

The results of this section (MSLR with leakage - no LIVs), based on preliminary scoping analyses, indicate that the actions provided in the Westinghouse ERGs for ECA-3.1 and ECA-3.2 will be successful in bringing Maine Yankee to a stable shutdown condition for four SGTRs and potentially more. Further evaluations and sensitivity analyses, particularly in the modelling of the primary to secondary heat transfer, and RCP operation, are necessary to validate the conclusion drawn above.

Bounding Radiological Assessment - MSLR with Induced Leakage

Draft NUREG-1477 suggested a method for conservatively determining offsite dose consequences versus effective leakage rate. Reference 3 provides a detailed description of the derivation of these curves. Reference 3 is provided as an attachment to this memo for completeness.

Maximum SG Tube Pressure Differential

The intent of this analysis was to determine the maximum differential pressure across the steam generator tubes for use as input to the leakage evaluation by ABB-CE. This information has been formally sent to ABB-CE in Reference 4. A copy of Reference 4 is provided in Attachment B of this memo for completeness.

The pressure challenges to the steam generator tubes occur at two distinct times. An initial peak differential pressure of 1739 psid occurs approximately four seconds into the event during the blowdown phase of the MSLR. A second peak pressure of 2520 psid occurs approximately 12 minutes into the event following the steam generator dryout and subsequent RCS repressurization via the HPSI pumps and post-dryout RCS heatup.

LIV Dynamic Pressure Response

On of the concerns for crediting the LIVs to close in the more severe MSLR leakage cases is the potential for the pressure differential across the LIVs to increase above 500 psid, which is the manufacturer's design specification for which the valves can close. As the second LIV approaches fully closed, the flow resistance through the valve will increase significantly until eventually, the flow through the valve is unable to match the flow through the tube leakage. Consequently, as the valve closes, the pressure differential across the valve will continuously increase. A detailed RETRAN model was put together to specifically model the LIV differential pressure response during the closure stroke. The results of the analysis showed that the maximum differential pressure developed across the valve remained below the 500 psid design pressure.

CONCLUSIONS

The focus of the thermal-hydraulic analyses has been on effective leakage in the range from 100 to 3,500 gpm (up to 4 complete tube severances). Preliminary data from ABB-CE put the effective leakage at less than 10 gpm. Since the expected leakage is so small, the results from the majority of this assessment are not required. Thus, for the purposes of evaluating the offsite consequences for the leakage observed, the curves provided in Figure 1 through 8 of Reference 3 can be used to conservatively evaluate the dose consequences for both realistic and conservative radiological assumptions. The information in Reference 3 demonstrates that for the relatively small leakages expected to occur (<10 gpm) the offsite consequences for this event are minimal and within 10 CFR 100 limits.

Note that the results provided in this memo are based on scoping analyses and are not intended to be construed as a replacement for any of the licensing analyses presented in the FSAR. All of the analyses supporting these assessments have been documented and have been reviewed for reasonableness. This assessment has not been prepared or reviewed under the YNSD Quality Assurance Program. Use of this information to support safety related activities or licensing correspondence should be reviewed by YNSD prior to its implementation.

SAFETY ASSESSMENT

This memo is not safety related. It provides a status of an assessment performed to determine the potential offsite consequences for both multiple SGTR events as well as MSLR induced steam generator tube leakage.

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REFERENCES

1. Calculation File MYC-1592, "Multiple Steam Generator Tube Rupture Assessment," October, 1994.

2. Service Request M-94-131, "Steam Generator Tube Leakage Report," dated September 19, 1994.

3. YNSD Memorandum REG 198/94 - Revision 1, "Offsite Doses as a Function of Primary-to-Secondary Leakage Due to MSLR with Induced Steam Generator Tube Leakage," dated October 26, 1994.

4. Letter YNSD to ABB-CE, TAG-MY-94-050, "Main Steam Line Rupture," dated October 17, 1994.

Table 1
Radiological Results of MSLR Induced Tube Leakage Evaluation and Multiple SGTR

Case Description	0-1 hr Eff. Leakage (gpm)	0-8 hr Eff. Leakage (gpm)	Realistic Assumptions (Coincident Spike)				Licensing Assumptions (Pre-existing Spike)			
			EAB 2-hr Thyroid (rem)	LPZ 8-hr Thyroid (rem)	EAB 2-hr Whole Body (rem)	LPZ 8-hr Whole Body (rem)	EAB 2-hr Thyroid (rem)	LPZ 8-hr Thyroid (rem	EAB 2-hr Whole Body (rem)	LPZ 8-hr Whole Body (rem)
MSLR 1/2-HL* SGTR	89	29	---	---	---	---	180.0	12.3	1.4	0.1
MSLR 1-HL SGTR	174	45	---	---	---	---	444.0	---	3.4	---
MSLR 1-DEGR SGTR	573	90	14.00	0.40	0.30	0.00	1149.0	83	10.5	0.8
MSLR 2-DEGR SGTR	869	122	16.70	0.48	0.40	0.01	1532.0	111.0	14.4	1.0
MSLR 3-DEGR SGTR	996	137	14.60	0.40	0.40	0.01	1552.0	112.0	14.0	1.1
MSLR 4-DEGR SGTR	1,073	145	17.40	0.47	0.40	0.01	1736.0	123.0	15.6	1.2

* MSLR - Main Steam Line Rupture
HL - Tube leakage equivalent to the leakage from the hot leg side of a complete DEGR SGTR
DEGR - Double-ended guillotine rupture of a single tube at the tube sheet

Table 2
Radiological Results for Multiple SGTR Events

Case Description	SGTR Isolation Method	Realistic Assumptions (Coincident Spike)				Licensing Assumptions (Pre-existing Spike)			
		EAB 2-hr Thyroid (rem)	LPZ 8-hr Thyroid (rem)	EAB 2-hr Whole Body (rem)	LPZ 8-hr Whole Body (rem)	EAB 2-hr Thyroid (rem)	LPZ 8-hr Thyroid (rem	EAB 2-hr Whole Body (rem)	LPZ 8-hr Whole Body (rem)
1-SGTR	Cooldown with ADV Depressurize with Aux Spray	---	---	---	---	5.1	0.38	0.74	0.08
1-SGTR	30 minute isolation Non-mechanistic FSAR Case (Auto initiation of EFW)	---	---	---	---	2.1	0.16	0.57	0.05
1-SGTR	30 minute isolation Non-mechanistic FSAR Case (Current FSAR reported case)	---	---	---	---	120.0	8.5	0.55	0.04
3-SGTR	Cooldown with SD&BP Depressurize with Normal Sprays	0.006	0.002	0.008	0.002	---	---	---	---
3-SGTR	Close LIV in 45 minutes SG overfill with stuck open MSSV	6.8	0.34	0.15	0.006	---	---	---	---
3-SGTR	Cooldown to atmospheric conditions via ADV and RHR SG overfill with stuck open MSSV Isolation (depressurized) in 8 hrs	36.1	4.2	0.70	0.06	---	---	---	---

Table 3
Summary of Estimated Allowed Equivalent Leakages from Assessment
for MSLR with Induced Leakage and Multiple SGTR Events

CASE	ASSUMPTIONS			Allowed Leakage	
	Radiological	LIV	SD & BP	Radiological (10CFR100 limit)	RWST Inventory
MSLR	Realistic	Yes	N/A	3367+ gpm	10+ SGTRs
MSLR	Realistic	No	N/A	Not calculated	4+ SGTRs
MSLR	Conservative	Yes	N/A	<167 gpm	10+ SGTRs
MSLR	Conservative	No	N/A	<100 gpm	4+ SGTRs
SGTR	Realistic	Yes	Yes	3+ SGTRs	10+ SGTRs
SGTR	Realistic	No	Yes	3+ SGTRs	≤3 SGTRs
SGTR	Realistic	Yes	No	3+ SGTRs	10+ SGTRs
SGTR	Realistic	No	No	3+ SGTRs	3+ SGTRs
SGTR	Conservative	Yes	Yes	3+ SGTRs	10+ SGTRs
SGTR	Conservative	No	Yes	≤3 SGTRs	≤3 SGTRs
SGTR	Conservative	Yes	No	1-<2 SGTRs	10+ SGTRs
SGTR	Conservative	No	No	1-<2 SGTRs	3+ SGTRs

Key

Realistic - Realistic initial primary coolant activity (maximum observed since Cycle 4) and dispersion factor - coincident spike - See Reference 3

Conservative - Conservative (i.e. licensing) assumptions on initial primary coolant activity and dispersion factor - pre-existing spike - See Reference 3

LIV = Yes - Operators assumed to close LIVs within 45 minutes of event - 15 gpm leakage each maximum

LIV = No - No Credit for LIV closure

SD & BP = Yes - Steam dump and bypass system available to perform RCS cooldown to equalize primary and secondary pressures during SGTR event.

SD & BP = No - Steam dump and bypass system not available

Table 4
Key Operator Actions - MSLR with Induced Tube Leakage

Operator Action	EOP	Time Assumed*
Operators complete E-0 immediate actions and throttle HPSI if throttling criteria are met (pressurizer level, subcooling, SG level) Note for larger leakages, pressurizer level criteria is not met within 15 minutes. Throttling delayed.	E-0	15 minutes
Operators identify faulted steam generator and initiate SG sample for coolant activity	E-2	15 minutes
Operators identify leakage into faulted steam generator and trip RCPs	E-1	40 minutes
Operators determine combined MSLR and SGTR event, isolate HPSI to faulted loop, and commence closing of LIVs	E-3/ AOP 2-45	45 minutes
Operators reestablish normal charging and letdown	ES-1.1	49 minutes
Operators establish a feed (EFW) and steam (ADV) path in the intact steam generators to maintain RCS temperature and pressure until 2.0 hours	ES-1.2	49 minutes
Operators depressurize RCS with aux spray to 50°F subcooling to minimize LIV leakage	ES-1.2	2.0 hours
Operators cool and depressurize RCS to RHR entry conditions, establish RHR and cool to Cold Shutdown	ES-1.2	2.0 - 8.0 hours

* Time from beginning of event

Table 5
Chronological Sequence of Events - MSLR with 3 SGTR

TIME (SEC)	EVENT
0 - 10.0	Steady state full power operation.
10.0	MSLR upstream of the NRV.
10.1	NRV closes preventing the other two SGs from blowing down.
11.2	Reactor trip signal on low steam line pressure (<500 psia)
11.4	EFCV Closure signal on SG #1 low steam line pressure (<415 psia). Isolate EFW to SG #1
11.4	MFWRVs begin to close - conservatively short stroke time (10 seconds) Low pressure in main steam line - all EFCVs begin to close Reactor scram commences
20.0	SG leakage equivalent to three double ended SGTR begins
31.7	SIAS on low RCS pressure (<1600 psia). Trip of feedwater pumps, condensate pumps and heater drain pumps with coincident EFCV signal.
220.0	SG #1 dryout occurs
443.0	EFW isolation to intact steam generators - low steam line pressure (<415 psia)
900.0	Operators check if HPSI can be throttled. Subcooling and level criteria met. Delay throttle until pressurizer level ≥50%. Operators identify faulted SG and sample for activity.
2400.0	RCPs tripped. Operators identify leakage in faulted SG.
2700.0	Operators enter AOP 2-45. Isolate HPSI to faulted SG and commence closing cold leg LIV (2 minute stroke time).
2820.0	Cold leg LIV closed, operators commence closing hot leg LIV (2 minute stroke time.
2940.0	Hot leg LIV closed. Leakage of <15 gpm in each valve.

Table 5 (continued)
Chronological Sequence of Events - MSLR with 3 SGTR

TIME (SEC)	EVENT
2940.0	EFW to intact SGs restored and throttled to maintain level in SG. ADV opened to remove decay heat, HPSI throttled, CVCS restored.
1.0 hr	RETRAN run terminated. Operators assumed to hold conditions (pressurizer pressure = 830 psia, CET = 348°F) until 2.0 hours
2.0 hr	Operators depressurize to establish 50°F subcooling at CET (250 psia)
2.0 - 8.0 hr	Operators cool down, establish RHR, cool and depressurize to cold shutdown conditions. Event terminated. Note the radiological analysis assumes the plant to be maintained at the 2.0 hr conditions throughout.

Table 6
Chronological Sequence of Events for MSLR with 3-SGTR - NO LIVs
RWST Inventory Assessment

TIME (SEC)	EVENT
0 - 10.0	Steady state full power operation.
10.0	MSLR upstream of the NRV.
10.1	NRV closes preventing the other two SGs from blowing down.
11.2	Reactor trip signal on low steam line pressure (<500 psia)
11.4	EFCV Closure signal on SG #1 low steam line pressure (<415 psia). Isolate EFW to SG #1
11.4	MFWRVs begin to close - conservatively short stroke time (10 seconds) Low pressure in main steam line - all EFCVs begin to close Reactor scram commences
20.0	SG leakage equivalent to three double ended SGTR begins
31.9	SIAS on low RCS pressure (<1600 psia). Trip of feedwater pumps, condensate pumps and heater drain pumps with coincident EFCV signal.
210.0	SG #1 dryout occurs
433.0	EFW isolation to intact steam generators - low steam line pressure (<415 psia)
1800.0	Operators check if HPSI can be throttled. Subcooling and level criteria met. Operators delay throttle to establish and maintain level of about 50%. Operators identify faulted SG and sample for activity. Operators throttle EFW to intact steam generator to maintain level above the tube bundle
2700.0	Operators fail to close LIVs, open ADV to commence cooldown to RHR entry conditions
2.0 hr	Operators establish aux spray to depressurize the RCS to 100°F of subcooling

Table 6 (continued)
Chronological Sequence of Events for MSLR with 3-SGTR - NO LIVs
RWST Inventory Assessment

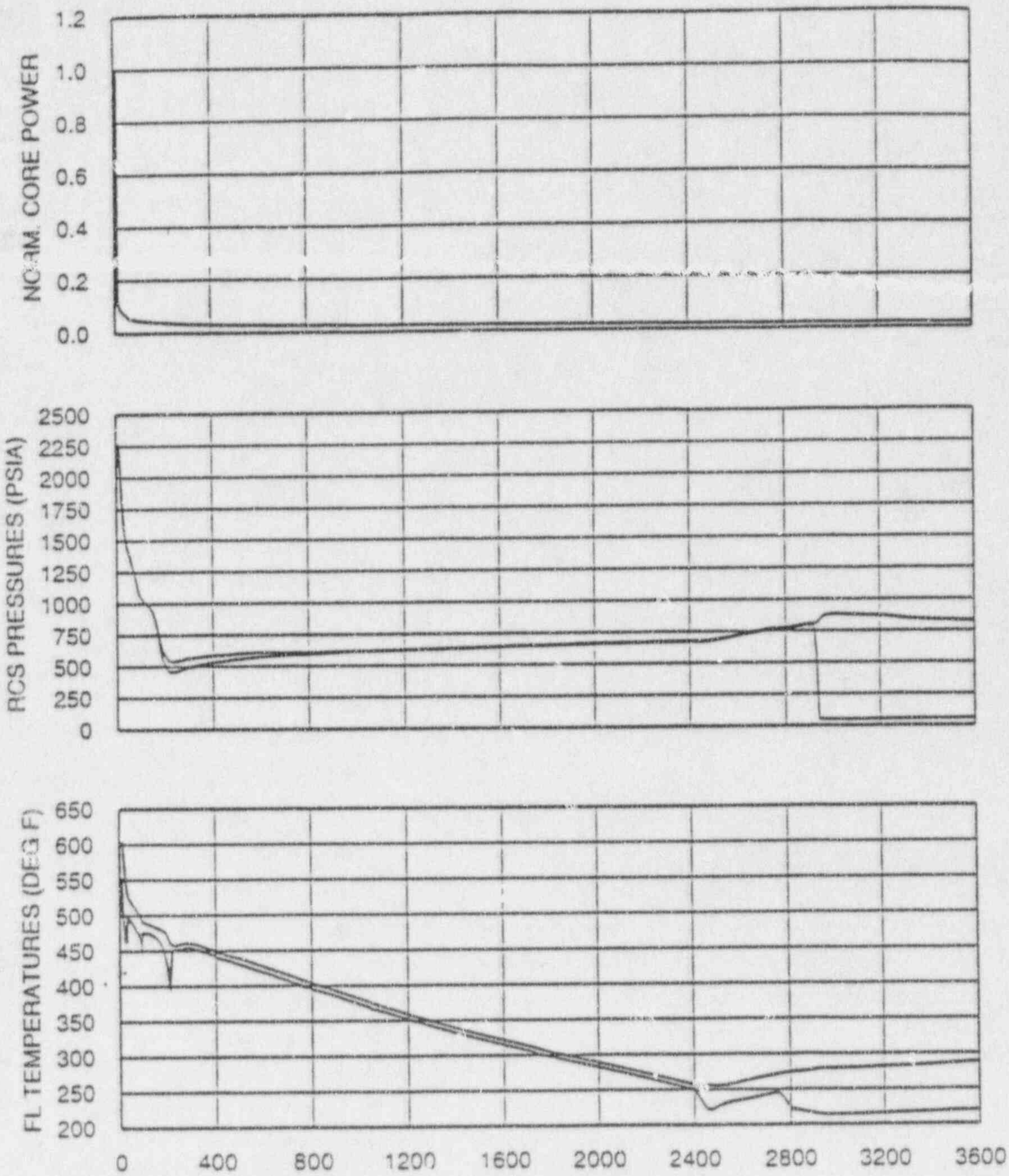
TIME (SEC)	EVENT
2.4 hr	100°F of subcooling achieved, aux spray terminated.
2.0-8.0 hr	RETRAN run terminated. Operators are assumed to have established RHR and cooled to cold shutdown conditions during this time period.
	RWST inventory used = 256,914 gal

Table 7
MSLR with Induced Leakage - No Credit for LIVs
Summary of RWST Requirement Versus Number SGTRs

Case Description	Effective Leakage (0-8 hr - gpm*)	RWST Inventory Requirement - RCPs Running** (0-8 hr)
MSLR/1-DEGR SGTR - No LV	326	129,701
MSLR/2-DEGR SGTR - No LIV	435	171,869
MSLR/3-DEGR SGTR - No LIV	666	256,914
MSLR/4-DEGR SGTR - No LIV	586	229,775

* At reference conditions of 552°F, 2250 psia

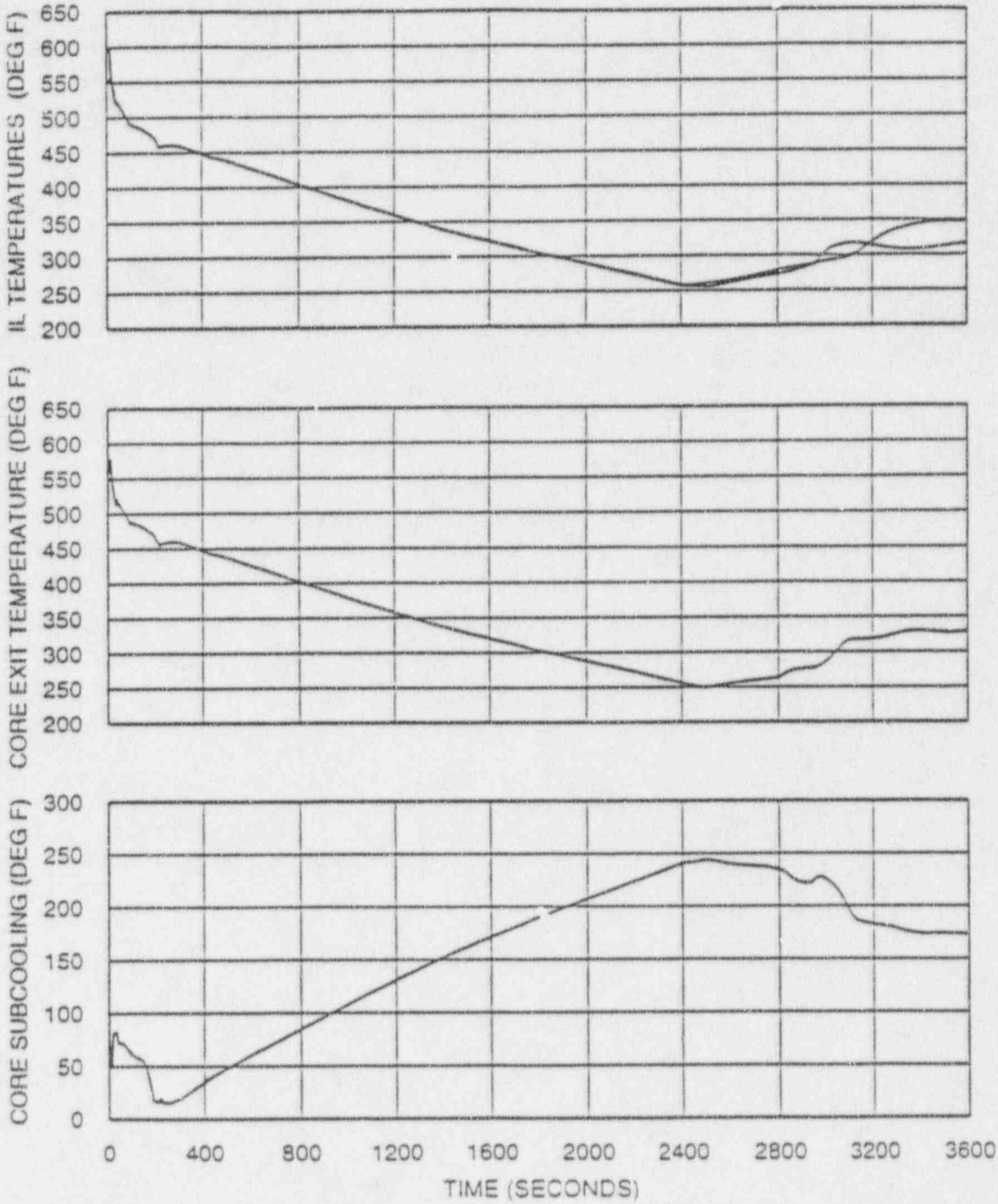
** With RCPs tripped, higher inventory requirements of approximately 40,000 gallons are necessary.



MAINE YANKEE
MSLR BASE SGTR ANALYSIS

MSLR WITH INDUCED TUBE LEAKAGE
LEAKAGE EQUIVALENT TO 3-SGTR
EOP ACTIONS - LIVS CLOSED AT 45 MIN

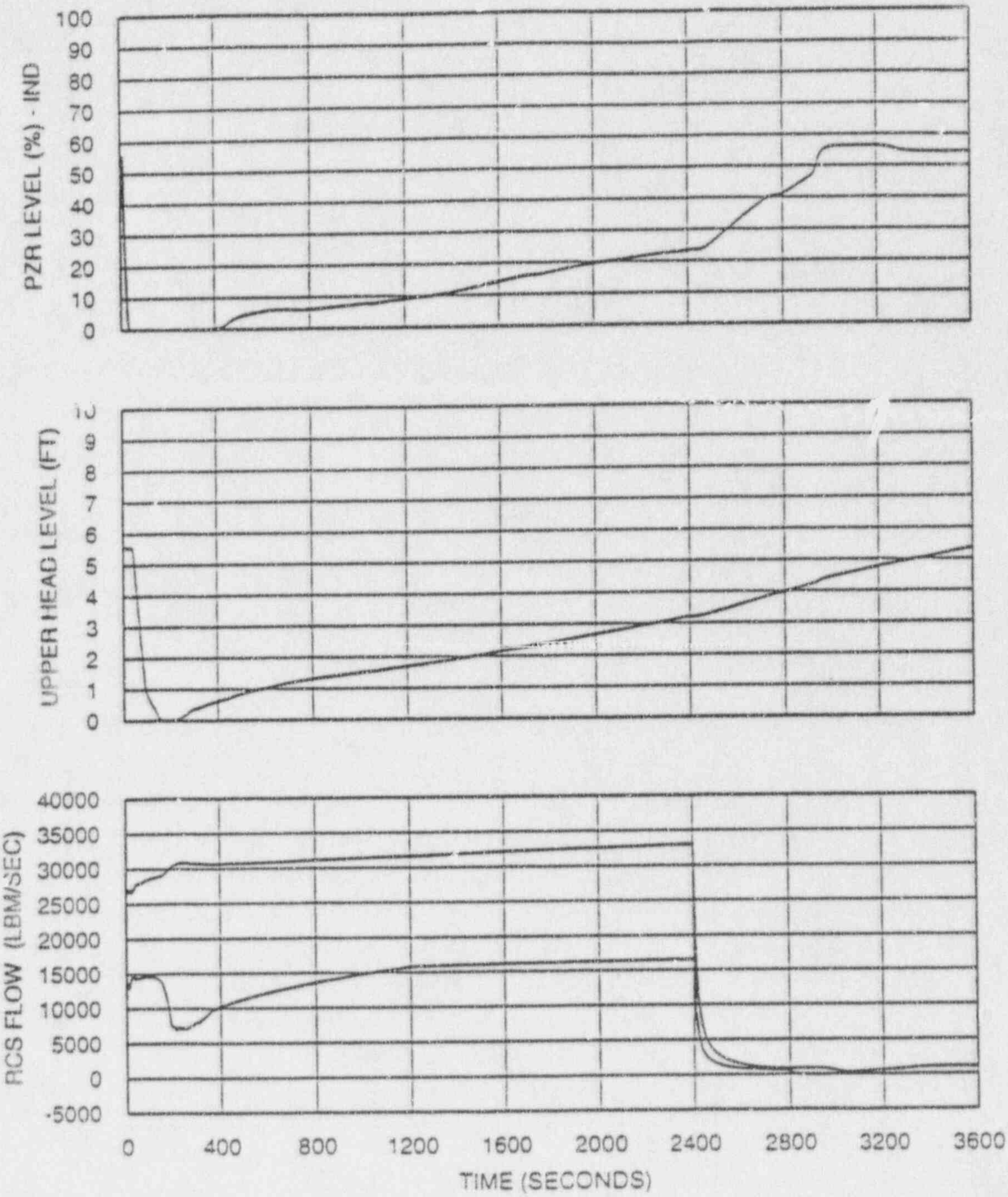
FIGURE 1 . Sh. 1



MAINE YANKEE
MSLR BASE SGTR ANALYSIS

MSLR WITH INDUCED TUBE LEAKAGE
LEAKAGE EQUIVALENT TO 3-SGTR
EOP ACTIONS - LIVS CLOSED AT 45 MIN

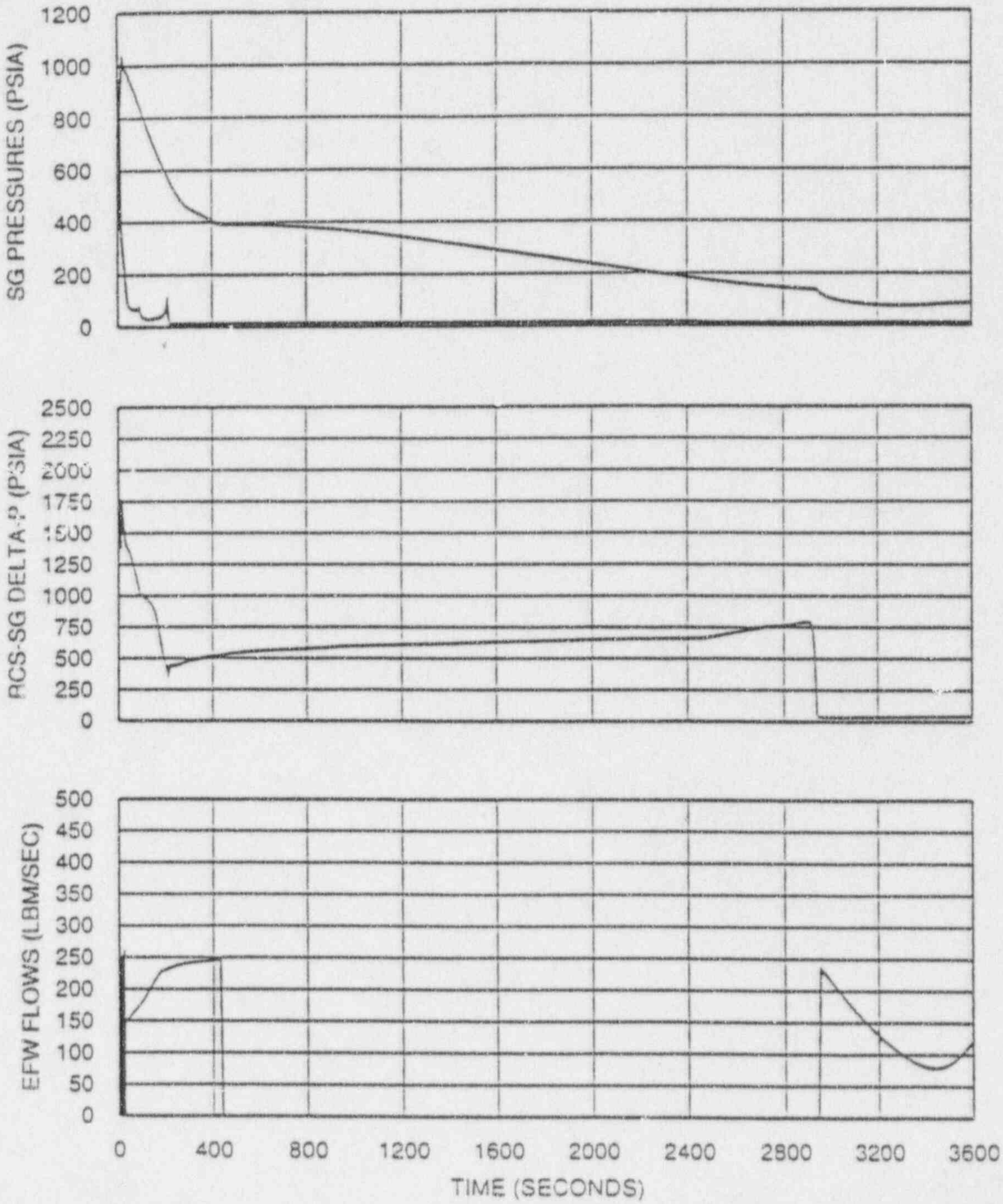
FIGURE 1 , Sh 2



MAINE YANKEE
MSLR BASE SGTR ANALYSIS

MSLR WITH INDUCED TUBE LEAKAGE
LEAKAGE EQUIVALENT TO 3-SGTR
EOP ACTIONS - LIVS CLOSED AT 45 MIN

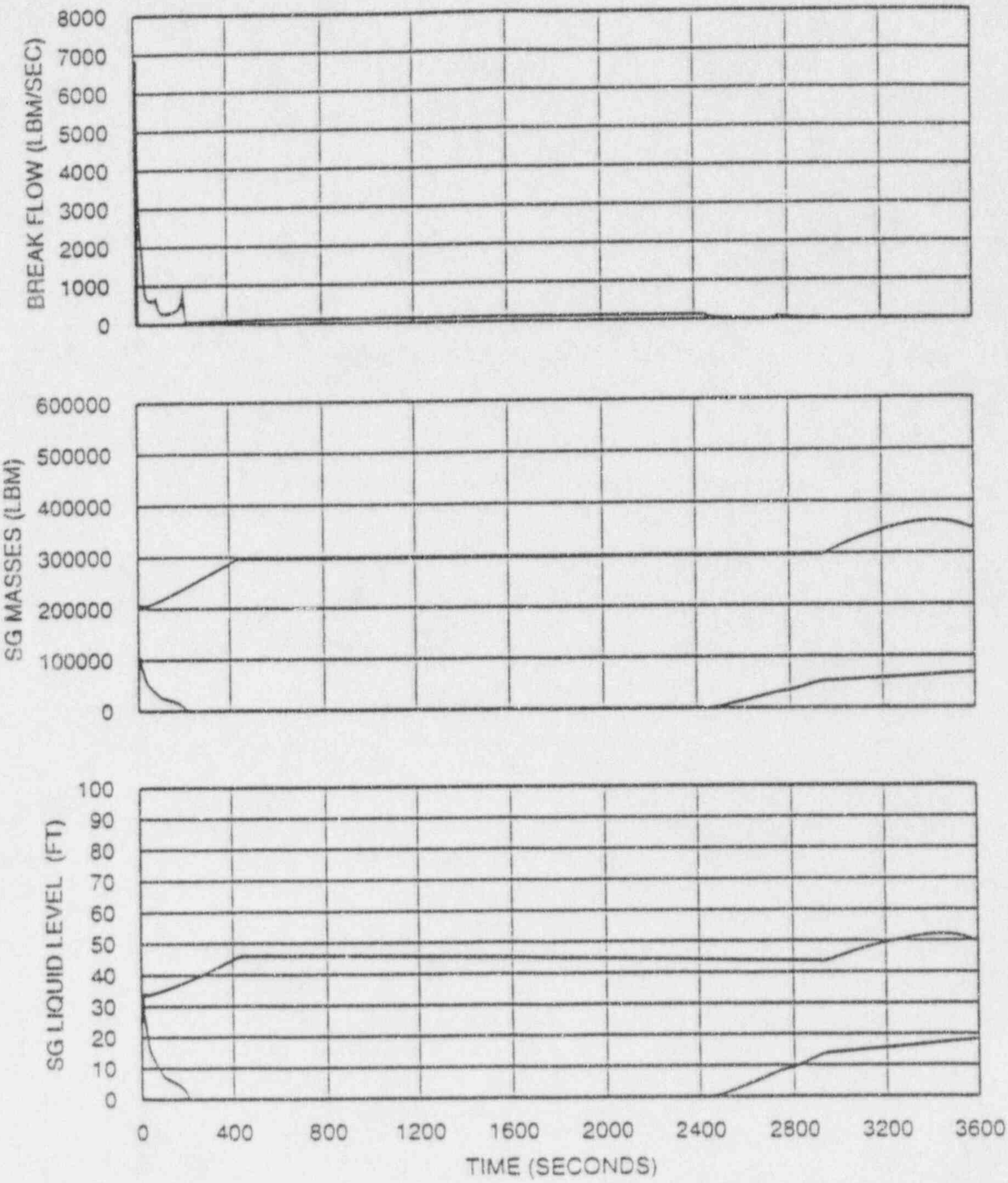
FIGURE 1 Sh 3



MAINE YANKEE
MSLR BASE SGTR ANALYSIS

MSLR WITH INDUCED TUBE LEAKAGE
LEAKAGE EQUIVALENT TO 3-SGTR
EOP ACTIONS - LIVS CLOSED AT 45 MIN

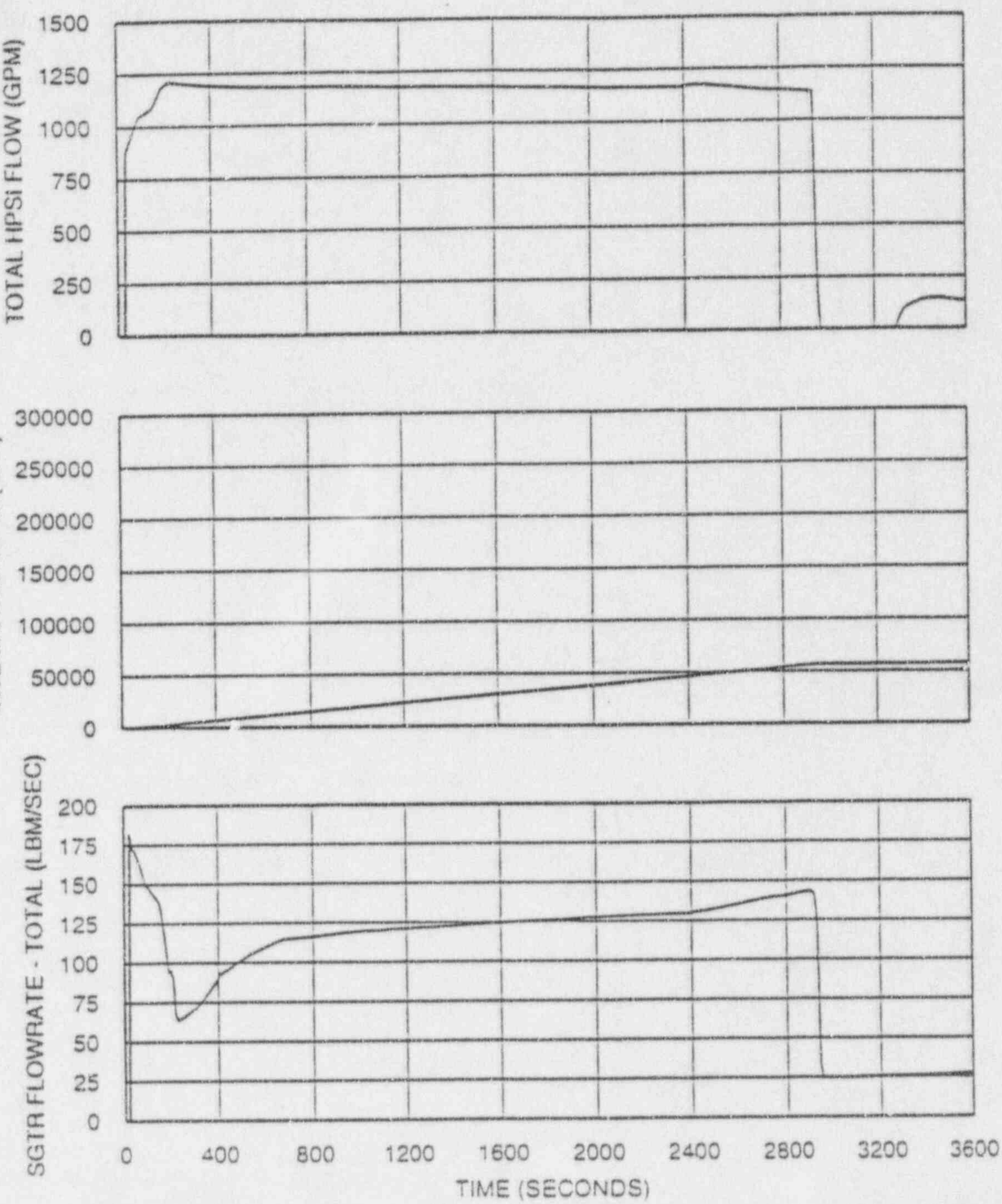
FIGURE 1 Sh. 4



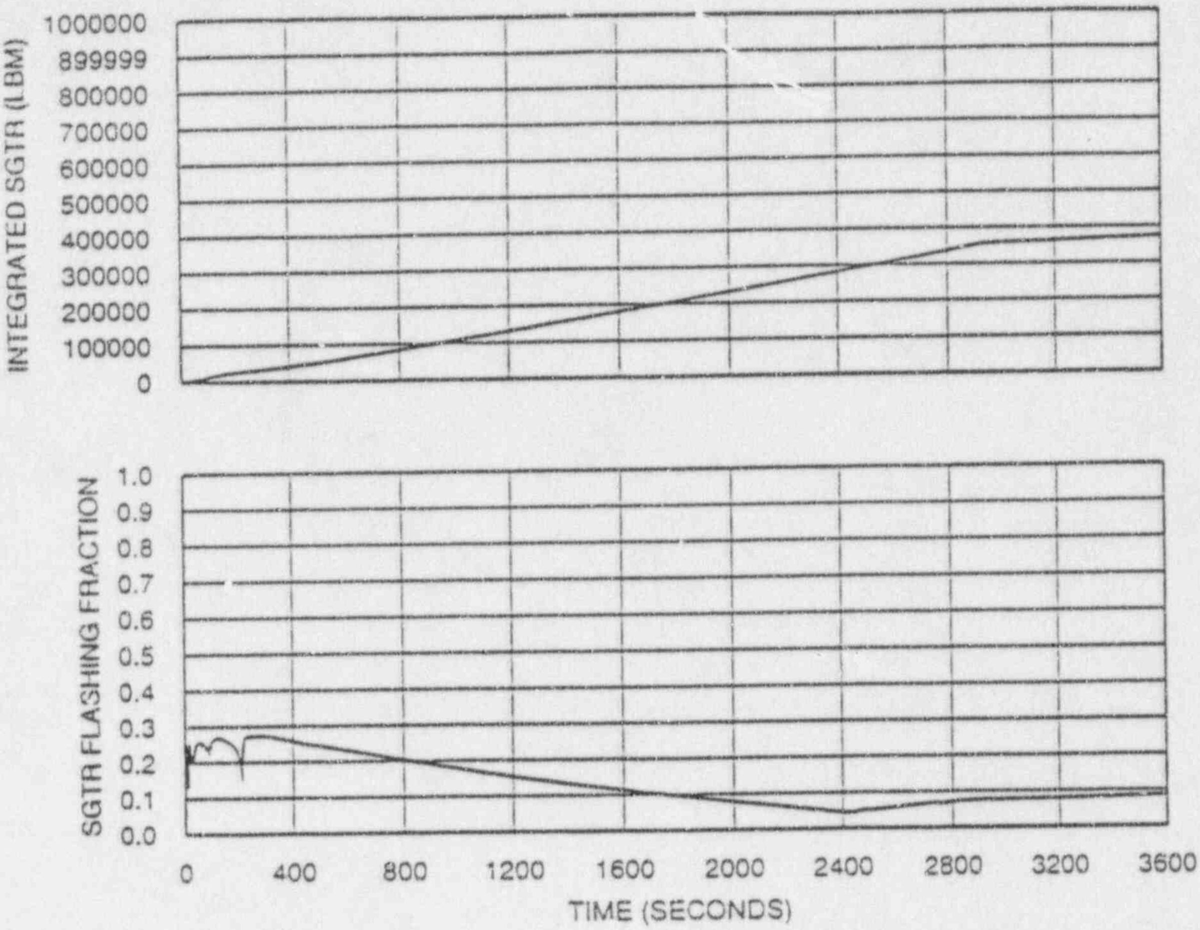
MAINE YANKEE
MSLR BASE SGTR ANALYSIS

MSLR WITH INDUCED TUBE LEAKAGE
LEAKAGE EQUIVALENT TO 3-SGTR
EOP ACTIONS - LIVS CLOSED AT 45 MIN

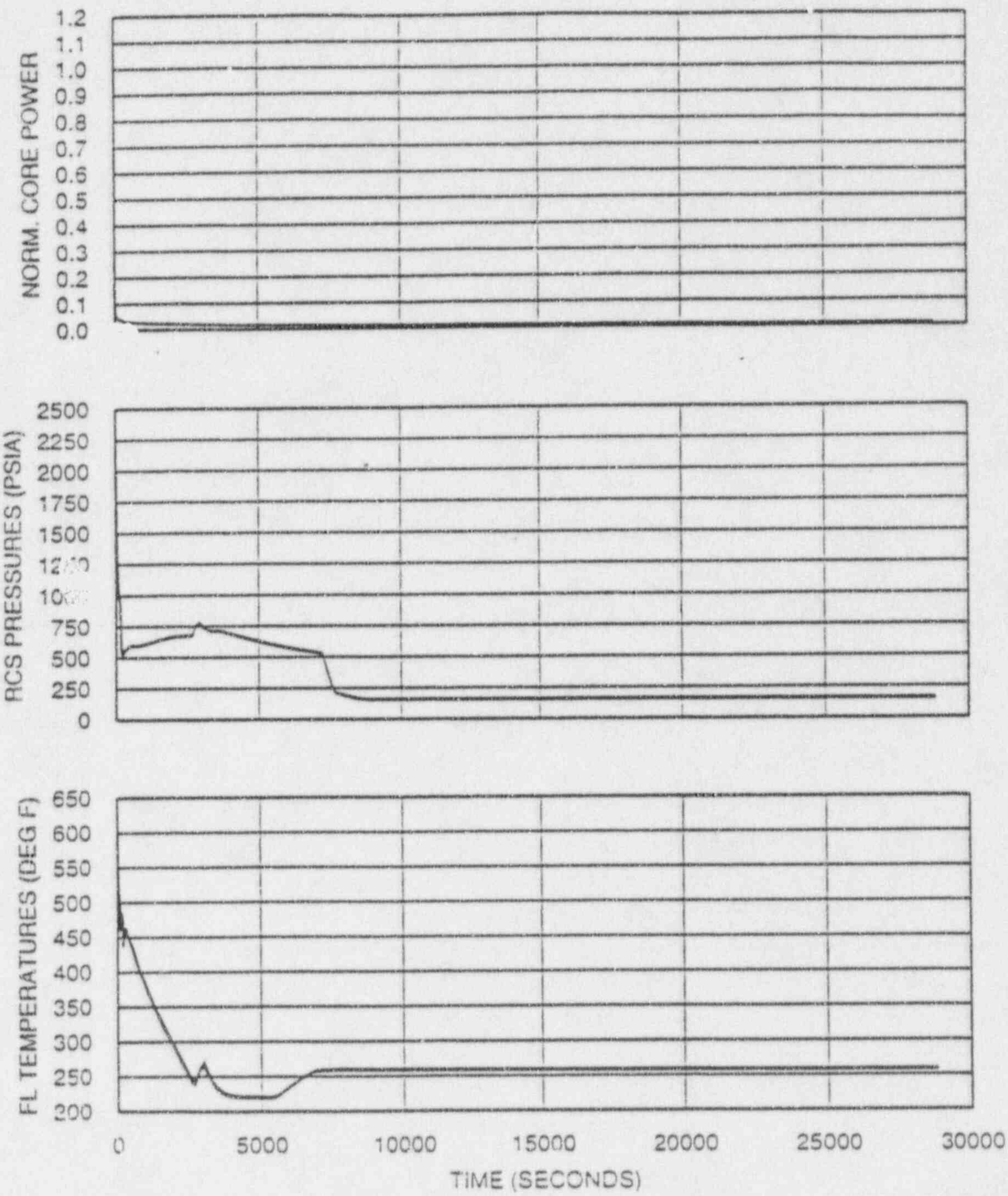
FIGURE 1 Sh 5



MAINE YANKEE MSLR BASE SGTR ANALYSIS	MSLR WITH INDUCED TUBE LEAKAGE LEAKAGE EQUIVALENT TO 3-SGTR EOP ACTIONS - LIVS CLOSED AT 45 MIN FIGURE 1 Sh. 5



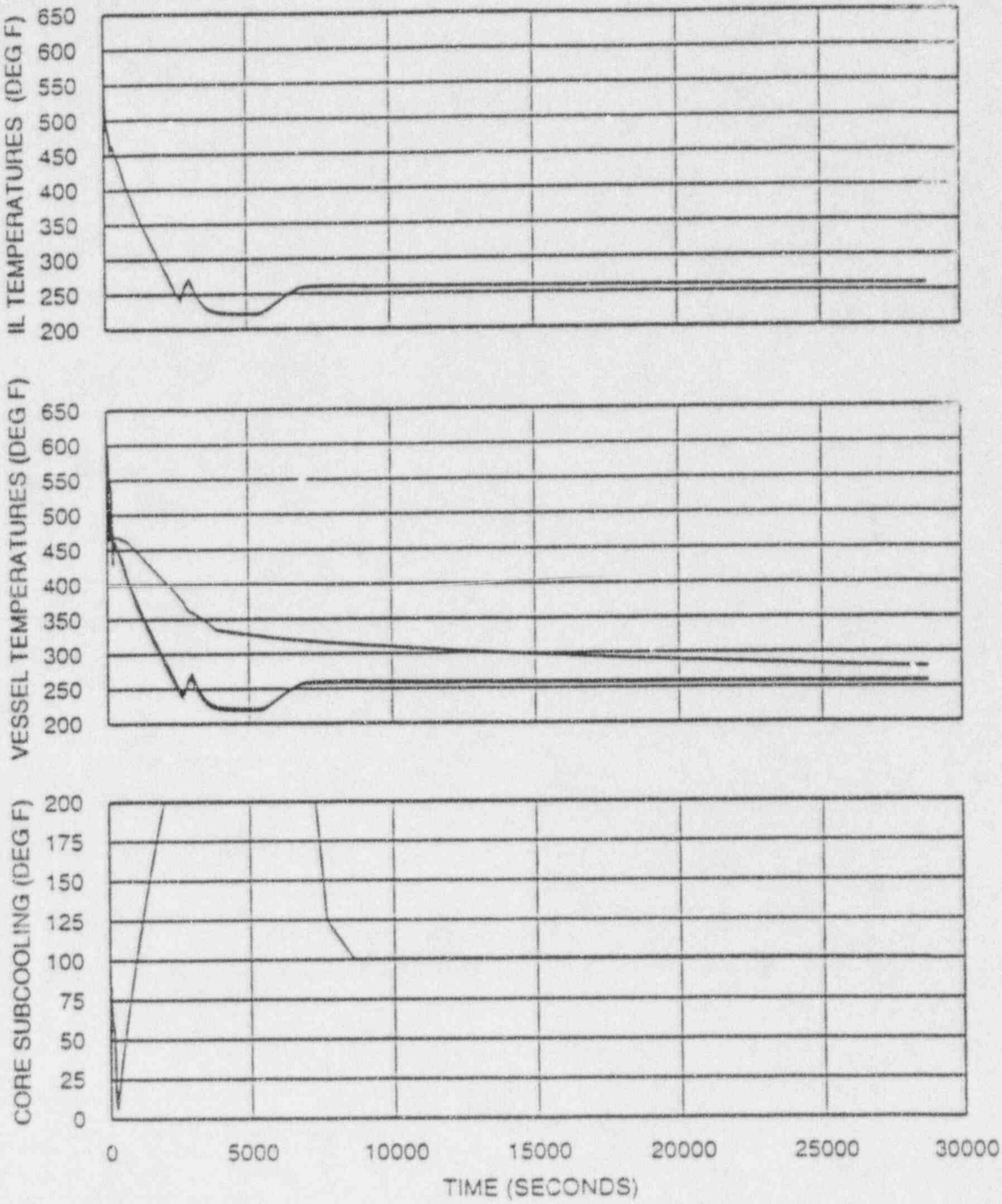
MAINE YANKEE MSLR BASE SGTR ANALYSIS	MSLR WITH INDUCED TUBE LEAKAGE LEAKAGE EQUIVALENT TO 3-SGTR EOP ACTIONS - LIVS CLOSED AT 45 MIN FIGURE 1 , Sh 7



MYC-1592 REV 0
MSLR BASE SGTR ANALYSIS

MY MSLR 3-SGTR - mvm303
NO LOOP - RCPs MAINTAINED
HPSI THROT 1800/ADV AT 1800.

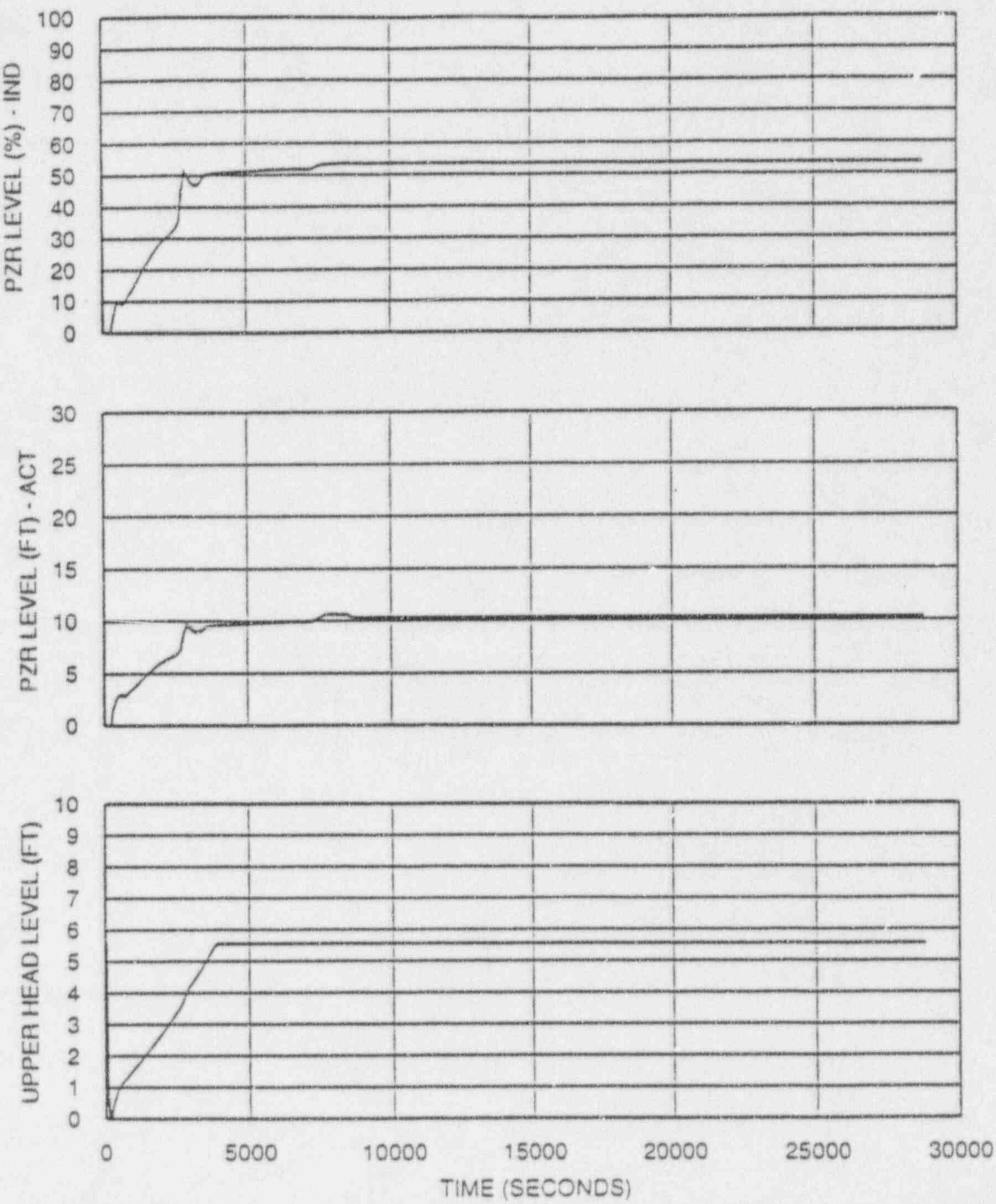
FIGURE 2 . Sh 1



MYC-1592 REV 0
MSLR BASE SGTR ANALYSIS

MY MSLR 3-SGTR - mvm303
NO LOOP - RCPs MAINTAINED
HPSI THROT 1800/ADV AT 1800.

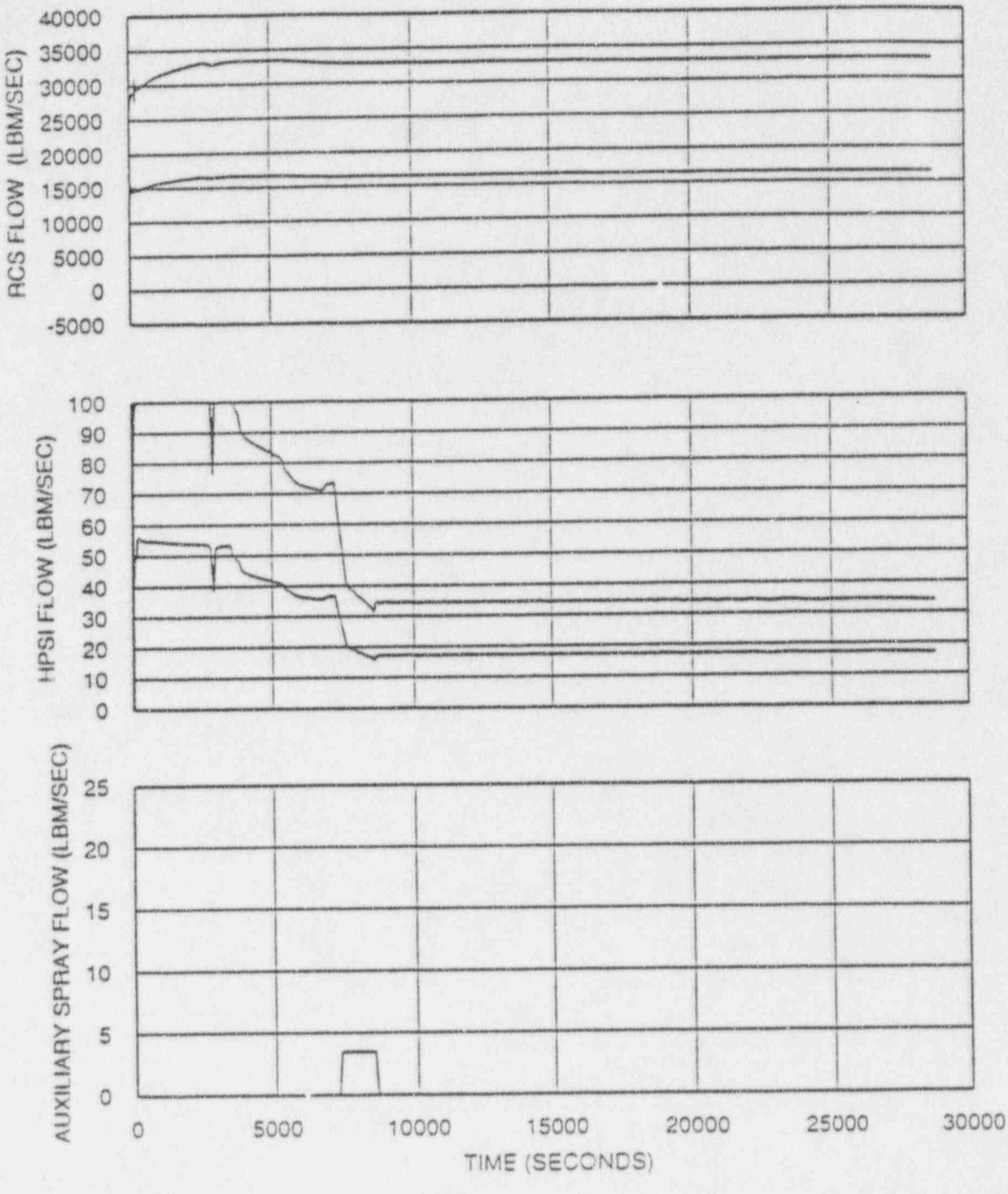
FIGURE 2 , Sh. 2



MYC-1592 REV 0
MSLR BASE SGTR ANALYSIS

MY MSLR 3-SGTR - mvm303
NO LOOP - RCPs MAINTAINED
HPSI THROT 1800/ADV AT 1800.

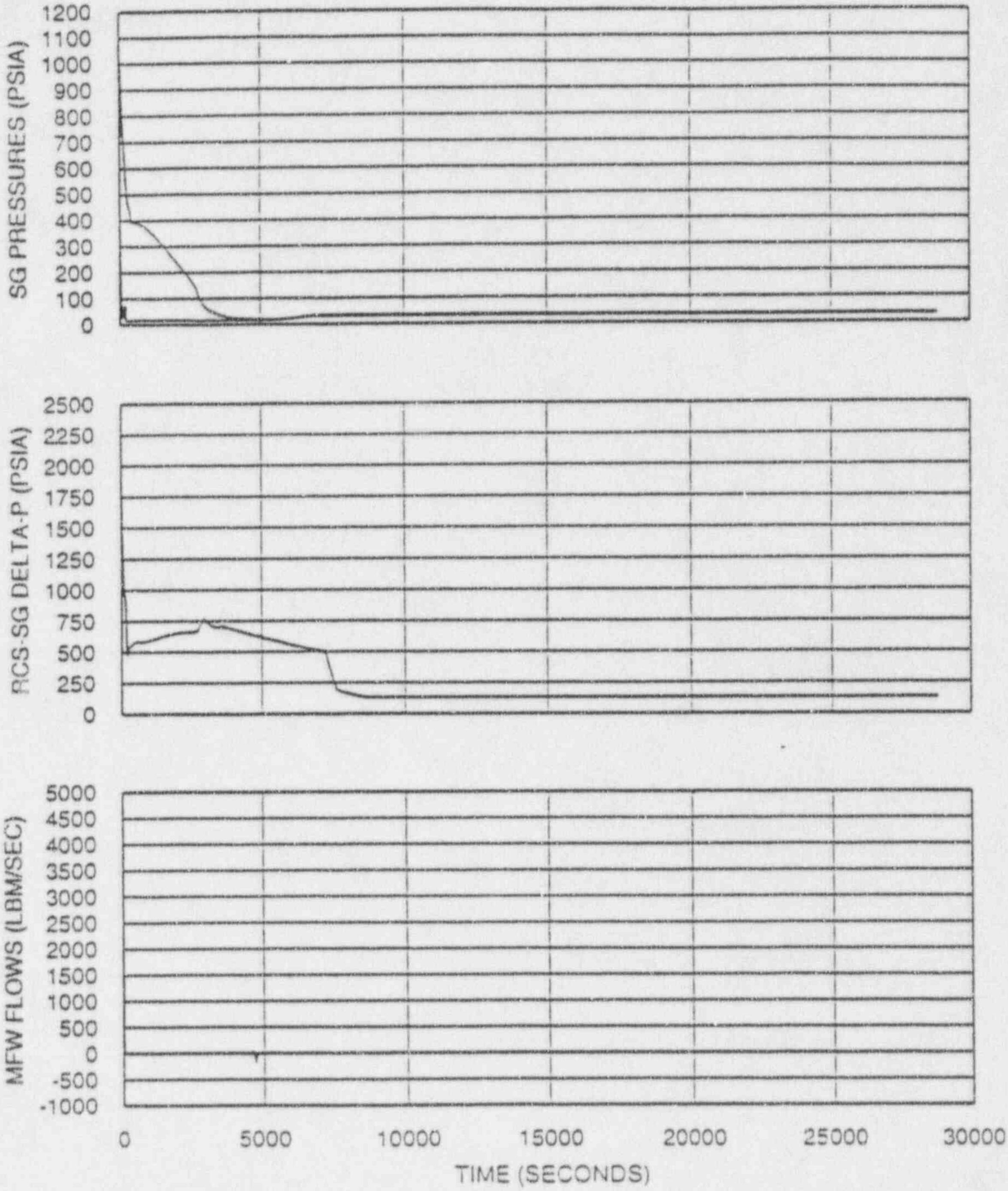
FIGURE 2 . Sh. 3



MYC-1592 REV 0
MSLR BASE SGTR ANALYSIS

MY MSLR 3-SGTR - mvm303
NO LOOP - RCPs MAINTAINED
HPSI THROT 1800/ADV AT 1800

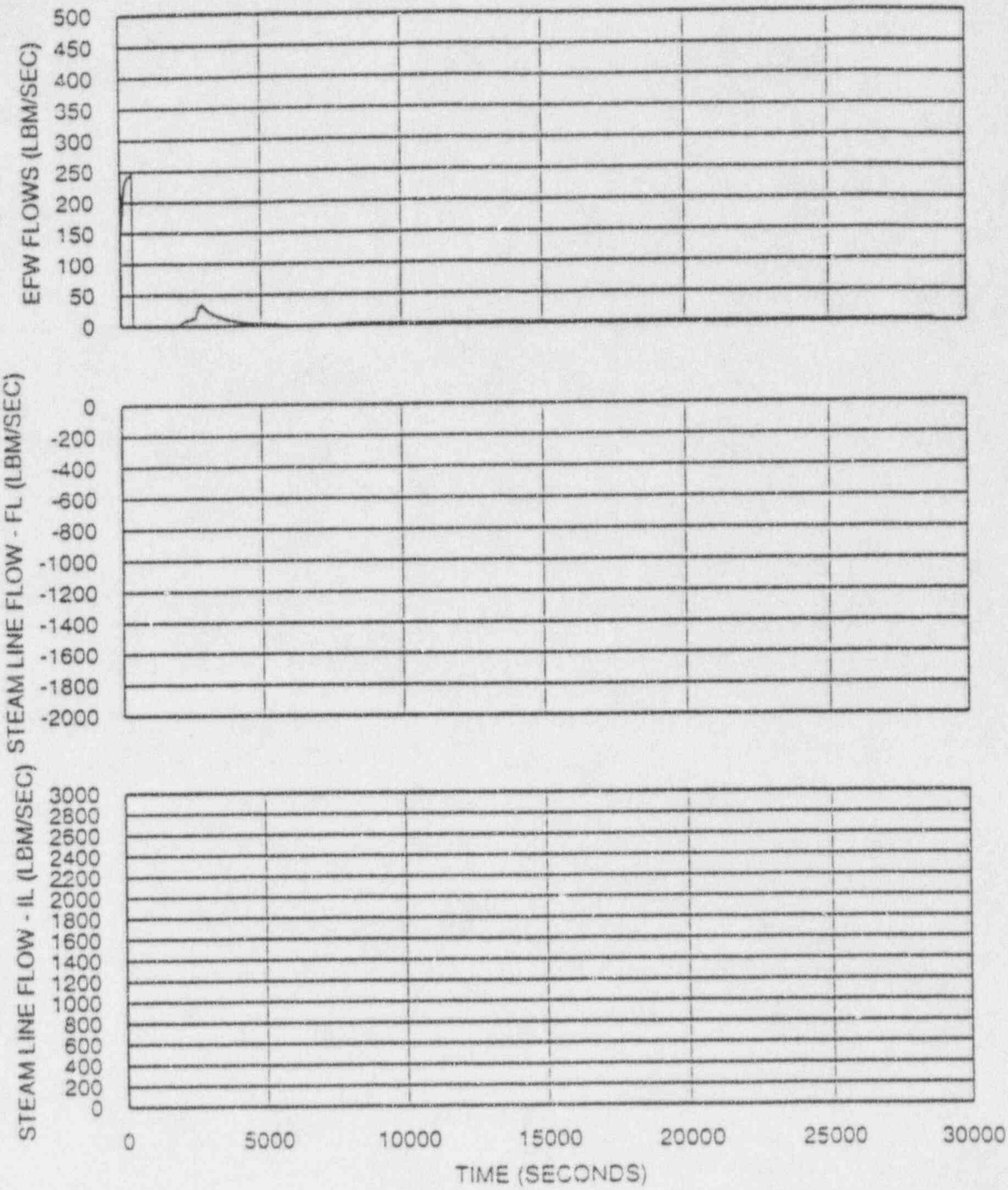
FIGURE 2 , Sh. 4



MYC-1592 REV 0
MSLR BASE SGTR ANALYSIS

MY MSLR 3-SGTR - rnvm303
NO LOOP - RCPs MAINTAINED
HPSI THROT 1800/ADV AT 1800

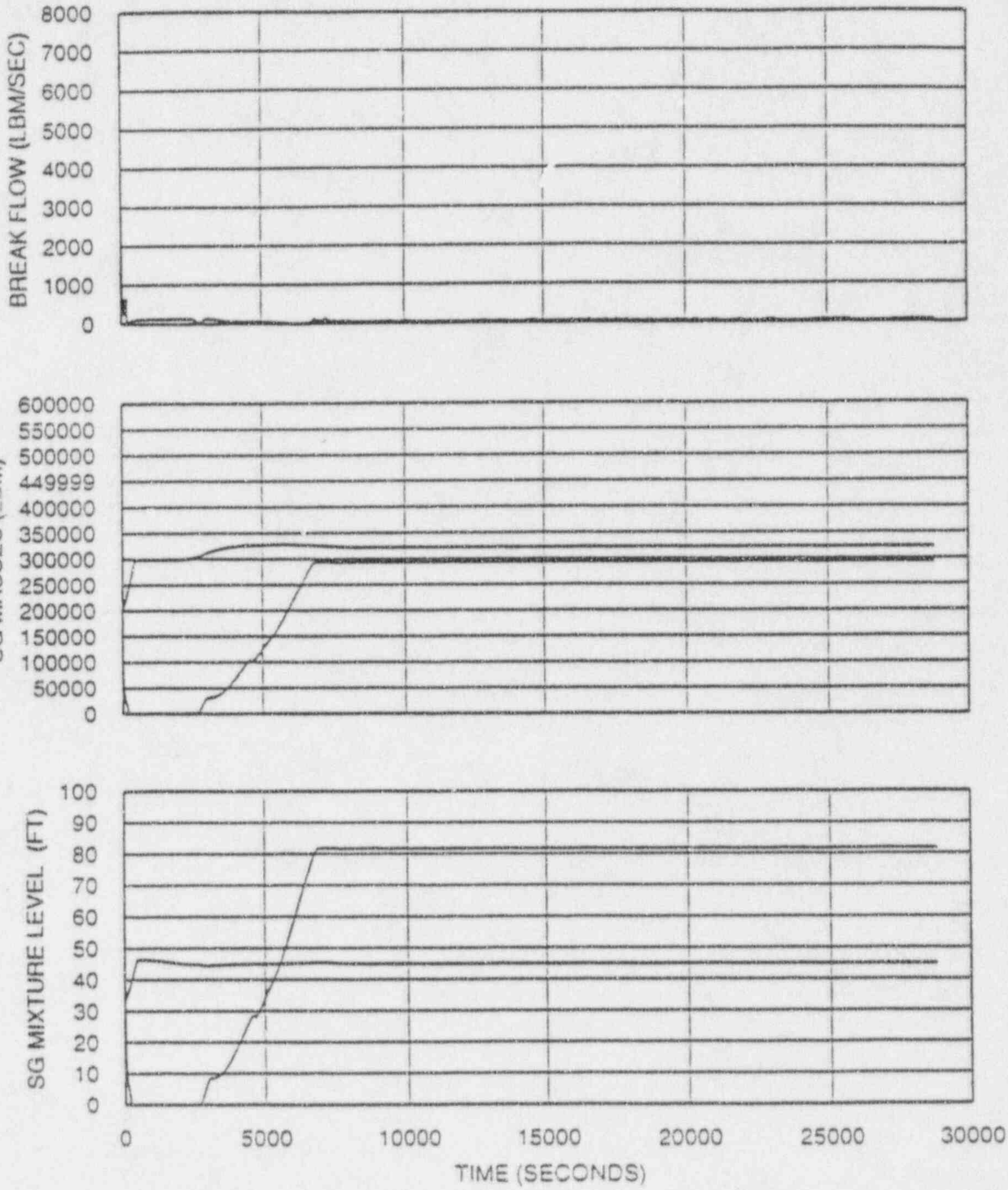
FIGURE 2 Sh 5



MYC-1592 REV 0
MSLR BASE SGTR ANALYSIS

MY MSLR 3-SGTR - rnvm303
NO LOOP - RCPs MAINTAINED
HPSI THROT 1800/ADV AT 1800

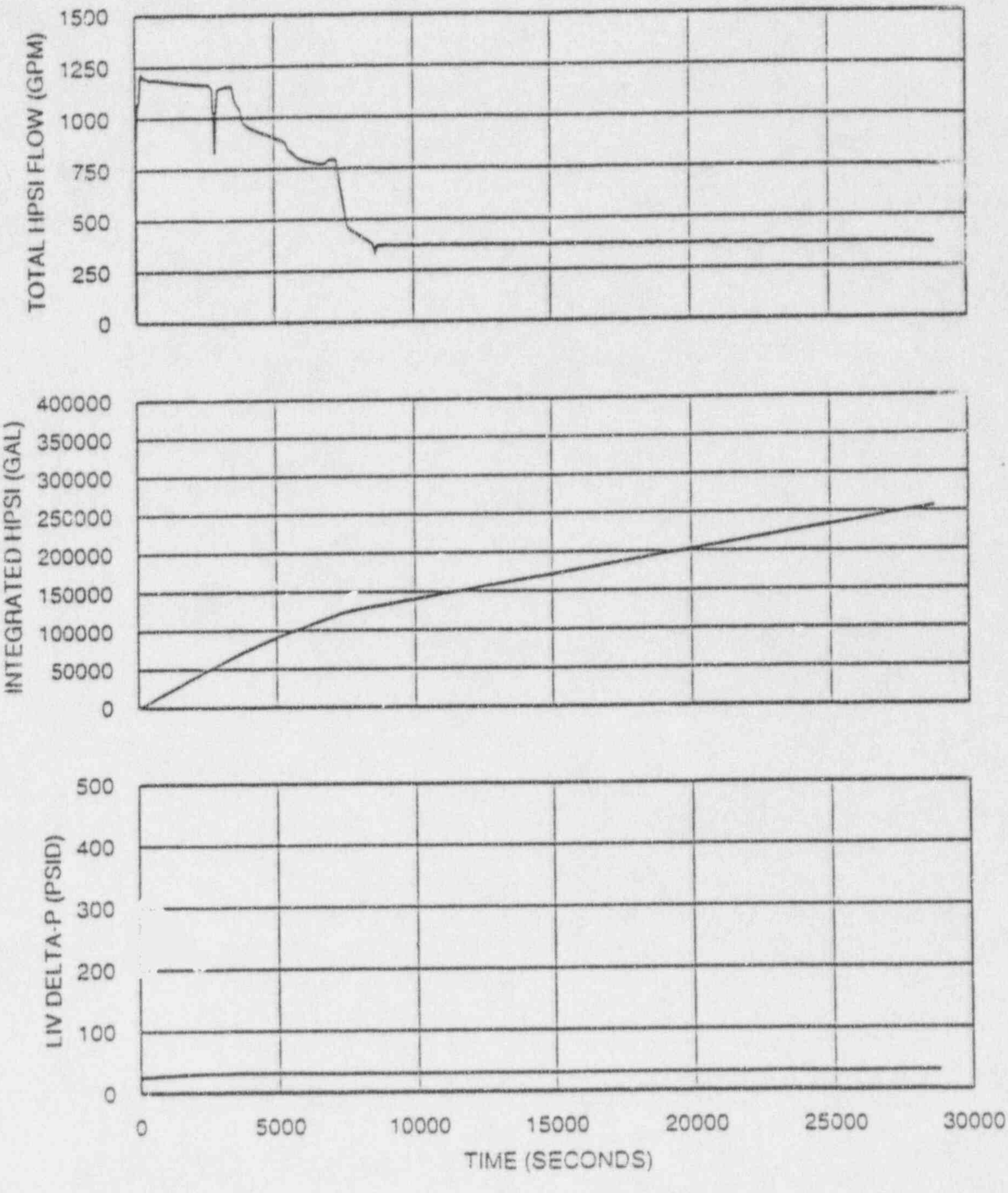
FIGURE 2 . Sh 6



MYC-1592 REV 0
MSLR BASE SGTR ANALYSIS

MY MSLR 3-SGTR - mvm303
NO LOOP - RCPs MAINTAINED
HPSI THROT 1800/ADV AT 1800

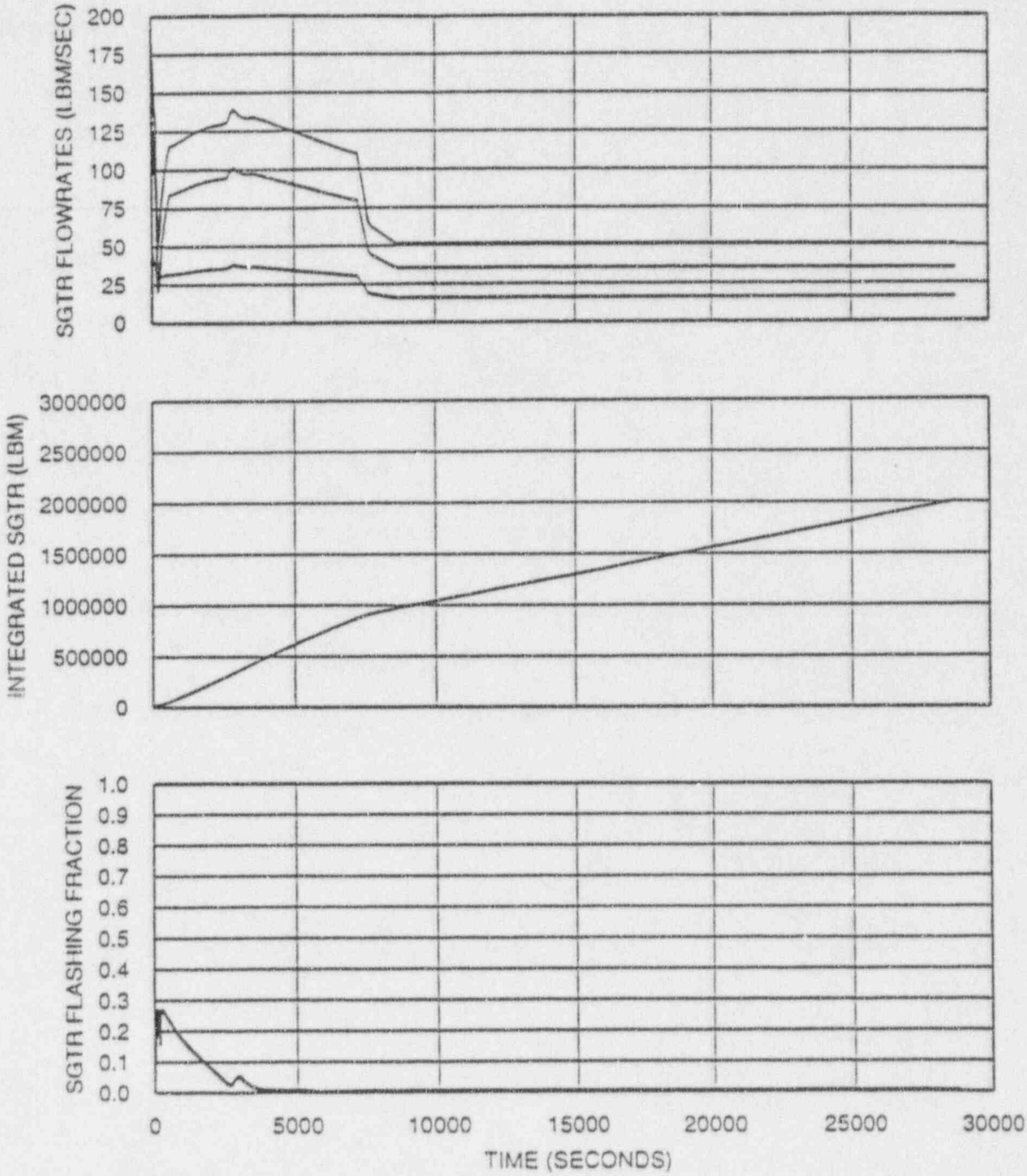
FIGURE 3 . Sh T



MYC-1592 REV 0
MSLR BASE SGTR ANALYSIS

MY MSLR 3-SGTR - mvm303
NO LOOP - RCPs MAINTAINED
HPSI THROT 1800/ADV AT 1800.

FIGURE 2 . Sn 8

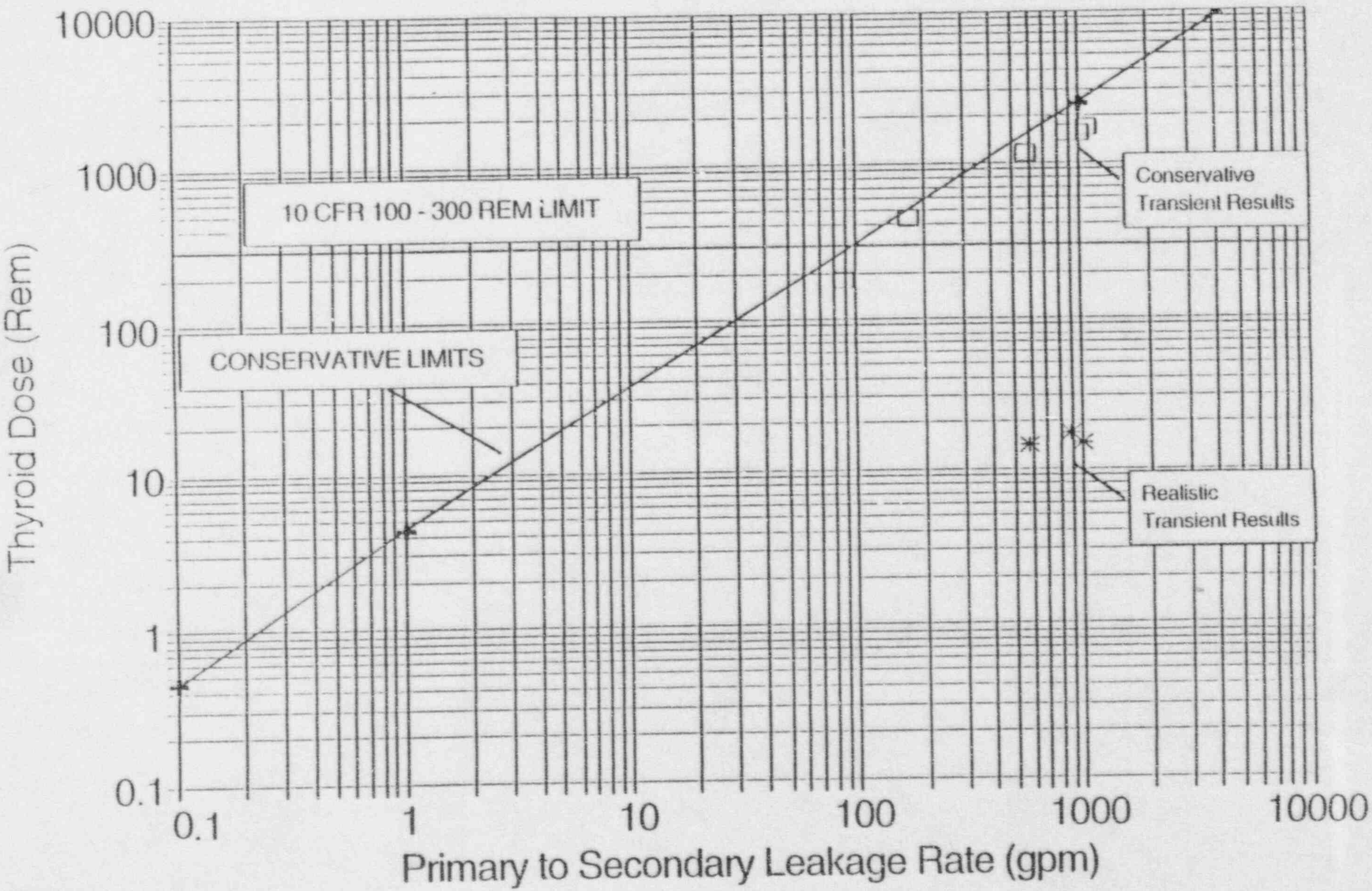


MYC-1592 REV 0
MSLR BASE SGTR ANALYSIS

MY MSLR 3-SGTR - mvm303
NO LOOP RCPs MAINTAINED
HPSI THRO. 1300/ADV AT 1800

FIGURE 2 Sh. 9

Figure 3
EAB 2-Hour Thyroid Dose - Pre-Spike



Attachment A

Offsite Doses as a Function of Primary-to-Secondary Leakage
Due to MSLR with Induced Steam Generator Tube Leakage
(Reference 3)

Yankee Atomic Electric Co. Memo REG203/94 may be observed in this letter's Enclosure 6 as Attachment A of Yankee Atomic Electric Co. Memo TAG-MY-94-51.

ENCLOSURE 7

MAINE YANKEE NOVEMBER 14, 1994 LETTER ENCLOSURE'S REFERENCE 10

Yankee Atomic Electric Co. Memo REG 203/94, "Offsite Doses as a Function of Primary-to-Secondary Leakage due to MSLB with Induced Steam Generator Tube Leakage (Rev. 1)", October 26, 1994

MEMORANDUM

YANKEE ATOMIC - BOLTON

To	P. L. Anderson	Date	October 26, 1994
		Group #	REG 203/94
From	Y. J. Yu	W.O.#	5037
Subject	OFFSITE DOSES AS A FUNCTION OF PRIMARY-TO-SECONDARY LEAKAGE DUE TO MSLR WITH INDUCED STEAM GENERATOR TUBE LEAKAGE (REV. 1)	I.M.S.#	
		File #	

REFERENCES

1. Service Request M-94-131, "Steam Generator Tube Leakage Report," dated September 19, 1994.
2. Draft NUREG-1477, "Voltage-Based Interim Plugging Criteria For Steam Generator Tubes--Task Group Report," June 1, 1993.
3. FSAR Table B.1, "Dilution Factors For Accident Analysis At Maine Yankee."
4. FSAR Table A.2, "Equilibrium Coolant Activity Concentrations."
5. Standard Review Plan 15.6.3.
6. Plant Data of Coolant Activity Concentrations on January 20, 1987, from John Mayer of Maine Yankee Project.

EXECUTIVE SUMMARY

A conservative analysis has been performed to determine the offsite dose as a function of the primary-to-secondary leakage rate following a postulated Main Steam Line Rupture (MSLR), outside containment and upstream of both the Non-Return Valve (NRV) and Excess Flow Check Valve (EFCV) over a range of effective steam generator tube leakages. Based on the analysis, a set of figures showing offsite doses as a function of the primary-to-secondary leakage rate was provided. These figures can be used to obtain a conservative estimate of offsite doses following a postulated MSLR for a constant primary-to-secondary leakage from 0.1 to 100 gpm.

BACKGROUND INFORMATION

In support of Reference 1, MY Project requested REG to provide figures similar to those in Section 4.1 of Reference 2 that shows offsite doses as a function of the primary-to-secondary leakage rate following a main steam line rupture with induced steam generator tube leakage.

DISCUSSION

A conservative analysis has been performed to determine the offsite dose as a function of the primary-to-secondary leakage rate following a postulated MSLR, outside containment and upstream of both the Non-Return Valve (NRV) and Excess Flow Check Valve (EFCV) over a range of effective steam generator tube leakages. The analysis conservatively assumed rapid dryout of the faulted steam generator, and a constant induced primary-to-secondary leakage from 0.1 to 100 gpm. Two cases were analyzed: the conservative case and the realistic case. The conservative case used Standard Review Plan (SRP) (Reference 5) assumptions in the calculation of offsite doses. In the realistic case, offsite doses were calculated using realistic assumptions (i.e., maximum plant coolant activity concentrations of Cycle 9 were used). These assumptions are listed in Table 1. In addition, it is assumed that (1) the primary coolant mass is constant and (2) the flashing of the leakage coolant is 100% for the duration of the accident. The results of the calculations are shown in Figures 1 through 12. These figures include:

1. EAB 2-Hour Thyroid Dose - Pre-existing Iodine Spike (Conservative)
2. LPZ 8-Hour Thyroid Dose - Pre-existing Iodine Spike (Conservative)
3. EAB 2-Hour Whole Body Dose - Pre-existing Iodine Spike (Conservative)
4. LPZ 8-Hour Whole Body Dose - Pre-existing Iodine Spike (Conservative)
5. EAB 2-Hour Thyroid Dose - Coincident Spike (Conservative & Realistic)
6. LPZ 8-Hour Thyroid Dose - Coincident Spike (Conservative & Realistic)
7. EAB 2-Hour Whole Body Dose - Coincident Spike (Conservative & Realistic)
8. LPZ 8-Hour Whole Body Dose - Coincident Spike (Conservative & Realistic)
9. EAB 2-Hour Thyroid Dose - Coincident Spike (Realistic)
10. LPZ 8-Hour Thyroid Dose - Coincident Spike (Realistic)
11. EAB 2-Hour Whole Body Dose - Coincident Spike (Realistic)
12. LPZ 2-Hour Whole Body Dose - Coincident Spike (Realistic)

It should be noted that a reference condition of the primary coolant of 2250 psia and 552°F was used in the development of these curves. If the primary-to-secondary leakage rate is at a different condition, the density must be corrected to the reference condition, prior to finding the doses for the corresponding leakage rate from these curves. For example, if the leak rate is measured to be 4.5 gpm at 2500 psia and 100°F, this would correspond to 6 gpm at the reference condition. This can be determined as follows:

$LR_{ref} = LR_{meas} \times \rho_{meas} / \rho_{ref}$

where

LR_{meas} = leak rate at the measured condition = 4.5 gpm
ρ_{meas} = density at the measured condition = 62.46 lbm/ft^3
ρ_{ref} = density at the reference condition = 46.73 lbm/ft^3

So, LR_{ref} = leak rate at the reference condition
= 4.5 gpm x 62.46 / 46.73 = 6.0 gpm.

The corresponding doses from Figures 1 to 8 are shown in Table 2 for the conservative case. These figures are most meaningful for leakage in the range of tens of gpm. At higher leakage rates the system's effects of fluid loss should be explicitly considered by thermal hydraulic analyses. The assumption of a uniform leak rate, upon which these curves are based, becomes less mechanistic as the rate increases.

The results provided in this memo are based on scoping analyses and are not intended to be construed as a replacement for any of the licensing analyses presented in the FSAR. All of the analyses supporting these assessments have been reviewed for reasonableness. This assessment has not been prepared or reviewed under the YNSD Quality Assurance Program.

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YJY/lll
Attachments

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V. M. Esquillo
R. A. Marcello
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K. R. Rousseau
S. Spanos

Table 1

Maine Yankee MSLR Radiological Analysis Assumptions

Radiological Parameter	Conservative Assumptions	Realistic Assumptions
Reactor Coolant Activity Concentrations ($\mu Ci/g$)		
D.E I-131 Iodine Activity	1.0	0.16
Total Noble Gas Activity	100/Ē	8.2
Pre-Existing Iodine Spiking Reactor Coolant D.E I-131 Activity Concentration ($\mu Ci/g$)	60	0.0
Coincident Iodine Spiking Factor	500	500
Steam Generator Liquid Activity Concentrations ($\mu Ci/g$)		
D.E. I-131 Iodine Activity	0.1	1.0E-05
Steam Generator Partition Factors		
Intact	100	100
Ruptured	100	100
Breathing Rate (m^3/sec)		
(0 - 8 hours)	3.47E-04	3.47E-04
(8 - 24 hours)	1.75E-04	1.75E-04
(> 24 hours	2.32E-04	2.32E-04
Concentration χ/Q's (sec/m^3)		
EAB (0 - 1 hour)	5.40E-04	9.92E-05
(1 - 2 hours)	3.11E-04	6.51E-05
LPZ (0 - 1 hour)	3.88E-05	2.58E-06
(1 - 2 hours)	2.15E-05	1.91E-06
(2 - 8 hours)	1.01E-05	1.30E-06

Table 2

Maine Yankee Offsite Doses (Rem) Following A Postulated MSLR
With 4.5 gpm Primary-to-Secondary Leakage At 2500 psia And 100°F

Offsite Doses	Pre-existing Spike	Coincident Spike
EAB 2-hour Thyroid Dose	21	17
LPZ 8-hour Thyroid Dose	2.5	4.8
EAB 2-hour Whole-Body Dose	0.19	0.7
LPZ 8-hour Whole-Body Dose	0.019	0.12

Figure 1
EAB 2-Hour Thyroid Dose - Pre-Spike

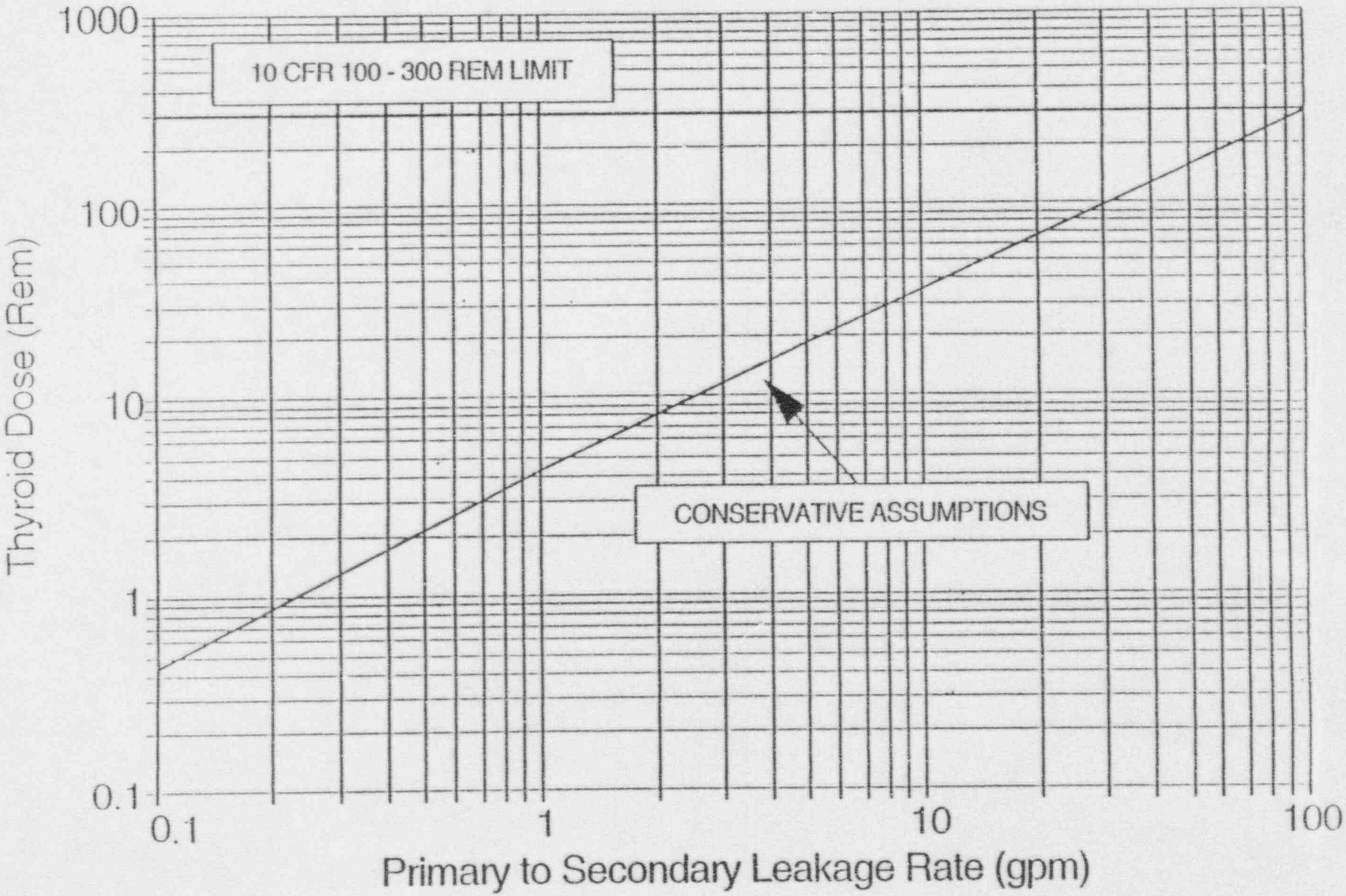


Figure 2
LPZ 8-Hour Thyroid Dose - Pre-Spike

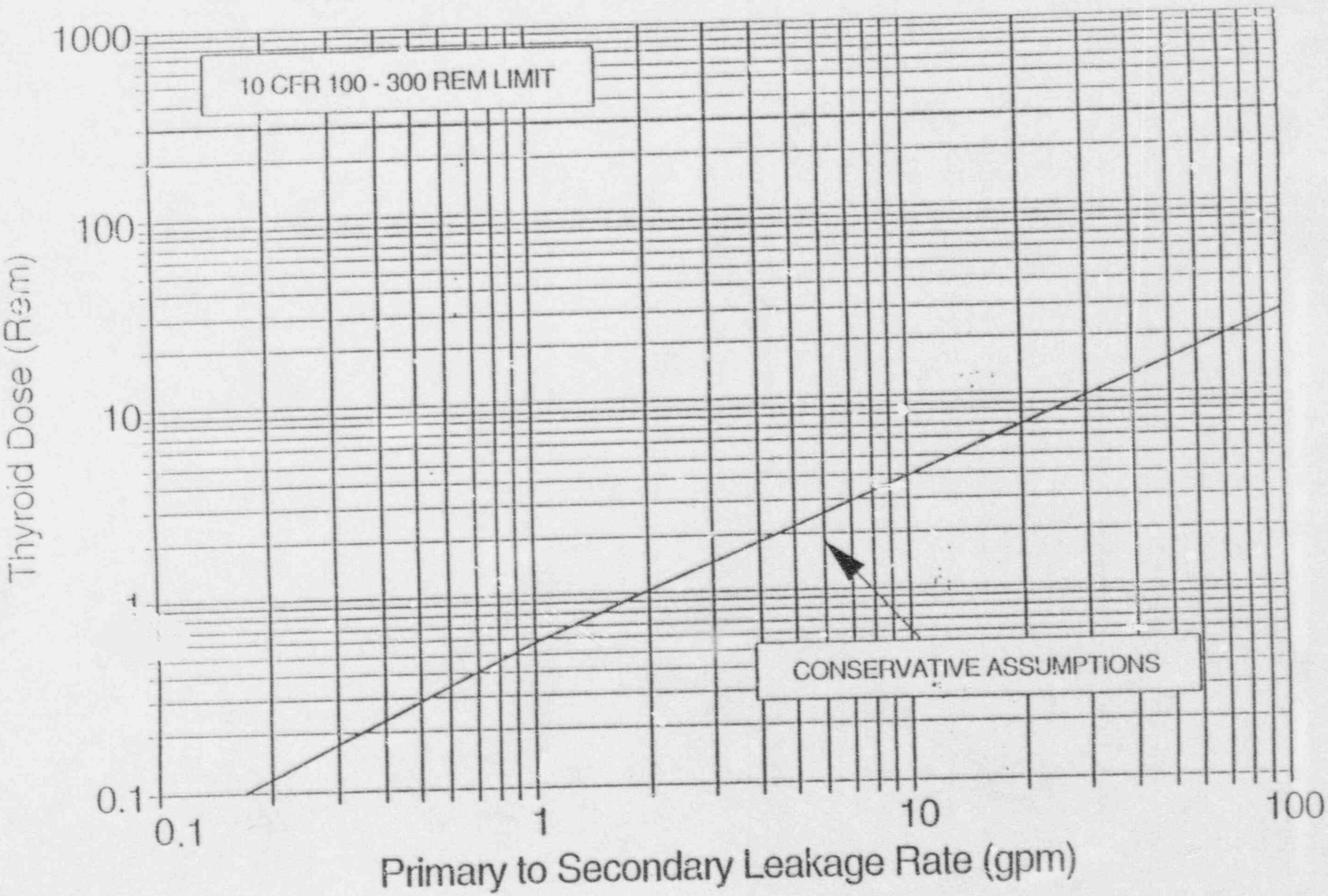


Figure 3
EAB 2-Hour Whole Body Dose - Pre-Spike

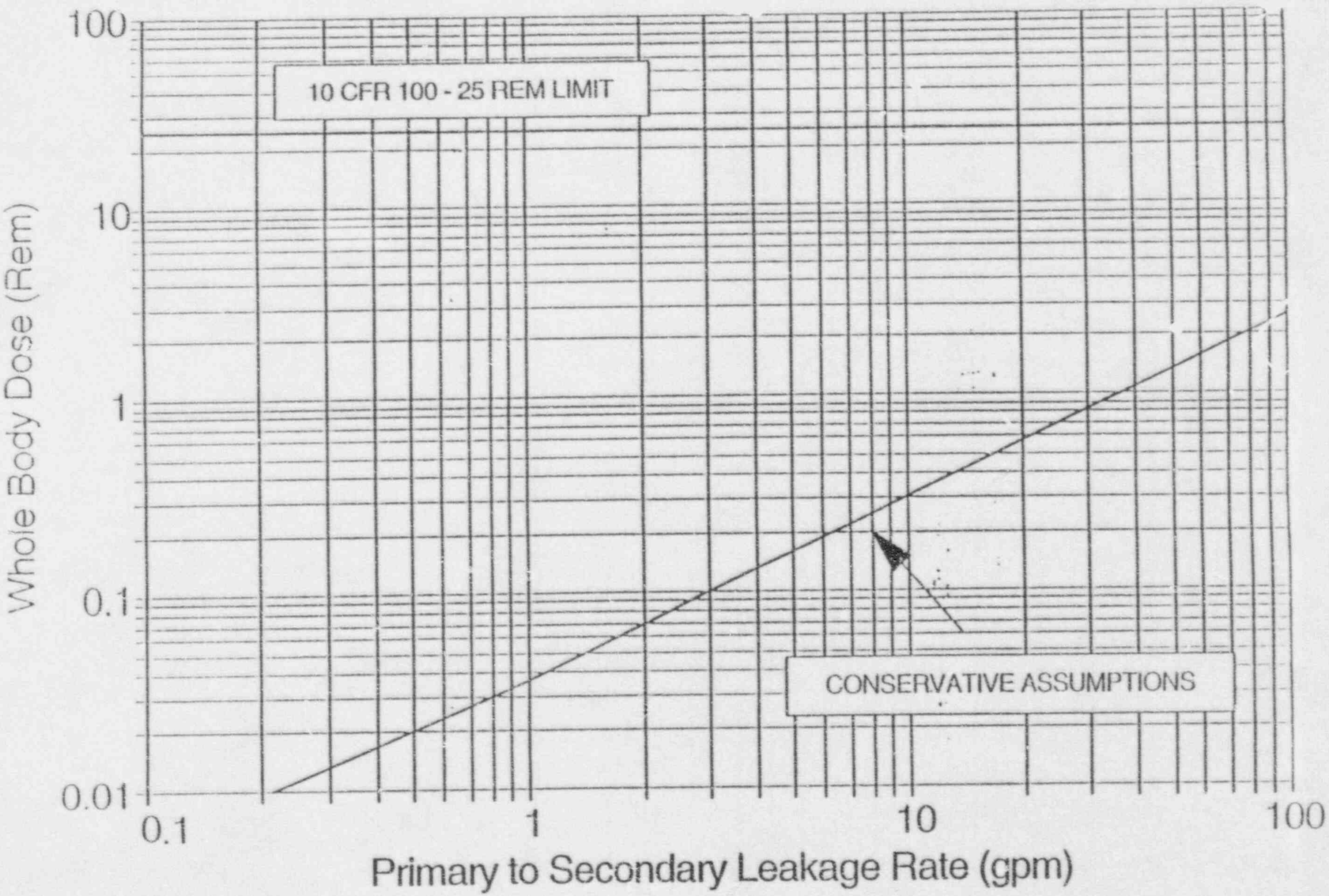


Figure 4
LPZ 8-Hour Whole Body Dose - Pre-Spike

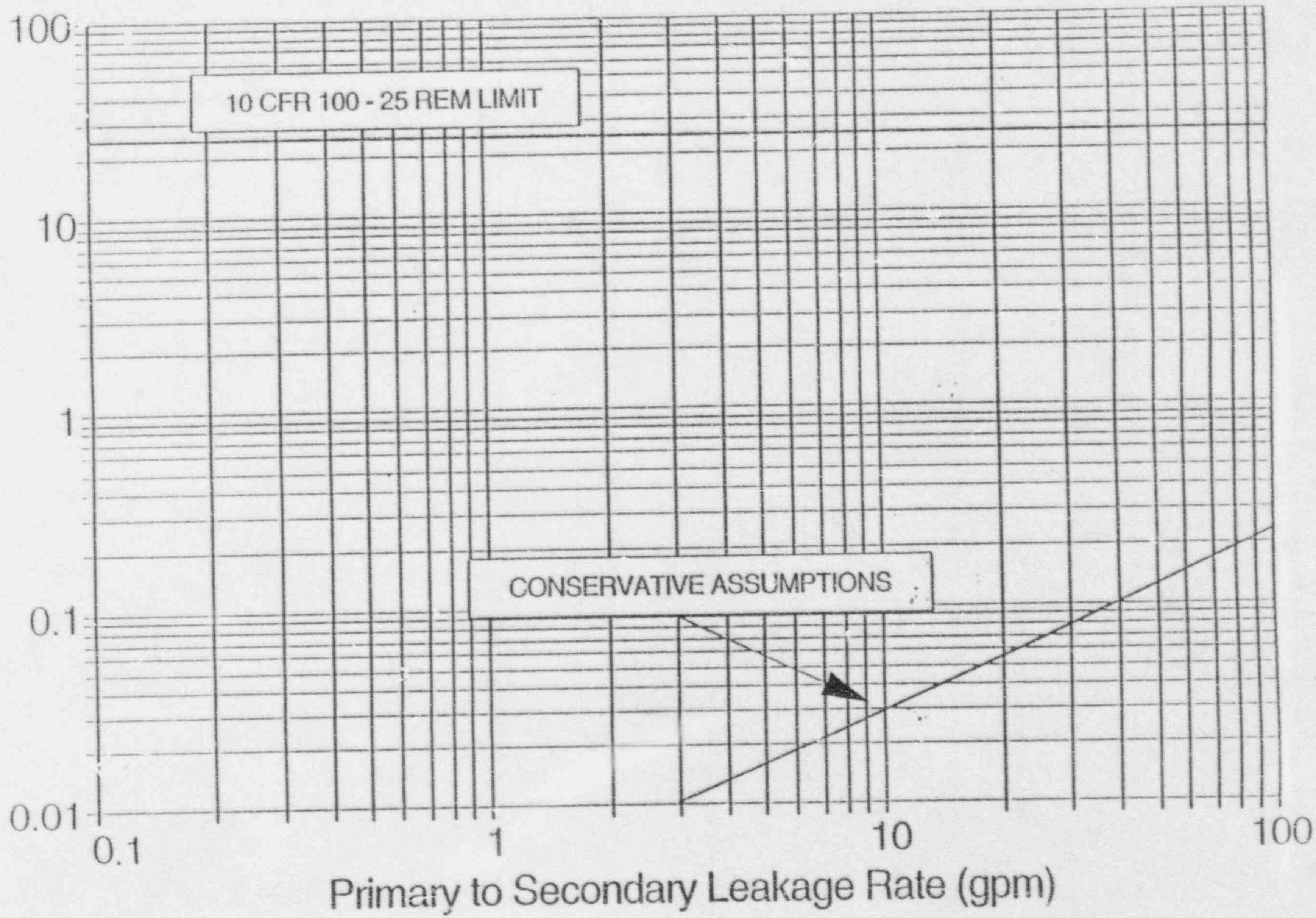


Figure 5
EAB 2-Hour Thyroid Dose - Co-Spike

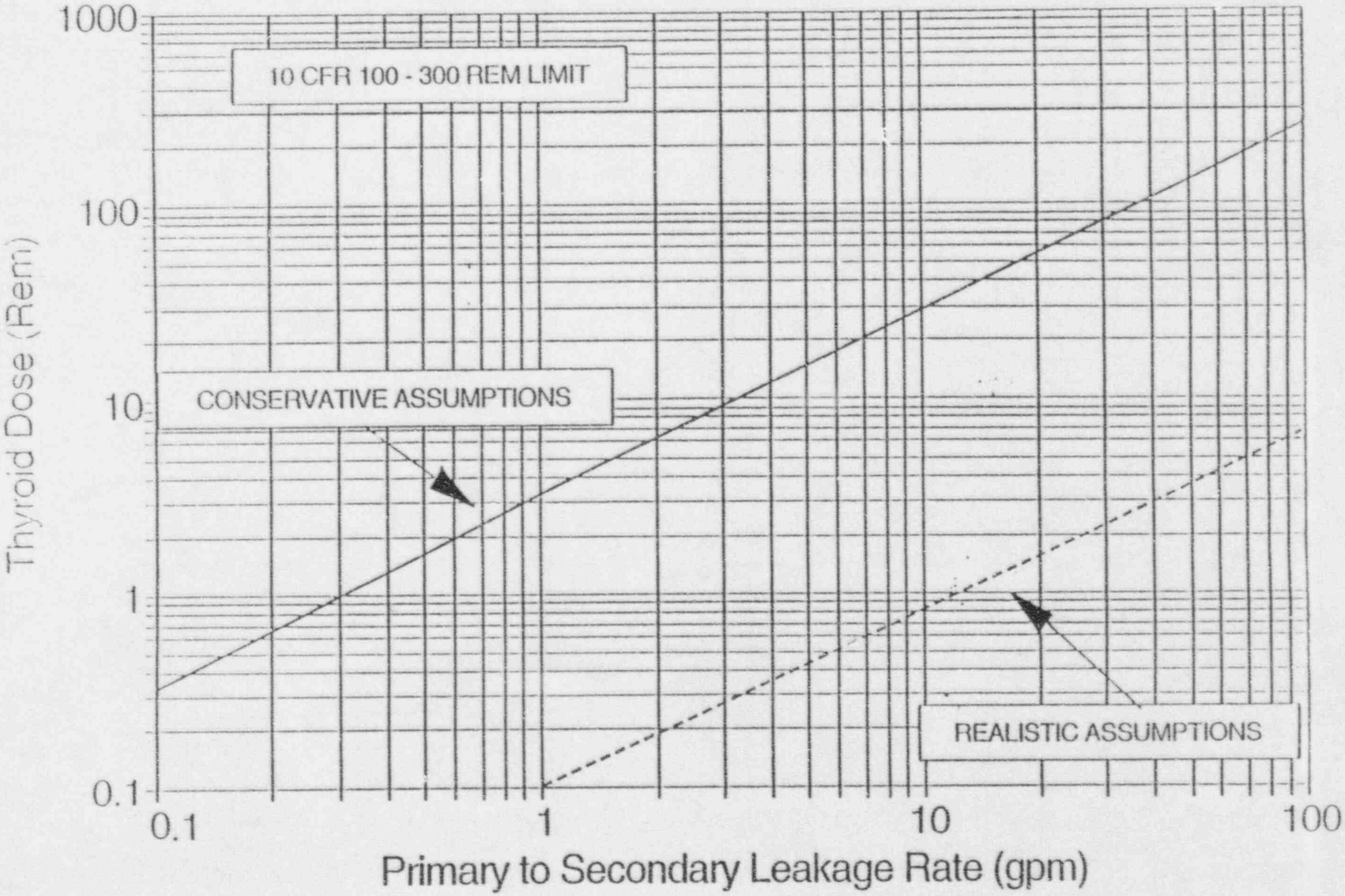


Figure 6
LPZ 8-Hour Thyroid Dose - Co-Spike

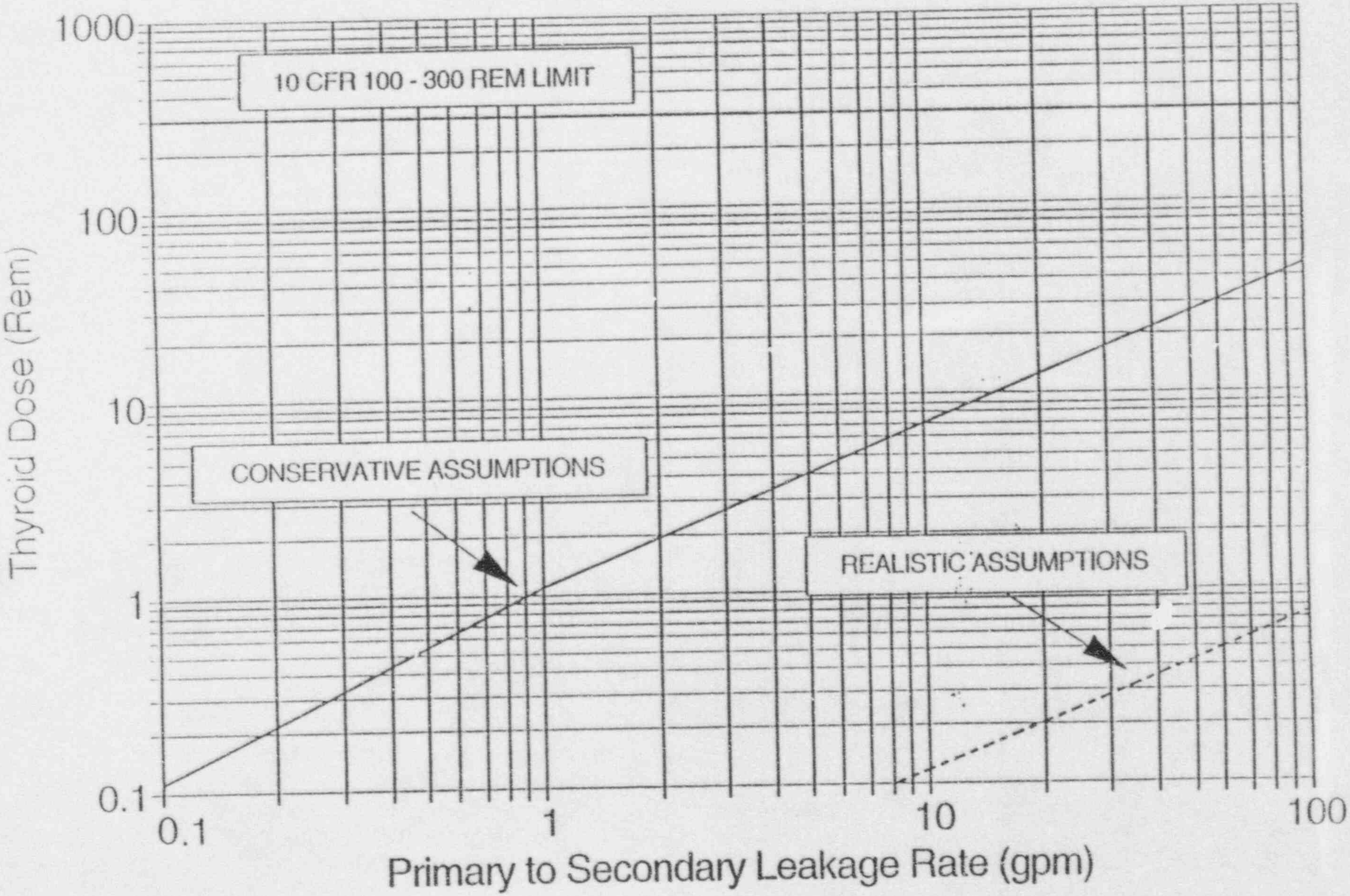


Figure 7
EAB 2-Hour Whole Body Dose - Co-Spike

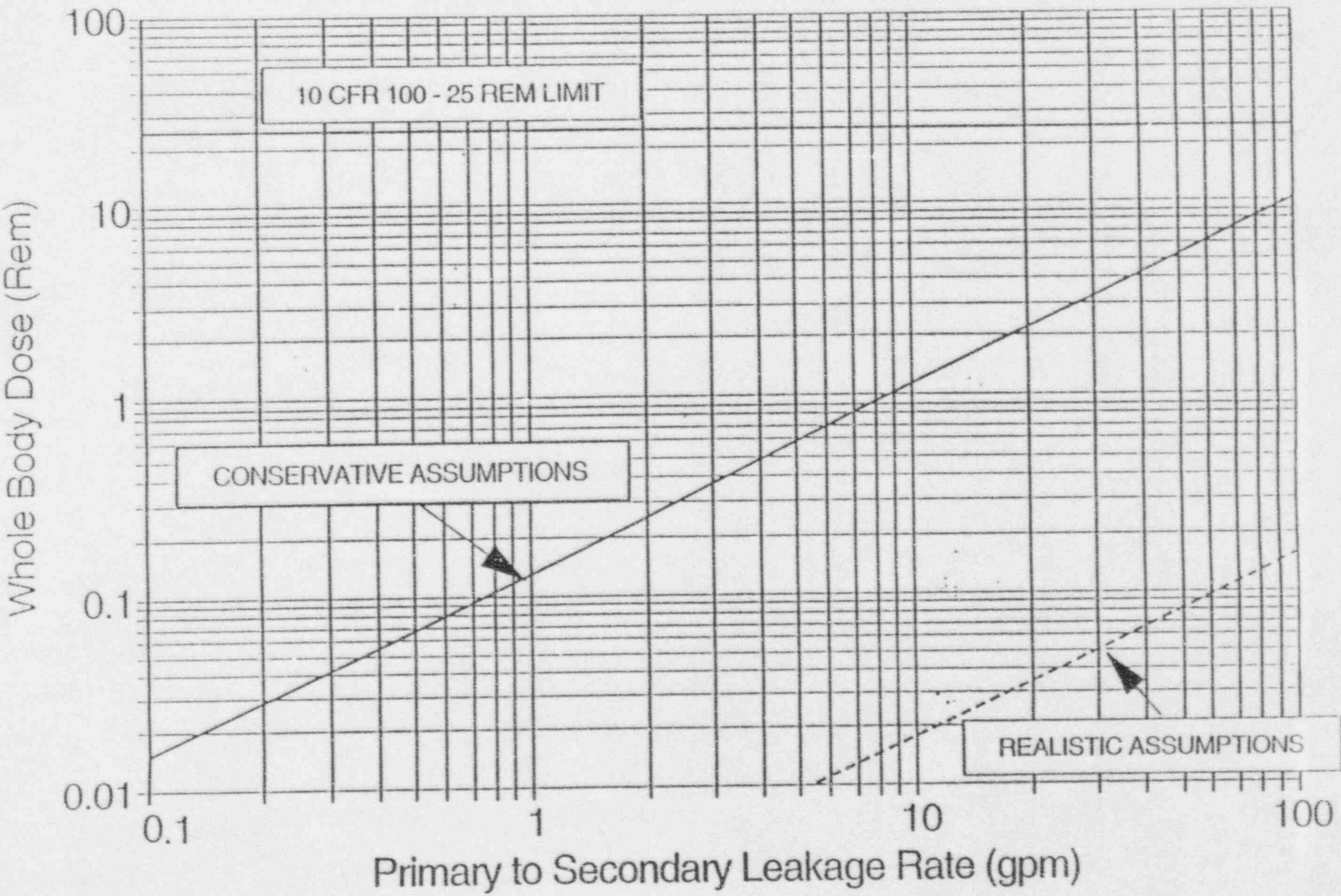


Figure 8
LPZ 8-Hour Whole Body Dose - Co-Spike

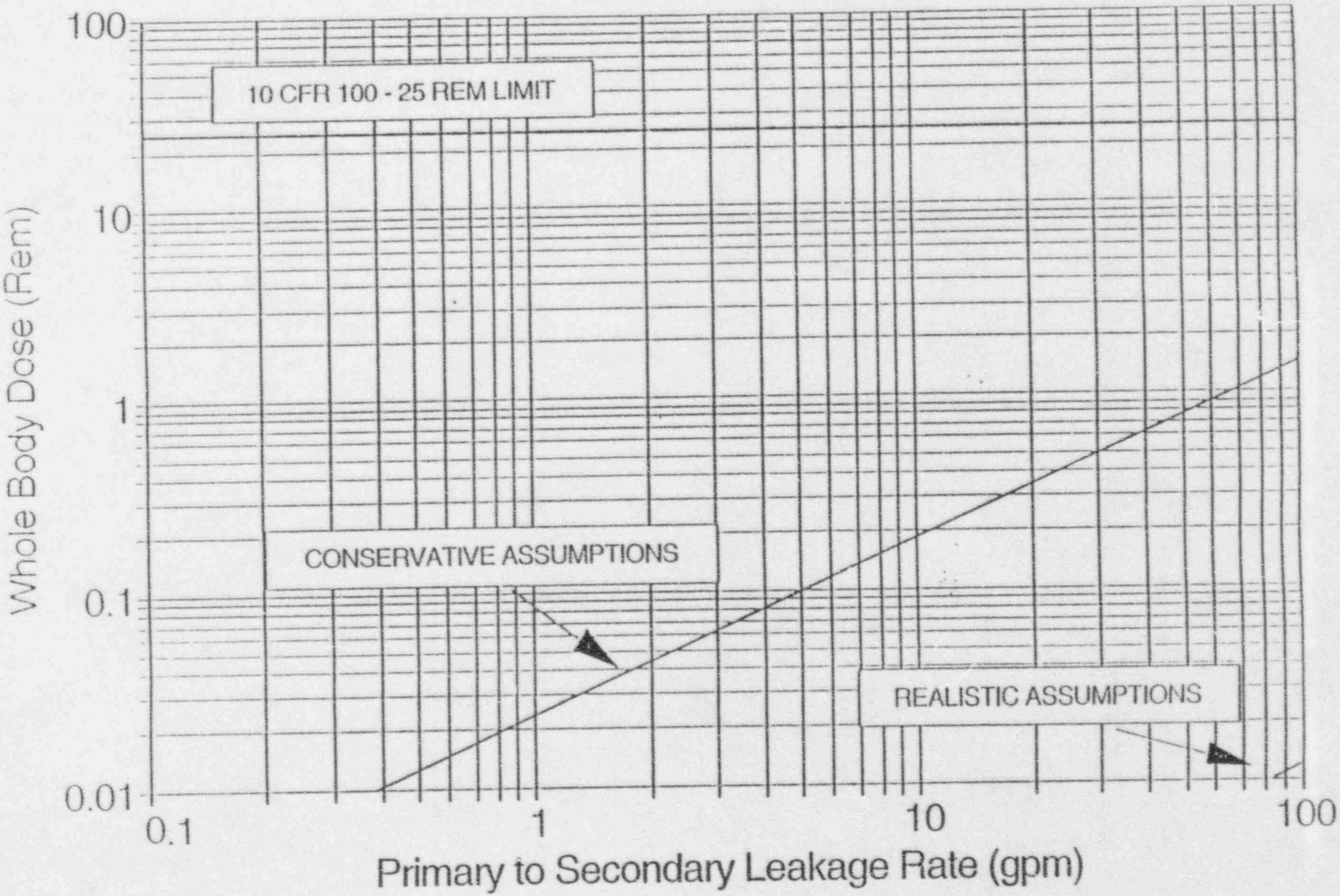


Figure 9
EAB 2-Hour Thyroid Dose - Co-Spike

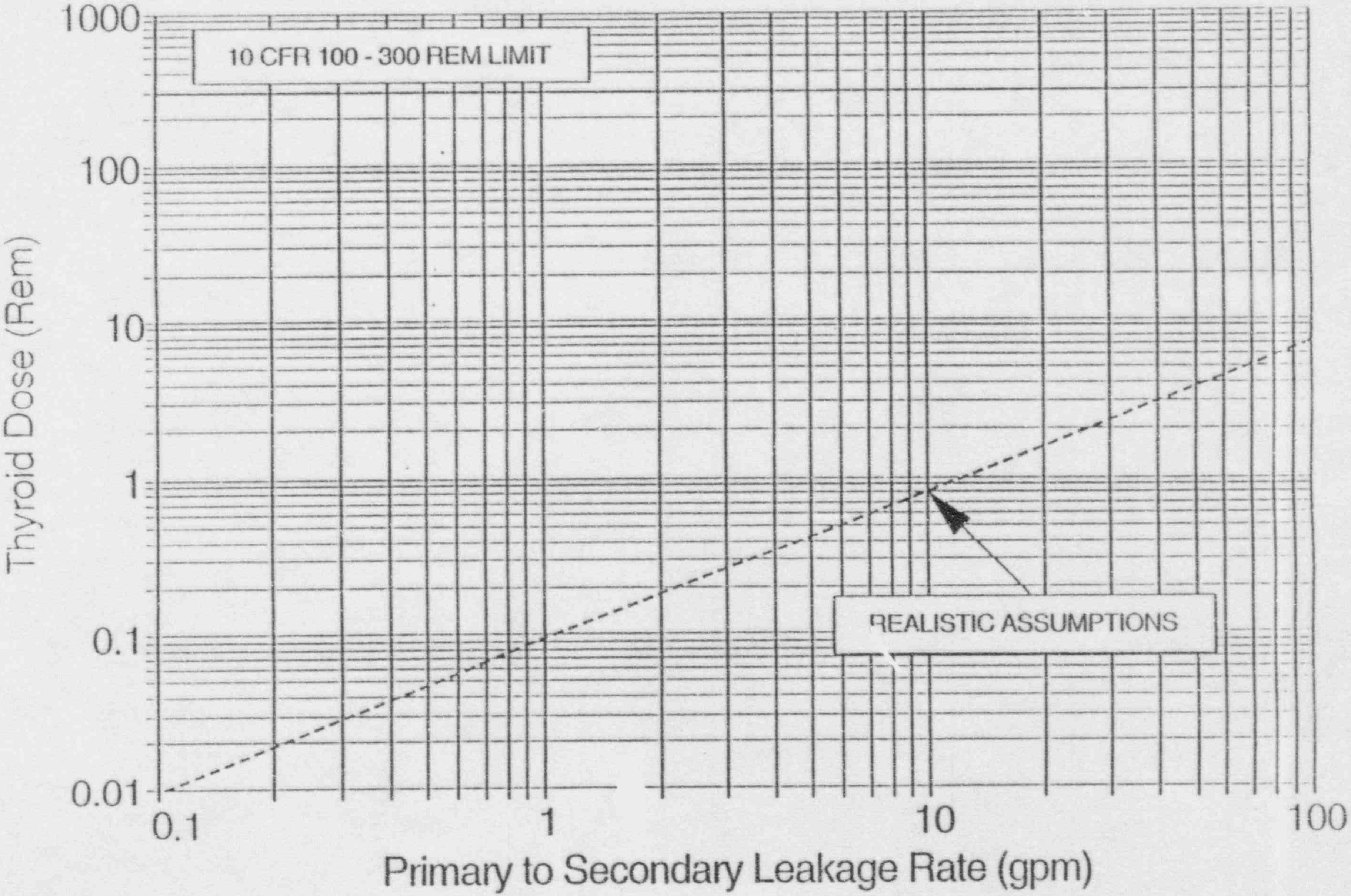


Figure 10
LPZ 8-Hour Thyroid Dose - Co-Spike

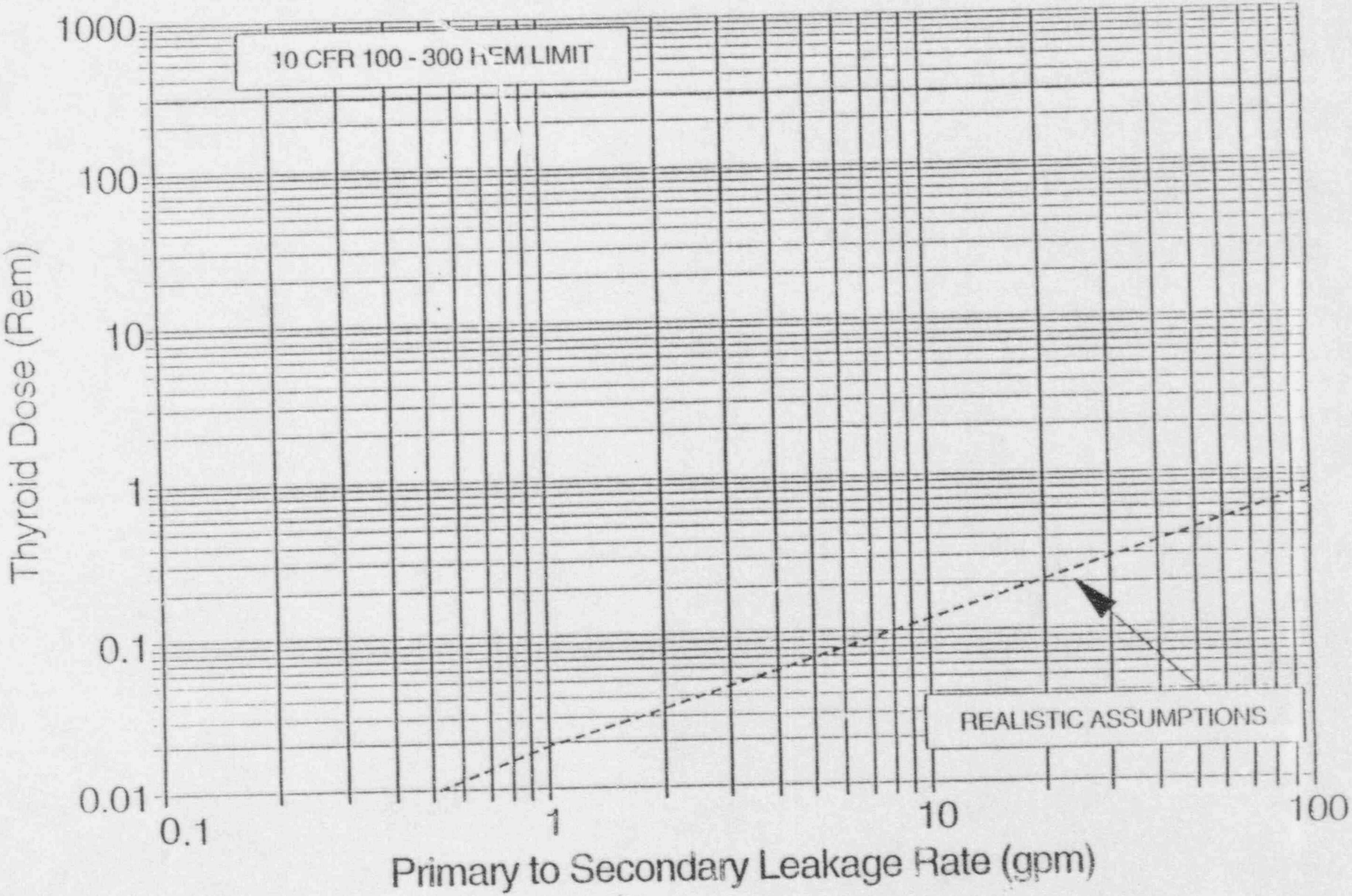


Figure 11
EAB 2-Hour Whole Body Dose - Co-Spike

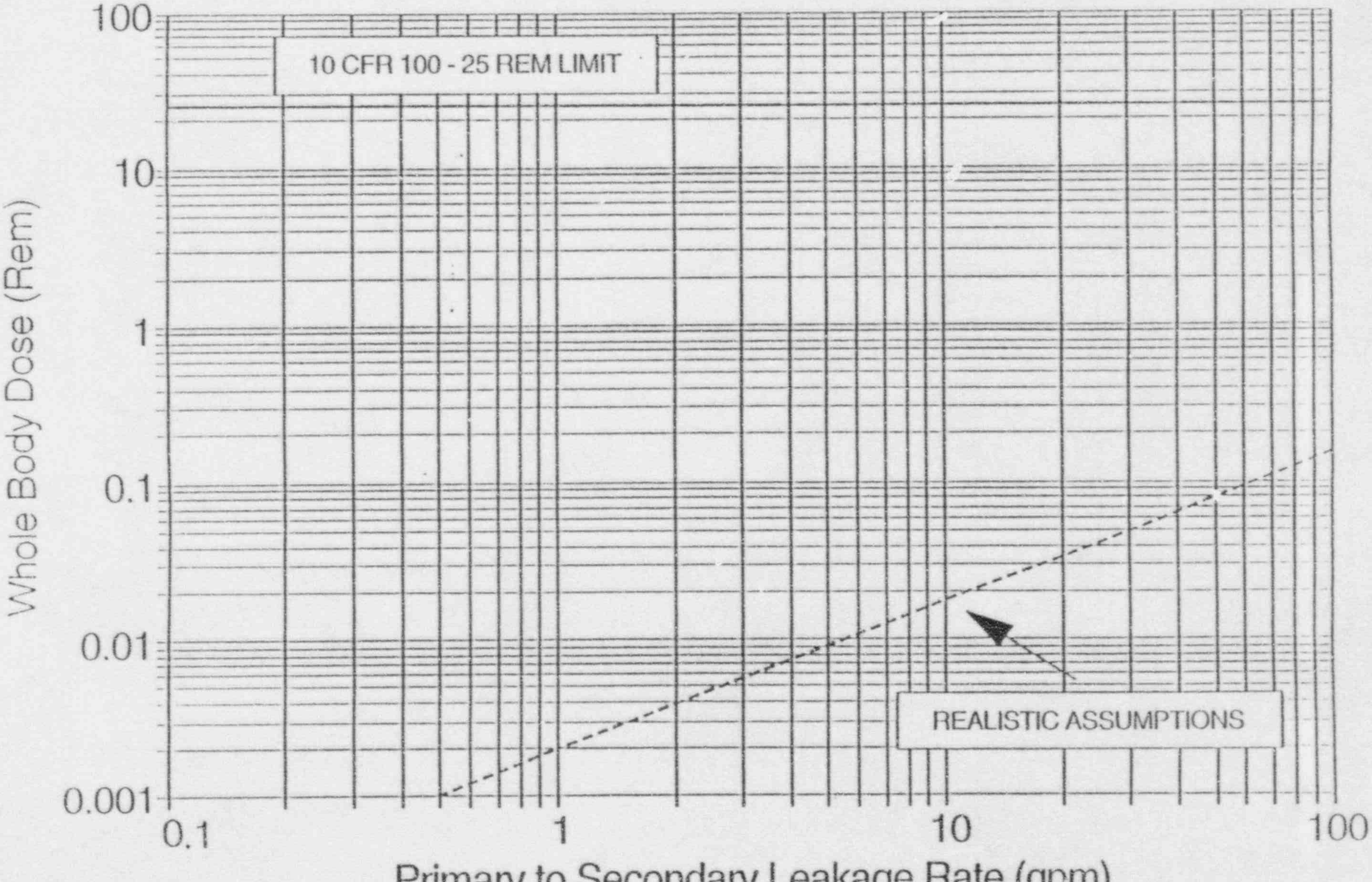
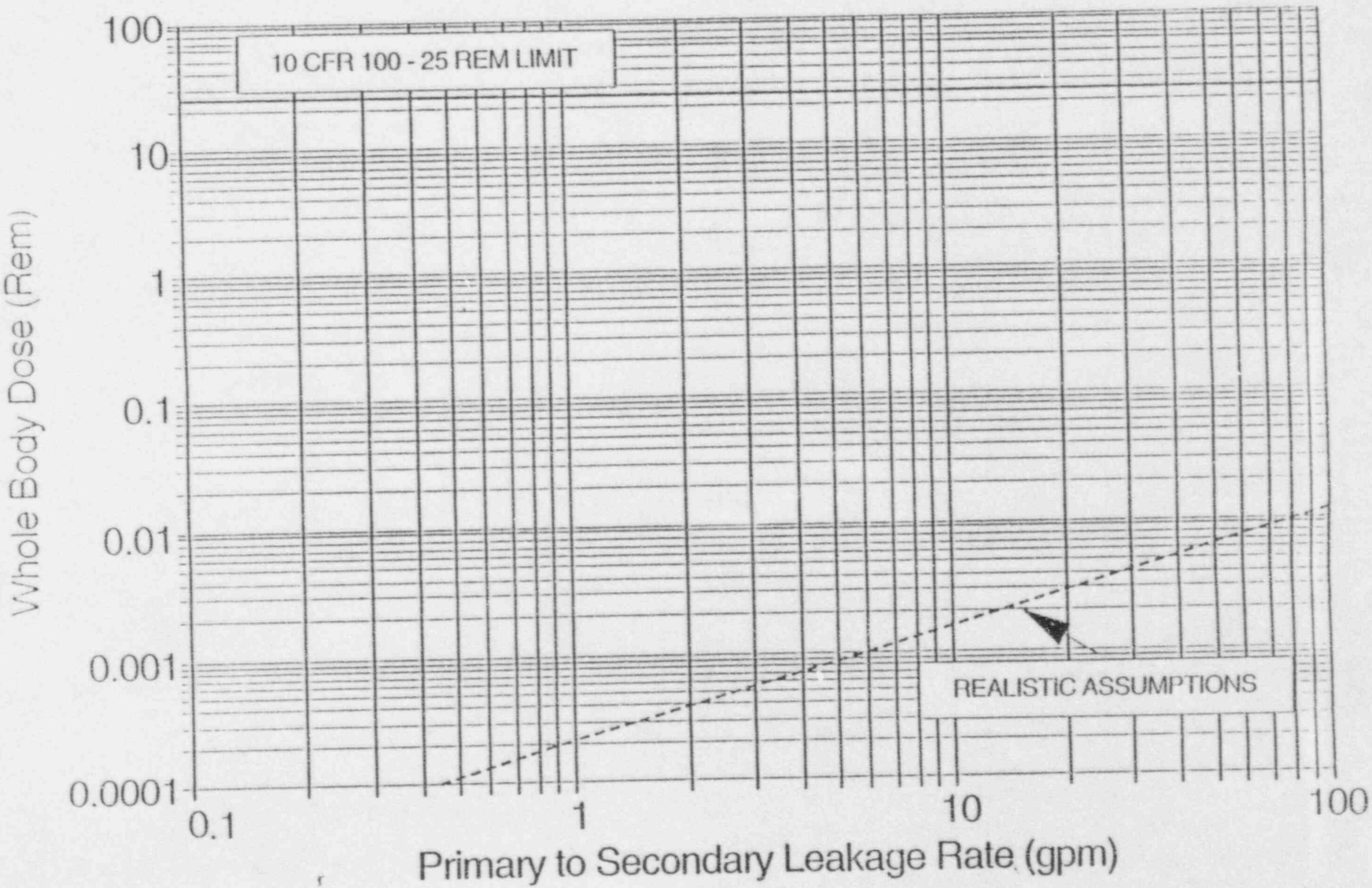


Figure 12
LPZ 8-Hour Whole Body Dose - Co-Spike



Attachment B

Letter to ABB-CE Describing Maximum Primary-to-Secondary Pressure Differential During MSLR Event
(Reference 4)

YANKEE ATOMIC ELECTRIC COMPANY

Telephone (508) 779-6711
TWX 710-380-7619

YANKEE

580 Main Street, Bolton, Massachusetts 01740-1398

October 17, 1994

TAG-MY-94-050

Mr. Paul L. Anderson
ABB-CE
1201 Riverfront Parkway
Chattenooga, TN 37403

Dear Paul,

Attached please find a series of plots depicting a Main Steam Line Rupture at Maine Yankee. The intent of the supporting analysis was to determine the maximum differential pressure across the steam generator tubes. Chick Eames, at Maine Yankee, has asked that I formally transmit this information for use as a reference in your leakage evaluation.

The pressure challenges to the steam generator tubes occur at two distinct times. An initial peak differential pressure of 1739 psid occurs approximately four seconds into the event during the blowdown phase of the MSLR.

A second peak pressure of 2520 psid occurs approximately 12 minutes into the event following the steam generator dryout and subsequent RCS repressurization via the HPSI pumps and post-dryout RCS heatup.

I have also attached a short sequence of events to help sort through the plots. If you have any questions or need additional information, please feel free to call me at (508) 779-6711, Extension 2156.

Sincerely,

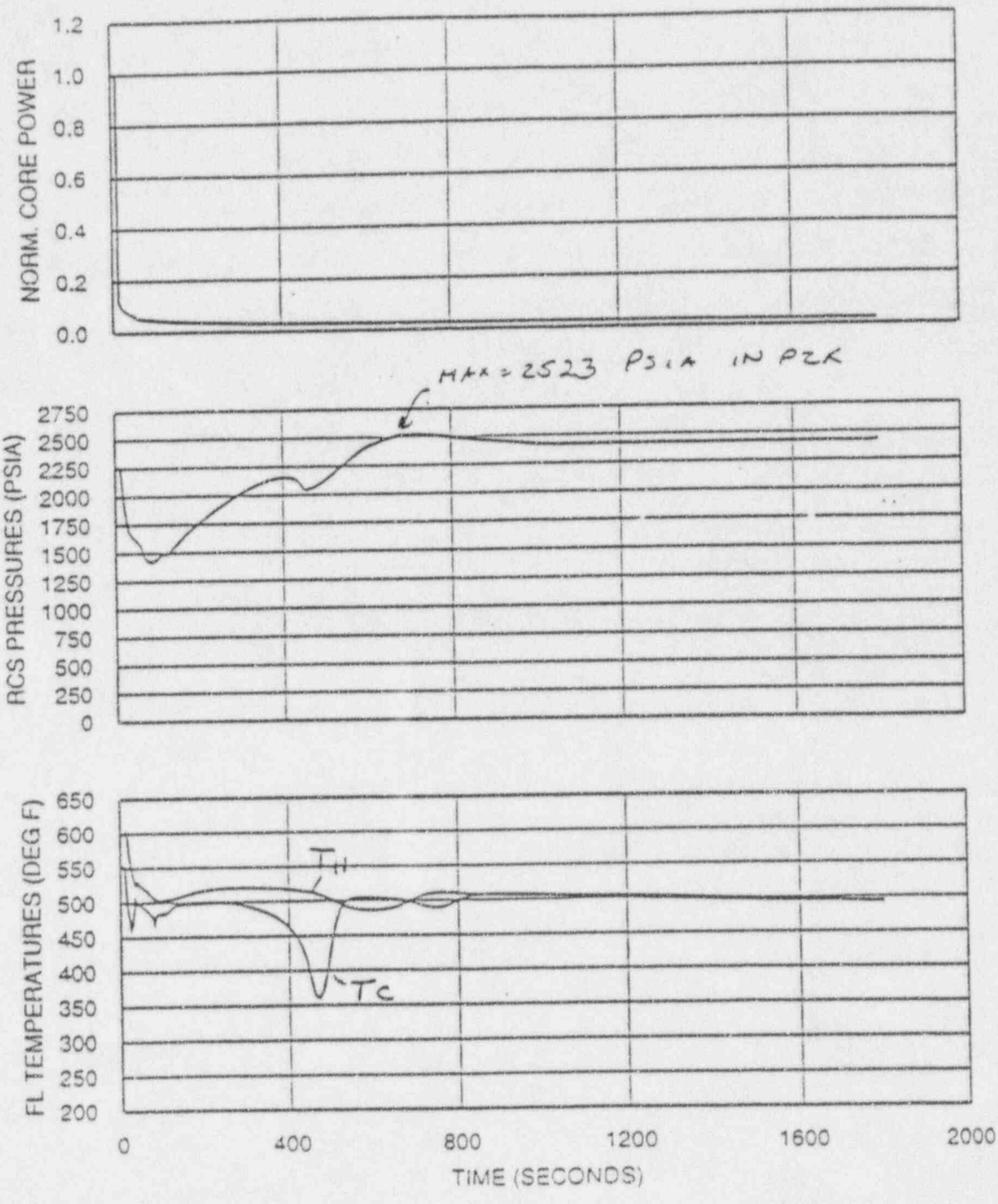
Kenneth R. Rousseau
Transient Analysis Group
Nuclear Engineering Department
Yankee Nuclear Services Division

KRR/meg
Attachments

bc: P. A. Bergeron
P. L. Anderson
J. R. Chapman

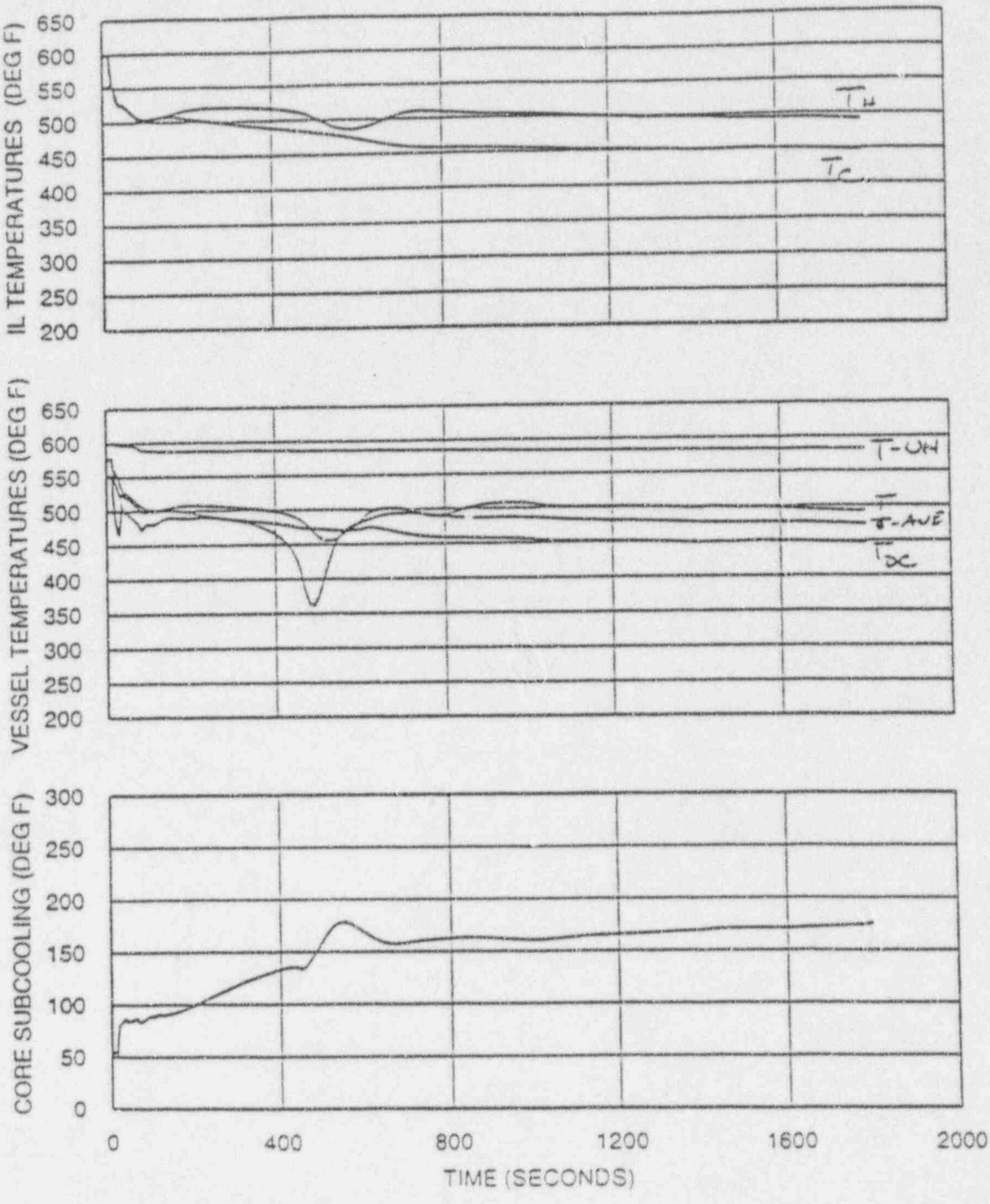
rnom500 - Main Steam Line Rupture - No SGTR

TIME (SEC)	EVENT
0 - 10.0	Steady state full power operation.
10.0	MSLR upstream of the NRV.
10.1	NRV closes preventing the other two SGs from blowing down.
11.2	Reactor trip signal on low steam line pressure (<500 psia)
11.4	EFCV Closure signal on SG #1 low steam line pressure (<415 psia). Isolate EFW to SG #1
11.4	MFWRVs begin to close - conservatively short stroke time (10 seconds) Low pressure in main steam line - all EFCVs begin to close Reactor scram commences
14.0	First peak primary-to-secondary pressure differential - 1739 psid
44.3	SIAS on low RCS pressure (<1600 psia). Trip of feedwater pumps, condensate pumps and heater drain pumps with coincident EFCV signal. HPSIs begin repressurization of the RCS.
460.0	SG #1 dryout occurs
390.0	EFW isolation to intact steam generators - low steam line pressure (<415 psia)
710.0	Maximim primary to secondary pressure differential = 2520.2 psid occurs
1800.0	RETRAN terminated

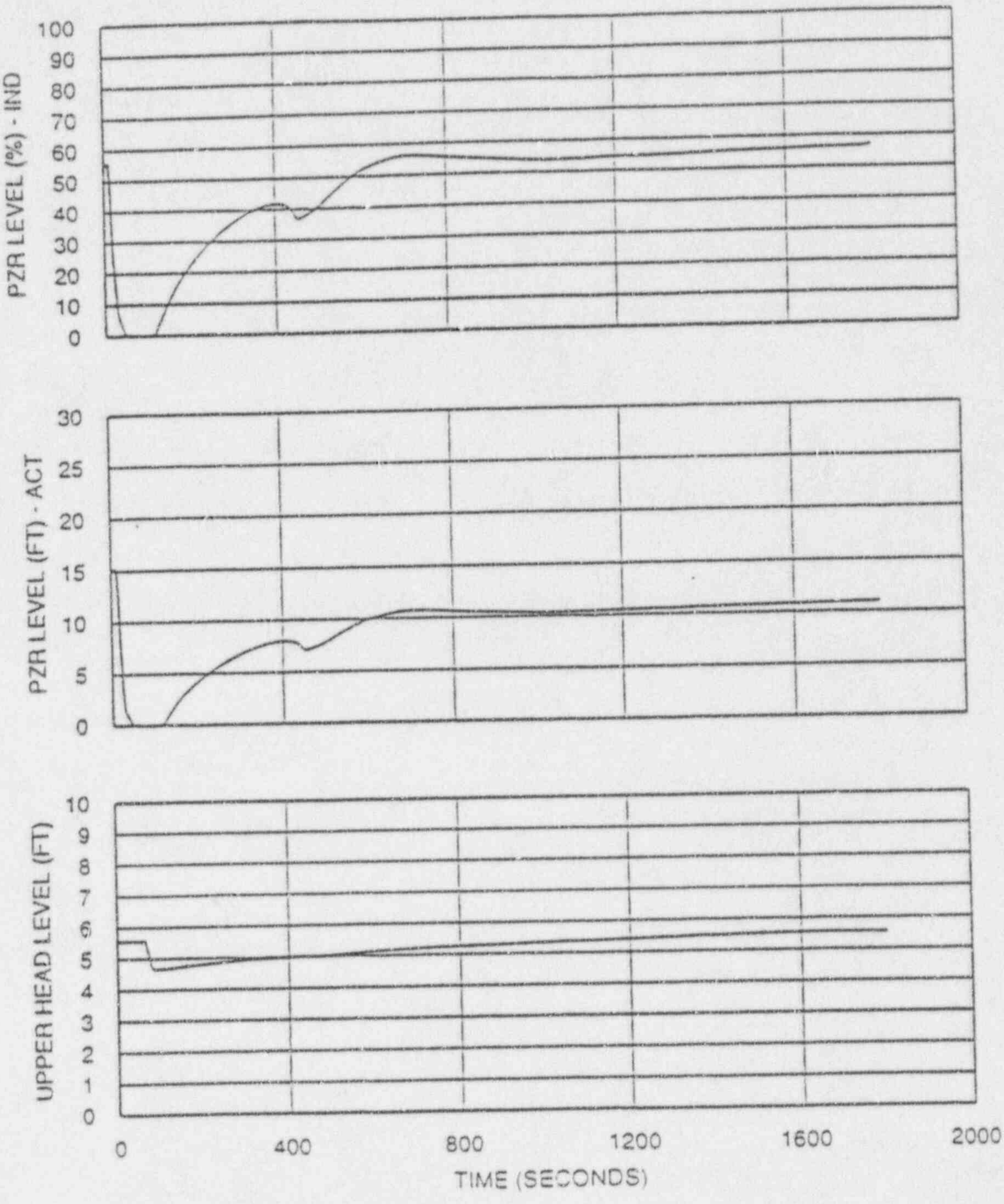


MYC-[illegible] REV [illegible]
MSLB [illegible] SGTR ANALYSIS

MY MSLB WITHOUT SGTR - mym500
RCPS TRIPPED AT 30 SECONDS
[illegible] ACTIONS: N/A

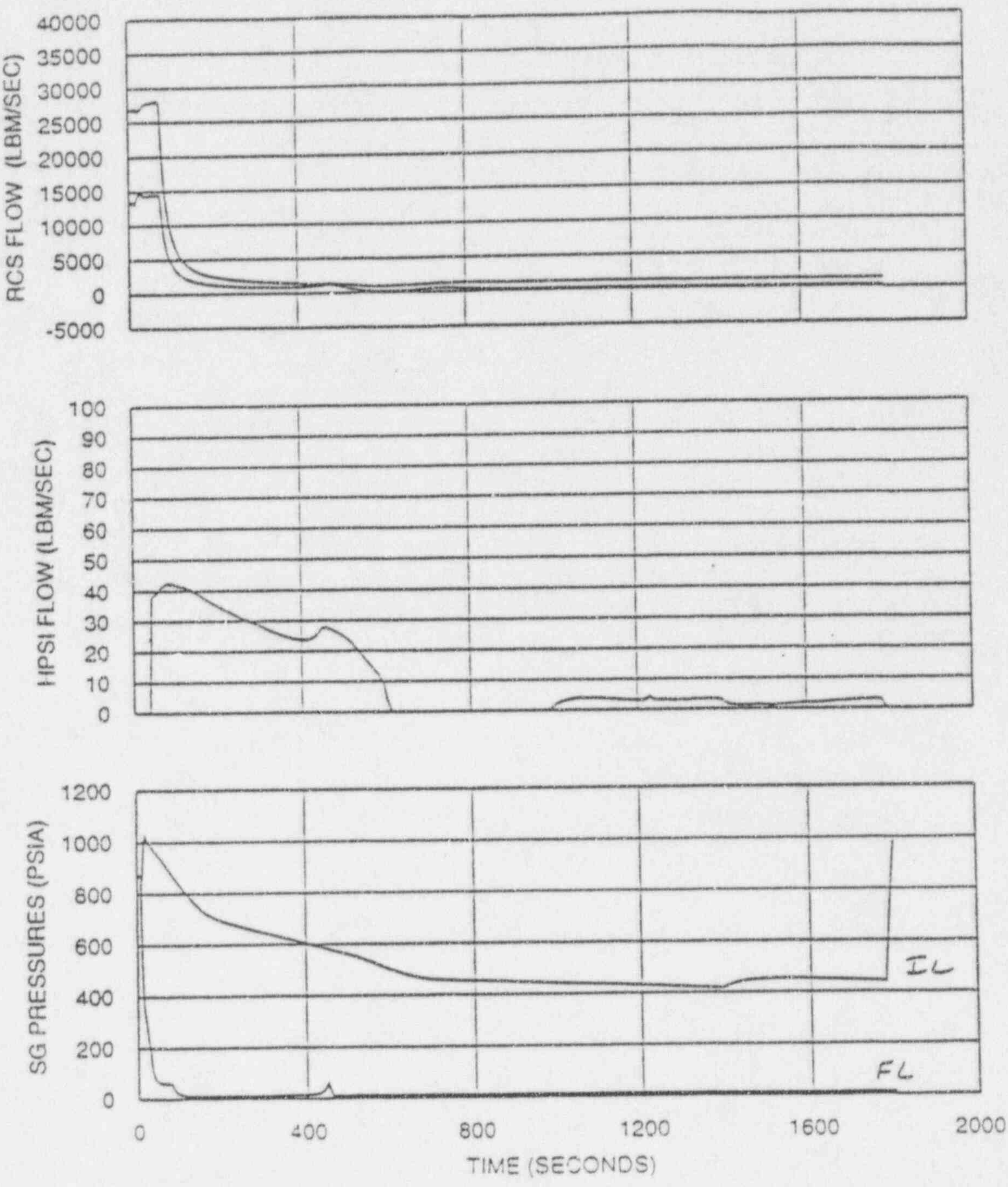


MYC-1592 REV 0 MSLR BASE SGTR ANALYSIS	MY MSLR WITHOUT SGTR - mym500 RCSS TRIPPED AT [illegible] SECONDS SOP ACTIONS: N/A



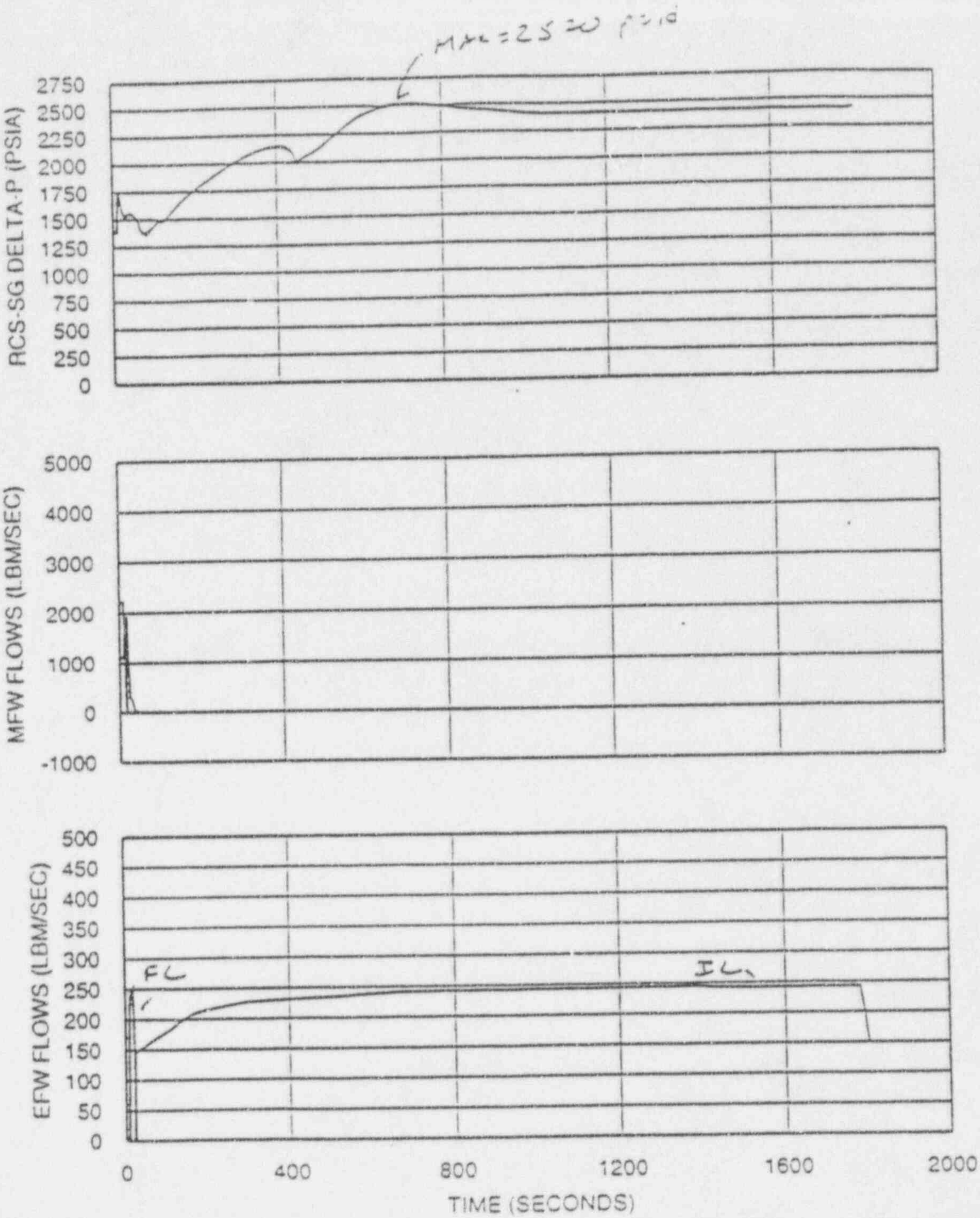
MYC-1592 REV 0
MSLR BASE SGTR ANALYSIS

MY MSLR WITHOUT SGTR - mvm500
RCSS TRIPPED AT 30 SECONDS
EOP ACTIONS - N/A



[illegible]
[illegible]

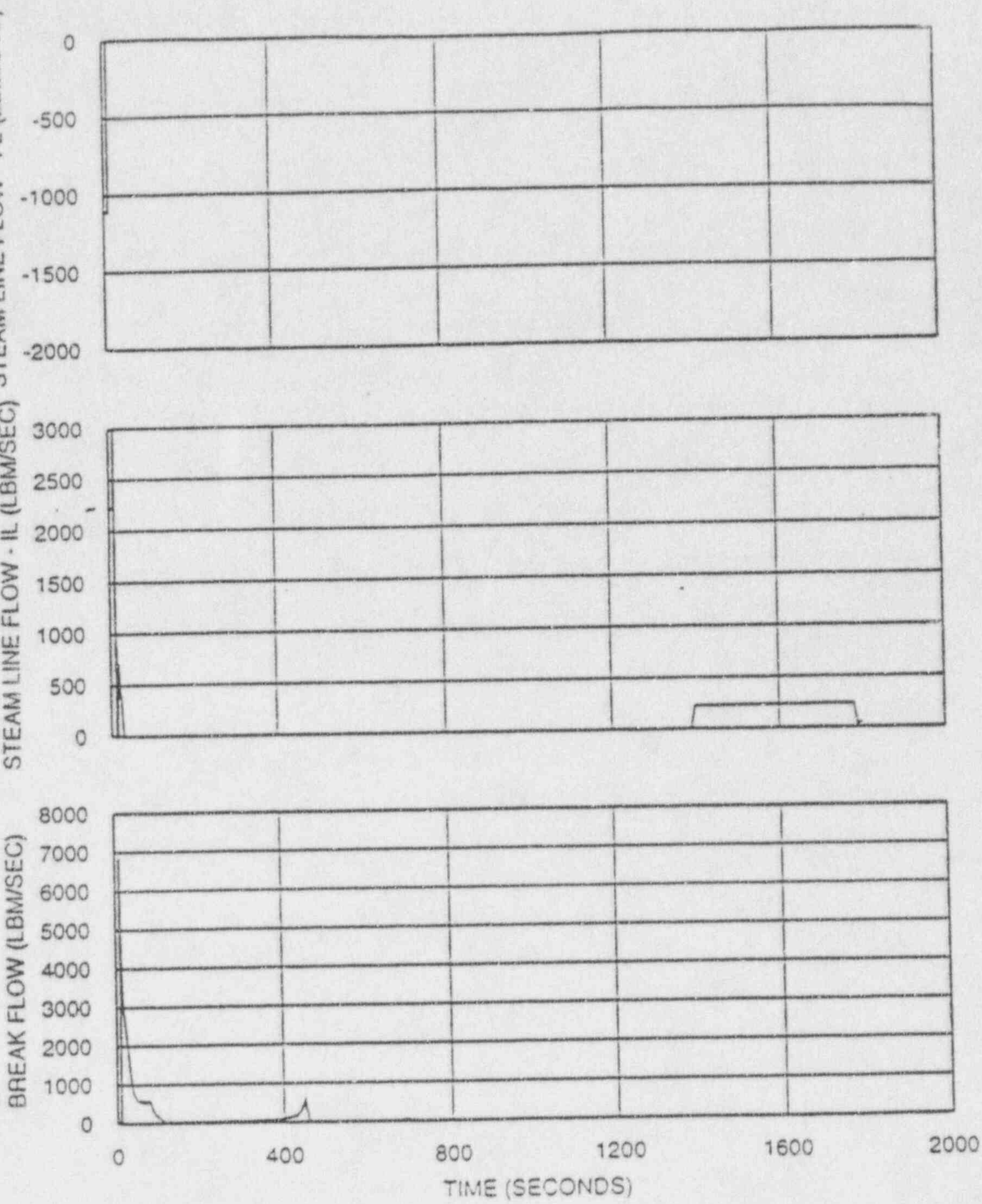
MY MSLB WITHOUT SGTR - [illegible]
RCSS TRIPPED AT 30 SECONDS
EOP ACTIONS [illegible]



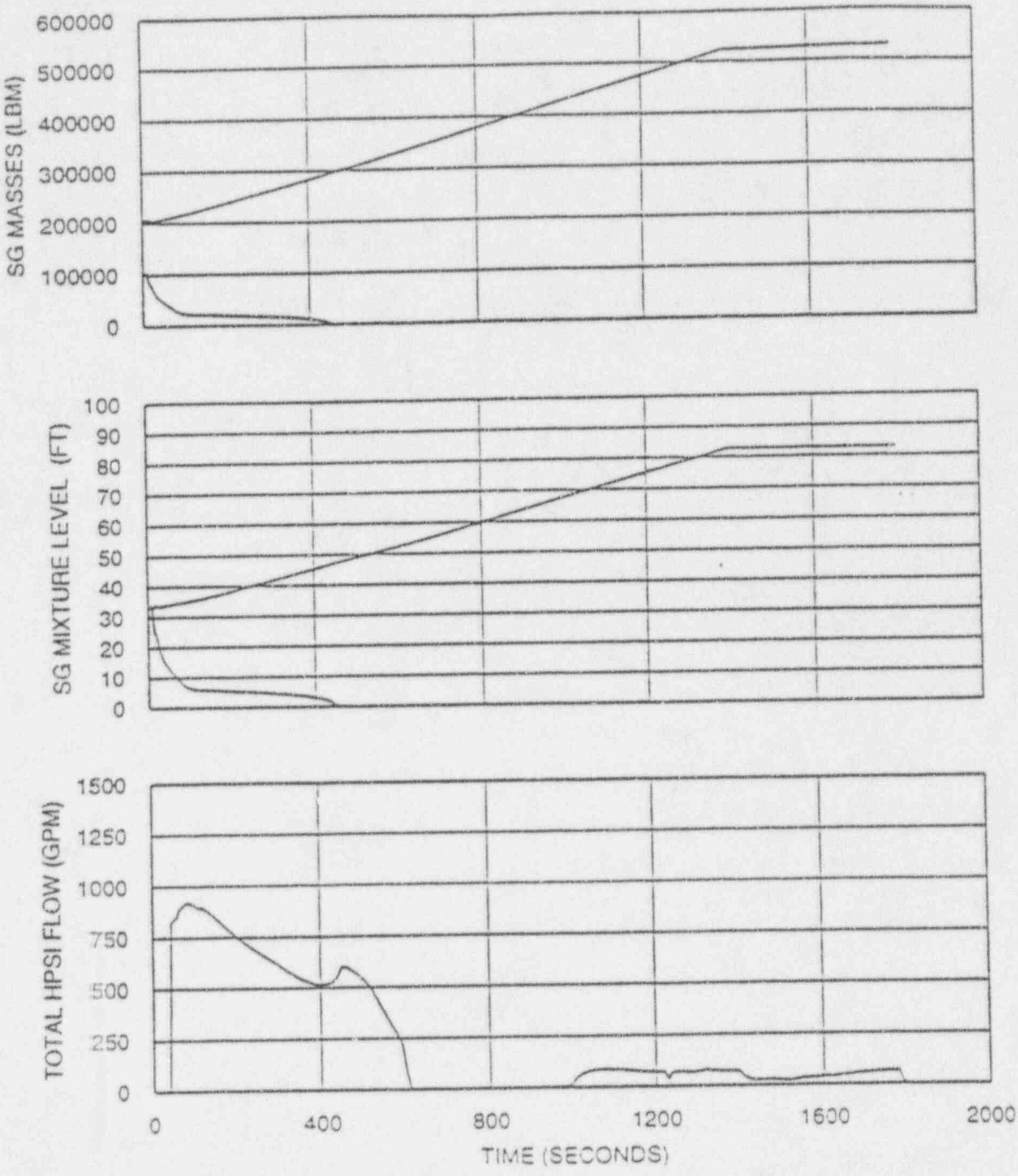
MYC-1592 REV 0
MSLR BASE [illegible]

MY MSLR WITHOUT SGTR - [illegible]
RCSS TRIPPED AT [illegible]
EOP ACTIONS - N/A

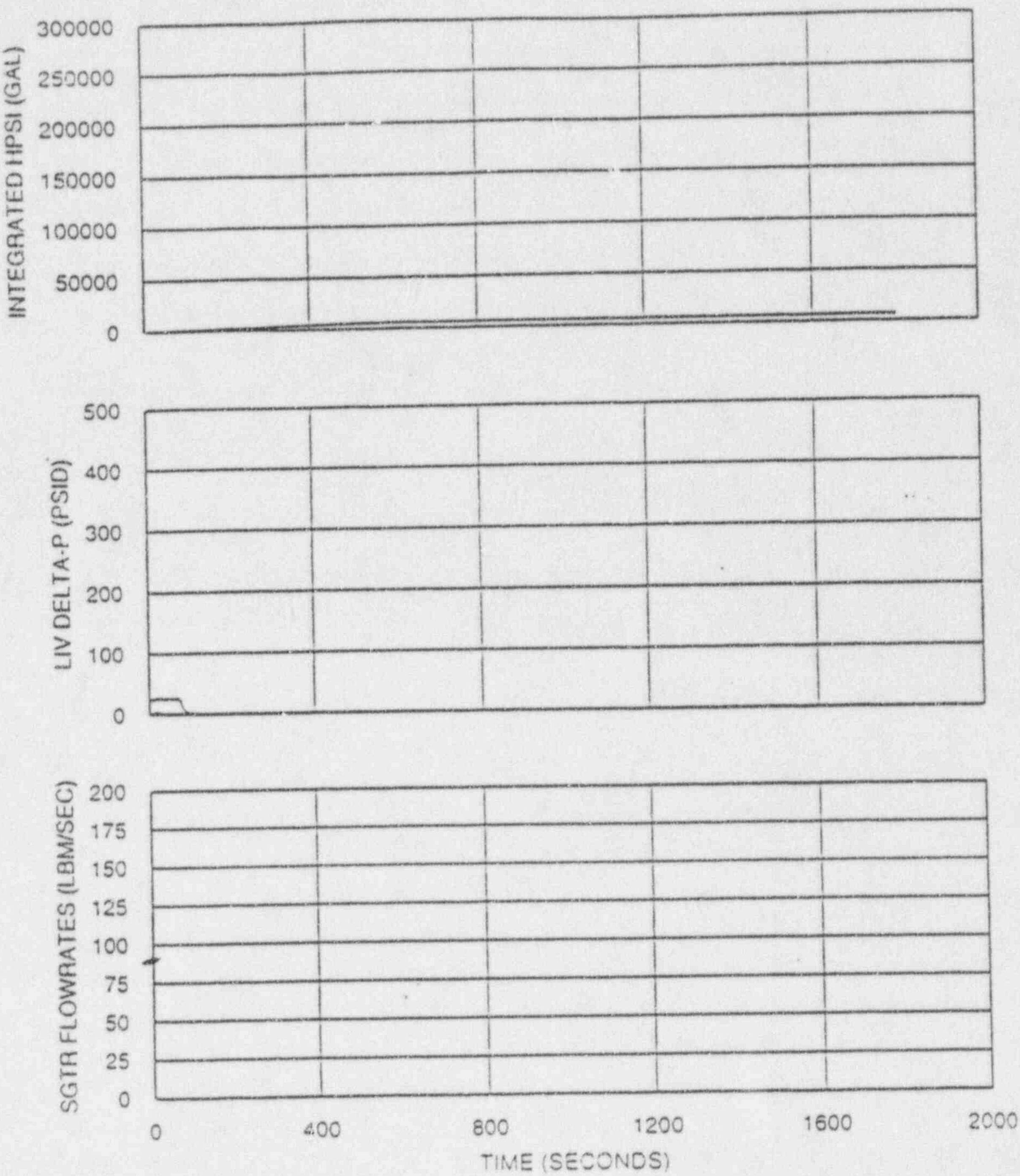
[illegible]



MYC-1592 REV 0 [illegible] SGTR ANALYSIS	MY MSLB WITHOUT SGTR - mvm500 RCSS TRIPPED AT 30 SECONDS EOP ACTIONS - N/A

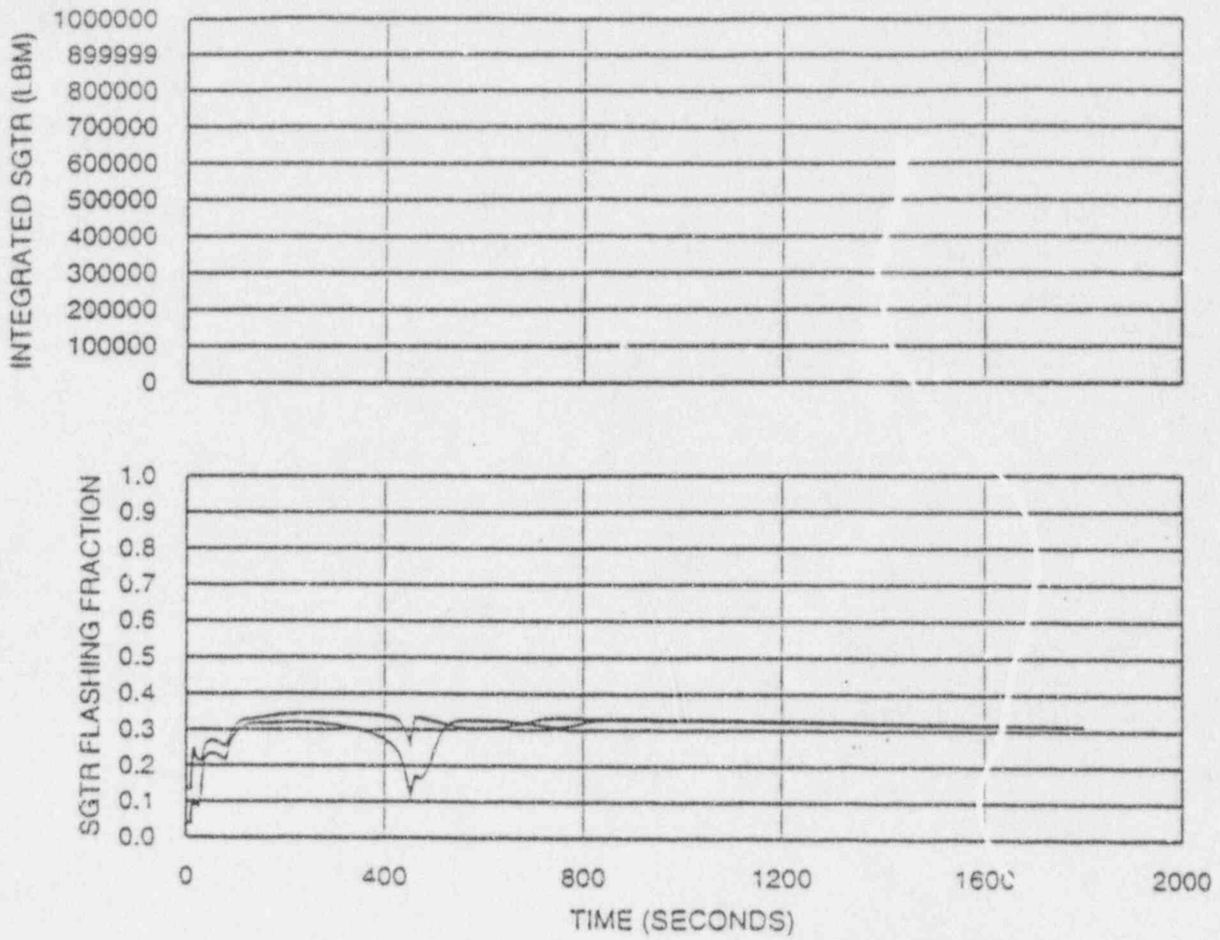


MYC-153[illegible] REV [illegible] MSLR [illegible]	MY MSLR WITHOUT SGTR [illegible] RCSS TRIPPED AT [illegible] [illegible] ACTIONS N/A



MYC-[illegible] REV 0
[illegible] ANALYSIS

MY MSLB WITHOUT SGTR - [illegible]
RCPS TRIPPED AT 30 SECONDS
[illegible] ACTION [illegible]


INTEGRATED SGTR (LBM)
1000000
899999
800000
700000
600000
500000
400000
300000
200000
100000
0
SGTR FLASHING FRACTION
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
0
400
800
1200
1600
2000
TIME (SECONDS)

ENCLOSURE 8

MAINE YANKEE NOVEMBER 14, 1994 LETTER ENCLOSURE'S REFERENCE 12

ABB C-E Letter WO94243.RM dated 11/10/94 to C. Eames of Maine Yankee; Subject: "Transmittal of Documents: MRPC Sizing of Circumferential Cracks"

ABB

Mr. Charles Eames
Corporate Engineering Department
Maine Yankee Atomic Power Company
Bath, Maine 04578

November 10, 1994
WO94243.RM

Subject: Transmittal of Document: MRPC Sizing of Circumferential Cracks

Dear Mr. Eames:
Please find enclosed the subject document which describes the Eddy Current Test technique for estimation of average cross-sectional wall loss. Should you desire additional information or require clarification on the application of this technique, please contact me at (203) 285-3055.

Very Truly Yours,
Rick Maurer
R. S. Maurer
ABB Combustion Engineering

Combustion Engineering Inc
P.O. Box [illegible]
1000 Prospect Hill Road
Windsor, CT 06095-0500
Telephone (203) 285-2941
Fax (203) 285-[illegible]

MRPC SIZING OF CIRCUMFERENTIAL CRACKS

The motorized rotating pancake coil (MRPC) has been used as the primary means of detection for circumferential cracks in C-E steam generators since 1989. Although not used as a bases for repair (all are repaired), measurements of length, depth, and voltage are recorded from which an assessment of remaining structural margins can be estimated. The following is a brief description of how these measurements are made:

Maximum Depth:
Estimates of maximum crack depth are made by phase analysis of the prime frequency (400Khz). A three point curve is generated based on the known depths of the ASME calibration standard. A software algorithm interpolates between these points to allow sizing of indications in the steam generator tube. Figure 1 shows the MRPC response to the ASME standard. It is significant to note that the resolution for inner diameter flaws is relatively poor, compared to outer diameter flaws.

Voltage measurements are an indirect reflection of volume loss. The voltage setpoint is the 60% OD flaw in the ASME standard set at 5.0 volts. Figure 2 shows this setpoint.

Length:
Estimates of crack length are made by isolating the vertical component of the flaw signal and comparing the time interval that the indication is sensed relative to a circumferentially staged series of trigger pulses input from the probe motor. Figures 3 through 5 illustrate ECT measurements of a circumferential crack.

Average Depth
Average cross-sectional wall loss is derived from depth and length values described above. AVG % = Depth(max %) x Length(degrees) / 360
This method is conservative in that the entire arc length is assumed to be uniformly deep(tubes removed from C-E steam generators have exhibited varying depths around the circumference). The method is non conservative in that the unseen area is assumed to be non flawed(removed tubes show that crack depths of 40% and less can be transparent to the MRPC). Scatter diagrams which plot average depth estimates against metallographic results for tubes removed from C-E units is presented in figure 6. Additional data from over 40 circumferentially cracked lab samples is also exhibits reasonable correlation between measured and actual values (see figure 7).

Figure 1

ASME Standard Response

ASME STANDARD SHOWING PHASE RESOLUTION FOR ID AND OD FLAWS

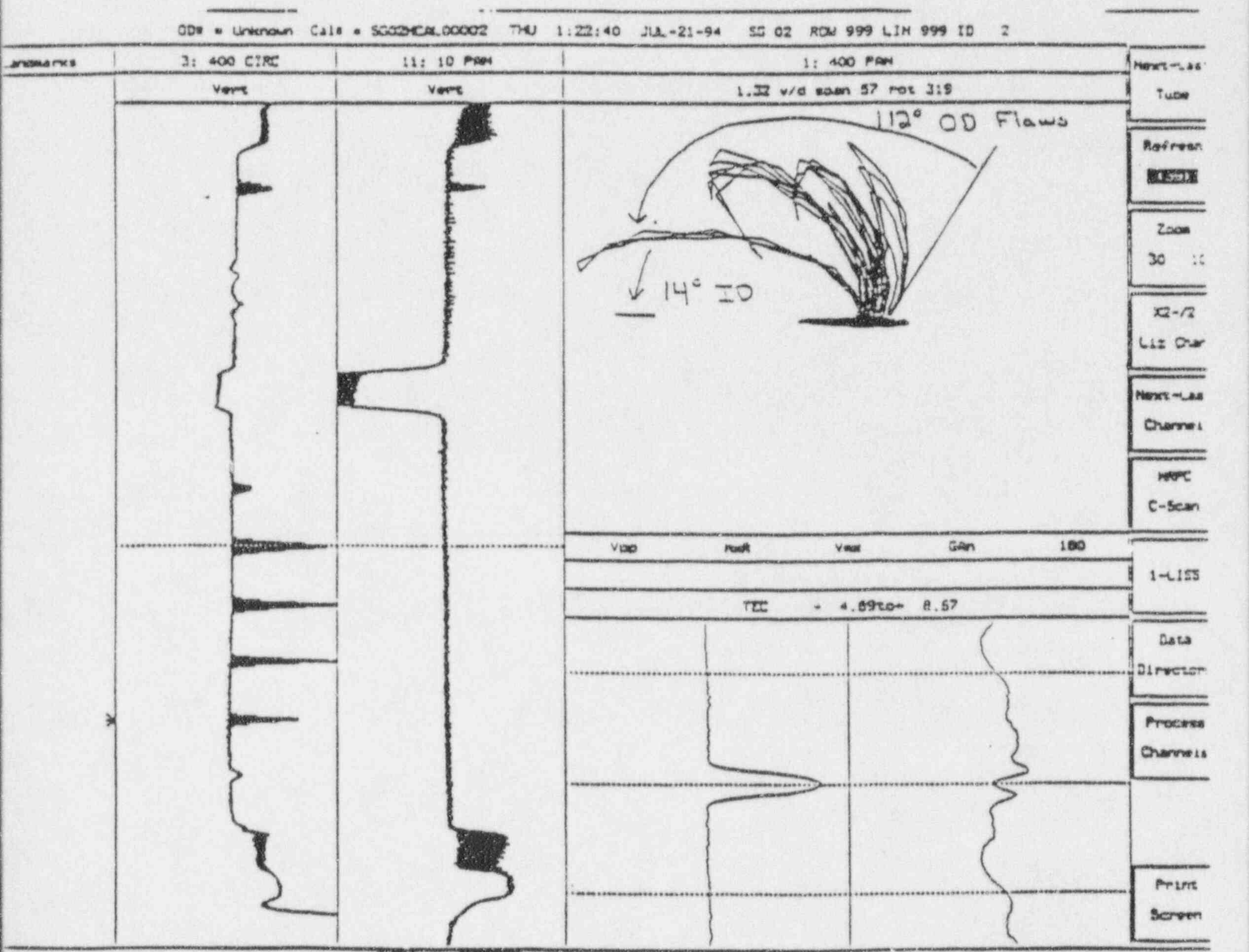


Figure 2

Voltage Setpoint

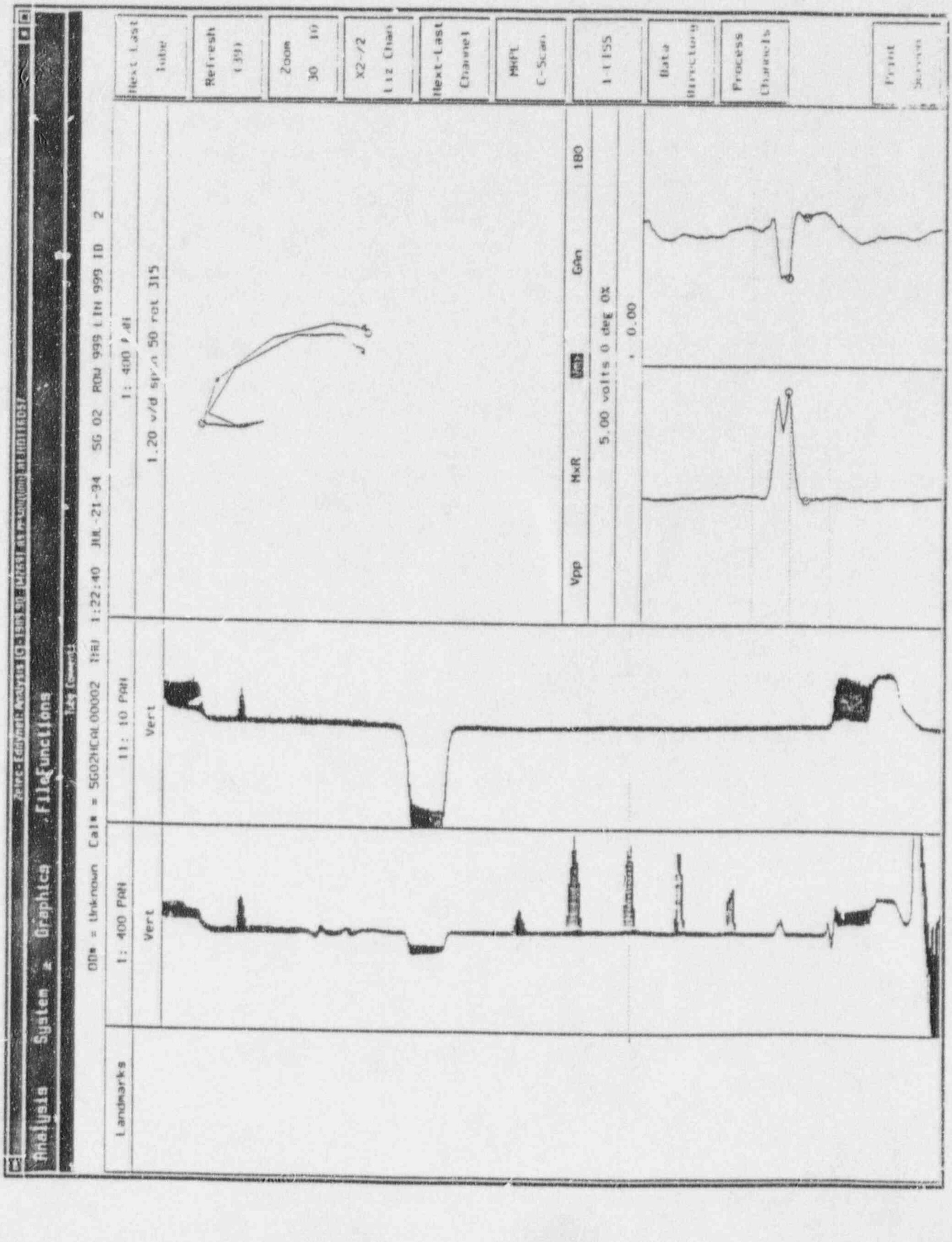


Figure 3

Maximum Depth Measurement

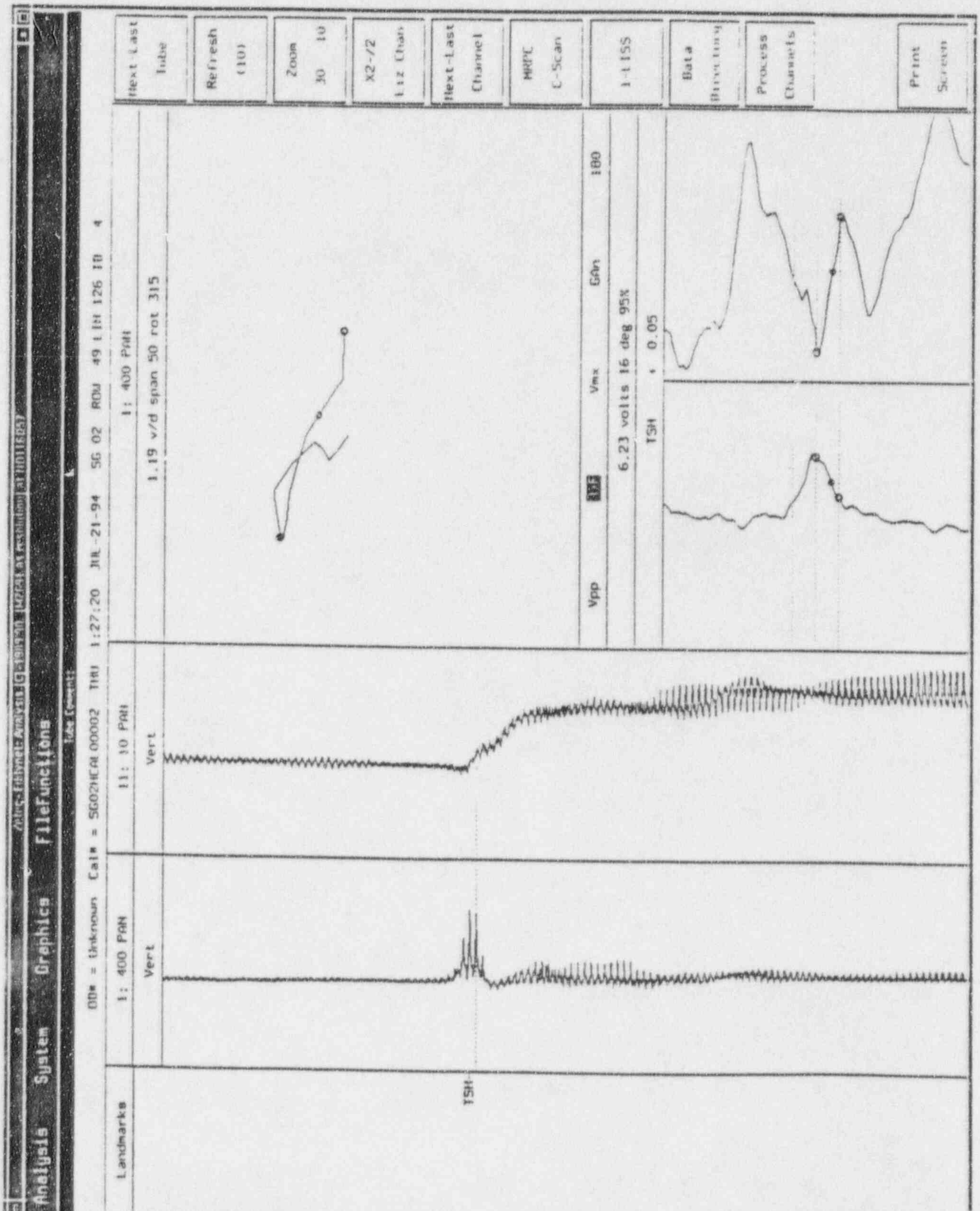


Figure 4
Isolation of Flaw Component

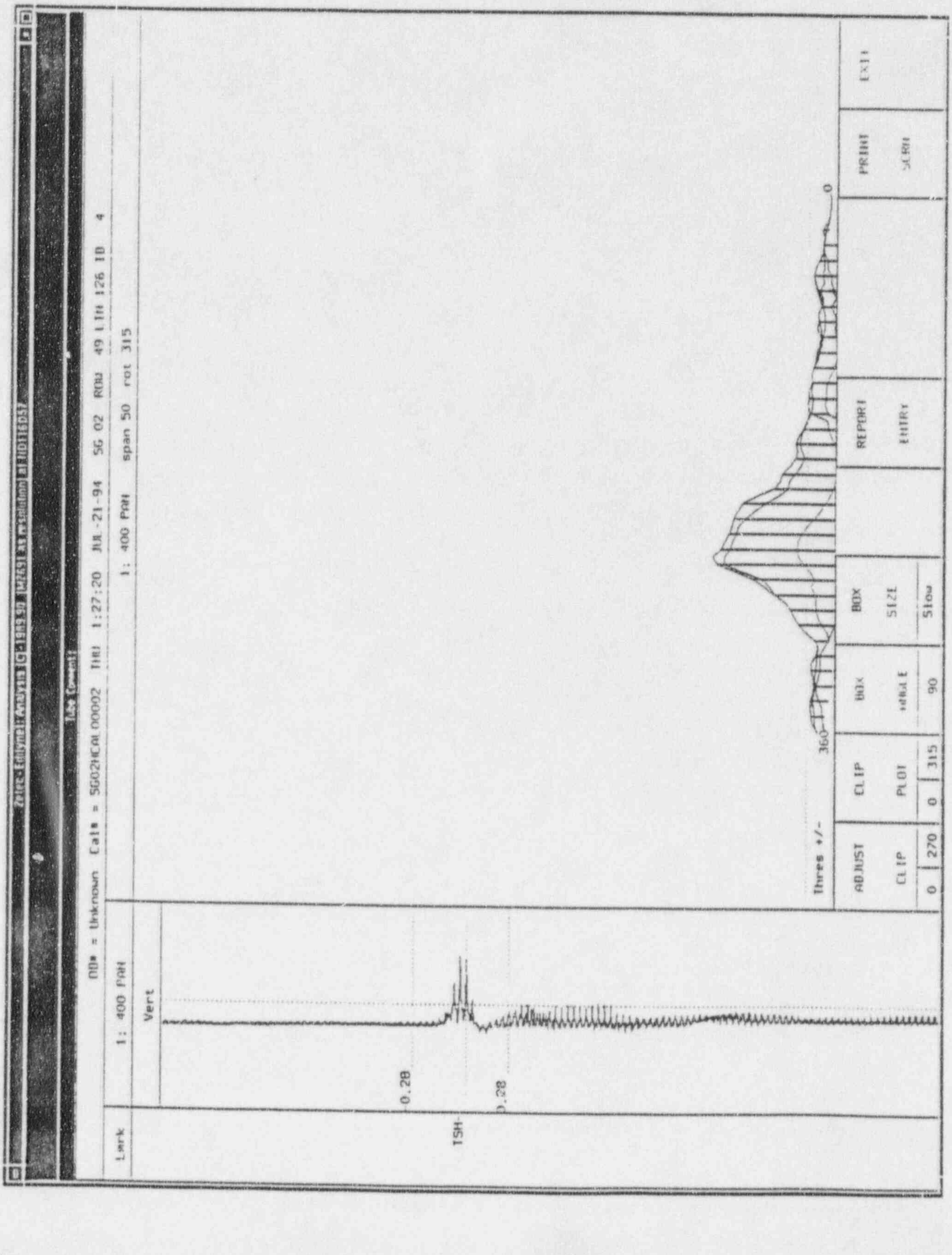
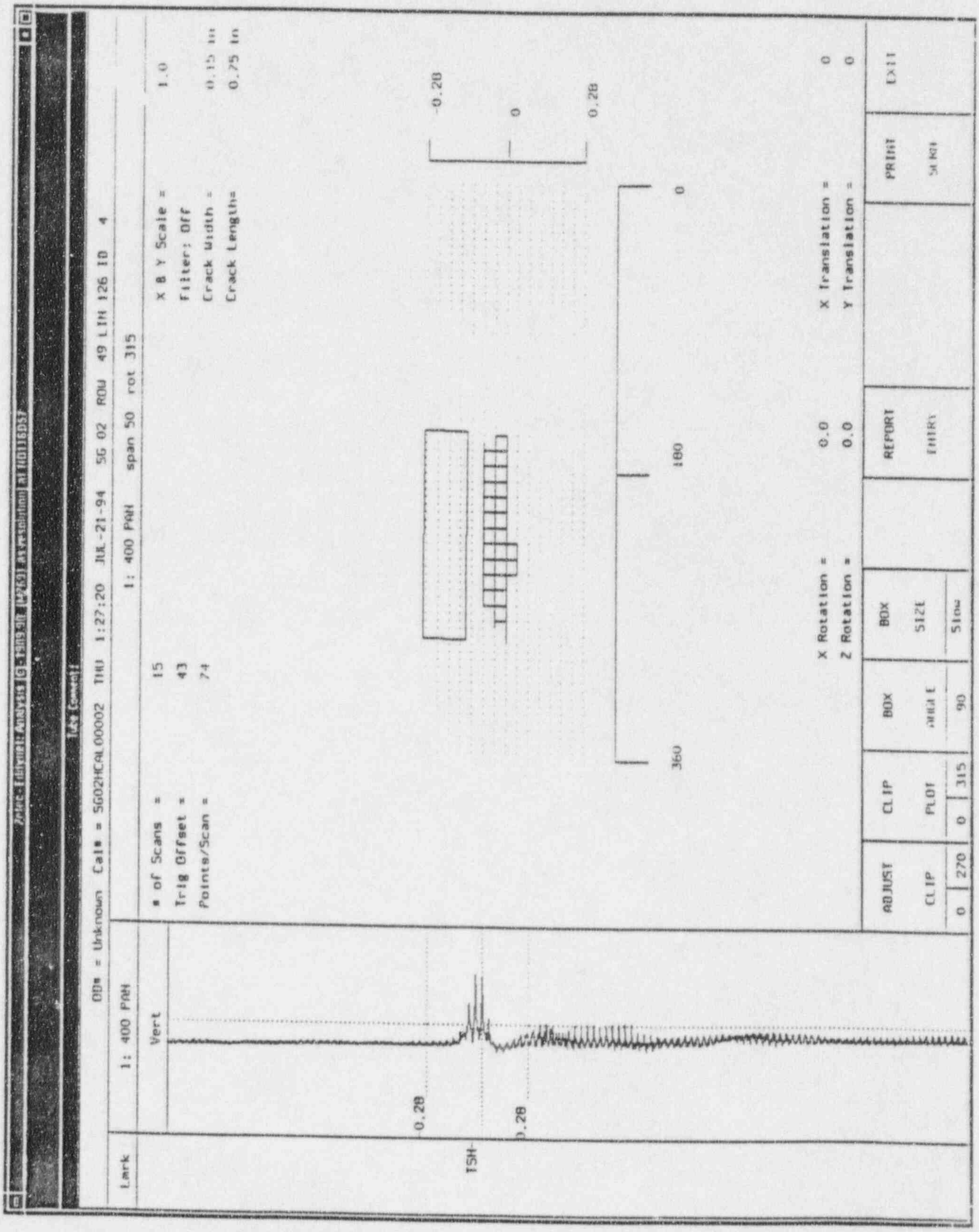


Figure 5

Flaw length measurement



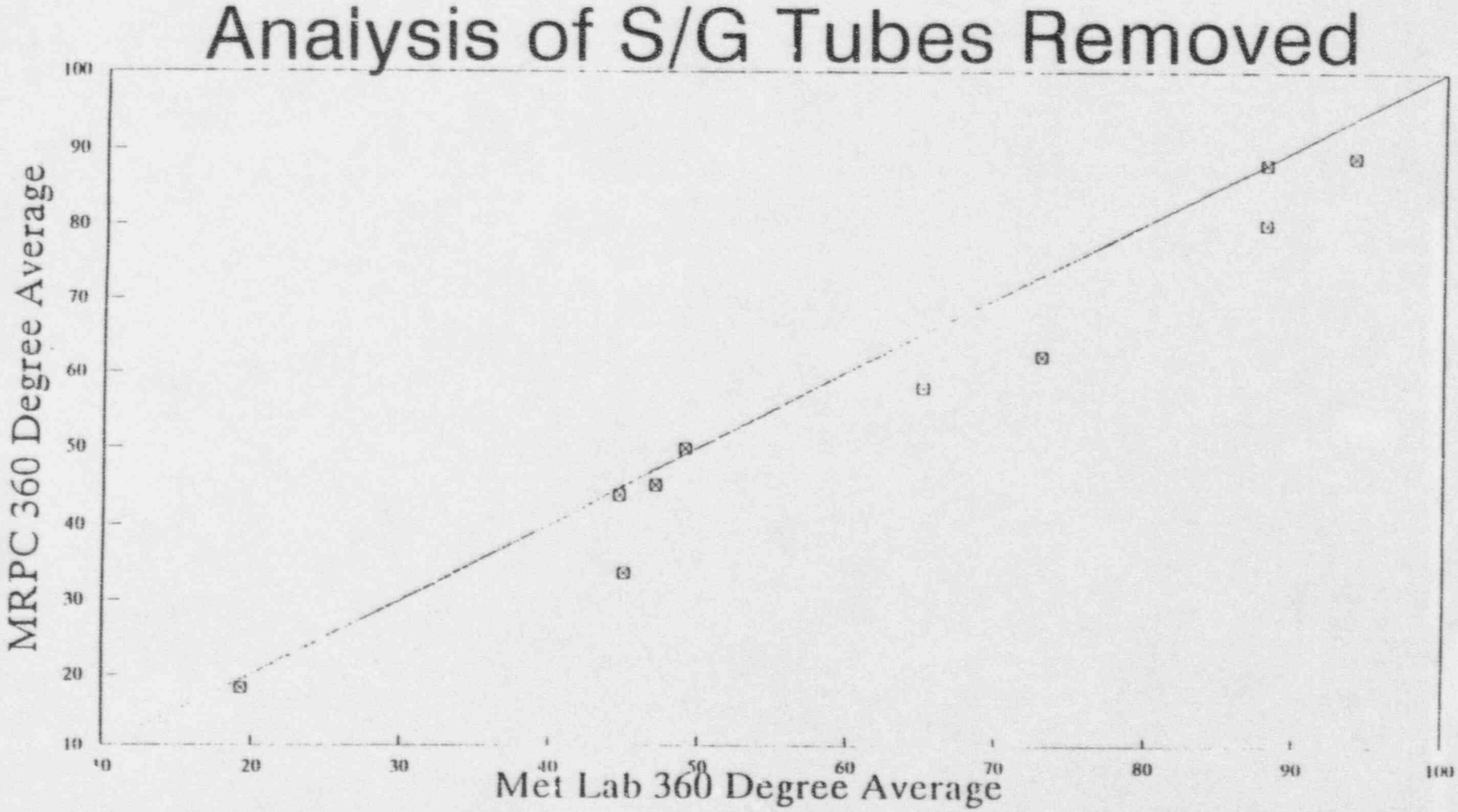
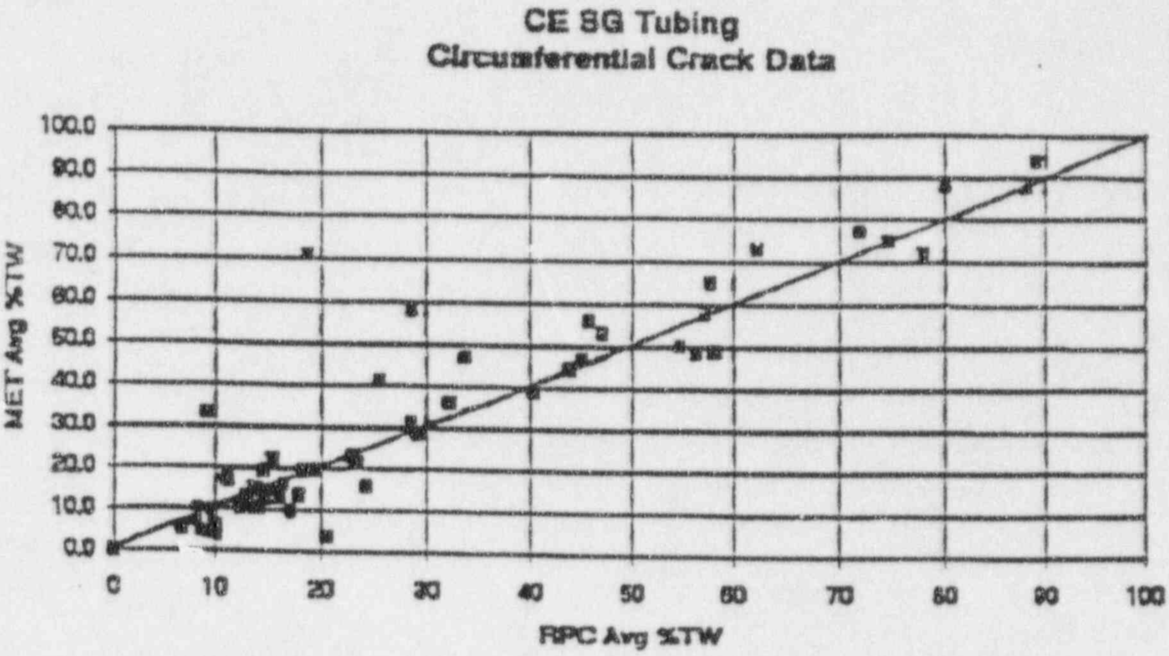


Figure 6
ECT VS Metallography
For Pulled Tubes

Figure 7

ECT VS Metallography

Removed Tubes and Lab Crack Samples



ENCLOSURE 9

MAINE YANKEE NOVEMBER 14, 1994 LETTER ENCLOSURE'S REFERENCE 13

Proceedings: 1992 EPRI Workshop on PWSCC of Alloy 600 in PWRs. EPRI TR-103345 dated December 1993 Paper C4 "Potential Benefits of Zinc Addition to PWR Coolant" by R. E. Gold

Potential Benefits of Zinc Addition to PWR Coolant

- *R. E. Gold, Westinghouse*

•

EPRI Workshop -
PWSCC of Alloy 600 in PWRs

Orlando, FL
Dec. 1-3, 1992

Background

- Westinghouse Evaluated the Potential Benefits of Zn Addition to the Primary Coolant - 1988-89
- Clear Benefits were Observed:
  - The Initiation of PWSCC of Alloy 600 was Inhibited
  - General Corrosion Rates/Metal Release Rates of Alloy 600 and Other Primary System Materials were Reduced (These Benefits are Consistent with Experience in BWR Environments)
- Provided Incentive for Further Evaluations under Aegis of the WOG-Materials Subcommittee - 1990-92

Solubility Studies

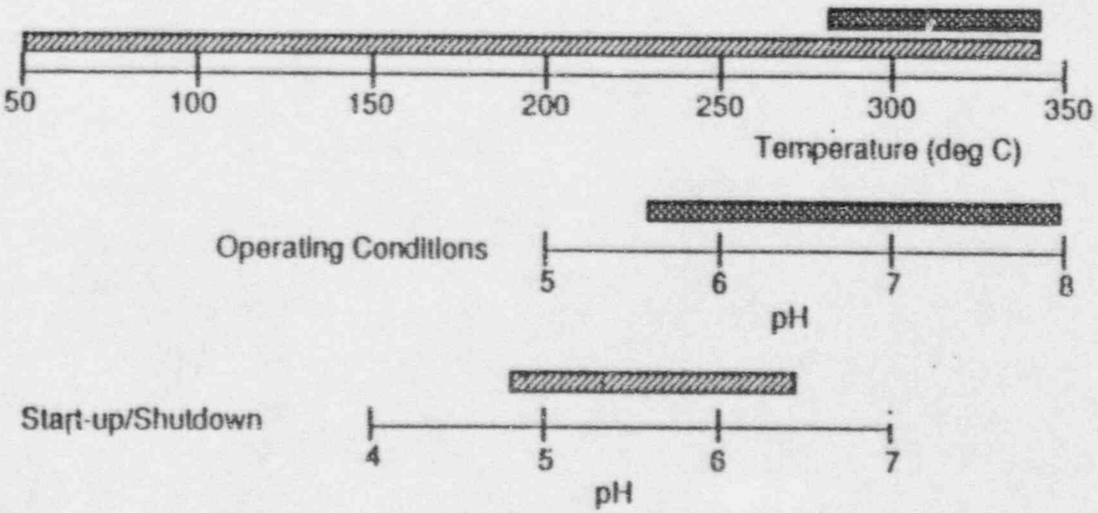
Purpose:

Develop sufficient understanding of the solubility characteristics of Zn in primary water to be able to project the performance characteristics in the various temperature and pH ranges expected in the primary loop.

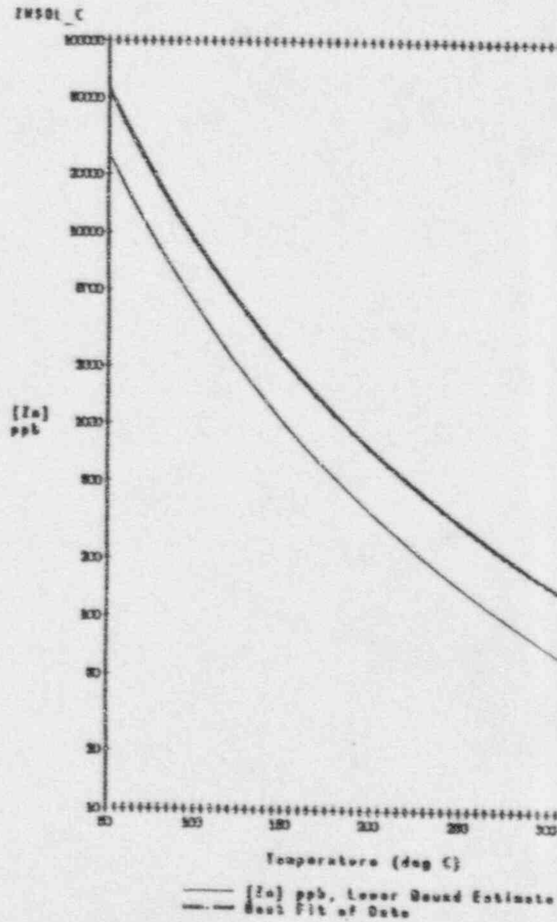
Techniques:

Obtain relevant BWR and literature data.

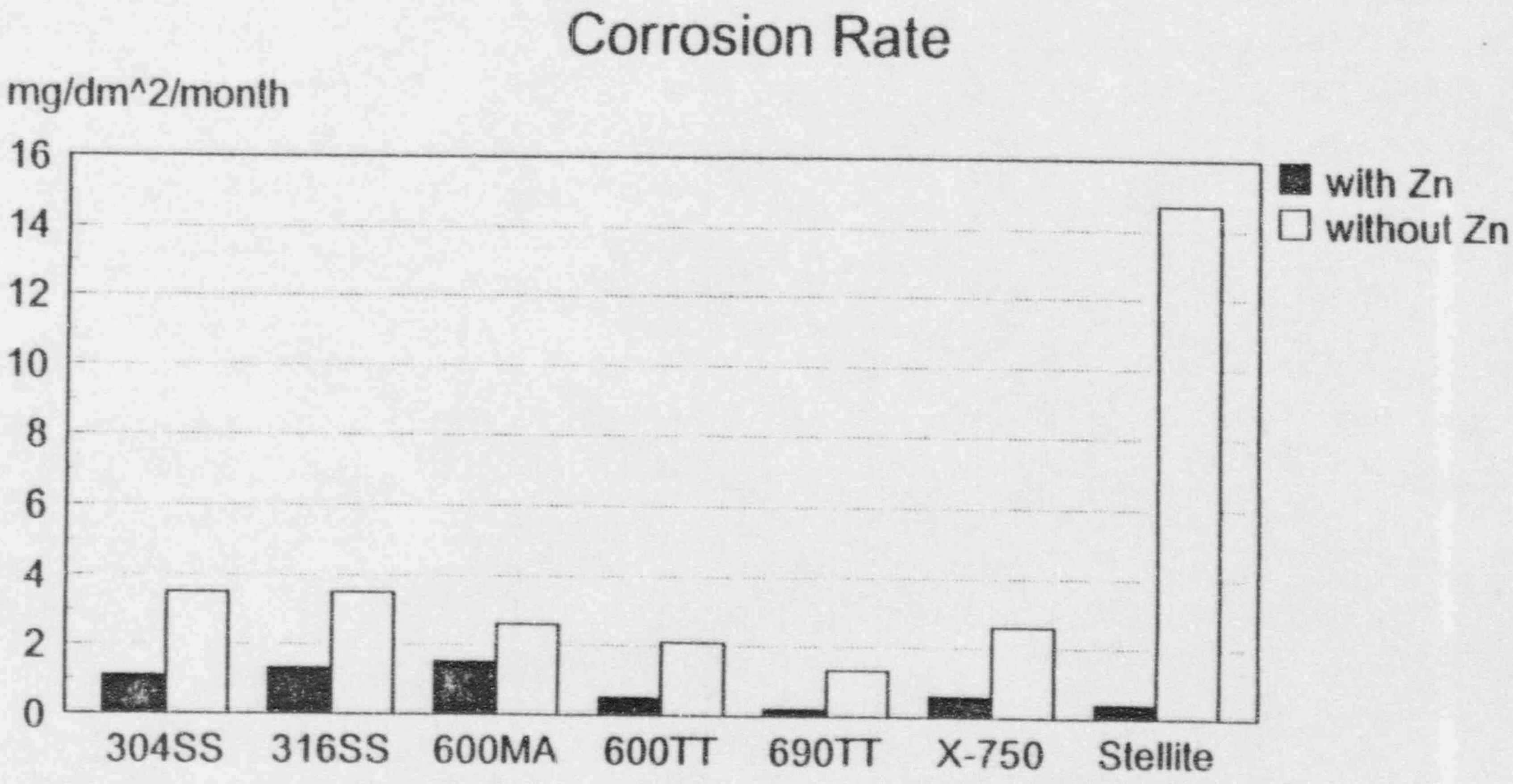
Make solubility measurements in autoclaves.



Limits of Application



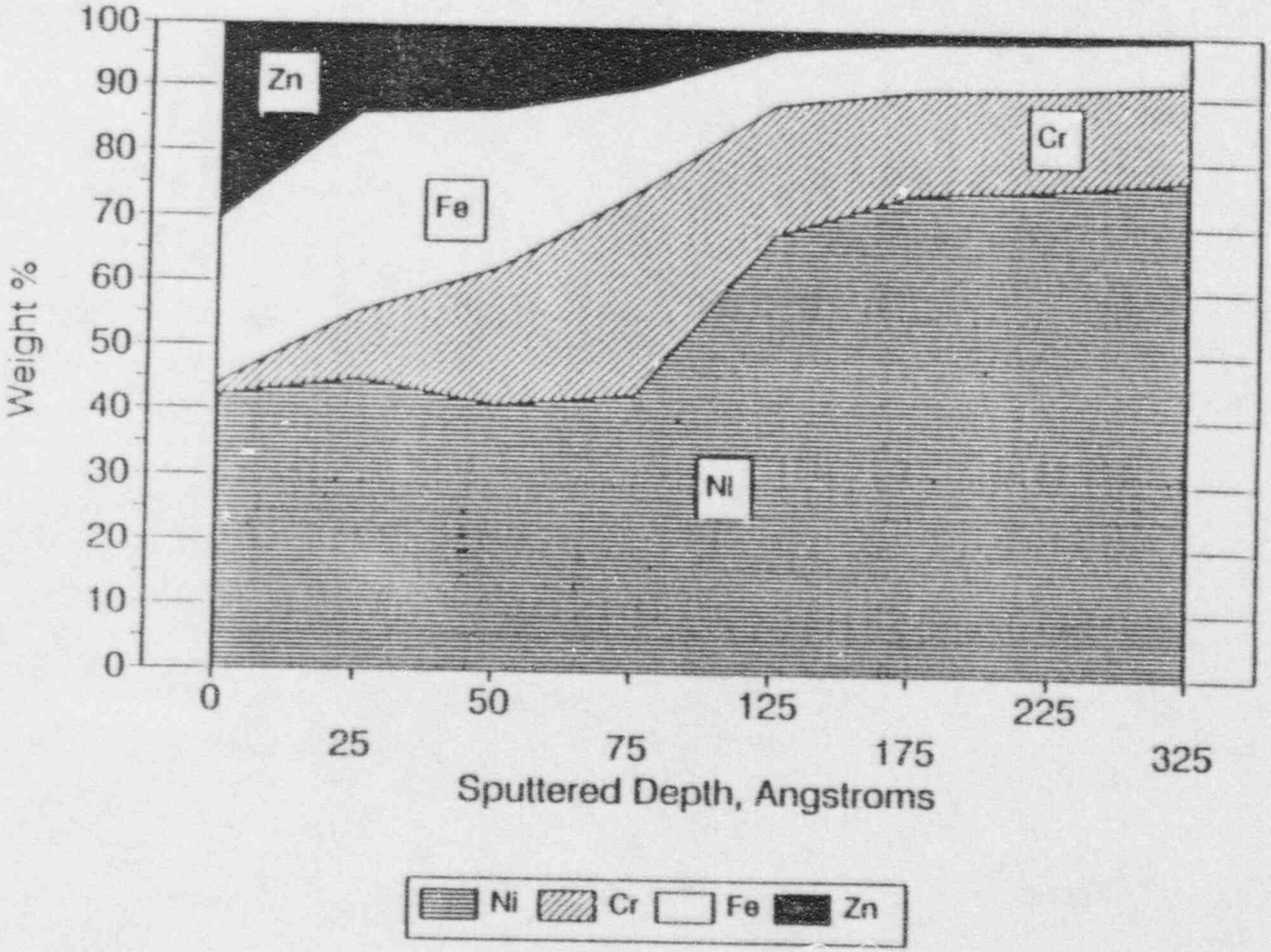
The Corrosion Rate of Primary Side Materials is Reduced by Zn Additions [2500 Hrs. at 330 Deg. C in BOL PWR Coolant Containing 20 ppb Zinc]



Zinc is Readily Incorporated into the Corrosion Films on Stainless Steels and Ni-Base Alloys

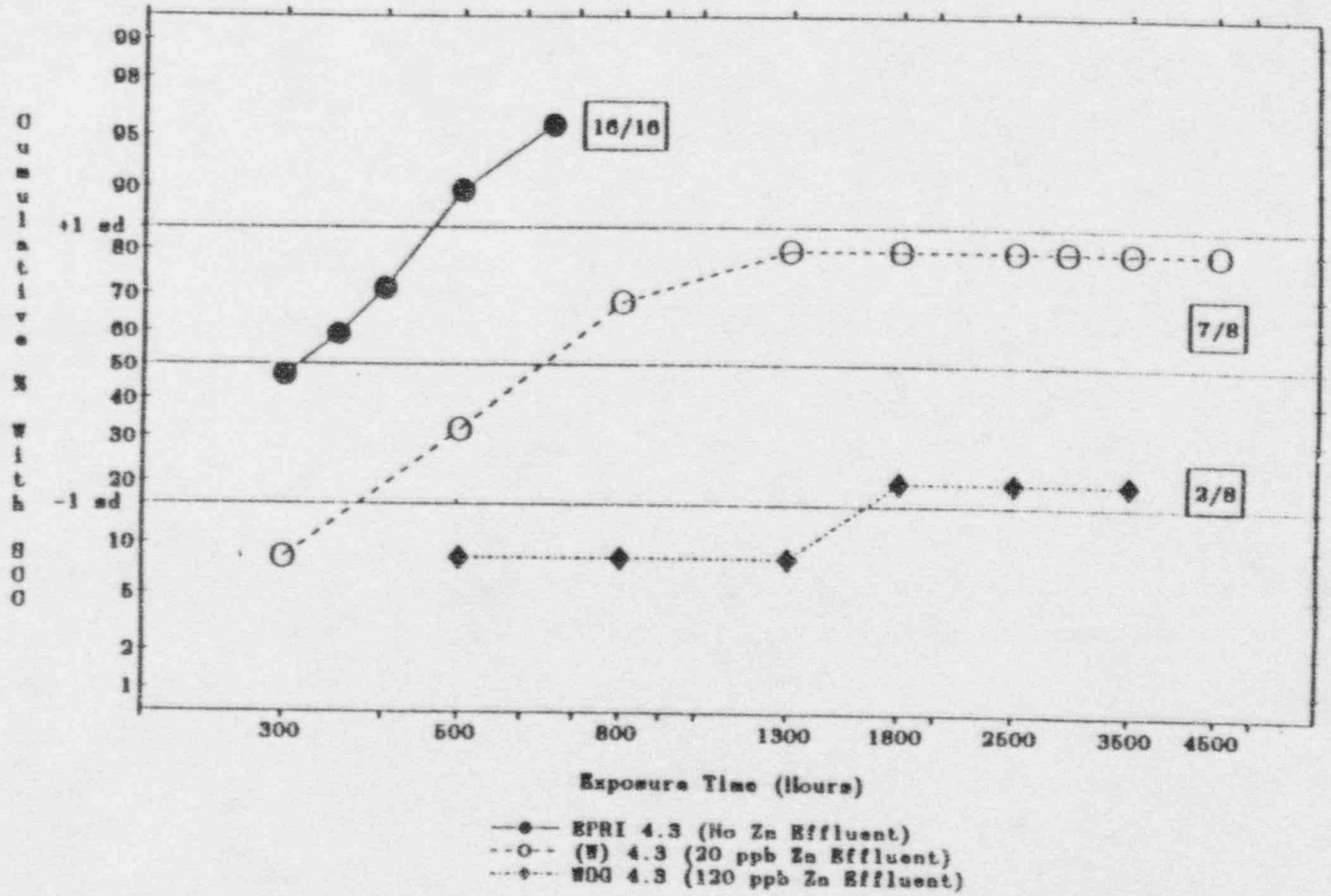
- Zinc Displaces Cobalt and Other Divalent Ions in Spinel (Chromite) and Inverse Spinel (Ferrite) Structures
  - The Preferential Binding Energy of Divalent Zn for Tetrahedral Sites in Spinels is Greater than that of Other Divalent Ions (such as Cobalt)
  - Zn Concentrations in the Coolant may be in the ppb Range; Co is usually Present in the ppt Range
- Oxide Layers are Much Thinner in the Presence of Zinc
  - Auger and ESCA Analyses Indicate Film Thicknesses on A600 are about 15 nm w/ Zn and about 75-80 nm w/o Zn
  - Values for 304SS are 40-55 nm w/Zn and Approx. Ten Times that Value w/o Zn

Results of Auger Analysis - Alloy 600 MA Oxide Film (Metals only) after 2500 Hrs. in BOL PWR Coolant Containing 20 ppb Zinc



eprl-fl dec92

Zn Provides Significant Benefit in Inhibiting PWSCC Initiation in Highly Stressed Alloy 600 MA RUBs - Heat 1019 at 330 Deg. C

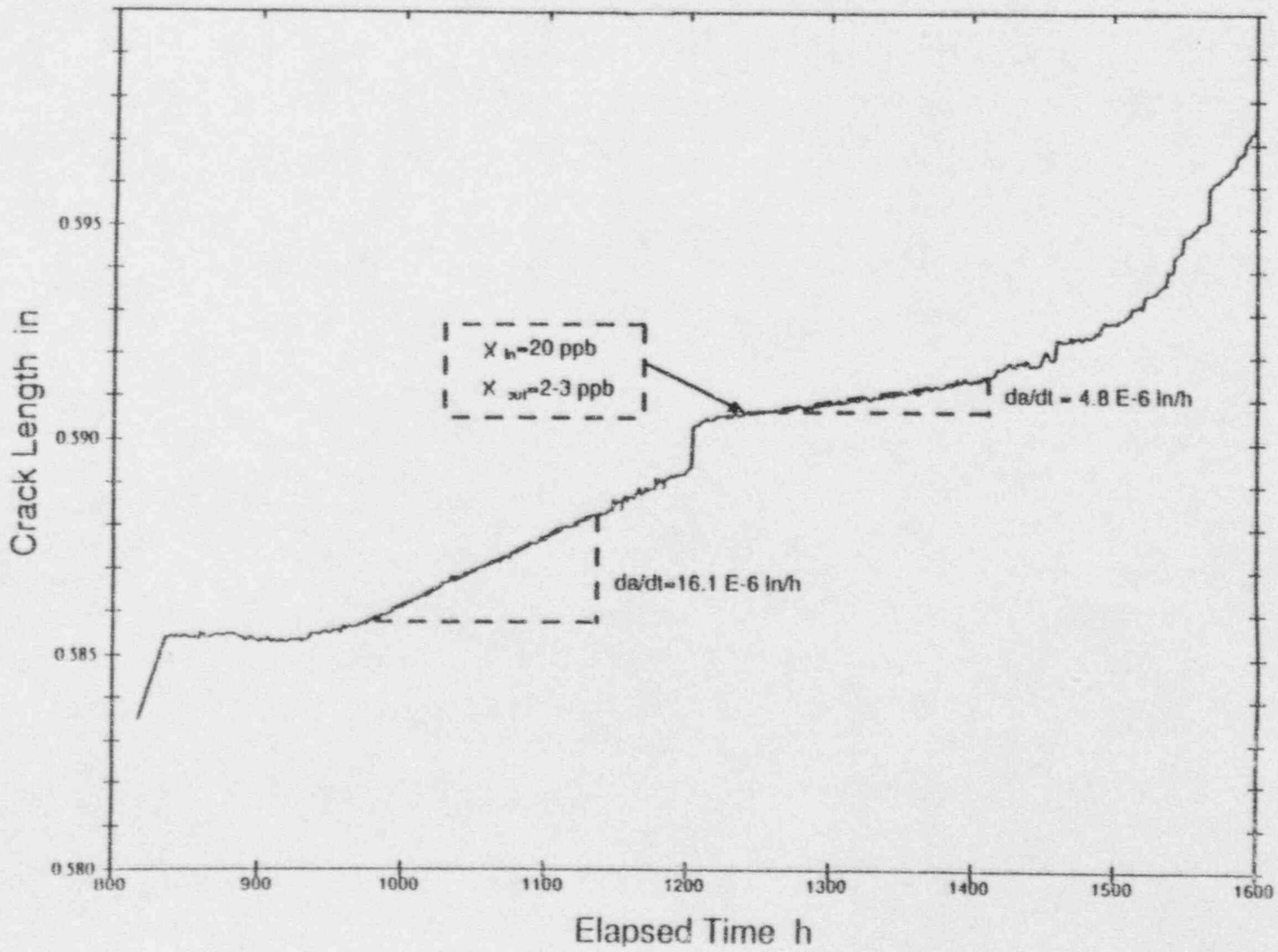


epri-fi dec92

Effects of Zinc on Crack Propagation in Alloy 600

- Westinghouse Evaluated Crack Propagation in Alloy 600 MA Tubing - The Reversed DC Potential Drop Technique was Used
  - Tests were Performed at 330 Deg. C in BOL PWR Coolant
  - For Stress Intensities < 40-50 MPa.m^1/2, 20 ppb Zn Reduced the Propagation Rate by a Factor of 3.3
- Andresen (GE) Has Performed Extensive Propagation Rate Tests with a Similar Reversed DC Potential Drop Technique for 304SS and Sensitized Alloy 600 under BWR Conditions (1992 BNES Paper)
  - For Alloy 600 CT Specimens [288 Deg. C @ 33 MPa.m^1/2], 10 ppb Zn Lowered the Propagation Rates by Factors of 2 - 3.3
  - Propagation Rates in 304 SS were Reduced by a Factor of 5

Zinc Additions Effect a Substantial Decrease in Crack Propagation in Alloy 600 Tubing [Tests at 330 Deg. C in BOL PWR Coolant]



epri-fl dec92

Mitigation of PWSCC of Primary System Materials Offers the Potential for Significant Savings

- In the 20-120 ppb Zn Range, PWSCC Initiation Times in Susceptible Alloy 600 RUBs are Reduced by Factors of 2.8 to > 10
- Reduced PWSCC of SG Tubing Translates into Savings in Direct Costs (Inspection, Plugging, Sleeving) and Indirect Costs (Outage Duration, Equip. Mobilization, Man-Rem)
- The Kinetics of PWSCC of Alloy 600 CRDM Penetrations Appear Similar to Those of SG Tubing (Operating Times and Temperatures); Similar Benefits Would be Expected
- Specific Cost Benefits Would Vary According to Plant Characteristics and Extent of Degradation

ENCLOSURE 10

MAINE YANKEE NOVEMBER 14, 1994 LETTER ENCLOSURE'S REFERENCE 14

ABB C-E Report CE NPSD-957 (CEOG Task 729) "S/G Tube Degradation at the Support Plates" by G. C. Fink and S. M. Schloss, October 1994

Room and operating temperature pressure burst tests of S/G tubes representive of those in C-E steam generators are described in ABB C-E Report CE NPSD-957, a C-E Owner's Group funded and owned report. These test results were referred to in Section 13.1 of the Enclosure of Maine Yankee November 14, 1994 letter to the NRC. The applicable testing material has been extracted from the CEOG's CE NPSD 957 and reported in the following C-E report M-PENG-TR-005, "Alloy 600 Tube Burst Tests":

Alloy 600 Tube Burst Tests

M-PENG-TR-005

March 1, 1995

D. J. Ayres
G. C. Fink

ABB COMBUSTION ENGINEERING NUCLEAR OPERATIONS
COMBUSTION ENGINEERING, INC.

Introduction

A test program was conducted to establish the temperature dependence of the burst pressure of the Alloy 600 tubes which are used in Combustion Engineering steam generators. This temperature dependence can then be used to establish the pressure required to demonstrate appropriate margins for potentially degraded tubes tested in operational steam generators.

Burst Test Program

The yield stress and the ultimate stress of Alloy 600 decrease with increased temperature. Therefore it is apparent that the burst pressure will also be lower at higher temperatures. In order to quantify the temperature effect, three burst tests were conducted at room temperature (Series 1) and three burst tests were conducted at 650F (Series 2). The tubes were 0.75 inch outside diameter Alloy 600 (high temperature mill annealed) tubing with wall thickness of 0.048 inches. The tubes were not constrained or supported and had no defects.

The burst test equipment for these tests include:

- an air driven positive displacement hydraulic pump to pump deionized water for pressurization (maximum capacity 1.0 liter per minute)
- a pressure gauge (0-20,000 psi)
- a pressure transducer connected to a strip chart recorder psi full scale)
- piping to connect test specimen to the burst cart.

The burst tests were conducted in accordance with ABB-CE Corrosion Technology Procedure 00000-MCC-994.

Burst Test Results

The burst pressures for the room temperature tests (Series 1) ranged between 11,690 psi and 11,615 psi as shown in Figure 1. The average for this series is 11,645 psi. The burst pressures for the tests conducted at 650F (Series 2) ranged form 10,256 psi and 10,073 psi as shown in Figure 2. The average for the tests at 650F is 10,159 psi. The average burst pressure at 650F is 12.75% below the average burst pressure at room temperature. Another way of expressing this relationship is that the average burst pressure at room temperature is about 15% higher than the average burst pressure at 650F.

Discussion of Results

The decrease in burst pressure at higher temperature is in the range of the decrease in yield strength and ultimate strength of Alloy 600 at higher temperature. The ASME Boiler and Pressure Vessel Code minimum specified yield strength for Alloy 600 (72Ni-15Cr-8Fe, SB163) is 35,000 psi at room temperature and 27,400 psi at 650F. The 650F yield strength is 21.7% below the yield strength at room temperature. The minimum specified ultimate strength is 80,000 psi for both temperatures. Since the burst pressure is controlled by a combination of the yield and ultimate strengths (often referred to as the flow stress) it is expected that the percentage decrease in burst pressure will lie between the percentage decrease of the yield strength and ultimate strength. Therefore the difference in the burst pressures determined in Test Series 1 and 2 is consistent with the anticipated difference in material properties for the two test temperatures.

Selection of In-Situ Pressure Test Pressure

The purpose if the in-situ pressure test is to demonstrate an acceptable margin against tube burst at operating and accident conditions. Since the in-situ test is performed at room temperature, the test pressure should be increased above the operating temperature target pressure in order to demonstrate the required margin at the higher temperature. Based on the test results of Series 1 and 2, it is recommended that the room temperature test pressure be increased to be 15% higher than the target pressure at operating temperatures. This pressure increase is conservative for operation temperatures below the test temperature of 650F.

FIGURE 1 - SERIES 1 BURST TEST

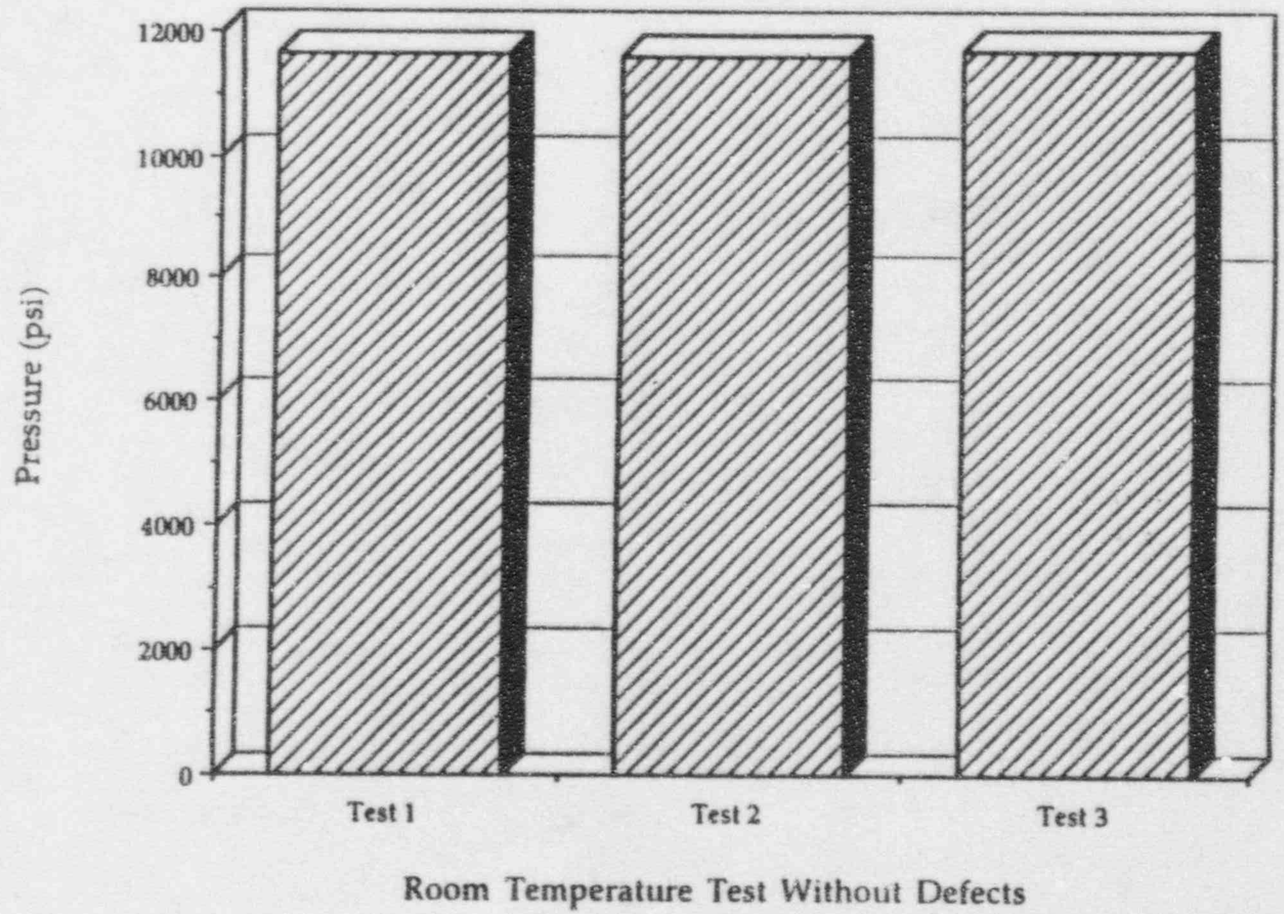
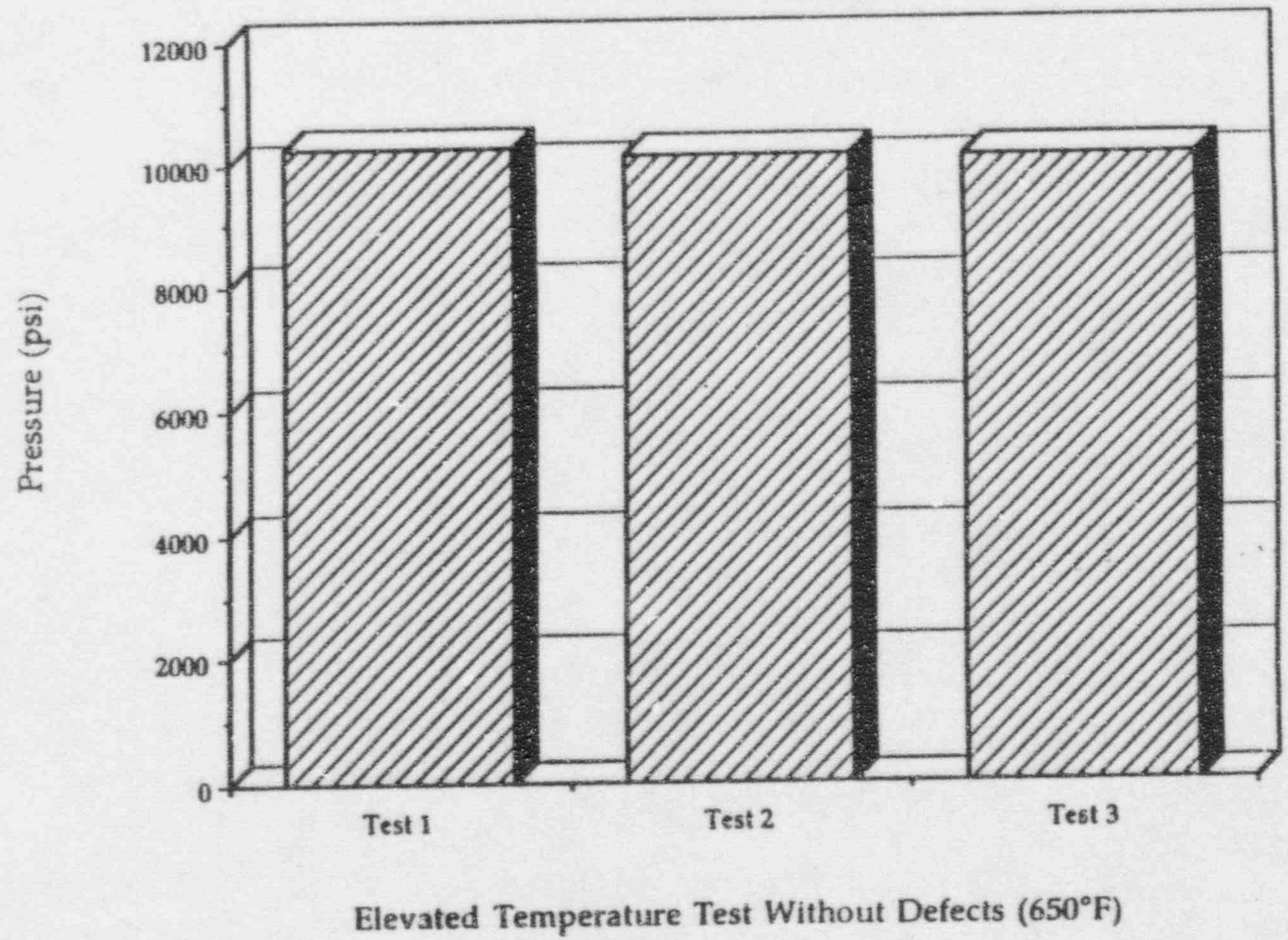


FIGURE 2 - SERIES 2 BURST TEST



ENCLOSURE 11

MAINE YANKEE NOVEMBER 14, 1994 LETTER ENCLOSURE'S REFERENCE 15

ABB C-E Letter M-PENG-94-13 dated November 10, 1994 to C. Eames of Maine Yankee, Subject: "S/G Tube Rupture in PWR Plants"

ABB

November 10, 1994
M-PENG-94-013

Mr. Charles Eames
Maine Yankee Atomic Power Company
329 Bath Road
Brunswick, ME 04011

Subject: **STEAM GENERATOR TUBE RUPTURES IN PWR PLANTS**

Dear Mr. Eames:

The purpose of this letter is to document the information that we discussed during our conversation of yesterday afternoon on steam generator tube ruptures. I have reviewed my files and discussed this matter with Mr. Allan McIlree of EPRI and Dr. James Begley of Packer Engineering to insure my information was correct. There have been eight steam generator tube ruptures (SGTRs) in US plants but there have not been an SGRTs where the rupture was in the circumferential orientation and the degradation was due to corrosion. The following table summarizes the SGTRs:

Plant	Date	Location	Mechanism
Point Beach-1	February, 1975	Tubesheet	Wastage, caustic IGSCC
Surry-2	September, 1976	U-bend	PWSCC, denting related
Prairie Island-1	October, 1979	Tubesheet	Wear by foreign object
Ginna	January, 1982	Tubesheet	Wear by foreign object
Fort Calhoun	May, 1984	U-bend	OD IGSCC
North Anna-1	July, 1987	Upper vertical leg	Fatigue with denting
McGuire-1	March, 1989	Lower vertical leg	OD IGSCC
Palo Verde	March., 1993	Upper vertical leg	OD IGSCC

These events are discussed in more detail below.

Point Beach-1

The Point Beach-1 tube rupture occurred in February 1975. Failure analysis indicated wastage and caustic intergranular stress corrosion cracking (OD initiated) in the tubesheet region were the mechanisms that caused the axial rupture.

Surry-2

The Surry-2 failure occurred in September, 1976. The failure, which was an axial failure in the U-bend region, was attributed to ID initiated stress corrosion cracking (PWSCC).

Prairie Island-1

The Prairie Island-1 tube rupture in October, 1979 was the result of foreign object damage which caused tube wall loss by a wear mechanism.

Ginna

The Ginna tube rupture in January, 1982, was the result of foreign object damage which caused excessive wear damage (tube wall loss) at a location just above the top of the tubesheet.

Fort Calhoun

The tube burst at Fort Calhoun occurred in May, 1984, as the plant was returning to service from a refueling outage. The failure was an axially oriented 1-1/4 inch long "fish mouth" type failure on the horizontal run of a tube that was near the periphery of the tube bundle. The failure was at the hot leg vertical strap location in an area that may have been steam blanketed. The tube was significantly deformed, apparently by a denting process, and the failure was at the location of maximum stress. Failure analysis indicated that the mechanism was OD initiated intergranular stress corrosion cracking.

North Anna-1

The tube burst at Surry-2 which occurred on July 15, 1987, was a circumferential failure on the cold-leg side of the steam generator, at the top of the seventh tube support plate. Failure analysis, after removal of the tube, indicated that the tube failed as a result of fatigue. Contributing factors were the presence of denting at the top side of the support plate, and the absence of an anti-vibration bar (AVB) for the tube in question.

McGuire-1

The tube burst at McGuire-1 occurred on March 7, 1989. The failure was an axial rupture that was 3-1/2 to 4 inches in length. The location was on the cold leg side of the steam generator in the economizer section near the number 20 support plate. The failure was OD initiated intergranular stress corrosion cracking that appeared to be associated with a shallow groove or scratch.

Palo Verde-2

The Palo Verde-2 steam generator tube burst occurred on March 14, 1993. The failure was an axial rupture, approximately two inches in length with additional part through-wall cracking above and below the burst for a total length of 8-9 inches. The burst was located above the number 9 (partial) eggcrate on the hot leg side of the steam generator. Failure analysis of sections of the burst tube indicated the mechanism was OD initiated intergranular stress corrosion cracking and intergranular attack.

In addition, there has been one SGTR in a foreign plant. That SGTR occurred at Mihama-2 in Japan in 1991 and was a situation similar to North Anna-1. The failure was a circumferential fatigue failure adjacent to a dented support. There was no evidence of corrosion degradation associated with the SGTR.

In summary, there have been no SGTRs to date where the failure was circumferentially oriented and resulted from a tube corrosion mechanism. There has been severe degradation, circumferentially oriented, in several tubes with average corrosion depth exceeding 90% through-wall.

I hope this information is sufficient to fulfill your request. If you need additional information, please contact me at (203) 285-4762.

Very truly yours,

John F. Hall

John F. Hall
Consultant
Primary Systems

JFH:prr

cc: D. Warren
P. Leombruni
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