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Oconee Nuclear Station

MASS AND ENERGY RELEASE AND CONTAINMENT RESPONSE METHODOLOGY

DPC-NE-3003-A

Revision 1

September 2004

Nuclear Engineering Division
Nuclear Generation Department
Duke Power Company

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 24, 2003

Mr. Ronald A. Jones
Vice President, Oconee Site
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P. O. Box 1439
Seneca, SC 29679

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 - SAFETY EVALUATION OF REVISIONS TO TOPICAL REPORTS DPC-NE-3000, -3003, AND -3005 (TAC NOS. MB5441, MB5442, AND MB5443)

Dear Mr. Jones:

By letter dated June 13, 2002, you submittal the following revisions to three topical reports: (1) DPC-NE-3000-P, Revision 3, "Thermal-Hydraulic Transient Analysis Methodology"; (2) DPC-NE-3003-P, Revision 1, "Mass and Energy Release and Containment Response Methodology"; and (3) DPC-NE-3005-PA, Revision 2, "UFSAR Chapter 15 Transient Analysis Methodology." You asked for approval of three revisions to support the replacement of the steam generators at Oconee Nuclear Station, Units 1, 2, and 3. You provided additional information in your letters dated May 21, July 7, and July 28, 2003.

Enclosure 1 contains our Safety Evaluation (SE) of DPC-NE-3000-P, Revision 3, and DPC-NE3005-PA, Revision 2; and Enclosure 2 contains our SE of DPC-NE-3003-P, Revision 1.

Sincerely,

Leonard N. Olshan, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosures: As stated

cc w/encls: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY OFFICE OF NUCLEAR REACTOR REGULATION

OF DUKE ENERGY CORPORATION TOPICAL REPORT DPC-NE-3003-P, REVISION 1

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

An August 11, 1993, letter from Duke Energy Corporation to the NRC requested review and approval of Duke Energy Corporation topical report DPC-NE-3003-P (Revision 0). The report describes methods of calculating the response of the Oconee Nuclear Station reactor building to a postulated design basis loss-of-coolant accident (LOCA) and a postulated design basis main steam line break (MSLB) accident. It includes the calculation of the mass and energy injected into the containment as a result of these events. The staff issued a March 15, 1995, safety evaluation (SE) approving this report for containment licensing calculations for the Oconee Nuclear Station.

The licensee revised this topical report in November 1997 and included Supplements 1 and 2. Supplement 1 is an MSLB analysis crediting the MSLB detection and feedwater isolation system. The addition of the MSLB detection and feedwater isolation system to the Oconee units and the accompanying analysis were approved by the NRC in a letter dated December 7, 1998. Supplement 1 replaced Chapters 5 and 6 of Revision 0. Supplement 2 is a comparison of the FATHOMS[1] containment analysis computer code with data from CVTR[2].

A June 13, 2002, letter from Duke Energy Corporation requested review and approval of DPC-NE-3003-P Revision 1 (Revision 1). The licensee provided supplemental information in letters dated July 7 and July 28, 2003. These supplements provide additional information and also discuss two additional changes to Revision 1 that were not included in the original Revision 1 report. Revision 1 incorporates changes to the methods described in the original topical report and Supplement 1, including a description of the proposed use of the GOTHIC 7.0 computer code[3] to perform containment thermal hydraulic calculations as well as the

[1]FATHOMS/Containment Analysis Package User's Manual, Version 2.4, Numerical Applications, Inc., April 20, 1989

[2]"Simulated DBA Tests of the Carolina Virginia Tube Reactor Containment - Final Report," R.C. Schmitt, G.E. Bingham, J.A. Norberg, Idaho Nuclear Corp., December 1970

[3]GOTHIC Containment Analysis Package Version 7.0, NAI-8907-02, Revision 13, July 2001

FATHOMS code and the replacement of RETRAN-02[4] with RETRAN-3D[5] for MSLB mass and energy release calculations. GOTHIC 7.0 is a general purpose thermal-hydraulics computer code developed by Numerical Applications, Incorporated for the Electric Power Research Institute (EPRI). RETRAN-3D was developed by Computer Simulation & Analysis, Inc. for EPRI. Its use for licensing calculations was approved by the NRC (with conditions and limitations) by a safety evaluation report (SER) dated January 25, 2001. Appendix A to Revision 1 discusses the application of RETRAN-3D to model the Oconee replacement steam generators. Appendix B to Revision 1 discusses compliance of this modeling with the conditions of the NRC SER approving RETRAN-3D. (The use of RETRAN-3D has been approved in Enclosure 1 that proceeds this SE.)

The computer codes discussed in Revision 1 are:

RELAP5/MOD2-B&W[6] is used to calculate the mass and energy released from the reactor coolant system (RCS) following both large and small break LOCAs.

The BFLOW code is used for long-term liquid and steam mass flow rates out of a large cold leg break in the reactor coolant system. Its description is provided in Revision 1.

The RETRAN-3D code is used to calculate the mass and energy release for a main steam line break. This code was reviewed and approved by the NRC[7]. Revision 0 utilized RETRAN-02. Attachment 3 to the licensee's June 13, 2002, letter, Item 9, discusses some differences between RETRAN-02 and RETRAN-3D. In addition, Appendix B of the June 13, 2002, letter evaluates the use of RETRAN-3D for this application in terms of conditions and limitations of the NRC SER on RETRAN-3D.

The FATHOMS/DUKE-RS[8] code is used to calculate the reactor building response to high energy line breaks.

The GOTHIC 7.0 computer code is also used to calculate the reactor building response to high energy line breaks. Attachment 1 to the licensee's July 7, 2003, letter describes the important differences between FATHOMS/DUKE-RS and GOTHIC 7.0. Appendix C to Revision 1 provides a summary description of the GOTHIC 7.0 code.

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[4]RETRAN-02 A Program for Transient Thermal Hydraulic analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCMA Revision 4 EPRI November 1988

[5]RETRAN-3D - A Program for Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP 7450 (A), Volumes 1-4, Revision 5, July 2001

[6]B&W-10164P, Revision 1, "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water reactor LOCA and Non-LOCA Transient Analysis," Babcock and Wilcox, October 1988

[7]Letter from S.A. Richards (USNRC) to G.L. Vine (Electric Power Research Institute-EPRI) Safety Evaluation report on EPRI topical report NP-7450(P), Revision 4, "RETRAN-3D-A Program for Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems," January 25, 2001.

[8]CAP - Containment Analysis Package (FATHOMS 2.4), Numerical Applications, Inc., October 10, 1989

Revision 1 discusses containment analysis using the FATHOMS code. However, it is Duke Energy Corporation's intention to have the option to use GOTHIC 7.0 for these calculations.

Since Revision 0 has previously been approved by the NRC, this review will concentrate on the differences between Revision 0 and Revision 1. These changes are listed and evaluated in the Technical Evaluation section of this SE. Changes to the topical report that are editorial or descriptive are not discussed. The NRC staff agrees that the changes designated editorial are indeed editorial. In addition, the two changes to Revision 1 that were included in the July 7, 2003, letter but not discussed in the topical are discussed after the discussion of the items in the June 13, 2002, topical report submittal.

2.0 REGULATORY ANALYSIS

Containment pressure analyses are required for Oconee Nuclear Station as part of the design basis evaluation (DBE). All three Oconee units were licensed for construction prior to May 21, 1971. Therefore, the General Design Criteria of 10 CFR Part 50, Appendix A are not applicable to the Oconee Nuclear Station. Section 3.0, "Design of Structures, Components, Equipment and Systems," of the Oconee Updated Final Safety Analysis Report (UFSAR) lists the principal design criteria "developed in consideration of the seventy General Design Criteria proposed by the AEC (Atomic Energy Commission) in the Federal Register of July 11, 1967." Of these, the following are relevant to this SE:

> Criterion 10: *Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity, and together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.*
>
> Criterion 15: *Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.*
>
> Criterion 49: *The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design basis leakage rate, the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for the effects from metal water or other chemical reactions that could occur as a consequence of failure of the Emergency Core Cooling Systems.*

The licensee utilized guidance from two sources in establishing the analytical methods and assumptions for mass and energy release calculations and containment calculations. The first source is the NRC Standard Review Plan Sections 6.2.1, "Containment Functional Design," 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," and 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures."

The second source of guidance is the American National Standards Institute Standard ANSI/ANS-56.4-1983, "Pressure and Temperature Transient Analysis for Light Water Reactor Containments." The use of this ANSI/ANS standard is not required by the NRC. However, the ANSI/ANS standard provides guidance in making conservative assumptions for safety

calculations. Revision 0 used this guidance. Revision 1 also uses this guidance. In addition to this guidance, Revision 0 and Revision 1 both use sensitivity calculations to ensure that the input assumptions are conservative.

3.0 TECHNICAL EVALUATION

Revision 1 describes 76 changes to Revision 0. These changes are evaluated in this section. Some changes are editorial and are not discussed. Some of the 76 changes are identical but found in different sections of the report. Each change is numbered consistent with the item numbers in Revision 1. The section of Revision 1 for each change is identified.

In addition, the two changes to Revision 1 discussed in the July 7, 2003, letter are evaluated.

Cover page and frontal pages

1. and 2. Editorial

Chapter 1 Introduction

3., 4., 5. Editorial

Chapter 2 RELAP5/MOD2-Babcock & Wilcox (B&W) - Primary System Mass and Energy Release

6. Section 2.1.2 Main feedwater piping

The RELAP5 nodalization for the main feedwater piping is revised to include the main feedwater piping between the last check valve and the steam generator. The licensee states that this enables modeling flashing of the main feedwater in the piping if the steam generator pressure decreases low enough for the flashing to occur. If flashing occurs, additional hot water will be expelled into the steam generator with the potential to increase primary-to-secondary heat transfer. Revision 1 states that calculations show this to be a minor effect. It is added for completeness. Since it makes the model more accurate by modeling a real effect, this change is acceptable.

7. Section 2.1.2 Replacement Once-Through Steam Generator (ROTSG) Modeling

Revision 1 states that the ROTSGs are modeled with the same noding as the original steam generators. The differences between the current and the replacement steam generators are described in Revision 1. In general, the licensee characterizes the replacement of the steam generators as "like-for-like." The input data for the ROTSGs has been recalculated to reflect the differences based on data supplied by the manufacturer, B&W Canada, Inc. Since the geometric configuration is very similar and the other input changes are minor, the NRC staff finds the proposed use of the same RELAP5 noding for the ROTSGs to be acceptable.

8. Editorial

9. New Section 2.5 RETRAN-3D code description

This section provides a brief discussion of RETRAN-3D. RETRAN-02 was used previously to model the mass and energy release from the MSLB. This was described in Revision 0. Appendix A of Revision 1, "Methodology Revision for Oconee Replacement Steam Generators," provides a description of the application of RETRAN-3D to steam line break mass and energy release calculations. Appendix B of Revision 1, "Evaluation of RETRAN-3D SER Conditions and Limitations for the Oconee RETRAN model with ROTSGs" discusses the compliance of the RETRAN-3D ROTSG modeling with the conditions of the January 25, 2001, NRC SER approving RETRAN-3D.

Modeling of Chapter 15 events with RETRAN-3D, including the MSLB mass and energy input for containment calculations, is evaluated in the SE contained in Enclosure 1 that proceeds this SE. The NRC staff considers this SE adequate to cover the types of analyses discussed in Revision 1.

10. A new Section 2.6 "GOTHIC 7.0 Code Description" is added to Revision 1.

Appendix C of Revision 1 provides a brief description of GOTHIC 7.0.

See the evaluation of Chapter 6 revisions later in this SE for an evaluation of the use of GOTHIC 7.0 for Oconee Nuclear Station containment thermal hydraulic analysis.

11. Editorial

Chapter 3 Large Break LOCA Mass and Energy Release Analyses

12. Section 3.1: The termination of the RELAP5 analysis is changed from 30 minutes to the end of the borated water storage tank (BWST) injection phase.

Revision 1 states that with further experience in applying the methods of Revision 0 it has become apparent that the appropriate time for terminating the RELAP5 analysis is at the transition from the BWST injection phase (that is, the time when the BWST has reached the level at which the emergency core cooling system (ECCS) pumps and reactor building spray (RBS) pumps switch suction to the emergency sump), not necessarily at 30 minutes, which was the criterion used in Revision 0.

At the transition from the BWST injection to recirculation, the water in the reactor vessel is in a steady state condition and the transition to the BFLOW code or the RELAP5 code (see Item 13), which calculates mass and energy release after blowdown, is more appropriately made. The NRC staff finds this change to be consistent with the accident scenario and it is therefore acceptable.

13. Section 3.1: Clarify that the BFLOW code is only used for cold leg break LOCA analysis and describe what is used for hot break analysis.

Revision 0 stated: "The BFLOW code is used to calculate the mass and energy release for the remainder of the [LOCA] analysis."

However, the BFLOW code was only intended for calculating the mass and energy release after blowdown for cold leg breaks. Revision 1 is revised to make this clear. RELAP5 calculates the post-blowdown mass and energy release for hot leg breaks. This change reflects current practice. RELAP5 is capable and acceptable for doing this type of calculation and the change is therefore acceptable.

14. Section 3.2: The initial pressurizer level is updated to be consistent with the technical specifications.

The licensee proposes to use a value for pressurizer level that is consistent with the high level value of the technical specifications, plus instrument uncertainty. This is conservative for mass and energy release calculations and is, therefore, acceptable.

15. Section 3.2: Revision 1 states that a conservative high initial mass is assumed for the ROTSGs. The values are not specified.

A conservatively high steam generator mass maximizes the energy transferred from the secondary to the primary during the LOCA mass and energy release and is therefore a conservative assumption. It is acceptable to not specify the numerical value of the steam generator operating level since this is an input value. The specification that the initial mass will be conservatively high is the significant condition for this topical report. Therefore, the licensee's change is acceptable.

16. Section 3.2: The temperature of the water in the core flood tanks is increased from 120 °F to 130 °F. Since the higher temperature is conservative for mass and energy release calculations, this is acceptable.

17. Section 3.2: Rather than specifying the RCS flow to be a low value, Revision 1 will specify that a sensitivity analysis must be done to determine whether low RCS flow or high RCS flow is more conservative with respect to peak containment pressure.

The NRC staff determined this approach is acceptable since the conservative RCS flow condition will be used.

18. Editorial

19. Section 3.3.1.1: Bounding high values for fuel stored energy will be used. In addition, the licensee's July 7, 2003, letter states that the gap conductance is held constant at its initial value.

> The NRC staff concludes that this is a bounding assumption since the gap conductance would actually decrease as the blowdown progresses and is, therefore, acceptable.

20. Section 3.3.1.1: The model of the main feedwater lines is revised to reflect the fact that the main feedwater system is realigned to feed through the upper auxilliary feedwater header following the trip of all four reactor coolant pumps. The licensee states that this is to enhance natural circulation heat transfer. This results in less feedwater flow following realignment.

> Since this modeling change reflects the design of the plant, the NRC staff finds this change to be acceptable.

21., 22., 23. Editorial

24. Section 3.3.2.1 of Revision 0 calculated the amount of metal-water reaction that takes place during the short term period of a postulated LOCA. Revision 1 assumes a constant amount of 1 percent, which is the limit allowable by 10 CFR 50.46 for core-wide reaction of coolant with zirconium.

> The NRC staff determined that since 1-percent metal-water reaction is the maximum allowed by regulation, this is conservative and acceptable.

25. Section 3.3.2.2: The modeling of the low pressure injection (LPI) flow is revised to be consistent with the current plant design. Following loss of 4160 V switchgear, flow from two high pressure injection (HPI) pumps and one LPI pump will be available for injection into the reactor vessel.

Previously a constant value for LPI flow was assumed. The LPI flow is now modeled as a function of the RCS backpressure until it is throttled by the operator.

> Modeling the LPI flow as a function of RCS backpressure is more accurate than assuming a constant value. Throttling the flow is consistent with the Oconee emergency operating procedures. Therefore, the NRC staff concludes this change is acceptable.

26. Section 3.3.2.2 of Revision 1 revises the reactor coolant pump two-phase head degradation model.

> The specific model is proprietary but it has been previously reviewed and approved by the NRC and is acceptable for this application.

27. Section 3.3.3.1: The calculation of the long term LOCA fission heat uses the same RELAP5 modeling used for the short term analyses.

The NRC staff determined that this is an improvement in the long term analysis. Revision 0 uses fission heat values calculated by a separate computer code. In Revision 1 these calculations are done directly in the RELAP5 code that calculates the LOCA response. This also better reflects the initial and boundary conditions and the transient behavior. Thus, this is an improvement in modeling and is acceptable to the NRC staff.

28. Section 3.3.3.2: The long term large break LOCA containment analysis considers both cold leg pump discharge breaks and hot leg breaks. Revision 0 considered only a double ended guillotine break located at the A1 cold leg pump discharge.

The NRC staff determined that this is an improvement in determining the limiting LOCA and is, therefore, acceptable.

29. Section 3.3.3.2: The assumed location and the size of the hot leg break for long term containment response to a LOCA is discussed. For the limiting containment environmental qualification response, the limiting break location is a double-ended guillotine break at the reactor vessel outlet nozzle. For the maximum containment sump water temperature analysis, the limiting break is a double-ended guillotine break located at the reactor vessel outlet nozzle.

Since these findings are the result of calculations examining possible locations for the most conservative results, the NRC staff finds this selection of breaks acceptable.

30. Section 3.3.3.2 of Revision 1 discusses the timing of the depletion of the BWST and the switch to pump recirculation. Depending on the objective of the analysis (e.g., peak pressure, sump temperature), high or low ECCS flow rates are chosen.

Since these findings are based on calculations examining possible break locations for the most conservative results, the NRC staff finds this selection of ECCS flow rates for BWST depletion to be acceptable.

31. Section 3.3.3.2: In Revision 0 the LPI pump flow was assumed constant. In Revision 1 the LPI flow rate is a function of RCS backpressure until the flow is throttled in accordance with the emergency operating procedures. In addition, in Revision 0 there is a discrepancy between the RBS flowrate assumed for the calculation of the rate of BWST depletion (the instrument uncertainty in flow rate is added to the nominal flow rate value) and the calculation of the RBS system flow rate in FATHOMS (the instrument uncertainty in flow rate is subtracted from the nominal flow rate value) in order to be conservative in each case. In Revision 1 these flow rates are the same.

Revision 1 states that the changes to these analyses are necessary since plant procedures and station design have changed. Although the flow rates have been revised, Revision 1 states that conservative values are still used. Since the analysis is revised to reflect changes to plant procedures and design, and the

resulting flow rates used in the accident analyses are still conservative, these changes are acceptable to the NRC staff.

32. Section 3.3.3.4: For the long term LOCA containment analysis, the termination of the RELAP5 analysis is changed from 30 minutes to the end of the BWST injection phase.

See the evaluation of Item 12.

33. Editorial

Chapter 4 Small Break LOCA Mass and Energy Release

34. Section 4.1: Revision 0 described the use of a simple containment model in RELAP5 for the determination of containment conditions in small break LOCA mass and energy calculations. This has been revised in Revision 1. RELAP5 small break LOCA mass and energy calculations now use containment conditions from a FATHOMS analysis rather than a containment model in RELAP5.

There are two FATHOMS results that do impact the RELAP5 calculation. These are the RBS actuation time (the RBS actuates on containment high pressure) that affects the timing of the switch to recirculation from the containment sump and the containment sump water temperature. Iterations are performed between FATHOMS and RELAP5 until the results converge or conservative agreement is achieved between the FATHOMS and RELAP5. The licensee provided Table 9-1 in the July 7, 2003, letter that demonstrates good agreement between the sump temperature at the start of recirculation and the peak sump temperature for both FATHOMS and RELAP5. The NRC staff noted there is good agreement.

The licensee's use of FATHOMS to determine the RELAP5 containment boundary conditions is acceptable to the NRC staff since FATHOMS is intended for calculations of this type (sump temperature and containment pressure).

35. Section 4.1: The transition from RELAP5 to FATHOMS for the small break LOCA mass and energy release modeling requires consideration of the stored energy at the time of the transition. This modeling has been revised in Revision 1. The stored energy in the secondary system fluid and structural metal is now included. FATHOMS now includes all heat sources. The modeling is done in FATHOMS with heater components. The licensee states that this is an improvement and a simplification.

The NRC staff concludes that inclusion of more heat sources is an improvement. Therefore, the NRC staff finds the change to be acceptable.

36. Section 4.3.1: Both Revision 0 and Revision 1 use the RELAP5 point kinetics model to calculate delayed neutron power as a function of time. Both limit the shutdown margin to the technical specification shutdown margin limit, 1 percent *K/K. Revision 0 calculated the shutdown reactivity but limited the amount to the technical specification minimum shutdown margin. In Revision 1 the thermal feedback calculation has been deleted and the shutdown margin of 1 percent *K/K is used.

Since the amount of shutdown reactivity has not changed, the NRC staff determined that the proposed revision is acceptable.

37. Section 4.3.1: Since no thermal feedback is modeled (See Item 36), the total rod worth is set to the technical specification shutdown margin value.

Since the amount of shutdown reactivity has not changed, the NRC staff finds this change acceptable.

38. Section 4.3.1: A new section, "Fission Products and Decay Heat," is added to Revision 1 to describe the decay heat modeling. ANSI/ANS Standard 5.1-1979 is used with a 2σ uncertainty.

ANSI/ANS Standard 5.1-1979 has been used acceptably for the determination of decay heat and the NRC staff determined it is acceptable to use in this case.

39. Section 4.3.2, "Reactor Building Model," is deleted in its entirety and replaced with a new subsection titled "Containment Boundary Conditions," since the model is no longer used in the Revision 1 methods. This section describes the use of FATHOMS to determine two boundary conditions for RELAP5 mass and energy release calculations. These two boundary conditions are: (1) the containment backpressure against the LOCA break flow discharges and (2) the sump water temperature used for the ECCS suction supply after transition to the sump recirculation mode. These quantities were determined by a RELAP5 containment model in Revision 0.

Revision 1 states that the boundary conditions are based on FATHOMS cases that are consistent with the RELAP5 analysis in terms of break size and intended application. Since FATHOMS calculates containment pressure and sump temperature conservatively, this change is acceptable.

40. Section 4.3.2: FATHOMS calculates boundary conditions for RELAP5 small break LOCA mass and energy release calculations. In addition to those boundary conditions listed in Item 39, spilled HPI flow is also modeled in FATHOMS for input into RELAP5 for small break LOCA mass and energy release calculations.

See Item 34.

41. Section 4.3.2: Revision 1 revises the modeling of emergency feedwater (EFW) actuation and control. The revision makes the modeling consistent with current station procedures and the UFSAR Chapter 15 small break LOCA peak cladding temperature analysis. The delay from reactor trip to EFW actuation is changed from 10 minutes to 20 minutes. The steam generator level is controlled to the loss of subcooled margin (LSCM) setpoint. Revision 0 specified the value of this setpoint. Revision 1 does not specify the value, only that the LSCM setpoint is used and must be adjusted for level instrument uncertainty.

The NRC staff determined that this change is consistent with current station procedures and the small break LOCA peak cladding temperature analysis and

conservatively accounts for instrument uncertainty. Therefore, it is acceptable to the NRC staff.

42. Section 4.3.2: The modeling of the steam generator pressure control is revised to change the time at which the atmospheric dump valves are credited and to change the target cooldown rate. An operator is assumed to begin manipulating these valves locally beginning at 60 minutes, rather than the previous time of 30 minutes, after break initiation. The cooldown rate is revised from 80 °F/hr to > 50 °F/hr, based upon the core exit temperature.

The NRC staff determined that these changes are consistent with current station procedures and are therefore acceptable.

43. Section 4.3.2: The modeling of the emergency feedwater system is revised to credit only one motor-driven emergency feedwater pump initiating at 20 minutes and feeding only one steam generator.

The single failure of a 4160 V switchgear results in the loss of one motor-driven emergency feedwater pump. No credit is taken for the turbine-driven emergency feedwater pump. Revision 1 states that this change is consistent with current station procedures and the small break LOCA peak cladding temperature analysis. The change is, therefore, acceptable to the NRC staff since it more accurately reflects licensing bases limitations. The NRC staff notes that taking no credit for the turbine driven emergency feedwater pump is consistent with the licensing basis for the MSLB detection and feedwater isolation system.

44., 45., 46. Editorial

Chapter 5 Steam line Break Mass and Energy Release Analyses

47., 48, 49., 50., 51., 52. Editorial

Chapter 6 Containment Analysis

Revision 1 proposes the use of two containment thermal-hydraulic analysis computer codes. The use of FATHOMS was proposed in Revision 0 and approved by the NRC in a letter dated March 15, 1995. Revision 1 proposes some changes to this analysis that are discussed below.

Revision 1 also proposes the use of the GOTHIC 7.0 computer code for Oconee containment thermal-hydraulic calculations. GOTHIC 7.0 was developed by Numerical Applications, Inc., for EPRI. GOTHIC 7.0 contains state-of-the-art models for the phenomena expected to occur during a high energy line break (LOCA or MSLB) in a nuclear reactor containment.

GOTHIC 7.0 has been validated against an extensive data base including analytical solutions and data from separate effects and integral tests[9].

[9]GOTHIC Containment Analysis Package: Qualification Report Version 7.0 July 2001

The licensee's July 7, 2003, letter provides a summary of the major differences between GOTHIC 7.0 and FATHOMS that are relevant to the licensee's application of GOTHIC 7.0 to Oconee. These include the modeling of drops and the modeling of interphase heat transfer.

The licensee's July 7, 2003, letter describes the modeling of the blowdown flow in GOTHIC 7.0. Separate flowrates and enthalpies for the steam and the liquid phases of the blowdown are calculated in the RELAP5 code and passed to GOTHIC 7.0 as separate boundary conditions. In GOTHIC 7.0 all liquid flow enters as droplets when the RCS pressure exceeds the containment pressure.

While FATHOMS solves one energy equation for the vapor phase (air and steam) and another for the liquid phase (drops and liquid), GOTHIC 7.0 models the drops and liquid within a computational control volume so that each can have a different temperature.

The GOTHIC 7.0 drop liquid conversion option signals the use of the GOTHIC 7.0 default models for drop behavior including entrainment, agglomeration, and deposition. The calculated bulk average velocities of the Oconee containment atmosphere are not great enough to cause entrainment. GOTHIC 7.0 does model the dripping of condensate from ceilings and equipment (1/6th of the condensation rate for the volume). The July 7, 2003, letter states that this dripping model was included to improve agreement with experimental data.

The licensee's July 7, 2003, letter states that the Uchida condensation heat transfer coefficient will be used for all heat slabs in GOTHIC 7.0 calculations. The Uchida correlation is recommended by ANSI/ANS Standard 56.4-1983, "Pressure and Temperature Transient Analysis for Light Water Reactor Containments." While Oconee is not considered to be an SRP plant, the use of the Uchida correlation is consistent with the guidance in SRP 6.2.1.1.A, "PWR Dry Containment, Including Subatmospheric Containments." The GOTHIC Uchida model was also previously accepted by the staff in approving Revision 0 and as part of the review of the application of GOTHIC 6.0 to Kewaunee (ADAMS Accession No. ML012490176). It is considered conservative by the NRC staff and is typically used for licensing calculations.

Revision 0 notes that ANSI/ANS Standard 56.4-1983 allows the Tagami and Uchida condensation heat transfer correlations. The Tagami correlation applies to blowdown conditions. However, since the Tagami correlation reverts to the Uchida correlation after blowdown, Revision 0 uses the Uchida correlation throughout. This remains the case in Revision 1. The NRC staff determined that this is conservative since the Tagami correlation typically predicts a higher rate of heat transfer.

The Oconee reactor building is modeled as two volumes; the containment air volume and a separate sump volume that consists of the lowest 6.35 feet of the reactor building. The two nodes are referred to as the atmosphere region and the sump region, respectively. The noding is analogous to the CONTEMPT[10] containment model except that they are separate volumes in FATHOMS whereas the atmosphere and sump regions are part of the same volume in CONTEMPT.

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[10]Hargroves, Don W., et al, CONTEMPT-LT/028 - A Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident, USNRC NUREG/CR-0255 March 1979.

Following a LOCA, the nitrogen cover gas in the core flood tanks is released to the containment through the break. The addition of this gas is, in general, conservative. However, the July 7, 2003, letter points out that because the peak containment pressure is reached prior to the time at which the nitrogen gas is injected into the containment for all cases analyzed, it has no impact on the calculated peak containment pressure for Oconee.

GOTHIC 7.0 offers other options for calculating condensation heat transfer. These have not been reviewed by the NRC staff for application to Oconee (e.g., the mist diffusion layer model and the Gido-Koestel correlation) and shall not be used for design basis licensing calculations.

The July 7, 2003, letter provided comparisons of GOTHIC 7.0 containment LOCA and MSLB calculations with those of FATHOMS. These comparisons show that:

GOTHIC 7.0 predicts LOCA peak pressures that are generally slightly lower (approximately 0.5 psi)

GOTHIC 7.0 predicts LOCA temperatures that are generally slightly lower (approximately 0.5 °F)

GOTHIC 7.0 predicts MSLB pressures that are generally slightly higher (approximately 0.5 psi)

GOTHIC 7.0 predicts MSLB peak temperatures that are generally lower (approximately 10 °F)

GOTHIC 7.0 predicts sump water volumes that are generally lower for both the LOCA and MSLB

GOTHIC 7.0 predicts sump water temperatures that are generally lower for both the LOCA and MSLB

The July 7, 2003, letter explains these trends as follows. For the LOCA events, both codes predict a saturated containment atmosphere due to the large amount of drops in the containment atmosphere. The lower containment pressure and temperature predicted by GOTHIC 7.0 are due to additional drop holdup in the containment atmosphere. The effective increase in the heat capacity of the atmosphere due to the difference in drops in the atmosphere explains the reduction in containment temperature compared with FATHOMS.

For the MSLB events, the higher pressure predicted by GOTHIC 7.0 is due to additional steam generation from the improved interface heat and mass transfer models and from the drops that are generated due to dripping. The additional vaporization takes heat from the atmosphere and reduces the atmosphere temperature.

With the restriction on the use of condensation heat transfer options, the NRC staff finds the use of GOTHIC 7.0 acceptable as described in Revision 1 and as discussed below.

53. Section 6.1: For the long term LOCA analyses, the large break and small break containment calculations are divided into two time segments. They were previously divided into three.

The large break and small break mass and energy release calculations are divided into two segments rather than three. The two time segments are: (1) ECCS injection flow coming from the BWST, and (2) after switchover to sump recirculation. In order to be consistent with this division, the containment analysis is also divided into two time segments so that they can be modeled with the same boundary conditions. The NRC staff determined that this approach makes the analysis more consistent and is, therefore, acceptable.

54. Section 6.2.3: The blowdown mass flow is assumed to consist of droplets. The size of these droplets is assumed to be 100μ[11].

The GOTHIC computer code assumes the liquid portion of the break flow consists of droplets and recommends a drop size of 100μ based on spray tests reported in the Journal of the American Institute of Chemical Engineering, Volume 8, No. 2, 1962. This is acceptable to the NRC staff based on the following:

> During a DBE LOCA, the water entering the containment as RCS break flow is at a temperature above the saturation temperature at the containment pressure. Upon entering the containment, a portion of the water flashes to steam, fracturing the water jet into fine droplets. The experimental data cited above have shown that when superheated water flashes to steam, the mean drop diameter is less than 100μ.
>
> The GOTHIC 7.0 qualification analyses, presented in the GOTHIC 7.0 code documentation qualification report, were performed using a drop diameter of 100μ. These qualification analyses showed that GOTHIC 7.0 calculations, with the 100μ assumption, agreed with, and typically bounded, the measured pressure and temperature response from blowdown tests and measured pressure drops from orifice pressure drop tests.
>
> A 100μ drop has a terminal velocity (rainout velocity) of between 1 and 2 ft/sec. This is a realistic terminal velocity and allows for the break drops to be in the containment atmosphere for a realistic time.

Therefore, the use of the 100μ drop size is acceptable.

55. Section 6.3.1: The RELAP5 analysis now runs through to the end of the injection phase rather than stopping at 1800 seconds.

See Item 12.

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[11] μ represent "micron," a millionth of a meter

56. Section 6.3.2: The RELAP5 analysis now runs through to the end of the injection phase rather than stopping at 1800 seconds. Also there are two time segments instead of three.

See Item 12.

57., 58., 59. Section 6.3.3: The large break and small break containment calculations are divided into two time segments. They were previously divided into three.

See Item 53.

60. Section 6.3.3: The blowdown mass flow is assumed to consist of droplets. The size of these droplets is assumed to be 100μ.

See item 54.

61. Section 6.3.3: The LPI flowpaths and mixing volumes are revised.

The LPI flowpath is revised to reflect the fact that for some failure scenarios, it is possible for only one LPI cooler to be supplied with cooling water although both will have LPI flow. The current model combines both LPI trains into one equivalent train. The NRC staff found that the proposed revision more accurately models the LPI system and is, therefore, acceptable.

62. Through 69. Revision 1 describes changes made to the noding of the FATHOMS model.

The NRC staff did not review the details of the revised noding. The NRC staff determined that the adequacy of the noding can be judged by the reasonableness of the results. On the basis of reasonable results, the noding is acceptable.

70. Section 6.4.3: Revision 1 revises the modeling of RCS and secondary side heat structures.

The change is proprietary. The NRC staff determined that the change makes the accounting of heat structures more complete and is, therefore, acceptable.

71. Section 6.4.3: The droplet size of the blowdown mass is changed to 100μ.

See item 54.

72. Section 6.4.5: The noding of the small break FATHOMS model has been revised.

The NRC staff did not review the detailed noding of the FATHOMS model. The NRC staff determined that the adequacy of the noding can be judged by the reasonableness of the results. On the basis of reasonable results, the noding is acceptable.

73. Section 6.4.5: Revision 1 discusses a boundary condition for a break in a high pressure injection line. A model for the BWST is eliminated from FATHOMS. The NRC staff determined that the event is adequately modeled without modeling the BWST. Therefore, this change is acceptable.

74. Section 6.6 Editorial

75. Chapter 7 Editorial

76. Supplement 1 Editorial

Additional revisions to Revision 1 identified in the July 7, 2003, letter from Duke Energy Corporation:

The long-term mass and energy release boundary condition modeling in FATHOMS has been installed in GOTHIC 7.0. This modeling approach enables the interpolation of the BFLOW code results (for mass and energy release after blowdown) as an input boundary condition. It is described in Sections 2.2.1.1, 2.4.1.2, and 6.3 of Revision 1.

This modification is made in lieu of using the control variables available in GOTHIC 7.0. The proposed modeling is similar to that used in FATHOMS that was approved by the NRC's March 15, 1995, SER and is acceptable.

Because GOTHIC 7.0 is being modified to accept BFLOW input, the licensee's version of GOTHIC 7.0 will be uniquely identified as GOTHIC 7.0/DUKE.

Environmental Qualification

The temperature envelope curve for environmental qualification was provided in a July 28, 2003, letter. The GOTHIC 7.0 results are compared to the FATHOMS curves for containment atmosphere temperature, pressure, and the mass weighted liquid temperature, which is a combination of the liquid and droplet phases. The NRC staff found that the agreement is good. The differences are explainable in terms of the difference in modeling between the two codes.

4.0 CONCLUSION

Duke topical report DPC-NE-3003-P, "Mass and Energy Release and Containment Methodology," has been revised. Two new computer codes are proposed for Revision 1: RETRAN-3D will replace RETRAN-02 for steam line break mass and energy release calculations and the GOTHIC 7.0 computer code will be used in addition to the FATHOMS code approved as part of DPC-NE-3003-P for containment thermal hydraulic calculations. Both new codes upgrade the calculation capabilities of the codes they replace. (RETRAN-3D has been approved by the NRC in Enclosure 1 that precedes this SE.)

DPC-NE-3003-P, Revision 1, proposes to use GOTHIC 7.0 with conservative input and heat transfer correlations. For these reasons, the NRC staff finds the use of GOTHIC 7.0 to be acceptable. GOTHIC 7.0 offers options for calculating condensation heat transfer other than

the Uchida correlation proposed in DPC-NE-3003-P, Revision 1. These correlations have not been reviewed by the NRC for application to Oconee (e.g., the mist diffusion layer model and the Gido-Koestel correlation) and shall not be used for design basis licensing calculations.

In addition, many changes are made in Revision 1 to calculation assumptions to reflect changes in plant operation and procedures. These changes ensure that the safety calculations will accurately predict the plant behavior following a design basis accident and that the relevant safety criteria will not be exceeded.

The changes made in Revision 1 comply with the AEC General Design Criteria 10, 15, and 49, published in the Federal Register of July 11, 1967, which apply to Oconee.

The methods described in Revision 1 may be used to perform analyses in support of licensing applications related to containment accident response for peak containment pressure and temperature and environmental qualification.

Principal Contributor: R. Lobel

Date: September 24, 2003

PROPRIETARY INFORMATION

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 15, 1995

Mr. M. S. Tuckman
Senior Vice President
Nuclear Generation
Duke Power Company
P. O. Box 1006
Charlotte, North Carolina 28201-1006

SUBJECT: SAFETY EVALUATION FOR TOPICAL REPORT DPC-NE-3003-P, "MASS AND ENERGY RELEASE AND CONTAINMENT RESPONSE METHODOLOGY" (TAC NOS. M87258, M87259, AND M87260)

Dear Mr. Tuckman:

By letter dated August 11, 1993, Duke Power Company (DPC), submitted the subject topical report for review. The report describes the DPC methodology for simulating the mass and energy release from high energy line breaks and the resulting containment response for the Oconee Nuclear Station. Supplemental information in response to our requests for additional information was submitted in letters dated February 16, June 1, and June 30, 1994.

The NRC staff has completed its review of the topical report and the supplemental submittals. The staff concludes that the DPC-NE-3003-P basic methodology may be used by DPC to perform future reanalyses in support of licensing applications related to containment accident response. The staff's Safety Evaluation (SE) is included as Enclosure 1. Proprietary information included in this SE was identified by your staff and documented by your letter dated February 23, 1995. The enclosure should be controlled and distribution limited to personnel with a "need to know." The enclosure is considered exempt from Public Disclosure in accordance with Title 10, Code of Federal Regulations, Part 2.790. However, a copy of this letter, with a non-proprietary version of the SE as an enclosure, will be placed in the NRC Public Document room. This completes NRC actions for TAC Nos. M87258, M87259, and M87260. If you have questions regarding this matter, contact me at (301) 415-1495.

Sincerely,

L. A. Wiens, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosure and cc w/o encl:
See next page

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Oconee Nuclear Station

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SAFETY EVALUATION REPORT
OFFICE OF NUCLEAR REACTOR REGULATION
DIVISION OF SYSTEMS SAFETY AND ANALYSIS
CONTAINMENT SYSTEMS AND SEVERE ACCIDENT BRANCH
MASS AND ENERGY RELEASE AND CONTAINMENT RESPONSE METHODOLOGY
OCONEE NUCLEAR STATION
DOCKETS NUMBERS 50-269, 50-270 AND 50-287
DUKE POWER COMPANY TOPICAL REPORT DPC-NE-3003-P

ENCLOSURE

TABLE OF CONTENTS

1.0 INTRODUCTION

Containment pressure and temperature analyses are performed to determine the pressure and temperature loads within the containment that would result from postulated pipe breaks inside containment. The results of the containment pressure and temperature analyses establish design criteria, test criteria and environmental qualification criteria for containment systems, structures and components.

By letter dated August 11, 1993, the licensee submitted topical report DPC-NE-3003-P, "Mass and Energy Release and Containment Response Methodology," for staff review and approval. The topical report describes the licensee's proposed new methodology for analysis of mass and energy releases from in-containment high-energy line breaks and the predicted containment pressure and temperature responses. The methodology is applicable to the licensee's three Oconee 2568 MWt Babcock & Wilcox (B&W) "lowered-loop" PWRs located 30 miles west of Greenville, SC.

The purpose of the new methodology is to overcome limitations in the existing methodology which dates to the early 1970s. In particular, the new methodology addresses the absence of an FSAR licensing analysis for the containment long-term response to cold leg breaks. The new analyses also address issues relating to LPI cooler degradation, fan cooler fouling, an increase in the ultimate heat sink temperature above the value assumed in the FSAR analyses, and measured containment temperatures in excess of values assumed in the FSAR analyses. The new methodology is not used for the ECCS net positive suction head analysis as that analysis is encompassed by an NSSS vendor generic analysis.

The staff has reviewed the licensee's submittal using the guidance and criteria of: (1) the Standard Review Plan (SRP), Sections 6.2.1, 6.2.1.1.A and 6.2.1.3 and 6.2.1.4, and (2) ANSI/ANS 56.4-1983, "Pressure and Temperature Transient Analysis for Light Water Reactor Containments." ANS-56.4-1983 has not been formally approved by the staff; however, the staff participated in its development and considers its guidance to be appropriate for use in licensing analyses. Both were used by the licensee for guidance.

Based on the licensee's history relating to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions," and previously-approved topical reports, the licensee's technical support organization is competent to develop, verify, validate, use and maintain computer codes for transient and accident analyses. The licensee's response to the Generic Letter was provided as part of topical report submittal DPC-NE-3000, "Thermal-Hydraulic Transient Analysis Methodology," May 1989.

The format and organization of this Safety Evaluation (SE), including the paragraph/section numeration is consistent with that of the topical report.

2.0 OVERVIEW OF COMPUTER CODES

The new Oconee containment analysis methodology makes use of four computer codes; (1) RELAP5/MOD2-B&W, (2) BFLOW, (3) RETRAN-02 MOD5,1DKE, and (4) FATHOMS-RS. RELAP5/MOD2-B&W is used to compute loss of coolant accident (LOCA) mass and energy releases. BFLOW is a licensee-developed code used to compute the long-term break flow quality for a large cold leg break. RETRAN-02/MOD5 is used to compute main steam line break mass and energy releases. Output from these codes is used by FATHOMS-RS to compute containment pressure and temperature responses.

It is noted that the RELAP and RETRAN codes are "best estimate" hydraulic-analysis codes. Conservatism of analyses, using these codes, is provided by the users' judicious selection of initial conditions, code options and boundary conditions (See Section 3.3 for a definition of the term "boundary conditions" as used in this report).

2.1 RELAP5/MOD2-B&W

2.1.1 CODE USE AND DESCRIPTION

RELAP5/MOD-B&W is a modified version of the RELAP5/MOD2 advanced thermal-hydraulic code developed by EG&G Idaho for the NRC staff to provide a tool for licensing analyses of small and large break LOCAs. B&W has modified the RELAP5/MOD2, cycle 36.04 code to include an Appendix K model and to improve code capability and execution. The code models the steady-state and transient behavior of any hydraulic system that may contain a mixture of steam, water, non-condensible gas or nonvolatile solute. The fluid system is modeled by discretizing the system into control nodes joined by junctions. The hydraulic flow field treats the liquid and steam phases as separate fluids in a nonhomogeneous, non-equilibrium manner, solving the mass, energy and momentum equations for each phase. Constitutive relationships are used to define flow regimes and model interphase drag, vapor generation and interphase heat and mass transfer, and horizontal and vertical stratification. Empirical relationships are used to model convective heat transfer, energy partitioning between phases, choked flow and wall friction. Various slip options are available. The code supports simulation of the primary system, secondary system, feedwater train, automatic control system and core neutronics. Component models include reactor point kinetics, pumps, valves, heat structures, heat exchangers, turbines, separators and accumulators.

RELAP5/MOD2-B&W has been reviewed by the staff and is the subject of a safety evaluation (Ref.: Letter from A. Thadani to J. Taylor, dated April 18, 1990). The staff found the code acceptable for use, subject to specified limitations, for calculation of transient response for reload analyses of large and small break LOCAs and operational transients for plants having recirculating steam generators. The staff is currently evaluating its use, for those purposes, for once-through steam generator (OTSG) plants. The fact that RELAP5/MOD2-B&W has not yet been approved for analysis of transient responses for OTSG facilities does not preclude its acceptance for use in calculating mass and energy releases for containment analysis, since the objectives of the analyses are different. In reload transient analysis, the focus is on conservative

analysis of RCS response and fuel performance rather than mass and energy release.

An emergency feedwater (EFW) module has been added to account for steam condensation inside OTSG tubes. The EFW module estimates the tube external area wetted by the EFW. This area is then used to estimate the internal and external heat and mass transfers. The resulting condensation within the OTSG tubes represents an energy exchange between the primary and secondary systems that is particularly significant in the simulation of small break LOCAs where the energy removed by the break flow may be insufficient to remove decay heat from the primary system.

2.1.2 SMALL BREAK AND LARGE BREAK LOCA MODELS

The topical report describes distinctly different RELAP5 nodalizations for large and small break mass and energy release models. With large breaks, large pressure gradients exist during the blowdown phase, resulting in tight coupling between the primary and secondary systems, thereby permitting a reduced level of noding for the steam generators and coolant loops. Instead, large break analyses require emphasis on the reactor vessel nodalization to reflect the in-vessel check valves, core region, and downcomers and replicate the axial progression on the quench front during reflood. For small breaks, there is increased emphasis on the coolant piping and OTSG nodalization to replicate the phenomena resulting from the "candy cane" hot leg configuration, the EFW flow, and boiler-condenser heat transfer. The licensee's modeling has been developed in consideration of these requirements. The OTSGs are modeled with [ ] effect of the EFW flow.

2.1.3 CODE AND MODEL VALIDATION

The RELAP series of codes was used extensively by its developers for both pre-test and post-test predictions in the LOFT and Semiscale tests. For RELAP5/MOD2-B&W, B&W has assessed the code against the Multiloop Integral System Test (MIST) transients (Ref.: "Multiloop Integral System Test - Final Summary," NUREG/CR-5395, Volume 1). The licensee states that the code results were in generally good agreement with the test results. The Oconee model has not been assessed against actual LOCA data due to lack of experimental data. However, the model has been validated against available data from plant transients. The transients include RCP coastdown data from startup tests, discharges of core flooding tanks, natural circulation data from similar plants. The licensee states that these efforts indicated reasonable agreement between code predications and experimental data.

2.1.4 USE OF RELAP5 FOR CONTAINMENT RESPONSE MASS AND ENERGY ANALYSIS

CRAFT-2 was used for the original Oconee licensing analysis described in the FSAR. The Standard Review Plan (SRP) states that the CRAFT-2 code is an acceptable code for calculation of mass and energy releases for use in containment design basis calculations for B&W plants. While CRAFT-2 is specifically referenced in the SRP, the SRP also states that other methods are acceptable if they are found to be conservative for these calculations.

RELAP5 provides an alternative means to perform mass and energy release calculations. RELAP5 thermal-hydraulic analyses take into account fluid momentum changes, wall friction, gravity forces, and geometric head losses due to bends, contractions and expansions, etc. RELAP5 contains the necessary modeling flexibility and computational capability for use in calculations of large break and small break LOCA mass and energy releases for containment analysis. A variety of break flow and slip option models are available to the user. Based on its known capabilities and the licensee's discussion of the code modifications, the staff acknowledges that the modified RELAP5/MOD2-B&W code is capable of generating conservative LOCA mass and energy release calculations. Additional discussions on the use of RELAP5/MOD2-B&W for the Oconee containment analyses are provided in Sections 3.0 and 4.0 below.

2.2 BFLOW

2.2.1 CODE USE AND DESCRIPTION

BFLOW was developed by the licensee as a tool for use in calculating the long-term liquid and steam mass flow rates out of a large cold leg break. A cold leg break at a lowered-loop B&W plant is characterized by continued steaming from the break during the long-term phase. This is because voiding in the core is necessary in order to induce core flow. The steam or steam-water mixture leaving the core enters the reactor vessel upper plenum, passes into the outlet annulus, then passes through internal vent valves into the upper downcomer region. The vent valve flow will be two-phase at higher decay heat levels when steam production is high enough to carry liquid up through the internal vent valves. At lower power levels, the flow is saturated steam. The steam or steam/liquid mixture (depending on power level) then leaves the vessel by entering the broken cold leg, then discharges into the containment where the steam adds heat to the containment atmosphere.

BFLOW models the core power shape, geometry and elevations of the flow volumes and passages. Mass, energy and momentum balances are utilized to compute values for the break flow quality for a wide spectrum of core power, boron dilution flow rate, LPI flow rate and temperature, and containment backpressure conditions. The qualities are used to partition the break mass flow rate into a steam component, which is added to the containment atmosphere and a liquid component which is added to the sump. The total mass flow rate is assumed to be the LPI mass flow rate minus, if any, the boron-dilution line mass flow rate. Additional description is provided in Sections 3.4 and 6.3.

2.2.2 CODE AND MODEL VALIDATION

The staff has not performed a detailed review of the BFLOW model and no actual experimental data are available to assess the results. However, because BFLOW mechanistically models the physical geometry of the reactor vessel and the nodal mass and energy balances (neglecting certain friction and momentum effects due to low flow rates), the staff acknowledges that BFLOW is suitable for use in Oconee long-term large cold leg break mass and energy release computations.

2.3 RETRAN

2.3.1 CODE DESCRIPTION

The RETRAN-02 MOD5.1DKE code is a modified version of the RETRAN-02 MOD005.0 code. RETRAN was developed by Energy Incorporated for the Electric Power Research Institute to provide utilities with a code capable of simulating thermal-hydraulic transients of interest for both PWRs and BWRs. RETRAN-02 can be used to model a general fluid system by partitioning the system into one-dimensional volumes and connecting flowpaths or junctions. The code solves the mass, energy and momentum equations using numerical methods. Although the RETRAN-02 equations describe homogenous equilibrium fluid volumes, phase separation can be modeled by separated bubble-rise volumes and by a dynamic slip model. Heat transfer across steam generators and to or from structures can be modeled. Component models for fans, heat exchangers, pumps, and valves, are available in RETRAN. Control system and trip logic capability are also provided. A general transport model capable of modeling the distribution of boron is included. RETRAN-02 MOD5.0 has been reviewed and approved by the staff for use in the analysis of non-LOCA transients (Ref.: Letter w/Safety Evaluation from A. Thadani to W.J Boatright, dated November 1, 1991).

2.3.2 OCONEE RETRAN MODEL

The licensee's RETRAN models for the Oconee, McGuire and Catawba facilities are described in topical report, "DPC-NE-3000." The staff has reviewed that submittal and issued safety evaluation reports approving their use. The Oconee OTSG model was found to overpredict primary to secondary heat transfer, however, this is not non-conservative the purpose of analysis of containment mass and energy release.

2.3.3 VALIDATION OF CODE AND MODEL

Validation of the Oconee RETRAN model against data from actual secondary break events or experiments has not been performed as there are no data available.

2.3.4 USE OF OCONEE RETRAN-02 MODEL FOR CONTAINMENT RESPONSE MASS AND ENERGY RELEASE ANALYSES

The SRP, Section 6.2.1.4, cites "TRAP-2" (Ref.: Babcox and Wilcox Company, Reference Safety Analysis Report, B-SAR-205, May 1978) as an acceptable code for use in mass and energy release analysis for postulated secondary steam system pipe ruptures. The SRP also states that other methods will be acceptable if they are found to be conservative for these calculations.

The RETRAN-02 code contains suitable modeling flexibility and computational capability for use in producing conservative calculations of steam line break mass and energy releases for containment analysis. Additional discussions on the use of the code is provided in Section 5 below.

2.4 FATHOMS/DUKE-RS

2.4.1 CODE USE AND DESCRIPTION

The FATHOMS/DUKE-RS code is used in the Oconee containment analysis methodology to determine the short and long term containment pressure and temperature response to the mass and energy inputs from high energy primary and secondary reactor coolant system breaks. The FATHOMS/DUKE-RS code is a derivative of the generic FATHOMS code which is, in turn, a derivative of the COBRA-NC/GOTHIC thermal-hydraulic code. The generic FATHOMS code, a product of Numerical Applications, Inc., is capable of modeling all containment types (i.e., ice condenser, large dry, subatmospheric and suppression pool).

To use FATHOMS, a containment is modeled as a network of control cells connected by fluid flow path junctions. FATHOMS solves the mass, energy and momentum equations for multi-component, two-phase flow. Velocity fields are provided for; (1) vapor/non-condensible gases, (2) continuous liquid, and (3) liquid droplet. Temperature fields are provided for (1) vapor/non-condensible gases, and (2) liquid/droplet mixture. (All liquid/droplets within the same volume must be at the same temperature, similarly, all vapor/non-condensibles within a calculational volume must be at the same temperature. However, vapor/non-condensible mixtures may be in thermal non-equilibrium with liquid/droplet mixtures within the same calculational volume.) Passive thermal conductors, flat plate, cylindrical tube or solid rod models, are simulated with finite-difference conduction models. Active heat sources and sinks may also be included in the volumes. Valves, heat exchangers, pumps, spray nozzles and fans may be included in the flow paths.

The licensee's Oconee FATHOMS model does not utilize all of the features available in FATHOMS. Thus, this evaluation does not encompass all of FATHOMS capabilities.

2.4.2 OCONEE FATHOMS MODEL

The significant difference between FATHOMS/DUKE-RS and the generic FATHOMS code is that FATHOMS/DUKE-RS has been modified to include a model to estimate the long-term mass and energy release used to calculate the long-term (post-reflood) LOCA containment response after RCS conditions have stabilized.

In subsequent chapters of this report, the nomenclature "FATHOMS" and "FATHOMS/DUKE-RS" are used interchangbly.

2.4.3 CODE AND MODEL VALIDATION

The topical report states that the FATHOMS code and its predessor COBRA-NC have been extensively benchmarked. To supplement those efforts, the licensee modeled three steam blowdown tests conducted at the Carolina Virginia Tube Reactor (CVTR) and compared the model predictions to the test results. The model predicted CVTR building temperature and pressure responses with good agreement between the code results and test results. Also, Oconee containment response predicted by the licensee's Oconee FATHOMS model was compared to that

of the FSAR analysis performed by the NSSS vendor using the CONTEMPT code. The FATHOMS results compared favorably to the CONTEMPT results.

2.4.4 ACCEPTABILITY OF FATHOMS/DUKE-RS FOR CONTAINMENT RESPONSE MASS AND ENERGY RELEASE ANALYSES

Based on; (a) our review the description of the FATHOMS/DUKE-RS model provided by the licensee, (b) the LOCA validation against the FSAR CONTEMPT analysis depicted in Figure 2.4-2 of the topical report, and (c) the CVTR benchmarking described above, the staff finds the FATHOMS/DUKE-RS code acceptable for use as described in the topical report. Additional information regarding FATHOMS/DUKE-RS usage is provided in Section 6.0 of this report.

3.0 LARGE BREAK LOCA MASS AND ENERGY RELEASE ANALYSIS

3.1 OVERVIEW

Large break LOCA (LB-LOCA) mass and energy release analyses are performed to generate FATHOMS analysis boundary conditions for determining the peak containment pressure following a hot or cold leg double-ended guillotine break (DEGB), and for determining the long-term containment response to a cold leg DEGB. The Oconee RELAP5/MOD2-B&W model was used to generate the mass and energy releases for all but the long-term (i.e., after 30 minutes) segment of a cold leg break for which BFLOW is used. DEGBs were examined for four locations; (1) a hot leg break at the vessel nozzle, (2) a hot leg break at the OTSG inlet, (3) a cold leg break at a RCP discharge, and (4) a cold leg break at a RCP suction. Split breaks and other breaks of lesser discharge were not examined as it is clear that they would produce a lower peak containment pressure. After determining the limiting break location, sensitivity studies were performed to investigate the effect of discharge coefficient selection, offsite power availability (RCP trip), ECCS delays, main feedwater availability, containment backpressure, and vessel refill time. A RELAP5 control system option was used to integrate the mass and energy release for each simulation into a form suitable for input to FATHOMS. For each case examined, the RELAP5 analysis was terminated when the RCS stabilized. BFLOW provides the mass and energy analysis for the remainder of the transient in cases where the analysis is continued further. These studies are described in greater detail in Section 6.0.

3.2 INITIAL CONDITIONS

Initial conditions for LB-LOCA mass and energy analyses were conservatively selected to maximize the energy stored in the primary and secondary systems at the start of the event (i.e., after a converged steady-state solution has been reached). Initial conditions were selected as follows:

Time of life: Time of life was chosen to bound all future reloads and fuel burnups in such a manner as to maximize the combined effects of stored energy (which is highest at BOC) and decay heat (which is highest at end-of-cycle [EOC]). See "core stored energy" boundary condition below.

Initial power level: 2619 Mwt, which is the rated power level with an additional 2% measurement uncertainty was used as the initial power level.

Initial T_{AVG}: 581°F, which is the average coolant temperature maintained by the Integrated Control System (ICS) plus a 2% measurement uncertainty factor was used as the initial T_{AVG}.

Initial RCS pressure: 2185 psig, which is the nominal RCS pressure plus a 30 psi measurement uncertainty allowance was used as the initial RCS pressure.

Initial OTSG pressure: 925 psig was used as the initial OTSG pressure. Turbine header pressure at Oconee is normally maintained at 885 psig. With a 25 psig pressure drop in the steam lines and a 40 psig measurement uncertainty allowance, 950 psig would be the desired initial OTSG pressure. However, in order to initialize a RELAP5 Oconee code run, some parameters must be allowed to vary. The licensee selected OTSG pressure as one such parameter, as it has a minor impact on the mass and energy results.

Initial pressurizer level: At Oconee, high level and high-high level alarms are provided. These alarms are set at 260 and 315 inches respectively. Measurement uncertainty is 25 inches Thus, the first alarm could come in at a level as high as 285 inches. It is assumed the operators would respond to the first alarm in time to maintain true level below 315 inches. Based on these features 315 inches was used as the initial pressurizer level.

Initial secondary coolant mass: [ ] lbm per steam generator was used as the initial OTSG secondary coolant mass inventory. This is the most that can be obtained under initialization conditions and is conservative with respect to the nominal full power steady-state value of [ ] lbm.

Initial CFT liquid volume: 972 ft^3 was selected as the initial core flood tank (CFT) liquid volume, 122 °F as the initial CFT temperature and 655 psig as the initial CFT pressure. Temperature

uncertainty was biased high to maximize the break steaming rate. The pressure was biased upward from the technical specifications limit to account for as-left error and instrument error and to maximize the non-condensible gas discharge. The initial volume is the technical specifications nominal volume biased downward in order to minimize the quantity of injection fluid.

Initial RCS flowrate: 366,080 gpm (104% of design flow) is used as the initial RCS flow rate. This value maximizes T_{HOT} and is conservative with respect to RCS stored energy.

The initial conditions selected by the licensee, and described above, have been selected in a manner consistent with the conservative guidance of ANS-56.4-1983, paragraph 3.2.2 to maximize the amount of energy available to be released to the containment. Accordingly, they are acceptable for use in the LB-LOCA containment response mass and energy release analyses.

3.3 BOUNDARY CONDITIONS

Boundary conditions are; (1) the energy sources that transfer mass and energy to the modeled system during the event, and (2) assumptions that specify how the mass and energy are distributed and/or how the components will respond during the event. Boundary conditions are specified by the code user for parameters governed by conditions outside of the problem boundaries. The code imposes the boundary conditions on the system model at the beginning of the transient after the initial conditions have been established.

3.3.1 COMMON BOUNDARY CONDITIONS

The licensee's LB-LOCA containment analysis methodology involves the following boundary conditions that are common to both peak pressure analyses and the RELAP5 segment of the long-term release analysis. (Different boundary conditions are used for peak pressure and long-term analyses in cases where use of a common boundary conditions would be unrealistic or non-conservative for either analysis.)

3.3.1.1 ENERGY SOURCES

When calculating LB-LOCA mass and energy releases for containment response analyses, the analyst considers all significant sources of mass and energy that may be added to the containment. The topical report identifies the following energy sources used as common boundary conditions in the calculation of LB-LOCA mass and energy releases for containment peak pressure analyses and long-term response.

RCS inventory: An initial pressurizer level of 315 inches, which represents an increase of approximately 2% above the normal operating RCS inventory was selected as a conservative boundary condition. Thermal expansion of the RCS volume, from the cold to hot condition, was accounted for increasing the calculated volume of the RCS piping by 1%.

Secondary system inventory: The physical volume of the secondary system was not increased to reflect thermal expansion. Instead, the main feedwater flow boundary condition was chosen so as to conservatively represent the secondary inventory. See 3.3.1.2 "OTSG level" below.

RCS and OTSG metal heat structures: The metal heat structures (e.g., reactor vessel and internals, RCS piping, OTSG) are initially assumed to be in thermal equilibrium with the surrounding coolant, each with a constant initial temperature distribution. This condition maximizes initial stored energy.

Core stored energy and decay heat: Core stored energy is a function of initial fuel average temperature which in turn is highly dependent on burnup. It is highest at beginning of cycle (BOC). [ ] is used as the initial average fuel temperature. This value is based on the weighted linear heat rate in each of three core regions and is consistent with the ECCS analyses. Fission product decay heat and actinide heat input are based on ANS-5.1-1979 with 2σ uncertainty allowance. These heat inputs are greatest at EOC. A time in life is selected that maximizes the combination of core stored energy and decay heat contributions to containment mass and energy releases.

Metal-Water reaction heat: A unique value, as discussed in 3.3.2.1 and 3.3.3.1, is used for each analysis.

Main steam isolation: The turbine stop valves are assumed to close instantaneously when the break occurs. The turbine bypass valves remain closed. All steam relief valves are assumed operable.

Main feedwater supply: Main feedwater, in cases where it is not lost due to loss of offsite power, is assumed to be under the post-trip control of the MFW level controller and thus responsive to OTSG level changes.

CFT nitrogen expansion: The nitrogen mass in the CFTs is assumed to directly enter the containment at at time calculated by RELAP to be equivalent to CFT depletion.

ANS-56.4-1983, para. 3.2.1 provides guidance for the identification and selection of conservative energy source conditions for calculation of mass and energy releases used on containment response to LOCAs. The above selections conform to the ANS guidance and are acceptable.

3.3.1.2 COMMON ASSUMPTIONS USED IN MASS AND ENERGY ANALYSES FOR CONTAINMENT PEAK PRESSURE AND LONG-TERM RESPONSE

Assumptions used in both the peak pressure and long term mass and energy release calculations are identified below.

Axial power distribution: A chopped cosine distribution with a [ ] has been shown to result in the greatest quantity of steam exiting the break and is thus assumed for the LB-LOCA analyses.

Borated Water Storage Tank (BWST): The ECCS and spray pumps draw from the BWST during the injection phase of a LOCA. Prior to the BWST becoming depleted, suction is switched to the containment sump. The sump is warmer and the switchover thus results in reduced heat removal from the RCS and containment atmosphere. For conservatism, the minimum allowable BWST level of 46 feet, plus a 20.2-inch measurement allowance is assumed. A relatively high BWST temperature of 115 °F is assumed.

Main feed water (MFW) temperature: The nominal full power MFW temperature at Oconee is 453°F. The uncertainty of this measurement is 4°F. The licensee has selected a MFW temperature of 460°F as a boundary condition. Also, the temperature is held constant through the transient for additional conservatism although the bleed steam supply to feedwater heaters is lost when the turbine trips. These assumptions increase the secondary system energy contribution and are conservative.

Emergency feedwater (EFW) temperature: For the large break peak pressure analyses, EFW (for sequences in which it is available), is assumed to be at 120°F.

Limiting single failure: Consistent with the guidance of ANS-56.4-1983, paragraph 3.2.3, the licensee evaluated the potential effects of various single active failures on containment long-term response. [Note: The short-term peak pressure response is relatively insensitive to ESF single-failures due to the fact that the pressure peaks quickly, before the containment cooling systems become fully effective.] Failure of a 4160 VAC switchboard was determined to be the single failure that disables the greatest number of containment cooling components and was assumed as the limiting single failure.

OTSG level: If offsite power is available, the MFW system is available and used to control OTSG level. Following reactor trip, the boundary condition assumes that level is controlled at 50% + 10.5% measurement error on the operating range with flow directed to the upper headers, and MFW temperature is 460°F. These assumptions are conservative, as they maximize the available amount of secondary energy available for heat transfer to the blowdown effluent.

The above assumptions are appropriately conservative boundary conditions for short-term (i.e., peak pressure) and long-term mass and energy release calculations.

3.3.2 LARGE BREAK LOCA PEAK PRESSURE MASS AND ENERGY RELEASE BOUNDARY CONDITIONS

Boundary conditions applicable to the LB-LOCA peak pressure analyses, but not common to the LB-LOCA RELAP5 long-term mass and energy release analyses are discussed in this section.

3.3.2.1 ENERGY SOURCES

The boundary conditions identified below describe the energy sources applied to the RELAP5 large-break LOCA mass and energy release analyses used for containment peak pressure response analysis. These boundary conditions supplement those for the common energy sources described in 3.3.1.1 above.

Fission heat: Fission heat calculations are generated by the RELAP5 built-in point kinetics model for the blowdown and reflood phases. The reactivity value used is computed by collapsing weighted nodal reactivities into a single (point) value. Feedback mechanisms for each node include the effects of moderator density, Doppler and boron. Control rods are assumed to remain out of the core.

For the blowdown phase, the combined reactivity feedback effects of the moderator density negative reactivity contribution and Doppler positive reactivity are based on analyses which bound previous and future reloads from beginning of cycle to end of cycle.

During refill boron concentration changes rapidly. For reactivity feedback during refill, a net reactivity function (a function of moderator density, boron concentration and fuel temperature) is developed from the known lower reactivity limit for a voided core at the end of blowdown, a calculated upper limit for the all-rods-out 1% shutdown margin condition, and intermediate points on a third order polynomial fitted to arbitrarily-selected intermediate points.

Metal-water reaction: The RELAP5/MOD2-B&W code includes a user-selectable metal-water reaction model which can be used in a manner similar to an ECCS evaluation model. However, remodeling of the core using this model is unnecessary for puposes of containment mass and energy analyses. The containment analysis methodology uses an alternative approximation methodology based on the results of ECCS analyses performed by B&W and applicable to B&W 177 Fuel Assembly Lowered-Loop B&W facilities. For a cold leg break, the method assumes a[

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earlier. In both cases the energy is assumed to be distributed in an axial profile consistent with neutron power and decay heat.

The above energy sources, along with the common energy sources described in paragraph 3.3.1.1, encompass the energy sources enumerated in ANS-56.4-1983, paragraph 3.2.1, which need be accounted for in conservative calculations of mass and energy releases for the analysis of containment peak pressure. The staff did not closely examine the refill reactivity model or the metal-water reaction approximation methodology described above. However, the staff notes that these simplified models are fundamentally sound in principle, and suitable for mass and energy release calculations for containment peak pressure analyses.

3.3.2.2 ASSUMPTIONS

Assumptions common to RELAP5 large break peak pressure and long-term analyses are described in 3.3.1.2 above. Additional assumptions specific to the peak pressure analyses are described below.

Break location: Four large breaks, two hot leg breaks and two cold leg breaks, were examined. These include hot leg breaks at the vessel nozzle and at the OTSG inlet, and, for the cold leg, one at the RCP discharge and one at the RCP suction. For the hot leg break vessel outlet case, core stored energy release is maximized and moderator density reactivity feedback is maximized, whereas OTSG energy contribution is reduced. [ ] Also a Loop A break is assumed in order to maximize the pressurizer fluid contribution to the break. For the hot leg break at the OTSG, secondary system energy contribution is maximized. These two locations are assumed to bound all intermediate hot leg locations.

For the cold leg case, breaks at the RCP suction and discharge in the A1 loop are assumed to bound all cold leg cases. Loop A1 is chosen because the pressurizer is attached to that loop. [ ]

Break flow model: RELAP5/MOD2-B&W has available two mutually exclusive choked flow models. The first option is its built-in Ransom-Trapp model. The other option permits use of table interpolation with any of four correlation tables [i.e, extended Henry-Fauske model (subcooled flow), Moody model (two-phase flow), homogenous equilibrium no-slip model (HEM), or Murdock-Bauman (superheated flow)]. The method employed in the Oconee methodology is to determine the differences between the Henry-Fauske and Moody correlations and the Ransom and Trapp model and adjust the flow coefficients as necessary to bound the Henry-Fauske and Moody correlations. (The suitability of this method was confirmed by sensitivity studies as described in Section 6.0 below.)

ECCS injection: An ECCS injection delay time of 15 seconds is assumed to be bounding for scenarios with offsite power available. For scenarios with loss of offsite power and failure of a 4160 VAC bus, a delay of 35 seconds is assumed. Low pressure injection is assumed to be constant at 3000 gpm plus 311 gpm instrument error. The positive allowance for instrument error is conservative due to the fact that it results in an increased reflood rate and accelerated energy release to containment. High pressure injection flow is represented as a function of RCS backpressure. All injection flow is modeled as being injected low in the vessel downcomer to conservatively minimize steam condensation. For the RCP discharge line break case, all HPI flow into the broken line is assumed to be spilled directly into the containment.

Reactor coolant pumps: Except for the loss of offsite power cases, the RCPs are assumed to continue operating. This conservatively maximizes secondary-to-primary heat transfer and pump heat. Degradation of pump capacity under two phase flow conditions is modeled using Semiscale and pump vendor data.

Emergency feedwater system: For loss of offsite power cases, MFW pumps are lost, coasting down in 10 seconds. At 17 seconds into the event, EFW flow is established to the OTSG upper header. For conservatism, only the steam-driven EFW pump, feeding both OTSGs, is assumed available although using the single-failure criterion one motor-driven MFW pump would also be available. The reduced EFW flow is conservative because relatively cold EFW flow absorbs primary system heat resulting in less break flow energy.

Containment backpressure sensitivity: RELAP5 is not iteratively coupled to FATHOMS in a manner that would enable a calculated backpressure to be used in the mass and energy calculation. Instead, a conservatively high backpressure (i.e., 59 psig, the containment design pressure) is assumed. Section 6.2.6 discusses backpressure sensitivity studies.

Refill phase modeling sensitivity: Vessel refill time (the period between the end of blowdown and the beginning of reflood) has typically, in the past, been considered to be zero seconds. The new analyses assume a 10-second refill period. A sensitivity calculation using a 0-second refill time was performed as described in Section 6.2.6, and confirmed that refill time has no effect on peak pressure response.

The above assumptions are conservative for purpose of analysis of LB-LOCA mass and energy releases for use in calculation of containment peak pressure response.

3.3.3 RELAP5 LARGE BREAK LONG-TERM MASS AND ENERGY RELEASE

Section 3.3.3.1 through 3.3.3.4 below discuss boundary conditions applicable to long-term LB-LOCA mass and energy analyses, but not to the peak pressure LB-LOCA mass and energy analyses.

3.3.3.1 ENERGY SOURCES

Fission heat: For the blowdown, refill and reflood phases, fission heat is calculated as described in 3.2 above. For the long-term segment after reflood and prior to achievement of quasi-steady state conditions, a less rigorous fission heat energy source boundary condition is used. A fission power history from a suitable large break Appendix K LOCA analysis was used, the results of that analysis being normalized to reflect the necessary power level and axial peak conditions appropriate for the new Oconee analyses. The licensee has validated this method against results using the peak pressure analysis method described above, and confirmed that the results are comparable for long-term mass and energy release.

Metal-water reaction: Exothermic heat from metal-water reaction is neglected for the long-term analyses. The licensee determined that the maximum expected energy release to containment from metal-water reaction in the first 30 minutes (RELAP portion) would be approximately 0.56% of the total energy released to containment during that period. Although it is non-conservative to neglect a heat source, this amount is so insignificantly small as to not warrant modeling for the long-term analysis.

These energy sources, along with the common energy sources described in 3.3.1.1 above, encompass all energy sources which must be considered in the analysis of large break long-term mass and energy releases.

3.3.3.2 ASSUMPTIONS

Assumptions utilized in the RELAP large break long-term mass and energy release analyses are discussd below.

Break location: Only one break size and location is analyzed for long-term containment response - a double-ended guillotine break (DEGB) at the A1 cold leg discharge. This selection is based on a consideration of the RCS geometry. An RCS cold leg pump discharge break will continue to steam during the long-term, whereas breaks at other locations will become flooded. This steaming results in a more severe long term containment temperature response notwithstanding the use of containment spray and containment coolers. [The current (original) licensing analyses did not analyze cold leg breaks for long-term response.]

Break flow: The break flow model assumed for the DEGB long-term analysis is the RELAP5 Ransom and Trapp model. This assumption is

based on the long-term analysis being insensitive to blowdown phase dynamics and break size.

Borated Water Storage Tank (BWST) depletion - ECCS and containment system flow rates: High ECCS pump flow rates are intentionally established in order to accelerate BWST depletion. Early BWST depletion results in early switchover to the sump recirculation phase. The higher injection temperature (less subcooling) of sump fluid increases the steam release to containment and thus provides a more conservative analysis. The higher ECCS flow rates are established by assuming a 4160 VAC switchgear failure and manual operator action (per guidance of Emergency Operating Procedures [EOPs]) to cross connect HPI trains and throttle containment spray flow. For additional conservatism, it is assumed that the HPI line serving the broken loop has also broken in such a manner as to preclude steam condensation on the spilled injection fluid. Conservative-direction allowances are included to account for possible instrument errors.

Reactor coolant pumps: For cases where offsite power is not lost, RCPs are assumed to be manually tripped two minutes after the break. This assumption is based on EOP requirements to trip the pumps upon loss of subcooling.

Emergency feedwater system: Because MFW temperature is warmer than EFW temperature, it is conservatively assumed that MFW pumps are used to maintain OTSG levels.

Containment backpressure: Containment temperature and pressure conditions for the long-term OEGB analysis are assumed to follow time-dependent profiles which are bounding on the high side. The profiles are based on a FATHOMS analysis which assumes operation of two RBCUs, one train of RBS, and one LPI cooler. The pressure values bound the high side of the FATHOMS response. The temperature profile is chosen so as to specify a slight superheat in order to reduce code instabilities due to the vacuum-creating effect of steam condensation. As noted in ANS-56.4-1983, paragraph 3.2.4.6.1, a conservatively high backpressure assumption is acceptable if the mass and energy release analysis is not coupled to the containment analysis in such a manner that a calculated containment backpressure is available.

The above modeling assumptions are consistent with the guidance of ANS-56.4-1983, paragraphs 3.2.3 through 3.2.5, and are conservative for the purpose of analysis of long-term mass and energy releases from large breaks.

3.3.3.3 BREAK FLOW OSCILLATIONS

Toward the end of blowdown, there is a potential for code instability. The high mass flux inertia of the break flow can cause fluid to continue exiting the break after the RCS pressure has dropped below the containment backpressure. A period of reverse flow, followed by additional, damped

oscillations may then occur. Modeling techniques attempt to minimize the number and severity of these oscillations. These measures include; (a) injecting ECCS flow into the vessel downcomer below the equilibrium liquid level, (b) time-averaging of heat transfer coefficients, and (c) holding the in-vessel vent valve open at a constant value and varying the downstream volume. Measures (b) and (c) are implemented only after quasi-steady state conditions are present, and in such a manner as to ensure that the resulting mass and energy release profiles are bounding.

3.3.3.4 HEAT STRUCTURE STORED ENERGY

Whereas a RELAP5 model can account for the energy stored in heat structures, and the heat transfer between the heat structures and fluids, BFLOW, which assumes quasi-stable conditions, cannot. Since RELAP5 is only used for the first segment (not beyond 30 minutes) of the cold leg DEGB mass and energy analysis, a means is necessary to account for, in the BFLOW portion of the analyses, the stored heat remaining at the termination of the RELAP5 portion of the analysis. To facilitate this action, RELAP5 heat structures are grouped.

Each RELAP5 heat conductor is grouped into[ ]from FATHOMS heat structures modeled as slabs based on nominal values for volume, initial temperature, specific heat, density and overall heat transfer coefficient. The FATHOMS heat slabs are modeled to be geometrically similar to the RELAP heat slabs. The total stored energy in each group is calculated by a RELAP5 control system and the results provided to FATHOMS. This technique provides a means of accounting for heat structure stored energy release rates while performing the analysis in separate segments as further described in Section 6.0.

3.4 BFLOW LONG-TERM MASS AND ENERGY RELEASE

As discussed in Section 2.0 above, the RELAP5 model calculates the mass and energy release from the assumed break for approximately the first 30 minutes following the break. For the subsequent period, FATHOMS obtains the break boundary conditions from a set of subroutines which utilize BFLOW results. BFLOW results are made available to FATHOMS in the form of[ ]of the four key parameters noted in Section 2.2 (i.e, containment pressure, reactor power, LPI flow, and LPI subcooling). At the end of the RELAP5 period, the FATHOMS subroutines use these four parameters to obtain from the matrices the break

quality and boron dilution flow for each time step. Through mass and energy balances, this information is sufficient to enable calculation of the liquid and vapor break flow rates and enthalpies. The vapor flow is added to the containment atmosphere and the liquid flow to the containment sump. [

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3.5 LARGE BREAK LOCA PEAK PRESSURE MASS AND ENERGY RELEASE RESULTS

The licensee analyzed thirteen LB-LOCA cases to investigate the sensitivity of peak containment pressure to break location, critical flow model, offsite power assumptions, containment backpressure and refill time interval. The limiting case was determined to a DEGB of the hot leg at the OTSG inlet, with offsite power available, using the Ransom and Trapp critical flow model with flow coefficient adjusted to match the Appendix K evaluation model. The licensee's methodology provides integrated mass and energy release results which are in good agreement with that of the NSSS vendor. The critical flow model sensitivity studies confirmed the validity of using the Ransom and Trapp model in conjunction with critical flow discharge coefficient multipliers to increase the mass flow from the Ransom and Trapp model to those of the Henry-Fauske/Moody (Appendix K) models. See further discussion in Section 6.2.6 below.

3.6 RELAP5 LONG-TERM MASS AND ENERGY RELEASE

RELAP5 is used to calculate large cold leg break mass and energy release for the short-term segment, and for the long-term segment until quasi-steady state conditions are established. As discussed in Section 3.3.3.2, a Loop A1 cold leg RCP discharge line break is modeled for the long-term analysis and is assumed to be the bounding LB-LOCA for long-term considerations. Cold leg break phenomena results in a 26-second blowdown phase. Refill occurs in approximately 35 seconds and reflood is complete at about 200 seconds.

The instantaneous mass and energy release data computed by RELAP5 are integrated and smoothed by a utility program which puts them in a form which facilitates use by FATHOMS.

4.0 SMALL BREAK LOCA MASS AND ENERGY RELEASE ANALYSIS

4.1 OVERVIEW

An SB-LOCA is considered to be a LOCA with a break size less than 0.5 ft^2 and greater than the capacity of the normal makeup system. The normal makeup system at Oconee has the capacity equivalent to a 0.0008 ft^2 break. SB-LOCAs do not challenge the containment from the standpoint of internal pressure stresses in the manner of large break LOCAs, but are investigated to verify that the post-accident containment heat removal systems have adequate capacity to assure that post-accident containment pressure is maintained at or below assumed values. Equipment environmental qualification is the limiting consideration. A deficiency in the current Oconee licensing basis is that

only a single break (i.e., 5.0 ft^2 large hot leg break) was analyzed for long-term response. The new DPC-NE-3003-P methodology examines a spectrum of small breaks to determine the limiting break for long-term containment response. Six cases having break areas from 0.0025 ft^2 to 0.10 ft^2 were considered.

For the small break containment analyses, the mass and energy boundary conditions, up to the time that the RCS has been cooled to the extent possible with the OTSGs, are calculated with RELAP5. For the subsequent period, wherein all decay heat removal is accomplished by the break flow, a simplified FATHOMS model is used to determine the mass and energy release.

This section discusses the derivation of the mass and energy release data. The containment responses are discussed in Section 6.4.

4.2 INITIAL CONDITIONS

The initial conditions for the SB-LOCA analyses are identical to those of the LB-LOCA described in Section 3.2 above.

4.3 BOUNDARY CONDITIONS

Boundary conditions for analysis of SB-LOCA mass and energy releases include the energy sources and assumptions described below.

4.3.1 ENERGY SOURCES

The SB-LOCA analyses consider the same energy sources as the large break LOCAs (Ref. Section 3.3.1.1 above) with the exception of metal-water reaction. Metal-water reaction need not be considered in the SB-LOCA because fuel cladding temperatures remain below the threshold for significant reaction.

Fission heat is calculated using the RELAP5 kinetics model in a manner similar to that of the large break peak pressure analysis described in Section 3.3.2.1 above, but the negative reactivity from boron in the ECCS injection fluid is not considered. The reactor is assumed to trip when the first trip setpoint condition is met. The rods begin to drop after an appropriate delay.

4.3.2 ASSUMPTIONS

This section discusses assumptions used in the SBLOCA analyses that differ from those of the large break mass and energy release analyses discussed in 3.3.2.1 above.

Break size and location: As noted above, a spectrum of SB-LOCA analyses (six sizes at one location) were analyzed to identify the most limiting mass and energy release. Each case assumed a split rupture in the A1 cold leg RCP discharge piping. This location was selected because it would result in the least core cooling and greatest steam release to containment for the given break size.

Break flow model: Break flow is calculated using the RELAP5 built-in Ransom and Trapp critical flow model with an assumed break discharge coefficient of 1.0. This is acceptable in lieu of a more precise method for each case, because the potential mass and energy release errors for each case are accounted for by the fact that a sufficiently broad spectrum of break sizes is analyzed.

Reactor building, ESF flow, and BWST depletion: For SB-LOCA case studies, a reactor building model is included in RELAP5 for estimating containment-related boundary conditions which may impact the mass and energy release calculations since FATHOMS is not coupled to RELAP5. The model is similar to the FATHOMS model and includes appropriate representations for; (a) heat structures for the building cylinder, dome, internal concrete, and internal steel, (b) fan coolers, (c) containment spray, and (d) LPI coolers. For the smaller breaks, the reactor building model is used in the calculation of when sump recirculation switchover occurs. For the larger SB-LOCA cases, the event terminates before recirculation is aligned. HPI, LPI and RBS system boundary conditions are chosen to reflect the instrumentation setpoints, single-failures, instrument uncertainties, EOP operator actions, pump performance, etc. as appropriate for each case to generate a correct sump mass and temperature profile.

RCPs: A 4160vac switchgear failure is the limiting single failure. Thus offsite power is assumed to be available. RCPs are conservatively assumed to trip when the reactor trips. This is conservative because the OTSGs are less effective in removing primary system heat.

OTSG level control: The MFW system is used to initially control OTSG level since off-site power is assumed to be available and because it is hotter. If subcooling margin is reduced, the EFW system is used to raise the level per EOPs, after a 10 minute delay.

EFW system: One turbine-driven EFW pump is assumed to be available for the operator to feed both OTSGs as necessary. EFW is injected into OTSGs through the upper header as noted earlier in Section 2.1.1.2.

OTSG pressure control: For the first 30 minutes, it is assumed that pressure is maintained by the relief valves. Secondary cooldown is then begun using ADS valves if the main condenser is unavailable.

Hot leg high point vents: The hot legs are provided with high point vents for venting non-condensibles and for use in restoring natural circulation. These vents are assumed to be closed at all times except for one case. The one case is at 30 minutes into the 0.005 ft^2 break when EOP conditions are met for use of the vents.

4.4 RESULTS OF SIX-CASE SPECTRUM OF SB-LOCA ANALYSES

For the two smallest of the six cases considered, it was clear that they were not limiting for containment response. Primary-to-secondary heat transfer can be maintained and an orderly cooldown accomplished. For the larger breaks, the RCS will depressurize to permit LPI and break cooling is sufficient. For the intermediate size breaks OTSG heat transfer can be interrupted and break cooling is limited. Breaks in this range are thus limiting.

5.0 STEAM LINE BREAK MASS AND ENERGY RELEASE ANALYSIS

5.1 OVERVIEW

In addition to primary coolant system breaks (i.e., LOCAs), containment responses to secondary system pipe ruptures are also postulated and analyzed. A main steam line break in containment presents a significant pressure and temperature challenge to the containment, possibly more limiting than the worst-case LOCA.

As noted in Section 2.3 above, the DPC-NE-3003-P methodology, uses a RETRAN-02 model to predict the mass and energy release for a MSLB. The mass and energy release calculated by RETRAN-02 is input to FATHOMS to determine the containment pressure and temperature response. This section discusses the mass and energy analyses. The containment response is discussed in Section 6.5.

5.2 THERMAL-HYDRAULIC ANALYSIS

5.2.1 MODIFICATIONS TO OCONEE BASE PLANT MODEL

The Oconee RETRAN-02 "base plant model" is a model developed by the licensee for reload transient analyses. That model is described in "Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, July 1987."

The RETRAN-02 containment mass and energy release analysis model differs from the base model in that it has certain provisions to; (a) model the break, (b) include containment backpressure, (c) include asymmetry and thermal mixing in the vessel, and to (d) add additional detail for condensate and feedwater piping systems as described below.

> Renodalization of reactor vessel: A MSLB produces an asymmetrical effect across the reactor core. In order to model the asymmetrical effects, the RETRAN base model is renodalized to

Condensate feedwater model: A condensate/feedwater system model is added to the RETRAN base deck to model the feedwater pumps, condensate pumps, feedwater heaters and feedwater heater drain pumps.

5.2.2 BREAK MODEL

A DEGB of the 34-inch diameter main steam line is modeled using the Moody critical flow model. The Moody model is appropriate for two-phase critical flow conditions. The licensee performed sensitivity studies which indicate that the 34-inch MSLB containment pressure and temperature response bounds smaller breaks.

5.3 INITIAL CONDITIONS

Initial conditions are chosen to maximize the stored energy in the primary and secondary systems. A summary of the initial conditions is provided below.

Power level: For purposes of containment analysis, higher power level is more conservative than lower power due to the larger initial OTSG inventory, higher feedwater flow rate and increased decay heat. Accordingly, an initial power level of 100%, plus 2% power measurement uncertainty, is assumed.

$T_{(AVG)}$: $T_{(AVG)}$ is normally controlled at 579°F. This, plus an additional 2°F added for uncertainty, results in 581°F being used as an initial condition.

RCS pressure: RCS pressure is normally maintained at 2155 psig. This, plus 30 psi for uncertainty, results in 2185 being used as the initial RCS pressure.

OTSG pressure: Turbine header pressure is normally maintained at 885 psig. Pressure loss between the OTSG and turbine header is 25 psi. The pressure measurement uncertainty is 40 psi. Based on these values, 950 psig would be the appropriate initial condition for OTSG pressure. However, 910 psig is a realistic pressure which provides for proper RETRAN initialization with the required steady-state balance, and is acceptable.

Pressurizer level: An initial pressurizer level of 245 inches is used, based on a normal level of 230 inches plus 25 inches to bound uncertainty.

Secondary system inventory: An upper bound total OTSG inventory of [ ] that the FSAR assumption value of 62,600, but has been analyzed by the licensee and determined to be a conservative value.

Initial conditions for time-of-life, boron and other core parameters are discussed under boundary conditions below.

5.4 BOUNDARY CONDITIONS

RCS inventory: The RCS inventory is the calculated volume of the RCS plus 1% to account for expansion between the cold and hot conditions. Zero OTSG tube plugging is assumed.

Primary system structures stored heat: Heat conductors are included in the RETRAN model for all structural metal in contact with the primary coolant. These heat conductors are assumed to be in equilibrium with the coolant and to have a uniform temperature.

Secondary system structure stored heat: Secondary structures are assumed to be in equilibrium with the surrounding coolant and to have a uniform temperature.

Core stored energy: Although core stored energy for a given power level is greatest at beginning of cycle (BOC), end-of-cycle conditions are used for the MSLB mass and energy analysis, because of the higher return to power resulting from the greater EOC negative moderator coefficient.

Fission heat: Fission heat is calculated by the RETRAN point kinetics model. The option employed uses one prompt neutron group, six delayed neutron groups, eleven delayed gamma emitters, plus U-239 and Np-239. A low effective delayed neutron fraction and prompt neutron lifetime are selected to maximize the reactivity addition rate. Reactivity feedback for moderator density and Doppler are specified by reactivity feedback tables. The most negative curves are tabulated.

Scram reactivity assumes a top-peaked power distribution and lower bound rod insertion times. These assumptions are non-conservative, but sensitivity analyses indicate the assumptions have little impact on the mass and energy release analyses. The most negative control rod is assumed to fail to insert.

Consistent with the assumed EOC condition, boron concentration is assumed to be zero. The negative reactivity inserted by boration is averaged for the six nodes to obtain a core average boron concentration. The core average concentration is multiplied by boron worth to give reactivity.

A high differential boron worth (i.e., 120 ppm/%$\Delta k/k$) is used to minimize the negative reactivity added by HPI and core flood tank discharge.

Decay of actinides and fission products: The ANSI/ANS-5.1-1979 decay heat values for EOC conditions are used.

Main steam line isolation: The turbine stop valves are assumed to trip closed in one second upon reactor trip. Closure of the turbine stop valves isolates the OTSGs from each other. The one second response time is a technical specification requirement.

Condensate and feedwater model: The feedwater/condensate model was previously noted. This model explicitly calculates the feedwater flow rate during the accident. The flow rate is based on pump curves and transient pressure response. Two booster pumps, two D-Heater drain pumps, and two MFW pumps are assumed operating. The third condensate booster pump is on automatic standby. No credit is taken for low suction pressure trips or for MFW pumps trip on high OTSG level. Condensate booster and drain pumps are assumed to trip with a 10-second coastdown if their suction inventory source is depleted.

The main feedwater heaters are supplied with extraction steam to heat the feedwater. The feedwater is heated from about 80°F to 455°F during normal operation. A main feedwater temperature of 458.3°F is used initially. These values provide a code initialization with secondary side heat removal equal to reactor power. When reactor trip occurs, extraction steam is no longer available and condensate temperature decreases. This is accomplished by adjusting the heat input from the feedwater heaters to zero immediately after the break occurs.

Emergency feedwater system: The Oconee EFW system includes three pumps, two motor-driven (one for each OTSG) and one shared steam turbine-driven. Flow from the shared pump is assumed to be discharged to the faulted OTSG. The pumps start automatically on loss of both MFW pumps or low level in either OTSG. EFW water temperature is assumed to be 120°F.

RCS flow rate: A high RCS flow rate maximizes primary-to-secondary heat transfer and is thus conservative. An RCS flow rate of slightly greater than 102% of the nominal best-estimate steady-state flow rate is assumed. This value corresponds to 115% of the design flow rate.

ECCS: The HPI system is modeled using a fill table with one HPI pump injecting through one train. This is conservative, since, in an emergency, all three pumps start, injecting into both loops, but use of one pump results in a higher return to power. HPI initiation is assumed to occur at 1480 psi, reflecting a 1600 psig setpoint with 120 psi measurement uncertainty. Boron concentrations in the BWST and CFTs are assumed to be at the Technical Specifications lower limits. The CFT temperatures are assumed to be 120 °F. The boron concentration in piping purge volumes (piping between CFTs and the RCS and between BWST and the RCS) is assumed to initially be 0.0 ppm.

Limiting single-failure: The failure of one 4160 VAC switchgear

is the limiting single failure for the MSLB analyses. This results in loss of one train of HPI and one train of LPI. The licensee plans to perform another single-failure analysis to address a new feedwater isolation system design. At present, as an interim measure, the licensee is being permitted to rely on, manual operator action for feedwater isolation.

Reactor protection system (RPS) trips: A delayed reactor trip is a conservative assumption for containment analysis. Accordingly, the variable low pressure trip setpoint assumptions include a 40 psig uncertainty allowance and 0.7 seconds delay time, and 5.04 second $T_{(HOT)}$ temperature sensor lag. The low RCS pressure trip is also modeled. The actual setpoint is 1810 psig. In the analysis, a setpoint of 1780, with a 0.5 second delay is assumed to provide conservatism and account for instrument uncertainty.

Integrated control system (ICS): The control rods and turbine control valves are assumed to be in manual control. For the rods, this is conservative since initial rod motion would otherwise be in the insert direction due to flux instrumentation errors resulting from downcomer temperature changes. For the turbine control valves, the manual control assumption is also conservative. The MFW flow is assumed to rapidly drop to zero within a few seconds after reactor trip reflecting the ICS response. After OTSG level drops to 25 inches, a RETRAN control system models the action of the low level controller to maintain a 25-inch level.

Loss of offsite power: MSLB is analyzed for both the loss of offsite power case and no loss of offsite power case, since it is not obvious or intuitive which case is more limiting for containment temperature or pressure responses.

Containment backpressure: A low backpressure is assumed in the RETRAN analyses. This provides a greater break flow, and is therefore conservative.

Liquid carryout and superheat: For conservatism, the break flow is adjusted such that the flow is[

]

The initial conditions, modeling assumptions and boundary conditions described above are consistent with the guidance of ANS-56.4-1983, paragraphs 3.3.2

through 3.3.4, and provide conservative results in the analysis of steam line break mass and energy releases.

5.5 RESULTS OF MSLB MASS AND ENERGY RELEASE ANALYSES

Mass and energy releases were calculated for a 6.3 ft^2 DEGB of the 24 in^2 MSL for the offsite power maintained conditions and for the loss of offsite power condition. In both cases operator action is credited for isolation of the affected OTSG at 170 seconds. (Note: Modifications are planned to provide for automatic feedwater isolation for the faulted OTSG.) These releases were input to FATHOMS for analysis of the containment response. The results are provided in Section 6.5 below.

6.0 CONTAINMENT PRESSURE AND TEMPERATURE RESPONSE ANALYSES

6.1 OVERVIEW

As discussed in Section 2.4 above, the FATHOMS/DUKE-RS (FATHOMS) code is used to determine the containment responses to the mass and energy releases predicted by RELAP5, BFLOW, and RETRAN-02 for postulated high energy lines breaks inside containment. The RELAP5 mass and energy release data for each time step are integrated by a RELAP5 control system, then averaged by a utility program which prepares a data file for use by FATHOMS. BFLOW results are supplied to FATHOMS in the form of matrices giving the quality of the flow exiting the break as a function of decay heat, containment pressure, LPI flow and LPI subcooling. RETRAN data are provided to FATHOMS in a manner similar to that of RELAP.

Each FATHOMS case is run in one or more time segments, each segment representing a different phase of the event. Peak pressure analyses and MSLB analyses are of short duration and therefore may be run in one segment. Long-term large break LOCAs and SB-LOCAs are run in three segments.

A large number of break cases must be examined. For PWRs, it is not clear or intuitive what primary or secondary system pipe rupture event will produce the most limiting peak containment pressure, or harshest temperature response for safety-related equipment. The results of the LB-LOCA analyses are examined carefully for both pressure and temperature responses to determine if the highest peak pressure is bounded by the containment design pressure, and the temperature response is bounded by the EQ envelope. For SB-LOCAs it is recognized that the pressure response is not likely to be limiting; however, it is necessary to analyze these events, due to the possibility that an SB-LOCA produces a more limiting temperature response than LOCAs and MSLBs. MSLBs must be examined closely for both containment pressure and containment temperature responses, either or both of which may possibly be more limiting than any LOCA. Thus, to ensure that the most limiting events for containment pressure and temperature are identified, several cases of different break sizes and locations must be analyzed for each class (i.e., LB-LOCA, SB-LOCA and MSLB) of event.

6.2 LOCA PRESSURE RESPONSE ANALYSES

6.2.1 OVERVIEW

The mass and energy release analyses for large break LOCAs are discussed in Section 3.0. Eleven FATHOMS cases were run to examine containment response for peak pressure sensitivity to break location, critical flow model, offsite power availability, containment back pressure, and refill time.

6.2.2 ANALYTICAL APPROACH

The analytical approach for examination of containment pressure response to LOCAs was to examine hot leg DEGBs at the vessel and at the S/G and cold leg DEGBs on both sides of the RCP. The cases are run for a period of time long enough to ensure that the highest peak for each case is identified (recognizing that a case can have more than one peak). Additional cases are also run to determine the sensitivity of the containment response to changes in break flow model selection and containment back pressure effects. The licensee examined a sufficient number of cases to reasonably conclude that the limiting (i.e., worst case) containment loading events have been identified.

6.2.3 MODIFICATIONS TO FATHOMS BASE MODEL

The Oconee base FATHOMS model is described in Section 2.4.2. For the peak pressure analyses, certain conservative modifications are made to the base model.

Containment volume: For LOCAs, the pressure response is particularly sensitive to containment volume. For conservatism, a 2% reduction in the containment free volume is assumed.

Passive heat sink surface area and heat transfer correlations: The surface area (and thus also, the volume) of passive heat sinks is reduced by 1% for conservatism. The Uchida heat transfer correlation is used to determine the heat transfer coefficients on passive heat structures. This is consistent with Standard Review Plan guidance.

Break discharge droplet size: A 20 μm droplet size is assumed based on a recommendation by Numerical Applications, Inc. (the FATHOMS vendor) for use as a nominal value for break discharge droplet size. A sensitivity study, using a 2 μm droplet size indicated that the containment peak pressure response is insensitive to droplet size selection in this range. The 20 μm droplet size selection is therefore acceptable.

Containment heat removal systems: In the peak pressure analyses no credit is taken for fan cooler or containment spray as these systems may not function early enough into the event to affect the peak pressure response.

6.2.4 INITIAL CONDITIONS FOR PEAK PRESSURE ANALYSES

The containment initial conditions have a potentially significant effect on the peak pressure response. The effect of assuming a cold initial temperature and a high initial pressure is to increase the initial mass of noncondensible gas in the containment. This results in an increased air partial pressure contribution to the peak total (steam+air) containment pressure and is conservative. The effect of low humidity (dry air) is to increase the initial air mass. For conservatism, the peak pressure analyses assume containment initial conditions of 1.5 psig (a Technical Specifications limit), 110 °F, and 0% relative humidity. The temperature and humidity values bound normal power operation.

6.2.5 BLOWDOWN BOUNDARY CONDITION FOR PEAK PRESSURE ANALYSES

The blowdown boundary conditions for the LOCA peak pressure analyses are the mass and energy results of the RELAP5 analyses. The combined steam and water mixture representing both sides of the break is injected into the FATHOMS model via a junction provided for that purpose.

6.2.6 RESULTS OF PEAK PRESSURE ANALYSES

As stated in 6.2.2 "Analytical Approach" above, a large number of cases are analyzed. The following table and discussion provides the results of the LOCA peak pressure case studies.

TABLE 6.2.6 - RESULTS OF PEAK PRESSURE ANALYSES

CASE	BREAK	TIME OF PEAK PRESSURE seconds	PEAK PRESSURE psig	PEAK TEMPERATURE °F
1A	14.1 SQ.FT. HOT LEG BREAK AT VESSEL OUTLET WITH OFFSITE POWER AVAILABLE, UNADJUSTED RANSOM AND TRAPP CRITICAL FLOW MODEL	13.9	57.7	283.5
1B	14.1 SQ.FT. HOT LEG BREAK AT OTSG INLET WITH OFFSITE POWER AVAILABLE, UNADJUSTED RANSOM AND TRAPP CRITICAL FLOW MODEL	14.9	58.4	284.4
1C	6.35 SQ.FT. COLD LEG PUMP DISCHARGE BREAK WITH OFFSITE POWER AVAILABLE, UNADJUSTED RANSOM AND TRAPP CRITICAL FLOW MODEL	18.9	51.9	276.1
1D	6.35 SQ.FT. COLD LEG PUMP SUCTION BREAK WITH OFFSITE POWER AVAILABLE, UNADJUSTED RANSOM AND TRAPP CRITICAL FLOW MODEL	23.2	54.4	279.3
2A	14.1 SQ.FT. HOT LEG BREAK AT VESSEL OUTLET USING HENRY-FAUSKE/MOODY CRITICAL FLOW MODEL WITH OFFSITE POWER AVAILABLE	THIS CASE RUN FOR MASS AND ENERGY ONLY		
2B	14.1 SQ.FT. HOT LEG BREAK AT VESSEL OUTLET WITH RANSOM AND TRAPP CRITICAL FLOW MODEL AND DISCHARGE COEFFICIENT ADJUSTED TO MATCH APPX MODEL, WITH OFFSITE POWER AVAILABLE	12.8	58.3	284.3
2C	14.1 SQ.FT. HOT LEG BREAK AT OTSG INLET USING HENRY-FAUSKE/MOODY CRITICAL FLOW MODEL WITH OFFSITE POWER AVAILABLE	THIS CASE RUN FOR MASS AND ENERGY ONLY		
2D	14.1 SQ.FT. HOT LEG BREAK AT OTSG INLET USING RANSOM AND TRAPP CRITICAL FLOW MODEL AND DISCHARGE COEFFICIENT ADJUSTED TO MATCH APPX MODEL, WITH OFFSITE POWER AVAILABLE	13.8	58.9	285.0
2D-80	SAME AS 2D BUT WITH CONTAINMENT AT 80°F INITIAL TEMPERATURE	13.7	58.8	284.0
3A	14.1 SQ.FT. HOT LEG BREAK AT VESSEL OUTLET WITH LOSS OF OFFSITE POWER AND WITH ADJUSTED DISCHARGE COEFFICIENT	12.3	57.9	283.8
3B	14.1 SQ.FT. HOT LEG BREAK AT OTSG INLET WITH LOSS OF OFFSITE POWER AND WITH ADJUSTED DISCHARGE COEFFICIENT	14.5	58.9	285.0
3B-80	14.1 SQ.FT. HOT LEG BREAK AT OTSG INLET WITH LOSS OF OFFSITE POWER AND WITH ADJUSTED DISCHARGE COEFFICIENT WITH CONTAINMENT AT 80°F INITIAL TEMPERATURE	14.3	58.8	284.0
3C	14.1 SQ.FT. HOT LEG BREAK AT VESSEL OUTLET WITH OFFSITE POWER AVAILABLE AND 30 PSIG CONTAINMENT INITIAL BACKPRESSURE	13.8	58.4	284.4
3D	14.1 SQ.FT. HOT LEG BREAK AT VESSEL OUTLET WITH OFFSITE POWER AVAILABLE AND 30 PSIG CONTAINMENT INITIAL BACKPRESSURE	13.5	58.4	284.4
3E	14.1 SQ.FT. HOT LEG BREAK AT VESSEL OUTLET WITH OFFSITE POWER AVAILABLE AND ZERO REFILL TIME	12.8	58.3	284.3

Four cases (1A, 1B, 1C and 1D) were analyzed to examine sensitivity to break location: hot leg breaks at the vessel and at the OTSG, and cold leg breaks on the RCP suction side and discharge side. These Series 1 cases utilized the Ransom and Trapp break flow model with unadjusted flow coefficients. Results of the analyses indicate that the hot leg break at the OTSG inlet produces the greatest peak containment pressure (58.4 psig). The peak pressure for the RPV outlet nozzle break was only slightly less (57.7 psig). Each hot leg break location produces a peak containment pressure earlier and higher than each cold leg break location.

Cases 2B and 2D investigated the two hot leg break locations to compare the Ransom and Trapp critical flow model with adjusted break discharge flow coefficients, to the Case 1 results which used the unadjusted flow coefficients. The Ransom and Trapp critical flow model with discharge coefficients adjusted by use of a multiplier to match the EM model (Appendix K LOCA Evaluation Model) produced slightly higher peak pressures and temperatures than the unadjusted Ransom and Trapp model. The results confirmed that the Ransome and Trapp model with properly adjusted discharge coefficients produces conservative results for peak pressure and temperature response.

The two Series 2 hot leg breaks (Case 2B and Case 2D for RPV outlet and OTSG inlet respectively) were rerun for loss of offsite power sensitivity. These cases, 3A and 3B, indicate that a loss of offsite power assumption results in slightly lower peak pressures and temperatures for the vessel outlet break, but has no significant effect for the OTSG inlet break.

Cases 3C and 3D investigated sensitivity to containment backpressure. As expected, containment backpressure was found to have no significant effect on peak pressure response.

Case 3E investigated sensitivity to refill time for the hot leg break at the RPV. Results were identical for cases with zero refill time and code-calculated refill time. (A zero refill time is simulated by stopping the code execution and fictiously filling the RPV lower plenum, then restarting.) This was expected since refill begins after the initial containment pressure peak.

Cases 2D-80 and 3B-80, were reruns of cases 2D and 3B to investigate the effect of initial containment conditions. 2D and 3B assumed initial containment conditions of 1.5 psig and 110°F. Cases 2D-80 and 3B-80 assumed 1.2 psig and 80°F. Results of these studies indicate that loss of offsite power is a conservative assumption for low containment temperatures and non-conservative for higher temperatures. These cases also indicate that low containment initial temperature combined with increased initial pressure could result in peak pressure exceeding the design limit of 59 psig. However, an 80°F containment temperature is considered unrealistic.

The most limiting case for LOCA peak pressure was 3B, the hot leg break at the OTSG inlet with loss of offsite power, for which the peak pressure 58.9 psig. The current FSAR analysis (Ref.: FSAR Table 15-9) indicates a worst case LOCA peak pressure of 53.5 psig for the (14.1 ft^2) hot leg break.

The licensee examined a sufficient spectrum of break cases to identify the worst case LOCA for containment peak pressure response.

6.3 LONG-TERM LARGE BREAK LOCA ANALYSES

6.3.1 OVERVIEW

In the long-term phase of a LOCA, heat continues to be added to the containment. Heat added in the form of steam is condensed by the containment spray and fan cooler systems. The sensible heat of the liquid recirculated from the sump is removed by the LPI coolers. These systems must maintain the containment temperature and pressure below specified limits. The long-term phase is analyzed for a period of 20 days. During this period, the containment conditions should be bounded by the defined set of equipment environmental qualification (EQ) curves. At Oconee, the fan cooler and LPI cooler capacities have been degraded by fouling and tube plugging. The goal of the long-term analyses is thus to ensure that the fan coolers and LPI coolers have adequate capacity.

6.3.2 ANALYTICAL APPROACH

The mass and energy release analyses were discussed in Section 3.0 above.

The long-term large cold leg break LOCA containment response FATHOMS analyses are run in three segments. Segment 1 utilizes RELAP5 mass and energy data and covers the period up to 30 minutes. Segment 2 utilizes BFLOW break quality data and covers the period from 30 minutes to initiation of sump recirculation at 2808 seconds. Segment 3 covers the recirculation phase out to 15 days. (This period is then extrapolated to 20 days.)

A series of FATHOMS runs is performed to determine the minimum fan cooler heat removal coefficient required for a given cooling water supply temperature (assuming constant LPI cooler heat transfer coefficient and spray flow). Additional runs are subsequently performed to determine the trade-off effect for lower fan cooler capacity and increased LPI cooler capacity and for increased spray flow.

6.3.3 MODIFICATIONS TO BASE CONTAINMENT MODEL

Containment free volume and passive heat sink data: The conservative assumptions for containment volume and heat sinks described in Section 6.2.3 for pressure analyses are also used in the long-term analyses for containment heat removal.

BWST volume and temperature: As noted in Section 2.4.1 above, the FATHOMS code used for Oconee containment analyses includes a long-term mass and energy release model. The BWST contents are initialized at a temperature of 115°F. During Segment 1, the injection flow coming from the BWST is accounted for by the RELAP5 break boundary conditions. During Segment 2, the FATHOMS/DUKE-RS long-term mass and energy model tracks the BWST volume. At the 30 minute point, the BWST volume is initialized using the RELAP5

information. The BWST volume is tracked until it drops to the recirculation setpoint (at approximately 2808 seconds) at which time Segment 2 is terminated. During Segment 3 sump recirculation is the injection flow source.

Heat transfer correlations: The Uchida correlation is used in heat transfer calculations for containment internal concrete and steel heat structures except for the containment base. A relatively large, constant heat transfer coefficient of 20/BTU/hr-ft^2-°F is assumed for the base. This value was selected as a weighted average between the pool-to-concrete heat transfer coefficient between the base and sumpwater (90%), and the atmosphere-to-concrete heat transfer coefficient between the containment atmosphere and unsubmerged portion of the base (10%). The large heat transfer coefficient is considered conservative since the base is being cooled rather than heated during the long-term phase.

[ ]

Droplet size of blowdown mass: As in the peak pressure analysis, a 20 μm droplet size is assumed. Use of the the blowdown phase droplet size for the long-term analysis is conservative.

LPI cooler data: LPI cooler performance data are inserted into tables in the FATHOMS input. The data in the tables are based on outage test data. The outage test data provides measured heat transfer performance for known conditions of service water supply temperature and LPI cooler outlet temperature.

Containment fan cooler (RBCU) data: The basic heat removal capacity for RBCUs is for a containment temperature of 110°F and service water inlet temperature of 90°F. A conversion factor is applied for each analysis, adjusted by the analyst, to reflect the degree of water-side fouling. Another factor, provided in the form of input tables is also applied to account for the effect on RBCU performance of changing containment atmosphere conditions. The tables reflect varying conditions of containment temperature and humidity. Another multiplier is applied when the RBCU service water inlet is other than 90°F. All cases assume two RBCUs

operating with a volumetric airflow rate of 108,000 cfm.

The 110°F-base heat transfer capacity value for RBCU performance is conservative in that it results in reduced RBCU heat transfer, relative to the heat transfer rate that would result if the 125°F initial containment temperature assumption was used.

Core decay heat: The FATHOMS/DUKE-RS long-term mass and energy model calculates core decay heat based on EOC conditions and using ANS-5.1-1979 with 2σ uncertainty. Heat released from the Group 1 heat structures is added to the decay heat.

6.3.4 INITIAL CONDITIONS

Initial conditions, applicable to the long-term analyses are described below.

Initial building temperature: The initial containment temperature (atmosphere and structure surfaces) is assumed to be 125°F. This value has been determined to be conservatively high for Oconee.

Initial pressure and humidity: The initial containment pressure and humidity are selected as 14.7 psia and 100% respectively.

The results of long-term analyses are not sensitive to small changes in these parameters.

6.3.5 BOUNDARY CONDITIONS

The FATHOMS/DUKE-RS long-term LB-LOCA model includes[ ] junctions, each assigned to a particular boundary condition. Each junction is used to account for a mass and energy sink or source interface with the containment. Certain junctions are not used during certain segments of the analysis (e.g., Junction [ ].

6.3.5.1 SEGMENT 1 FLOW PATH

During Segment 1, [four boundary conditions (#1-#4) are in effect for the break flow - a vapor and liquid condition for each of the two sides of the DEGB.] These boundary conditions calculated by the RELAP5 mass and energy release analyses discussed in Section 3.6 above.

One boundary condition[ ] is in effect for building spray during Segment 1. The flow is initiated at 96 seconds after the break. The assumed temperature is 115°F (BWST temperature) and droplet size is 7000 μm (a conservative value, as containment spray nozzle nominal droplet size would not be expected to exceed 1000μm).

One boundary condition[ ] is used to account for the injection of the CFT nitrogen. The flow is assumed to begin at 37.6 seconds at a constant rate of 356 lbm/sec and terminates at 44.5 seconds. The nitrogen pressure and temperature conditions are the same as for the CFTs in the RELAP5 analyses.

One boundary condition (#18) supplies make-up water to the sump. This flow corrects for mass imbalances between the RELAP5 analysis and FATHOMS analysis. In RELAP5 the BWST outlet (injection) flow was assumed to be conservatively high to rapidly deplete the tank. In FATHOMS, conservatively low flow rates are assumed for injection and containment spray. The Junction 18 boundary condition makes up the mass imbalance. The spilled HPI fluid from the broken loop (the fluid that doesn't make it to the core) is also added to this flowpath.

6.3.5.2 SEGMENT 2 FLOW PATHS

In Section 2.4.1, it was noted that FATHOMS/DUKE-RS includes a long-term mass and energy release model. This model is used beginning in Segment 2 when sump recirculation begins. There are[

]

6.3.5.3 SEGMENT 3 FLOW PATHS

For Segment 3 of the long-term analyses,[

]

6.3.6 RESULTS OF LONG-TERM LB-LOCA ANALYSES

Twenty-seven long-term analysis cases were run for various combinations of containment spray flow, LPI cooler capacity, fan cooler (RBCU) capacity units, and service water supply temperatures. The analyses demonstrate the relative containment temperature control effectiveness of containment spray, LPI coolers and RBCUs. Unlike the peak pressure analyses, for which the specific containment design pressure value serves as an acceptance criterion for peak pressure, the long-term analyses have a time-dependent EQ temperature profile to be satisfied. The new long-term analyses do not result in a greater peak pressure than the initial short-term blowdown response for hot leg breaks. The long-term analyses demonstrate that the RBCU's contribution to overall long-term containment heat removal is very sensitive to service water temperature which in turn varies seasonally.

6.4 SB-LOCA ANALYSES

6.4.1 OVERVIEW

SB-LOCAs are a long-term containment cooling concern because the RCS pressure and temperature may remain high, with considerable flashing at the break, for an extended period of time. The goal of SB-LOCA analyses is to determine if the required RBCU (containment fan cooler system) capacity is more limiting than for the large-break LOCA. For the SB-LOCA analyses, RELAP5 calculates the mass and energy releases and FATHOMS/DUKE-RS calculates the containment pressure and temperature response.

6.4.2 ANALYTICAL APPROACH

The mass and energy analyses for SB-LOCA were discussed in Section 4.0 above. Six break sizes are analyzed for mass and energy release. If the results indicate by inspection that the resulting containment response would be bounded by a larger break, the case is not run. The RELAP5 mass and energy responses for the two smallest breaks indicate by inspection that the containment responses for those breaks are bounded by more limiting LB-LOCA results.

Operator actions assumed in the analyses are selected and timed to be consistent with Emergency Operating Procedures.

Four analyses were run long enough to determine if the LB-LOCA EQ profile is satisfied. In some cases, the analyses were run for a period of time beyond the available RELAP5 mass and energy data. As discussed in Section 4.1 above, RELAP5 data are[

]

6.4.3 MODIFICATIONS TO BASE CONTAINMENT MODEL

The following modifications are made to the base Oconee FATHOMS model for SB-LOCA containment response analyses.

Containment free volume and passive heat sink data: For the SB-LOCA analyses, the FATHOMS base model is modified, as in the peak pressure analyses, for a 2% reduction in containment volume, a 1% reduction in passive heat sink surface areas and volumes and use of the Uchida heat transfer correlation.

Reactor coolant system volumes: As noted in 6.4.2 above, a [ ] is added to FATHOMS/DUKE-RS for SB-LOCA analyses.

BWST: For the larger two breaks of the six SB-LOCA cases, a BWST model was added to the FATHOMS model because the RELAP5 analysis

had not reached the sump recirculation point when it was terminated and the BWST was still being drained.

Junction changes: During the first part of the FATHOMS analysis, FATHOMS uses RELAP5 mass and energy results as a boundary condition. [ ] to provide the mass and energy data. For two of the break cases, a junction for the boron dilution line is opened at 15 hours.

Heat transfer correlation: The Uchida heat transfer correlation is used for passive heat structures.

RCS heat structures: RCS heat structures are modeled as [ ].

Droplet size of blowdown mass: Consistent with the peak pressure analyses (See Section 6.2.3 above), a 20 μm average droplet diameter value is used. However, due to the short duration of two-phase flow from the break, the SB-LOCA FATHOMS results are not highly sensitive to droplet size assumption.

LPI cooler data: Section 6.3.3 above discusses how, for long-term LB-LOCA FATHOMS analyses, spreadsheet-generated data are used for LPI cooler performance. The same spreadsheet algorithm is used for SB-LOCA FATHOMS analyses. Lower injection fluid flow rates are needed for analysis of portions of the SB-LOCA cases where the HPI pumps alone are supplying the injection flow. The spreadsheet data are given to FATHOMS in tables of data for various flow rates.

RBCU data: The base model modifications described in Section 6.3.3 for long-term large break analyses are also applicable to the SB-LOCA analyses.

6.4.4 INITIAL CONDITIONS

The base model modifications described in Section 6.3.4 for long-term large break analyses are also applicable to the SB-LOCA analyses.

6.4.5 BOUNDARY CONDITIONS

[ ] are utilized in the FATHOMS model, [ ] of which are used for flow path boundary conditions. However, not all junctions are used for all segments of the analyses or for all break sizes. A discussion of the various flow paths follows.

Cold leg break flow junctions: Four junctions are provided for cold leg break flow into FATHOMS, two vapor and two liquid. For the smaller two of the four analyzed break sizes, two pairs of junctions are needed. For the larger small breaks, only one pair is needed. The need for two pairs of junctions in the cases of the smaller breaks is a result of the longer time frame of the RELAP5 analysis.

Containment spray flow junction: During the initial portion of a FATHOMS analysis, while it is using RELAP5 mass and energy data and RELAP5 is tracking the BWST, spray flow into the containment is modeled with one boundary condition junction. The junction becomes active at the point in time when containment pressure rises to 20 psig, plus a time delay and continues to the end of the RELAP5 portion, after which other junctions are used. The spray temperature is 115°F and the flow rate is 1500 gpm minus an uncertainty allowance.

Containment spray - recirculation: Two junctions, one for containment spray sump suction and one for dome discharge, are provided for use during the recirculation phase for the intermediate size small break cases. The spray flow during this period is reduced to 600 gpm. Recirculation phase spray is not needed for the larger break cases.

HPI spill: During the RELAP5 portion of a FATHOMS analysis, one of the four HPI lines is assumed to be broken for the small break. BWST fluid is thus added to the containment sump using this junction. After the RELAP5 portion of the analysis, HPI spill fluid is being drawn from FATHOMS' internal BWST, instead of the RELAP5 BWST, so this junction is no longer used.

HPI junctions: As noted earlier, for SB-LOCAs, RELAP5 is run until the RCS is cooled and depressurized (i.e., the heat removal load is shifted from the OTSGs to the LPI coolers), after which, the [ ] For the larger of the small breaks cases (Cases 5 & 6), the RELAP5 portion of the analysis is relatively short (≈1 hour into the event), due to the more rapid cooldown/depressurization. For the intermediate small breaks (Cases 3 & 4), the RELAP5 data are generated for about 10 hours (several hours into the recirculation phase). When the RCS is depressurized, LPI becomes available and HPI is off. Recirculation initiation is determined by which comes first; (a) BWST depletion, or (b) depressurization. If the BWST is

depleted prior to depressurization, HPI is continued in the recirculation mode using "piggyback" LPI/HPI pump alignment operation. In order to properly model and account for mass and energy transfer under the various HPI/LPI alignments, six FATHOMS junctions are provided.

Non-boundary condition junctions: There are[

]

6.4.6 RESULTS OF SB-LOCA CONTAINMENT ANALYSES

Case 1, 0.0025 ft^2: The RELAP5 mass and energy data indicate that the containment response for this case is clearly bounded by the large break cases. Therefore FATHOMS was not run for this case.

Case 2, 0.005 ft^2: The RELAP5 mass and energy data indicate that the containment response for this case is clearly bounded by the large break cases. Therefore FATHOMS was not run for this case.

Case 3, 0.01 ft^2: Two sets of data were generated, one for the case with 35% fan cooler capacity with 55°F service water, and the other for 90% fan cooler capacity and 90°F service water. The containment EQ temperature and pressure limitations were met for both cases at all points in time. In addition, the results of these plus some additional runs, confirmed that, for the complete range of service water temperatures, the containment response for this break is bounded by the large break cases.

Case 4, 0.025 ft^2: This break is similar to the Case 3 break. Fan cooler capacity and service water temperature are less limiting than for Case 3 and, as for Case 3, are bounded by large break LOCAs.

Case 5, 0.05 ft^2: For Case 5, the RCS cools rapidly and the containment response is mild. EQ limits are not exceeded. As with the above SB-LOCA cases, the containment response for this case is bounded by large break LOCAs.

Case 6, 0.10 ft^2: The containment response was similar to but slightly more severe than for Case 5. As with the above smaller SB-LOCA cases, Case 6 results are within EQ limits and are bounded by LB-LOCA results.

6.4.6.5 SUMMARY - SB-LOCA CONTAINMENT RESPONSE ANALYSES

The results indicate that SB-LOCA containment response is bounded by LB-LOCA response. As with the large breaks, fan cooler fouling and high service water temperature can result in unacceptable results. As a result, SB-LOCAs require

a reduction in the containment spray actuation setpoint and opening of the boron dilution flowpath. The Technical Specification for the spray actuation setpoint is 30 psig. However, as a result of the new analysis, the licensee has administratively implemented a lower spray actuation pressure setpoint of 10 psig. Emergency Operating Procedures will be revised; and the Technical Specifications amendment, subject to, and following, approval of the topical report.

6.5 MSLB ANALYSES

6.5.1 OVERVIEW

The mass and energy releases for a MSLB are calculated with the RETRAN-02 code as discussed in Section 5.0 above. A 34-inch MSLB is analyzed with and without offsite power available. The containment response for smaller break sizes would be bounded by the 34-inch diameter break case.

6.5.2 ANALYTICAL APPROACH

The FATHOMS/DUKE-RS (FATHOMS) code is used to calcuate the containment temperature and pressure profiles for a MSLB using the Oconee base FATHOMS model with minor modifications as described below. The objective of the analysis is to verify that the containment pressure and temperature responses are bounded by the containment design conditions and electrical equipment environment profile.

6.5.3 MODIFICATIONS TO OCONEE BASE CONTAINMENT MODEL

Containment volume: The peak pressure analyses are particularly sensitive to containment volume. For conservatism, a 2% reduction in the containment free volume cross-sectional area is assumed.

Passive heat sink surface area and heat transfer correlation: The surface area/volume of passive heat sinks is reduced 1%. The Uchida heat transfer correlation is used for condensing heat transfer coefficients on passive heat structures. Consistent with NUREG-0588 guidance, T_{SAT} rather than T_{BULK} is used in calculations of heat transfer from the containment atmosphere to structures.

RBCUs (fan coolers): Two RBCUs with a reduced capacity of 50% each are credited in the analyses.

6.5.4 INITIAL CONDITIONS

For the MSLB analyses, FATHOMS is initialized with a containment pressure of 16.2 psia (1.5 psig) and temperature of 125°F. The above-atmospheric pressure is conservative because it increases the initial air mass. The 125°F temperature is based on a consideration of the relative effects of the initial temperature assumption on both air mass and heat sink capacity of heat structures. An initial humidity of 0% is assumed to maximize the air mass. These initial conditions are conservative for the MSLB analysis.

6.5.5 BOUNDARY CONDITIONS

The break mass flow, pressure and enthalpy data from the RETRAN-02 mass and energy analyses are supplied to FATHOMS as a boundary condition. Containment spray is a second boundary condition. The spray flow is (1500 gpm minus instrument error of 143 gpm) 1357 gpm. Spray is tripped on at 92 seconds when the containment reaches a pressure of 30 psig. A spray droplet size of 7000 μm is assumed.

6.5.6 RESULTS OF MSLB CONTAINMENT RESPONSE ANALYSES

The following table summarizes the results of the MSLB analyses.

CASE	TIME OF PEAK PRESSURE (sec)	PEAK PRESSURE (psig)	PEAK TEMPERATURE (°F)
Offsite Power Maintained	200	58	397
Loss of Offsite Power	375	58	418

The containment pressure remained below the design pressure. The containment atmosphere temperature exceeded the EQ profile, in both cases, for a short period of time early in the event. As noted in Section 5, operator action is credited for termination of feedwater flow to the affected OTSG at 170 seconds, in both cases. Without this action, containment pressure and temperature response would be more severe.

Because the containment temperature response exceeded the EQ profile, the staff evaluated the potential effect on equipment. Environmental qualification (EQ) of electric equipment ensures that equipment will perform its safety function during and following a design basis event. The licensee prepared calculation number OSC-5460, "Oconee MSLB/EQ Analysis" to determine whether the new MSLB containment temperature response affects the qualification of electrical equipment inside containment. The containment temperature exceeds the EQ test temperature for approximately 50 seconds in the early part of the Main Steam Line Break (MSLB) transient. The licensee performed a thermal lag analysis and concluded that equipment internals would not experience the peak temperatures associated with the MSLB vapor temperature spike. Although the MSLB peak temperature is higher than that of a LOCA, the time period is short. Equipment internal temperatures would actually be higher when subjected to a LOCA. The equipment required to mitigate the consequences of the MSLB is qualified and would perform its safety function. For documentation of qualification, the new MSLB profile will be included in the Oconee EQ Criteria Manual.

7.0 SUMMARY AND CONCLUSIONS

The new analyses described in the topical report expand the scope of analyzed piping failures in containment for the Oconee facilities. The licensee has utilized new methods to reanalyze existing licensing basis pipe failure events in containment, and to examine the potential effects of previously unanalyzed pipe failures in containment. The new methodology utilizes modeling assumptions and initial conditions which the staff finds to be consistent with current staff acceptance criteria or produce equally conservative results.

The staff has reviewed the licensee's analysis of the new MSLB containment temperature response profiles for their impact on environmental qualification of equipment. The licensee responded to staff questions on this subject in a letter dated June 9, 1994. The staff agrees with the licensee's conclusion that the equipment is qualified and will perform its safety function during an MSLB in containment.

The DPC-NE-3003-P basic methodology discussed in this evaluation, with appropriate adjustments to reflect potential plant modifications, may be used by the licensee to perform future reanalyses in support of licensing applications related to containment accident response.

Duke Power Company
Oconee Nuclear Station

MASS AND ENERGY RELEASE AND CONTAINMENT RESPONSE METHODOLOGY

DPC-NE-3003-A

Revision 1

September 2004

Nuclear Engineering Division
Nuclear Generation Department
Duke Power Company

Abstract

This report describes the Duke Power Company methodology for simulating the mass and energy release from high energy line breaks and the resulting containment response for the Oconee Nuclear Station. The mass and energy release resulting from LOCAs is simulated with the RELAP5/MOD2-B&W and BFLOW codes. The mass and energy release resulting from a steam line break is simulated with the RETRAN-02 MOD5.1DKE code. The containment response is simulated with the FATHOMS/DUKE-RS code. A spectrum of break sizes and locations are analyzed to determine the limiting cases. The results are then compared to the Reactor Building design pressure limit and the pressure and temperature profiles which establish the equipment qualification limits for the post-accident harsh environment. These reanalyses of the UFSAR licensing basis address many shortcomings in the original analyses and determine limits and requirements on plant initial conditions and engineered safeguards performance.

Revision 1

Revision 1 updates the methodology for the replacement steam generators, the use of the RETRAN-3D MOD003.1DKE code for the steam line break analysis, the use of the GOTHIC Version 7.0 code for the containment analysis, and other minor methodology revisions.

MASS AND ENERGY RELEASE AND
CONTAINMENT RESPONSE METHODOLOGY

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1. INTRODUCTION

This report describes the methodologies to be used by Duke Power Company to simulate the mass and energy release from high-energy line breaks, and the resulting containment pressure and temperature response. The analyses are applicable to the Oconee Nuclear Station, which is a 2568 MW_{th} pressurized water reactor of the Babcock & Wilcox design. The Oconee Reactor Building is of the large dry containment design.

Chapter 6.2 of the Oconee Final Safety Analysis Report (Reference 1-1) documents the licensing basis analyses related to high-energy line breaks inside containment. The original analyses were performed prior to the initial startup of the station in 1973, and were not reanalyzed until application of the methodology in this topical report in 1995. The UFSAR analyses fall into the following categories:

- LOCA analyses which determine the peak pressure inside containment as a function of break size and location.
- LOCA analyses which determine the long-term containment response and show the effectiveness of containment engineered safeguards systems.
- LOCA analyses which determine the minimum containment backpressure to be used for evaluating the effectiveness of the Emergency Core Cooling Systems in limiting the peak cladding temperature.
- A steam line break analysis to show that the LOCA response is bounding.

The UFSAR analyses were performed by B&W and employed assumptions which were expected to result in a conservative prediction of the containment response. Without the benefit of any operating experience at Oconee, the selection of analysis inputs was based on expected plant conditions and design data. Beginning in April 1986, several issues were identified which impacted the boundary conditions assumed in the analysis of the containment response. The containment engineered safeguards at Oconee include fan coolers, sprays, and the Low Pressure Injection System (LPI) coolers which cool the containment sump water prior to returning it to the reactor vessel for core cooling. The first issue was an observed degradation of the heat transfer capability of the LPI coolers due to fouling. The second issue, which was identified as a follow-up action of the first, was the detection of fouling of the fan coolers. The third issue was an increase in the temperature of the ultimate heat sink to a

value slightly above that assumed in the UFSAR analyses. The fourth issue was the temperature of the containment atmosphere, with measured temperatures exceeding the value assumed as the initial condition in the UFSAR analyses. The fifth issue was an accumulation of changes and corrections to the data base of passive structural heat sinks. Each of these issues was evaluated and resolved in the short-term by analyzing the impact on the UFSAR analyses using simplified energy balance or other methods. A longer term effort to develop more sophisticated methods using simulation codes was in progress.

During the evaluation of the above issues it was recognized that the only long-term post-LOCA containment response analyzed in the UFSAR was for a hot leg break. A review of the physical processes that occur following both hot and cold leg breaks led to the conclusion that the cold leg break could be more limiting. A detailed analysis of the long-term cold leg break containment response was necessary. Resolution of this issue could not be undertaken using simple methods.

This report is the culmination of the development effort initiated to 1) address the absence of a long-term cold leg break containment response analysis in the UFSAR, and 2) to replace the existing UFSAR analyses with revised analyses using modern computer codes that utilize assumptions which are conservative and bounding based on operating experience. One existing UFSAR containment analysis is not reanalyzed and is beyond the scope of this report. That analysis is the minimum backpressure analysis used in the LOCA ECCS analysis as required by 10CFR50 Appendix K (Reference 1-2). The minimum backpressure analysis has been analyzed generically by B&W as part of the LOCA analysis for all B&W plants. Since this analysis remains conservative and bounding, there is no need for reanalysis in this report. Therefore, this report focuses on conservative predictions of the peak containment pressure and temperature response for LOCA and steam line break.

Chapter 2 describes the computer codes used in each part of the analyses. The LOCA mass and energy releases are simulated with the RELAP5/MOD2-B&W code (Reference 1-3). The long-term post-LOCA boiloff phase is modeled with the BFLOW code, which was developed by Duke specifically for this application. The steam line break mass and energy release is modeled with the RETRAN-02/MOD005 code (Reference 1-4). Duke Power has submitted for NRC review a topical report which includes details of RETRAN-02 transient analysis models for Oconee (Reference 1-5). The containment analyses employ the FATHOMS/DUKE code, which is a customized version of the FATHOMS code (Reference 1-6). A brief description of each of these codes is provided along with a summary of validation efforts. Details of the simulation model for each code application are discussed.

Chapter 3 describes the details and results of the large break LOCA mass and energy release analyses. RELAP5 analyses of different break locations are shown in order to determine the worst cases for peak containment pressure and temperature response. The long-term boiloff results from the BFLOW code are also discussed.

Chapter 4 presents the results of the RELAP5 small break LOCA analyses. A spectrum of six break sizes were analyzed. This chapter illustrates the effect of break size on the mass and energy release.

Chapter 5 shows the results of the RETRAN-02 steam line break analysis, which is a double-ended rupture of a 34 inch I.D. steam line.

Chapter 6 details the containment pressure and temperature response for each of the line breaks discussed above. Sensitivity study results show the impact of varying initial and boundary conditions, and modeling assumptions. Margin to the containment design pressure, and the long-term temperature and pressure profiles for equipment qualification are the key results.

A summary of the report is given in Chapter 7.

The guidance provided in the Standard Review Plan Section 6.2 (Reference 1-7), Regulatory Guide 1.89 (Reference 1-8), and ANSI/ANS-56.4-1983 (Reference 1-9) was considered in the development of the methodology in this report. It should be noted that the reactor vessel internals vent valves, which are unique to a B&W design NSSS, and of critical importance in this methodology, are not discussed in these references. Since the mass and energy release methodology in this report differs from the industry norm in a few areas, those areas will be highlighted and justified. Consistent with the ANS Standard, Section 3.3, it is concluded that the main feedwater line break is bounded by the LOCA or steam line break, and so the feedwater line break is not analyzed in this report.

Supplement 1 was added with the November 1997 version of Revision 0 of the report to present updated steam line break mass and energy release and containment response analysis results that supercede the results in Chapters 5 and 6 of the original report.

Supplement 2 was added with the November 1997 version of Revision 0 of the report to present validation analyses comparing the FATHOMS prediction to data from the CVTR facility.

Appendix A was added in Revision 1 to describe the replacement steam generator (ROTSG) steam line break mass and energy release modeling with RETRAN-3D (Reference 1-10).

Appendix B was added in Revision 1 to address the RETRAN-3D SER (Reference 1-11) conditions and limitations as relates to the replacement steam generator (ROTSG) steam line break mass and energy release modeling with RETRAN-3D MOD003.1/DKE.

Appendix C was added in Revision 1 to include the use of the GOTHIC 7.0/DUKE code (Reference 1-12) for containment response analysis.

1.1. References

1-1 Oconee Nuclear Station Updated Final Safety Analysis Report, Duke Power Company

1-2 10CFR50, Appendix K - ECCS Evaluation Models

1-3 RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis, BAW-10164P, Babcock & Wilcox, December 1987

1-4 RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCMA Revision 4, EPRI, November 1988

1-5 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision 3, Duke Power Company, September 2004

1-6 CAP-Containment Analysis Package (FATHOMS), Numerical Applications, Inc., October 10, 1989

1-7 USNRC Standard Review Plan, Section 6.2, NUREG-0800, USNRC

1-8 Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants, Reg. Guide 1.89, USNRC

1-9 Pressure and Temperature Transient Analysis for Light Water Reactor Containments, ANSI/ANS-56.4-1983

1-10 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI, NP-7450(A), Volumes 1-4, Revision 5, July 2001

1-11 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

1-12 GOTHIC Containment Analysis Package Version 7.0, NAI-8907-02, Revision 13, July 2001

2. COMPUTER CODES

2.1 RELAP5/MOD2-B&W - Primary System Mass and Energy Release

2.1.1 Code Description

2.1.1.1 Overview

The RELAP5/MOD2-B&W (Reference 2-1) thermal-hydraulic computer code is used to calculate mass and energy released from the Reactor Coolant System (RCS) following both large and small break loss of coolant accidents (LOCAs). RELAP5/MOD2-B&W is derived from RELAP5/MOD2 Cycle 36.05, which is an advanced thermal-hydraulic computer code developed by EG&G Idaho for the Nuclear Regulatory Commission (NRC). The code was originally developed to provide the NRC with a tool for auditing licensing analyses of both large and small break LOCAs. Babcock & Wilcox (B&W) has modified RELAP5/MOD2 by including the evaluation model correlations and methods required by 10CFR50 Appendix K. In addition, certain modifications have been made to add to the predictive capabilities of the constitutive models and improve code execution. The models required for Appendix K licensing calculations are optionally selected, and are not used in the mass and energy release analyses described in this document. The RELAP5/MOD2-B&W code is accepted for use by the NRC. The safety evaluation report (SER) written for the code is included in Section 5.3 of Reference 2-1.

The RELAP5/MOD2-B&W code can be used to simulate the steady-state and transient behavior of any general hydraulic system that may contain a mixture of steam, water, noncondensible gas, or a nonvolatile solute. Analysis of a particular fluid system is performed by first discretizing the system into control volumes or nodes which are joined by momentum cells or junctions. The code solves the continuity and energy conservation equations for each control volume and the momentum conservation equation for each junction during a time advancement. Heat structures are used to model energy exchange between two hydraulically separate systems or between the control volume and insulated or internal heat sources.

The RELAP5/MOD2-B&W hydraulic flow field model treats the liquid and steam phases as two separate fluids in a non-homogeneous, non-equilibrium fashion; the continuity, momentum, and energy conservation equations are solved for each phase. Constitutive relationships are used to model interactions between phases such as interfacial drag, vapor generation, and interfacial mass transfer. Other empirical relationships are used to model convective heat transfer, energy partitioning between the

liquid and vapor phases, choked flow, and two-phase wall friction. The code includes many generic component models from which general systems can be simulated. Component models include pumps, valves, pipes, heat structures, tee connections, reactor point kinetics, electric heaters, jet pumps, turbines, separators, accumulators, trip logic, and control system components. In addition, special process models are included to simulate the influence of high emergency feedwater injection on once through steam generator (OTSG) heat transfer and degradation of heat transfer due to the presence of noncondensible gas.

RELAP5/MOD2-B&W is one of the computer codes used by the B&W Fuel Company (BWFC) to perform licensing calculations in support of nuclear fuel reload design for pressurized water reactors. The BWFC has submitted a topical report to the NRC describing the physical models and numerical solution techniques employed by RELAP5/MOD2-B&W. This topical report has been reviewed and approved by the NRC and a Safety Evaluation Report for the code has been issued (Reference 2-2). The BWFC currently uses RELAP5/MOD2-B&W to calculate the RCS transient response during the blowdown phase of a large break LOCA, to characterize the entire transient response following a small break LOCA, and for analyzing non-LOCA transients.

2.1.1.2 Models and Features Unique to RELAP5/MOD2-B&W

As discussed previously, RELAP5/MOD2 has been modified by B&W to include all of the models and features required by 10CFR50 Appendix K for ECCS evaluation models. These modifications enable the code to be used for licensing LOCA analyses. The modifications included in the code to satisfy these requirements are:

- Critical Flow Models - The Moody, extended Henry-Fauske, and Murdock-Bauman critical flow models have been added in addition to the existing Ransom and Trapp model
- Core Heat Transfer Model - Heat transfer correlations have been included to model all regions of the boiling curve consistent with 10CFR50 Appendix K: pre-CHF heat flux correlations, CHF correlations, and post-CHF correlations
- Heat Transfer Switching Logic - This logic prevents a return to nucleate boiling and transition boiling as required by 10CFR50 Appendix K
- Fuel Pin Model - This model considers fuel rod deformation. fuel cladding swelling and rupture, gap conductance, and zircaloy-water reaction

These features must be specifically requested by the user before they are activated. RELAP5/MOD2-B&W preserves all of the original RELAP5/MOD2 correlations and models related to critical flow and core heat transfer unless specified otherwise. None of the models listed above are used in analyses described in this document.

Other coding changes have been made to improve the overall predictive capabilities of RELAP5/MOD2, particularly OTSG modeling. The most significant of these changes is the addition of an emergency feedwater (EFW) model which is used to calculate the heat transfer which results when EFW is injected into the upper regions of an OTSG. The ability to model this phenomenon is an extremely important prerequisite for simulating small break LOCAs because there is the potential for considerable steam condensation inside the OTSG tubes. The condensation of primary system steam inside the OTSG tubes can result in a significant energy exchange between the primary and secondary systems.

The EFW model consists of [

]. This model has not been reviewed for use in licensing calculations since B&W currently applies RELAP5/MOD2-B&W to licensing analyses for recirculating steam generators. However, this model is used in the RELAP5 analyses described in this document since it represents a mechanistic tool for predicting heat transfer due to EFW injection.

2.1.2 Simulation Model

This section provides a brief description of the simulation model utilized for the RELAP5 mass and energy release analyses. This description assumes the reader has some familiarity with the design and transient response of a B&W 177 FA plant to LOCAs. A nodalization diagram of the Oconee RELAP5 base model is shown in Figure 2.1-1.

<u>Nodalization</u>

The Oconee RELAP5 model is designed primarily for use with small and large break LOCA applications. Because the transient responses of these applications are dominated by distinctly different phenomena, the model has been designed to capture these phenomena by including greater levels of noding detail, and special models where required. For example, during a small break LOCA, break cooling alone is usually not sufficient to totally remove decay heat and cool the primary system. Because the steam

generators must also be used, the ability to accurately predict primary-to-secondary heat transfer is an important consideration. Sustained primary-to-secondary coupling is generally no concern in plants which use recirculating steam generators. However, because of the hot leg configuration of a B&W plant, it is likely that steam will collect in the U-bend region of the hot legs and effectively block all natural circulation flow, thereby terminating heat transfer. This condition will persist until the hot legs refill with liquid, or until the level in the primary system drops low enough to support boiler-condenser mode (BCM) heat transfer.

To accurately predict the interruption (and restoration) of heat transfer, detailed noding has been included in the [ ] steam generator model is discussed later in this section.

The transient response of a large break LOCA is considerably different from that of a small break LOCA. The large pressure gradients which exist during the blowdown phase are sufficient to provide fluid flow rates well in excess of normal system flow rates. These flow rates will maintain tight coupling between the primary and secondary systems. As a consequence, the level of noding required in loops and steam generators are much less than is necessary for a small break. Instead, the emphasis for large break LOCA modeling shifts from the steam generators and loops to the reactor vessel. Detailed noding is necessary in the [ ] during the reflood phase of the transient. The Oconee base model uses [ ] of the core reflood phase. It should be noted that [ ] is modeled. There is no attempt in these analyses to model the transient response in the hot channel or hot assembly which is typical for licensing analyses which calculate peak cladding temperatures. This is because the primary objective of these analyses is to calculate the rate of mass and energy release from the RCS. A detailed core thermal response is not required to generate these boundary conditions.

Reactor Vessel Vent Valves

The reactor vessel noding also includes a model for the internal vent valves. The reactor vessel vent valves are eight vertically oriented swing check valves which effectively vent fluid from the core outlet to the reactor vessel downcomer. These valves are instrumental in minimizing the core back-pressure during the reflood phase of a large cold leg break LOCA by venting steam generated in the core to the break location without the need to clear loop seals. They serve a similar role following a small cold leg break LOCA by venting two-phase liquid to the break site. In general, the vent valves are effective in maintaining pressure equilibrium within the reactor vessel so that balanced fluid levels are assured throughout the primary system.

A RELAP5 servo-valve junction is used to model all eight reactor vessel vent valves. The valve normalized area and corresponding flow coefficient are a function of the pressure differential across the valve, which is calculated by a control system. Data obtained from full size prototype vent valve testing establishes the bases for the valve characterization used in the base deck (References 2-3 and 2-4).

Radial Region Steam Generator

The OTSG tubes are modeled with [

]. This nodalization scheme is consistent with the recommended implementation of the RELAP5/MOD2-B&W EFW model (Reference 2-1).

During EFW System operation, feedwater is sprayed onto the outer periphery of the OTSG tubes through each of six spray nozzles. As the feedwater penetrates the tube bundle, it spreads and pools on the upper tube support plate before flowing downward through the broached openings around each tube. The distance the feedwater penetrates into the tube bundle, and hence the number of tubes wetted, [

].

These observations establish the basis for choosing how the OTSG [

]

The EFW model allows the user to [

] does not occur.

Main Feedwater Piping

The RELAP5 nodalization is revised to include the main feedwater piping between the last check valve and the steam generator. This enables modeling flashing of the main feedwater in the piping if the steam generator pressure decreases low enough for the flashing to occur. Should this occur additional hot water will be expelled into the steam generator with the potential to increase secondary-to-primary heat transfer. This additional modeling detail is necessary to accurately model the main feedwater boundary condition."

ROTSG Model

The replacement steam generators (ROTSGs) are modeled in RELAP5 with the same nodalization as the original steam generators. All of the RELAP5 model input data have been recalculated for the ROTSGs based on data supplied by B&W Canada, the manufacturer. Refer to Appendix A for a description of the design differences.

2.1.3 Validation of Code and Model

In order to properly simulate the transient response of a reactor system to a LOCA, it is necessary to represent the thermal-hydraulic processes which occur during the transient. The RELAP5 code provides the generic tool which is used to model these processes. Furthermore, the code user must provide geometric and component input which describes the reactor system of interest. The Oconee RELAP5 model satisfies this requirement. Finally, the code user must ensure that both the code and the model have been properly validated before the results of any transient simulation should be considered valid.

This section summaries the validation efforts completed to date for both the RELAP5 code and the Oconee RELAP5 model.

Code Validation

Efforts have been underway to validate the RELAP5 family of computer codes since the inception of the code. The code has been used extensively by the code developers for both pre-test and post-test predictions of the Semiscale and LOFT experiments. RELAP5 validation was also the primary impetus of the International Code Assessment Program (ICAP) which lead to the creation of RELAP5/MOD3. Because the various efforts performed thus far to validate the RELAP5/MOD2 code are generally well recognized, a summary of these activities is not provided in this document.

B&W has validated RELAP5/MOD2-B&W against numerous experiments performed at the Multiloop Integral System Test (MIST) facility. This facility was originally designed and built to provide integral system test data from a scaled test facility which emulates a B&W Reactor Coolant System. The MIST facility was built as part of a multiphase program initiated in 1983 to address small break LOCA licensing issues which are unique to B&W plants. The data acquired from the MIST facility was used to resolve these issues and provide data for code benchmarking. These benchmarks demonstrated the small break LOCA predictive capabilities of the code and provided B&W with an indication of what models and correlations needed to be strengthened. A summary of the MIST program is provided in Reference 2-7.

Model Validation

The RELAP5 code has been used by Duke Power Company since 1985. The initial RELAP5 efforts focused on developing and testing a RELAP5 model of Oconee Nuclear Station. This model has since been applied to numerous LOCA and non-LOCA transient simulations in support of emergency procedure development and simulator validation. The most recent application of the model is the mass and energy release simulations described in this document.

Validation of the Oconee RELAP5 model is necessary to ensure that the mass and energy predictions described in this document are reasonable. Ideally, the model should be validated against LOCA test data from a full scale prototypical experimental facility, or from an actual LOCA, since the model is being used to simulate LOCA transients. Realistically, data of this nature does not exist so it is impossible to validate the Oconee model against similar LOCA transients. Data are, however, available

from operational transients. These transients therefore provide the only realistic way to validate the Oconee RELAP5 model.

Several formal and informal benchmark analyses have been performed to validate the Oconee model. Three of the formal benchmarks validate features of the model which are most applicable to the LOCA simulations. These benchmarks are discussed briefly in this section to provide an overview of the validation efforts performed to date on the Oconee RELAP5 model.

The first benchmark of interest compares predictions from the Oconee RELAP5 model to reactor coolant pump (RCP) coastdown data obtained during hot zero power startup tests. The intent of this benchmark is to validate the single phase reactor coolant pump model and to evaluate the single phase loop hydraulic losses. The RCPs are generally tripped shortly after loosing subcooling. Even if they are left running in a saturated system, pump performance degrades considerably from the single phase performance. Thus the validation efforts of the pump model are marginally applicable to the LOCA simulations. The loop hydraulic losses do, however, have a significant impact on the predicted flow rates around the loop. These losses are particularly important during a large break LOCA since flow velocities are generally several times the normal velocity during the blowdown phase. Favorable comparisons between test data and the Oconee RELAP5 model results indicate that the loop hydraulic losses are consistent.

Another benchmark has been performed to characterize the discharge of the core flood tanks (CFT) following a large LOCA. Once again, data obtained from tests of the CFTs performed during startup are compared to predictions of the accumulator model used in the Oconee RELAP5 model. This benchmark compares the measured pressure and level in the tank during the test to code predictions to validate the RELAP5 accumulator model and the hydraulic losses in the CFT surge line. The performance of the CFTs will have an impact on all of the classic phases of a large LOCA. The results of this benchmark effort confirm that the accumulator component in the Oconee RELAP5 adequately represents the performance of the actual CFT during discharge.

Finally, analyses have been performed to predict steady-state natural circulation flow rates at various power levels and at different steam generator levels. These predictions are compared to natural circulation data obtained from sister plants to validate the implementation of the RELAP5/MOD2-B&W EFW model into the Oconee RELAP5 model. As discussed in Section 2.1.2, operation of the EFW System influences the location of the steam generator thermal center which has a direct impact on the resulting natural circulation flow rate. Since primary-to-secondary heat transfer is generally required to cool and depressurize the RCS following a small break LOCA, it is important to accurately predict the location of the secondary thermal center.

The results from this study indicates that the natural circulation flow predicted by RELAP5 for the various combinations of test conditions are slightly high compared to data, but trended the data favorably. Furthermore, the model predicts a degradation in RCS flow as the steam generator levels are lowered, although the decrease is not as great as shown by the data. Since these discrepancies are sufficiently small, the results of this benchmark validate the application of the RELAP5/MOD2-B&W EFW model for predicting single phase natural circulation flow.

2.2 BFLOW - Long-Term Large Break LOCA Mass and Energy Release

2.2.1 Code Description

2.2.1.1 Overview

The BFLOW code calculates the long-term liquid and steam mass flow rates out of a large cold leg break. In the long-term phase of a large break, the Reactor Coolant System and containment conditions [

] BFLOW calculates the resulting break mass flow rate and quality.

The Oconee reactor vessel (Figure 2.2-1) contains eight internal vent valves which vent steam and water directly from the upper plenum to the upper downcomer and then to the broken cold leg. Therefore, in B&W plants like Oconee there is no need to vent coolant through the steam generators and the intact or broken loops to the break. In the post-reflood phase of a cold leg break, the Oconee reactor vessel should be in a "boiling-pot" mode, as illustrated on Figure 2.2-2. Low Pressure Injection (LPI) flow enters the vessel through the core flood tank nozzles in the upper downcomer. The nominal LPI flow rate is 3000 gpm for one train, or less if some of the flow is diverted to the Reactor Building Spray System. The LPI water comes directly from the borated water storage tank (BWST) or from the Reactor Building sump via the LPI coolers; in either case, the injection water should be significantly subcooled. LPI flow is sufficient to maintain the downcomer full of subcooled water up to the elevation of the break. Some of the injection water flows into the vessel lower head and up through the core, while the remainder spills out the break. The water in the core flows up and eventually begins to boil due to the residual decay heat. A two-phase mixture or saturated steam leaves the core and enters the vessel upper plenum. From

there, the fluid flows through the reactor vessel vent valves and back into the vessel downcomer, where it goes out the break or circulates back through the vessel.

In steady-state the head of water in the downcomer must equal the head of water in the inner vessel plus the pressure drop across the inner vessel and the reactor vessel vent valves. The downcomer level can never appreciably exceed the bottom of the cold leg since the injection water will predominantly spill out the double-ended break. The vent valves are located significantly higher than the bottom of the cold leg. For liquid to flow through the vent valves there must be voiding in the reactor core and upper plenum regions so that the downcomer and inner vessel heads can balance. Thus a cold leg break at a B&W lowered-loop plant is characterized by continued steaming from the reactor core. The relative amounts of steam and liquid flowing out the break are important because the vapor flow adds energy directly to the containment atmosphere and is therefore deleterious with respect to the Reactor Building temperature response. At higher core power levels the steam production rate in the core is sufficient to carry liquid up through the vent valves with the steam. If the core power level is very low then all of the vent valve flow is steam. In [

]. BFLOW determines the break quality - the fraction of the total break flow which is steam.

The two phase flow in the reactor vessel upper plenum reaches the outlet annulus (see Figure 2.2-1) through two main flow paths. One flow path consists of the large 22 inch and 34 inch flow holes in the top of the plenum assembly. The bottom of these flow holes roughly corresponds to the bottom of the reactor vessel vent valves. It should be noted that none of these flow holes lines up with the eight reactor vessel vent valves. The other flow path consists of the 3 inch holes in the plenum assembly reinforcing plates. Each hot leg nozzle has a reinforcing plate in its flow stream, with 24 three inch holes per plate. Since the large flow holes do not line up with the reactor vessel vent valves, liquid flowing through the big 22 inch and 34 inch flow holes in the top of the plenum assembly will hit the core support shield and fall into the outlet annulus. Thus, the geometry of the outlet annulus will act as a separator, collecting liquid and only allowing steam to pass through the reactor vessel vent valves. Liquid accumulating in the outlet annulus flows back into the upper plenum through the 3 inch holes in the reinforcing plates.

The BFLOW upper plenum model assumes

In the following paragraph and in other sections of this report the manual opening of the boron dilution line is described. Opening the boron dilution line is no longer credited in the methodology. The original text is maintained so that the results presented can be understood.

Provisions are made at Oconee for opening a flow path from the hot leg to the sump within 9 hours of a large LOCA. This flow path, known as the boron dilution line, ensures continuous liquid flow through the core and prevents excessive boron concentrations from developing in the reactor vessel (see Figure 2.2-3). The effects of the opening of this flow path are accounted for in BFLOW.

The BFLOW code uses somewhat different solution techniques for the break conditions, depending on whether or not the flow through the reactor vessel vent valves is two-phase. Both techniques are described below.

2.2.1.2 Solution Technique for Two-Phase Vent Valve Flow

An iterative solution technique is used to determine the break flow boundary condition if the vent valve flow is two-phase. This technique is shown on Figure 2.2-4 and summarized below.

Vent Valve Steam Condensation Rate

Downcomer Enthalpy

Reactor Vessel Nodal Energy Balance

Reactor Vessel Nodal Momentum Balance

Nodal Void Fraction

Upper Plenum Recirculating Flow Rate

Reactor Vessel Vent Valve Flow

Break Conditions

2.2.1.3 Solution Technique for Single-Phase Vent Valve Flow

2.2.2 Simulation Model

2.2.3 Validation of Code and Model

2.3 RETRAN-02 MOD5.1DKE - Secondary System Mass and Energy Release

2.3.1 Code Description

The RETRAN-02 MOD5.1 DKE code is a modified version of the NRC-approved RETRAN-02 MOD005.0 code (References 2-16 and 2-17). The version used in this topical report differs from the NRC-approved version in two ways. Firstly, EPRI released the MOD5.1 version to correct errors and other internal coding problems in the MOD5.0 version. Secondly, two additional error corrections to the MOD5.1 version have been inserted to obtain the MOD5.1DKE version. These two error corrections, which will be included in future versions of the RETRAN code are:

- Parameter LPOOL in subroutine INPUT was changed from 125,000 to 175,000. This modification was necessary to allow a large RETRAN input deck to execute.

- The DO statement in loop 920 of subroutine GENTRN is changed from 'DO 920 IGS = 1,10' to 'DO 920 IGS = 1,30'. This correction enhances convergence in the generalized transport model.

Since the RETRAN version used is NRC-approved with the exception of error corrections, it is concluded that this report need not justify the validity of the RETRAN code itself. Therefore, details regarding the theory of the RETRAN code are left to the references.

2.3.2 Simulation Model

The Oconee RETRAN Model is documented in Reference 2-18. The nodalization diagram (Figure 2.3-1) is generally applicable for non-LOCA transient simulation. For certain transients, transient-specific modeling revisions and/or additions must be included. For the steam line break mass and energy release analysis, model revisions are required to model the break, to model a containment backpressure, to model the asymmetry and thermal mixing in the reactor vessel, and to add detail to the modeling of the condensate and feedwater piping and related pumps and controls. Additional details are provided in Section 5.2.1. Other than these modifications, the RETRAN model used for the steam line break analysis is the same as that currently under NRC review per Reference 2-18.

2.3.3 Validation of Code and Model

Validation of the RETRAN code and the Oconee RETRAN Model for transient analysis was submitted to the NRC as part of Reference 2-18. These validation efforts employed a range of plant transient data and demonstrated the capability of the methodology. Specific validation of a steam line break transient in a B&W NSSS has not been performed due to an absence of applicable data. There have been no representative events in B&W plants, and very limited scaled test facility data Consequently, no validation results are presented to support the steam line break analyses in this report. The validity and conservatism of the steam line break analysis is ensured by careful selection of initial and boundary conditions. These modeling details have been conservatively applied to maximize the worst case mass and energy release scenario. Further discussion is provided in Chapter 5

2.4 FATHOMS/DUKE-RS - Containment Response

2.4.1 Code Description

2.4.1.1 Overview

The FATHOMS/DUKE-RS computer code is used to determine the containment response to high-energy line breaks. FATHOMS/DUKE-RS is a derivative of the FATHOMS code (Reference 2-19), developed by Numerical Applications, Inc., for performing thermal-hydraulic analysis of nuclear power plant containment and auxiliary buildings. The FATHOMS code, in turn, is derived from the COBRA-NC thermal-hydraulic code (Reference 2-20). It solves the conservation equations for mass, energy, and momentum for multi-component, two-phase flow. These conservation equations are solved numerically on a finite-volume mesh made up of numerous computational cells. The code features a nodalization scheme in which lumped parameter, one-, two-, or three-dimensional analysis or any combination of these may be performed. Velocity fields are included for three phases:

- Vapor / non-condensable gas mixture
- Continuous liquid
- Liquid droplet

Interfacial heat transfer, mass, and momentum transfer are modeled; the effects of two-phase slip on pressure drop are also accounted for. Two temperature fields are included:

- Vapor / non-condensable gas mixture
- Liquid / droplet mixture

These two mixtures may be in thermal non-equilibrium within the same calculational volume. A total of eight non-condensable gases may be modeled; mass transport equations are solved for each component of the non-condensable gas mixture.

FATHOMS includes finite-difference conduction models for passive thermal conductors. Concrete walls and structural steel within the Reactor Building are simulated with these models. These conductors may have the following geometrical shapes:

- Flat plate (wall)
- Cylindrical tube
- Solid rod

Heat is conducted through the thickness or radius of each of these conductor models. They may be thermally connected to fluid volumes on either surface. A variety of condensation heat transfer coefficient options is available for the surfaces of these thermal structures. Heaters and coolers may also be modeled within calculational volumes in FATHOMS. Any volumetric heat sources or cooling equipment may be simulated with these models.

Each FATHOMS analysis model consists of a network of volumes connected by flow paths. Typically, each volume represents a room or area within the Reactor Building. Any volume may be partitioned into a two- or three-dimensional mesh for which mass, energy, and momentum balance are maintained. This allows the fluid property distribution within a room to be calculated and flow patterns within a room to be predicted. The flow paths represent doorways, pipes or piping systems, vents, etc. Within each subdivided volume, the momentum equations for the vapor/gas mixture, droplets, and liquid are solved for all flow paths, with each phase having its own velocity. A variety of mechanical components to simulate various plant equipment can be modeled along these flow paths. These include:

- Valves
- Pumps
- Heat exchangers
- Spray nozzles
- Volumetric fans

Boundary conditions for the containment model are implemented using these flow paths. The user may specify mass and energy sources or sinks with these boundary conditions. The boundary conditions may be coupled in groups to model flowstreams which are sent to different calculational volumes. The user has the ability to model cooling and spray systems associated with containment safety features using these options.

The FATHOMS code was adopted by Electric Power Research Institute (EPRI) in 1989 for a project to provide interested nuclear utilities with an efficient, state-of-the-art containment analysis code. The code was renamed GOTHIC and, together with a pre/post-processor package, has been refined even further. The FATHOMS/DUKE-RS source code is based from FATHOMS 2.4, the version of FATHOMS existing prior to the inception of the EPRI project. The code had undergone extensive testing and

benchmarking at that stage. This version of the code is sufficient to provide accurate answers for dry containment modeling.

2.4.1.2 Models and Features Unique to FATHOMS/DUKE-RS

Modifications were made by Duke Power Company to implement a long-term mass and energy release model into FATHOMS. This customized version is referred to as FATHOMS/DUKE-RS. The long-term mass and energy release modeling capability allows the simulation of a hot or cold leg pipe break during the post-reflood stage of a loss-of-coolant accident (LOCA). A series of flow paths may be set up with this capability to model core injection water, decay heat, and flow spilling back out the pipe break after the Reactor Coolant System and the Reactor Building have equalized in pressure.

In the case of a hot-leg break, the core decay heat is [

]

For a cold leg break, all of the injection flow will not reach the core, so different algorithms are utilized. The output from the BFLOW code (Section 2.2) is used in this case to determine the quality of the water exiting the break. The BFLOW [

]

Another coding change allows the desired flow in a boundary condition to be input as a function of the pressure in the discharge volume of a coupled boundary condition rather than as a function of a time. The pressure used is the mid-volume elevation pressure; this includes the contribution of any static head of water. This model is used in long-term small break LOCA simulations for a simple HPI pump model.

Other minor modifications have been made to enhance the printed code output and to provide for certain new modeling capabilities. These include an option to provide heat transfer rates to the heater/cooler models as a function of pressure and humidity and an option to terminate any simulation when a given trip occurs (problem end trip). Unsupported code in the FATHOMS/DUKE-RS routines (coding left over from the COBRA-NC modeling schemes) has been removed.

2.4.2 Simulation Model

This section gives a brief description of the simulation models used for the FATHOMS containment analyses. A nodalization diagram of the FATHOMS base model is shown in Figure 2.4-1.

Nodalization

Large dry containments have traditionally been modeled with one large volume, with all pressures and temperatures being considered volume-averaged quantities. The Oconee FATHOMS base model uses a similar modeling approach. Since FATHOMS has only one set of energy equations for the liquid phase, all droplets and continuous (pooled) liquid within any volume or sub-volume is considered to be at the same temperature. This is inadequate for a containment volume, since the water in the sump region will typically be at a different temperature than any water existing in droplet form in the containment atmosphere or in a film on the building walls or internal structures. Therefore, a separate sump volume is represented in the FATHOMS base model. This consists of the lowest 6.35 feet of the building space; the remaining 195 feet represents the remainder of the containment space. These two nodes are referred to here as the atmosphere region and the sump region; these are analogous to the similar regions in a CONTEMPT containment model, but are actually in different calculational volumes in a FATHOMS model, where they are actually in the same volume in a CONTEMPT model. The sump temperature refers to the liquid temperature in the sump volume, and the vapor temperature refers to the temperature of the vapor phase of the atmosphere volume. There is a small vapor space in the sump region, but it is so small that it would not affect the temperature of the entire containment vapor space if it were factored in.

The containment free volume (CFV) and passive heat sink data used for the base model include a 1% conservative allowance for uncertainty.

Initial Conditions

The initial pressure and temperature assumed in the Reactor Building varies with each analysis. For the peak pressure analysis, it was discovered that variations in the initial air mass had a large impact on the calculated peak pressure. A higher initial air mass leads to higher calculated peak pressures. The initial air mass can be increased by assuming a low initial building temperature The early time at which the peak pressure occurred (within 30 seconds of each case executed) meant that the increased initial air mass had a larger effect than the decreased heat transfer into the passive heat structures at a high initial temperature. Therefore, conservatively low initial temperatures are assumed in the peak pressure analyses. In all long-term analyses, as well as the steam line break analysis, a high initial building temperature of 125°F is assumed.

Likewise, the initial building humidity is assumed to be at 0% to maximize the air mass in the peak pressure analyses, and at 100% in the long-term analyses. The initial building humidity has very little impact on the results of either set of analyses.

A high initial building pressure of 1.5 psig is conservatively assumed for all peak pressure analyses. This maximizes the calculated peak building pressure. In the long-term analyses, this initial pressure has no significant impact on results, so an initial pressure of 0 psig is assumed.

Heat Structures

The concrete and steel heat structures present in the Oconee containment building are modeled using the heat slab models in FATHOMS. They are modeled with an insulated boundary condition for one surface. For the building walls, dome, and base, it would be many days before significant quantities of heat would conduct through the thickness of the structure, so this insulated boundary condition is used on each outer surface. The remaining structures, all internal to the Reactor Building, are exposed to the containment atmosphere on all sides, but the entire surface area is combined onto one surface for the FATHOMS heat slab model. The zero heat fluxes on the second surface here represent a symmetry condition at the midplane of the structures. The slabs are modeled in full thickness, however, so that the total volume will be conserved. All structures are initially at 125°F throughout their thicknesses, the same as the initial building temperature.

All structures are modeled as walls, with one-dimensional heat transfer occurring in the direction perpendicular to the structure surface. The noding divisions are finer closer to the surface, where larger temperature gradients are expected.

The Uchida heat transfer correlation is used for heat transfer between the subcooled surfaces of the structure and the containment atmosphere. The standard heat transfer logic present in FATHOMS is used for other heat transfer. For the building floor, which would be covered with a pool of water, a constant heat transfer coefficient of 20 Btu/hr-ft^2-°F is assumed.

Reactor Building Cooling Units (RBCUs)

The RBCUs are modeled using the standard cooler model in FATHOMS. Input parameters are chosen from design values for the coolers. The air flow for each cooler modeled is 54,000 cfm. The heat removal characteristics of the coolers vary with each analysis. Heat transfer rates are given in million Btu/hr. The coolers are initiated at 9 psig, plus a 78 second delay time; these setpoints include allowances for instrument uncertainties and all other associated delays.

Reactor Building Spray System

The Reactor Building Spray System is modeled with a single junction for injection from the borated water storage tank (BWST), or with a set of junctions for sump recirculation mode. The nominal flow rate for injection from the BWST is 1500 gpm per train, while the nominal flow rate during recirculation mode is 1000 gpm per train. Both these rates are usually adjusted for instrument error, depending on the analysis goal.

All spray is assumed to enter the containment atmosphere with an average droplet size of 700 mm. No spray efficiency parameter is necessary in the FATHOMS flow models. The spray is initiated at 30 psig, with a delay time of 92 seconds for the spray header fill time.

2.4.3 Validation of Code and Model

Code Validation

The FATHOMS code authors, Numerical Applications, Inc., subjected the FATHOMS code to extensive benchmarking efforts throughout the process of refining and updating the code. The code has been compared against analytic solutions as well as experimental data (Reference 2-21). This set of 25 tests includes comparisons with data from the Battelle Frankfurt HDR containment experimental facility and specially designed analytic problems to test specific components and code capabilities. The FATHOMS

code is an extension of the COBRA-NC code and retains all of the modeling capabilities of COBRA-NC. The COBRA-NC code was also benchmarked against analytic solutions and experimental data, including steam and steam/water blowdowns in the Battelle Frankfurt Model Containment, Containment Analysis Standard Problems 1, 2, and 3, blowdown tests in the HDR facility and hydrogen distribution tests in the Battelle Frankfurt Model Containment (Volume 7, Reference 2-20). These verification results can be reasonably assumed to the FATHOMS code as well. All efforts have validated the code's ability to accurately predict pressure and temperature transients within containment structures.

The FATHOMS code has also been tested by Duke Power Company by comparing code results with data from steam injection tests performed at the Carolinas Virginia Tube Reactor (CVTR) (Reference 2-22). This set of data is particularly appropriate for this purpose because it applies to the pressurization of a large, dry containment building similar to that of Oconee. These experiments studied the pressurization of the CVTR building due to a saturated steam blowdown, with passive heat structures and building spray aiding in the de-pressurization. A FATHOMS model was developed for the CVTR containment building, and simulations of three steam blowdown tests were performed. The pressure and temperature responses for all three tests are predicted reasonably well for all three cases. See Supplement 2 for additional details.

Model Validation

The FATHOMS code has been used by Duke Power Company since 1987. Once a model of the Oconee containment was put together, efforts to compare the results obtained from that model were undertaken. Since no test data exists for the pressurization rate of the Oconee Reactor Building following a high-energy line break, the only means of comparison of the FATHOMS model are the results of the CONTEMPT analyses performed by Babcock and Wilcox (B&W) for the Oconee U. Mass and energy release data is available for several different break sizes; the one of main interest is the 14.1 ft^2 hot leg break, representing a double-ended guillotine break of the hot leg. This transient gives the highest peak pressure of any break, according to the B&W CONTEMPT analyses.

The Oconee FATHOMS model developed by Duke Power uses more recent building free volume and heat sink data, so the data used in the UFSAR analyses for these features was substituted into the input deck for the model validation effort. Also, the appropriate condensation heat transfer correlation for the passive heat structures within containment (described in Chapter 15 of the Oconee UFSAR) was implemented. Initiation times for the Reactor Building Cooling Units (RBCUs) and the Building Spray System (BS) were also adjusted to match the B&W assumptions.

The results of this comparison (Figure 2.4-2) showed that FATHOMS predicted a remarkably similar pressure transient following this large-break LOCA as that predicted by CONTEMPT. Two pressure peaks were predicted by both codes: an initial peak occurred within 20 seconds, and a second, larger pressure peak occurred at about 90 seconds. The rate at which the pressure decreased following the second peak was slightly greater in the FATHOMS analysis than the CONTEMPT analysis, but this was expected due to differences in the RBCU models in the two codes. This comparison showed that given similar models, the results of the FATHOMS containment analysis code compare very favorably with the CONTEMPT code. Since FATHOMS has many more modeling capabilities, is much easier to use, and is considered to be state-of-the-art in containment transient analysis, it is the code of choice for this analysis effort.

2.5 RETRAN-3D MOD003.1/DKE Code Description

RETRAN-3D (Reference 2-23) was developed by Computer Simulation & Analysis, Inc. for EPRI to enhance and extend the simulation capabilities of the RETRAN-02 code. RETRAN-3D has many new and enhanced capabilities relative to RETRAN-02, in particular, a 3-D kinetics core model, improved two-phase models, an improved heat transfer correlation package, and an implicit numerical solution method. Most of the capabilities of the RETRAN-02 code have been retained within RETRAN-3D as options, except for a limited number of models and correlations that were not in use. The NRC SER for RETRAN-3D is dated January 25, 2001 (Reference 2-24). The SER includes limitations and conditions on the use of the code for licensing applications. RETRAN-3D MOD003.1/DKE is the Duke-specific version that includes code revisions described in Appendix A. Refer to Appendix A for details on the application of RETRAN-3D MOD003.1/DKE for steam line break mass and energy release analyses. Appendix B is an evaluation of the conditions and limitations in the RETRAN-3D SER."

2.6 GOTHIC 7.0/DUKE Code Description

The GOTHIC 7.0 code (Reference 2-25) was developed by Numerical Applications, Inc. for EPRI as the next version of the GOTHIC family of codes for thermal-hydraulic analysis of nuclear power plant containment and auxiliary buildings. The GOTHIC code was developed from the FATHOMS code (Reference 2-19), which in turn was developed from the COBRA-NC code (Reference 2-20). Similar to FATHOMS, GOTHIC 7.0 solves the conservation equations for mass, energy, and momentum for multi-component two-phase flow. GOTHIC 7.0/DUKE is the Duke-specific version that includes code

revisions as described in Appendix C. Refer to Appendix C for details on the application of GOTHIC 7.0/DUKE for LOCA and steam line break containment response analyses.

2.7 References

2-1 B&W-10164P, Revision 1, "RELAP5/MOD2-B&W - An Advanced Computer Program for Light-Water Reactor LOCA and Non-LOCA Transient Analysis", Babcock & Wilcox, October, 1988.

2-2 Letter from A. C. Thadani (NRC) to J. H. Taylor (BWFC), "Acceptance for Referencing of Topical Report BAW-10164P, Revision 1", April 18, 1990.

2-3 BAW-10005, Revision 1, "Internals Vent Valve Evaluation", Babcock & Wilcox, June, 1970.

2-4 Letter from J. D. Agar (B&W) to G. B. Swindlehurst (Duke), "RC Loop Hydraulic Losses", transmittal of B&W Document 51-1176025-00, dated August 14, 1989.

2-5 B&W Document #12-1132513, "Supporting Data for AFW (EFW) Models - Auxiliary Feedwater Penetration", Babcock & Wilcox, April 1982.

2-6 K. G. Condie et al., "Flooding, AFW Penetration and Visual Investigations in Scaled Once Through Steam Generator Facilities", Idaho National Engineering Laboratory, July, 1988.

2-7 Gloudemans et al., "Multiloop Integral System Test (MIST): Final Summary", NUREG/CR-5395, April, 1991.

2-8 Yeh, Hsu-Chieh and Hochreiter, L. E., Mass Effluence During FLECHT Forced Reflood Experiments, Nuclear Engineering and Design Volume 60, 1980, pp. 413-429.

2-9 Collier, John G., Convective Boiling and Condensation, McGraw-Hill Book Company, 2nd Edition, 1972.

2-10 Cohen, Paul, Water Coolant Technology of Power Reactors, Gordon and Breach Science Publishers, New York, 1969.

2-11 Brater, E. F. and King, H. W., Handbook of Hydraulics, McGraw-Hill Book Company, 6th Edition, 1976.

2-12 Kreith, Frank, Principles of Heat Transfer, Harper and Row, 3rd Edition, 1973.

2-13 NUREG/CR-2289, Concurrent Steam/Water Flow in a Horizontal Chamber, Lim, Bankoff, Tankin, and Yuen, U.S. Nuclear Regulatory Commission, August, 1981.

2-14 EPRI NP-3989-SR, Revision 1, A Full Range Drift-Flux Correlation for Vertical Flows, Electric Power Research Institute, September, 1986.

2-15 NSAC-107, An Assessment of Eight Void Fraction Models for Vertical Flows, Electric Power Research Institute, December, 1986.

2-16 RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCMA Revision 4, EPRI, November 1988

2-17 Letter, A. C. Thadani, (NRC), to W. J. Boatwright, Texas Utilities Electric Company, RETRAN-02 MOD5.0 NRC SER, November 1, 1991

2-18 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision 3 Duke Power Company, September 2004

2-19 CAP-Containment Analysis Package (FATHOMS 2.4), Numerical Applications, Inc., October 10, 1989

2-20 NUREG/CR 3262, "COBRA-NC: A Thermal Hydraulics Code for Transient Analysis of Nuclear Reactor Components", Volumes 1 to 7, Wheeler, et.al., Pacific Northwest Laboratory, April 1986.

2-21 "CAP Test Problem Notebooks: Tests 1-25", Numerical Applications, Inc., February 1988.

2-22 "Simulated DBA Tests of the Carolinas Virginia Tube Reactor Containment - Final Report", Schmitt, et.al., Idaho Nuclear Corp., December 1970.

2-23 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI, NP-7450(A), Volumes 1-4, Revision 5, July 2001

2-24 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

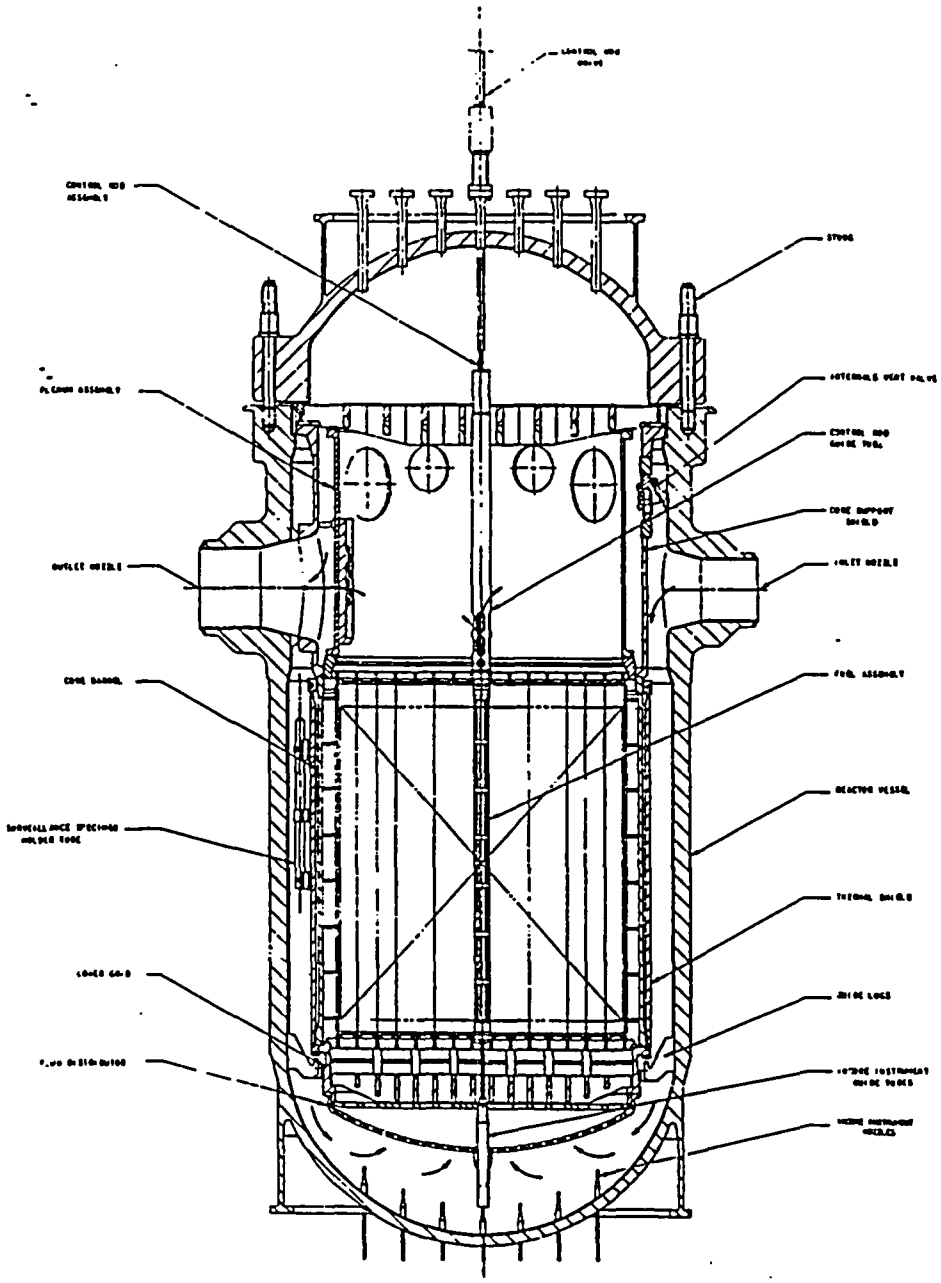
2-25 GOTHIC Containment Analysis Package Version 7.0, NAI-8907-02, Revision 13, July 2001

FIGURE 2.1-1

ONS RELAP5 Model

[(page intentionally blank)]

FIGURE 2.2-1



REACTOR VESSEL AND
INTERNALS GENERAL ARRANGEMENT
OCONEE NUCLEAR STATION

FIGURE 2.2-2
Cold Leg Break - Post-Reflood Stage

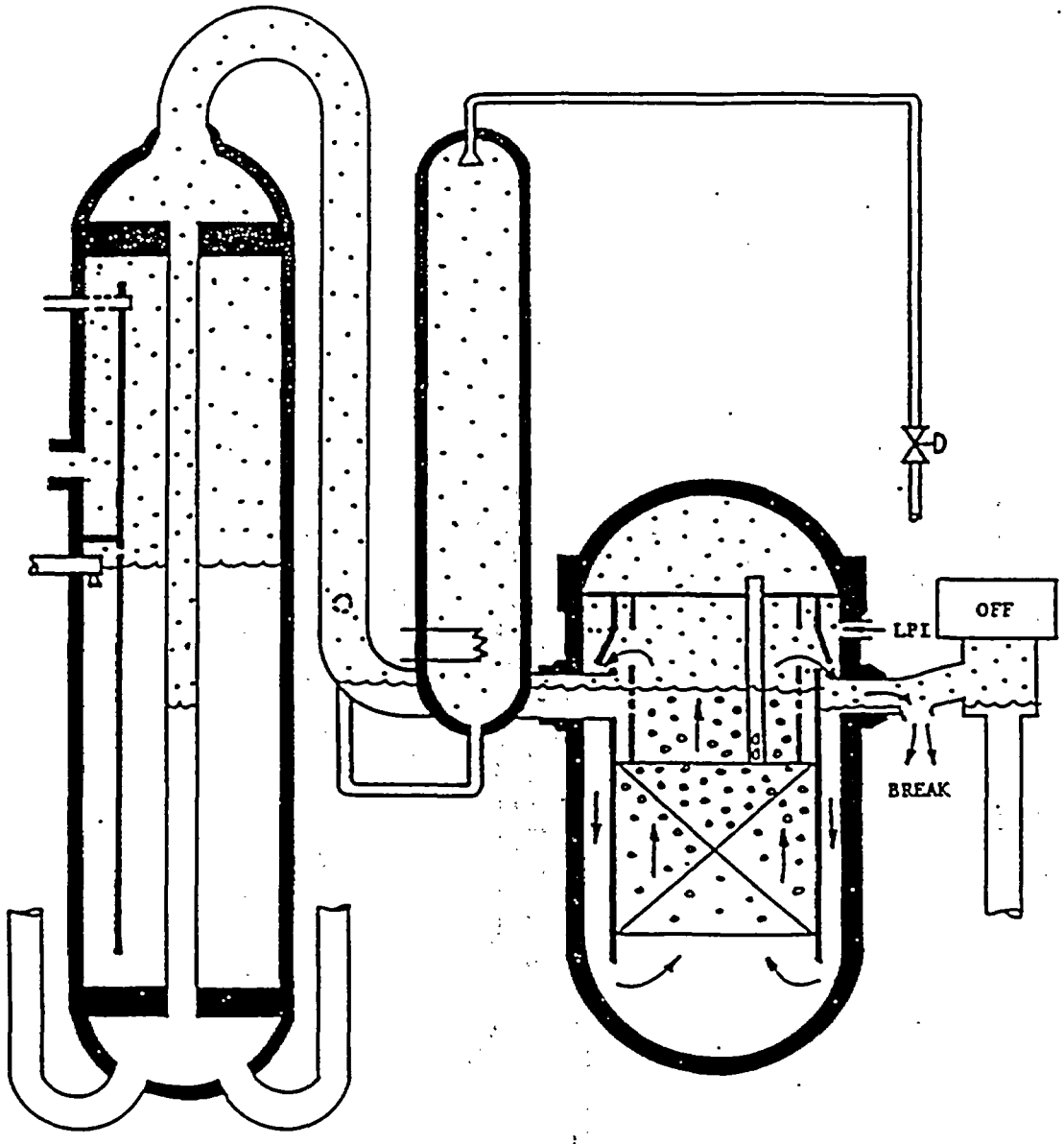


FIGURE 2.2-3
Boron Dilution Flowpath

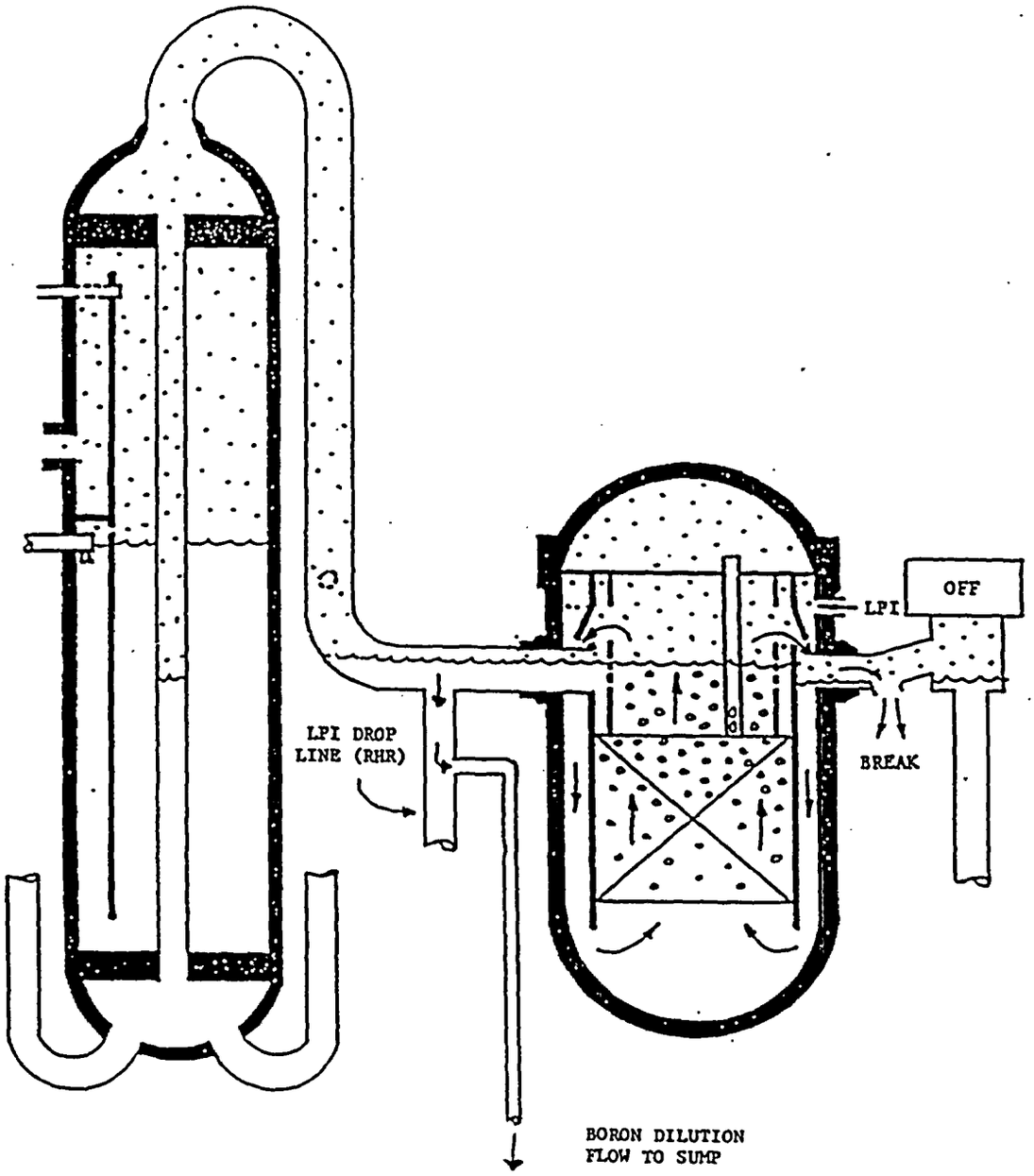


FIGURE 2.2-4
Two-Phase Vent Valve Flow - Solution Technique

FIGURE 2.2-5
Core and Upper Plenum Nodes

FIGURE 2.2-6
Downcomer Mass Balance

FIGURE 2.2-7
RCS Cold Leg Nozzle

FIGURE 2.2-8
Reactor Vessel Nodal Energy Balance

FIGURE 2.2-9
Reactor Vessel Nodal Momentum Balance

FIGURE 2.2-10
Overall System Mass and Energy Balance

FIGURE 2.2-11
Break Flow Solution Technique

FIGURE 2.2-12
Void Correlation Comparison

FIGURE 2.3-1

[(pages 2-41- 2-50 intentionally blank)]

FIGURE 2.4-1

OCONEE CONTAINMENT

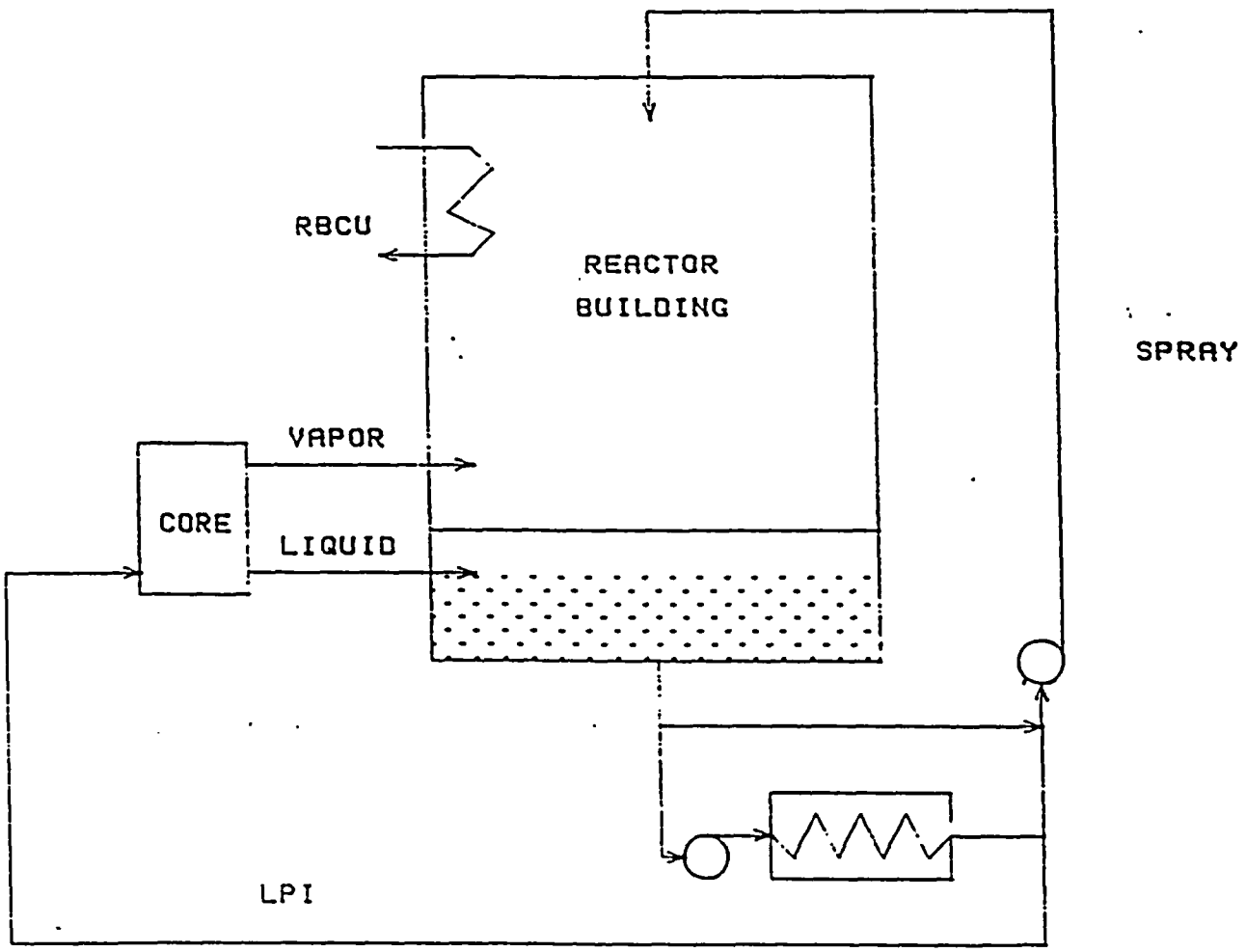
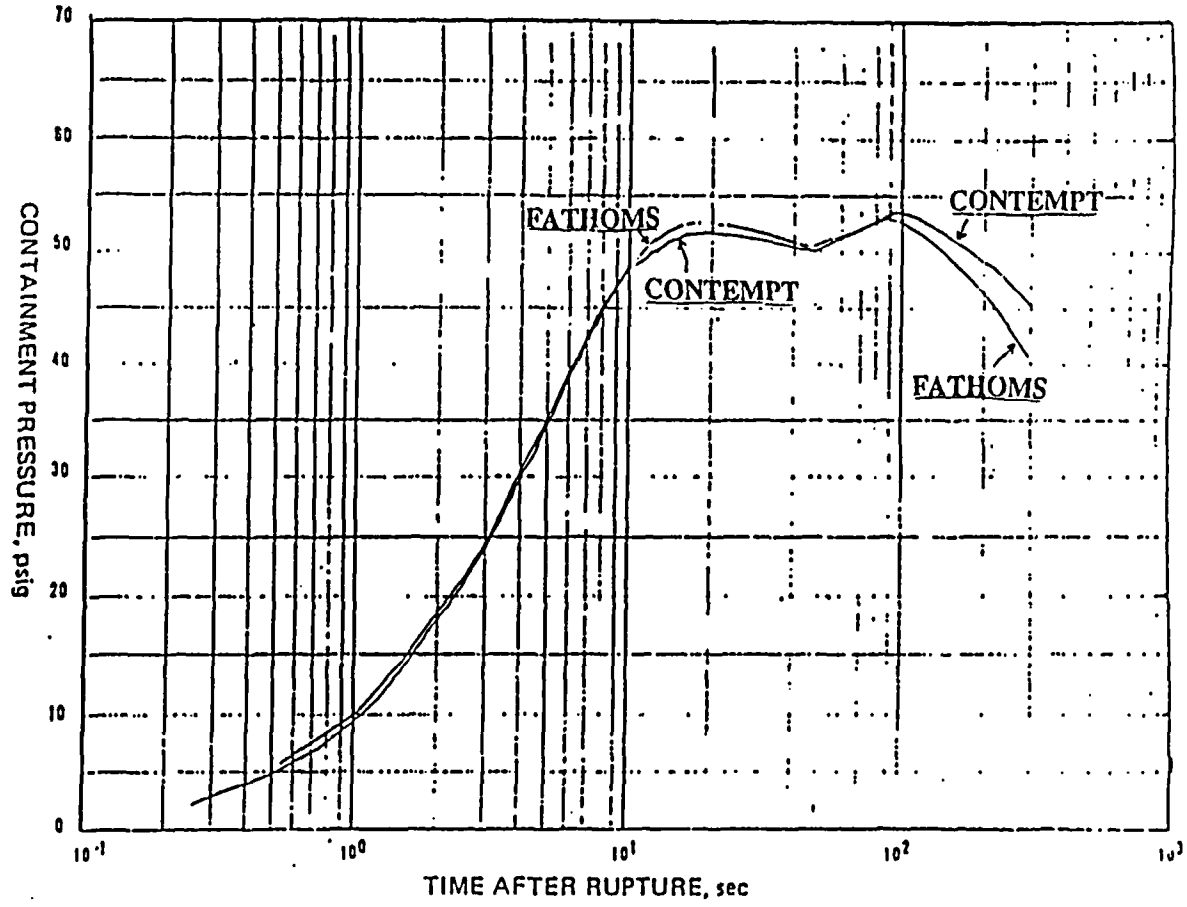


FIGURE 2.4-2



FATHOMS - CONTEMPT Comparisons For
Oconee Large Break Post-LOCA Pressure

3. LARGE BREAK LOCA MASS AND ENERGY RELEASE ANALYSES

3.1 Overview

Large break LOCA analyses are performed to generate mass and energy release boundary conditions for 1) determining the peak pressure following a double-ended rupture of the RCS piping and 2) determining the long-term containment response to a double-ended rupture of the RCS cold leg piping. The RELAP5/MOD2-B&W computer code is used to generate these boundary conditions for all peak pressure analyses, and through the blowdown and post-reflood phases of the long-term analysis. Break flow is calculated by the BFLOW code once quasi-steady-state conditions have evolved in the RCS (~30 minutes). The stored energy in the RELAP5 heat structures is tracked during the simulation and a methodology for releasing this energy to containment during the remainder of the analysis is developed. The stored energy in the primary and secondary liquid is also released to the containment using this methodology, which is described in Section 3.3.3.4.

To ensure conservatism, the initial conditions and boundary conditions chosen for all LOCA analyses generally conform to the ANSI/ANS-56.4-1983 (Reference 3-1). These initial conditions are the same for all large and small break LOCA analyses presented in this document and are described in Section 3.2. The assumed boundary conditions vary depending upon the stated objectives of the analysis. There are, however, many boundary conditions which are the same for both the peak pressure and the long-term mass and energy release analyses. The boundary conditions for these analyses are discussed in detail in Section 3.3.

For the peak pressure analysis, a series of break sizes, locations and boundary conditions are examined to determine the combination which yields the peak containment pressure following a rupture of the RCS piping. Four double-ended guillotine pipe break locations are examined; hot leg at the reactor vessel outlet (14.1 ft^2), hot leg at the steam generator inlet (14.1 ft^2), cold leg pump discharge (8.55 ft^2), and cold leg pump suction (8.55 ft^2). Split breaks and breaks of lesser flow areas are not examined because it is well recognized that these breaks will not yield the peak containment pressure. The mass and energy release boundary conditions generated during these analyses are used as input to the FATHOMS code to identify the most limiting break area and location.

Once the limiting break area and location has been identified, a series of sensitivity studies are performed. The first sensitivity compares break flow rates calculated by the Ransom and Trapp critical flow model (RELAP5 model) to flow predicted by the Moody/Henry-Fauske critical flow models. Critical flow discharge coefficients are determined which yield comparable results. Next, the availability of off-site power is examined. The availability of offsite power determines when the reactor coolant pumps (RCP) trip, the delays associated with emergency core cooling (ECCS) equipment, and whether main feedwater is available. Finally, several sensitivities are performed to examine the impact of the assumed containment back-pressure and to examine the impact of a non-zero refill time. The mass and energy release data from each sensitivity analysis is used as input to the FATHOMS code to determine the most limiting boundary conditions. Table 3.1-1 summarizes each peak pressure case analyzed and provides a short description of the type of sensitivity considered.

The long-term mass and energy release analysis is divided into two segments. Mass and energy release data during the initial blowdown phase following a double-ended guillotine rupture of the cold leg piping (8.55 ft^2) is calculated using the RELAP5/MOD2-B&W code. These calculations are continued through the refill, reflood, and post-reflood phases with RELAP5 until the end of the cold leg injection phase. The mass and energy release data generated during this analysis provides the boundary conditions for the initial containment response. The BFLOW code is used to calculate the mass and energy release for the remainder of the analysis for cold leg break LOCAs. RELAP5 is run for hot leg LOCAs until the break flow has reached a quasi-steady condition. The remainder of the mass and energy release is performed in FATHOMS using an input boundary condition based on decay heat.

An averaging process is used to convert the RELAP5 mass and energy release data into a form suitable for the FATHOMS code. The mass and energy exiting each break junction is integrated during each simulation. These integral totals include the thermal energy of the flow stream as well as the kinetic energy imparted to the fluid as it exits the break. The vapor and liquid phases are tracked separately to ensure that the energy is distributed correctly between the containment atmosphere and the fluid in the emergency sump.

3.2 Initial Conditions

The initial conditions for all large and small break LOCA analyses are chosen to maximize the stored energy in both the primary and secondary systems. Maximizing the stored energy will ensure that conservative mass and energy release boundary conditions are provided for the corresponding containment analyses. ANSI/ANS-56.4-1983 (Reference 3-1) provides criteria and guidance for selecting the initial values for the principal system parameters (e.g. pressures, temperatures, etc.) and the type of uncertainties which should be applied to these parameters. Every attempt is made to adhere to the criteria presented in this standard. If the initial value chosen does not conform to the standard, then justification is provided for the deviation. Otherwise, the initial conditions selected meet or exceed the requirements of the standard.

The initial conditions for the principal parameters chosen for this analysis are presented in this section in the same order as they appear in Reference 3-1, Section 3.2.2. A summary of the initial conditions is shown in Table 3.2-1.

Time of Life

See Section 3.3.1.1, Core Stored Energy.

Power Level

An uncertainty of 2% is bounding for thermal power based upon plant performance data. This uncertainty is applied to the 2568 MW_{th} rated power for Oconee to yield the initial power assumed in these analyses (102% rated power or 2619 MW_{th}).

RCS Temperature

The Integrated Control System (ICS) controls the average coolant temperature at a constant value of 579°F whenever power is greater than 15%. Because analyses indicate that a 2°F uncertainty allowance is bounding for the average coolant temperature. an initial value of 581°F is assumed. This value is also consistent with the current ECCS licensing analysis assumptions.

RCS Pressure

An allowance of 30 psi has been determined to bound the pressure indication uncertainty. Since the nominal RCS pressure is controlled at 2155 psig, an initial pressure of 2185 psig is assumed.

Steam Generator Pressure

Turbine header pressure is normally controlled at a nominal setpoint of 885 psig, yielding a pressure of approximately 910 psig at the steam generator outlet. With an uncertainty of 40 psi associated with the steam generator pressure indication, a pressure of 950 psig would appear the be the desired initial condition. However, because of modeling limitations, this value is not used. Instead, steam generator pressure is adjusted during the initialization process to obtain the desired reactor coolant average temperature and coolant flow rate. The basis for this deviation is discussed below.

When initializing a RELAP5 model to the desired initial conditions, the user must choose the parameters which are considered fixed, and allow other parameters to vary in order to obtain an converged steady-state solution. The fixed parameters chosen are usually the ones which are most significant for the analysis (e.g. average temperature, coolant flow rate, etc.). Steam generator pressure is one of the parameters that is allowed to vary during initialization because it has the least impact on the initial stored energy in the primary system. The actual value used for steam generator pressure is 925 psig as opposed to the desired 950 psig. This difference yields a secondary saturation temperature which is approximately 3°F lower than the desired saturation temperature when including uncertainties. This difference has a slight impact on the stored energy in the secondary system and an even lesser impact on the calculated energy release to containment. However, the initial secondary system conditions are consistent with the heat transfer required at this power level.

Pressurizer Level

The level control system normally maintains pressurizer level at the 220 inch setpoint. High level and high-high level alarms exists at 260 and 315 inches to warn the operator of a high level condition. Technical specifications require pressurizer level to be <285 inches. The uncertainty on the level instrument is 25 inches. Therefore, an initial condition of 310 inches, corresponding the technical specification value plus uncertainty is assumed in the analyses.

Steam Generator Operating Level

Main feedwater flow in a once-through steam generator is not controlled based upon steam generator level. Instead, feedwater flow is controlled by the ICS and is based primarily upon the desired power output. As a consequence, level is not a true indication of the steam generator inventory because phenomena such as tube fouling and two-phase pressure drop variations will affect the level indications. A more reliable measure of steam generator inventory is the total mass of liquid and steam present. The nominal full power steady-state mass is [ ] lbm per steam generator for the original steam generators.

The OTSG includes orifice plates in the downcomer region which are designed to allow hydraulic tuning of the generator. These plates are modeled in the Oconee RELAP5 base deck as a junction with an orifice restriction. The abrupt area change model is used to calculated the hydraulic resistance across the junction. Varying the orifice area can significantly change the steady-state steam generator mass inventory. Reducing the area has the effect of increasing the steady-state mass. In order to obtain the greatest initial mass inventory, the flow area through the steam generator downcomer orifice plate junction is specified as the area of the orifice plates in the closed position. An initial steam generator mass of approximately [ ] lbm for the original steam generators is obtained when the orifice plates are modeled in this fashion. This mass is the most that can be obtained when initializing the RELAP5 model at these operating conditions, and is sufficiently conservative. A conservative high initial mass is assumed for the replacement steam generators.

Core Parameters

See Section 3.3.1.1, Core Stored Energy, and Sections 3.3.2.1 and 3.3.3.1, Fission Heat.

Core Flood Tanks

Core Flood Tank (CFT) initial conditions must be chosen for pressure, temperature and liquid volume. According to Reference 3-1, these parameters should be consistent with normal operating values, with uncertainties biased in the direction which leads to the maximum primary containment pressure. Determining the normal operating values is straight-forward. Selecting the bias to apply to these values is less obvious. The following discussions present the rationale for selecting the appropriate biases.

The CFTs are located inside containment and as such, are subjected to containment ambient temperatures. An initial fluid temperature of 130°F is assumed since this value bounds the initial containment temperature. It is correct to bias the initial temperature on the high side since the available fluid sensible heating is minimized thereby maximizing the break steaming rates.

The initial CFT pressure is biased upward, based upon the nominal pressure required by Technical Specifications (600 psig +/- 25 psig) and an instrument uncertainty of 4%. After summing these components, a value of 655 psig is obtained as the initial CFT pressure.

By maximizing CFT pressure, the mass of non-condensible gas initially available for discharge into containment is maximized. Non-condensible gases within containment will boost the pressure by adding an additional partial pressure component to the containment atmosphere. Furthermore, non-condensible gases cannot be removed by the operation of the containment cooling equipment, unlike steam.

Biasing pressure high will allow the CFTs to discharge into the RCS somewhat earlier during the blowdown phase. Because RCS depressurization during a large break LOCA is so sudden, small variations in the initial pressure about the nominal will have a insignificant impact on the transient results. The impact on small break LOCAs will be more noticeable since fluid will begin to inject at higher pressures. Once again, the actual impact of a higher CFT pressure on the thermal-hydraulic response is believed to be insignificant.

The initial CFT liquid volume is biased downward, based upon the nominal volume required by Technical Specifications (1040 +/- 30 ft^3) and an instrument uncertainty of 4%. After summing these components, a value of 972 ft^3 is calculated for the initial CFT liquid volume. By minimizing the volume of subcooled liquid available for injection, less energy can be absorbed by sensibly heating the injected liquid. As a result, more steam will be relieved through the break and containment pressure and temperature will be maximized.

RCS Flow

Primary system flow is not specifically addressed in Reference 3-1 although it does have an impact on the initial stored energy of the coolant. If the flow rate is changed from the nominal value, then the DT required to remove the same quantity of energy from the core must change. Assuming a constant average temperature (which is how temperature is controlled in a B&W plant), a lower flow rate results in a higher T_{hot} and a lower T_{cold}. Similarly, a higher flow rate will result in a lower T_{hot} and a higher T_{cold}. Thus, the distribution of RCS mass between hot and cold leg temperatures, and the associated fluid volume at each temperature, appears to be the factor which determines whether an assumed flow rate is conservative with respect to the initial stored energy.

The nominal best-estimate steady-state flow rate is approximately 112.5% of design flow (88,000 gpm/pump) or 396,000 gpm. The uncertainty associated with this value is approximately 2%. A value of 104% design flow (366,080 gpm) is consistent with current Oconee LOCA analysis assumptions. Assuming a flow rate of 104%, approximately 50.6% of the primary system is at a temperature greater than T_{ave} while 49.4% is at a temperature below T_{ave}. Although this distribution is similar, it indicates that a primary system flow rate which maximizes T_{hot} is conservative with respect to the initial coolant stored energy. An initial flow rate of 104% design flow is therefore assumed.

The above discussion is applicable to the methodology used in the analysis of the original steam generators. For the replacement steam generator analyses RCS flow is a sensitivity parameter and both bounding high and low initial RCS flow values are analyzed to determine the limiting case.

3.3 Boundary Conditions

This section presents the boundary conditions assumed for the peak pressure analyses and for the RELAP5 phase of the long-term mass and energy release analysis. The boundary conditions have been classified as energy sources, which are present at the initiation of the transient or transfer energy to the RCS during the transient, and assumptions, which determine how the energy sources are distributed and/or how the various equipment will respond during the transient. Because the objectives of these analyses are similar, many of the assumed boundary conditions are the same. Identical boundary conditions are presented in Section 3.3.1. Boundary

conditions which are unique to the peak pressure analyses are presented in Section 3.3.2 while RELAP5 long-term boundary conditions are presented in Section 3.3.3. A summary of the common boundary conditions is shown in Table 3.3-1.

3.3.1 Common Boundary Conditions

3.3.1.1 Energy Sources

The energy sources assumed for the large break LOCA mass and energy release calculations are described in this section in the order they appear in Reference 3-1, Section 3.2.1.

RCS and Steam Generator Inventory

The volume of the RCS increases due to thermal expansion of the pressure boundary when the system is heated to operating conditions. This thermal expansion has been accounted for in the Oconee RELAP5 model by increasing the calculated volume of the RCS piping by a nominal 1%, which is representative of the expansion resulting from a temperature change from cold to operating conditions. Furthermore, the initial pressurizer level is assumed to be 310 inches (see Pressurizer Level, Section 3.2) to maximize the liquid volume present in the RCS. This level is well above the normal operating level plus instrument errors and results in an increase in volume of approximately 220 ft^3 or 2% of the total RCS volume. This assumption is particularly conservative because the pressurizer liquid is initialized at saturated conditions (i.e. temperature is much greater than the bulk RCS liquid temperature).

The physical volume of the secondary system has not been increased to account for thermal expansion. Instead, the main feedwater flow boundary condition is chosen so that secondary inventory is conservatively represented.

RCS and Steam Generator Metal

RELAP5 heat structures are included to model all structural metal which contacts the primary and secondary coolant. These heat structures are assumed to be in equilibrium with the surrounding coolant and maintain an initial temperature distribution which is constant across each structure to maximize the initial stored energy.

Core Stored Energy

Stored energy in the core is a function of the initial average fuel temperature, which is a function of burnup. The maximum fuel temperature occurs near beginning-of-cycle. Core decay heat, on the other hand, is at a maximum at end-of-cycle. Reference 3-1 therefore suggests that the time in life be chosen such that the combination of core stored energy and decay heat is maximized.

The initial average fuel temperature is chosen to ensure that the assumed value will remain conservative with respect to future fuel cycle designs. This is accomplished by choosing a temperature which is consistent with the value assumed in the licensing ECCS analyses. Duke currently verifies that the predicted hot pin temperature for each reload design remains less than the value assumed by B&W in the ECCS LOCA analysis. Reference 3-2 represents typical ECCS interface criteria used by Duke. This document contains hot pin characteristics, but does not contain information pertaining to the remaining fuel pins. However, Reference 3-3 does contain dated information on all fuel pins. A comparison of hot pin characteristics in References 3-2 and 3-3 indicates that there is not a significant difference between the data. Thus, the Reference 3-3 data are used to obtain a estimate of the average fuel temperature assumed in the ECCS LOCA analysis.

B&W models the reactor core in their ECCS analyses in the following fashion:

From Reference 3-4, the average linear heat rate is 5.74 kw/ft at 100% full power (5.85 kw/ft @ 102% full power). This value remains essentially constant between reloads because the licensed power level, total fuel rod length, and fraction of direct moderator heating do not change. Thus, the average fuel temperature can be estimated based upon the linear heat rate in each of the three fuel regions.

The following table provides the linear heat rate and average fuel temperature for the three core sections at 102% full power.

Core Section	Linear Heat Rate	Ave. Fuel Temp
Average pin	5.85 kw/ft	
90% hot pin	8.96 kw/ft	
Hot pin	9.95 kw/ft	

The average fuel temperature is estimated by averaging the above temperatures using weighting factors based upon the number of fuel assemblies in each region (i.e. [

]

The above fuel temperatures were assumed in the original analyses. New analyses use bounding values for the current fuel design.

Fission Product and Actinides Decay

The heat produced from the radioactive decay of fission products and actinides is calculated for end-of-cycle conditions. This calculation determines decay heat as a function of time after trip for Oconee Units 1, 2, and 3 using the ANSI/ANS-5.1-1979 standard (Reference 3-5). For additional conservatism, uncertainties (2σ) are calculated and added to the mean decay heat values.

Metal-Water Reaction Rate

The application of this energy source is unique to each large break LOCA analysis. See Section 3.3.2.1 and 3.3.3.1 for details.

Main Steam Lines

The turbine stop valves are assumed to close immediately without delay upon break initiation and the turbine bypass valves are assumed to be unavailable. Therefore, the secondary coolant is isolated in the steam generators and steam lines and is available to transfer energy to the primary fluid. The main steam relief valves are assumed to be operable and will open briefly following closure of the turbine stop valves.

Main Feedwater System

The Main Feedwater System is used to control steam generator level during each analysis (except the peak pressure off-site power sensitivity study). The upper limit placed upon the post-trip MFW level controller corresponds to the nominal MFW flow at 100% full power. This assumption allows the MFW controller to respond to level changes as rapidly as possible and eliminates the need to model the MFW piping and the stroke time for MFW block valves to close.

The Main Feedwater System is realigned to feed through the upper auxiliary feedwater header following the trip of all reactor coolant pumps. Flow through this alignment is restricted to much lower flowrates to prevent exceeding the steam generator tube flow-induced vibration limit.

Emergency Feedwater and ECCS Flow

These items are addressed individually in Sections 3.3.2.2 and 3.3.3.2.

Core Flood Tank Nitrogen Expansion

The mass of the CFT nitrogen over pressure is released directly to containment once the CFTs have emptied. The time frame in which this release occurs is determined from the RELAP5 analysis. Because RELAP5/MOD2 fails whenever the accumulator component is depleted quickly, this component is isolated just before the accumulator is empty. The nitrogen gas is then assumed to enter containment directly at this time.

3.3.1.2 Assumptions

Axial Power Distribution

The axial power distributions evaluated in the Oconee LOCA limits analyses (Reference 3-6) are chopped cosine distributions with a 1.7 peak at the 2, 4, 6, 8, and 10 core elevations. The same axial flux shape is chosen for the all LOCA analyses. A [

Borated Water Storage Tank (BWST) level and Temperature

The ECCS injections pumps and Reactor Building spray pumps draw from the BWST until the available inventory is exhausted. At this point, the pump suctions are aligned to the containment emergency sump in a recirculation mode. The temperature of the ECCS water increases when sump recirculation is aligned because the liquid collected in the sump has been heated by the RCS, or has condensed steam in the containment atmosphere. Although the LPI coolers remove energy from the recirculated liquid, the fluid temperature exiting the cooler is always greater than BWST ambient temperature. Because less sensible heating of the ECCS liquid is possible subsequent to sump recirculation, more steam will be present in the break flow which is deleterious to the containment long-term response.

For these reasons, the minimum BWST level allowed by Technical Specifications is assumed. This assumption will ensure that the swap to sump recirculation will occur at the earliest possible time. Technical Specifications require a minimum level of 46 feet.

A conservatively high BWST temperature of 115°F is assumed since this will minimize the potential sensible heating energy of the injection fluid. As discussed previously, higher fluid temperatures will generally increase the rate of steam release from the break.

MFW Temperature

The assumed MFW temperature will impact the energy removed from the RCS during the LOCA blowdown phase, and the stored energy in the secondary which could be released to containment. A conservatively high MFW temperature will maximize the potential energy release to the containment. The nominal full power MFW temperature is approximately 453°F and the uncertainty associated with this value is 4°F. Based upon these numbers, a bounding value of 460°F is assumed. For additional conservatism, this temperature is held constant throughout the transient even though the temperature will decrease as bleed steam is no longer available for preheating.

EFW Temperature

The EFW System is activated during all peak pressure loss of off-site power sensitivity analyses and during all small break analyses. As with the MFW temperature, the assumed EFW temperature can impact the energy exchange rate between the primary and secondary systems. A

constant EFW temperature of 120°F is assumed for the peak pressure sensitivity analyses. For additional conservatism, this value is increased to 130°F for the small break LOCA analyses since EFW spray through the upper steam generator nozzles can significantly influence the small break transient response.

Most Limiting Single Failure

A component single failure will generally have little impact on the peak pressure analysis, but a much greater impact on the long-term containment response. This is because peak pressures usually occur before the engineered safeguards equipment has time to activate and become effective. The most limiting single failure is therefore chosen to yield a conservative long-term containment response (this assumption will have little or no impact on the peak pressure analysis). The most restrictive single failure is chosen as the one which disables the greatest number of containment cooling components.

An evaluation was performed to determine the most limiting single failure with respect to containment cooling. This evaluation indicated that the failure of a 4160 V switchgear represents the most limiting single failure. Electrical switchgear power a myriad of safety related equipment including injection systems and containment cooling systems. The failure of one of the three available switchgear will result in the loss of the following components:

- one HPI pump
- one LPI pump
- one RBS pump
- one RBCU

All other ECCS equipment is available following a nominal actuation delay which is transient specific.

The switchgear failure is more limiting than a loss of offsite power (LOOP) and failure of one Keowee hydroelectric unit because the second hydroelectric unit is available to power the standby busses through CT-4 (underground) or through the switchyard (overhead). Therefore, all ECCS equipment would be available after a small time delay. This assumption is a departure from the classic LOOP and diesel failure assumed in typical LOCA analyses although Oconee's unique electrical design warrants this departure.

Steam Generator Level

If off-site power is available, the MFW System is used to control steam generator level. Following reactor trip, the MFW pumps are assumed to control level at 50% on the operating range plus instrument errors with flow directed through the upper headers. In reality, the level control setpoint is 25" on the startup range until the RCPs are tripped. MFW temperature is assumed constant at a bounding high value of 460°F. These assumptions are conservative because high energy liquid is available for reverse heat transfer and possible superheating of steam in the RCS. Feeding the SGs with MFW through the upper header conservatively maximizes the amount of secondary energy available for reverse heat transfer during blowdown.

3.3.2 Peak Pressure Mass and Energy Release

3.3.2.1 Energy Sources

Fission Heat

The RELAP5 kinetics model, coupled with reactivity feedback from moderator density, Doppler, and boron, is used to determine the neutron fission power as a function of time after break initiation. A mechanistic calculation of fission power is desired since as with most large break LOCA modeling practices, all control rods are assumed to remain out of the core throughout the simulation. The reactor is therefore brought subcritical with the available reactivity feedback mechanisms. The most dominant negative feedback during blowdown is moderator density while boron is dominant during the refill phase. The positive feedback introduced by decreasing fuel temperatures during blowdown (Doppler feedback) is also significant. All of these mechanisms are modeled.

Analyses were performed to determine reactivity as a function of fluid density for a typical Oconee core. The analyses examined several combinations of burnup, enrichment, and time-in-cycle. The combination which resulted in the least negative reactivity feedback function was chosen for conservatism. This data is listed in Table 3.3-2.

All reactivity values expressed in terms of %Δk/k must be converted to dollars ($) for input to RELAP5. This conversion factor used for this purpose is ß, the effective delayed neutron fraction. A beginning-of-cycle (BOC) value of 0.0071 is chosen for moderator density feedback.

This value conservatively bounds (on the high side) the BOC delayed neutron fractions calculated for previous reloads and should bound future reloads. Although a high BOC ß value will minimize the density feedback contribution, a low end-of-cycle ß will maximize Doppler feedback. A bounding BOC ß value is used in this analysis since moderator reactivity feedback due to core void formation dominates Doppler reactivity feedback (by an order of magnitude). Thus, a bounding BOC ß is used to conservatively yield the least negative density reactivity. A constant Doppler feedback of -2.0×10^{-5} %Δk/k/°F is assumed. This value conservatively bounds (more negative) the Doppler coefficients calculated for previous reloads and should bound future reloads.

The data calculated from the moderator density reactivity feedback analysis is only applicable during blowdown since it is generated assuming equilibrium boron. During the refill phase, however, this function is no longer valid since the boron concentration of the RCS drastically changes. Another method for predicting the combined effect of moderator and boron reactivity feedback during reflood is necessary.

Boron is dissolved in the injected ECCS fluid and introduces negative reactivity during reflood, which maintains the reactor subcritical. The minimum ECCS boron concentration required by technical specifications and specified in the COLR ensures sufficient negative reactivity to offset the positive reactivity introduced through changes in moderator density and fuel temperatures (Doppler). This concentration corresponds to the minimum value necessary to maintain the core 1% subcritical at 70°F with all rods out. This concentration is used to establish the minimum required shutdown margin following a large break LOCA.

As the core is reflooded with borated ECCS liquid, the net reactivity is a function of boron concentration, moderator density, and fuel temperature. The specific nature of this function is unknown and would be difficult to calculate. However, the lower and upper limits of the function are known or can be calculated. The lower limit corresponds to the moderator reactivity feedback at the end of blowdown when the core is completely voided. From Table 3.3-2, this value is -11.324 Δk/k. The upper limit is calculated such that a 1% shutdown margin is maintained with all rods out at 70°F. The reactivity corresponding to this value is described below.

Doppler is the only reactivity feedback explicitly modeled in this analysis besides moderator feedback. Therefore, to ensure that a 1% shutdown margin is maintained, the positive feedback which results when the fuel is cooled from the initial average fuel temperature of [ ]

(initial average fuel temperature) to 70°F must be calculated. This value is calculated to be 2.76 %Δk/k. Thus, to compensate for Doppler, a shutdown margin of 3.76 %Δk/k (5.2958$) is required to maintain a 1% Δk/k shutdown margin at 70°F. This value establishes the upper limit to the refill reactivity feedback function.

The lower and upper limits of the reflood reactivity function have been determined. The reactivity during refill will range from [

] Since the shape of this function between these points is unknown, a 3rd order polynomial curve shape is assumed. Arbitrary points are picked in between the endpoints to generate the curve. This approximation maintains a curve shape similar to the blowdown function. The resulting blowdown and refill moderator reactivity curves are listed in Table 3.3-2. A plot of these functions is shown in Figure 3.3-1.

A RELAP5 control system is used to calculate changes in the moderator density reactivity. This control system first determines an average core fluid density using the following equation.

$$\overline{\rho} = K \frac{\sum_i V_i \rho_i \Phi_i^2}{\sum_i V_T \Phi_i^2}$$

Where V_i = Volume of ith active core node

V_T = Total active core volume

ρ_i = Average density of the core node

Φ_i = Flux in the ith node

K = Conversion factor

The average core density is then input into the blowdown or reflood table (Table 3.3-2) to determine the moderator density feedback. The reactivity contribution due to Doppler is determined directly by RELAP5.

Since the point kinetics model is not capable of calculating spatial power distributions, the reactivity contributions from each core heat structure and volume must be collapsed from a two-dimensional to a zero-dimension (point) representation. These nodal reactivities are combined by flux-squared and volume weighting to obtain the single reactivity value used in the

point kinetics model. The flux in each core volume is assumed to remain constant and proportional to the initial axial power distribution.

Metal-Water Reaction Rate

The publicly available RELAP5/MOD2 code version does not include a metal-water reaction model as a standard feature. This model has been added to RELAP5/MOD2-B&W as part of the evaluation model upgrade (see Section 2.1.1). By selecting this feature, the user is required to model the core in a manner similar to ECCS LOCA evaluations (i.e. hot channel, hot assembly, average assembly). Because this level of core modeling is unnecessary for containment mass and energy release analyses, an alternative approach is used to conservatively bound the expected metal-water reaction rate.

Significant metal-water reaction rates do not occur until cladding temperatures exceed approximately 1800°F. This temperature will be exceeded only when the core has been uncovered for some period of time after blowdown. Therefore, the timing of cladding heatup is more dependent upon time after blowdown than the size and location of the break.

3.3.2.2 Assumptions

The assumptions which are unique to the peak pressure mass and energy release are described in this section.

Break Location

Hot Leg Breaks:

A double-ended guillotine hot leg break (14.1 ft^2) has been demonstrated by B&W to yield the greatest containment peak pressure (Reference 3-7). This result is expected since the hot leg piping represents the largest cross-sectional piping area in the RCS. Two hot leg break locations are examined in this analysis; the reactor vessel outlet, and the inlet to the steam generators. The basis for examining these and no other hot leg break locations is discussed below.

If the break occurs at the reactor vessel outlet, a large percentage of coolant passes through the core as it flows toward the break. The majority of the core stored energy will be released quickly and the resulting high moderator density may contribute to greater delayed neutron power generation. If the break occurs at the inlet to the steam generator, core stored energy release rates and delayed neutron power may be less than the vessel outlet break, but the coolant flow through the steam generator will be greater thereby extracting more energy from the secondary system. These offsetting effects can only be examined through analysis.

Cold Leg Breaks:

A cold leg pump discharge guillotine (8.55 ft^2) break yields significant secondary-to-primary heat transfer during blowdown, and results in a reasonably short flow path from the core exit, through the vent valves, to the break. It should therefore represent the most bounding cold leg break. However, a break at the cold leg pump suction could result in greater secondary-to-primary heat transfer during blowdown. Both of these break locations are analyzed.

The A1 cold leg is chosen as the location for the pump discharge and pump suction breaks because the pressurizer is attached to this loop. [

]

Break Flow Model

The Zaloudek or Henry-Fauske correlations are considered acceptable for predicting subcooled critical flow, and the Moody correlation is acceptable for two-phase critical flow (Reference 3-1). None of these correlations exist in RELAP5 and are generally incompatible with the RELAP5 implementation of the momentum equation. RELAP5/MOD2-B&W does, however, include the Henry-Fauske and Moody models as optional critical flow models. These optional models are used to examine the difference in mass flux predictions between the Ransom and Trapp model and the evaluation model correlations. Based upon the results of this examination, the break junction critical flow discharge coefficients (C_D) are adjusted upward to bound the mass flux predicted by the Henry-Fauske/Moody models.

ECCS Injection

A delay is usually assumed to account for the time required for an ECCS system to actuate and deliver flow to the RCS. This delay results from practical considerations such as valve actuator stroke times and pump start times. If electrical power must first be restored before these actions can occur, then additional delays will be realized. A survey of the times required for the necessary HPI and LPI System components to operate indicates that a delay time of 15 seconds is bounding. When off-site power is not available and a failure of a 4160 V switchgear is assumed, a delay time of 35 seconds is appropriate.

A cross-tie modification has been installed that connects the two LPI trains at the pump discharge. This modification deletes the requirement for manual throttling of LPI flow. Conservative LPI flowrates are now used in the methodology consistent with this design change. The following original text is maintained so that the analysis results presented can be understood."

Injection flow will be available from two HPI pumps and one LPI pump following the failure of a 4160V switchgear. The injected HPI flow is represented as a function of RCS back pressure. LPI flow is assumed to be a function of RCS backpressure until operator action is assumed based upon throttling guidance in the emergency operating procedure. The throttled flow value is biased to account for instrument uncertainty.

All HPI and LPI fluid is assumed to be injected directly into the reactor vessel downcomer below the equilibrium liquid level. This approach is used to conservatively minimize the amount of steam condensation on the ECCS fluid. For the cold leg pump discharge analysis, all HPI injected into the broken cold leg is assumed to spill directly into containment. Thus, for this case, only 75% of the HPI flow is assumed to be delivered to the RCS.

Reactor Coolant Pumps

For cases which assume off-site power is available, the RCPs continue to run throughout the simulation. For the cases which examine the impact of a LOOP, the RCPs are assumed to trip immediately upon reactor trip. Continued RCP operation should be conservative because of the resulting reverse heat transfer from the SGs to the RCS (and hence containment) and additional pump heat.

Emergency Feedwater System

If a loss of offsite power (LOOP) has occurred, MFW is no longer available and the EFW System is used to control steam generator level. MFW pumps are assumed to conservatively coast down within 10 seconds of the LOOP. Then, there is a 17 second delay from time zero before flow from the EFW system is delivered to the SG upper header. With the assumed failure of a 4160V switchgear, one motor-driven and the turbine-driven EFW pumps would be available. However, for conservatism, flow from only the turbine driven EFW pump to both steam generators is assumed. EFW flow as a function of steam generator pressure is based upon Unit 3 values, which are slightly less than the other units. Lesser EFW flow rates are conservative because the EFW fluid temperature is much less than the saturation temperature in the steam

generators. The EFW fluid will therefore draw energy from the primary system when it contacts the steam generator tubes instead of transferring energy to the primary system.

Containment Back-Pressure Sensitivity

Reference 3-1 suggests directly coupling a containment model to the RCS model to mechanistically calculate containment pressure, or assuming a conservatively high back-pressure. The latter of these two approaches in used in the peak pressure analyses. A constant back-pressure at the containment design pressure of 59 psig (73.7 psia) has been used for conservatism for all but the back-pressure sensitivity studies. These studies examine the impact of containment pressure on the predicted mass and energy release for back-pressures of 30 psia (15.3 psig) and 50 psia (35.3 psig).

Refill Phase Modeling Sensitivity

Previous mass and energy release methodologies have assumed that the vessel refill phase does not exist. This philosophy has also been introduced into industry standards. Reference 3-1 states that "A non-zero refill time shall be justified if used in the calculation of long-term releases." It is not obvious that the time required for vessel refill (10s of seconds) has a significant impact on the containment response, especially for long-term releases. The only conceivable impact appears to be on the containment response during the short-term. For this reason, the impact of a zero refill time is evaluated in a sensitivity study. A simulation is restarted immediately following blowdown and the reactor vessel lower plenum is fictitiously filled the up to the bottom of the active core to effectively bypass the refill phase. The fictitious fluid is assumed to be saturated liquid at the lower plenum pressure. The results of this evaluation should help quantify the impact of a non-zero refill time on the containment response.

3.3.3 RELAP5 Long-Term Mass and Energy Release

3.3.3.1 Energy Sources

Fission Heat

Refer to Section 3.3.2.1.

Metal-Water Reaction

The arguments presented above regarding the significance of the fission power contribution to the total energy released to containment are equally applicable to energy generated from the metal-water reaction. Consequently, energy released by the steam/cladding metal-water reaction is not considered in this analysis. The potential impact of this non-conservative assumption is quantified below.

To examine the relative magnitude of the energy released during the exothermic Zr-H_2O reaction, the energy released from this reaction is calculated and compared to the total energy released to containment at the termination of the RELAP5 long-term analysis. A bounding value for the whole-core oxidation rate of [ ] is listed in Reference 3-9. The oxidation of this percentage of cladding will generate approximately 3.13×10^6 Btu of energy. An integrated energy release total of 558.107×10^6 Btu has been predicted by the termination of the RELAP5 analysis. The maximum expected energy addition from the metal-water reaction therefore represents 3.134 / 558.107 or 0.56% of the total energy released up to 1800 seconds. This value clearly represents a small fraction of the total energy released at 30 minutes and will decrease considerably with time. Thus, the energy addition from the metal water reaction is insignificant for long-term mass and energy release calculations which span a 20 day time frame.

3.3.3.2 Assumptions

The long-term large break containment analysis considers both cold leg pump discharge and hot leg breaks. There is no need to analyze a spectrum of break sizes, locations, and boundary conditions since suitably bounding assumptions can be chosen by inspection. The qualitative bases for these assumptions are presented in this section.

Break Location

The break size and location is dependent upon the analysis result of interest. For the limiting containment EQ response, the limiting break location is a double-ended guillotine break (8.55 ft^2) at the reactor vessel outlet nozzle. For the maximum containment sump water temperature analysis, the limiting break location is a double-ended guillotine break (14.1 ft^2) located at the reactor vessel outlet nozzle. Although it is true that an identical quantity of decay heat will be generated regardless of the break location, the manner in which this energy is partitioned

between the vapor and liquid break flow streams is the dominant consideration. Because the long-term containment response is concerned with temperature in containment as a function of time, it is expected that an energy release profile which is dominated by steam relief will generate a more severe containment response. This is because steam relief to the atmosphere will have a greater impact on containment temperature than if the energy is released primarily in the liquid phase which has only a slight interaction with the containment atmosphere (i.e. convection at the pool surface). Indeed, this observation has been validated with FATHOMS in numerous analyses. It might appear that the RBS System acts to homogenize the containment atmosphere such that the phase in which the energy is released in is insignificant. However, when the complicated interaction between the equipment used to cool the containment atmosphere (RBCUs and RBS) and the equipment used to cool the containment sump (LPI coolers) are examined by analysis, it is apparent that containment will never become completely homogenized. Therefore, the partitioning of energy released to containment between the vapor and liquid phases will be the dominant factor in the long-term containment response.

Due to the geometry of a B&W reactor system, a cold leg break site will never become completely flooded. This means that steam will always exit the break site no matter how much the decay heat power drops (see Section 2.2.1). In contrast, it is possible to flood a hot leg break site as decay power decreases. Decay heat will eventually be absorbed as sensible heat by the injection fluid and thus steaming from the break will cease. Naturally, when the break site is flooded, decay heat will be transferred to containment in the liquid phase resulting in a less severe containment pressure response, but a higher sump water temperature.

The cold leg pump discharge break location is selected rather than the pump suction location for two reasons. For a pump suction break the cold HPI fluid injected into the broken cold leg pump discharge piping will interact with steam exiting the core through the vent valves and condense a large portion of this steam before it reaches the break. Thus, the steam release will be less for a pump suction break. The pump suction break will also support a higher equilibrium liquid level in the hot leg so that it should be possible to align the LPI system in the decay heat removal mode which would terminate all steam relief through the break. Consequently, the pump discharge break location is limiting.

The break flow model assumed for the long-term mass and energy release calculations utilizes the Ransom and Trapp critical flow model. No adjustments are made to the break junction critical flow discharge coefficients. This assumption is based on the long-term analysis being insensitive to the blowdown phase dynamics and the exact break size.

BWST Depletion - ECCS and Containment System Flow Rates

The BWST is the common suction source for the HPI, LPI and RBS Systems. These systems will draw upon the BWST until level in the tank drops to a predetermined setpoint. At this point, the pump suction must be swapped to the containment emergency sump. The liquid which collects in the sump is much warmer than the initial BWST temperature. Although this fluid is cooled by the LPI coolers before it injected into the RCS, the liquid will still be hotter than the original BWST liquid. At these higher temperatures (and hence enthalpy), less sensible heating of the ECCS fluid is possible. Therefore, for the same decay heat power level, more steam will be generated. BFLOW analyses substantiate that the break steam flow rate is much more sensitive to ECCS subcooling than it is ECCS flowrate. Consequently, depleting the BWST rapidly more than compensates for the slight reduction in break steaming rate which will occur with the higher ECCS flow rates assumed to deplete the BWST. However, higher ECCS flow rates tend to enhance LPI cooler performance. These considerations are included in the selection of the ECCS boundary conditions."

The objective of this boundary condition is to then deplete the BWST to determine the time that sump recirculation is aligned. This is accomplished by assuming conservatively high or low flow rates from the ECCS systems which are consistent with the assumed single failure of a 4160V switchgear. Depletion flow from the HPI, LPI and RBS Systems is modeled in order to track the rate of BWST depletion and determine the time at which sump recirculation is aligned.

A cross-tie modification has been installed that connects the two LPI trains at the pump discharge. This modification deletes the requirement for manual throttling of LPI flow to 3000 gpm. Conservative LPI flowrates are now used in the methodology consistent with this design change. The following original text is maintained so that the analysis results presented can be understood.

The single failure of a 4160V switchgear implies that two HPI pumps will be available to inject through one train. There is guidance, however, in the Emergency Operating Procedure which requires the operator to cross-connect HPI trains within 10 minutes if flow is not verified through each train. Cross-connecting trains will allow more HPI fluid to enter the RCS compared to the situation where no cross-connecting is assumed (such as the peak pressure analyses). The HPI trains are therefore assumed to be cross-connected immediately after ES actuation to maximize the injection flow.

A consequence of choosing the cold leg pump discharge as the break location means that HPI flow in the broken loop will be lost out the break. In order to minimize non-conservative condensation of steam on the subcooled HPI fluid near the break site, it is assumed that only 3 of the 4 HPI injection lines are intact. Therefore, 25% of the total HPI flow is assumed to spill directly to containment.

The assumed failure of a 4160V switchgear will eliminate all but one LPI and one RBS pump. LPI flow is assumed to be a function of RCS backpressure until operator action is assumed based upon throttling guidance in the Emergency Operating Procedure. The throttled flow value is biased to account for instrument uncertainties. RBS flow is modeled in this analysis to calculate the BWST depletion rate. The RBS System is assumed to deplete the BWST at a constant rate using a conservative flow rate. It should be noted that consistent assumptions are made between the spray flow assumed in the BWST depletion calculation and the flow assumed in the FATHOMS analysis. BWST water that is assumed to be spilled directly to the sump is accounted for in the FATHOMS analysis in order to conserve mass inventory in containment.

The emergency operating procedure instructs the operator to switch to sump recirculation at a nominal BWST level of 6 feet. An uncertainty adjustment associated with the BWST level indication is made to conservatively minimize the amount of available BWST inventory. Although this uncertainty adjustment is applied in the negative sense when determining the initial BWST level (see Section 3.3.1.2, BWST Level and Temperature), it is applied in the positive sense when determining the setpoint to align sump recirculation. When the assumed setpoint is reached in the analysis, the switch to sump recirculation is assumed to occur instantaneously (i.e. no delay for valve stroke times and operator actions).

Reactor Coolant Pumps

Because offsite power is assumed to be available, the RCPs will continue to run. Two minutes after break initiation, the operators are assumed to trip all RCPs. This action should occur immediately upon an indication of a loss of subcooling to comply with guidance in the Emergency Operating Procedure. Although RCP operation in a highly voided system is believed to have little significance on the mass and energy release, it will result in the transfer of secondary energy to the steam which is circulated through the loops. The RELAP5 two-phase head degradation model is employed as discussed in Section 3.3.2.2, Reactor Coolant Pumps.

Emergency Feedwater System

Because EFW fluid temperatures are less than MFW temperatures and off-site power is assumed available in this analysis, it is conservatively assumed that EFW is not activated during the long-term mass and energy release analysis. MFW remains operable and is used to raise and maintain the steam generator levels at the appropriate setpoint as discussed in Section 3.3.1.2, Steam Generator Level.

Containment Backpressure

As noted previously, Reference 3-1 suggests directly coupling a containment model to the RCS model to mechanistically calculate containment pressure, or assuming a conservatively high back-pressure. Once again, the latter approach is used in this analysis. The containment boundary conditions differ slightly from the peak pressure case in that a time varying pressure is assumed. This variance recognizes that containment pressure will decrease considerably during the transient as the RBS System condenses steam in the containment atmosphere.

The containment time dependent volume pressure assumed is based upon FATHOMS results which assumes operation of the following equipment: two RBCUs, one train of RBS and one LPI cooler. The pressure values chosen bound (on the high side) the pressure response from the FATHOMS analysis. The particular pressure and temperature values assumed are listed below.

Time (sec)	Pressure (psia)	Temperature (°F)
0.0	63.1	300
25.0	63.1	300
80.0	58.6	295
185.0	58.4	295
705.0	47.0	280
1705.0	37.6	270
3009.0	27.3	270

The properties of the time dependent volume are specified as slightly superheated steam. Superheated steam, as opposed to saturated steam, is chosen to preclude known RELAP5 code instabilities which can occur if saturated steam from containment is drawn into the RCS and condensed. Condensation of steam from containment can create a low pressure region which

will draw in more steam and thus propagate steam flow from containment. The potential for this positive feedback effect is reduced by superheating the steam in containment.

3.3.3.3 Break Flow Oscillations

When assuming that containment can be represented as a time dependent volume, a non-physical boundary condition is introduced at the break junction plane. This condition leads to instabilities in the predicted break mass flow rate which have a direct impact on the RCS energy balance. An example of this phenomenon is discussed and the efforts taken to minimize the consequences are described in this section.

Whenever the mass flows through the break junctions reverse direction and flow from the containment boundary condition into the RCS, an inconsistency in the RCS stored energy is introduced. This phenomena will occur whenever the assumed containment pressure is greater than the RCS pressure and is termed reverse flow for the purpose of this discussion. This phenomenon is obvious at the termination of blowdown when RCS pressure is dropping rapidly and the containment pressure is essentially constant. The inertia of the break effluent causes fluid to continue flowing out of the break even though RCS pressure falls below containment pressure. This results in an undershoot in RCS pressure. As a result, a large influx of superheated steam from the containment boundary conditions is necessary to increase RCS pressure to the containment pressure. An oscillatory RCS pressure response is observed until the break flow oscillations dampen out. This initial series of flow oscillations account for over one-half of the total reverse flow energy entering the RCS from containment.

Although the bulk of the reverse flow occurs immediately after blowdown, break flow oscillations during the post reflood phase sometimes cause oscillations in the DP between the RCS and containment volumes. As a result, break flow oscillations have also been observed in the post reflood phase of the long-term analyses. These oscillations are in both the forward and reverse direction. Reverse flow from containment to the RCS results in the addition of significant quantities of mass and energy which does not preserve the initial RCS mass and energy. An attempt has been made in the analysis to minimize the number and severity of these post reflood oscillations by adopting the conservative modeling techniques listed below.

- All active ECCS is injected into the reactor vessel downcomer beneath the equilibrium liquid level
- The inter-phase heat transfer coefficients are time averaged
- The vent valve junction is held open at a constant value and the downstream volume connection is changed

The last two bullets are implemented only after quasi-steady state conditions have evolved. These techniques yield more stable code results when compared to preliminary analyses and are the topic of the following discussion.

RELAP5 predicts steam generation/condensation via the constitutive vapor generation model. The predictions of this model are particularly significant at low pressures since the change in specific volume from liquid to steam (or steam to liquid) is much greater than at higher pressures. As a result, the vapor generation rate will have a greater impact upon pressure and interphase drag predictions. Therefore, if the predicted vapor generation rates are oscillatory, the overall code predictions will be oscillatory.

In this analysis, all injection flow (HPI and LPI) is assumed [

] in the cold legs and reactor vessel downcomer regions.

The RELAP5/MOD2-B&W code includes an option which allows the user to time average the interphase heat transfer coefficients that are used in the vapor generation model. The option averages the predicted values at the old (n-1) time step with the current time step based upon a user supplied fraction. For this analysis, a value of 0.1 is used which averages 10% of the old time heat transfer coefficient with 90% of the current heat transfer coefficient. This option is not invoked until blowdown is complete and quasi-steady state conditions are attained.

Because of numerical instabilities, the fact that no inertia is modeled in the movement of the vent valves, and only one vent valve is modeled, flow through the vent valve junction can oscillate considerably with time. Flow oscillations have a direct impact on vapor generation rates in the core and downcomer because of the influence of flow on pressure in these regions. Also,

because of the tight coupling between vent valve flow and the pressure at the core exit and downcomer, it is possible to introduce a positive feedback effect which may ensure a perpetual instability. The vent valve is assumed to remain open at a fixed value during this analysis once blowdown is complete and quasi-steady state conditions are attained. Although the predicted vent valve flow does not attain a steady value even when the valve is held open, flow oscillations are much smaller than observed when the valves are allowed to move freely. This modeling approach is justified because of the small DP necessary to hold the valves open, and the fact that a reasonably constant time average value of flow could be expected through eight independent valves as exist in the reactor vessel.

By restarting the analysis after blowdown is complete and incorporating the above changes, break flow oscillations and reverse break flow are greatly reduced. As a result, the transient results are much more stable. The data averaging techniques used ensure that bounding mass and energy releases result.

3.3.3.4 Heat Structure Stored Energy

Overview

Heat structures are included in the RELAP5 input deck to model the heat capacity of the structural components in the primary and secondary systems. At the initiation of the simulation, a significant quantity of energy is stored in these heat structures and available for release to containment over time. The energy release rate is calculated mechanistically during the RELAP5 simulation. However, at the termination of this simulation, considerable stored energy remains in the heat structures and in the primary and secondary fluid inventory. At the end of the RELAP5 analysis a conservative method must be derived to release this stored energy during the FATHOMS analysis. This problem is complicated by the fact that the [

] This section describes how the heat structure and fluid stored energy is assumed to be released to containment from the end of the RELAP5 simulation to the end of the containment analysis.

Heat Structure Groups

In order to track of the stored energy in the primary and secondary heat structures, each RELAP5 conductor is categorized into [ ] based upon the projected long-term mode of heat transfer to containment. These groups are defined in general terms below.

The total energy stored in each group is calculated during the RELAP5 analysis using control system components. These calculations are referenced to a "basal" temperature of 125°F, which is currently the long-term upper limit for electrical equipment aging calculations. For all conductors within a specific group (n), the total stored energy is calculated as follows:

$$E_T = \sum_{i=1}^{n} C_{pi} V_i (T_i - 125°F) \qquad \text{Eq. 3.3.3-1}$$

Where: V_i = Metal volume of the ith heat structure

C_{pi} = Specific heat of the ith heat structure

T_i = Temperature of the ith heat structure

All RELAP5 heat structures within a group are lumped together and represented as an single heat structure with average properties.

Stored Energy Release Rates

The average overall heat transfer coefficient can be approximated by using the cooldown data associated with each group shown in Figure 3.3-4, and applying Newton's law of cooling to this data to solve for the overall heat transfer coefficient. Newton's law of cooling is typically written as follows:

$$Q = h\,A\,\Delta T$$

Where,	Q	=	Cooldown rate
	ΔT	=	Average temperature difference
	h	=	Heat transfer coefficient
	A	=	Area of conductor

This equation can be solved to calculate the overall heat transfer coefficient hA.

$$hA = \frac{Q}{\Delta T}$$

3.4 BFLOW Long-Term Mass and Energy Release

The methodology used by the BFLOW code to calculate the long-term break conditions is described in Section 2.3. The break conditions may be [

]

3.5 Peak Pressure Mass and Energy Release Results

The transient response of large break LOCAs have characteristics which are similar regardless of where the break is located. The most obvious characteristic is the rapid discharge of primary coolant into containment. Naturally, the time required to expel this coolant will vary depending primarily upon break area, but this time is usually less than 25 seconds for all hot and cold leg pipe breaks. Because of these similarities, there is no attempt to describe the transient response of each of the 13 cases executed in this section. Instead, the contents of this section will focus on the results of the various sensitivity studies used to determine the break location and boundary conditions which yield the peak containment pressure. The qualitative discussion of a large cold leg break LOCA transient response is reserved for Section 3.6.

3.5.1 Break Location Sensitivity

The Case 1 series analyses examine the effect of break location on the resulting containment peak pressure response. The integrated mass and energy totals from these analyses, as with all other analyses described in this document, are averaged with a utility computer program developed in-house. This program averages the mass and energy release data, while conserving each quantity, and provides an automated means for transferring data to the FATHOMS computer code. FATHOMS analyses were performed using data processed by this utility program to examine the resulting peak containment pressure. All other FATHOMS initial and boundary conditions remained constant in order to isolate the impact of the mass and energy release data on peak pressure.

<u>Hot Leg Break at Vessel Outlet and Steam Generator Inlet - Cases 1A and 1B</u>

The Case 1A and 1B breaks are expected to yield similar results since each case simulates a double-ended guillotine break of the hot leg. Although the break areas are identical, the location of the break (reactor vessel outlet - Case 1A, or the SG inlet - Case 1B) is significant enough to yield a different peak containment pressure.

Figures 3.5-1 and 3.5-2 compare the integrated mass and energy releases, respectively, from all Case 1 analyses. From Figure 3.5-1, it is obvious that except for approximately 10 to 30 seconds, the integrated mass release from Case 1A exceeds all other cases. Only the Case 1B integrated mass release briefly exceeds the Case 1A results. During and immediately after blowdown, it is also apparent that the Case 1A and 1B energy releases are nearly identical. The Case 1B energy slightly exceeds the Case 1A results (see Figure 3.5-2).

The differences between the Case 1A and 1B analyses are easily discerned. Because the Case 1B break is located at the inlet to the Loop A steam generator, a large fraction of the primary system coolant reverses flow direction, and passes through the Loop A steam generator. Naturally, this massive coolant flow extracts more energy from the secondary side coolant than is extracted in the Case 1A simulation. This observation is illustrated by comparing pressure in the steam generators at the end of blowdown. These pressures are listed below.

Steam Generator	Case 1A	Case 1B
Loop A	572.5 psig	461.1 psig
Loop B	831.1 psig	821.8 psig

Although pressure in the Loop B steam generators are comparable, there is a difference of approximately 110 psi between the Loop A steam generators. Because the steam generator secondary side is maintained at or near saturation, this difference in pressure implies a difference in saturation temperature and thus fluid internal energy. More energy has therefore been transferred from the Case 1B Loop A steam generator to the primary coolant, and out into containment.

Differences between the Case 1A and Case 1B simulations are highlighted by examining the rate at which energy flows into containment. Figures 3.5-3 and 3.5-4 illustrate the energy flow rates from the vessel side and SG side breaks. The total energy flow rate is shown in Figure 3.5-5. From these plots, it is apparent that the energy flow from Case 1B exceeds Case 1A, especially from the SG side break. This difference is a direct impact of the enhanced reverse heat transfer from the steam generators.

It was expected that a hot leg break at the reactor vessel outlet will result in greater delayed neutron power than a break at the inlet to the steam generator (Section 3.3.2.2, Break Location). This expectation is substantiated with the results of this analysis (see Figure 3.5-6). In addition, the analysis results indicate that the reverse heat transfer from the steam generators is greater than the added neutron power and core stored energy removal resulting from a break at the vessel outlet.

Cold Leg Break at Pump Discharge and Suction - Cases 1C and 1D

The integrated energy totals for Cases 1C and 1D are shown in Figures 3.5-1 and 3.5-2. From these figures, Case 1C clearly releases more mass and energy during the initial phase of the transient.

As illustrated in Figure 3.5-7, a break at the cold leg pump discharge yields a greater release of energy from the core region since the flow resistance from the vessel to the break is minimized. Furthermore, Figure 3.5-8 demonstrates that steam generator reverse heat transfer significantly influences the energy content of the fluid exiting a cold leg suction break. When the total energy

release rates are compared (Figure 3.5-9), it is apparent that for the first 8 seconds, Case 1C energy flow is greater. However, because of the influence of the steam generators on the exit quality of the SG side break, the Case 1D energy flow rate exceeds the Case 1C flow rates.

3.5.2 Critical Flow Model Evaluation

The Case 2 series simulations examine the differences in mass flux predictions between the Henry-Fauske and Moody evaluation models (EM) and the RELAP5 Ransom and Trapp model. The results from these comparisons indicate that the Henry-Fauske and Moody model combination yields higher critical mass fluxes than the Ransom and Trapp model. This result is expected since the Henry-Fauske and Moody models are generally regarded as conservative while the Ransom and Trapp model is designed for best estimate use. Therefore, to yield conservative critical flow predictions, the break junction discharge coefficients must be increased. The process used to approximate these coefficients is described below.

The nominal discharge coefficient for each break junction is approximated by comparing the results from the Case 1 series with those obtained from the Cases 2 EM series and shifting the Case 1 series curves upward. For example, Figures 3.5-10 and 3.5-11 compare the mass flux predictions from Cases 1B and 2C for the vessel side and SG side break junctions, respectively. Because the end of blowdown occurs at different times for each case, the mass flux curves are plotted with respect to normalized time. The normalization times for each curve are chosen near the end of blowdown when the indicated RCS pressure is the same for each case. Then, a constant multiplier is applied to adjust the Case 1 curves upward (see Figures 3.5-10 and 3.5-11). These multipliers are determined mathematically to yield the best curve fits. It should be noted that although this process approximates the required critical flow discharge coefficients, it does not ensure that the transient results with the adjusted discharge coefficients will match or bound the data from the EM cases.

Using the above process, critical flow discharge coefficients were determined for Cases 2B and 2D. These coefficients are summarized below.

Case	Vessel Side Break Junction	SG Side Break Junction
2B	1.2	1.1
2D	1.1	1.2

The resulting mass flux for these cases are compared to the corresponding EM case. Figures 3.5-12 and 3.5-13 compare the break mass flux predictions for Case 2A (HF/Moody) and Case 2B (Ransom and Trapp). Figures 3.5-14 and 3.5-15 compare the break mass flux predictions for Case 2C (HF/Moody) and Case 2D (Ransom and Trapp). From these figures, it is evident that the shape of the curves are generally similar.

The integrated mass and energy releases for all Case 2 series transients are compared in Table 3.5.2-1 near the end of blowdown. The results of this comparison indicate that the predicted mass and energy release from the Ransom and Trapp model, with adjusted critical flow discharge coefficients, is within 1% of the corresponding Henry-Fauske/Moody predictions. These comparisons and the mass flux comparisons demonstrate that the Ransom and Trapp model can be used to yield similar results to the Henry-Fauske/Moody models. They also confirm the validity of applying a constant multiplier (critical flow discharge coefficients) to each break junction to boost the mass flux predictions from the Ransom and Trapp model to those of the Henry-Fauske/Moody models.

3.5.3 Offsite Power Sensitivity

Sensitivity analyses are performed to examine the impact of offsite power on the calculated mass and energy release rates. Offsite power is a potentially important boundary condition since the operation of reactor coolant pumps, delays associated with ECCS equipment, and availability of main feedwater are impacted by offsite power.

Cases 2B and 3A examine a loss of offsite power (with and without offsite power, respectively) for a hot leg break at the vessel outlet while Cases 2D and 3B examine the same scenario for a hot leg break at the steam generator inlet (with and without offsite power, respectively).

The offsite power assumption has a greater influence when the hot leg break is located at the vessel outlet when compared to the steam generator inlet. Continuous reactor coolant pump operation tends to force more primary coolant through the core when compared to an immediate pump trip. This phenomena is illustrated in Figure 3.5-16 which shows that the Case 2B vessel side break energy flow exceeds the Case 3A flow throughout blowdown. In a related observation, the immediate pump trip allows the coolant in the loops to stop and reverse flow easier than if the pumps continue to operate. As a result, a slightly greater energy flow results from Case 3A (Figure 3.5-17) steam generator side break. Because the energy flow through the

vessel side break exceeds that of the steam generator side break, the net result is that to Case 2B energy release exceeds the Case 3A energy release (Figure 3.5-18).

The differences between the Case 2D and 3B results (SG inlet break) are not as pronounced as the Case 2B and 3A results (vessel outlet break). Because the break is located further away from the reactor coolant pumps than the vessel outlet break (higher flow resistances), the influence of the reactor coolant pump flow diminishes. Figures 3.5-19 through 3.5-21 compare the energy flow rates from the break junctions. These figures suggest that the availability of offsite power has little impact on the energy release rate from a hot leg break at the steam generator inlet.

It should be noted that the delays in ECCS and EFW actuation have no impact on peak pressure since the peak occurs before these systems are assumed to actuate. Furthermore, the main feedwater boundary condition appears to have little influence on peak pressure. For both LOOP cases, a main feedwater coastdown of 10 seconds is assumed. Thus, main feedwater flow is continuous for the majority of both the LOOP and non-LOOP blowdowns. The assumed flow rate is the only real difference between these cases.

3.5.4 Containment Backpressure Sensitivity

Sensitivity studies are performed to examine the impact of the assumed containment backpressure on the predicted containment peak pressure. Three backpressures are examined; 30 psia (Case 3C - 15.3 psig), 50 psia (Case 3D - 35.3 psig), and 74.3 psia (Case 2B - 59 psig).

Each transient response is identical up to the time when the break junctions are no longer choked, which is near the end of blowdown (the backpressure impacts the time at which this occurs). Since the peak pressure occurs near the end of blowdown, significantly different containment peak pressures are not expected.

Slight differences in peak pressure are expected. These differences will occur since with a low containment backpressure, more mass and energy must be expelled from the RCS before the RCS pressure equilibrates with the containment pressure. In addition, the specific volume of steam at low pressures is greater than at high pressures. Therefore, less steam mass exists in the RCS at the end of blowdown for the low backpressure cases. This difference in steam volume is thus displaced into the containment. Finally, the longer blowdown associated with a low containment

backpressure results in more direct spilling of the injected core flood tank fluid. Each of these phenomena would result in slightly higher peak containment pressures.

The majority of cases performed in this analysis assume a backpressure of 59 psig. Although this backpressure does not yield the greatest peak containment pressure, it is expected that the peak containment pressure is not sensitive to the assumed backpressure. Therefore, because the peak containment pressure calculated by FATHOMS are near 59 psig, this backpressure is considered appropriate for peak containment pressure analyses.

3.5.5 Refill Sensitivity

Reference 3-1 states that "A non-zero refill time shall be justified if used in the calculation of long-term releases." For this reason, the impact of a zero refill time is evaluated in a sensitivity study performed by Case 3E. This case is restarted from Case 2D immediately following blowdown. The reactor vessel lower plenum is fictitiously filled up to the bottom of the active core with saturated liquid to effectively bypass the refill phase.

The mass and energy release of Case 3E and Case 2D are by definition identical through the end of blowdown. Since peak containment pressure will occur prior to the end of blowdown, it is reasonable to expect that the peak pressure will be unaffected in Case 3E. Given that the peak containment pressure determined for Case 2D occurs prior to the end of blowdown, a containment analysis for Case 3E is not necessary. Thus, a non-zero refill time is justified.

3.5.6 Summary And Conclusions

The objective of the peak pressure analysis is to determine the break size, location, and boundary conditions which yield the highest peak containment pressure. Sensitivity studies are performed to determine these parameters. Analyses are then performed to determine the critical flow discharge coefficients necessary for the Ransom and Trapp critical flow model to generate mass flux data similar to the Henry-Fauske/Moody models.

Sensitivity studies are also performed to examine the impact of a LOOP, containment back-pressure, and zero-refill times. Results from the LOOP sensitivity indicate that for a hot leg

break at the steam generator inlet, the availability of off-site power has little impact of the peak pressure. A LOOP has a greater influence on the peak pressure from a hot leg break located at the reactor vessel outlet.

The containment backpressure sensitivity studies indicate that the containment back-pressure also has little impact on peak pressure. In general, a lower backpressure allows slightly more energy to enter containment from the RCS during the pressure equalization process. The differences observed are not great enough to utilize a backpressure which is less than physically reasonable.

Finally, the refill sensitivity study indicates that a zero-refill time has no impact on the calculated peak pressure. This is because the peak pressure is calculated to occur near the end of blowdown, before the refill phase begins.

Based upon the FATHOMS results presented for this analysis, Case 2D results in the most limiting mass and energy release rates for peak containment pressure. The mass and energy release data for this case are shown in Table 3.5-1.

As mentioned in Section 3.3.2.2, the mass and energy release from a 14.1 ft^2 hot leg break was found by B&W to yield the greatest peak containment pressure (Reference 3-7, Section 15.14.5). A comparison of the UFSAR case with Case 2D will indicate the differences in the mass and energy release methodologies. This comparison is made in a tabular form below at 20 seconds into the transient or shortly after the end of blowdown. This data is also compared graphically in Figure 3.5-22.

Comparison of UFSAR Data with Case 2D Results at 20 seconds

	Integrated Mass (lb)	Integrated Energy (Btu)
UFSAR Data	493,593	3.10179×10^8
Case 2D Results	482,912	3.23162×10^8

This table illustrates that the total mass release in the B&W analysis exceeds that predicted by Case 2D by 2%. The total energy release from Case 2D exceeds the B&W data by approximately 4% at 20 seconds. Figure 3.5-22 also indicates that the Case 2D energy release

rate exceeds the B&W data until approximately 45 seconds, at which time the B&W data is higher. These results indicate that the mass and energy release methodologies for the peak pressure analysis are very similar.

3.6 RELAP5 Long-Term Mass and Energy Release

The RCS transient response to a large break LOCA can be divided into four distinct phases; blowdown, refill, reflood, and post-reflood. A discussion of these phases as they apply to the long-term RELAP5 LOCA simulation are presented in this section.

RCS blowdown resulting from a LOCA in the Loop A1 cold leg pump discharge is characterized by a very rapid and violent depressurization of the RCS. Pressure drops almost immediately to the saturation pressure corresponding to the hot leg fluid temperature as liquid flow out of the pressurizer chokes, effectively decoupling this component from the RCS. The subsequent depressurization rate is more gradual, trending the liquid cooldown rate, until RCS pressure is equal to containment pressure and blowdown is concluded. Extensive liquid flashing occurs immediately throughout the RCS, especially in the core region where voiding alone is sufficient to bring the reactor subcritical.

Liquid in Loop B decelerates quickly during blowdown and changes flow direction. Similarly, flow reverses direction in the core and begins to move toward the break. Substantial loop flow rates, coupled with low RCS fluid temperatures, results in rapid cooling and depressurization of the steam generators. Loop A flow rates, influenced by the break location, greatly exceed Loop B flow rates causing SG A to depressurize further than SG B. These results clearly demonstrate that the SGs act as a substantial heat source, particularly during the blowdown phase. The influence of the RCPs diminishes rapidly during blowdown because of two-phase head degradation and because only steam is available to pump. The Loop A1 pump drastically overspeeds as the Loop A fluid races towards break location. The remaining RCPs continue to operate near 1200 RPM until they are tripped at two minutes. An extended coastdown follows because of minimal fluid resistance. The blowdown phase is essentially complete approximately 26 seconds after break initiation leaving the reactor vessel and loops completely voided.

The refill phase commences at the end of blowdown from CFT injection and LPI and HPI System injection. Subcooled liquid from these systems flows into the reactor vessel downcomer and begins to accumulate in the lower plenum. HPI flow into the A1 cold leg is assumed to spill

completely out the break. The refill phase is concluded at approximately 35 seconds when the injection systems refill the reactor vessel to the elevation corresponding to the bottom of the core.

The reflood phase begins once the reactor vessel fluid level rises into the core region. This phase is typically characterized by a flooding rate which is dependent upon many factors. As the core is reflooded, a quench front moves upward along the fuel assemblies, eventually rewetting the cladding surface. The quench front initiates the transition from film boiling to nucleate boiling heat transfer modes. Once core reflood is complete, core exit superheating, which is observed since early into the transient, is no longer indicated. Superheated steam temperatures, however, are observed throughout the simulation in both hot legs.

The reflood phase continues until approximately 200 seconds into the transient. As reflood progresses, the liquid level continues to rise in the reactor vessel and eventually liquid spills into the hot and cold legs. The post-reflood phase commences at this time. Post-reflood is characterized by quasi-steady state liquid levels, flow, pressures, etc. An equilibrium water inventory is established where the mass flow out the break is approximately equal to the total mass injection flow.

As with the peak pressure analysis results, the instantaneous mass and energy release rates generated by RELAP5 are not used directly as containment analysis boundary conditions. Instead, the mass and energy flow from each of the four flow streams (liquid and vapor flow from each break) is integrated by RELAP5 control systems during the simulation. These integral values are then used as input to a utility program which averages the data. Data averaging is necessary to smooth break flow oscillations (see Section 3.3.3.3) and provide for automated data transfer between RELAP5 and FATHOMS while preserving the integrated mass and energy totals.

The tabular boundary conditions provided for use in the FATHOMS containment analysis are shown in Table 3.6-1. The flow stream mass flow data is plotted against the corresponding mass flow values calculated by RELAP5 in Figures 3.6-1 through 3.6-4. From these figures, it is apparent that the calculated mass flow rates for each flow stream is accurately represented by the average values.

3.7 References

3-1 ANSI/ANS-56.4-1983, "American National Standard Pressure and Temperature Transient Analysis for Light Water Reactor Containments". American Nuclear Society, December 23, 1983

3-2 B&W Document 51-1152464-00, "Duke's ECCS Analysis Interface Criteria", Babcock & Wilcox, June 14, 1984

3-3 B&W Document 51-1152464-03, "Duke's ECCS Analysis Interface Criteria", Babcock & Wilcox, May 16, 1989

3-4 DPC-RD-2012, "O1C12 Reload Report", Duke Power Company, January 1989

3-5 ANSI/ANS-5.1-1979, "Decay Heat Power in Light Water Reactors", American Nuclear Society

3-6 BAW-10103A, Revision 3, "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS", Babcock & Wilcox, July 1977

3-7 Oconee Nuclear Station Final Safety Analysis Report, Table 14.14-4, Duke Power Company

3-8 Deleted

3-9 Deleted

3-10 BWNT LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants, BAW-10192-PA, Framatome Advance Nuclear Products, Revision 0, June 1998

Note: Page 3-47 has been deleted

TABLE 3.1-1

Summary of Peak Pressure Cases Analyzed

Case	Description	Sensitivity
1A	14.1 ft^2 hot leg break at vessel outlet w/ off-site power	Break size/location
1B	14.1 ft^2 hot leg break at SG inlet w/ off-site power	Break size/location
1C	8.55 ft^2 cold leg pump discharge break w/ off-site power	Break size/location
1D	8.55 ft^2 cold leg pump suction break w/ off-site power	Break size/location
2A	14.1 ft^2 hot leg break at vessel outlet w/ Moody/Henry-Fauske critical flow model (w/ off-site power)	Critical flow model
2B	14.1 ft^2 hot leg break at vessel outlet w/ Ransom & Trapp model and C_D adjusted to match EM critical flow (w/ off-site power)	Critical flow model
2C	14.1 ft^2 hot leg break at SG inlet w/ Moody/Henry-Fauske critical flow model (w/ off-site power)	Critical flow model
2D	14.1 ft^2 hot leg break at SG inlet w/ Ransom & Trapp model and C_D adjusted to match EM critical flow (w/ off-site power)	Critical flow model
3A	14.1 ft^2 hot leg break at vessel outlet w/o off-site power and w/ adjusted C_D's.	Off-site power sensitivity
3B	14.1 ft^2 hot leg break at SG inlet w/o off-site power and w/ adjusted C_D's.	Off-site power sensitivity
3C	14.1 ft^2 hot leg break at vessel outlet w/ off-site power & 30 psia containment back-pressure.	Containment back-pressure
3D	14.1 ft^2 hot leg break at vessel outlet w/ off-site power & 50 psia containment back-pressure.	Containment back-pressure
3E	14.1 ft^2 hot leg break at vessel outlet w/ off-site power and zero refill time.	Refill sensitivity

TABLE 3.2-1

Principal Parameter Initial Conditions

Parameter	Value
Power level	2619.4 MW_{th}
RCS T-ave	581°F
RCS pressure	2185 psig
RCS flow	104% design flow
Pressurizer level	315 inches
CFT pressure	655 psig
CFT liquid volume	972 ft^3
CFT temperature	120°F
SG mass	[ ] lb/SG
SG Pressure	925 psig

TABLE 3.3-1

Common Boundary Conditions

Parameter	Value
Average fuel temperature	[ ]°F
Decay heat	1979 ANS + 2σ uncertainty
Axial power profile	[ ]
BWST level	44.3 ft
BWST temperature	115°F
MFW temperature	460°F
EFW temperature	120°F (130°F for small break analyses)
Single failure	4160V switchgear

TABLE 3.3-2

Moderator Density Reactivity Feedback

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TABLE 3.5-1 (1 of 4)
Peak Pressure Mass and Energy Release Data

	BREAK 1					BREAK 2				
	Gas		Liquid			Gas		Liquid		
Time (sec)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Pressure (psia)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Pressure (psia)
0.00	3271.4	1223.6	48383.5	583.1	1933.3	3550.6	1224.4	51140.0	582.4	1925.4
0.10	3271.4	1223.6	48383.5	583.1	1933.3	3550.6	1224.4	51140.0	582.4	1925.4
0.30	3643.2	1222.8	43916.5	574.1	1560.9	3341.9	1229.0	43030.0	578.3	1541.0
0.55	3709.0	1225.9	38740.0	569.0	1502.8	4403.3	1251.3	31366.7	550.7	1486.3
0.80	3811.5	1221.7	38435.0	569.1	1459.2	5324.5	1279.3	28600.0	513.5	1452.3
1.00	4027.0	1220.9	36905.0	568.2	1426.2	4687.5	1287.6	28530.0	504.4	1423.1
1.25	4762.0	1223.8	33393.3	561.1	1384.4	3858.3	1282.7	30653.3	502.2	1380.9
1.50	6174.5	1245.1	28600.0	532.5	1342.7	3203.5	1273.5	32845.0	503.3	1338.3
1.75	6775.0	1257.6	25976.7	522.5	1301.0	2251.0	1265.5	33950.0	514.6	1297.4
2.00	7173.5	1267.9	22770.0	511.0	1256.3	1775.0	1259.2	34090.0	520.0	1253.9
2.25	7163.3	1271.3	21260.0	505.8	1207.6	1815.0	1253.4	33380.0	519.4	1208.2
2.50	7405.0	1278.9	18345.0	498.2	1161.0	1917.5	1249.0	32185.0	518.4	1163.3
2.75	7466.7	1285.3	16163.3	491.2	1117.2	2020.3	1244.0	31046.7	517.2	1115.9
3.00	7270.0	1288.9	15345.0	486.2	1085.9	2135.0	1243.6	29785.0	515.9	1083.3
3.25	6973.3	1287.8	15593.3	483.5	1061.9	2271.7	1239.9	28266.7	513.0	1058.6
3.50	6615.0	1284.2	16150.0	481.1	1037.1	2405.0	1241.2	26750.0	511.0	1030.1
3.75	6236.7	1281.1	16906.7	479.5	1018.3	2836.7	1245.6	27066.7	497.9	1012.2
4.05	5930.0	1279.4	17200.0	476.6	994.5	3596.7	1255.8	26266.7	483.6	990.5
4.30	5795.0	1277.8	17000.0	474.4	975.0	3790.0	1261.2	24450.0	477.5	970.6
4.55	5713.3	1280.1	16533.3	472.8	954.5	3886.7	1264.2	22933.3	473.7	950.3
4.80	5665.0	1278.9	16050.0	468.5	934.2	3985.0	1266.0	21400.0	468.7	930.1
5.05	5613.3	1282.1	15466.7	467.2	917.9	4056.7	1272.0	19900.0	465.0	914.4
5.30	5590.0	1281.8	15150.0	464.0	900.2	4115.0	1278.3	18500.0	460.0	896.8
5.55	5463.3	1281.3	14600.0	461.9	876.0	4163.3	1278.6	17266.7	455.8	871.9
5.80	5310.0	1282.5	14350.0	457.8	853.0	4185.0	1282.0	16000.0	451.3	848.3

TABLE 3.5-1 (2 of 4)
Peak Pressure Mass and Energy Release Data

	BREAK 1					BREAK 2				
	Gas		Liquid			Gas		Liquid		
Time (sec)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Pressure (psia)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Pressure (psia)
6.05	5116.7	1280.1	14400.0	455.8	833.0	4233.3	1287.4	14633.3	445.3	827.4
6.30	5005.0	1279.7	14100.0	454.6	812.3	4305.0	1293.8	13100.0	441.2	806.4
6.55	5083.3	1282.6	12966.7	450.1	789.0	4363.3	1299.5	11666.7	435.7	783.4
6.80	5240.0	1286.3	11500.0	444.8	763.9	4370.0	1305.5	10450.0	432.5	758.8
7.05	5380.0	1291.8	9966.7	442.5	737.6	4356.7	1308.3	9433.3	425.8	732.8
7.30	5465.0	1296.4	8650.0	439.9	710.3	4320.0	1313.7	8450.0	421.3	705.7
7.55	5473.3	1299.0	7666.7	433.9	681.4	4273.3	1315.9	7566.7	417.6	676.8
7.80	5415.0	1302.9	6800.0	433.8	652.0	4205.0	1318.7	6850.0	413.1	647.7
8.05	5293.3	1301.6	6200.0	426.3	616.1	4136.7	1319.1	6166.7	408.1	613.5
8.30	5115.0	1301.1	5650.0	420.4	573.9	4075.0	1320.3	5400.0	402.8	574.7
8.55	4873.3	1300.3	5066.7	414.5	535.0	4023.3	1323.1	4400.0	398.5	537.8
8.80	4610.0	1299.4	4550.0	408.8	500.1	3950.0	1330.4	3350.0	395.5	502.8
9.05	4360.0	1301.2	4100.0	402.4	465.5	3786.7	1338.0	2633.3	383.5	467.4
9.30	4115.0	1302.6	3650.0	398.6	432.0	3535.0	1340.9	2250.0	382.2	433.2
9.55	3890.0	1299.1	3300.0	386.9	400.2	3270.0	1337.4	2166.7	370.8	401.3
9.80	3675.0	1300.7	2800.0	389.3	367.6	2995.0	1333.9	2100.0	361.9	368.8
10.05	3446.7	1300.8	2300.0	375.4	333.8	2710.0	1322.3	2200.0	359.1	336.1
10.30	3135.0	1298.3	2100.0	369.1	302.9	2630.0	1319.4	1900.0	347.4	305.3
10.55	2770.0	1288.8	2233.3	355.2	268.6	2476.7	1309.6	1700.0	349.0	269.3
10.80	2490.0	1289.2	2100.0	350.0	247.7	2340.0	1312.0	1650.0	345.5	254.3
11.05	2333.3	1288.6	1900.0	347.4	240.0	2360.0	1309.3	1233.3	343.2	251.7
11.30	2140.0	1287.4	1900.0	336.8	223.5	2400.0	1318.8	700.0	342.9	230.6
11.55	2013.3	1283.1	1800.0	331.5	209.0	2420.0	1338.8	300.0	333.3	211.6
11.80	1860.0	1279.6	1700.0	326.5	194.4	2160.0	1321.8	100.0	400.0	196.6
12.05	1680.0	1269.8	1600.0	320.8	165.7	1990.0	1291.5	200.0	333.3	169.6

TABLE 3.5-1 (3 of 4)
Peak Pressure Mass and Energy Release Data

	BREAK 1					BREAK 2				
	Gas		Liquid			Gas		Liquid		
Time (sec)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Pressure (psia)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Pressure (psia)
12.30	1420.0	1264.1	1500.0	310.0	143.0	1885.0	1270.6	150.0	333.3	149.5
12.55	1226.7	1241.9	1433.3	295.4	134.6	1720.0	1255.8	66.7	300.0	136.5
12.80	1050.0	1233.3	1350.0	292.6	114.6	1550.0	1248.4	50.0	400.0	118.9
13.05	916.7	1232.7	1266.7	289.5	106.1	1366.7	1241.5	33.3	200.0	113.5
13.30	855.0	1216.4	1250.0	288.0	102.8	1210.0	1231.4	50.0	200.0	107.3
13.55	736.7	1203.6	1066.7	284.4	88.1	996.7	1220.7	33.3	500.0	93.3
13.80	570.0	1193.0	800.0	275.0	73.2	750.0	1200.0	100.0	200.0	76.6
14.05	400.0	1166.7	600.0	277.8	64.4	410.0	1178.9	66.7	250.0	66.9
14.30	310.0	1161.3	450.0	300.0	60.3	15.0	1333.3	0.0	180.0	61.4
14.55	230.0	1188.4	400.0	258.3	55.9	0.0	1150.0	0.0	180.0	57.6
14.80	180.0	1138.9	250.0	300.0	53.9	0.0	1150.0	0.0	180.0	55.8
14.90	180.0	1138.9	250.0	300.0	53.9	0.0	1150.0	0.0	180.0	55.8
14.91	445.7	1188.3	715.7	279.7	87.6	13.3	1191.2	0.0	180.0	80.9
17.45	445.7	1188.3	715.7	279.7	87.6	13.3	1191.2	0.0	180.0	80.9
22.50	774.0	1215.3	1196.0	285.5	112.9	11.2	1214.3	0.0	180.0	105.1
27.50	543.4	1207.2	1358.0	280.9	93.5	15.6	1230.8	0.0	180.0	93.1
32.50	219.0	1190.9	1948.0	277.2	79.4	10.6	1245.3	0.0	180.0	78.8
37.50	121.6	1187.5	1384.0	276.7	78.2	15.2	1223.7	0.0	180.0	78.0
42.50	98.6	1188.6	2390.0	276.5	78.2	19.8	1252.5	0.0	180.0	78.4
47.50	49.8	1184.7	980.0	276.1	75.7	18.6	1268.8	0.0	180.0	75.7
50.00	49.8	1184.7	980.0	276.1	75.7	18.6	1268.8	0.0	180.0	75.7
50.01	27.6	1183.6	320.0	277.1	75.6	20.9	1258.0	0.0	180.0	75.6
57.50	27.6	1183.6	320.0	277.1	75.6	20.9	1258.0	0.0	180.0	75.6
73.00	14.4	1186.2	47.5	276.3	75.8	21.3	1261.8	0.0	180.0	75.8
89.00	12.6	1183.2	34.4	272.7	77.2	22.3	1261.2	0.0	180.0	77.2

TABLE 3.5-1 (4 of 4)
Peak Pressure Mass and Energy Release Data

	BREAK 1					BREAK 2				
	Gas		Liquid			Gas		Liquid		
Time (sec)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Pressure (psia)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Pressure (psia)
105.00	24.8	1189.4	994.4	276.6	78.5	26.8	1268.7	0.0	180.0	78.3
121.00	14.4	1187.0	324.4	275.5	77.1	19.3	1263.0	0.0	180.0	76.9
137.00	6.5	1182.7	0.0	180.0	77.8	23.2	1264.2	0.0	180.0	77.6
153.00	15.9	1192.2	610.6	276.4	80.1	28.9	1272.1	0.0	180.0	79.8
169.00	18.8	1186.1	845.6	277.2	78.5	22.8	1261.0	0.0	180.0	78.2
178.55	2.9	1222.2	0.0	180.0	76.4	19.4	1283.3	0.0	180.0	76.2
180.10	2.9	1222.2	0.0	180.0	76.4	19.4	1283.3	0.0	180.0	76.2

TABLE 3.6-1 (1 of 3)
RELAP5 Long-Term Mass and Energy Release Data

	BREAK 1					BREAK 2				
	Gas		Liquid			Gas		Liquid		
Time (sec)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Pressure (psia)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Pressure (psia)
0.50	2.0	1190.4	47261.0	557.6	1683.3	456.0	1194.1	25759.0	541.4	1636.0
1.50	43.4	1189.9	41385.0	555.6	1099.5	684.5	1193.4	25238.0	544.5	998.8
2.50	271.0	1189.4	32454.0	558.7	1103.1	1152.0	1196.1	19507.0	539.7	934.8
3.50	757.9	1190.5	27480.0	557.0	1085.1	1511.2	1198.5	14977.0	528.6	810.8
4.50	1335.2	1192.3	22590.0	552.9	1036.0	2419.4	1206.0	11285.0	492.7	690.1
5.50	1554.8	1193.7	19020.0	546.1	964.5	2607.0	1208.6	8904.0	478.0	611.6
6.50	1544.2	1195.9	18040.0	531.6	899.2	2553.9	1209.5	8150.0	471.2	562.4
7.50	1551.3	1197.8	16760.0	520.9	843.6	2743.0	1212.5	5980.0	457.7	506.4
8.50	1712.8	1200.1	14220.0	512.7	777.7	2724.0	1213.3	4970.0	447.9	461.5
9.50	2075.4	1205.1	13030.0	491.9	694.1	2607.0	1214.0	4340.0	439.2	421.9
10.50	2120.0	1207.1	10960.0	478.1	611.3	2504.0	1215.3	3500.0	427.7	377.5
11.50	2014.0	1208.5	9230.0	459.4	514.9	2383.0	1216.1	2610.0	414.6	329.4
12.50	985.0	1207.1	11760.0	420.1	393.8	2216.0	1217.1	1830.0	403.3	281.8
13.50	770.0	1205.2	12220.0	392.0	289.6	2042.0	1217.9	1220.0	386.9	239.4
14.50	459.0	1198.3	10560.0	365.5	216.8	1849.0	1217.4	720.0	372.2	198.6
15.50	220.0	1195.5	8950.0	336.3	165.5	1664.0	1220.0	240.0	370.8	160.4
16.50	143.0	1202.8	8690.0	316.5	151.6	1428.0	1217.1	120.0	333.3	130.7
17.50	222.0	1193.7	8660.0	308.3	145.8	1182.0	1215.7	50.0	340.0	105.3
18.50	227.0	1185.0	8620.0	277.3	104.7	983.0	1211.6	30.0	266.7	84.7
19.50	258.0	1189.9	7090.0	273.6	71.2	744.0	1205.7	40.0	325.0	72.7
20.50	84.0	1178.6	7750.0	242.6	55.9	527.0	1195.5	130.0	284.6	67.3
21.50	23.0	1130.4	6440.0	201.9	42.2	437.0	1187.6	170.0	270.6	65.6

TABLE 3.6-1 (2 of 3)
RELAP5 Long-Term Mass and Energy Release Data

	BREAK 1					BREAK 2				
	Gas		Liquid			Gas		Liquid		
Time (sec)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Pressure (psia)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Pressure (psia)
22.50	14.0	1214.3	6100.0	177.1	43.4	387.0	1191.2	180.0	272.2	64.9
23.50	11.0	1090.9	5400.0	168.5	46.2	352.0	1190.3	180.0	266.7	64.2
24.50	9.0	1333.3	4730.0	169.1	48.7	308.0	1188.3	170.0	288.2	63.7
25.50	8.0	1250.0	4120.0	172.3	48.2	257.0	1206.2	140.0	264.3	63.4
26.00	8.0	1250.0	4120.0	172.3	48.2	257.0	1206.2	140.0	264.3	63.4
26.01	0.5	1000.0	179.0	173.2	53.0	53.4	1282.8	8.0	250.0	62.7
31.00	0.5	1000.0	179.0	173.2	53.0	53.4	1282.8	8.0	250.0	62.7
41.00	210.6	1184.2	1163.0	221.0	61.4	240.9	1210.1	88.0	269.3	61.9
51.00	213.1	1181.1	278.0	266.2	61.6	193.5	1195.4	182.0	264.8	61.1
61.00	132.5	1182.6	173.0	265.9	60.4	152.0	1194.1	157.0	264.3	60.2
71.00	111.0	1182.9	141.0	255.3	59.4	132.8	1192.0	166.0	261.5	59.3
81.00	72.7	1182.9	93.0	268.8	58.8	113.2	1188.2	179.0	262.6	58.7
91.00	58.1	1184.2	57.0	263.2	58.6	101.6	1188.0	157.0	260.5	58.5
101.00	37.5	1186.7	37.0	243.2	58.6	91.5	1185.8	144.0	261.8	58.5
111.00	66.8	1187.1	101.0	267.3	58.8	93.2	1185.6	137.0	260.6	58.5
121.00	177.1	1183.5	230.0	260.9	59.4	121.3	1183.0	200.0	261.0	58.6
131.00	166.7	1180.6	209.0	263.2	59.2	115.7	1185.0	148.0	262.8	58.6
141.00	114.1	1183.2	81.0	259.3	58.5	85.7	1182.0	131.0	259.5	58.5
151.00	97.2	1186.2	108.0	259.3	58.5	79.1	1183.3	141.0	261.7	58.5
161.00	95.2	1180.7	110.0	254.6	58.4	74.9	1180.2	149.0	261.1	58.5
171.00	52.4	1185.1	32.0	281.3	58.4	57.2	1181.8	134.0	260.5	58.4
176.00	52.4	1185.1	32.0	281.3	58.4	57.2	1181.8	134.0	260.5	58.4
176.01	29.4	1182.2	114.1	257.7	57.2	18.7	1206.7	4.8	260.3	57.2
236.05	29.4	1182.2	114.1	257.7	57.2	18.7	1206.7	4.8	260.3	57.2

TABLE 3.6-1 (3 of 3)
RELAP5 Long-Term Mass and Energy Release Data

	BREAK 1					BREAK 2				
	Gas		Liquid			Gas		Liquid		
Time (sec)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Pressure (psia)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Mass Flow (lb/sec)	Enthalpy (btu/lb)	Pressure (psia)
356.10	21.5	1180.0	336.5	240.7	54.6	8.9	1188.7	8.3	256.6	54.7
476.10	23.4	1177.3	452.8	217.6	52.0	9.3	1179.7	22.3	253.6	52.0
596.10	23.8	1179.9	505.9	214.5	49.4	4.5	1187.5	1.8	247.6	49.4
716.10	12.2	1176.9	520.3	202.3	47.2	3.8	1186.4	0.8	260.0	47.2
836.10	10.5	1177.6	538.9	186.6	45.7	2.8	1188.8	0.1	100.0	45.8
956.10	9.3	1178.6	543.3	185.3	44.6	2.2	1193.8	0.0	180.0	44.6
1077.10	8.4	1175.8	529.0	180.2	43.5	1.5	1196.8	0.0	180.0	43.5
1198.10	7.9	1174.6	548.0	178.7	42.4	1.1	1209.0	0.0	180.0	42.4
1318.10	6.3	1173.5	547.3	178.2	41.2	1.3	1206.5	0.0	180.0	41.2
1438.10	8.0	1173.8	527.3	171.7	40.1	0.9	1213.6	0.0	180.0	40.1
1558.10	13.6	1170.9	595.8	195.4	39.0	16.8	1183.7	35.3	235.4	39.0
1678.35	6.4	1172.3	429.9	187.6	37.9	2.5	1207.9	0.1	100.0	37.9
1769.35	35.0	1171.1	897.6	192.0	38.0	37.7	1208.5	51.9	233.2	37.1
1800.10	35.0	1171.1	897.6	192.0	38.0	37.7	1208.5	51.9	233.2	37.1

FIGURE 3.3-1

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FIGURE 3.3-2

[(page intentionally deleted)]

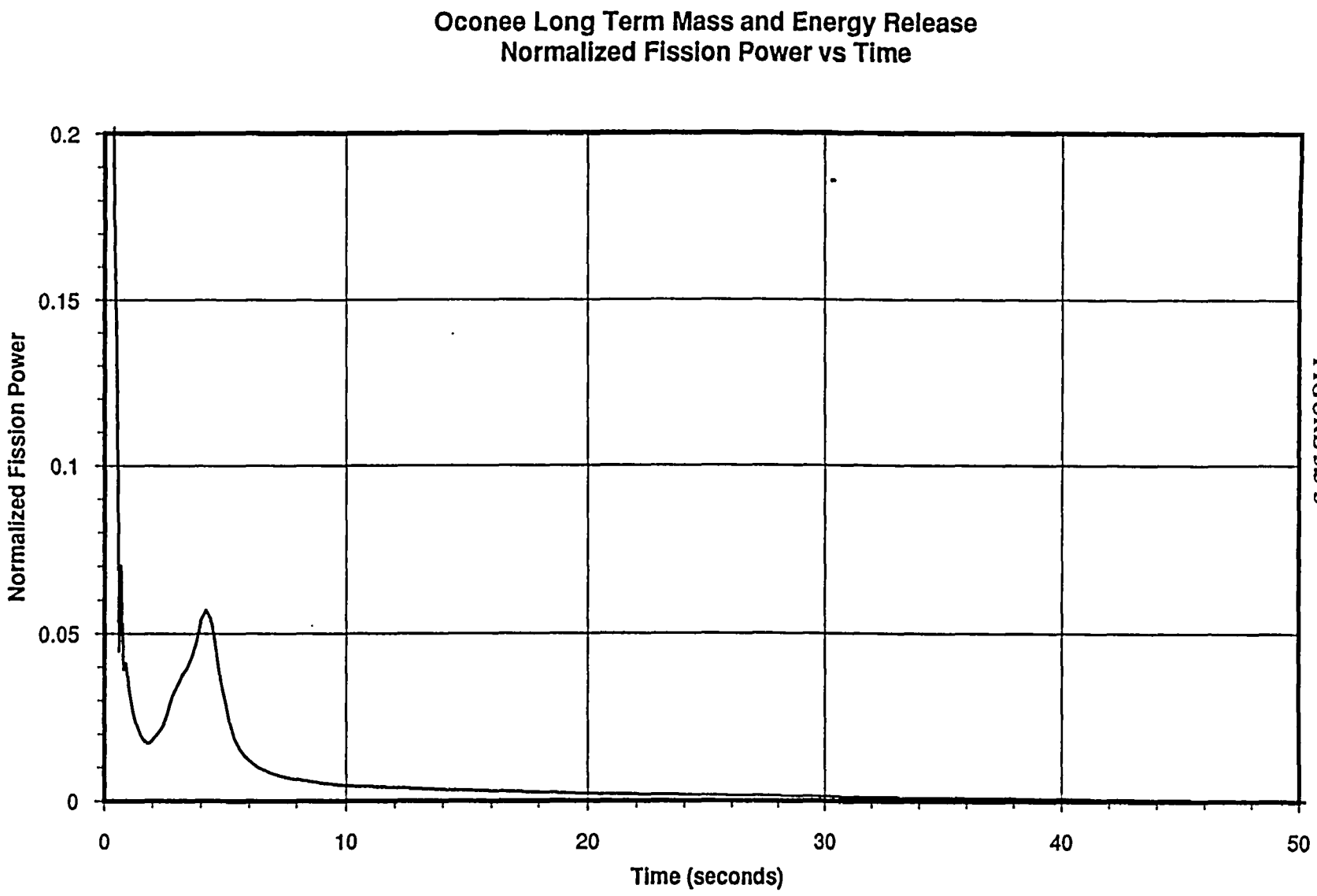

Oconee Long Term Mass and Energy Release
Normalized Fission Power vs Time
Normalized Fission Power
0.2
0.15
0.1
0.05
0
0
10
20
30
40
50
Time (seconds)

FIGURE 3.3-3

FIGURE 3.3-4

[(page intentionally deleted)]

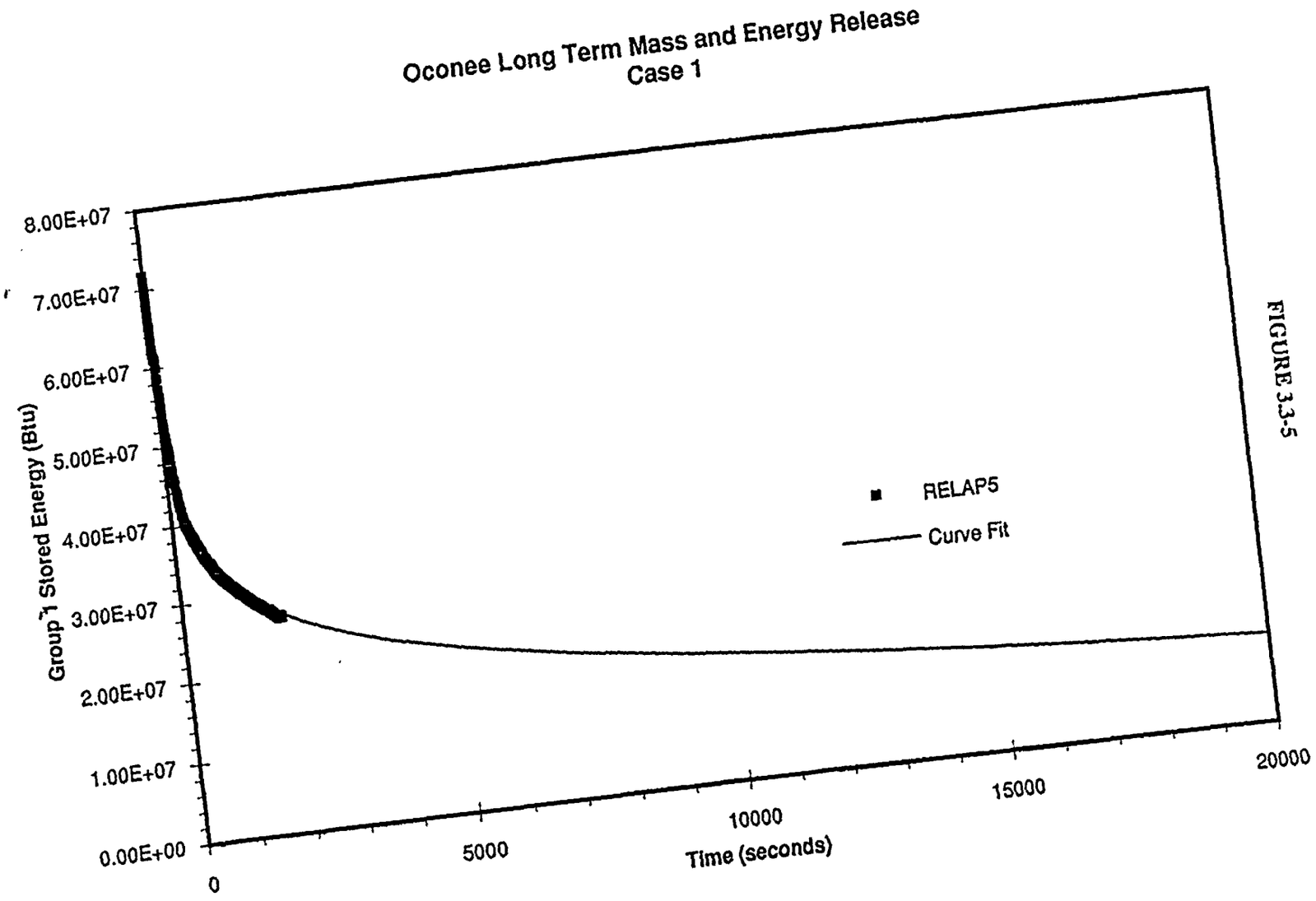

Oconee Long Term Mass and Energy Release
Case 1
Group 1 Stored Energy (Btu)
8.00E+07
7.00E+07
6.00E+07
5.00E+07
4.00E+07
3.00E+07
2.00E+07
1.00E+07
0.00E+00
0
5000
10000
15000
20000
Time (seconds)
RELAP5
Curve Fit

FIGURE 3.3-5

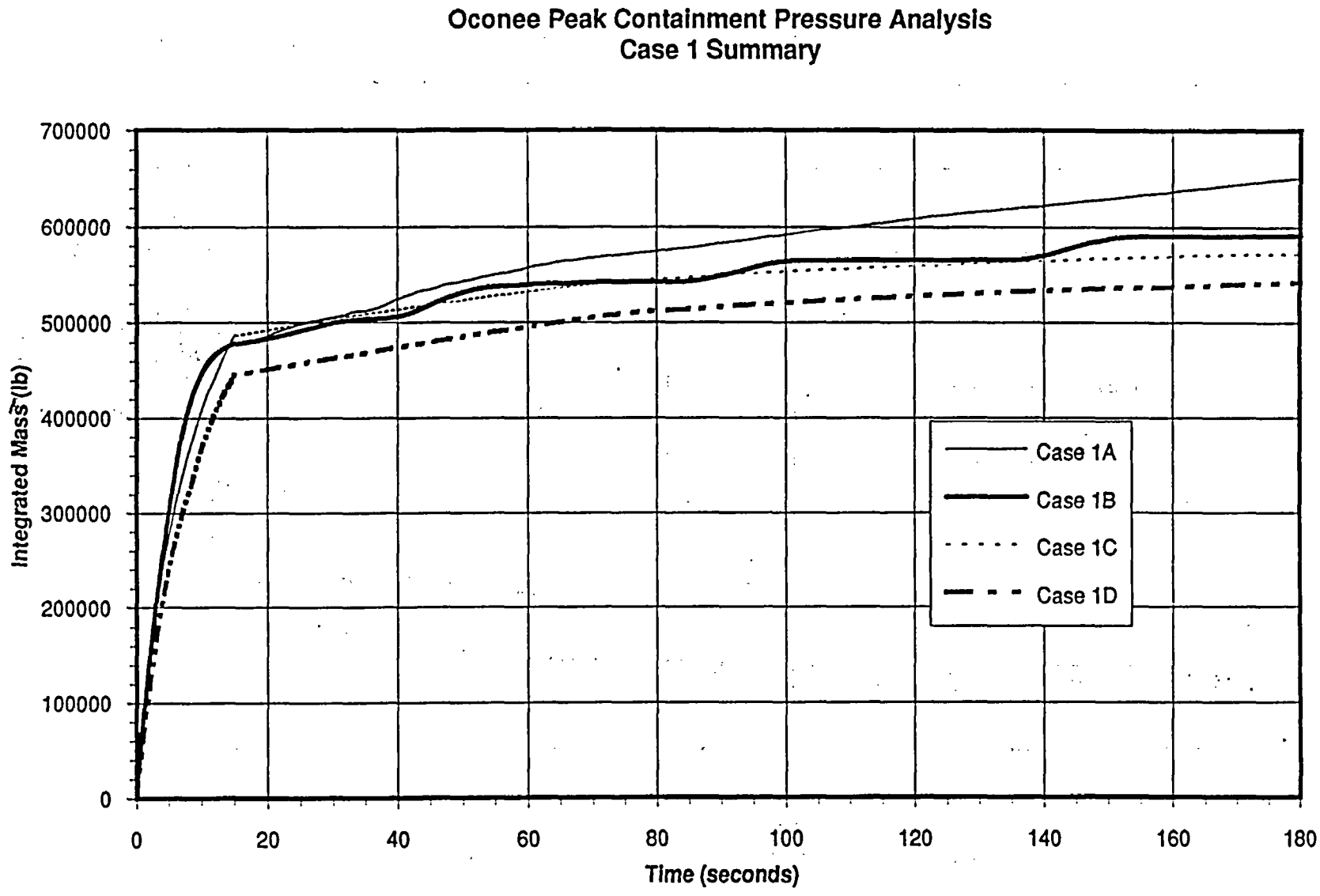

Oconee Peak Containment Pressure Analysis
Case 1 Summary
Integrated Mass (lb)
700000
600000
500000
400000
300000
200000
100000
0
0
20
40
60
80
100
120
140
160
180
Time (seconds)
Case 1A
Case 1B
Case 1C
Case 1D

FIGURE 3.5-1

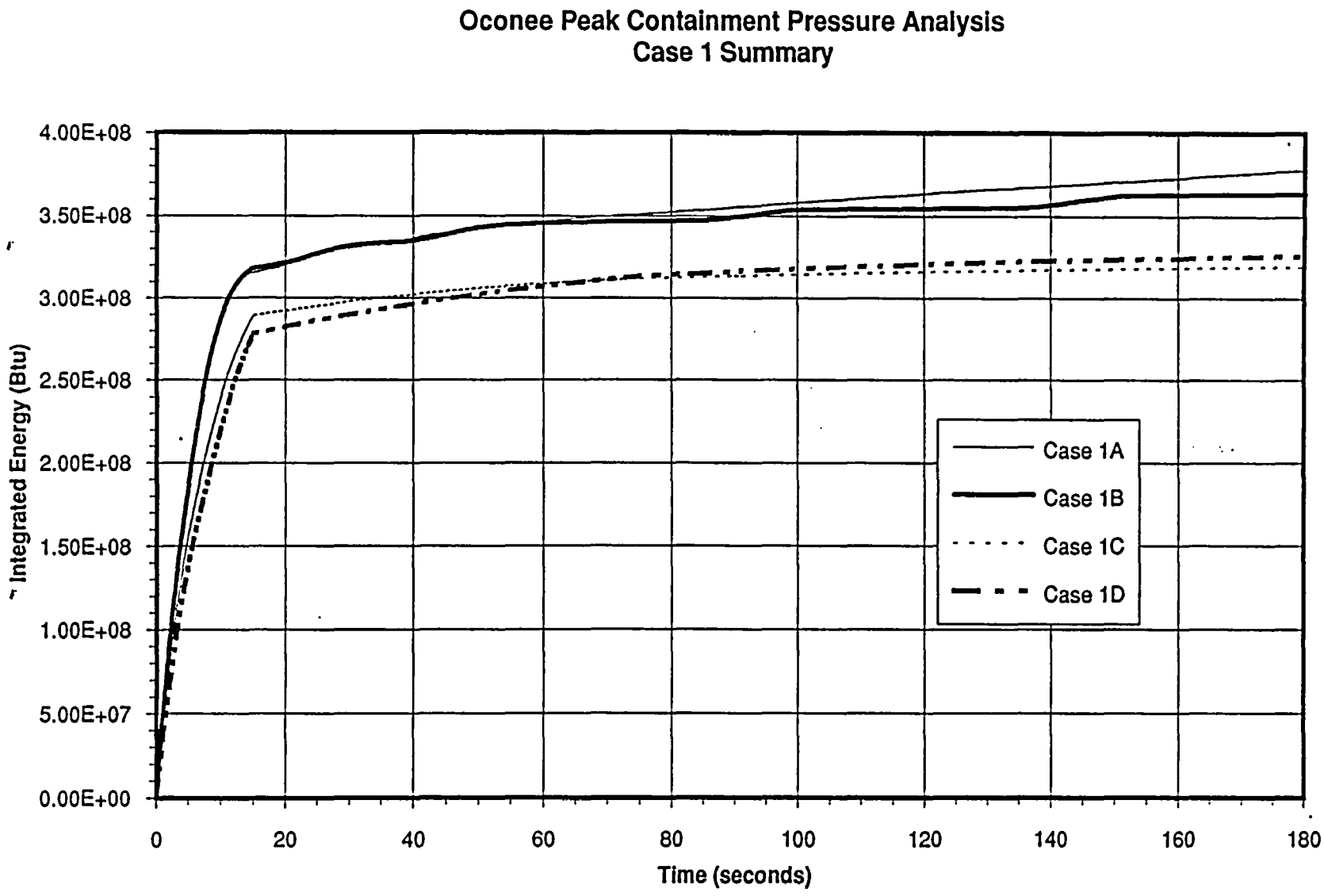

Oconee Peak Containment Pressure Analysis
Case 1 Summary
Integrated Energy (Btu)
4.00E+08
3.50E+08
3.00E+08
2.50E+08
2.00E+08
1.50E+08
1.00E+08
5.00E+07
0.00E+00
0
20
40
60
80
100
120
140
160
180
Time (seconds)
Case 1A
Case 1B
Case 1C
Case 1D

FIGURE 3.5-2

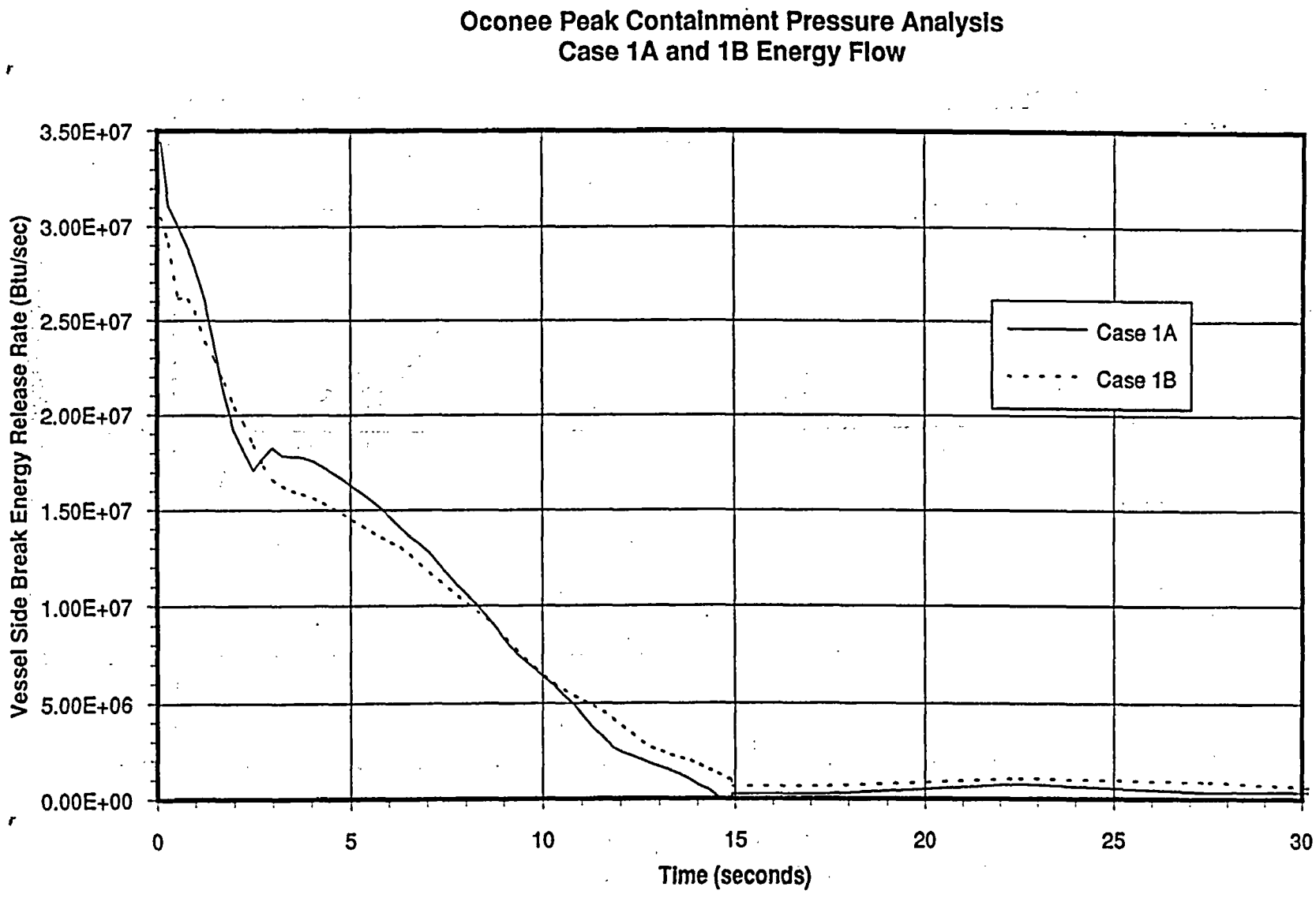

Oconee Peak Containment Pressure Analysis
Case 1A and 1B Energy Flow
Vessel Side Break Energy Release Rate (Btu/sec)
3.50E+07
3.00E+07
2.50E+07
2.00E+07
1.50E+07
1.00E+07
5.00E+06
0.00E+00
0
5
10
15
20
25
30
Time (seconds)
Case 1A
Case 1B

FIGURE 3.5-3

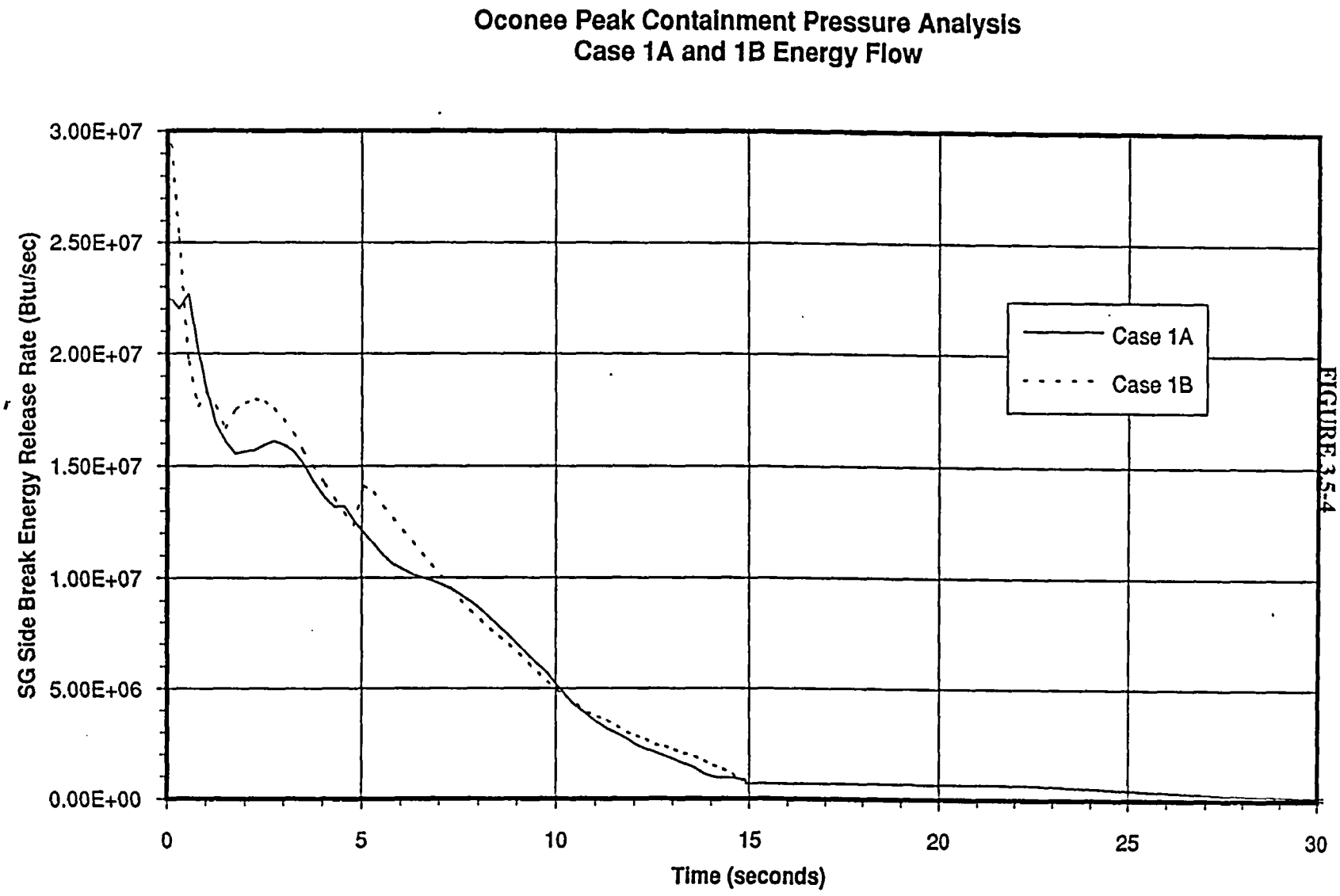

Oconee Peak Containment Pressure Analysis
Case 1A and 1B Energy Flow
SG Side Break Energy Release Rate (Btu/sec)
3.00E+07
2.50E+07
2.00E+07
1.50E+07
1.00E+07
5.00E+06
0.00E+00
0
5
10
15
20
25
30
Time (seconds)
Case 1A
Case 1B

FIGURE 3.5-4

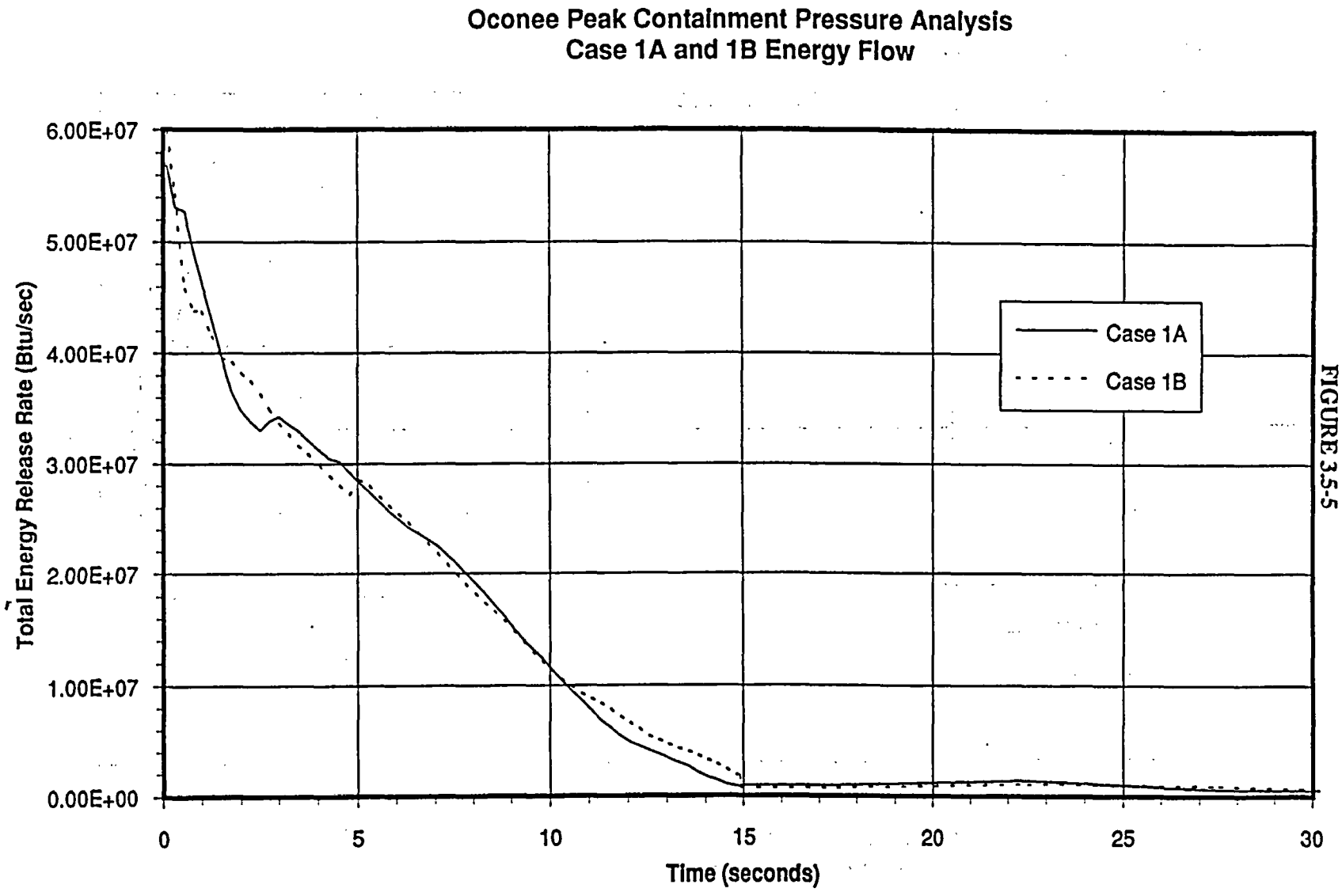

Oconee Peak Containment Pressure Analysis
Case 1A and 1B Energy Flow
Total Energy Release Rate (Btu/sec)
6.00E+07
5.00E+07
4.00E+07
3.00E+07
2.00E+07
1.00E+07
0.00E+00
0
5
10
15
20
25
30
Time (seconds)
Case 1A
Case 1B

FIGURE 3.5-5

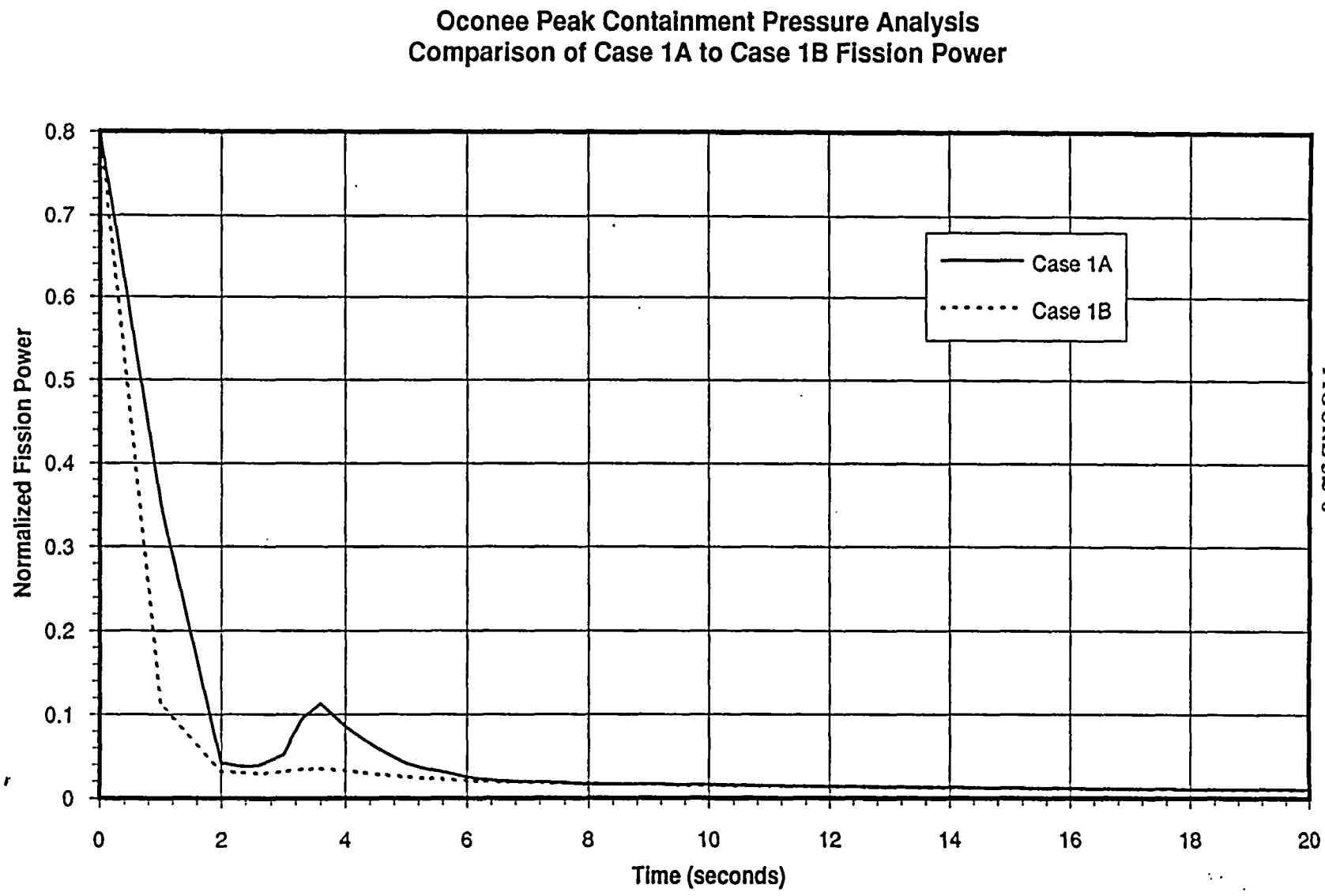

Oconee Peak Containment Pressure Analysis
Comparison of Case 1A to Case 1B Fission Power
Normalized Fission Power
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0
2
4
6
8
10
12
14
16
18
20
Time (seconds)
Case 1A
Case 1B

FIGURE 3.5-6

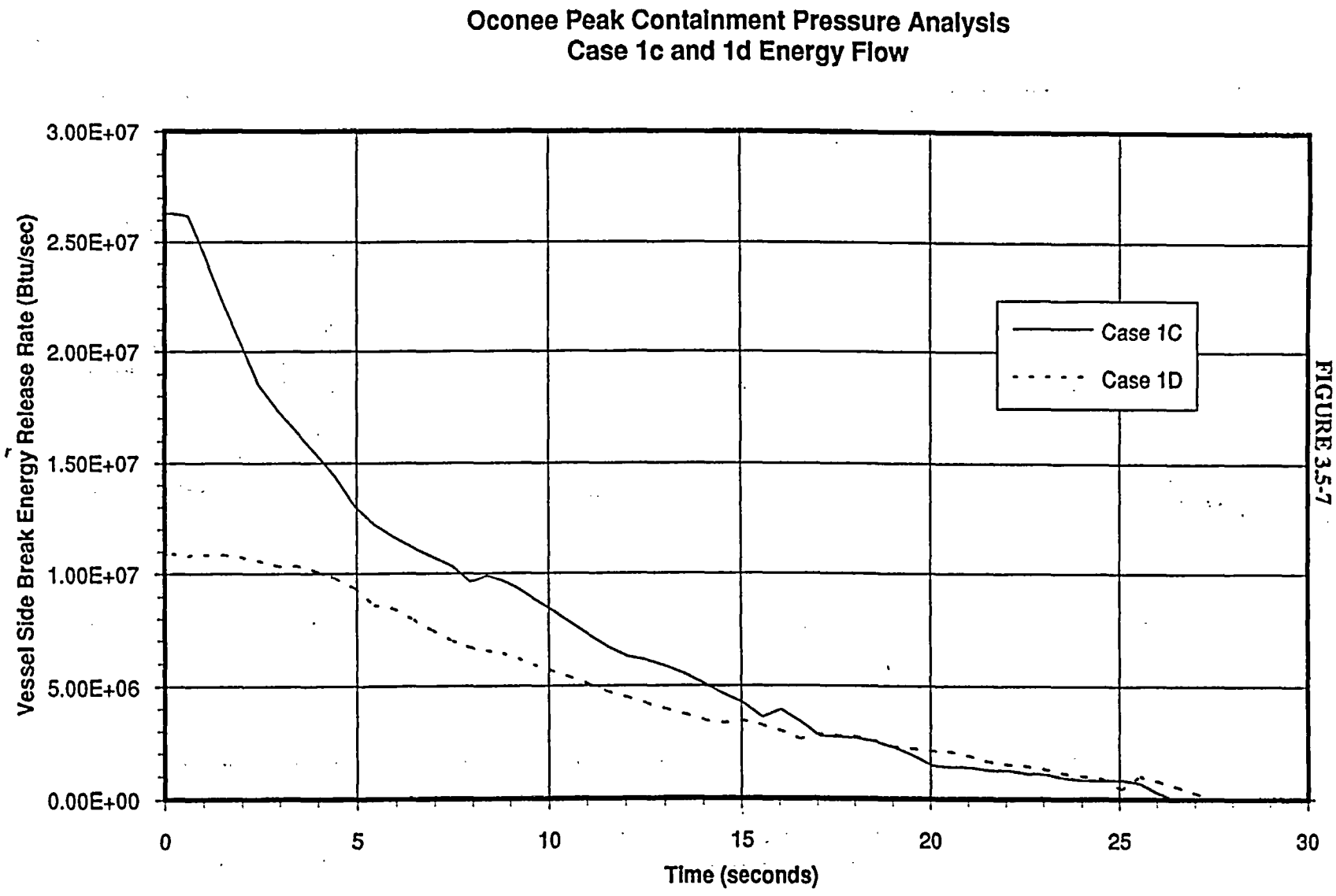

Oconee Peak Containment Pressure Analysis
Case 1c and 1d Energy Flow
Vessel Side Break Energy Release Rate (Btu/sec)
3.00E+07
2.50E+07
2.00E+07
1.50E+07
1.00E+07
5.00E+06
0.00E+00
0
5
10
15
20
25
30
Time (seconds)
Case 1C
Case 1D

FIGURE 3.5-7

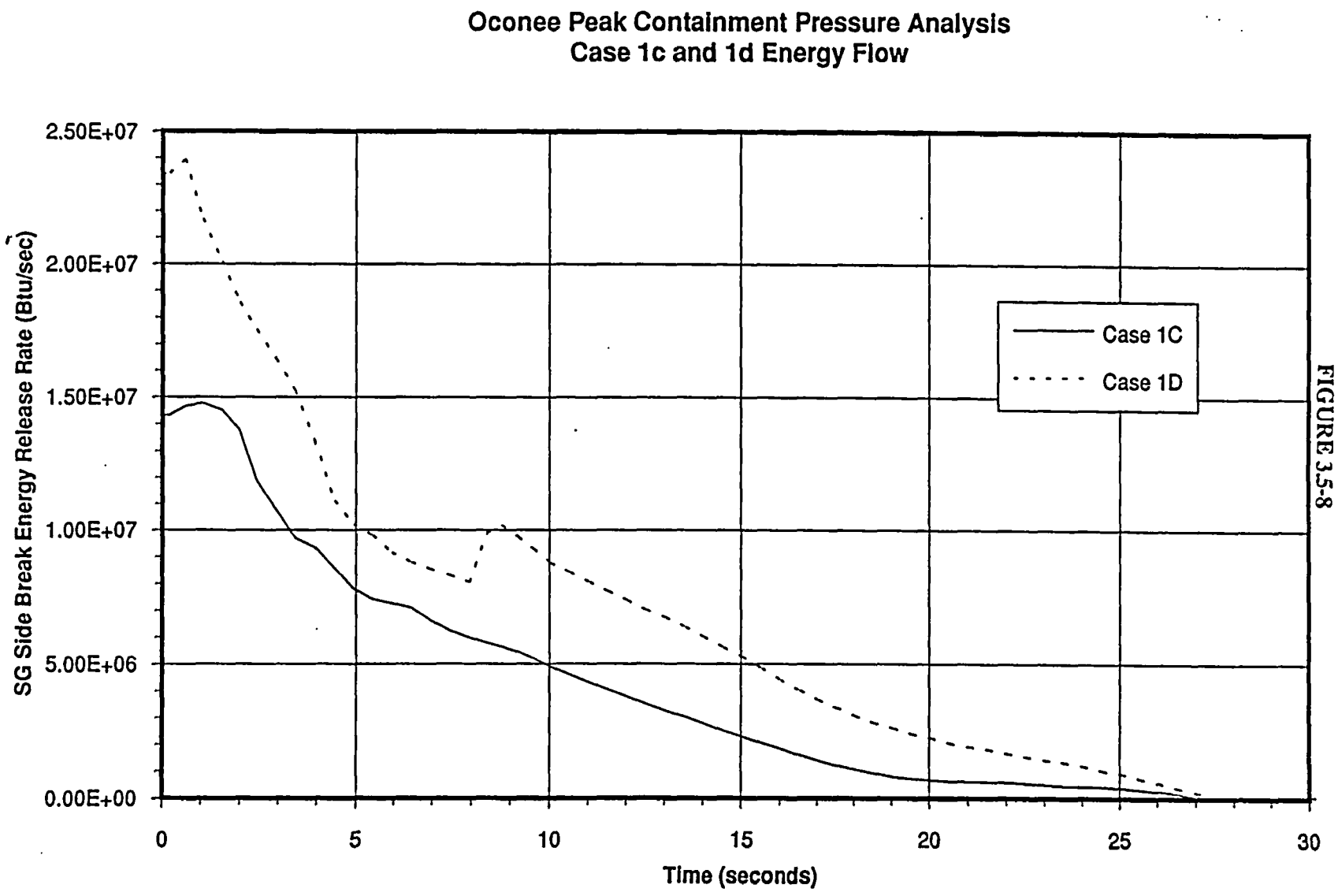

Oconee Peak Containment Pressure Analysis
Case 1c and 1d Energy Flow
SG Side Break Energy Release Rate (Btu/sec)
2.50E+07
2.00E+07
1.50E+07
1.00E+07
5.00E+06
0.00E+00
0
5
10
15
20
25
30
Time (seconds)
Case 1C
Case 1D

FIGURE 3.5-8

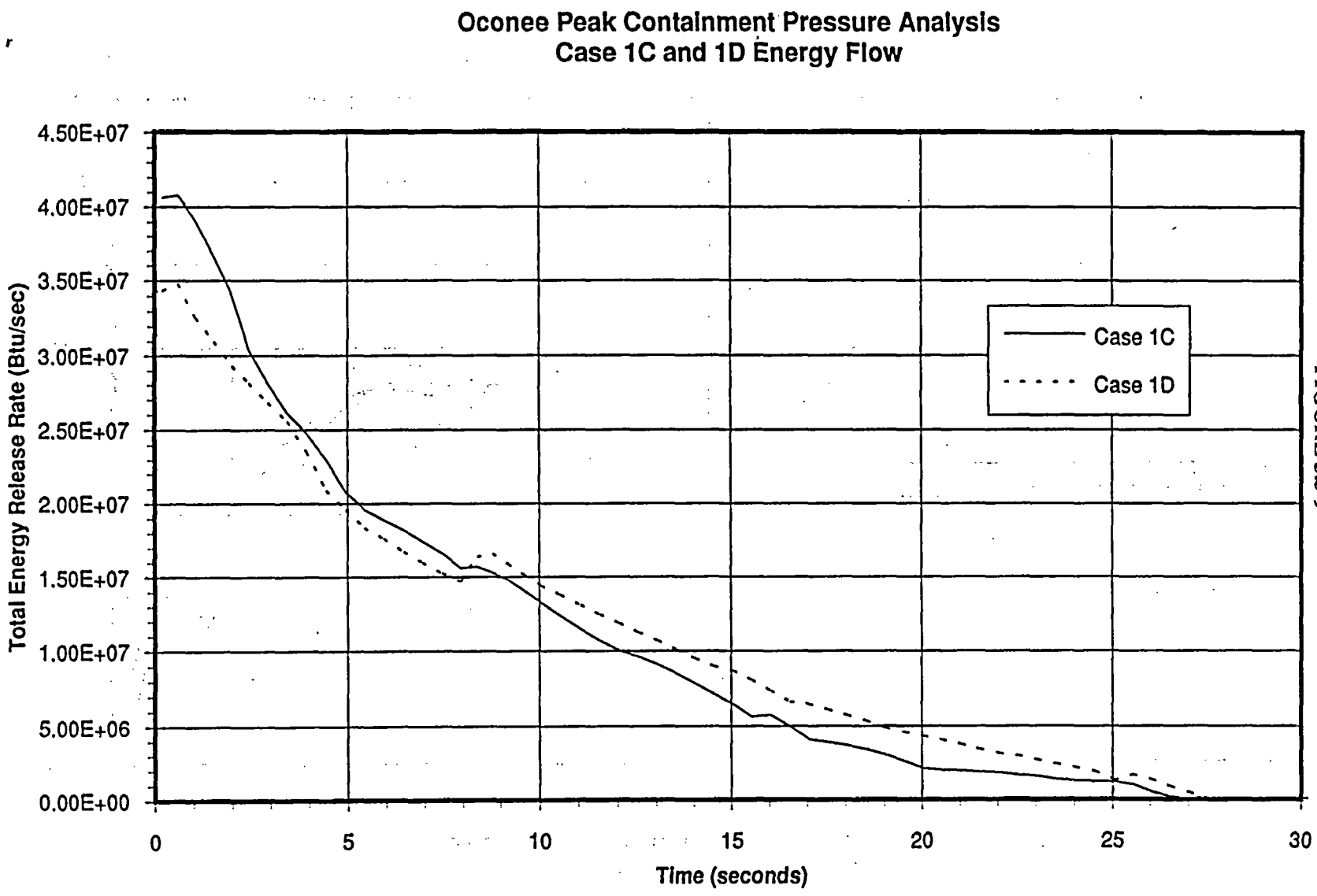

Oconee Peak Containment Pressure Analysis
Case 1C and 1D Energy Flow
Total Energy Release Rate (Btu/sec)
4.50E+07
4.00E+07
3.50E+07
3.00E+07
2.50E+07
2.00E+07
1.50E+07
1.00E+07
5.00E+06
0.00E+00
0
5
10
15
20
25
30
Time (seconds)
Case 1C
Case 1D

FIGURE 3.5-9

Oconee Peak Pressure Analysis
Vessel Side Break - Determination of Discharge Coefficient

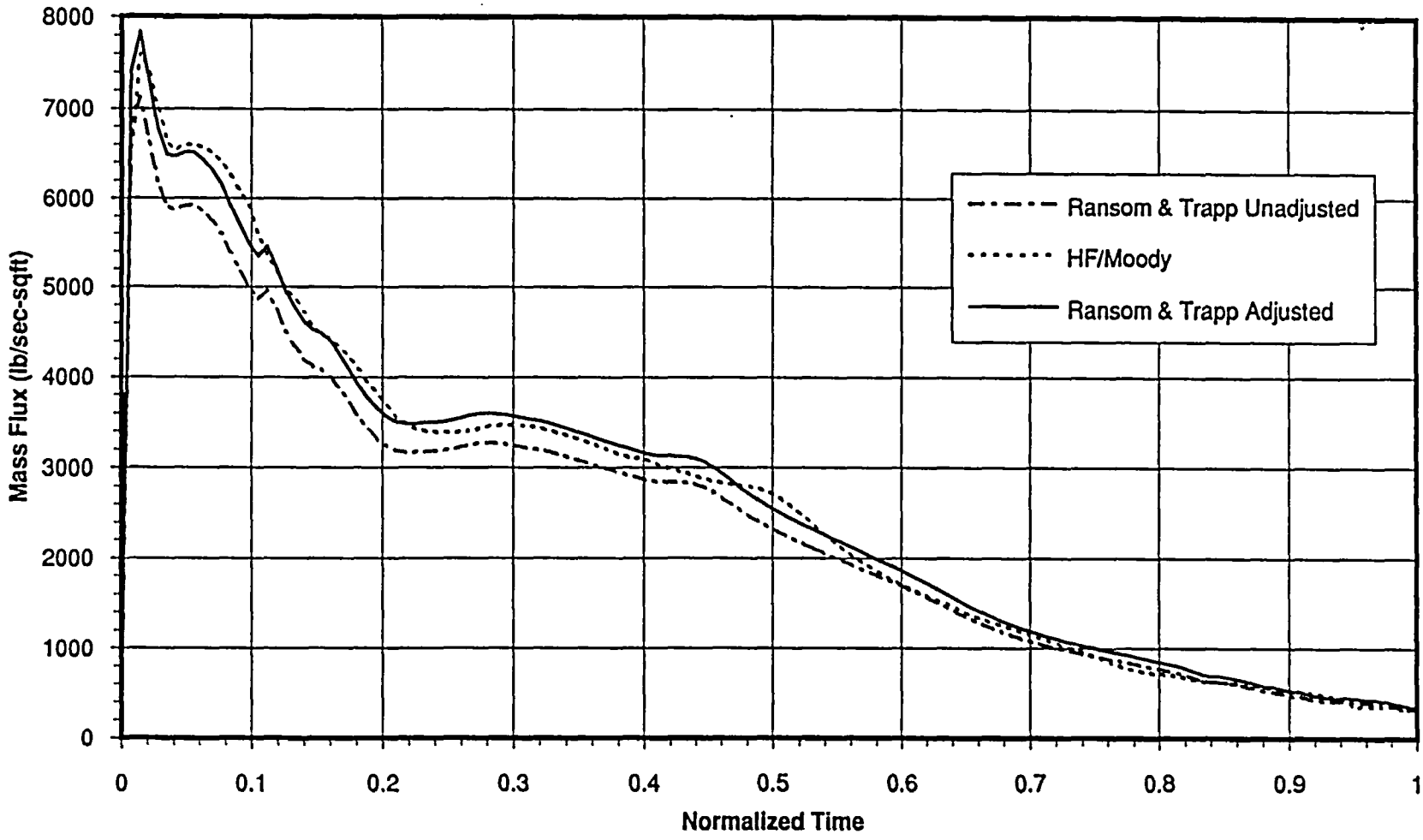


FIGURE 3.5-10

Oconee Peak Pressure Analysis
SG Side Break - Determination of Discharge Coefficient

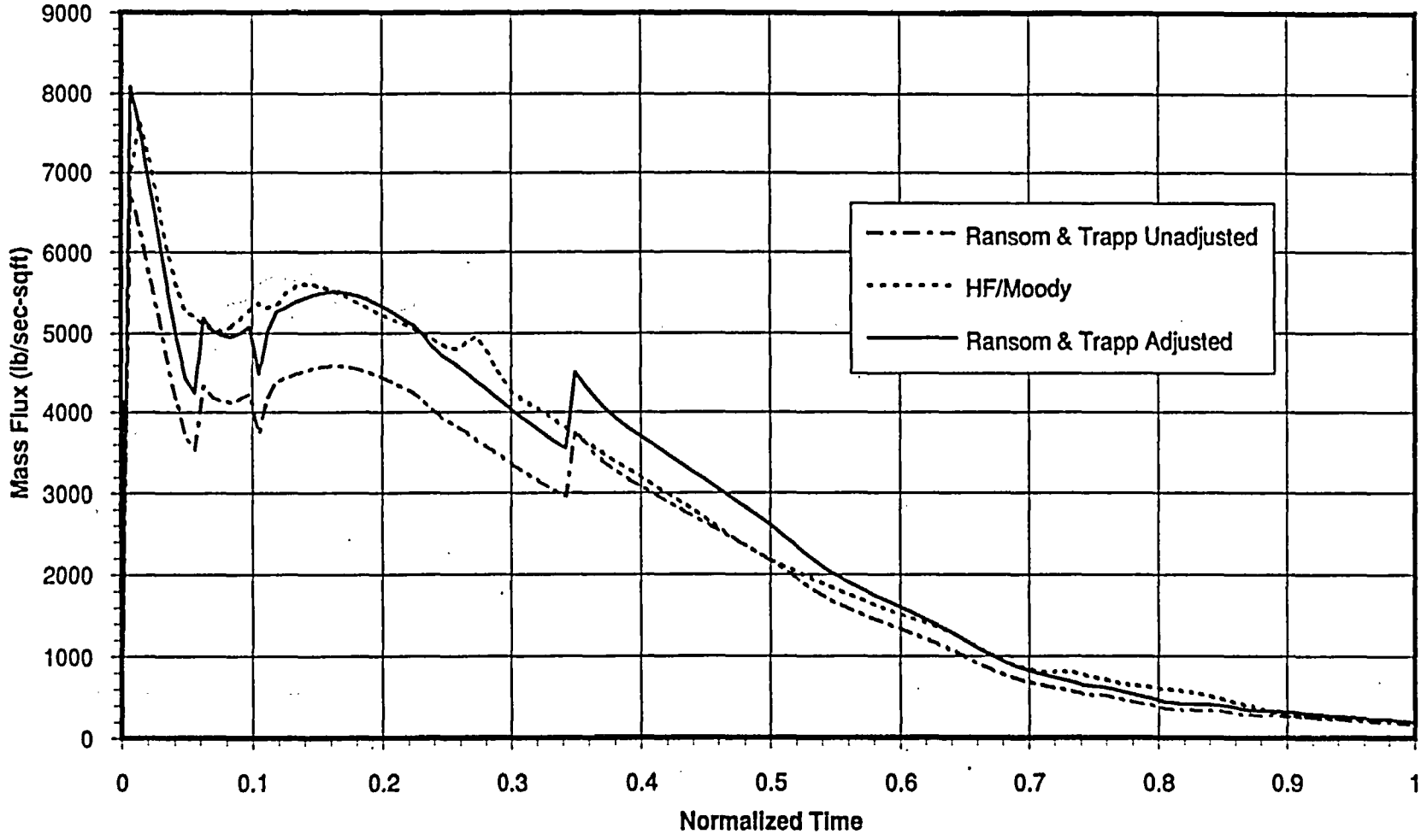


FIGURE 3.5-11

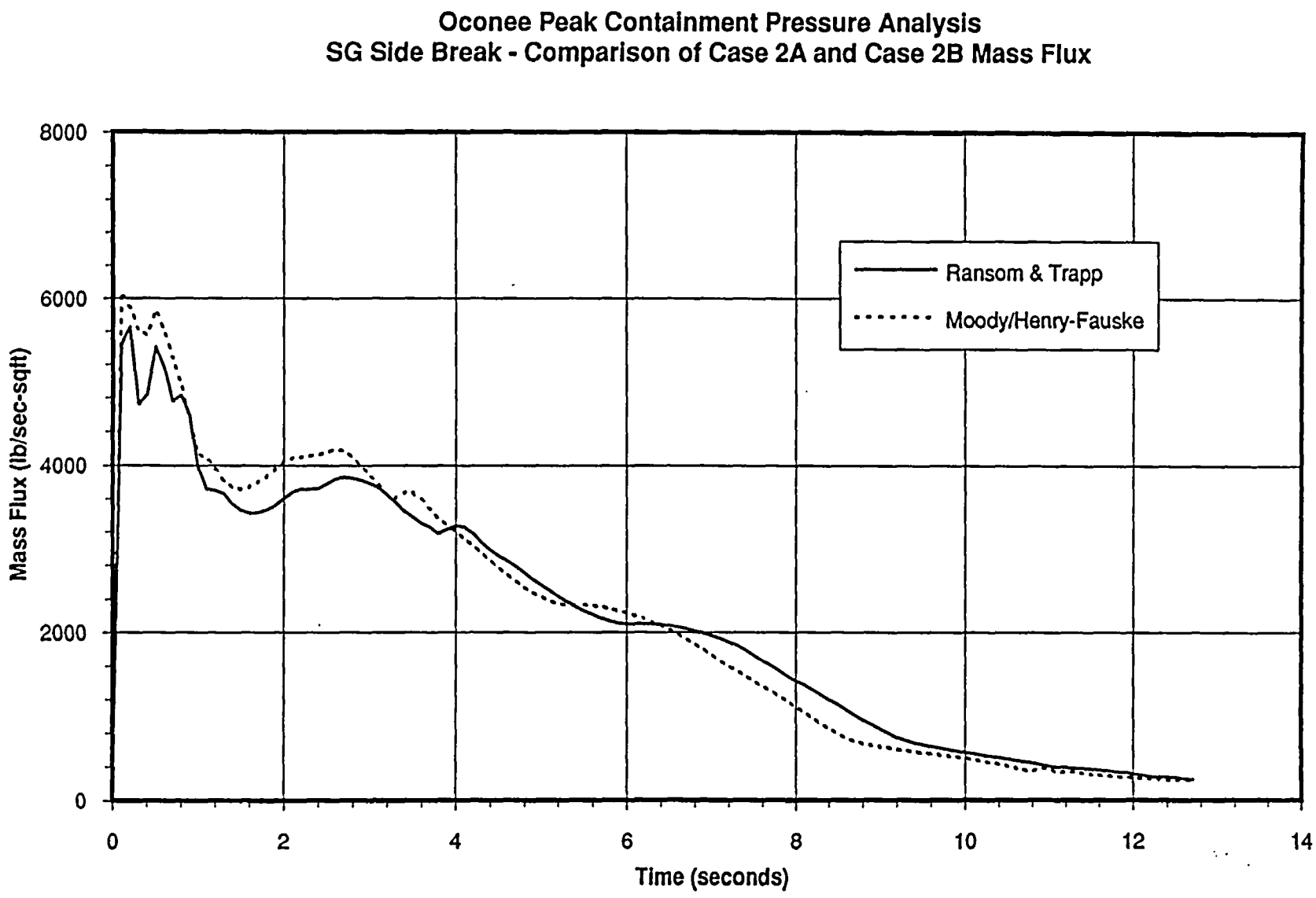

Oconee Peak Containment Pressure Analysis
SG Side Break - Comparison of Case 2A and Case 2B Mass Flux
Mass Flux (lb/sec-sqft)
8000
6000
4000
2000
0
0
2
4
6
8
10
12
14
Time (seconds)
Ransom & Trapp
Moody/Henry-Fauske

FIGURE 3.5-12

Oconee Peak Containment Pressure Analysis
Vessel Side Break - Comparison of Case 2A and Case 2B Mass Flux

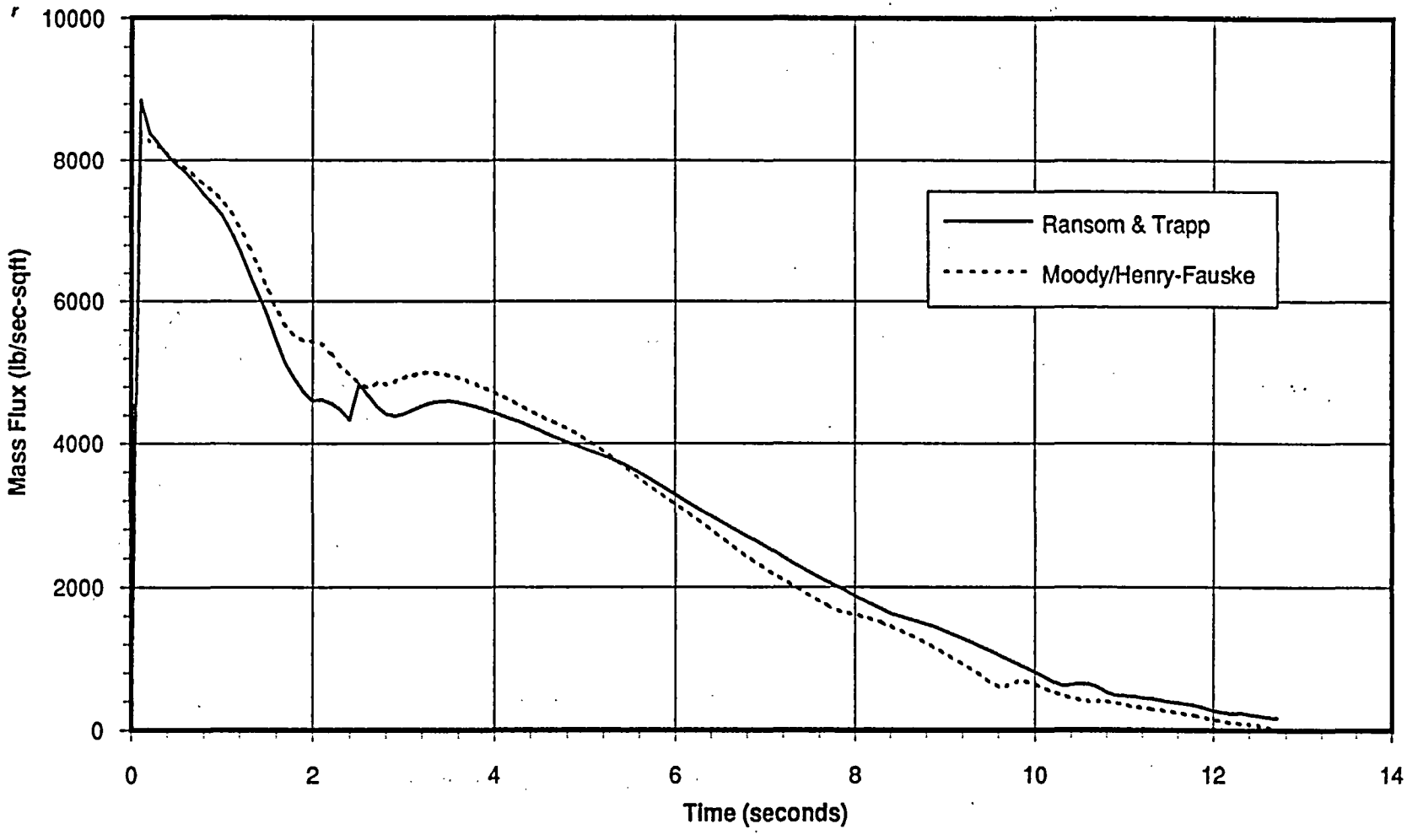


FIGURE 3.5-13

Oconee Peak Containment Pressure Analysis
SG Side Break - Comparison of Case 2C and Case 2D Mass Flux

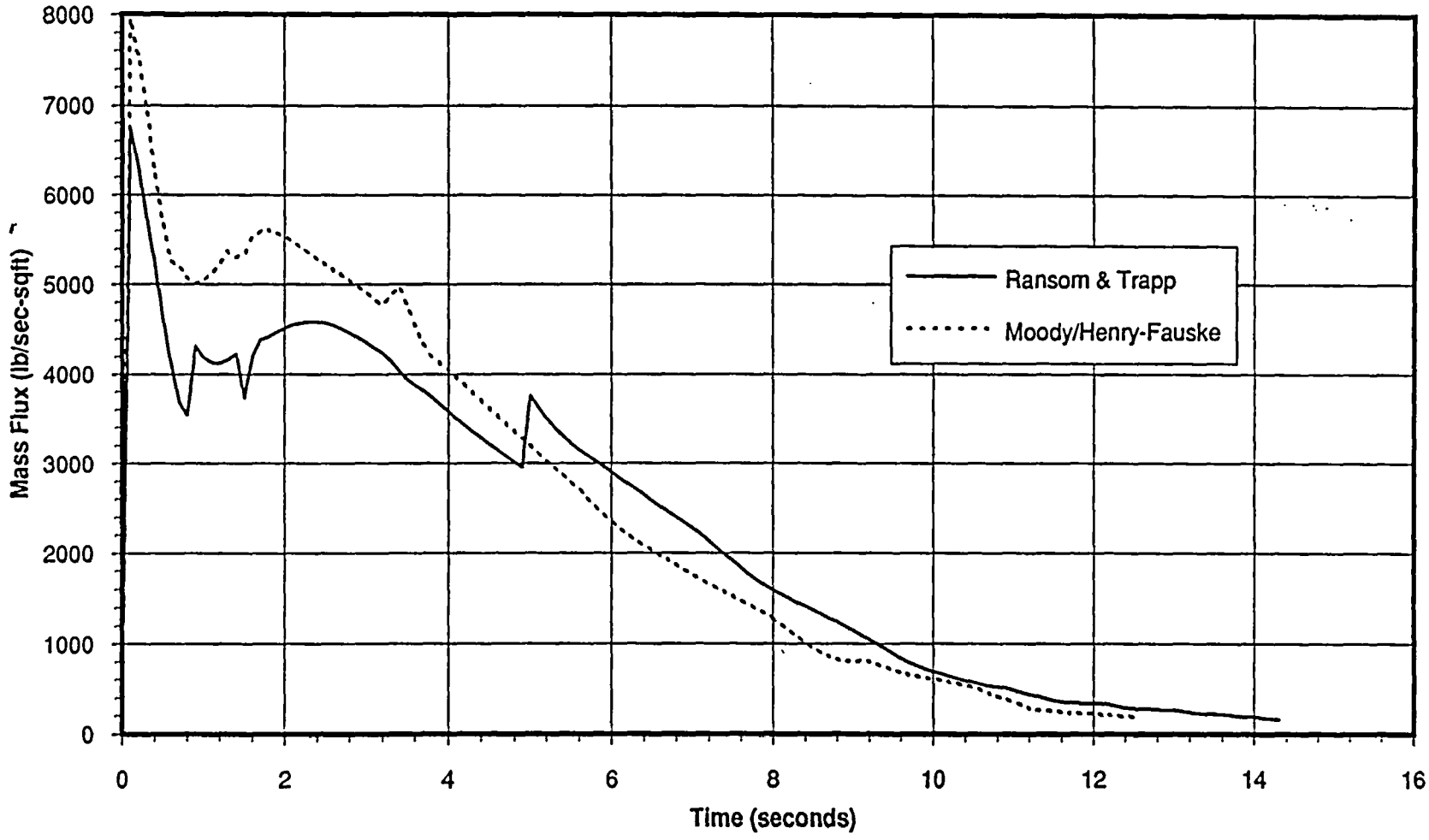


FIGURE 3.5-14

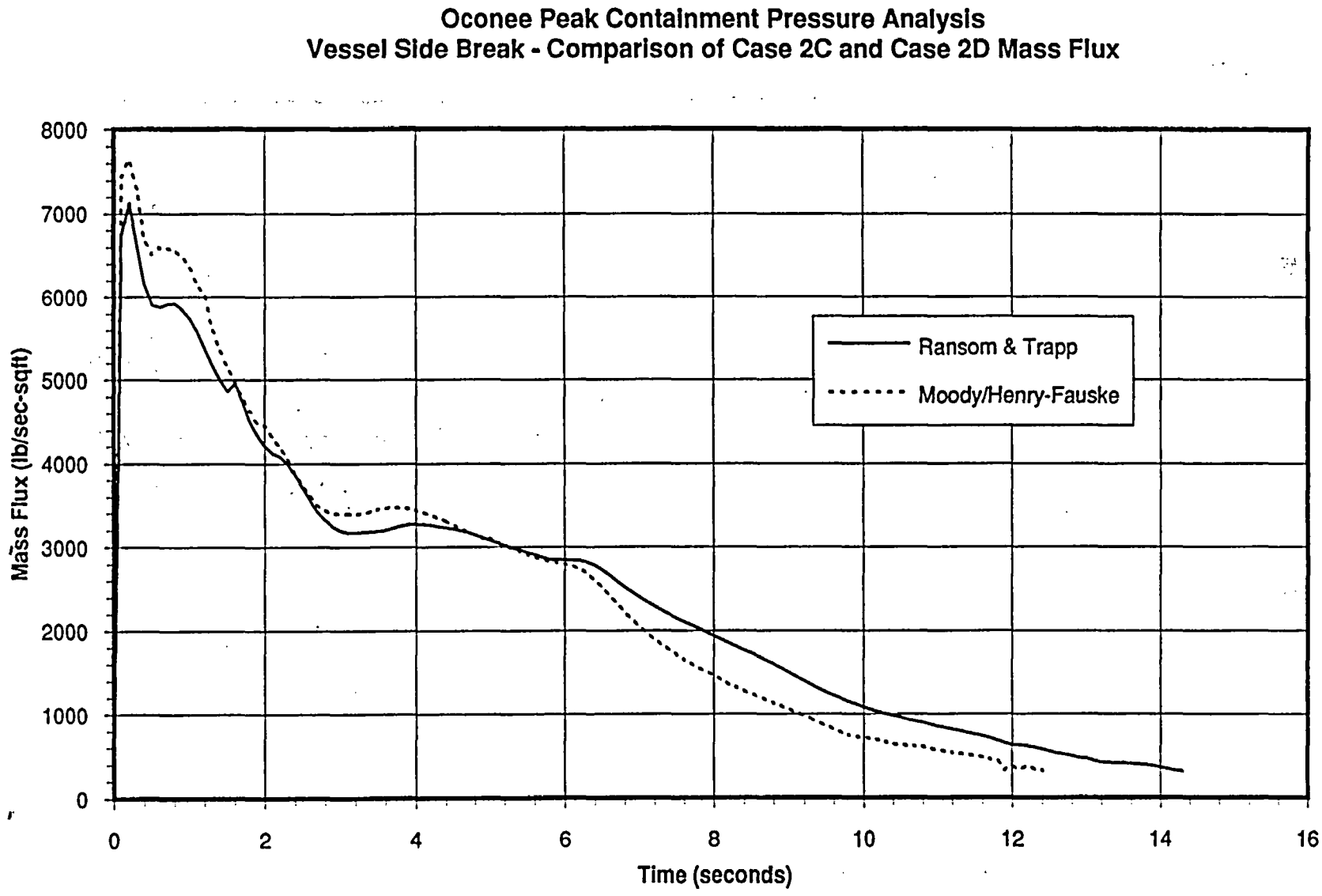

Oconee Peak Containment Pressure Analysis
Vessel Side Break - Comparison of Case 2C and Case 2D Mass Flux
Mass Flux (lb/sec-sqft)
8000
7000
6000
5000
4000
3000
2000
1000
0
0
2
4
6
8
10
12
14
16
Time (seconds)
Ransom & Trapp
Moody/Henry-Fauske

FIGURE 3.5-15

Oconee Peak Containment Pressure Analysis
Case 2B and 3A Energy Flow

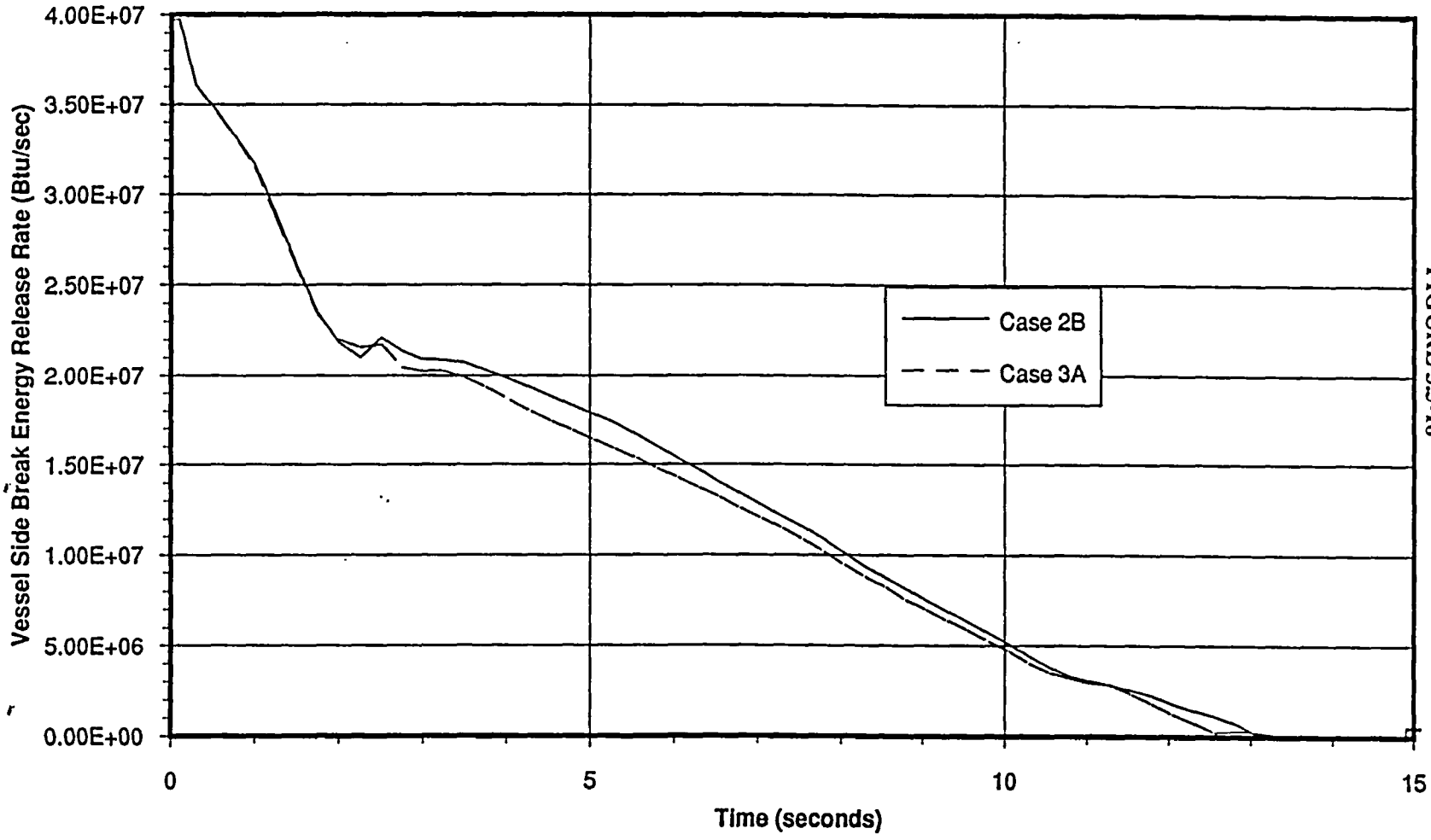


FIGURE 3.5-16

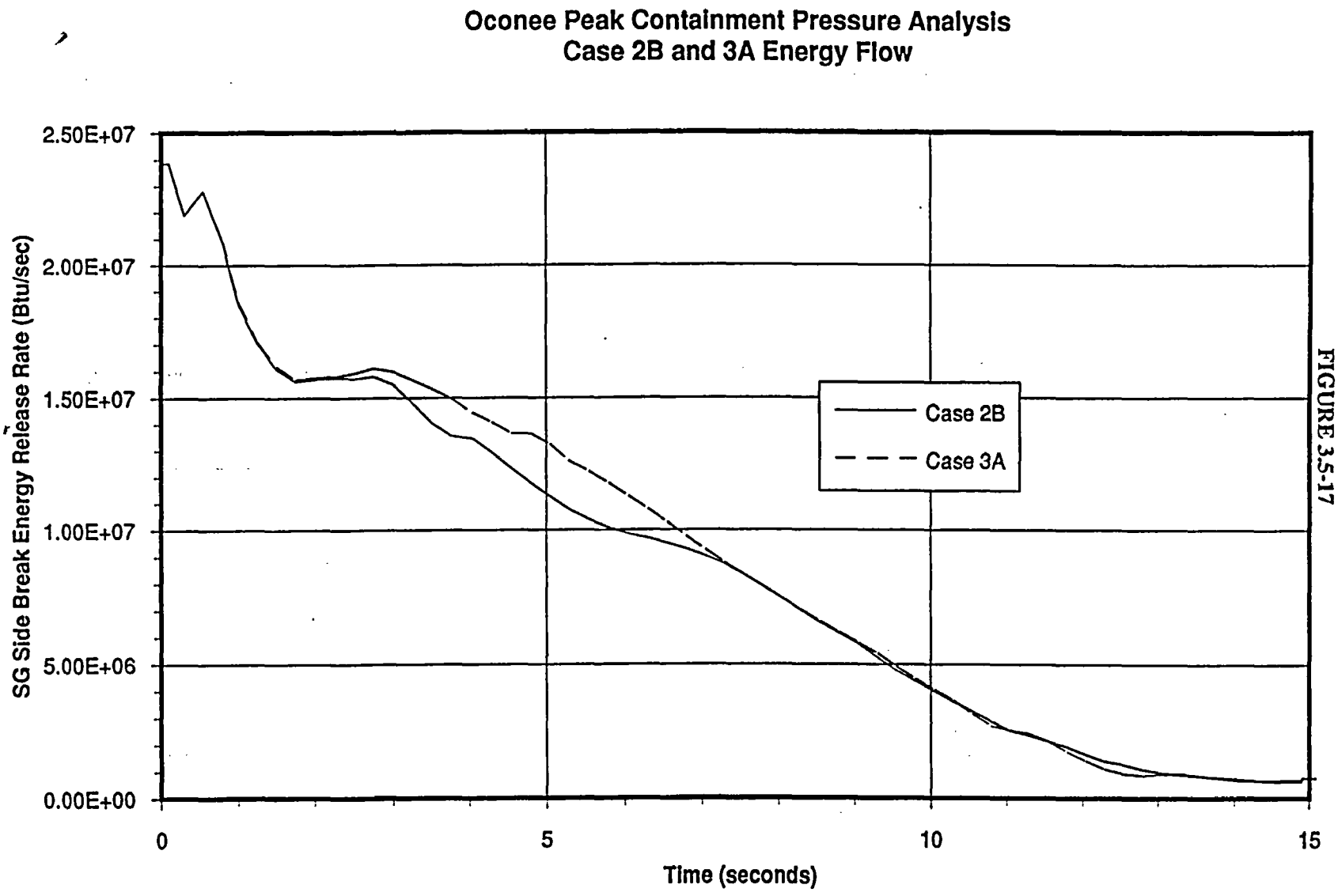

Oconee Peak Containment Pressure Analysis
Case 2B and 3A Energy Flow
SG Side Break Energy Release Rate (Btu/sec)
2.50E+07
2.00E+07
1.50E+07
1.00E+07
5.00E+06
0.00E+00
0
5
10
15
Time (seconds)
Case 2B
Case 3A

FIGURE 3.5-17

Oconee Peak Containment Pressure Analysis
Case 2B and 3A Energy Flow

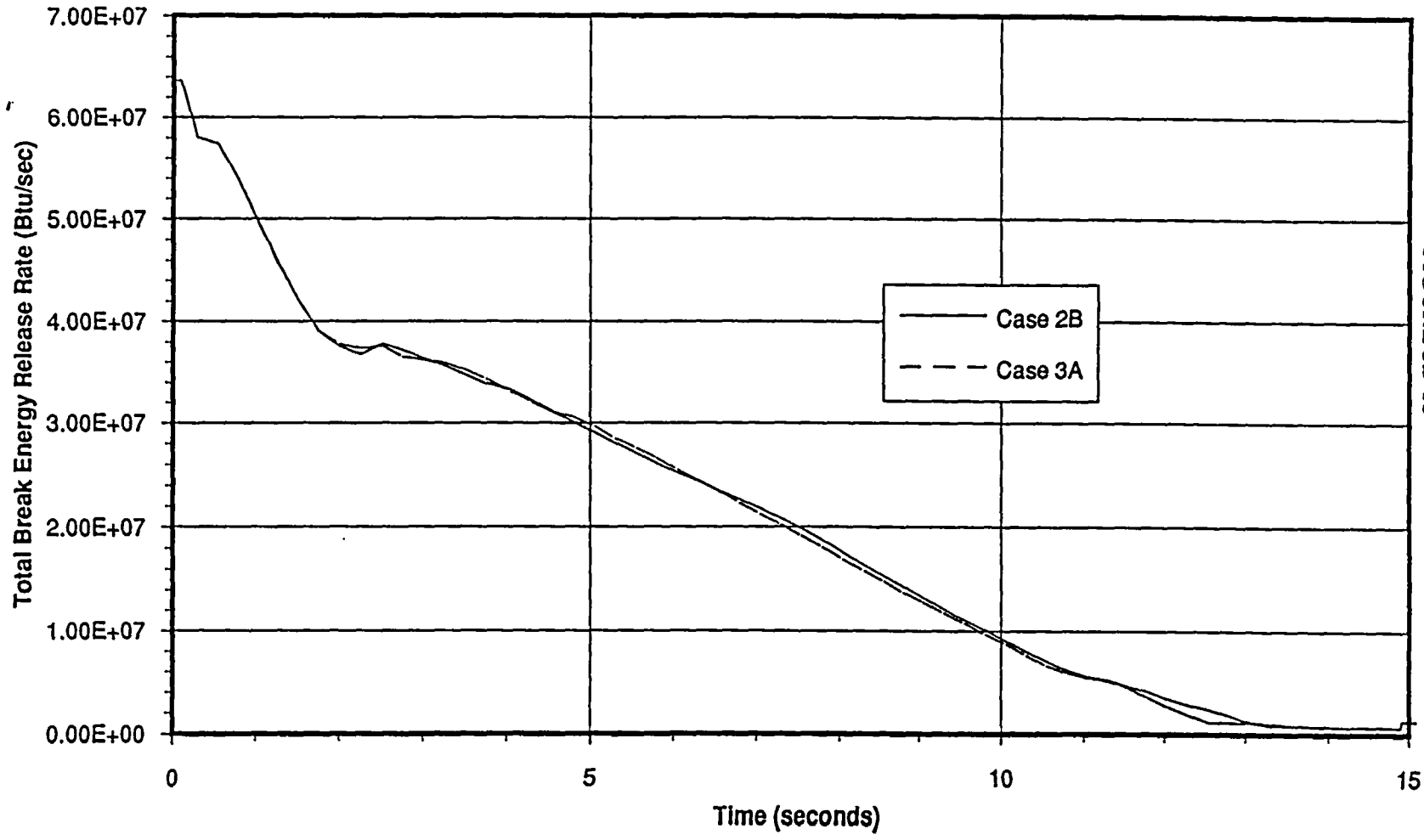


FIGURE 3.5-18

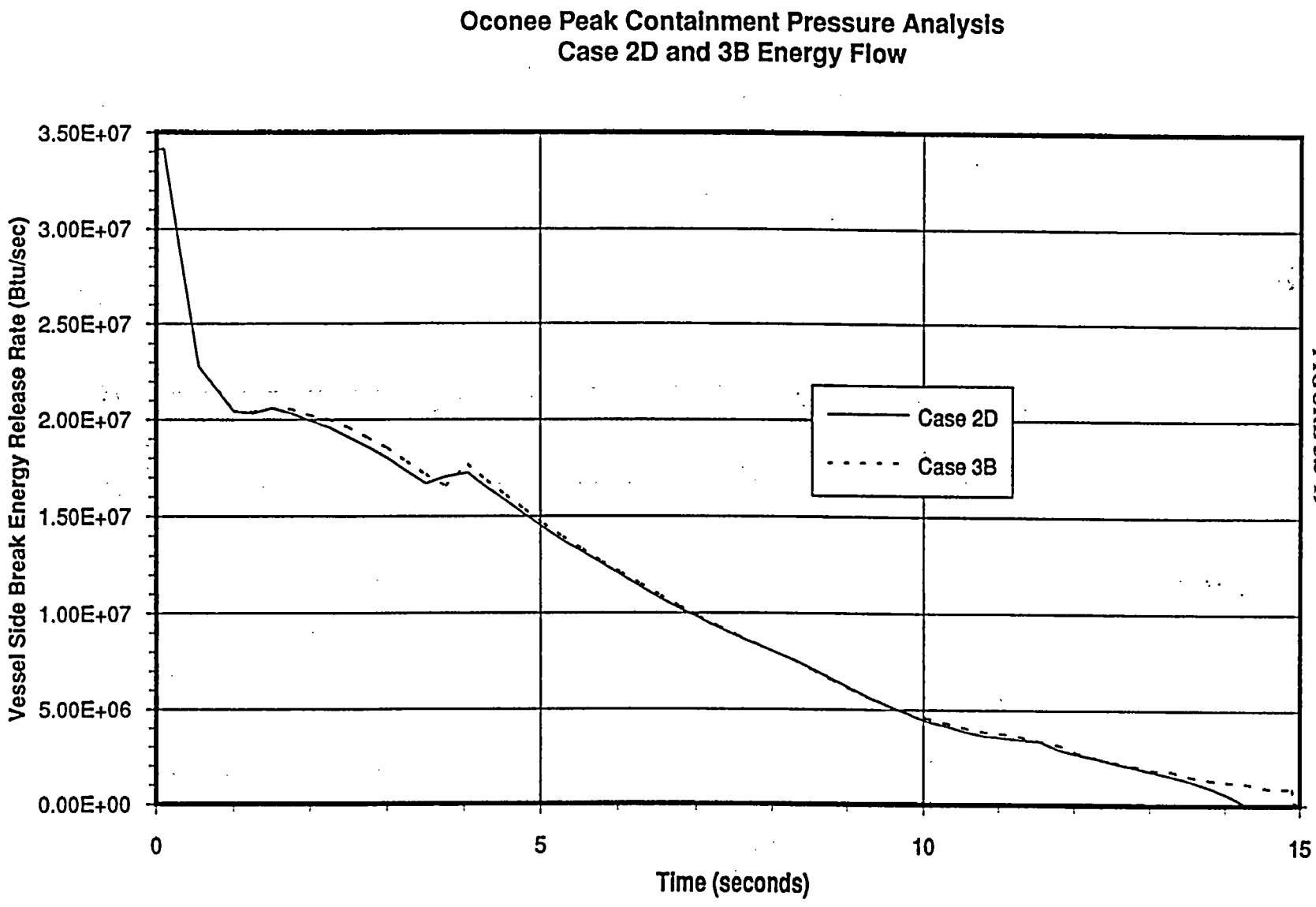

Oconee Peak Containment Pressure Analysis
Case 2D and 3B Energy Flow
Vessel Side Break Energy Release Rate (Btu/sec)
3.50E+07
3.00E+07
2.50E+07
2.00E+07
1.50E+07
1.00E+07
5.00E+06
0.00E+00
0
5
10
15
Time (seconds)
Case 2D
Case 3B

FIGURE 3.5-19

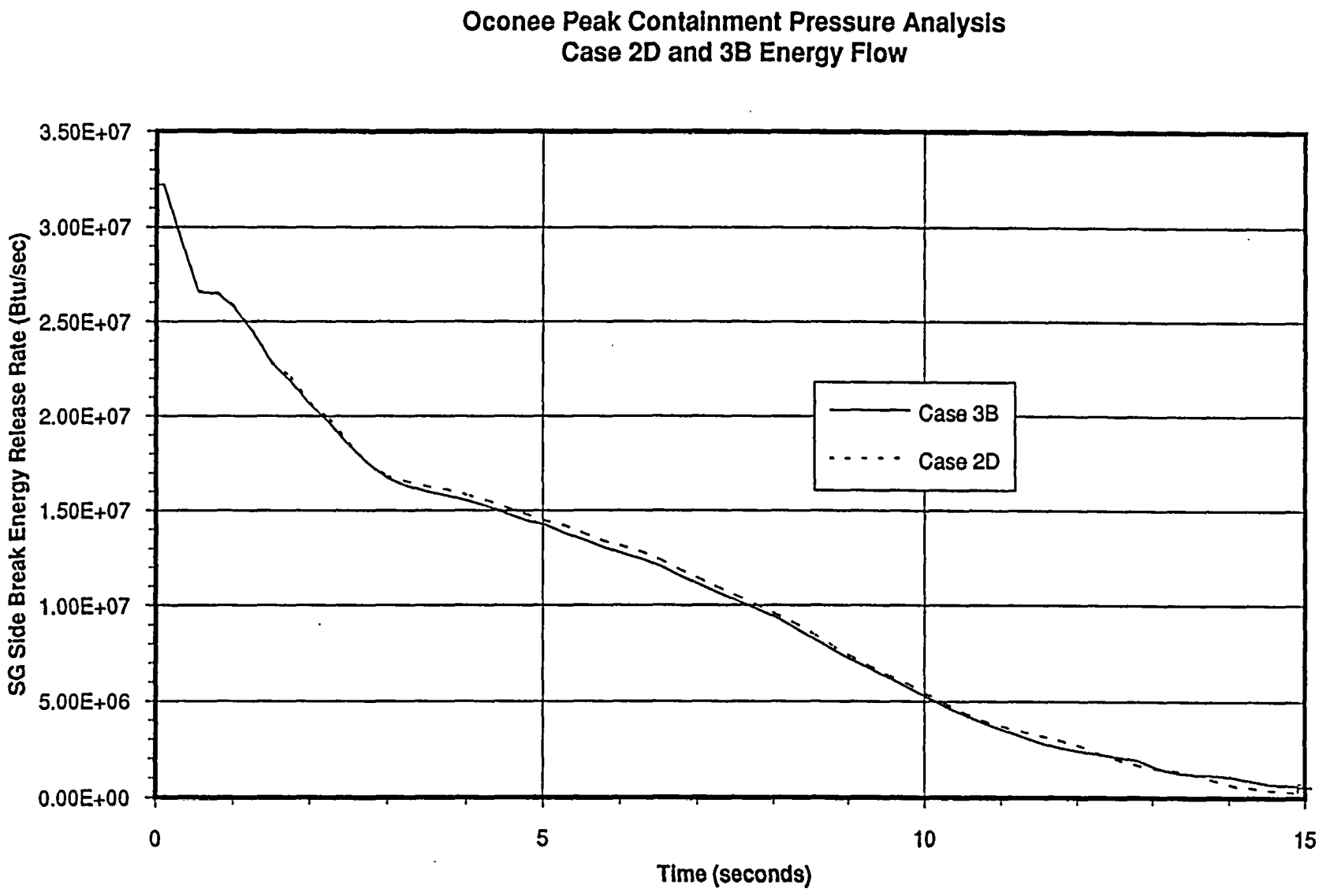

Oconee Peak Containment Pressure Analysis
Case 2D and 3B Energy Flow
SG Side Break Energy Release Rate (Btu/sec)
3.50E+07
3.00E+07
2.50E+07
2.00E+07
1.50E+07
1.00E+07
5.00E+06
0.00E+00
0
5
10
15
Time (seconds)
Case 3B
Case 2D

FIGURE 3.5-20

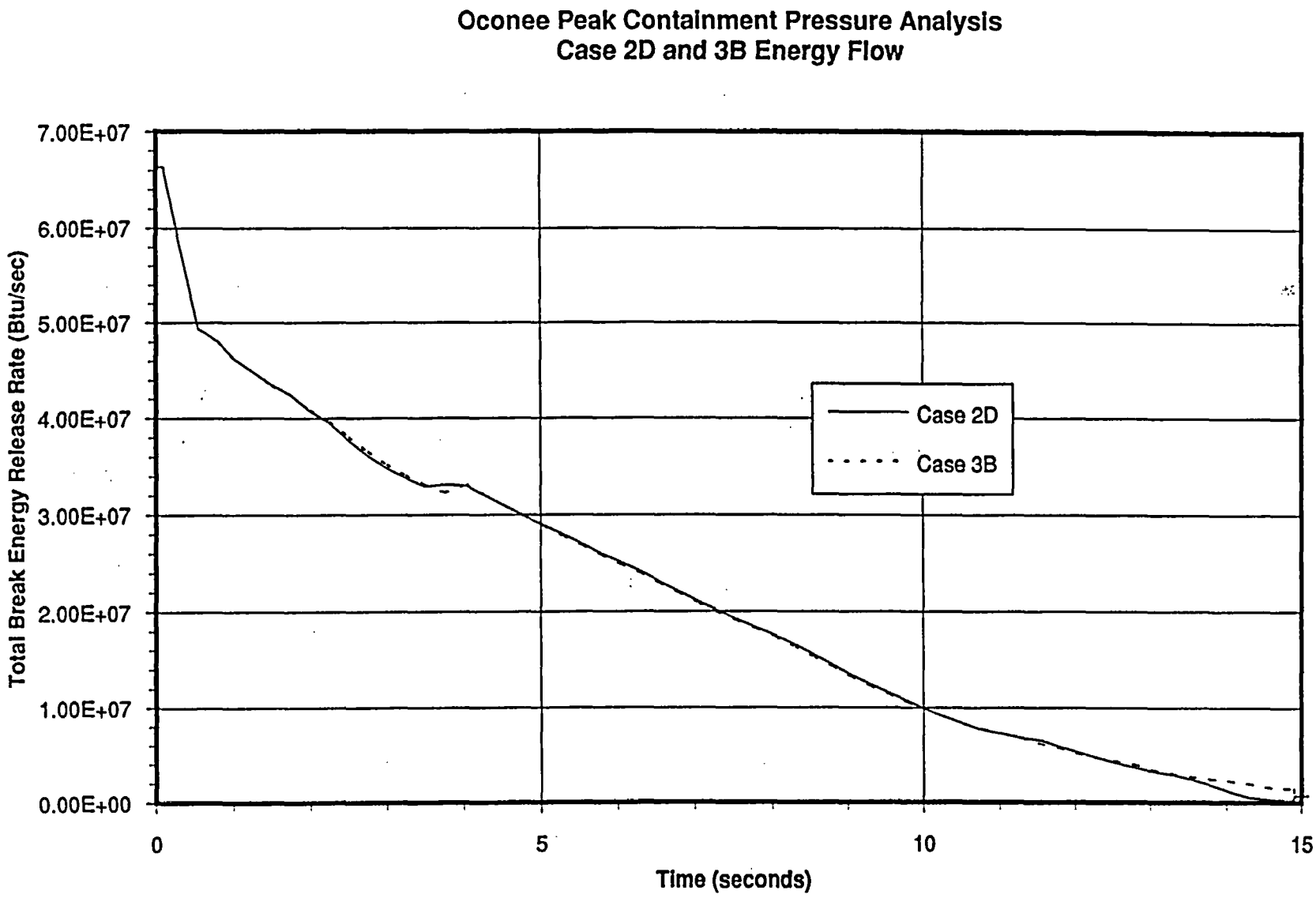

Oconee Peak Containment Pressure Analysis
Case 2D and 3B Energy Flow
Total Break Energy Release Rate (Btu/sec)
7.00E+07
6.00E+07
5.00E+07
4.00E+07
3.00E+07
2.00E+07
1.00E+07
0.00E+00
0
5
10
15
Time (seconds)
Case 2D
Case 3B

FIGURE 3.5-21

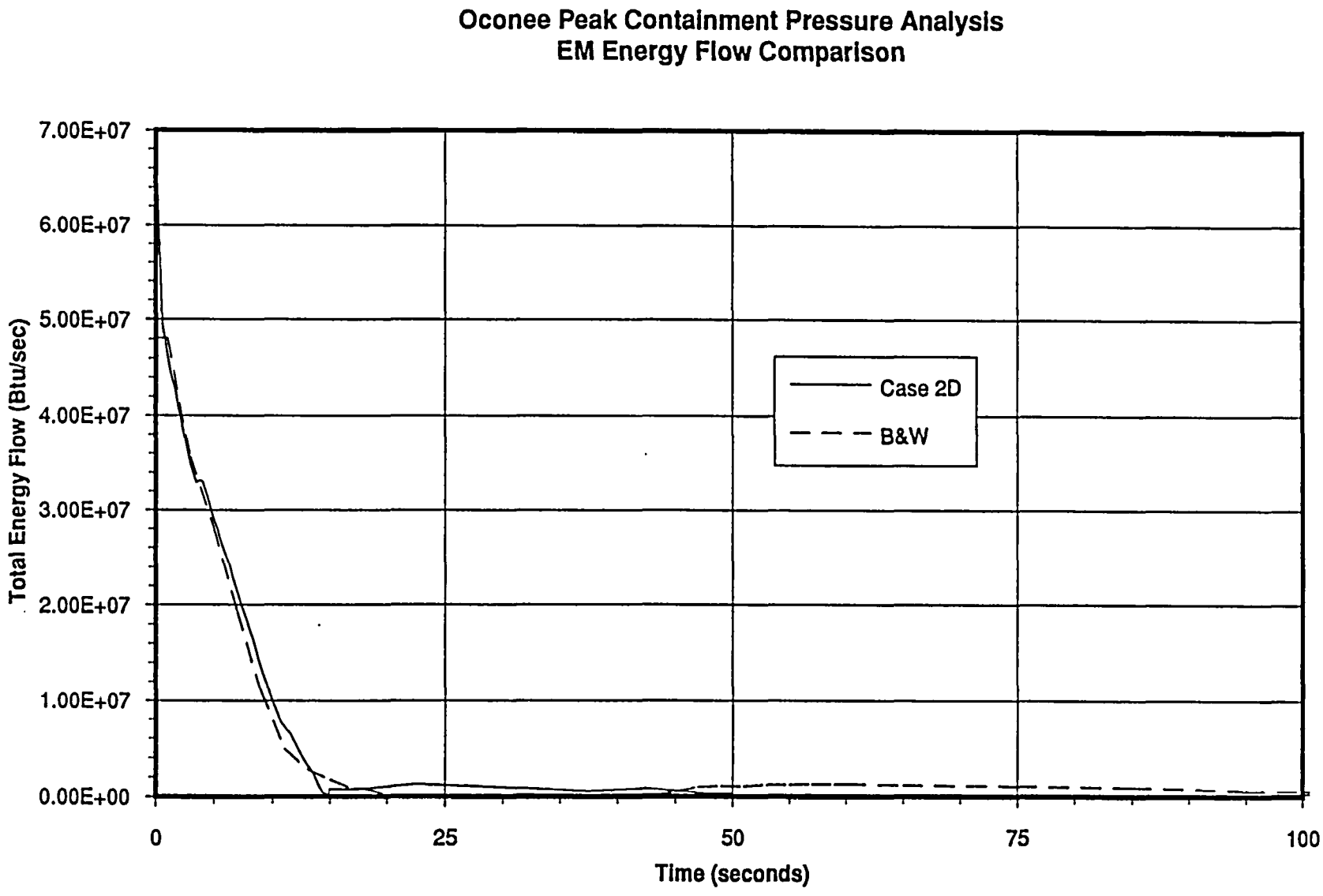

Oconee Peak Containment Pressure Analysis
EM Energy Flow Comparison
7.00E+07
6.00E+07
5.00E+07
4.00E+07
3.00E+07
2.00E+07
1.00E+07
0.00E+00
Total Energy Flow (Btu/sec)
0
25
50
75
100
Time (seconds)
Case 2D
B&W

FIGURE 3.5-22

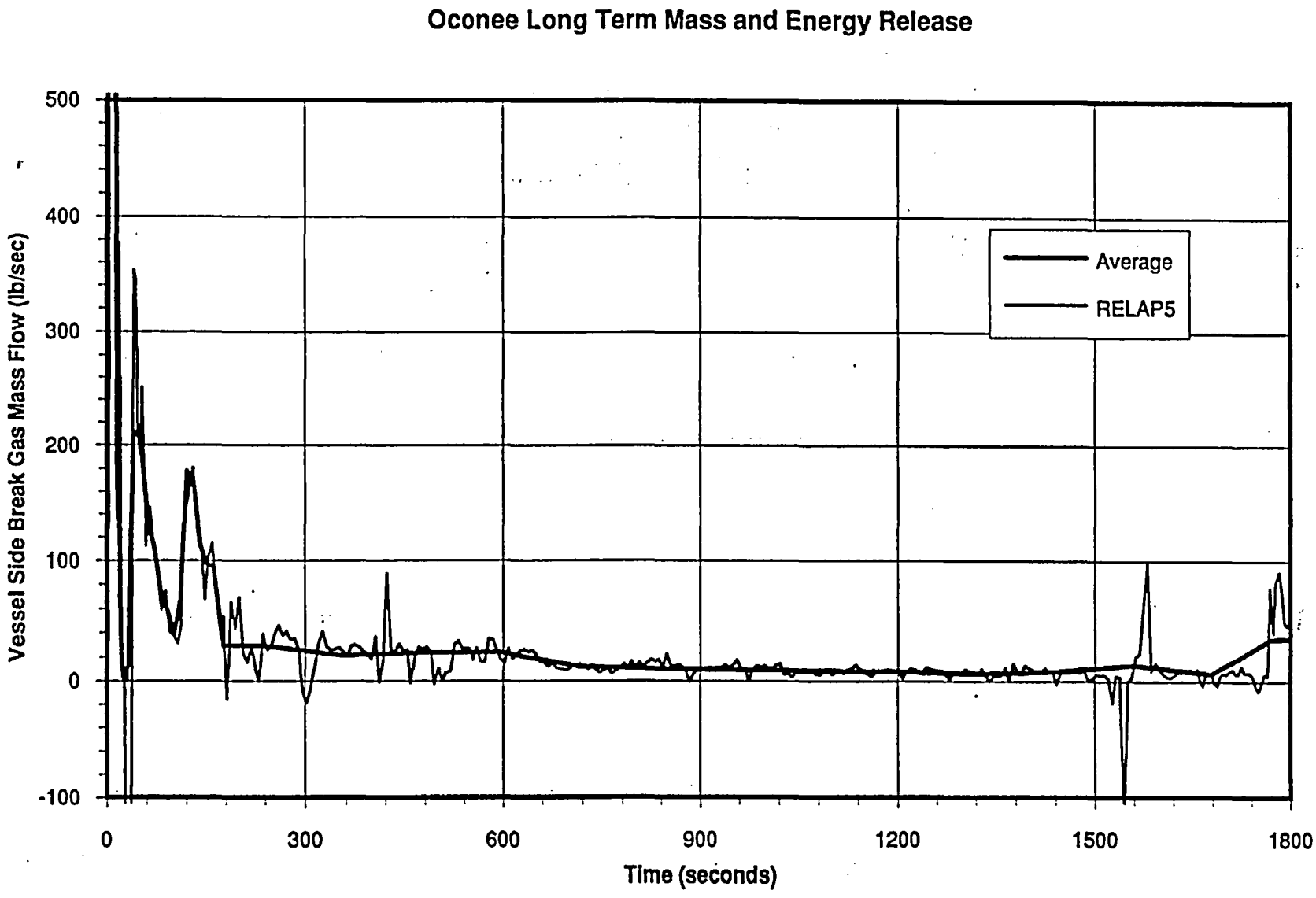

Oconee Long Term Mass and Energy Release
Vessel Side Break Gas Mass Flow (lb/sec)
500
400
300
200
100
0
-100
0
300
600
900
1200
1500
1800
Time (seconds)
Average
RELAP5

FIGURE 3.6-1

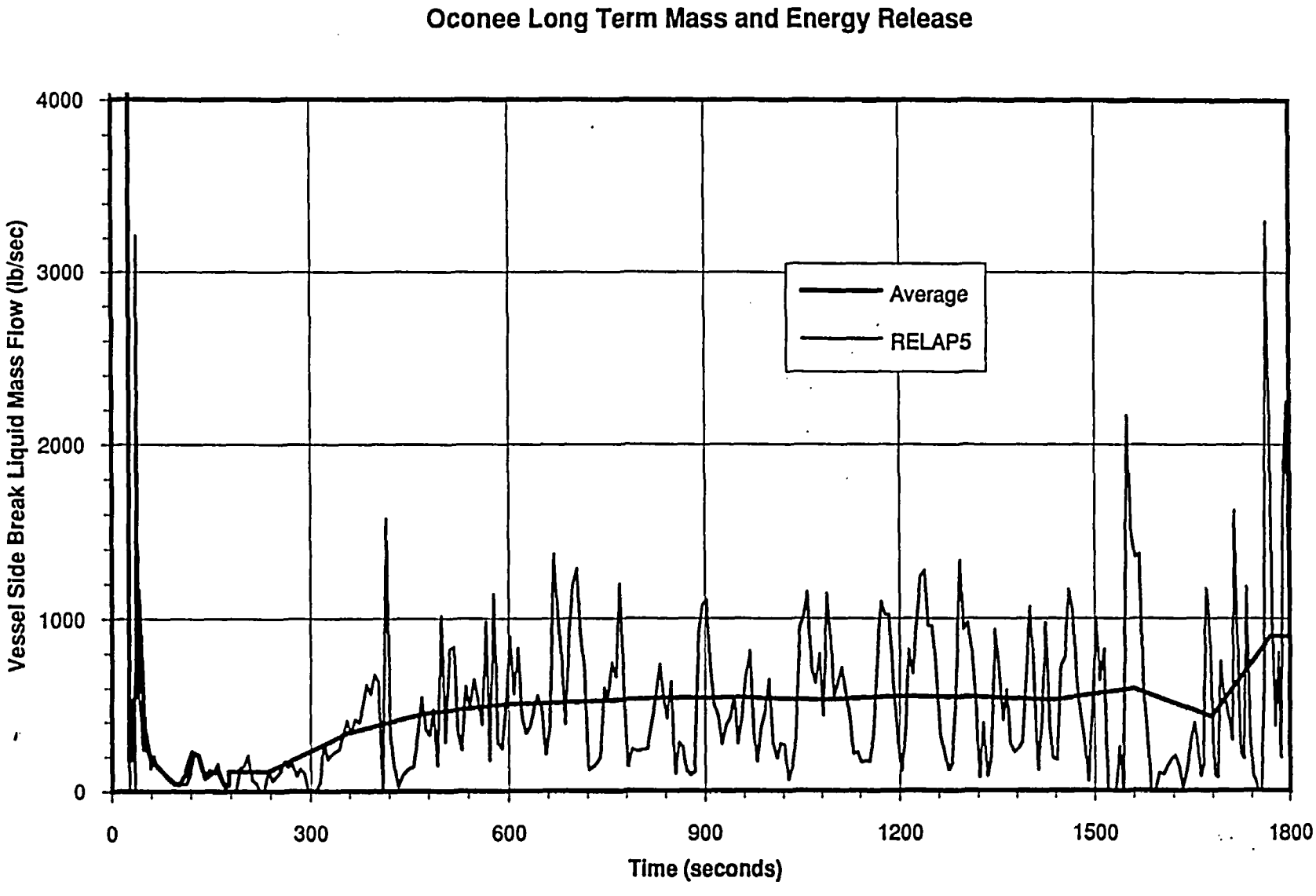

Oconee Long Term Mass and Energy Release
Vessel Side Break Liquid Mass Flow (lb/sec)
4000
3000
2000
1000
0
0
300
600
900
1200
1500
1800
Time (seconds)
Average
RELAP5

FIGURE 3.6-2

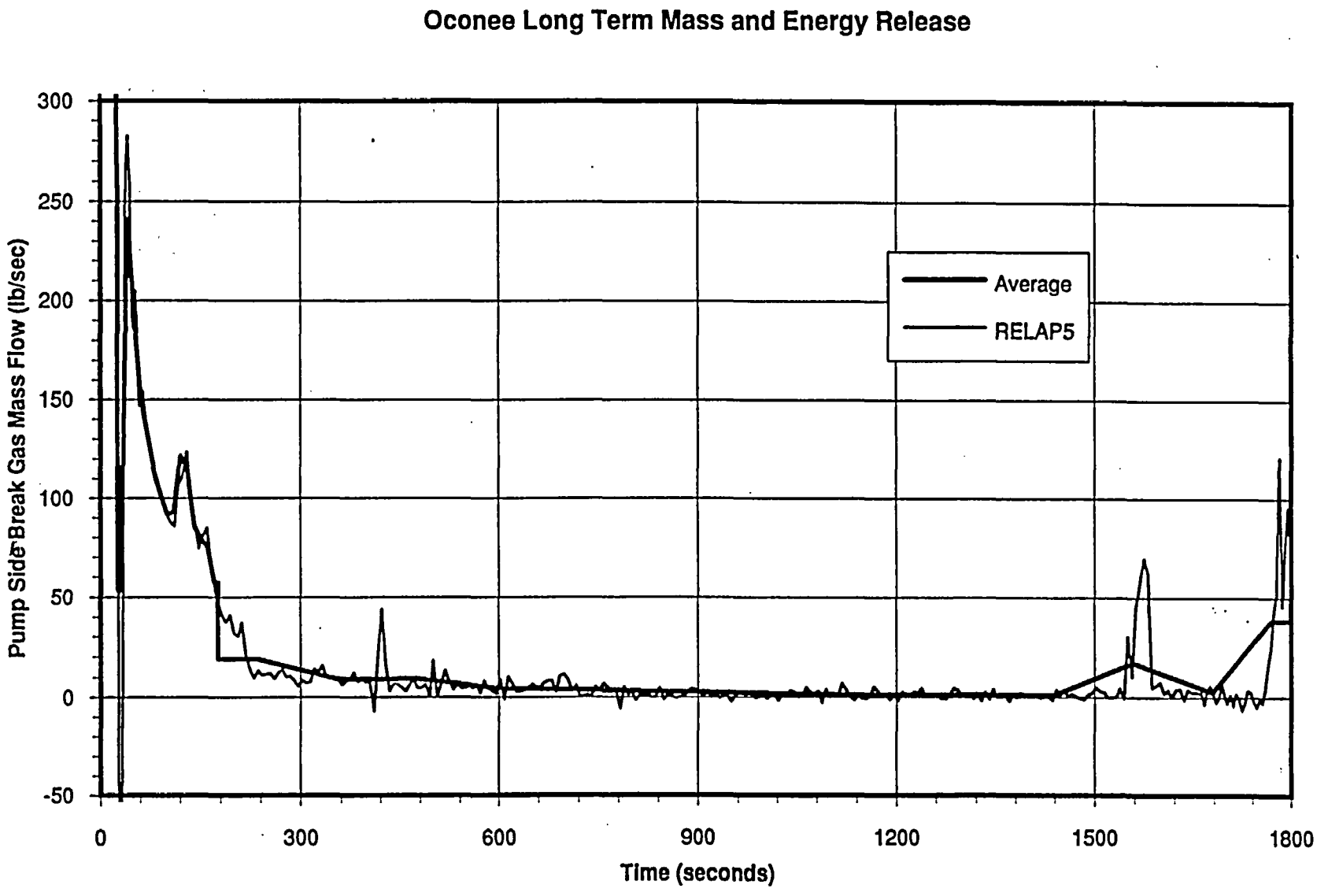

Oconee Long Term Mass and Energy Release
Pump Side Break Gas Mass Flow (lb/sec)
300
250
200
150
100
50
0
-50
0
300
600
900
1200
1500
1800
Time (seconds)
Average
RELAP5

FIGURE 3.6-3

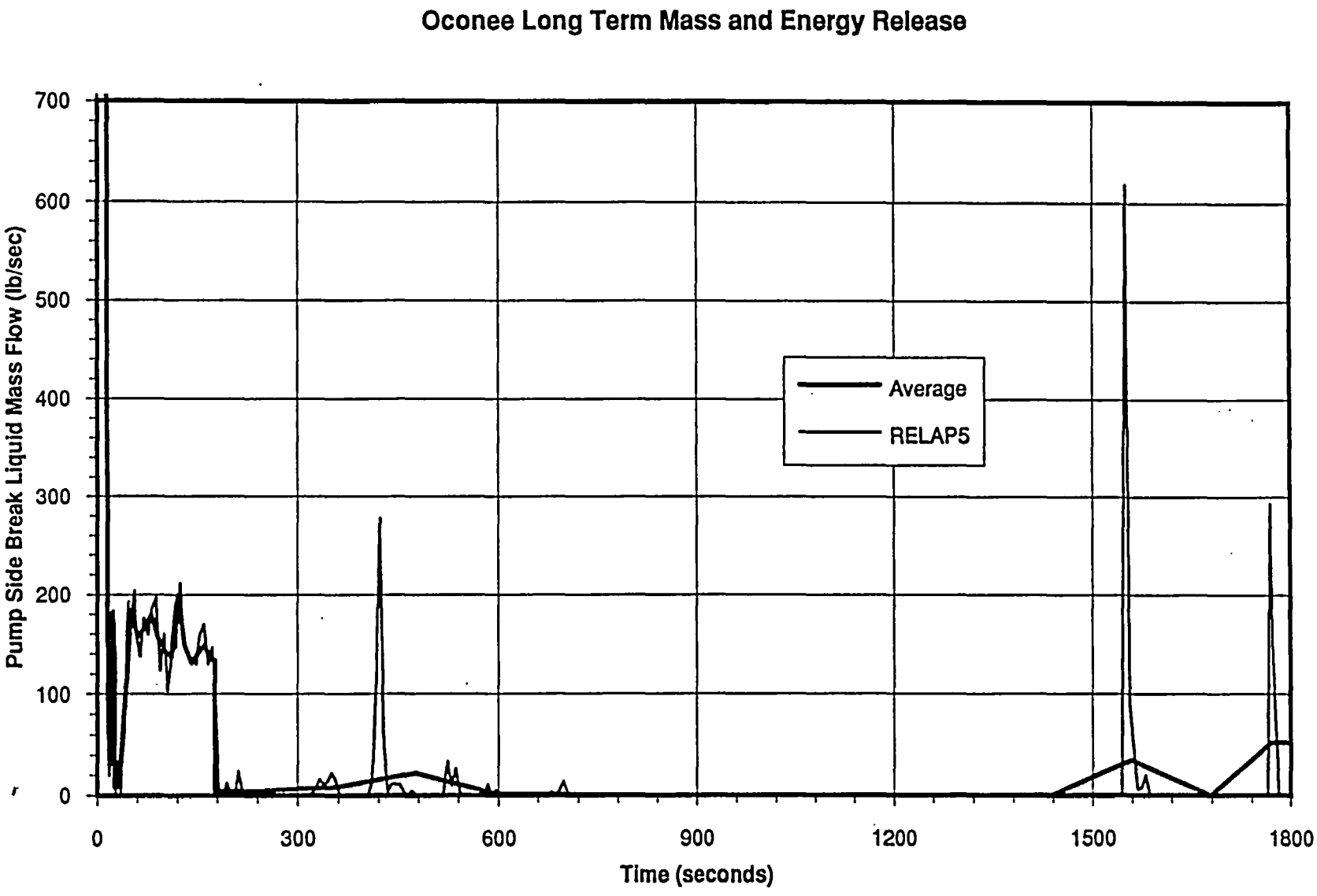

Oconee Long Term Mass and Energy Release
Pump Side Break Liquid Mass Flow (lb/sec)
700
600
500
400
300
200
100
0
0
300
600
900
1200
1500
1800
Time (seconds)
Average
RELAP5

FIGURE 3.6-4

4.0 SMALL BREAK LOCA MASS AND ENERGY RELEASE

4.1 Overview

Introduction

Small break LOCA analyses are performed to generate conservative mass and energy release boundary conditions for evaluating the long-term containment response. Small breaks are not analyzed with regard to peak containment pressure since it is intuitive that a large break LOCA releases energy much faster than a small break and will therefore yield a greater peak pressure. Small break LOCAs can, however, potentially result in the bounding long-term containment response. This is due mainly to the absence of LPI flow with respect to core cooling and rejecting heat from the Reactor Building.

The original Oconee licensing basis containment analyses considered a spectrum of break areas (0.5 - 14.1 ft^2) to evaluate peak pressure, but only one break area (5.0 ft^2 hot leg break) to evaluate the long-term containment response. Based upon the above discussion, it is apparent that a single analysis which assumes such a large break area may not produce the bounding long-term containment response. It is the intent of this analysis to determine the most limiting long-term mass and energy release whether it is the double-ended guillotine cold leg break discussed in Section 3.3.3, or one of the small breaks presented in this section.

Analysis Overview

The transient response of a small break LOCA is much less defined than that of a large break. During a large break, the primary system will depressurize rapidly towards containment pressure without operator action or without relying on steam generator heat transfer. This scenario is not necessary valid for a small break LOCA, particularly the breaks with smaller flow areas. Operator action may be necessary to cool and depressurize the RCS by utilizing steam generator heat transfer. This is because break cooling alone may not be sufficient to remove the decay heat and stored energy. As a result, when simulating small break LOCAs, the analyst must select a set of assumptions which will yield conservative results while allowing for reasonable operator actions. All analysis boundary conditions described in this section are selected to ensure that the mass and energy release is maximized while allowing for reasonable operator actions.

A spectrum of small break analyses is performed to identify the most limiting long-term mass and energy release. This spectrum ranges from break areas which are too small to interrupt primary to secondary heat transfer, to break areas which do not require steam generator heat transfer to cool to LPI conditions. In all, six cases have been analyzed. These cases analyze breaks with flow areas which range from 0.0025 ft^2 to 0.10 ft^2. All initial conditions and boundary conditions are the same for each case, only the break area differs between cases.

The small break LOCA simulations are executed much longer than the large break analyses. Consequently, containment boundary conditions will become more significant and must be considered. Containment pressure and sump water temperature are input to RELAP5 as boundary conditions based on FATHOMS analysis results. Pressure in containment as a function of time is desired to estimate when the RBCUs initiate and when the RBS System is activated. Sump fluid temperature as a function of time is desired to calculate the temperature of the ECCS injection fluid after sump recirculation is aligned.

An averaging process, which is identical to the methods developed for the large break analyses, is used to convert the RELAP5 mass and energy release data into a form suitable for the FATHOMS code. The mass and energy exiting the break junction is integrated during each simulation. These integral totals include the thermal energy of the flow stream as well as the kinetic energy imparted to the fluid as it exits the break. The vapor and liquid phases are tracked separately to ensure that the energy is distributed correctly between the containment atmosphere and the fluid in the emergency sump.

Long-Term Mass and Energy Release

4.2 Initial Conditions

The initial conditions assumed for all small break LOCA analyses are identical to the values described in Section 3.2. Since these values are conservative regardless of break size, they are chosen to maintain consistency with the large break LOCA analyses.

4.3 Boundary Conditions

4.3.1 Energy Sources

All of the common energy sources described in Section 3.3.1.1 are assumed for the small break analyses. The metal-water reaction is not modeled since cladding temperatures do not increase sufficiently during a small break to substantiate this reaction. Fission heat is calculated with a method similar to the large break peak pressure analysis, which is described in Section 3.3.2.1. This section describes the methods used to calculate fission power for all small break analyses.

Fission Heat

The RELAP5 point kinetics model is used to calculate delayed neutron power as a function of time. Upon reactor trip a negative reactivity insertion equal to the technical specification minimum shutdown margin, currently 1.0% ?k/k, using a normalized rod worth vs. position curve is modeled. The reactor is assumed to trip on the first RPS setpoint encountered. This setpoint is generally the variable low RCS pressure trip which is modeled with the equation given in the technical specifications. Instrument uncertainties are also applied to this equation. The rod insertion rate is based upon Technical Specification values. For conservatism, a delay is

added so that the rods begin to fall 0.7 seconds after the trip signal is received. No thermal feedback is modeled. This approach is consistent with the core reload design process which ensures that the minimum shutdown margin is maintained post-trip for LOCA-type events. This is a conservative model since greater than the minimum shutdown margin always exists."

Total Rod Worth

The total rod worth is set to the technical specification minimum shutdown margin, currently 1.0% ?k/k, converted to dollar units by dividing by the value for beta effective.

Fission Product and Actinides Decay

The heat produced from the radioactive decay of fission products and actinides is calculated for end-of-cycle conditions. This calculation determines decay heat as a function of time after trip for Oconee Units 1, 2, and 3 using the ANSI/ANS-5.1-1979 standard (Reference 3-5). For additional conservatism, uncertainties (2s) are calculated and added to the mean decay heat values.

4.3.2 Assumptions

For consistency, the boundary conditions and assumptions of the small break analyses are similar to those described in Section 3.3.1.2 for the large break analyses. However, because the assumptions used to analyze small break LOCAs are much different than a large break LOCA, many of the large break assumptions are not valid. The assumptions which differ are discussed in this section.

Break Location

A spectrum of small break analyses is performed to identify the most limiting long-term mass an energy release. This spectrum covers break areas which are too small to interrupt primary to secondary heat transfer due to hot leg U-bend voiding, to break areas which do not require steam generator heat transfer to cool to LPI conditions. Six cases have been analyzed. These cases are listed below along with the corresponding break area.

Case	Break Area (ft^2)
1	0.0025
2	0.005
3	0.01
4	0.025
5	0.05
6	0.10

Each case assumes a split rupture in the A1 cold leg pump discharge piping and uses the same set of initial conditions and boundary conditions. A cold break site is chosen over a hot leg break site because 1) a fraction of the total available HPI flow is assumed to be spilled directly out the break and into containment (less HPI fluid available for core cooling) and 2) the core is not vented directly to the break so that steam generated in the core must pass through the vent valves before exiting the break. These considerations tend to make cold leg LOCAs worse with respect to core cooling than hot leg LOCAs. The break is assumed to be oriented in an upward direction and the RELAP5 horizontal stratification model is used to conservatively maximize the steam release through the break.

Break Flow Model

Break flow is calculated by the standard RELAP5 Ransom and Trapp critical flow model. A subcooled and saturated discharge coefficient of 1.0 is assumed. Reference 3-1 suggests using a conservative correlation such as Moody to calculate the two-phase critical mass flux. This guidance is much less significant for small break analyses, particularly if a spectrum of break areas is analyzed, since decreasing the break area will have the same effect as increasing the mass flow. If the Moody correlation were used in place of the Ransom and Trapp model, the calculated mass flow rate would be greater than the mass flow calculated by RELAP5. However, because a spectrum of break sizes is examined in these analyses, greater mass flow rates are inherently considered due to variations in break flow area. Therefore, there is no need to use the Moody correlation for the small break analyses.

Containment Boundary Conditions

Results from FATHOMS analyses are used as inputs to the RELAP5 analyses to model two containment boundary conditions. One boundary condition is the containment backpressure that

the LOCA break flow discharges into. The second boundary condition is the containment sump water temperature used for the ECCS suction supply after the transition to the sump recirculation mode. These are selected based on FATHOMS cases that are consistent with the RELAP5 analysis in terms of break size and intended application.

BWST Depletion - ECCS and Containment System Flow Rates

The BWST is the common suction source for the HPI, LPI, and RBS Systems. These systems will deplete the tank until a predetermined level is reached and sump recirculation is aligned. In order to calculate when sump recirculation is aligned, the depletion rate from all of the ECCS systems must be integrated. The boundary conditions assumed for the HPI System determine when this system is activated and how much flow the system will deliver as a function of RCS back-pressure. Similarly, boundary conditions for the LPI System dictates flow as a function of RCS back-pressure. The RBS System flow rate is assumed, but the time at which this system initiates is a function of containment pressure. Containment analysis inputs required for the RELAP5 analysis are obtained from FATHOMS analysis results. This section presents the boundary conditions associated with all of the ECCS systems which deplete the BWST.

The HPI System is assumed to initiate upon receiving an ES actuation signal. Technical Specifications require the HPI System to actuate when RCS pressure is $\leq$1500 psig although the actual setpoint is somewhat higher (1600 psig). The 1500 psig setpoint is used in this analysis. A 15 second ECCS delay time is assumed consistent with the assumptions described in Section 3.3.2.2.

The HPI flow boundary condition chosen for the analyses minimizes HPI flow injected into the RCS as a function of time. For the first ten minutes following ES actuation, flow from one HPI pump is assumed to inject into the RCS through only one cold leg injection line. After ten minutes, the operator is assumed to cross-connect HPI trains to allow flow from two HPI pumps to inject into the RCS through three cold leg injection lines. This boundary condition inherently assumes that a) a single failure of a 4160V switchgear has occurred, and b) a break on or including the HPI line has occurred. A discussion of these assumptions is presented below.

As mentioned previously, the assumed single failure for all LOCA analyses is the failure of a 4160V switchgear. The train A HPI pump and flow control valve receive power from the same 4160V switchgear. If this switchgear fails, there will be no flow delivered through train A of the HPI System. The only flow available will be from pump C through train B. In response to this

condition, the operator must cross-connect HPI trains to allow flow to be delivered to both trains by two pumps. The time required for the operator to perform this action is reflected in the assumed ten minute delay. This assumption is supported by the Emergency Operating Procedure which currently requires this action to be completed within ten minutes of ES actuation, and is used in the licensing basis small break LOCA analyses.

Severing one of the HPI injection lines causes the severed line to discharge freely to containment while the other flow paths discharge to the RCS. Because containment pressure is usually much lower than the RCS pressure, there is a disproportionate amount of flow delivered to the broken line. Consequently, a HPI line break assumption results in a more rapid depletion of the BWST and minimizes flow delivered to the RCS.

For the first ten minutes of each simulation, flow will be delivered to the RCS through only one injection line and spilled directly to containment in the other line. Once the HPI trains have been cross-connected, three of the four injection lines will supply flow to the RCS while one line will spill flow to containment. The spilled HPI flow is modeled in the RELAP5 analysis and is used in the FATHOMS analysis as a flow boundary condition. This boundary condition is necessary to generate the correct sump mass and temperature profile.

The LPI System can deliver flow to the RCS only when pressure remains below the LPI pump discharge head, which corresponds to approximately 180 psig. In most small break cases, sustained LPI flow cannot be maintained. However, for the larger small breaks, a LPI System boundary condition is necessary.

The LPI flow versus pressure curve used in the analyses is a composite of the LPI pump head curve, and the LPI flow assumed in design basis analyses. The head/flow curves from all LPI pumps were surveyed to find the curve with the lowest flow. Pump head is converted to pressure and the pressure is reduced by 10% to account for line losses and potential pump degradation. The LPI flow assumed in the design basis analysis (Reference 4-1) is also examined. When the flow rates from these two sources are compared, the LPI pump head is bounding at low flow rates while the design basis flow is bounding at higher flow rates (Figure 4.3.2). Therefore, the LPI flow assumed in this analysis chooses the most bounding curve which is a composite of the pump head and the design basis flows.

A cross-tie modification has been installed that connects the two LPI trains at the pump discharge. This modification deletes the requirement for manual throttling of LPI flow as

described in the following paragraph and throughout this chapter. Conservative LPI flowrates are used in the methodology consistent with this design change. The original text is maintained so that the analysis results presented can be understood.

The operator is assumed to follow EOP guidance which instructs him to throttle LPI flow to 3000 gpm. When potential instrument errors are subtracted (311 gpm), the maximum injected flow is 2689 gpm. The maximum flow is limited to this value.

The RBS System is assumed to initiate consistent with the FATHOMS results. A 92 second delay is assumed once the setpoint is reached. Each of these assumptions is consistent with the assumed FATHOMS boundary conditions. Once initiated, RBS flow is injected from the BWST or the emergency sump, depending upon whether sump recirculation is aligned, into the RELAP5 upper containment volume. The assumed RBS System flow rate corresponds to a constant 1500 gpm minus instrument errors or 1357 gpm.

Reactor Coolant Pumps

Because a single failure of a 4160V switchgear yields the most limiting single failure, off-site power will be available throughout the transient. Electrical power will therefore be available to the RCPs. The EOP currently instructs the operator to trip all RCPs whenever subcooling is lost. This action is usually completed within two minutes of losing subcooling. However, for conservatism, the RCPs are assumed to trip immediately upon reactor trip. This assumption minimizes the primary system energy transferred to the steam generators during the initial phase of the transient.

Steam Generator Level

The MFW System is used to initially control steam generator level since off-site power is assumed to be available, and because MFW is hotter than EFW. Once the RCPs have tripped, the MFW System controls level at the natural circulation setpoint of 50% OR. An instrument error of 10.5% is subtracted from this setpoint to yield a setpoint of 39.5%. Instrument error is subtracted from the nominal setpoint to minimize the pool heat transfer area available for primary-to-secondary heat transfer during natural circulation.

The EOP requires the operator to raise steam generator levels to the loss of subcooled margin (LSCM) setpoint (which is higher than the 50% OR setpoint) using the EFW System whenever

subcooling is lost. This action is usually performed immediately. However, a 20 minute delay from reactor trip is assumed in the methodology. Furthermore, only one motor-driven EFW pump is credited to feed one steam generator to the LSCM setpoint minus an allowance for level instrument uncertainty.

Steam Generator Pressure Control

The turbine stop valves are assumed to close instantaneously at reactor trip and the turbine bypass valves are assumed to be unavailable. The main steam safety relief valves are assumed to operate at their nominal setpoint plus uncertainties to account for relief valve drift. Thus, the only means available for the operator to depressurize the steam generators are the atmospheric dump valves (ADVs). These are manually operated valves which cannot be manipulated from the control room. An operator is assumed to begin manipulating these valves locally beginning 60 minutes after break initiation in an attempt to maintain a cooldown rate >50°F/hr, based upon the core exit temperature. These valves have more flow capacity than is required to achieve the target cooldown rate.

Emergency Feedwater System

The single failure of a 4160V switchgear will disable one motor-driven emergency feedwater (MDEFW) pump, leaving one MDEFW pump and the turbine-driven emergency feedwater (TDEFW) pump available. No credit is taken for the TDEFW pump, leaving one MDEFW pump feeding one steam generator. A RELAP5 control system is used to control the actual EFW flow based upon the desired steam generator setpoint. The B&W EFW spreading model is used to model the phenomena of EFW spraying through the upper header onto the tubes in the outer periphery of the steam generators (see Section 2.1.1.2). Since MFW is conservatively used to feed to the natural circulation level setpoint, EFW is only used to raise the steam generator levels to the loss of subcooled margin setpoint. This is assumed to begin at 20 minutes based on manual operator action consistent with procedures. This is also consistent with the UFSAR Chapter 15 SBLOCA peak cladding temperature analysis.

Hot Leg High Point Vent Valves

The hot leg high point vent valves are designed to vent non-condensible gases which may accumulate in the hot leg U-bend region. The same vents are also used in small break LOCA mitigation to help restore heat transfer and provide for additional RCS relief capacity. These

small capacity vent paths simply connect the top of the hot leg U-bend region to the containment atmosphere.

The EOP includes guidance which instructs the operator to open the hot leg high point vents if primary-to-secondary heat transfer has been interrupted and the reactor coolant pumps are unavailable to perform pump bumps. These prerequisites are satisfied during many of the small break analyses but the high point vents are usually not assumed to be opened. The high point vents are, however, assumed to be opened two hours into the 0.005 ft^2 analysis since all of the conditions required by the EOP exist at this time.

4.4 Results

The range of break sizes analyzed (Figure 4.4-1) bound the possible transient responses of small break LOCAs. For the smaller break sizes, primary-to-secondary heat transfer is maintained and an orderly cooldown of the unit can be achieved. A significant percentage of the heat load is transferred outside of the containment, and the containment response is non-limiting. For intermediate small break sizes, steam generator heat transfer can be interrupted. In addition, RCS pressure will remain too high for LPI injection to occur. Plant cooldown via break cooling will proceed very slowly, and heat transfer from the Reactor Building will be limited. These are the limiting small break cases. For larger small break sizes the RCS will depressurize to LPI injection conditions since break cooling is sufficient. Plant cooldown for this scenario, although off-normal, is achievable and is not unlike that following a large break LOCA. This section provides details of the mass and energy release for each case analyzed.

4.4.1 Case 1 - 0.0025 ft^2 Break

This case establishes the lower bound for all of the break sizes examined since natural circulation heat transfer is never lost in both primary system loops. Consequently, it is reasonable to infer that a break of less than 0.0025 ft^2 will yield similar if not better results than this case, assuming the same set of initial conditions and boundary conditions. Furthermore, this case will not yield the most limiting long-term containment temperature response since the operator can maintain control of the primary system cooldown rate and will be able to cool and depressurize the RCS until the LPI System is aligned. As a result, the containment temperature response to this break is not examined with the FATHOMS code. The transient response is discussed briefly to provide an indication of what phenomena were observed and the significance of these phenomena.

Once the break is initiated, liquid slowly drains from the RCS. The reactor continues to operate at full power until the RPS trips the reactor on variable pressure approximately 5 minutes after the break is opened. The combination of leakage and coolant shrinkage due to the trip causes the RCS to eventually lose subcooling. The HPI System is initiated to replace the lost coolant but the makeup capacity is not sufficient to offset the break flow until the HPI trains are cross-connected. As a result, the RCS voids in the reactor vessel head, and eventually in the Loop A hot leg (fluid exiting the pressurizer usually causes this loop to void first) thereby interrupting primary-to-secondary heat transfer in the loop. However, once the HPI System is cross-connected (10 minutes), HPI flow exceeds break flow marking the end of a decreasing RCS trend.

Although the HPI System is capable of compensating for all leakage, voids remain in the reactor vessel head and the Loop A hot leg as these regions stagnate. The active regions of the RCS, however, remain subcooled and natural circulation flow is easily maintained in Loop B. At the termination of the transient (2700 seconds), the HPI System is maintaining a positive mass balance and natural circulation flow in the B Loop is effectively removing decay heat. As a result, the RCS cooldown can proceed to LPI conditions without loosing natural circulation flow due to hot leg voiding. A summary of events for this case is shown in Table 4.4-1.

4.4.2 Case 2 - 0.005 ft^2 Break

The break area assumed in this case is twice as large as Case 1. Since many of the cases exhibit similar trends during the initial phase of the transient, a more detailed discussion of this case is provided to describe how the transient response will vary as a function of break area. A summary of events for this case is shown in Table 4.4-1.

Upon break initiation, pressure and pressurizer level begin to decrease as primary coolant is slowly drained from the RCS. Pressure continues to drop until the RPS trips the reactor on variable-low pressure. This trip setpoint is reached in Loop A first due to the hot liquid which drains from the pressurizer and influences the hot leg temperature indication. The reactor trip results in a mismatch in primary-to-secondary heat transfer which causes the coolant to cool and shrink, further lowering pressure.

Since the RCPs are assumed to trip coincident with the reactor trip, primary system flow begins to coast down immediately. This assumption has a significant impact on the post-trip pressure

response. Normally, RCS pressure will decrease to approximately 1100 psig as the RCS saturation temperature approaches the secondary saturation temperature. However, since forced flow is lost early in the transient, energy cannot be transferred to the secondary system as readily. As a result, the minimum post-trip pressure is approximately 1450 psig instead of 1100 psig (Figure 4.4-2).

Post-trip saturated natural circulation flow is maintained briefly. However, as level in the hot legs begin to fall, liquid can no longer spill over the hot leg U-bend. As a result, primary-to-secondary heat transfer is interrupted and the RCS begins to reheat and repressurize (Figure 4.4-2). As the fluid at the core exit begins to heat and expand, liquid is forced into the pressurizer and up the hot leg piping. When the hot leg level expands sufficiently, two-phase fluid spills into the steam generator tubes where the steam is condensed. This spillover phenomenon results in periodic primary-to-secondary heat transfer which condenses the hot leg voids and causes large drops in RCS pressure. These flow surges are temporary since continuous flow cannot be maintained because of inadequate hot leg levels. Once the hot leg voids condense, phase separation once again occurs in the hot leg and the reheat and expansion process repeats.

Beginning 10 minutes after reactor trip, the steam generator levels are raised to the LSCM setpoint (Figures 4.4-3). The process of injecting EFW flow through the upper header into the steam space causes the steam generators to depressurize (Figure 4.4-4). Steam generator pressure eventually recovers when EFW is throttled after reaching the SG level and hot leg liquid spillovers transfer energy to the steam generators.

Once the HPI System is cross-connected, the makeup flow is approximately equal to the break flow. This causes the hot leg levels to remain above the elevation necessary for high-elevation boiler-condenser mode (BCM) heat transfer but too low for liquid spill over. BCM heat transfer occurs when steam inside steam generator tubes condenses due to cooler secondary coolant on the outside of the tubes. High-elevation BCM heat transfer occurs if EFW is spraying on the tubes. Pool BCM occurs if the secondary level water is higher than the primary water level. As a result, sustained primary-to-secondary heat transfer is never established. Instead, the RCS evolves to a reheat and flush condition where the fluid at the core exit swells and expands into the pressurizer and hot legs until levels are high enough to spill into the steam generators. This mode of heat transfer, combined with break cooling, it is sufficient to remove decay heat for the first 8000 seconds of the transient (Figure 4.4-5).

Beginning at approximately 8000 seconds, hot leg levels dip low enough to allow for high-elevation BCM heat transfer. Renewed primary-to-secondary heat transfer means the RCS must no longer rely on the cyclic reheat and swell process to assist with the break in removing decay heat. Consequently, RCS pressure (and temperature) stabilizes but does not decrease constantly since the heat transfer rate is not sufficient to remove decay heat and cool the primary. There is a positive RCS mass balance during this period which causes the loops to slowly fill and the available heat transfer area to shrink. At about 20,000 seconds, the loop levels have increased sufficiently to terminate high elevation heat transfer. While the primary and secondary systems are coupled, the MSRVs cycle to maintain steam generator pressure (Figure 4.4-4).

Between 20,000 and 26,0000 seconds, the positive mass balance causes the loops to continue filling. After 26,000 seconds, these levels are almost high enough to continuously spill over the hot leg U-bends. The reheat and flush cycle is reestablished once again with a greater frequency than before since the required swell is less with higher hot leg liquid levels.

The hot legs did not refill in the 12 hours of simulation time. Instead, the RCS continues the swell and flush process. The reasons behind this behavior are clear. Each time liquid spills over the hot leg U-bend, liquid also flows over the RCP internal lip and towards the reactor vessel. This causes the void fraction in the break volume to decrease which shifts the break flow from two-phase to single phase liquid. The additional liquid mass lost out of the break depletes the mass gained via the HPI System over the refill cycle. In the end, the RCS mass balance is established at a quasi-equilibrium which supports the swell and flush process. Based upon these observations, the potential for refilling the hot legs and establishing continuous primary-to-secondary heat transfer in the short-term is unlikely. The hot metal in the U-bend region keeps this region hot. However, the periodic spill over process does help remove energy from the piping metal faster than if no flow is present at all. It is speculated that the hot legs will eventually refill although this may occur in the two or three day time frame.

EOP Heat Transfer Restoration Guidance

The Emergency Operating Procedure includes guidance which instructs the operator to open the loop high point vent valves (HPVs) if primary-to-secondary heat transfer does not exist and reactor coolant pumps are unavailable to perform pump bumps. These conditions have evolved but no operator actions are assumed to open the valves. Because this assumption is somewhat unrealistic, credit is taken for this procedural guidance by restarting of the transient described above at an earlier time assuming the operators open the HPVs as instructed by the EOP.

The hot leg HPVs are assumed to be opened 7200 seconds into the transient. By this time, it is clear that primary-to-secondary heat transfer has been interrupted for quite some time. Based upon these symptoms and the fact that no RCPs are available, the operator would be expected to open the HPVs.

Opening the HPVs has an immediate effect on the transient response. The volumetric relief capacity of the combined vents is initially equivalent to approximately 25% of the steam volume exiting the break, effectively increasing the available break cooling capacity. The RCS responds favorably to the additional heat removal capacity and begins to cool down at approximately 40°F/hr. The additional mass relief from the system also causes the liquid levels in the hot legs to slowly drop, thereby exposing a portion of the tubes for high elevation BCM. Finally, the low pressure regions formed by the HPVs near the top of the U-bend tend to draw liquid up into the hot leg which aids in spilling the two-phase mixture into the steam generator tubes where the steam can be condensed.

By approximately 12,000 seconds, decreasing RCS pressure has caused break flow to decrease, injection flow to increase, and thus loop levels to increase above the top of the tube sheet terminating high elevation BCM. In all other cases without the HPVs opened, the liquid at the core exit would heat and expand into the hot leg until a spillover into the steam generator occurred. The same trend is exhibited here although the amount of expansion required to prompt the spillover is diminished in the presence of the HPVs. This difference results in many smaller spillovers instead of the larger ones observed without the HPVs open. These small spillovers continue with a high frequency until approximately 20,000 seconds.

At approximately 20,000 seconds, primary system inventory has recovered sufficiently for the Loop B hot leg to refill with a two-phase mixture. Although the ensuing natural circulation flow in Loop B is far less stable than the subcooled counterpart, it is nevertheless quite effective in maintaining the desired cooldown rate (Figures 4.4-6, 4.4-7). Saturated natural circulation in Loop B cools and depressurizes the RCS low enough for the CFTs to begin to inject. The additional makeup capability from the CFTs help to refill the RCS and ensure that natural circulation is maintained in Loop B. Cool liquid from the CFTs and the cold legs merge in the vessel downcomer and flow towards the core. The combined cooling capacity is sufficient for the core exit temperatures to fluctuate between several degrees subcooled and saturation. Toward the end of this transient, it is apparent that Loop A is beginning to refill.

The positive contribution of opening the high point vents for this case is clearly evident. Since sustained primary-to-secondary heat transfer is likely, the long-term mass and energy release will be non-limiting. Consequently, the FATHOMS code is not executed for this break size.

4.4.3 Case 3 - 0.01 ft^2 Break

As expected, the Case 3 transient progression is very similar to Case 2. Since the break size is larger, many of the phenomena observed in Case 2 will occur in an earlier time frame. This is apparent from Table 4.4-2. The initial transient response for this case is essentially identical to Case 2 in terms of phenomena therefore a description of this phase will not be repeated.

After a reactor trip on variable low RCS pressure, saturated natural circulation flow is maintained briefly. However, as level in the hot legs begin to fall, liquid can no longer spill over the hot leg U-bend. As a result, primary-to-secondary heat transfer is interrupted and the RCS begins to reheat and repressurize (Figure 4.4-8). As the fluid at the core exit begins to heat and expand, liquid is forced into the pressurizer and up the hot leg piping. When the hot leg level expands sufficiently, two-phase fluid spills into the steam generator tubes where the steam is condensed. This spillover phenomena results in periodic primary-to-secondary heat transfer which condenses the hot leg voids and causes large drops in RCS pressure. These flow surges are temporary since continuous flow cannot be maintained because of inadequate hot leg levels. Once the hot leg voids condense, phase separation once again occurs in the hot leg and the reheat and expansion process repeats.

Beginning 10 minutes after reactor trip, the steam generator levels are raised to the LSCM set-point (Figures 4.4-9). The process of injecting EFW flow through the upper header into the steam space causes the steam generators to depressurize (Figure 4.4-10). Steam generator pressure eventually recovers when EFW is throttled and hot leg liquid spillovers transfer energy to the steam generators. Pressure remains near the secondary safety valve setpoint until sustained heat transfer is established and the steam generators are manually depressurized. Several safety valve lifts occur prior to this time to maintain secondary pressure.

For a period of time, the RCS and the steam generators are essentially uncoupled. RCS pressure remains well above the steam generator pressure and the steam generators cannot be used to lower the RCS pressure. As decay heat decreases, the combination of break cooling and some steam generator heat transfer is sufficient to gradually cool and depressurize the RCS. At

approximately 9000 seconds, levels in the primary system drop low enough to allow steam on the inside of the steam generator tubes to condense and transfer energy to the subcooled EFW liquid flowing down the outside of the tubes. This phenomenon is possible only when EFW is injected onto the upper portion of the steam generator tube bundle periphery and steam is available to condense on the inside of the tubes. Once high elevation BCM is established, the primary and secondary systems are effectively coupled. Between 9000 and 16000 seconds, the combined effects of break cooling and steam generator heat transfer are sufficient to cool the RCS down at approximately 50 ^{o}F/hr (Figure 4.4-11), somewhat below the target value of 80°F/hr assumed for the ADV control system.

As RCS pressure decreases, the CFTs begin to inject and the makeup flow rate from the HPI System increases. Increased makeup causes levels in the primary system to slowly rise above the elevation necessary to support high elevation BCM. As the levels increase, the available surface area for condensation is reduced. Eventually, the primary and secondary systems once again decouple and the primary cooling mechanism is break cooling. As a result, the cooldown rate decreases to approximately 5°F/hr. The transient is terminated with hot leg level too high to support high elevation BCM but too low to spill over the U-bend. It is suspected that the RCS will remain in this configuration for quite some time for the following reasons:

- An equilibrium has been established where break cooling alone is removing energy from the RCS. Thus, there are no physical phenomena which will force the loops to refill in short order.
- The metal temperatures in the U-bend regions and steam generator upper tube sheet are well above the equilibrium saturation temperature. As a result, steam bubbles will be maintained in this region until the structural metal has cooled. Since ambient cooling is the primary heat transfer mechanism (no natural circulation flow), this process could take a significant amount of time.

Furthermore, noncondensible gases which are absorbed by the coolant (predominately hydrogen which is used to scavenge oxygen) will be released as the RCS pressure decreases during a cooldown. The radiolysis process will also generate noncondensible gases (hydrogen and oxygen). The buoyancy of these gases will cause them to accumulate in the high points of the RCS, namely the reactor vessel head and the hot leg U-bend region. The introduction of noncondensible gases in the U-bend region can only prolong the interruption of natural circulation since these gases will occupy volume which needs to be filled with liquid before natural circulation can be restored. Although noncondensible gases are not modeled in these

analyses, the potential accumulation of these gases in the U-bend region is mentioned to highlight additional complications involved in refilling the hot legs following a small break LOCA.

4.4.4 Case 4 - 0.025 ft^2 Break

The break area assumed in this case is 250% the area assumed in the previous case. Even with this difference, the Case 4 transient progression is very similar to Case 3. Once again, because the break size is larger, many of the phenomena observed in Case 3 will occur in an earlier time frame. This is apparent from Table 4.4-2.

After the reactor trip and RCP coastdown, the RCS demonstrates the same reheat and repressurization phenomena observed in Case 3 (Figure 4.4-12). Fluid swells into the pressurizer and into the hot legs until it spills over the hot leg U-bend and into the steam generator. This spillover results in an abrupt reduction in RCS pressure. At about 600 seconds, EFW flow begins filling the steam generators to the LSCM setpoint. As before, the introduction of cool EFW spray into the steam space causes the steam generator level to increase (Figure 4.4-13) and steam generator pressure to decrease (Figure 4.4-14). In addition, the fluid levels in the hot legs are low enough by this time to allow high elevation BCM heat transfer to gradually cool and depressurize the RCS during the fill process.

After the LSCM level is established, steam generator pressure slowly increases towards RCS pressure as primary energy is transferred to the steam generators (Figure 4.4-14). Although this provides some decay heat removal, the bulk of the primary system energy removal is due to break cooling. The RCS continues to cool and depressurize while the steam generators continue to repressurize. At approximately 2400 seconds, levels in the RCS drop low enough to support pool BCM and the primary and secondary systems become coupled. Thus, sustained primary-to-secondary heat transfer is established and the steam generators are used to gradually cool and depressurize the RCS.

The RCS continues to cool and depressurize. A cooldown rate of approximately 34 $^{o}F/hr$ is maintained between 4000 and 11600 seconds. Eventually, the RCS depressurizes low enough for the CFTs to begin to inject. Mass inventory from the CFTs begin to refill the RCS, effectively eliminating the condensation area necessary for pool BCM. Heat transfer is then maintained by break cooling and high elevation BCM. Eventually, the loops fill until there is no condensation

area available for high elevation BCM. Beyond this point and through the end of the transient, decay heat is removed primarily by break cooling. The cooldown rate resulting from break cooling is approximately 9 ^{o}F/hr. The hot leg U-bend region remains voided and natural circulation flow is never established for the reasons discussed above.

Oscillatory break mass flow rates are observed during this transient beyond 8000 seconds. The effect of these oscillations can be observed in many of the primary system parameters and the containment response. These oscillations are apparently caused by fluctuations in the void fraction in the volume upstream of the break which is used by the horizontal stratification model to determine when liquid pull through will occur. These oscillations are not observed in Case 3 since the column of water in the steam generator tube region was sufficient to maintain the break site flooded. However, with the larger break area assumed in this case, the break site did not remain flooded and fluctuations in void fraction resulted. Although the oscillations do have a slight impact on the transient results, the mass and energy release is not affected.

4.4.5 Case 5 - 0.05 ft^2 Break

As the break area increases, the dominance of break cooling becomes more pronounced. The initial transient response for this case is similar to the previous two cases except the post-trip RCS pressure increase is much smaller (Figure 4.4-15). In fact, at about 250 seconds, the fluid quality at the break site increases sufficiently to begin a steady cooldown and depressurization of the RCS. This steady cooldown continues until EFW is initiated to raise the steam generator levels to the LSCM setpoint. During this fill time, the rate of depressurization increases because of the additional heat removal which results from high-elevation BCM. The slope of the cooldown decreases at approximately 900 seconds once the LSCM setpoint is reached and the EFW System is secured. The RCS continues to cool and depressurize via break cooling until the ADVs are opened at 1800 seconds.

When the ADVs are initially opened, steam generator pressure is greater than RCS pressure. As steam pressure is lowered, feedwater begins to flash and level drops. In response to this drop, the EFW System is actuated to maintain level (Figure 4.4-16). As noted previously, actuation of EFW condenses steam on the primary side of the steam generator tubes and removes energy from the RCS which lowers RCS pressure. Pressure in the steam generators is also reduced by the introduction of cool EFW spray (Figure 4.4-17). The combination of break cooling and steam generator heat transfer causes the RCS to cool and depressurize steadily.

Once RCS pressure drops below the over-pressure in the CFTs, the CFTs inject until they are empty. Almost immediately after the CFTs have emptied, the LPI system begins to inject. This behavior is very similar to what is expected during a LBLOCA and shows the continuity between the CFTs and LPI. When LPI begins to inject, the break site becomes single phase and the RCS is reflooded. The break area is sufficient to relieve all of the mass that HPI and LPI Systems are injecting. As a result, the liquid level in the RCS establishes an equilibrium where the head required to force liquid out the break is balanced with the injection mass flow. At the end of this simulation, core cooling is maintained by break cooling alone. A summary of events for this case is shown in Table 4.4-3

4.4.6 Case 6 - 0.10 ft^2 Break

The assumed break size for this case approaches the size which is traditionally termed an intermediate size break. Indeed, the transient signature is very similar to an intermediate size break. After an immediate reactor trip, the RCS sustains a fairly steady cooldown and depressurization as a large percentage of the initial primary system mass is expelled through the break (Figure 4.4-18). Because break cooling is so great, the influence of high elevation boiler condenser mode heat transfer is barely noticeable during the time frame in which the steam generator level is raised to the LSCM setpoint (Figures 4.4-19 and 4.4-20). Furthermore, because the RCS voids extensively during the blowdown, there is no means for readily transferring energy from the secondary to the primary system. Thus, the steam generators do not act as a heat load on the RCS. The RCS cools and depressurizes to a little over 300 psig when steaming of the steam generators begins at 1800 seconds.

The initial phase of steam generator steaming lowers pressure to the existing RCS pressure. Naturally, as the generators are steamed, level drops and the EFW System actuates to attempt to maintain level (Figure 4.4.6-3). Some energy is removed from the primary system during this period through high elevation BCM heat transfer. As steam generator pressure drops below the RCS pressure, the steam generators begin to remove even more energy from the primary system. The steam generators assist in cooling down the primary system until the LPI System begins injecting.

When the LPI System begins to inject, the break site is quickly reflooded. Even though single phase liquid begins to exit the break, the break area is sufficient to relieve all of the mass that the HPI and LPI Systems are injecting. As a result, the liquid level in the RCS establishes an

equilibrium where the head required to force liquid out the break is balanced with the injection mass flow. A summary of events for this case is shown in Table 4.4-3.

4.5 Long Term Mass and Energy Release

The mass and energy release rates generated by RELAP5 for Cases 3 through 6 are shown in Tables 4.5-1 through 4.5-4. Since it was determined that Cases 1 and 2 would not represent the limiting small break cases, the mass and energy release data is not included. As in the large break LOCA simulations, the mass and energy flow from each flow stream (liquid and vapor flow from each break) is integrated by RELAP5 control systems during the simulation. These integral values are then used as input to a utility program which averages the data.

4.6 References

4-1 Letter from G. B. Swindlehurst (Duke) to R. J. Shomaker (B&W) dated April 22, 1986, commenting on SBLOCA assumptions for II.K.3.31 analysis work.

Note: Page 4-22 has been deleted

TABLE 4.4-1

Case 1 and 2 Summary of Events

Time (seconds)			
Case 1 0.0025 ft^2	Case 2 0.005 ft^2	Case 2 w/ HPVs 0.005 ft^2	Action
0.0	0.0	0.0	Break located at the A1 RCP cold leg pump discharge is initiated
288	117	117	Reactor trip on variable low pressure in Loop A
			All RCPs are tripped
675	307	307	Loss of indicated subcooled margin
635	340	340	Pressurizer level off-scale low
796	413	413	HPI flow is initiated through 1 train (after 15 sec delay)
890	720	720	EFW begins to raise SG levels to the LSCM setpoint
1396	1,020	1,020	HPI System is cross-connected
NA	3,787	3,787	RBCUs are activated
NA	36,735	NA	RBS is activated
NA	20,225	23,200	Switch to sump recirculation
2700	43,200	32,400	Job terminated

TABLE 4.4-2

Case 3 and 4 Summary of Events

Time (seconds) Case 3 0.01 ft^2	Case 4 0.025 ft^2	Action
0	0	Break located at the A1 RCP cold leg pump discharge is initiated
51	15.9	Reactor trip on variable low pressure in Loop A All RCPs are tripped
138	44	Loss of indicated subcooled margin
185	90	Pressurizer level off-scale low
215	97	HPI flow is initiated through 1 train (after 15 sec delay)
655	620	EFW begins to raise SG levels to the LSCM setpoint
815	697	HPI System is cross-connected
1,603	556	RBCUs are activated
6,980	3,772	RBS is activated
12,504	11,088	Switch to sump recirculation
~9,000	2,400	Sustained BCM mode is established
9,380	2,400	ADVs are manually opened
~14,480	7,270	CFTs begin to inject
NA	NA	LPI begins to inject into the RCS
Not Empty	Not Empty	CFTs are empty
32,900	31,900	Job terminated

TABLE 4.4-3

Case 5 and 6 Summary of Events

Time (seconds) Case 2 0.05 ft^2	Case 3 0.10 ft^2	Action
0	0	0.01 ft^2 break located at the A1 RCP cold leg pump discharge is initiated
2.4	.65	Reactor trip on variable low pressure in Loop A All RCPs are tripped
15	6	Loss of indicated subcooled margin
50	42	Pressurizer level off-scale low
50	37	HPI flow is initiated through 1 train (after 15 sec delay)
604	600	EFW begins to raise SG levels to the LSCM setpoint
650	638	HPI System is cross-connected
N/A	N/A	Sustained pool BCM mode is established
1,800	1,800	ADVs are manually opened
1,959	903	CFTs begin to inject
2,955	2,721	LPI begins to inject into the RCS
2,964	2,724	CFTs are empty
3,600	3,004	Job terminated

TABLE 4.5-1 (pg. 1 of 4)

Case 3 M&E Release - 0.01 ft² SBLOCA

Time	Mass flow	Mass enthalpy	Liquid flow	Liquid enthalpy	Pressure	Time	Mass flow	Mass enthalpy	Liquid flow	Liquid enthalpy	Pressure
(sec)	(lbm/sec)	(Btu/lbm)	(lbm/sec)	(Btu/lbm)	(psia)	(sec)	(lbm/sec)	(Btu/lbm)	(lbm/sec)	(Btu/lbm)	(psia)
0	0	1150	204.27	564.34	2033.35	3255	38.9	1159.77	2.57	662.96	1886.15
60	0	1150	204.27	564.34	2033.35	3465	41.67	1163.43	0	180	1866.7
180	0	1150	166.24	559.23	1630.85	3675	37.86	1162.26	3.43	662.5	1847.25
300	0	1150	155.59	558.62	1496.95	3885	35.48	1161.07	5.95	656.8	1825.9
420	0	1150	149.94	569	1551.85	4095	37.57	1166.03	2.1	668.18	1799.75
540	0	1150	148.52	582.17	1622.65	4305	36.57	1167.97	1.24	650	1775.2
660	12.46	1166.36	62.85	609.65	1651.85	4515	36.33	1170.38	0.81	658.82	1753.3
780	8.15	1165.8	86.5	609.92	1718.75	4725	35.43	1170.7	1.86	653.85	1731.95
840	8.15	1165.8	86.5	609.92	1718.75	4935	36.43	1171.24	0.29	533.33	1710.8
840.01	36.07	1164.11	4.62	647.42	1830.05	5145	35.9	1175.07	0.29	733.33	1689.4
945	36.07	1164.11	4.62	647.42	1830.05	5355	35.81	1175.53	0.19	550	1667.95
1155	20.41	1161.92	65	578.02	1806.8	5565	36.48	1177.55	0	180	1647.1
1365	34.51	1167.49	6.57	642.75	1791.3	5775	35.9	1180.37	0	180	1631.05
1575	34.86	1161.75	7.67	653.42	1859.55	5985	35.57	1179.38	0	180	1618.75
1785	37.15	1159.59	4.95	662.5	1877.85	6195	35.24	1179.73	0.05	100	1599.45
1995	38.06	1160.54	3.14	665.15	1879.05	6405	23.05	1177.69	14.57	601.96	1573.65
2205	36.67	1157.64	7.05	668.24	1901	6615	13.1	1170.91	39.81	569.74	1548.45
2415	36.66	1156.53	7.43	669.23	1920	6825	4.76	1170	96.95	578.39	1491.9
2625	37.23	1155.28	7.14	673.33	1927.8	7035	12	1182.54	52.43	564.03	1443.25
2835	39.13	1157.48	3.43	665.28	1910.4	7255.1	15.99	1179.35	25.98	535.12	1437.15
3045	37.35	1158.61	5.19	670.64	1894.7	7475.2	15.52	1180.98	28.67	531.56	1433.3

TABLE 4.5-1 (pg. 2 of 4)

Case 3 M&E Release - 0.01 ft² SBLOCA

Time (sec)	Mass flow (lbm/sec)	Mass enthalpy (Btu/lbm)	Liquid flow (lbm/sec)	Liquid enthalpy (Btu/lbm)	Pressure (psia)	Time (sec)	Mass flow (lbm/sec)	Mass enthalpy (Btu/lbm)	Liquid flow (lbm/sec)	Liquid enthalpy (Btu/lbm)	Pressure (psia)
7685.2	12.1	1181.1	38.14	545.57	1419.45	12125	5.33	1196.43	86.48	476.87	994.65
7895.2	8.33	1171.43	46.19	564.95	1397.4	12335	6.62	1201.44	64.14	482.55	954.3
8105.2	9.9	1182.69	42.29	549.55	1380.45	12555	7.35	1201.18	22.3	497.08	899
8315.2	12.95	1180.15	34.9	525.24	1373.8	12775	4.62	1195.88	25.86	517.5	860.9
8525.2	13.57	1185.96	32.57	523.39	1372.4	12985	3.29	1202.9	32.62	518.25	852
8735.2	12.52	1182.51	35.19	529.09	1367.95	13195	0	1150	126.38	472.49	820.2
8945.2	11.1	1180.26	39.14	536.5	1357.9	13405	0	1150	106.29	484.77	769.2
9155.2	10.67	1183.04	40.29	530.73	1343.1	13615	2	1214.29	59.24	487.94	738
9365.2	12.14	1188.24	36	513.23	1315.85	13825	3.1	1200	34.43	491.01	721.7
9575.2	13.57	1185.96	27.33	506.97	1279.05	14035	2.76	1206.9	35.57	488.62	713.95
9785.2	11.14	1183.76	35.57	511.38	1258.75	14245	1.29	1185.19	64.62	479.73	704.2
10010.1	11.26	1188.89	34.74	505.4	1240.8	14455	0.05	1150	89.24	480.79	684.7
10235	12.19	1191.41	29	494.25	1208.25	14665	0.05	2000	83.67	480.93	662.65
10445	12.14	1192.16	27.9	488.05	1180.4	14875	1.67	1228.57	48.43	479.84	655.95
10655	11.33	1193.28	30.14	486.57	1159.65	15085	1.52	1187.5	53.57	470.22	656.9
10865	10.76	1190.27	31.33	483.28	1135.75	15295	0.05	1150	92.52	467.32	644.25
11075	10.81	1198.24	30.57	475.08	1112.2	15505	0	1150	83.38	471.16	624.95
11285	8.76	1195.65	38	482.46	1094	15715	0.05	1150	81.95	468.33	607.1
11495	2.43	1196.08	133.29	495.89	1042.85	15895	0.07	1150	80.73	465.73	591.9
11705	14.57	1205.88	15.48	526.15	1004.25	15970	0.07	1150	80.73	465.73	591.9
11915	12.48	1202.29	15.57	480.12	1009.45	16000	0.1	1150	79.95	462.18	579.55

TABLE 4.5-1 (pg. 3 of 4)

Case 3 M&E Release - 0.01 ft² SBLOCA

Time	Mass flow	Mass enthalpy	Liquid flow	Liquid enthalpy	Pressure	Time	Mass flow	Mass enthalpy	Liquid flow	Liquid enthalpy	Pressure
(sec)	(lbm/sec)	(Btu/lbm)	(lbm/sec)	(Btu/lbm)	(psia)	(sec)	(lbm/sec)	(Btu/lbm)	(lbm/sec)	(Btu/lbm)	(psia)
16105	0.1	1150	79.95	462.18	579.55	20326	0.05	1150	76.86	429.37	457.15
16315	0.1	1150	78.14	460.09	565.85	20536	0	1150	76.86	429.37	454.1
16525	0.05	1150	75.33	457.65	554.3	20746	0	1150	77.05	427.07	451.1
16735	0.05	1150	77.76	454.99	543.2	20956	0.05	1150	77.14	426.54	448.7
16945	0.05	1150	77.43	453.26	536.9	21166	0	1150	76.67	426.71	444.7
17155	0	1150	78.43	451.12	533.05	21376	0	1150	76.67	423.6	440.95
17365	0.05	1150	77.29	449.17	523.45	21586	0	1150	76.19	425	437.8
17575	0	1150	76.48	447.7	513.9	21796	0.05	1150	76.19	423.12	435.1
17785	0.05	1150	75.71	445.28	506.6	22006	0	1150	76.19	422.5	433.15
17995	0	1150	76.33	443.54	500.45	22216	0	1150	76.19	420.62	430.15
18205	0	1150	77	442.18	496.15	22426	0.05	1150	75.71	418.87	425.7
18415	0.05	1150	77.24	440.2	492.95	22636	0	1150	75.71	419.5	421.3
18625	0	1150	76.48	439.6	487.65	22846	0	1150	75.24	418.99	419.4
18835	0.05	1150	76.05	437.7	480.05	23056	0	1150	75.24	418.99	417.4
19045	0	1150	77.05	435.72	475.3	23266	0	1150	75.24	416.46	414.1
19255	0	1150	77.14	434.57	473.3	23476	0.05	1150	75.24	415.19	412
19465	0.05	1150	77.19	433.68	470.55	23686	0	1150	75.24	414.56	410.05
19675	0	1150	77.1	432.98	467.15	23896	0	1150	75.71	413.84	408.25
19885	0	1150	77.14	432.1	464.45	24106	0	1150	75.24	415.82	407.25
20105.5	0	1150	76.62	431.07	460.7	24316	0.05	1150	75.71	412.58	404.55

TABLE 4.5-1 (pg. 4 of 4)

Case 3 M&E Release - 0.01 ft² SBLOCA

Time	Mass flow	Mass enthalpy	Liquid flow	Liquid enthalpy	Pressure	Time	Mass flow	Mass enthalpy	Liquid flow	Liquid enthalpy	Pressure
(sec)	(lbm/sec)	(Btu/lbm)	(lbm/sec)	(Btu/lbm)	(psia)	(sec)	(lbm/sec)	(Btu/lbm)	(lbm/sec)	(Btu/lbm)	(psia)
24526	0	1150	75.24	413.29	401.25	28746	0	1150	77.62	404.91	388.55
24736	0	1150	75.24	409.49	398.75	28956	0	1150	77.62	403.68	386.8
24946	0	1150	75.24	412.03	397.55	29166	0	1150	78.57	401.21	383.05
25156	0	1150	76.19	409.38	396.8	29376	0	1150	79.52	398.8	380.9
25366	0.05	1150	76.19	408.75	396.75	29586	0	1150	79.52	401.2	380.9
25576	0	1150	76.19	410.62	398.4	29796	0	1150	79.52	399.4	380.3
25786	0	1150	76.19	410.62	397.7	30006	0	1150	78.57	400	379.3
25996	0	1150	77.62	406.13	396	30216	0	1150	79.52	397.01	378.05
26206	0	1150	76.67	409.94	396.25	30426	0	1150	79.05	399.4	377.35
26416	0	1150	77.62	406.75	396.85	30636	0	1150	78.57	398.79	375.05
26626	0.05	1150	77.14	408.64	396.75	30846	0	1150	78.57	396.97	372
26836	0	1150	77.14	407.41	396.2	31056	0	1150	78.57	395.15	370.9
27046	0	1150	77.14	409.26	395.7	31266	0	1150	78.57	397.58	370.35
27266	0	1150	77.83	406.15	394.95	31476	0	1150	78.57	396.97	370
27486	0	1150	77.14	408.02	393.7	31686	0	1150	79.05	394.58	369.9
27696	0	1150	77.62	406.13	391.7	31896	0	1150	78.1	395.73	368.45
27906	0.05	1150	78.1	404.27	390.7	32106	0	1150	78.57	396.36	367.05
28116	0	1150	77.62	405.52	390.75	32316	0	1150	78.57	392.73	365.75
28326	0	1150	77.62	405.52	390	32526	0	1150	78.57	394.55	364.75
28536	0	1150	77.62	404.91	389.25	32721	0	1150	78.33	392.91	363.45

TABLE 4.5-2 (pg. 1 of 4)

Case 4 M&E Release - 0.025 ft² SBLOCA

Time (sec)	Mass flow (lbm/sec)	Mass enthalpy (Btu/lbm)	Liquid flow (lbm/sec)	Liquid enthalpy (Btu/lbm)	Pressure (psia)	Time (sec)	Mass flow (lbm/sec)	Mass enthalpy (Btu/lbm)	Liquid flow (lbm/sec)	Liquid enthalpy (Btu/lbm)	Pressure (psia)
0	0	1150	440.04	564.47	1871.4	3340	21.3	1199.53	47.95	505.74	945.6
60	0	1150	440.04	564.47	1871.4	3540	18.35	1201.63	56.1	501.78	922.25
180	0.26	1167.28	318.39	578.72	1519.5	3740	18.7	1200.53	52.3	495.22	898.1
300	34.6	1166.18	103.32	624.86	1657.3	3940	18.5	1202.7	50.85	488.69	876.8
420	55.24	1162.72	60.75	638	1767.75	4140	16.05	1199.38	56.8	492.08	861.35
540	55.39	1163.83	119.83	595.55	1711.6	4340	14.1	1202.13	62.05	493.96	849.35
660	33.89	1170.15	101.75	615.56	1624.6	4540	14.6	1202.05	59.25	488.61	836.25
780	71.18	1176.79	23.42	607.47	1588.5	4740	15.05	1202.66	56	483.04	822.85
840	71.18	1176.79	23.42	607.47	1588.5	4940	14.75	1200	55.45	482.42	809.6
840.01	65.11	1181.45	23.4	607.91	1510.95	5140	13.95	1204.3	56.25	480.89	796.3
940	65.11	1181.45	23.4	607.91	1510.95	5340	13.2	1204.55	57.15	480.31	782.95
1140	58.08	1184.13	27.9	599.28	1436.6	5540	12.4	1201.61	58.25	479.83	770
1340	58.3	1188.34	20.8	594.71	1380.25	5740	11.65	1201.72	59.15	478.44	757.75
1540	56.52	1191.1	21.5	579.3	1316.3	5940	11	1204.55	59.55	478.59	745.75
1740	47.86	1191.6	29.8	561.58	1257.35	6140	10.35	1207.73	60.05	478.77	733.65
1940	40.31	1193.25	32.75	535.27	1204.5	6340	9.75	1205.13	60.7	476.94	721.75
2140	31.18	1193.26	43.35	529.41	1152.95	6540	9.15	1196.72	61.45	477.62	710.25
2340	32.8	1193.6	33.2	519.58	1108.6	6740	8.5	1205.88	62	476.61	699.15
2540	34.45	1198.84	30.15	509.12	1067.75	6940	7.95	1207.55	62.6	476.04	688.15
2740	31.75	1198.43	31.3	503.19	1031.45	7145.1	7.42	1198.72	63.23	474.79	676.75
2940	22.7	1198.24	52.85	507.1	999.25	7350.2	7.15	1209.79	60	478.33	660.55
3140	23.25	1200	47.15	498.41	970.15	7550.2	7.2	1208.33	55.35	477.87	644.25

TABLE 4.5-2 (pg. 2 of 4)

Case 4 M&E Release - 0.025 ft² SBLOCA

Time (sec)	Mass flow (lbm/sec)	Mass enthalpy (Btu/lbm)	Liquid flow (lbm/sec)	Liquid enthalpy (Btu/lbm)	Pressure (psia)	Time (sec)	Mass flow (lbm/sec)	Mass enthalpy (Btu/lbm)	Liquid flow (lbm/sec)	Liquid enthalpy (Btu/lbm)	Pressure (psia)
7750.2	6.05	1198.35	62.4	471.96	633.75	12180	5.4	1194.44	37.05	430.5	444.9
7950.2	6.4	1203.12	82.6	466.71	622.15	12380	4.35	1206.9	42.8	427.57	442.2
8150.2	6.05	1206.61	75.05	466.36	607.85	12580	4.65	1215.05	40.9	427.87	440.7
8350.2	6.2	1209.68	54.8	466.24	596.75	12780	0.85	1176.47	153.65	422.39	435.25
8550.2	3.05	1196.72	122.5	459.59	581.75	12980	6.3	1206.35	32	418.75	425.6
8750.2	8.35	1203.59	63.1	448.49	566.2	13180	5.5	1209.09	32.55	428.57	423.8
8950.2	8.25	1212.12	44.45	442.07	559.75	13380	5.25	1209.52	34.45	426.71	423.8
9150.2	5.35	1205.61	52.1	457.77	551.95	13580	4.8	1208.33	37.75	421.19	422.75
9350.2	4	1200	78.95	451.55	539.5	13780	1.35	1185.19	139.85	415.8	415.85
9550.2	4.2	1214.29	124.9	445.16	524.5	13980	5.7	1201.75	30.5	418.03	407.3
9750.2	8.25	1206.06	37.4	431.82	513.6	14180	4.2	1214.29	38.35	419.82	408.9
9960.1	5.55	1204.92	45.54	445.55	504.15	14380	1.9	1210.53	99.8	412.83	409.1
10170	4.6	1217.39	48.5	443.3	494.15	14580	4.65	1193.55	51	411.76	405.95
10370	1.45	1172.41	163	436.2	481.25	14780	3.05	1229.51	77.85	416.18	404
10570	8.05	1204.97	32.2	420.81	470.25	14980	2.45	1183.67	115.35	411.36	399.1
10770	5.45	1211.01	40.5	432.1	465.3	15180	4.75	1221.05	33.6	416.67	397.05
10970	4.3	1209.3	44.95	436.04	459.7	15380	2.75	1181.82	63.05	415.54	398.75
11175	4.57	1208.33	45.86	428.87	460.2	15580	2.95	1203.39	103.1	407.37	397.3
11380	1.15	1217.39	161.65	425.92	455	15780	3.7	1229.73	47.5	414.74	395.3
11580	6.7	1208.96	33.65	423.48	447.05	15920	0.12	1150	182.5	410.96	389.85
11780	6	1200	32.95	432.47	445.25	15960	0.12	1150	182.5	410.96	389.85
11980	5.75	1217.39	34.2	432.75	444.45	15980	3.45	1202.9	78.5	412.1	380.55

TABLE 4.5-2 (pg. 3 of 4)

Case 4 M&E Release - 0.025 ft² SBLOCA

Time	Mass flow	Mass enthalpy	Liquid flow	Liquid enthalpy	Pressure	Time	Mass flow	Mass enthalpy	Liquid flow	Liquid enthalpy	Pressure
(sec)	(lbm/sec)	(Btu/lbm)	(lbm/sec)	(Btu/lbm)	(psia)	(sec)	(lbm/sec)	(Btu/lbm)	(lbm/sec)	(Btu/lbm)	(psia)
16080	3.45	1202.9	78.5	412.1	370.05	20280	0.45	1333.33	147.5	392.54	371.35
16280	3.6	1208.33	39.5	413.92	370.2	20480	2	1200	99.5	398.99	365.65
16480	0.05	2000	168.5	407.72	363.2	20680	4.4	1204.55	25.5	396.08	358.9
16680	5.85	1205.13	27	401.85	347.9	20880	2.55	1215.69	47	397.87	348.3
16880	4.05	1209.88	33.5	408.96	341.35	21080	0.05	1150	171.5	390.38	344.2
17080	2.7	1203.7	67	407.46	332.35	21280	2.95	1203.39	64.5	393.8	333.5
17280	3.6	1194.44	74.5	401.34	322.35	21480	3.5	1200	30	391.67	322.85
17480	3.7	1216.22	36	408.33	318.75	21685.5	1.23	1230.77	89.57	392.59	314.05
17680	0.55	1181.82	154	401.95	313.95	21891	1.65	1212.12	112	388.39	311.15
17880	3.1	1225.81	75	402.67	302.55	22091	3.85	1194.81	26.5	392.45	304.6
18080	3.8	1197.37	32.5	412.31	299.85	22291	1.9	1210.53	66	390.91	294.65
18280	2.55	1196.08	43.5	404.6	292.9	22491	0.5	1200	146	385.27	288.95
18480	0.1	1500	173.5	398.27	284.6	22691	3.65	1205.48	26.5	392.45	283.75
18680	4.25	1200	29.5	406.78	278.9	22891	2.3	1195.65	36.5	397.26	280.5
18880	1.3	1192.31	111	400.45	274.8	23091	0.3	1333.33	152	384.54	274.8
19080	2.55	1215.69	85.5	401.75	273.15	23291	3.7	1189.19	29	393.1	270.15
19280	4.05	1197.53	30	400	272.95	23491	2.4	1229.17	37.5	394.67	270.9
19480	2.4	1208.33	42	402.38	266.5	23691	0.05	1150	166.5	384.38	263.45
19680	0.3	1333.33	155.5	395.18	258.65	23891	2.05	1195.12	77	388.31	262.65
19880	3.4	1191.18	59	400.85	255.05	24091	3.2	1203.12	27.5	390.91	255.05
20080	3.45	1202.9	32.5	404.62	380.55	24291	0.3	1333.33	142.5	382.81	380.2

TABLE 4.5-2 (pg. 4 of 4)

Case 4 M&E Release - 0.025 ft² SBLOCA

Time (sec)	Mass flow (lbm/sec)	Mass enthalpy (Btu/lbm)	Liquid flow (lbm/sec)	Liquid enthalpy (Btu/lbm)	Pressure (psia)	Time (sec)	Mass flow (lbm/sec)	Mass enthalpy (Btu/lbm)	Liquid flow (lbm/sec)	Liquid enthalpy (Btu/lbm)	Pressure (psia)
24491	2.75	1200	53.5	384.11	371.05	28691	2.25	1177.78	30	385	358.65
24691	2.45	1183.67	46	388.04	356.75	28891	0.05	1150	152.5	371.15	352.8
24891	0	1150	160.5	381.31	354.75	29091	1.7	1205.88	82.5	375.15	346.9
25091	3.3	1212.12	37.5	384	351.2	29291	3.15	1206.35	22.5	380	333
25291	2.65	1207.55	30	386.67	340.75	29491	1.6	1218.75	34	383.82	320.5
25491	0.8	1187.5	110	377.73	327.05	29691	0	1150	161	368.01	318.6
25691	0.7	1150	126.5	380.63	319.4	29891	1	1150	102.5	376.59	316.45
25891	3.55	1211.27	23	389.13	312.85	30091	3.4	1205.88	21	378.57	309.15
26091	1.15	1173.91	94.5	379.37	314.25	30291	0.45	1222.22	111.5	371.3	299.1
26291	0.75	1266.67	123	379.27	302.3	30491	2.2	1204.55	59	374.58	293.65
26491	3.6	1180.56	23	376.09	290.9	30691	1.75	1200	45	375.56	288.05
26691	1.95	1230.77	34	389.71	288.65	30891	0	1150	152.5	369.51	283.5
26891	0.05	1150	159.5	375.24	286.45	31091	2.55	1196.08	54	371.3	274.65
27091	2.2	1227.27	69	378.99	278.15	31291	2.25	1222.22	28	376.79	270.65
27291	3.05	1196.72	25	386	271.9	31491	0.05	1150	149	368.46	267.6
27491	0.3	1166.67	129	376.74	265.95	31691	1.45	1172.41	85.5	373.1	268
27691	1.05	1190.48	104	378.37	265.45	31851	3.08	1216.22	20.83	372	262.35
27891	3.3	1196.97	23	378.26	265.6	31911	3.08	1216.22	20.83	372	255.6
28091	1.2	1208.33	81	374.69	260.65						
28291	0.05	1150	151	374.83	376.05						
28491	3.4	1220.59	27	370.37	368.55						

TABLE 4.5-3 (pg. 1 of 2)

Case 5 M&E Release - 0.05 ft² SBLOCA

Time	Mass flow	Mass enthalpy	Liquid flow	Liquid enthalpy	Pressure	Time	Mass flow	Mass enthalpy	Liquid flow	Liquid enthalpy	Pressure
(sec)	(lbm/sec)	(Btu/lbm)	(lbm/sec)	(Btu/lbm)	(psia)	(sec)	(lbm/sec)	(Btu/lbm)	(lbm/sec)	(Btu/lbm)	(psia)
0	0	1150	890.61	565.9	1878.25	859	87.17	1205.41	29.52	541.94	1022.5
20.5	0	1150	890.61	565.9	1878.25	902	85.07	1208.12	27.27	527.5	986.67
61.5	0	1150	736.05	557.09	1435.65	946.5	80.02	1209.66	37.56	492.31	965.28
103	0.02	1178.05	677.17	563.55	1385.6	991.5	81.84	1210.7	32.44	473.97	950.51
145	16.19	1177.99	350.62	587.94	1420.25	1036.5	76.38	1209.78	43.33	465.64	937.92
187	55.22	1178.54	200	580.95	1482	1081.5	75.71	1212.21	40.89	454.35	926.8
229	87.79	1177.62	106.19	585.2	1504.25	1126.5	75.73	1211.85	42.22	421.05	911.82
271	120.93	1181.51	57.38	586.72	1479.6	1171.5	78.87	1211.61	45.56	326.83	899.31
313	101.05	1181.43	67.14	598.58	1443	1216.5	74.64	1214.65	40.22	434.81	890.19
355	111.67	1184.43	58.57	573.58	1420.8	1261.5	72.27	1211.56	45.56	404.39	880.7
397	98.57	1183.82	62.86	587.88	1405.75	1306.5	72.67	1214.07	45.11	391.13	872.76
439	95.55	1184.15	62.62	582.51	1396.8	1351.5	67.11	1208.61	64.89	379.11	866.31
481	97.57	1184.48	62.62	584.03	1383.7	1396.5	70.22	1215.19	57.56	355.6	860.15
523	106.1	1187.16	58.33	576.33	1365.5	1441.5	66.67	1210	59.56	379.48	853.79
565	101.14	1187.85	60.71	573.73	1346.25	1486.5	68.22	1211.73	53.56	372.2	846.24
607	99.62	1188.58	56.67	568.91	1320.05	1531.5	68.89	1206.45	58.22	344.66	837.65
649	102.57	1191.74	47.62	560	1280.25	1576.5	66.44	1214.05	57.11	354.09	829.69
691	102.62	1195.13	38.81	562.58	1227.5	1621.5	67.78	1213.11	57.78	338.46	823.03
733	95.45	1197.56	37.62	566.46	1172.3	1666.5	66.89	1212.62	58.22	347.33	817.32
775	97	1200.79	26.9	560.18	1120	1711.5	65.33	1210.88	51.78	369.1	810.91
817	89.29	1202.93	31.67	551.13	1068.85	1756.5	66.22	1214.77	56.44	330.71	804.33

TABLE 4.5-3 (pg. 2 of 2)

Case 5 M&E Release - 0.05 ft² SBLOCA

Time	Mass flow	Mass enthalpy	Liquid flow	Liquid enthalpy	Pressure	Time	Mass flow	Mass enthalpy	Liquid flow	Liquid enthalpy	Pressure
(sec)	(lbm/sec)	(Btu/lbm)	(lbm/sec)	(Btu/lbm)	(psia)	(sec)	(lbm/sec)	(Btu/lbm)	(lbm/sec)	(Btu/lbm)	(psia)
1800.08	69.28	1208.9	61.21	275.19	796.8	2740.8	3.11	1214.29	42.44	371.73	235.82
1843.65	66.89	1219.27	40.22	342.54	784.88	2785.8	2	1222.22	207.33	333.33	220.13
1888.65	62.44	1213.52	28.89	461.54	753.74	2830.8	3.11	1214.29	335.11	326.92	207.64
1933.65	61.11	1210.91	11.56	480.77	700.62	2875.8	1.78	1250	339.33	322.86	196.68
1978.65	47.56	1214.95	40	394.44	647.18	2920.8	2.44	1181.82	307.78	324.19	185.97
2023.65	49.33	1216.22	37.78	317.65	602.26	2965.8	1.33	1166.67	327.56	308.68	175.68
2068.65	43.78	1223.35	40.89	331.52	564.75	3010.8	1.33	1166.67	306	309.37	166.46
2113.65	40.22	1209.94	38.22	354.65	531.44	3055.8	1.11	1400	306.89	303.4	158.21
2158.65	37.11	1221.56	36	351.85	499.24	3100.8	1.33	1000	304.89	295.92	151.13
2203.65	34.67	1217.95	37.56	337.28	471.31	3145.8	1.11	1200	316.67	286.32	145.12
2248.65	33.33	1213.33	36.89	325.3	445.76	3190.8	0.89	1250	322.22	277.24	139.73
2293.65	31.56	1218.31	27.56	362.9	422.8	3235.8	0.89	1000	331.78	267.25	134.9
2338.65	28.67	1217.05	30.67	376.81	401.98	3279.38	0.47	2000	343.3	252.94	130.66
2382.23	19.45	1219.51	43.42	415.3	378.73	3322.95	2	1111.11	333.56	251.17	125.66
2425.8	24.22	1211.01	29.11	366.41	353.52	3367.95	0.89	1250	346	237.64	121.96
2470.8	17.56	1215.19	34.67	391.03	330.34	3412.95	0.67	1000	358.89	225.39	121.03
2515.8	21.56	1216.49	27.78	336	310.11	3457.95	1.11	1200	156.89	259.21	118.56
2560.8	20.67	1215.05	27.56	306.45	294.28	3502.95	0	1150	254.89	238.88	113.95
2605.8	20	1222.22	25.56	295.65	279.59	3547.95	0	1150	227.56	230.47	111.59
2650.8	18.67	1202.38	25.56	304.35	264.85	3585.38	0	1150	224.79	223.55	110.88
2695.8	11.78	1207.55	30.44	372.26	251.83	3600.3	0	1150	224.79	223.55	110.88

TABLE 4.5-4 (pg. 1 of 2)

Case 6 M&E Release - 0.10 ft² SBLOCA

Time (sec)	Mass flow (lbm/sec)	Mass enthalpy (Btu/lbm)	Liquid flow (lbm/sec)	Liquid enthalpy (Btu/lbm)	Pressure (psia)	Time (sec)	Mass flow (lbm/sec)	Mass enthalpy (Btu/lbm)	Liquid flow (lbm/sec)	Liquid enthalpy (Btu/lbm)	Pressure (psia)
0	0	1150	1642.22	567.26	1868.35	679	112.7	1202.9	111.67	510.45	921.79
16	0	1150	1642.22	567.26	1868.35	709	118.6	1205.73	113.33	494.12	886.2
48	0	1179.07	1423.97	558.65	1410.75	739	125.1	1208.37	106	481.13	849.93
79	13.02	1179.55	956.73	573.44	1359	769	127.2	1209.91	83.33	472	813.62
109	81.72	1181.22	407.67	578.41	1359.6	799	127.33	1214.66	62.67	462.77	777.86
139	130.57	1182.86	253.33	569.08	1351.2	829	126.9	1213.55	42	460.32	743.26
169	147.32	1184.7	194.33	555.4	1327.15	859	123.5	1217.27	33.67	455.45	710.47
199	143.23	1186.18	192	551.56	1300.35	890	121	1219.01	30	458.33	680.12
229	138.47	1187.29	192.67	549.65	1274.2	924	112.5	1219.75	41.94	430.46	649.43
259	135.17	1188.16	188.67	547.7	1249.3	960	111.67	1223.88	30	453.7	619.48
289	132.57	1189.59	183.33	544.73	1225.6	996	105.56	1221.05	31.11	446.43	593.04
319	130.8	1190.62	175.33	541.44	1202.1	1032	98.61	1225.35	36.11	423.08	568.51
349	130.13	1191.6	163.67	539.71	1177.15	1068	90.56	1220.86	45	407.41	546.06
379	125.73	1192.74	160.67	539.42	1153.05	1104	81.94	1216.95	59.17	403.76	526.58
409	124.23	1194.26	153	535.95	1130.6	1140	75	1218.52	69.17	397.59	509.36
439	121.63	1194.85	141.67	541.18	1105.25	1176	71.39	1217.9	71.94	393.82	493.56
469	118.6	1196.46	131.33	548.22	1078.7	1212	68.33	1215.45	72.5	390.8	478.69
499	118.93	1197.59	113.67	548.39	1054.75	1248	65.83	1219.41	72.78	385.5	464.77
529	108.9	1199.27	160.67	539.42	1037.7	1284	63.33	1219.3	71.11	386.72	451.42
559	125.5	1201.06	121	523.42	1024.85	1320	61.11	1218.18	71.11	378.91	438.72
589	144.8	1203.5	109.33	515.24	1008.72	1356	58.61	1218.01	71.11	382.81	427.53
619	141.93	1203.62	114	523.39	985.95	1392	56.94	1219.51	71.39	377.43	416.74
649	118.67	1202.25	113	516.22	955.8	1428	55.28	1216.08	70	373.02	406.19

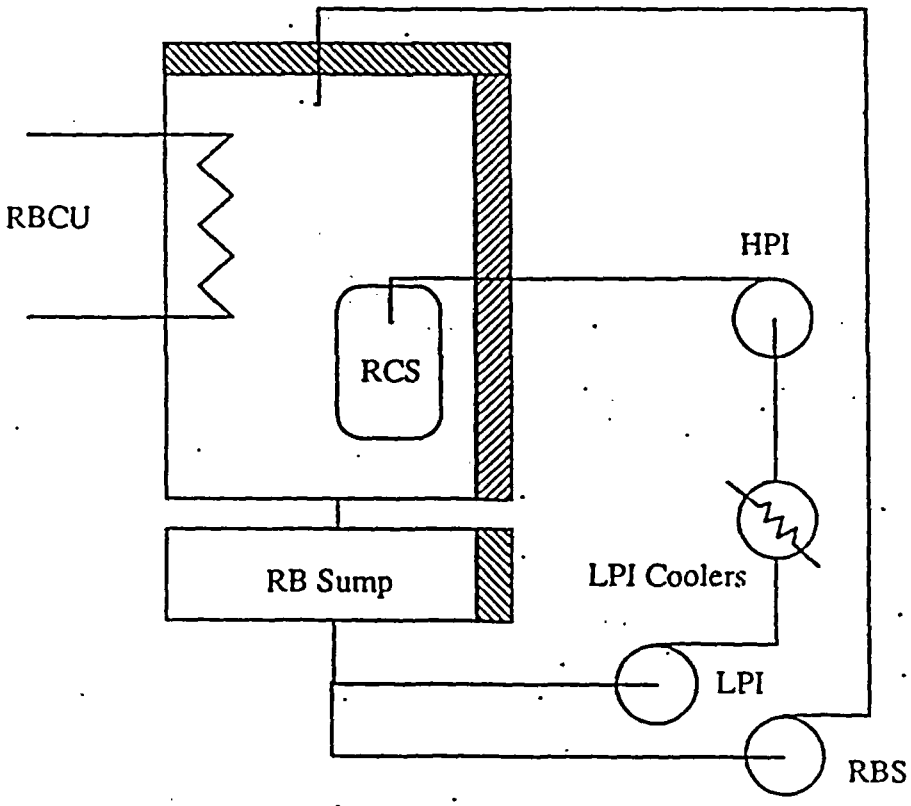
TABLE 4.5-4 (pg. 2 of 2)

Case 6 M&E Release - 0.10 ft² SBLOCA

Time (sec)	Mass flow (lbm/sec)	Mass enthalpy (Btu/lbm)	Liquid flow (lbm/sec)	Liquid enthalpy (Btu/lbm)	Pressure (psia)	Time (sec)	Mass flow (lbm/sec)	Mass enthalpy (Btu/lbm)	Liquid flow (lbm/sec)	Liquid enthalpy (Btu/lbm)	Pressure (psia)
1464	54.17	1215.38	68.61	364.37	396.47	2289.15	22.22	1212.5	83.06	327.76	245.94
1500	52.22	1218.09	65.83	371.31	387.41	2325.15	25	1211.11	78.33	326.24	246.32
1536	50.83	1218.58	63.89	373.91	378.17	2361.15	25.28	1208.79	77.22	312.95	244.18
1572	49.44	1207.87	62.78	376.11	369.48	2397.15	23.61	1211.76	78.06	320.28	241.54
1608	46.11	1222.89	66.94	377.59	361.59	2433.15	21.11	1210.53	81.67	323.13	238.19
1644	50.56	1208.79	45.28	392.64	354.71	2469.15	22.22	1212.5	86.67	304.49	233.66
1680	42.78	1220.78	70	376.98	349.45	2505.15	21.39	1194.81	84.17	310.23	229.92
1716	41.94	1205.3	72.5	371.65	344.12	2541.15	18.61	1223.88	78.61	314.49	226.06
1752	40.56	1212.33	70.56	377.95	338.1	2577.15	20	1208.33	85.83	304.21	221.17
1786.58	43.44	1215.28	56.11	381.72	333.44	2613.15	21.11	1197.37	82.5	292.93	216.2
1821.15	42.5	1215.69	59.72	372.09	331.53	2649.15	20	1208.33	83.06	287.63	209.34
1857.15	41.94	1211.92	58.61	383.89	330.62	2683.73	17.8	1203.39	293.51	305.24	197.79
1893.15	44.17	1213.84	48.33	390.8	330.29	2718.3	4.72	1235.29	779.44	297.22	180.77
1929.15	42.5	1215.69	58.61	379.15	332.17	2754.3	2.5	1222.22	517.22	323.31	163.09
1965.15	40.28	1213.79	61.94	376.68	328.8	2790.3	2.5	1111.11	467.78	318.29	148.15
2001.15	34.72	1208	92.78	323.35	312.8	2826.3	1.94	1285.71	446.94	312	135.76
2037.15	32.22	1215.52	88.33	327.04	293.74	2862.3	2.22	1125	440.56	305.17	125.63
2073.15	31.94	1217.39	71.67	337.21	278.98	2898.3	1.67	1333.33	437.22	296.7	116.59
2109.15	30.83	1198.2	71.39	319.07	266.39	2934.3	1.94	1000	428.61	291.64	108.23
2145.15	29.17	1219.05	77.22	302.16	256.15	2970.3	1.39	1600	417.22	284.95	100.96
2181.15	23.61	1200	85.83	323.62	250.36	2996.33	1.87	666.67	437.11	280.63	93.52
2217.15	19.72	1225.35	112.22	321.78	246.27	3004.36	1.87	666.67	437.11	280.63	93.52
2253.15	21.39	1194.81	90.83	327.22	244.18						

FIGURE 4.3-1

RELAP5 Containment Model



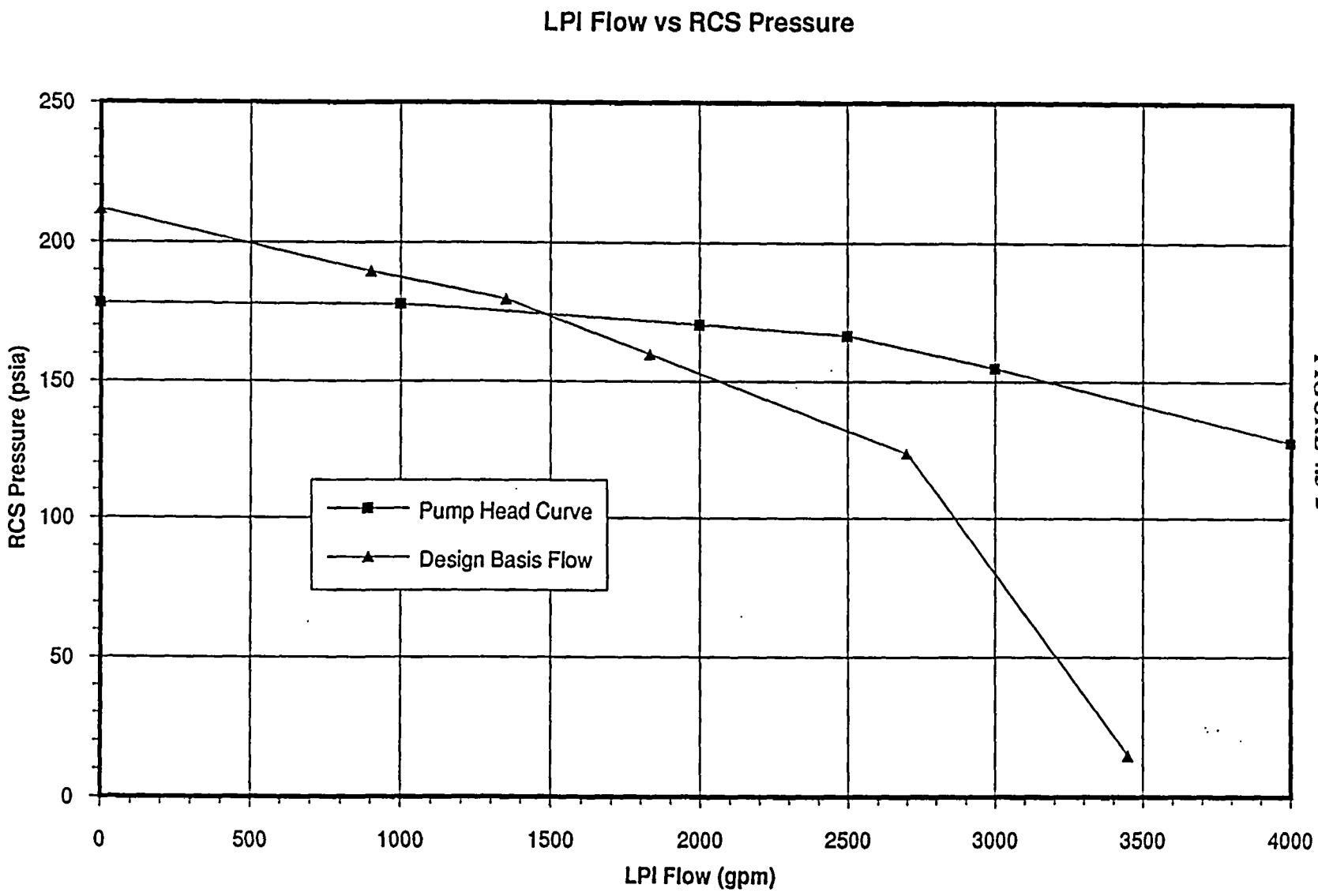
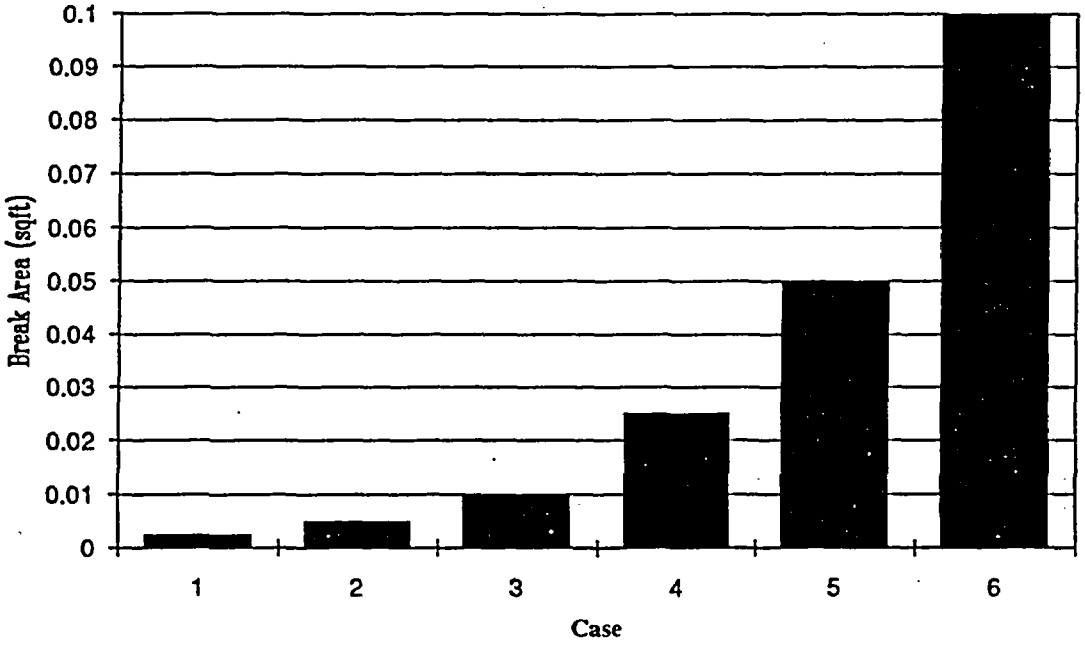

LPI Flow vs RCS Pressure
RCS Pressure (psia)
250
200
150
100
50
0
0
500
1000
1500
2000
2500
3000
3500
4000
LPI Flow (gpm)
Pump Head Curve
Design Basis Flow

FIGURE 4.3-2

FIGURE 4.4-1

Spectrum of Small Break Sizes Analyzed



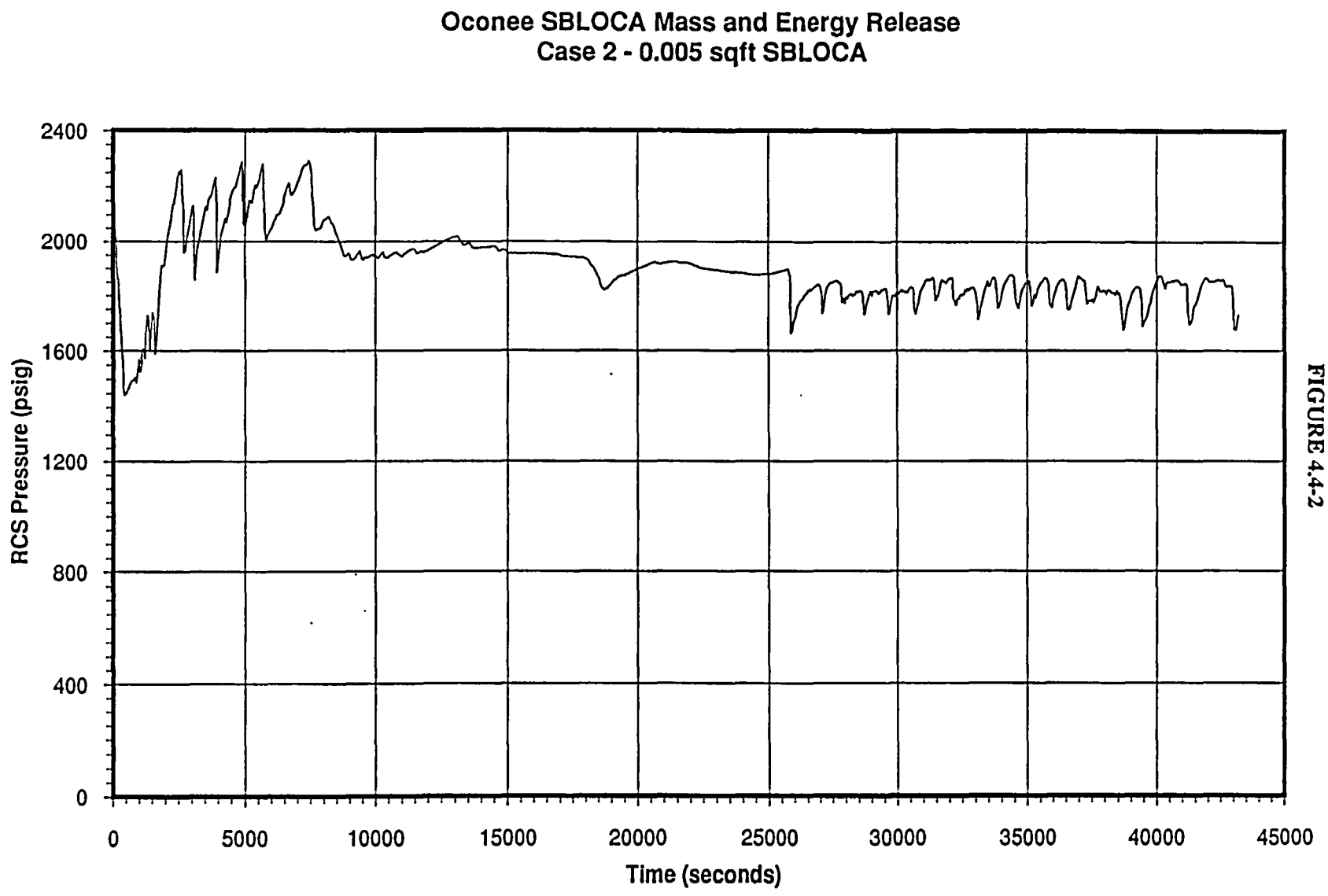

Oconee SBLOCA Mass and Energy Release
Case 2 - 0.005 sqft SBLOCA
RCS Pressure (psig)
2400
2000
1600
1200
800
400
0
0
5000
10000
15000
20000
25000
30000
35000
40000
45000
Time (seconds)

FIGURE 4.4-2

Oconee SBLOCA Mass and Energy Release
Case 2 - 0.005 sqft SBLOCA

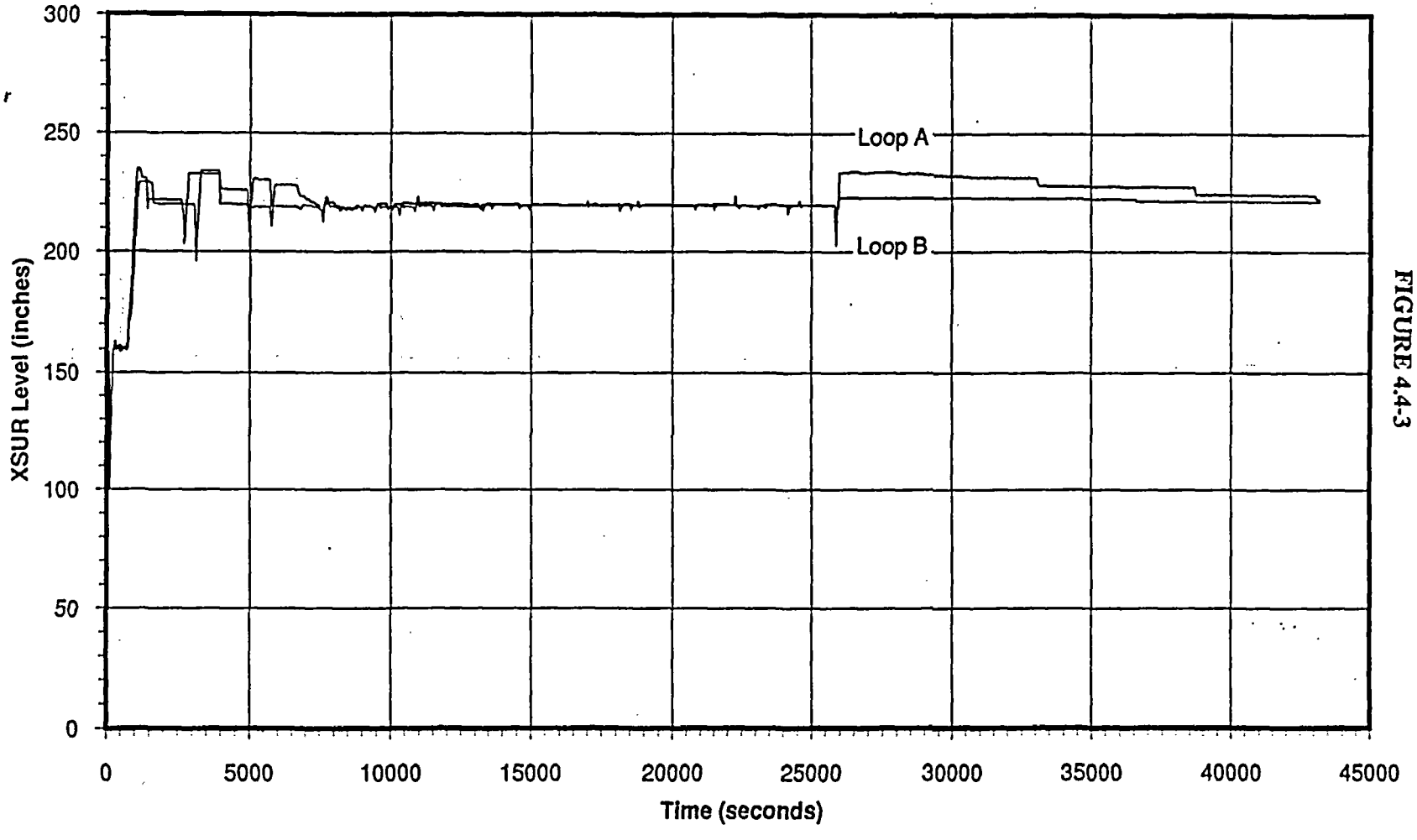


FIGURE 4.4-3

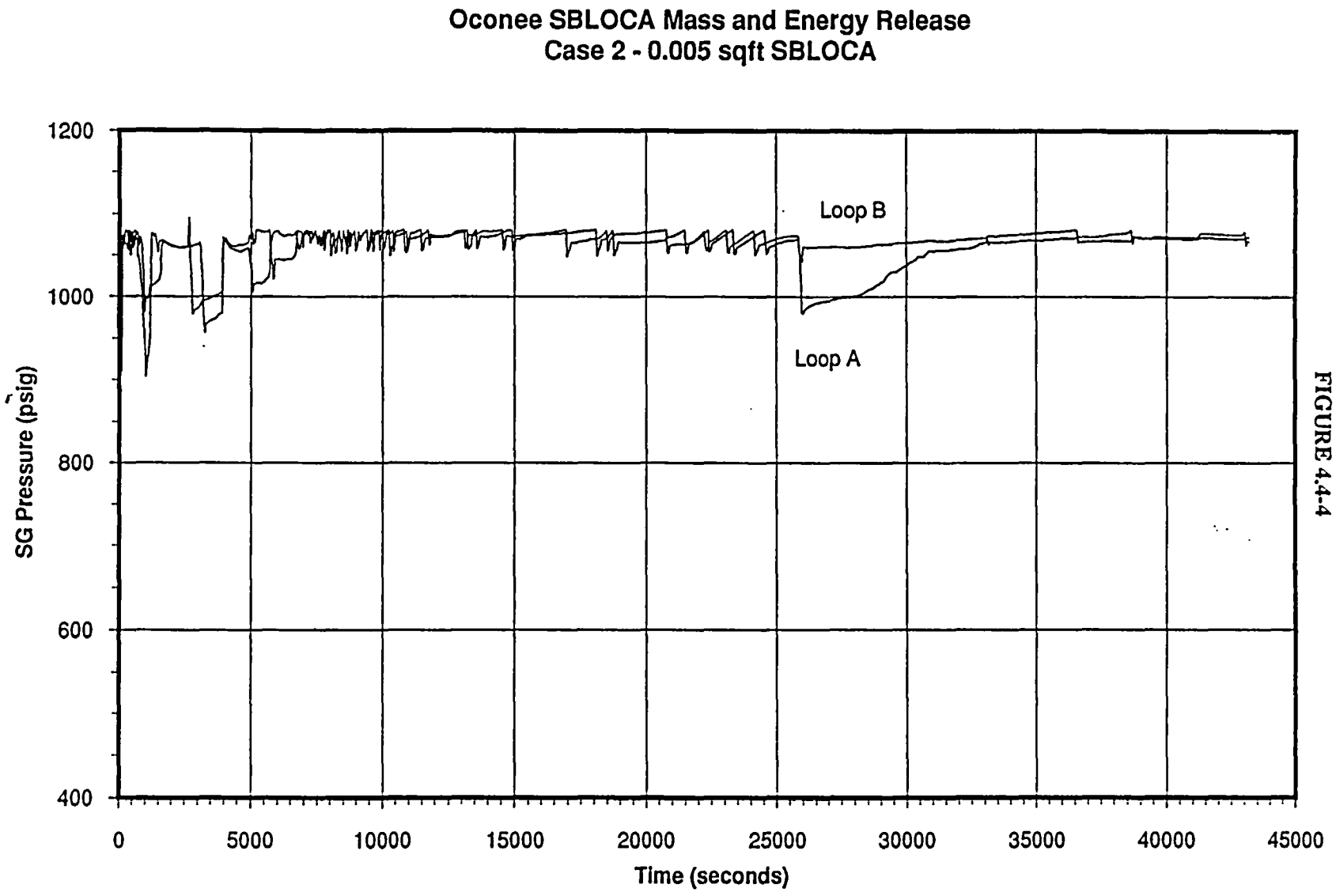

Oconee SBLOCA Mass and Energy Release
Case 2 - 0.005 sqft SBLOCA
SG Pressure (psig)
1200
1000
800
600
400
0
5000
10000
15000
20000
25000
30000
35000
40000
45000
Time (seconds)
Loop B
Loop A

FIGURE 4.4-4

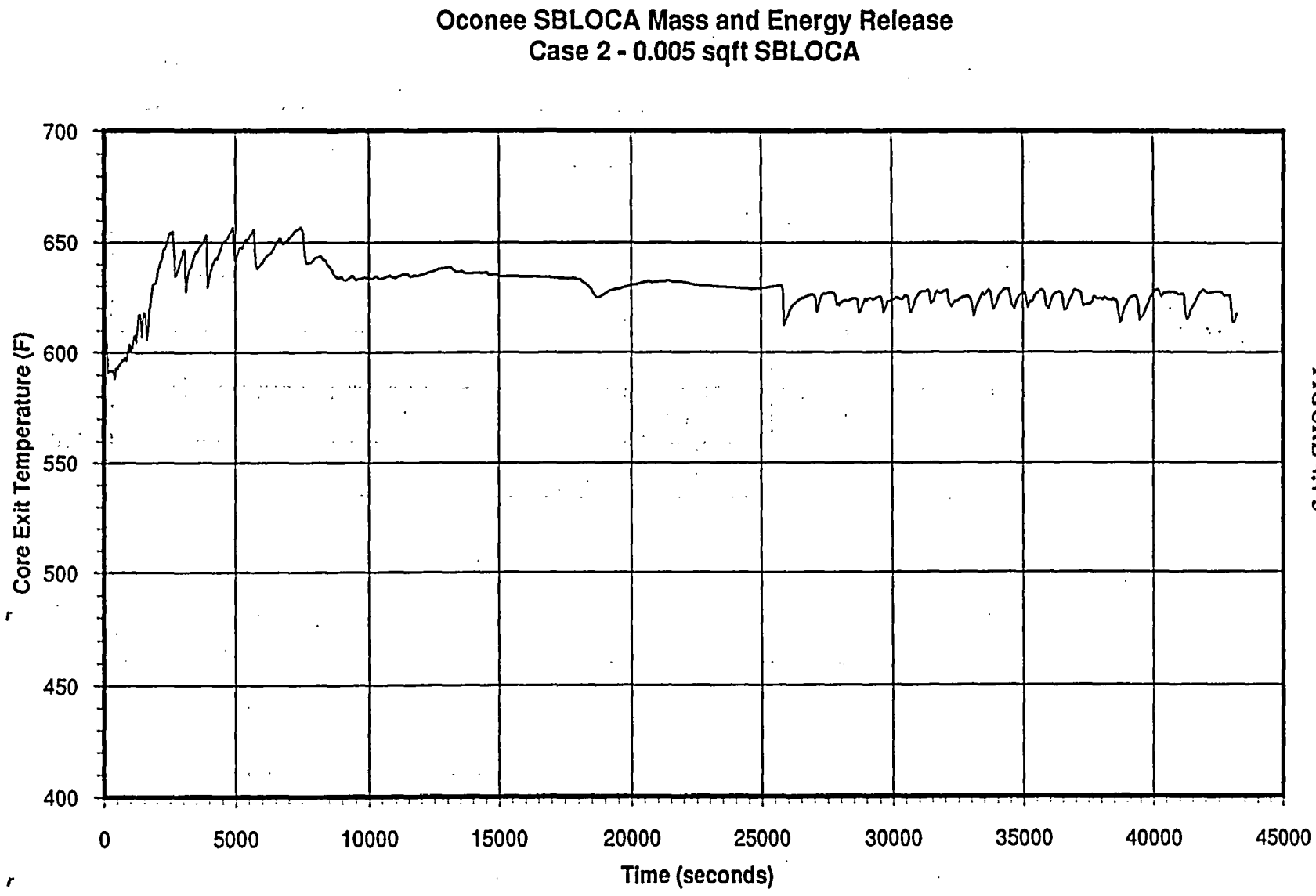

Oconee SBLOCA Mass and Energy Release
Case 2 - 0.005 sqft SBLOCA
Core Exit Temperature (F)
700
650
600
550
500
450
400
0
5000
10000
15000
20000
25000
30000
35000
40000
45000
Time (seconds)

FIGURE 4.4-5

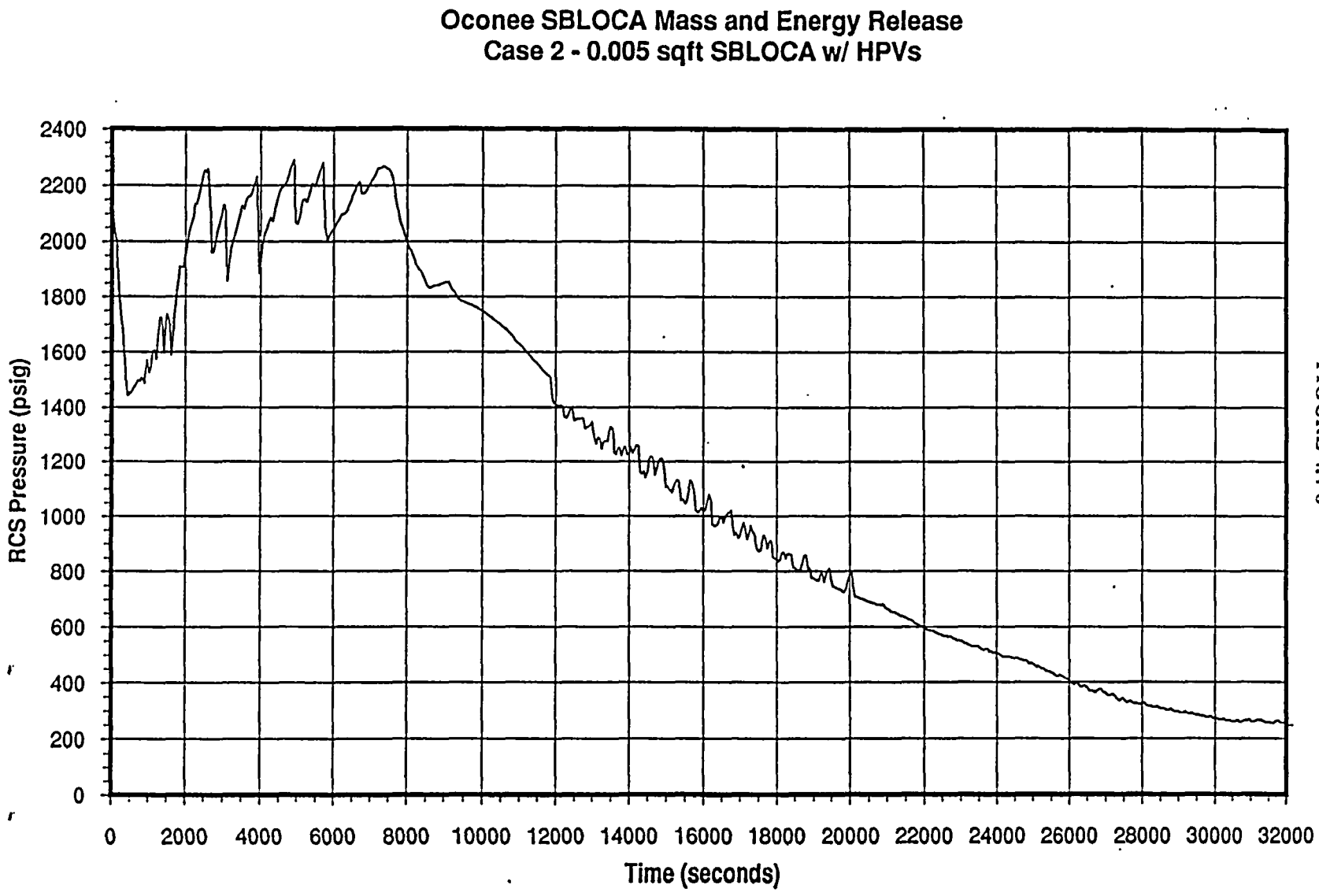

Oconee SBLOCA Mass and Energy Release
Case 2 - 0.005 sqft SBLOCA w/ HPVs
RCS Pressure (psig)
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
2000
4000
6000
8000
10000
12000
14000
16000
18000
20000
22000
24000
26000
28000
30000
32000
Time (seconds)

FIGURE 4.4-6

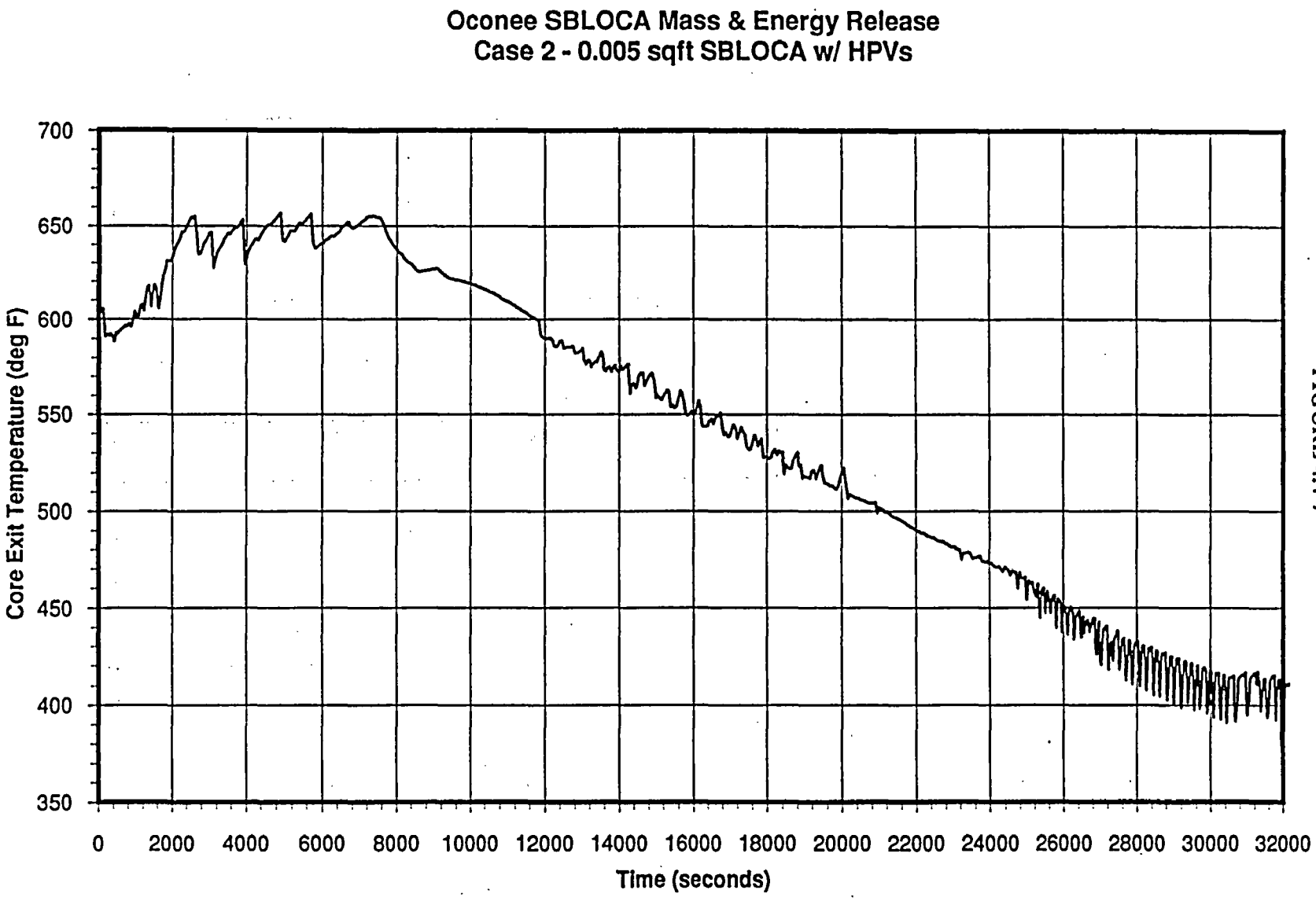

Oconee SBLOCA Mass & Energy Release
Case 2 - 0.005 sqft SBLOCA w/ HPVs
Core Exit Temperature (deg F)
700
650
600
550
500
450
400
350
0
2000
4000
6000
8000
10000
12000
14000
16000
18000
20000
22000
24000
26000
28000
30000
32000
Time (seconds)

FIGURE 4.4-7

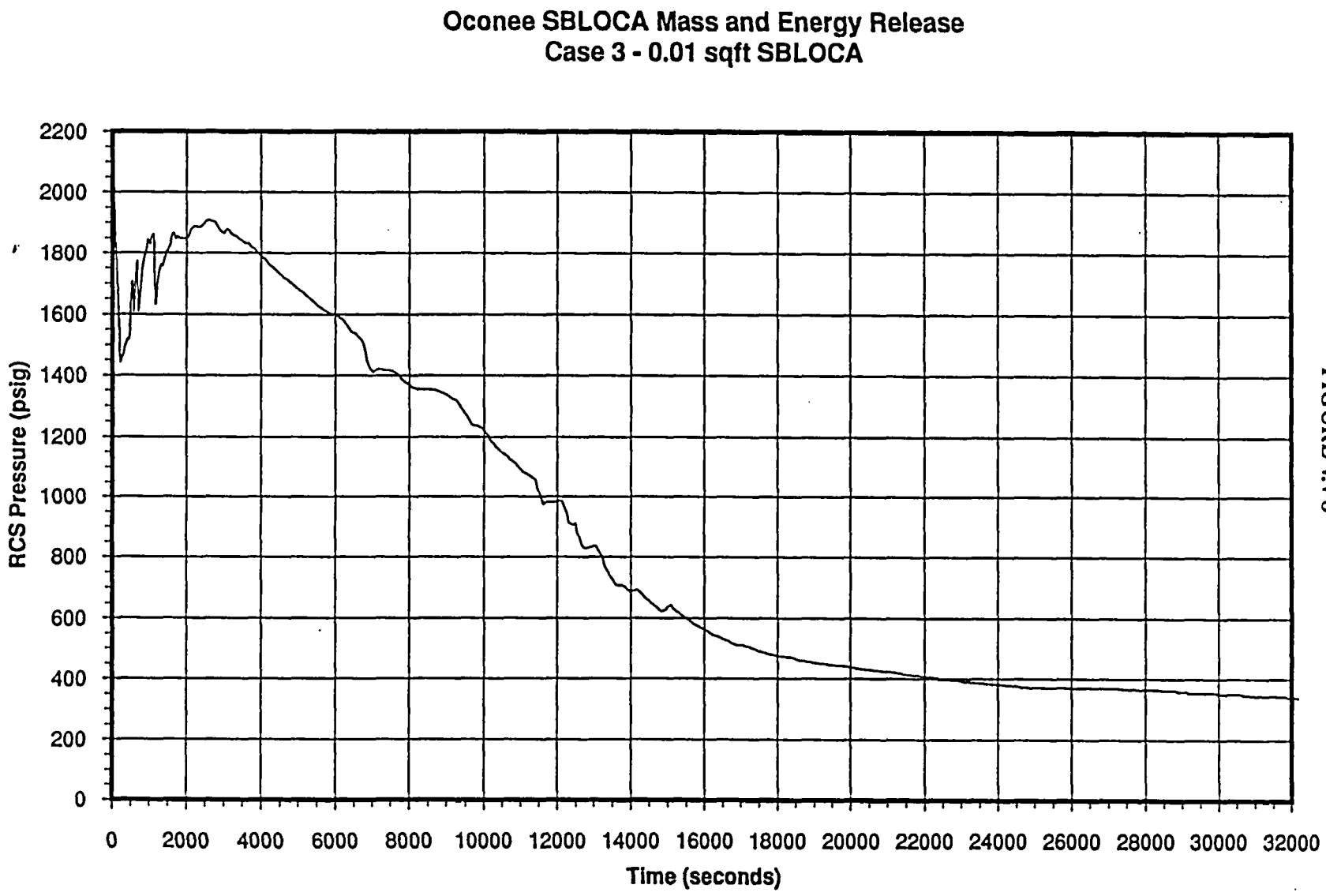

Oconee SBLOCA Mass and Energy Release
Case 3 - 0.01 sqft SBLOCA
RCS Pressure (psig)
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
2000
4000
6000
8000
10000
12000
14000
16000
18000
20000
22000
24000
26000
28000
30000
32000
Time (seconds)

FIGURE 4.4-8

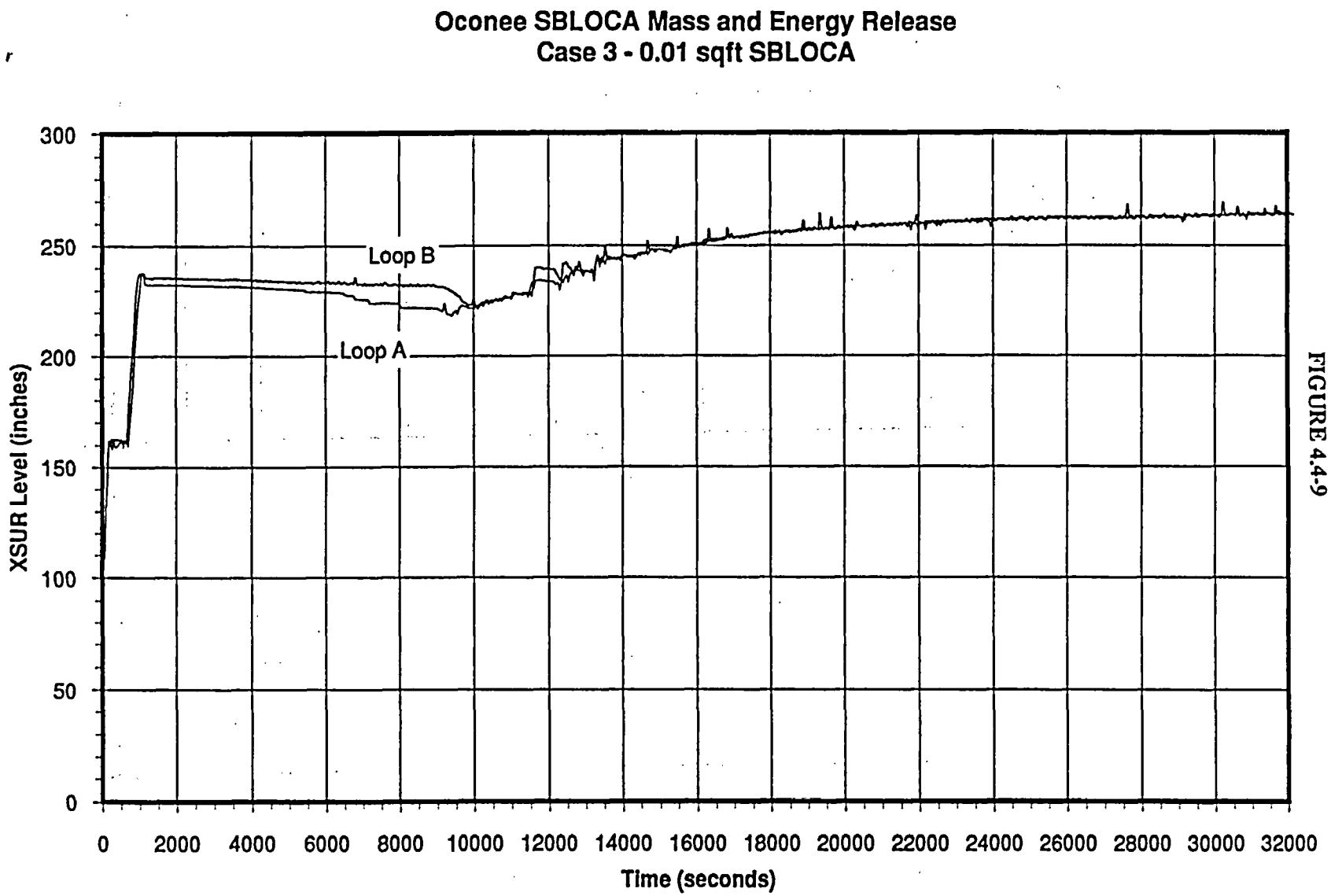

Oconee SBLOCA Mass and Energy Release
Case 3 - 0.01 sqft SBLOCA
Loop B
Loop A
XSUR Level (inches)
300
250
200
150
100
50
0
0
2000
4000
6000
8000
10000
12000
14000
16000
18000
20000
22000
24000
26000
28000
30000
32000
Time (seconds)

FIGURE 4.4-9

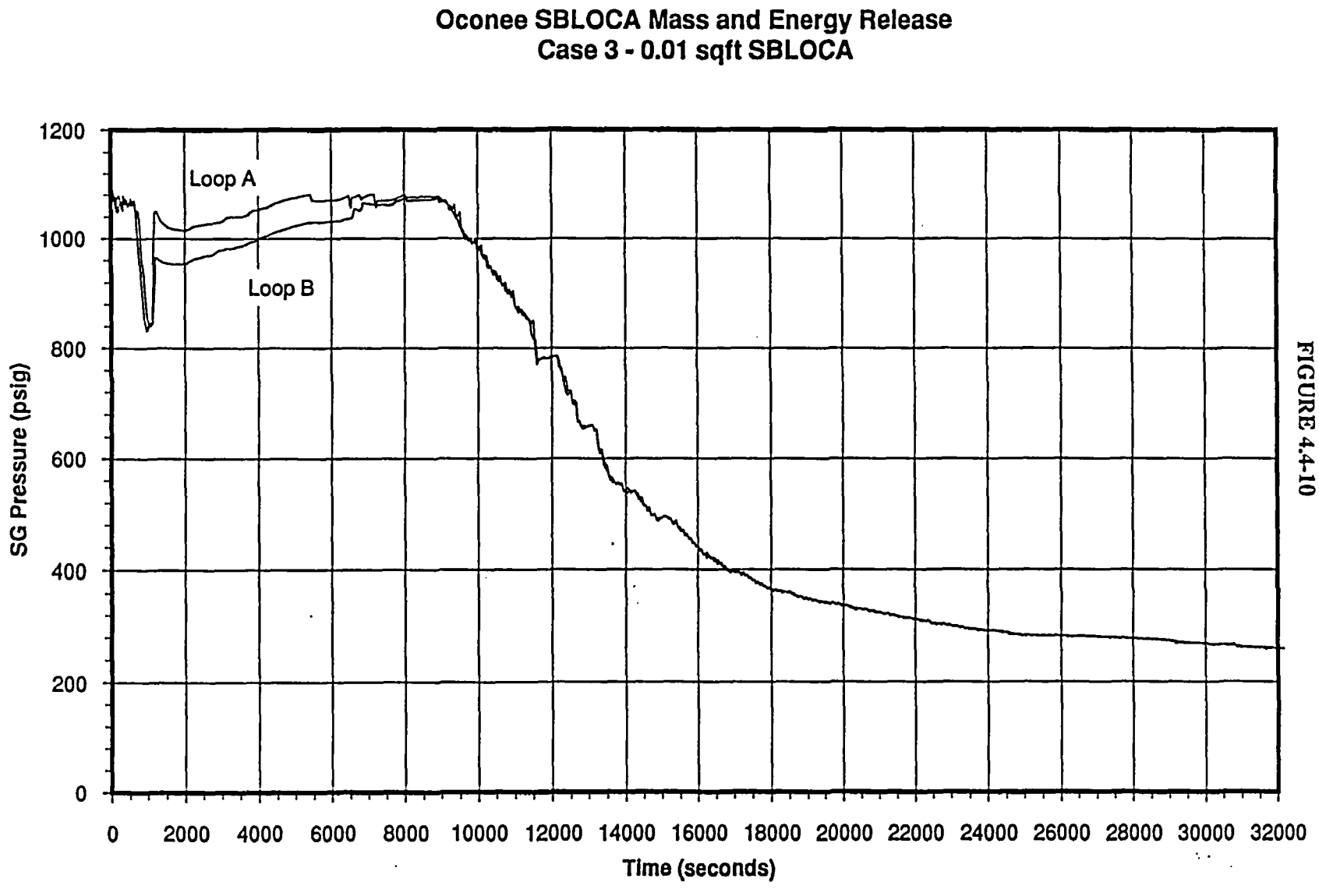

Oconee SBLOCA Mass and Energy Release
Case 3 - 0.01 sqft SBLOCA
Loop A
Loop B
SG Pressure (psig)
1200
1000
800
600
400
200
0
0
2000
4000
6000
8000
10000
12000
14000
16000
18000
20000
22000
24000
26000
28000
30000
32000
Time (seconds)

FIGURE 4.4-10

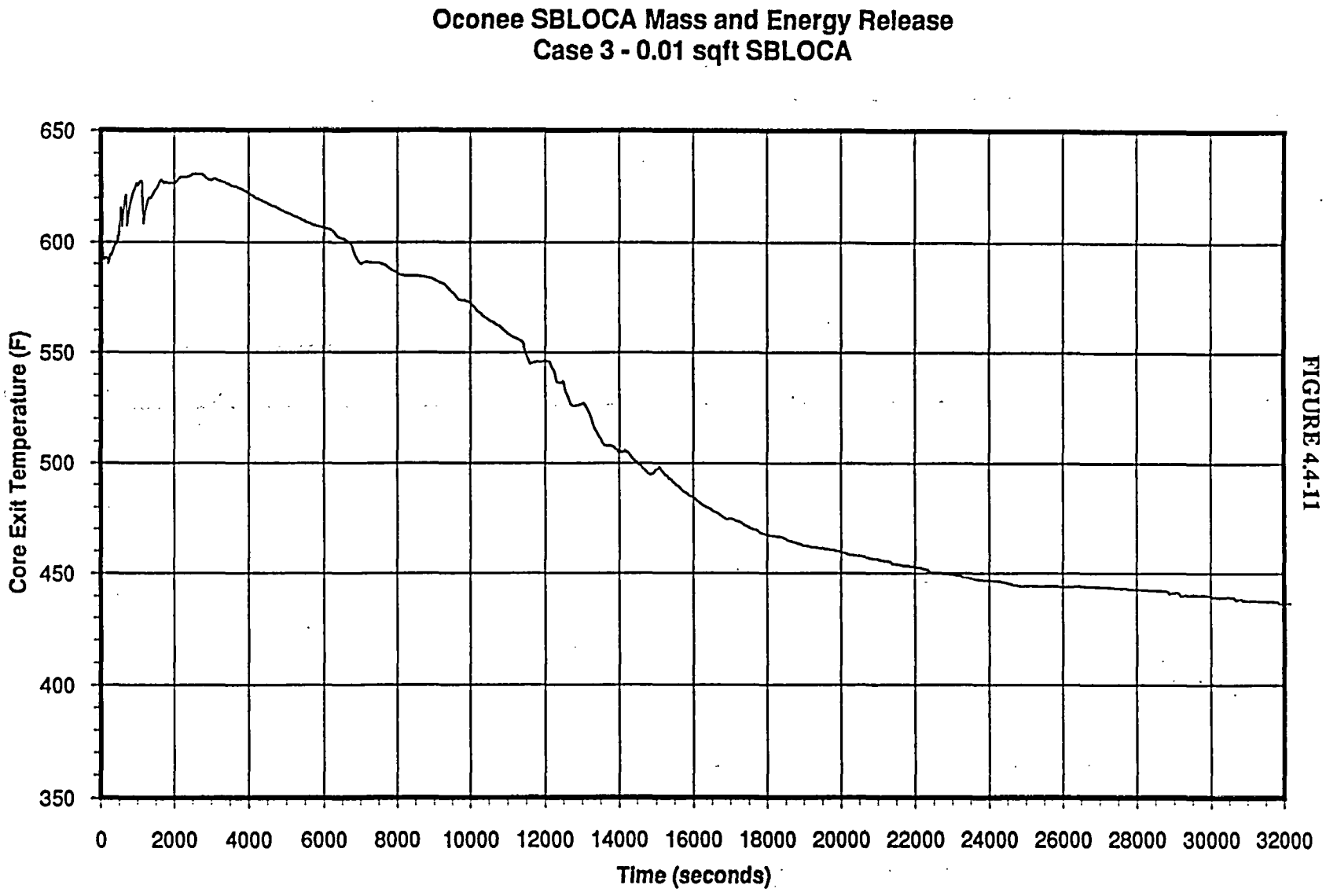

Oconee SBLOCA Mass and Energy Release
Case 3 - 0.01 sqft SBLOCA
Core Exit Temperature (F)
650
600
550
500
450
400
350
0
2000
4000
6000
8000
10000
12000
14000
16000
18000
20000
22000
24000
26000
28000
30000
32000
Time (seconds)

FIGURE 4.4-11

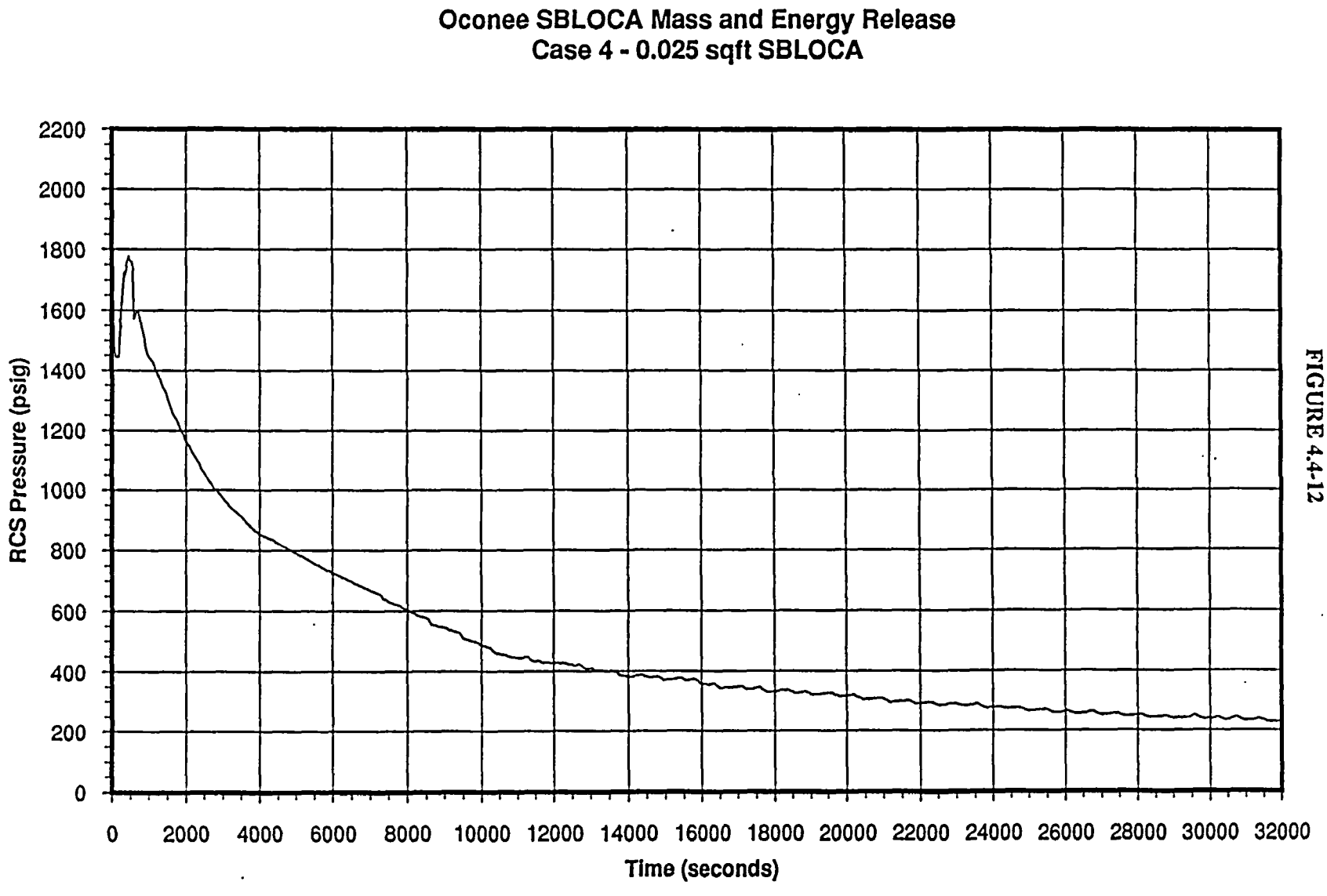

Oconee SBLOCA Mass and Energy Release
Case 4 - 0.025 sqft SBLOCA
RCS Pressure (psig)
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
2000
4000
6000
8000
10000
12000
14000
16000
18000
20000
22000
24000
26000
28000
30000
32000
Time (seconds)

FIGURE 4.4-12

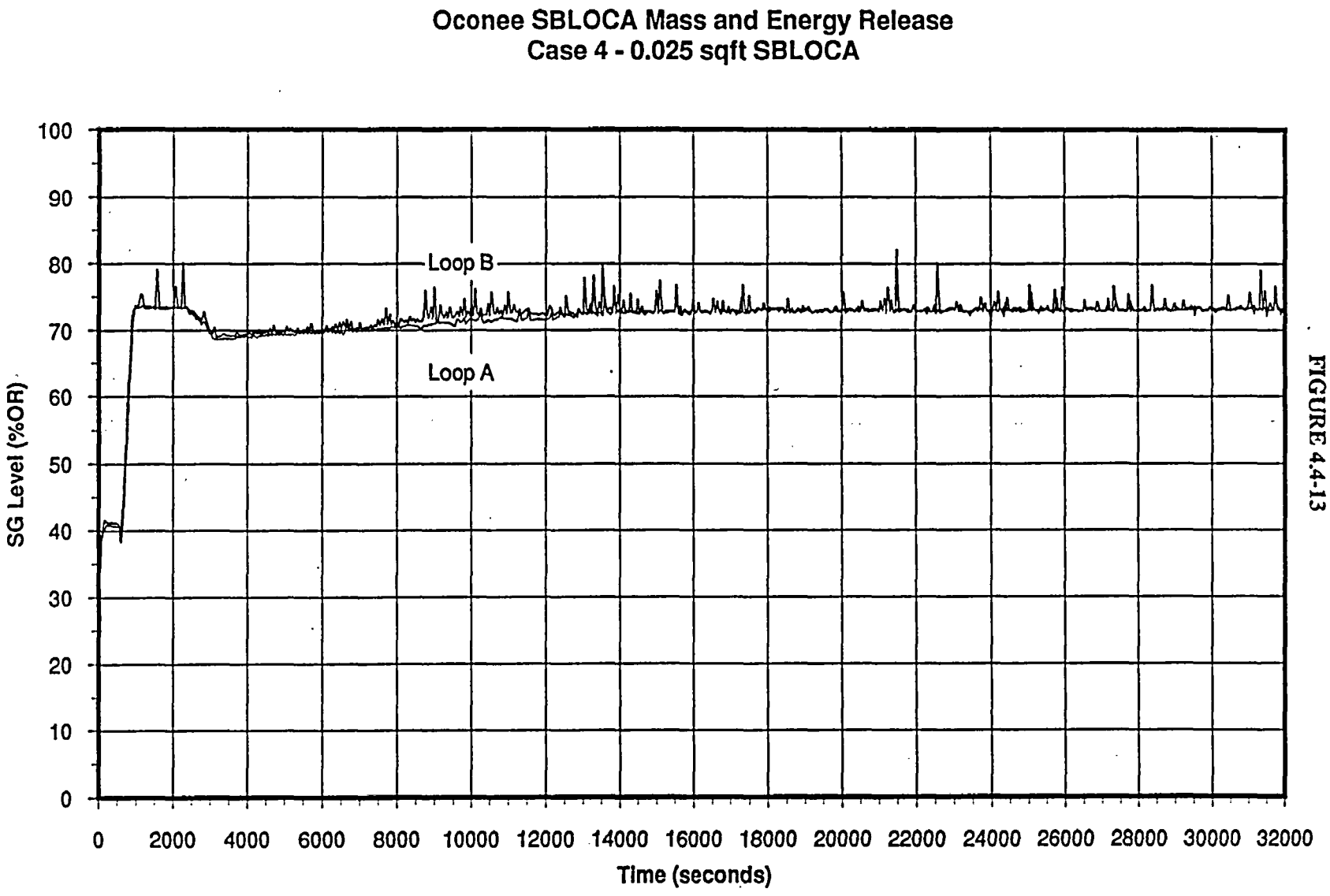

Oconee SBLOCA Mass and Energy Release
Case 4 - 0.025 sqft SBLOCA
Loop B
Loop A
SG Level (%OR)
100
90
80
70
60
50
40
30
20
10
0
0
2000
4000
6000
8000
10000
12000
14000
16000
18000
20000
22000
24000
26000
28000
30000
32000
Time (seconds)

FIGURE 4.4-13

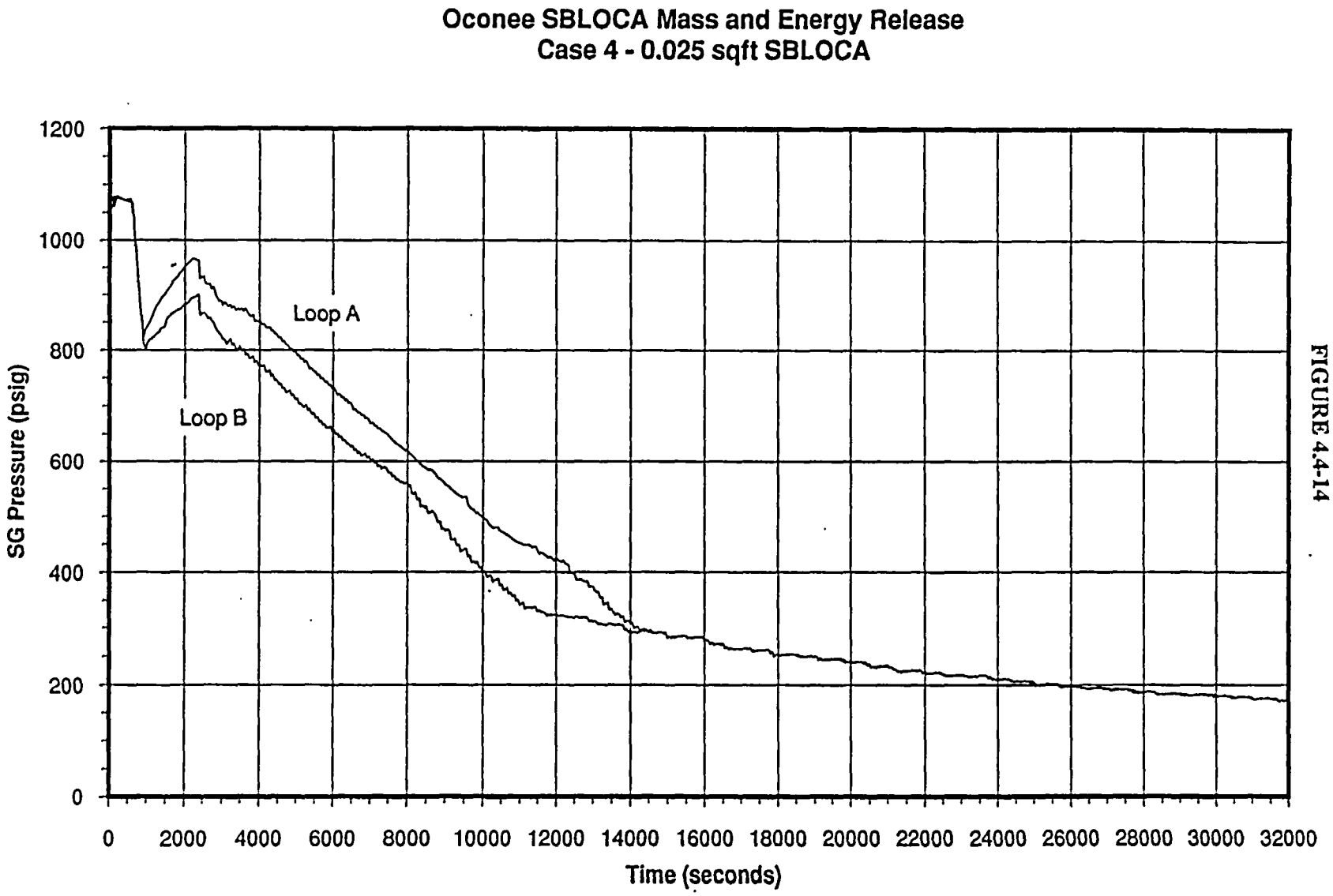

Oconee SBLOCA Mass and Energy Release
Case 4 - 0.025 sqft SBLOCA
SG Pressure (psig)
1200
1000
800
600
400
200
0
Loop A
Loop B
0
2000
4000
6000
8000
10000
12000
14000
16000
18000
20000
22000
24000
26000
28000
30000
32000
Time (seconds)

FIGURE 4.4-14

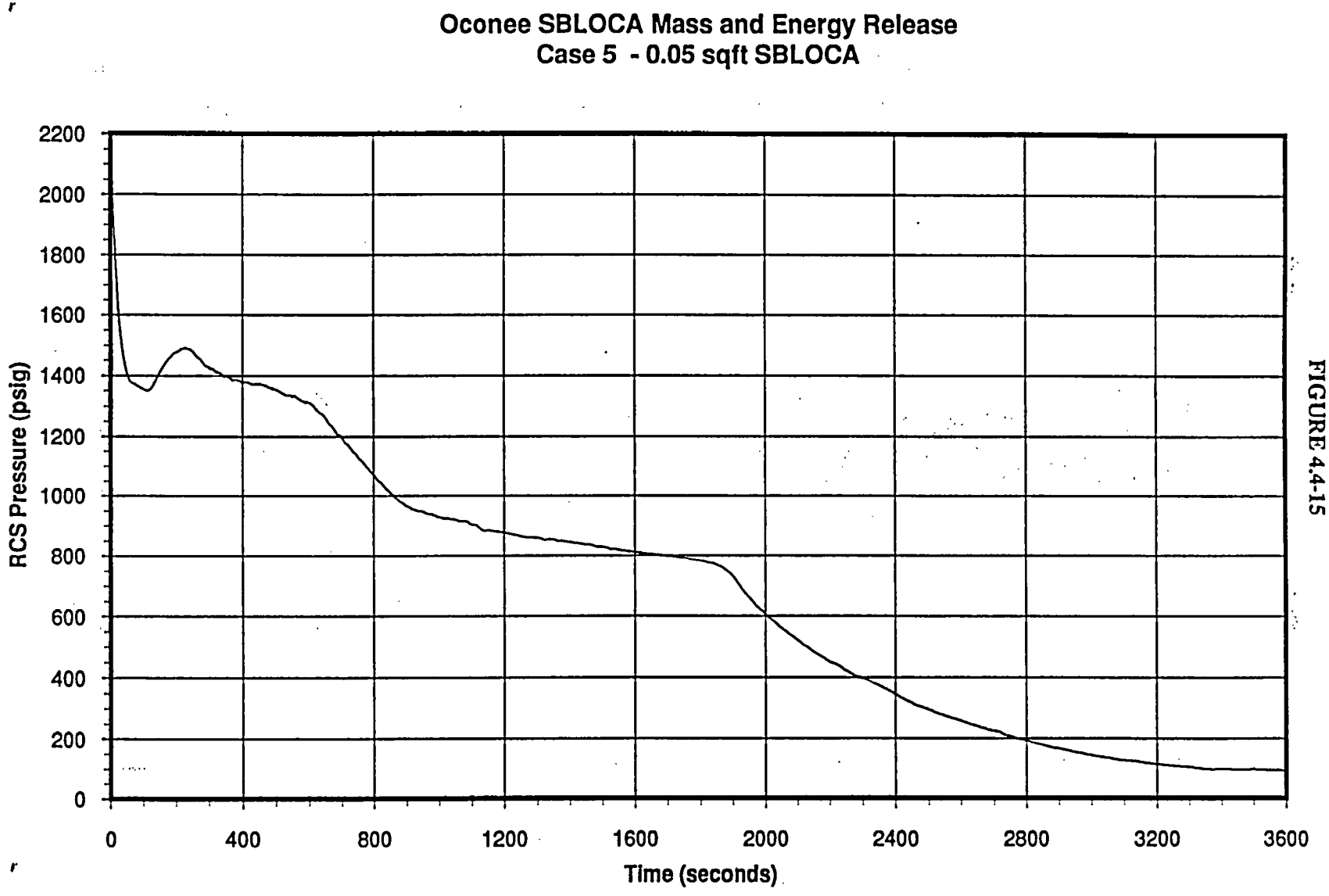

Oconee SBLOCA Mass and Energy Release
Case 5 - 0.05 sqft SBLOCA
RCS Pressure (psig)
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
400
800
1200
1600
2000
2400
2800
3200
3600
Time (seconds)

FIGURE 4.4-15

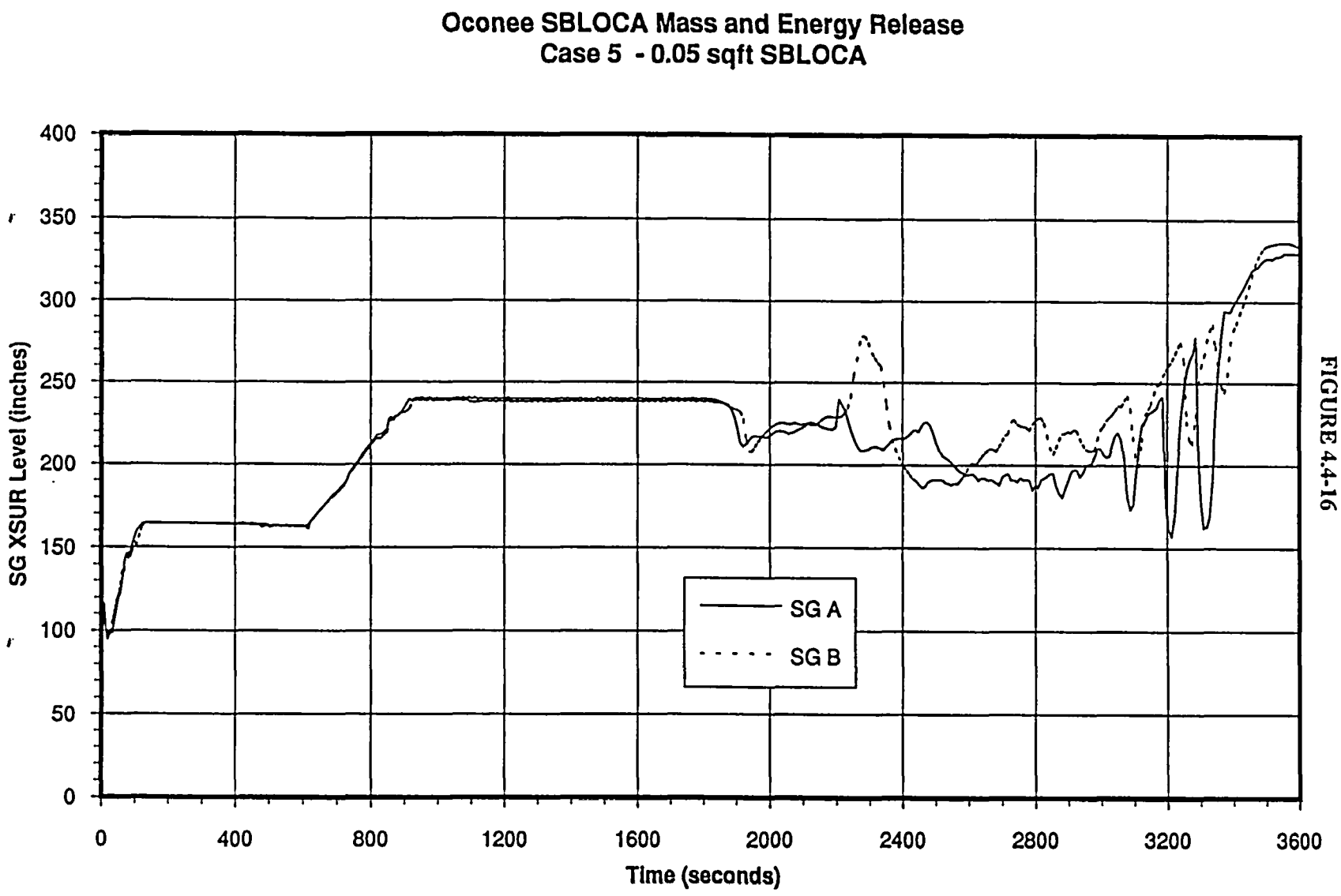

Oconee SBLOCA Mass and Energy Release
Case 5 - 0.05 sqft SBLOCA
SG XSUR Level (inches)
400
350
300
250
200
150
100
50
0
0
400
800
1200
1600
2000
2400
2800
3200
3600
Time (seconds)
SG A
SG B

FIGURE 4.4-16

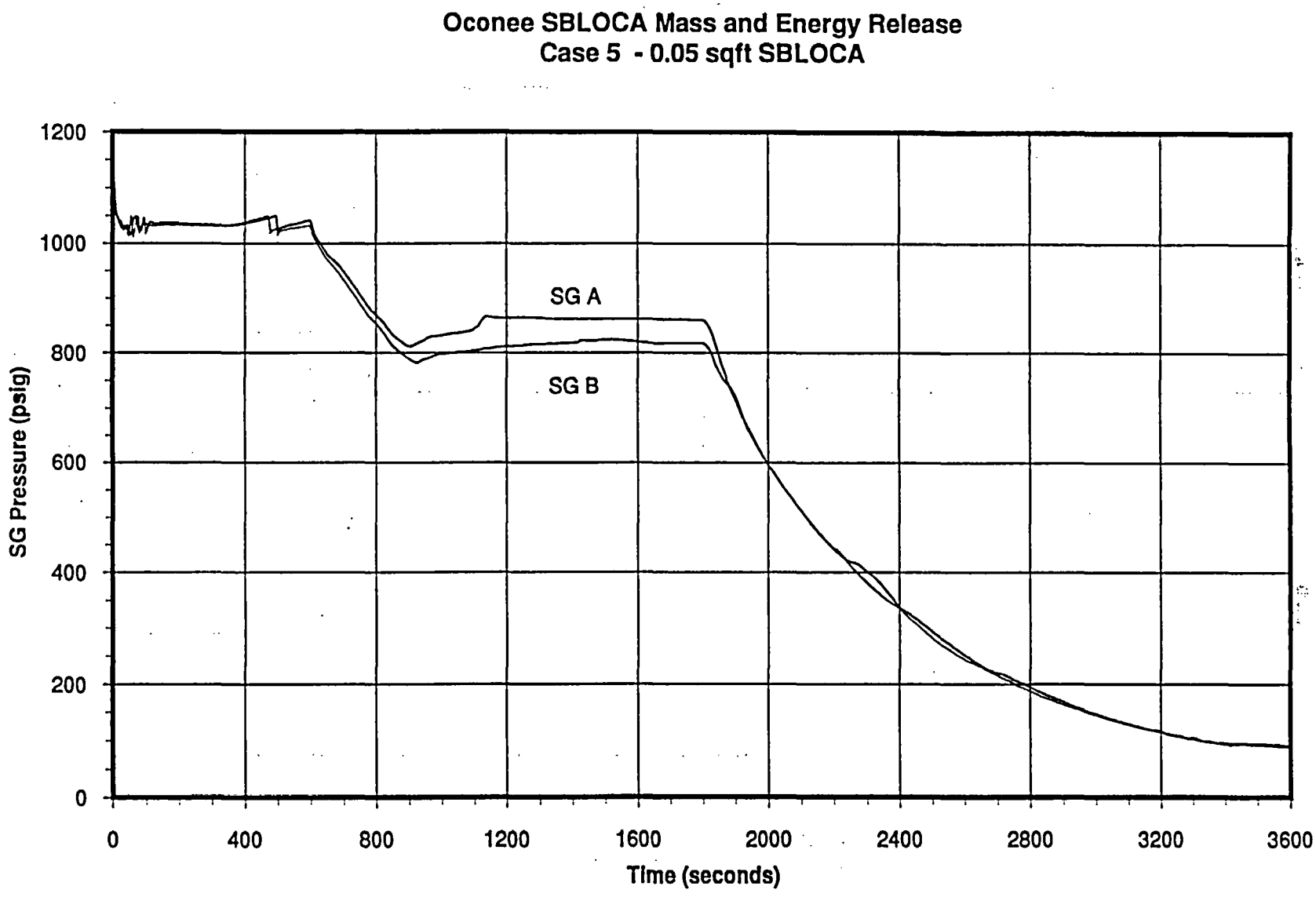

Oconee SBLOCA Mass and Energy Release
Case 5 - 0.05 sqft SBLOCA
SG A
SG B
SG Pressure (psig)
1200
1000
800
600
400
200
0
0
400
800
1200
1600
2000
2400
2800
3200
3600
Time (seconds)

FIGURE 4.4-17

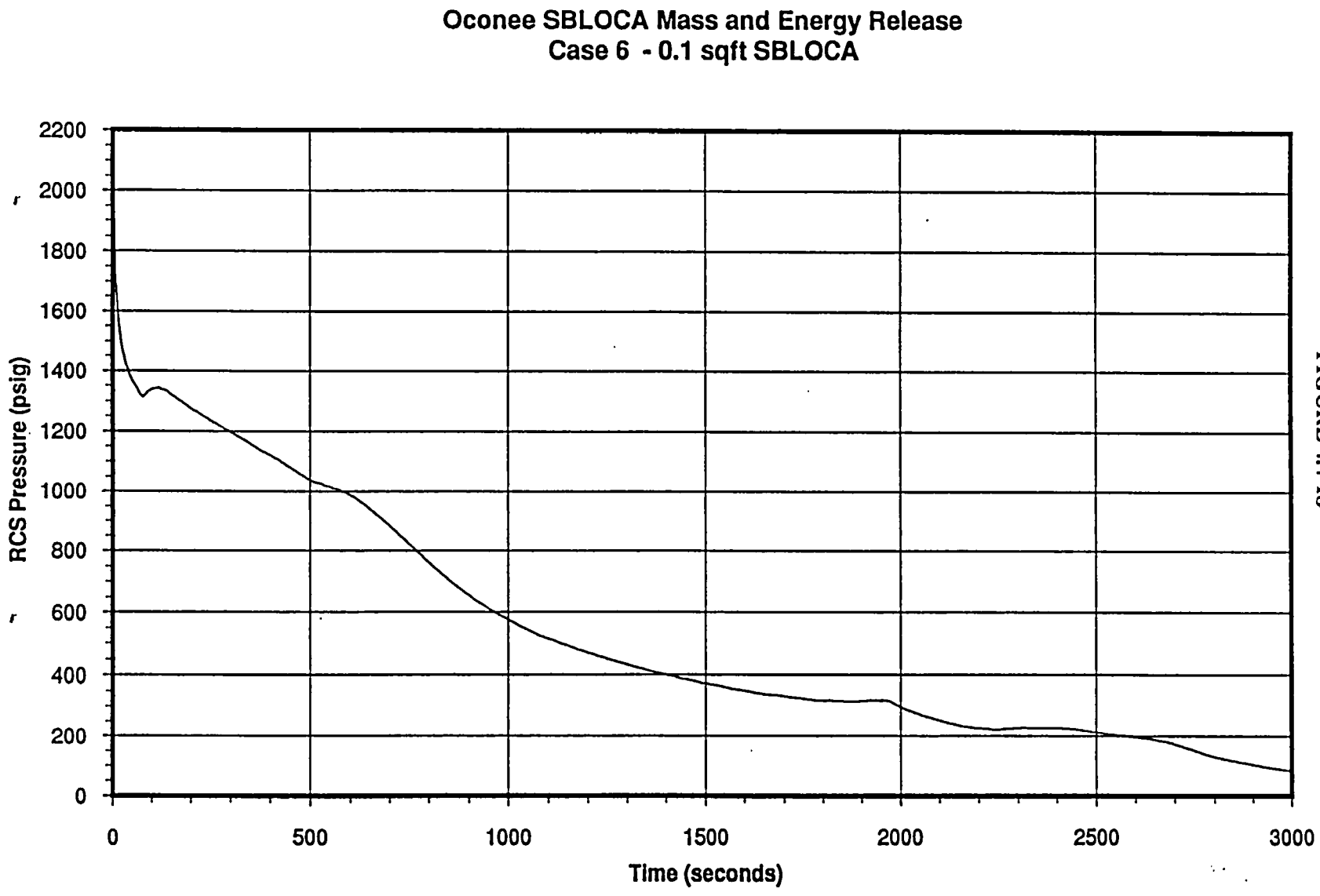

Oconee SBLOCA Mass and Energy Release
Case 6 - 0.1 sqft SBLOCA
RCS Pressure (psig)
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
500
1000
1500
2000
2500
3000
Time (seconds)

FIGURE 4.4-18

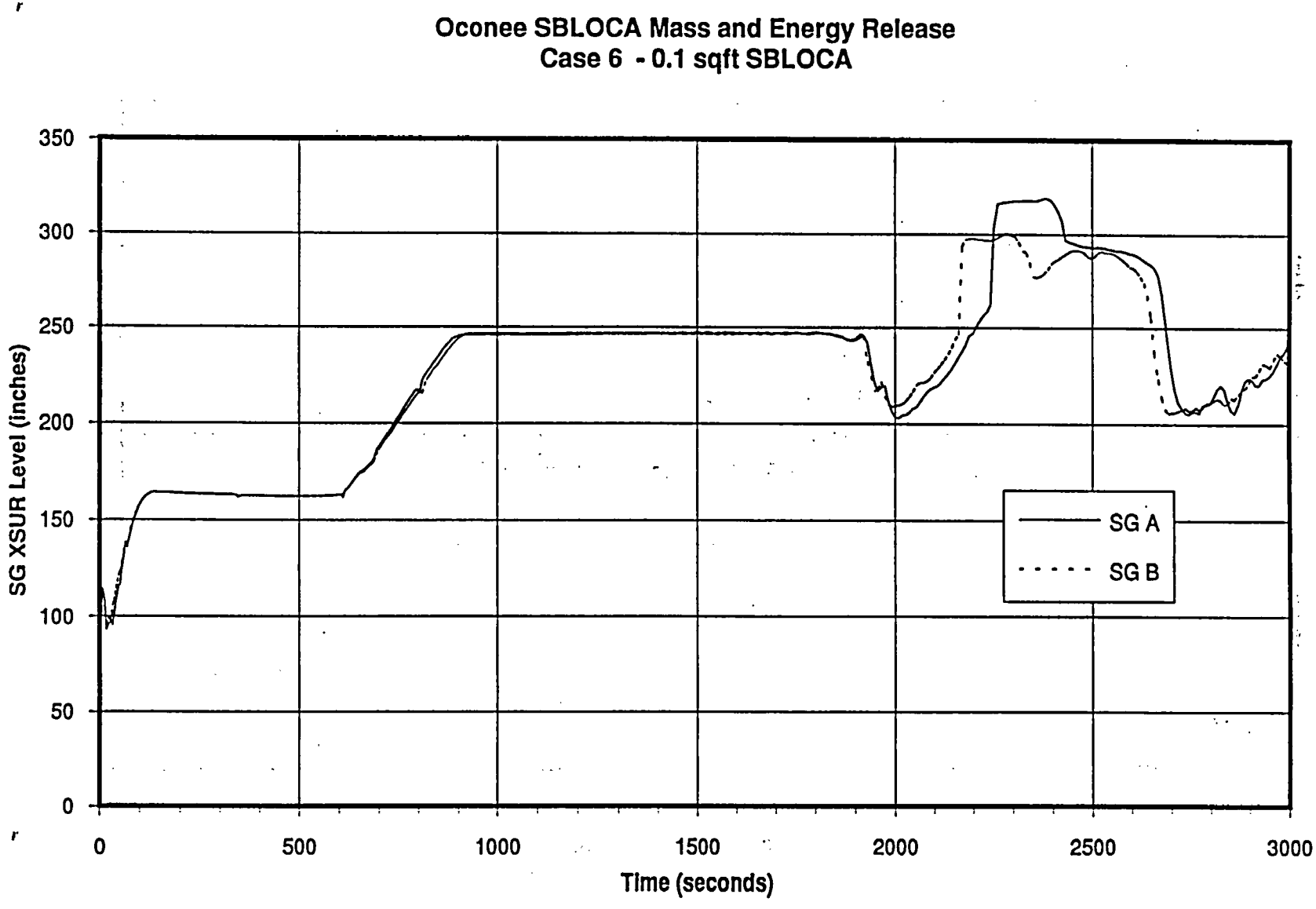

Oconee SBLOCA Mass and Energy Release
Case 6 - 0.1 sqft SBLOCA
SG XSUR Level (inches)
350
300
250
200
150
100
50
0
0
500
1000
1500
2000
2500
3000
Time (seconds)
SG A
SG B

FIGURE 4.4-19

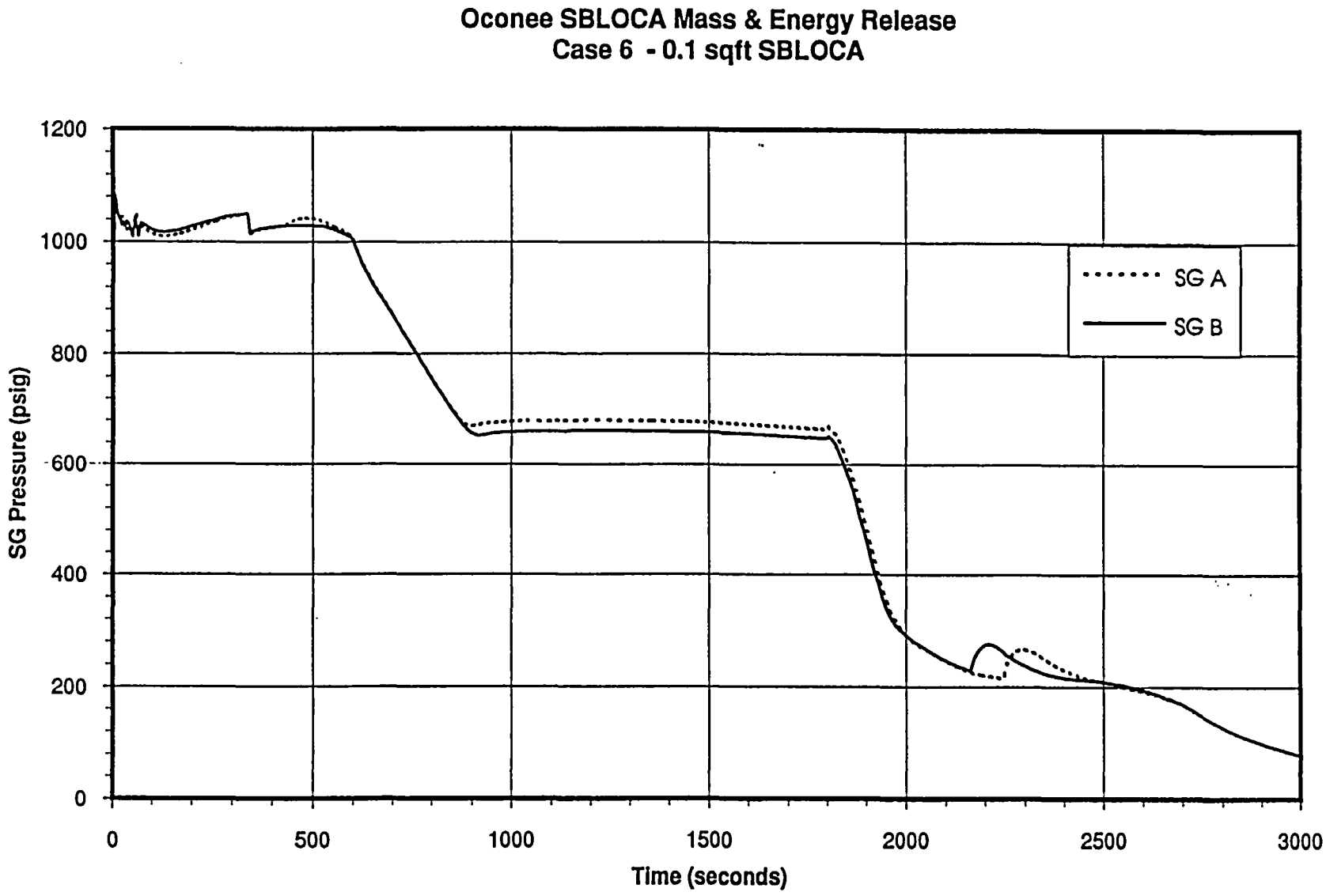

Oconee SBLOCA Mass & Energy Release
Case 6 - 0.1 sqft SBLOCA
SG Pressure (psig)
1200
1000
800
600
400
200
0
0
500
1000
1500
2000
2500
3000
Time (seconds)
SG A
SG B

FIGURE 4.4-20

5.0 STEAM LINE BREAK MASS AND ENERGY RELEASE ANALYSES

5.1 Overview

5.1.1 Description of Steam Line Break Accident

The steam line break transient is described in UFSAR Section 6.2 (Reference 5-1). The steam release arising from a break in a main steam line would result in an initial increase in steam flow, with a subsequent decrease during the accident as steam generator pressure falls. The energy removal from the Reactor Coolant System (RCS) causes a reduction in RCS temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive control rod is assumed stuck in its fully withdrawn position after reactor trip, the core may become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the High Pressure Injection (HPI) System and the core flood tanks (CFTs). Break flow continues until feedwater is isolated to the affected steam generator and any remaining inventory is boiled off.

5.1.2 Acceptance Criteria

The acceptance criteria for the steam line break mass and energy release and containment response analyses are that the containment pressure will not exceed the design pressure of 59 psig, and that safety-related equipment inside containment will survive the harsh environment resulting from the transient (equipment qualification).

5.1.3 Analytical Approach

The steam line break transient requires a limiting set of physics parameters to be determined for use as initial and boundary conditions. These parameters are input to the Oconee RETRAN-02 (Reference 5-3) model for the system thermal-hydraulic analysis. The RETRAN-02 analysis generates the mass and energy releases that are input to the Oconee FATHOMS (Reference 5-4) model to calculate the containment pressure response to the steam line break accident.

5.2 System Thermal-Hydraulic Analysis

5.2.1 Modifications to Base Plant Model

Renodalization of Reactor Vessel

Condensate/Feedwater System Model

A Condensate/Feedwater System model is added to the RETRAN base deck to accurately predict the feedwater flow boundary condition during the steam line break transient. The Condensate/Feedwater System model (Figure 5.2-2) contains fill tables to simulate the condensate booster pumps and the D heater drain pumps. Homologous pump curves are included to accurately model the main feedwater pumps. Non-conducting heat exchangers are used to model all of the feedwater heaters.

5.2.2 Break Modeling

The full cross-sectional area of the 34" main steam line is 6.3 ft^2. Thus, the double-ended guillotine break of the 34" main steam line results in a total initial break flow area of 12.6 ft^2. This analysis uses the Moody critical flow model. The steam line nodalization is given in Figure 5.2-3.

5.3 Initial Conditions

The initial conditions for the steam line break mass and energy release analyses are chosen to maximize the stored energy in both the primary and secondary systems. Maximizing the stored energy will ensure that conservative mass and energy release boundary conditions are provided for the containment analyses. ANSI/ANS-56.4-1983 (Reference 5-6) provides criteria and guidance for selecting the initial values for the principal system parameters (e.g. pressures, temperatures, etc.) and the type of uncertainties which should be applied to these parameters. Every attempt is made to adhere to the criteria presented in this standard. If the initial value chosen does not conform to the standard, then justification is provided for the deviation. Otherwise, the initial conditions selected meet or exceed the requirements of the standard.

The initial conditions for the principal parameters chosen for this analysis are presented in this section in the same order as they appear in Reference 5-6, Section 3.3.2. A summary of the initial conditions is shown in Table 5.3-1.

Time of Life

See Section 5.4, Core Stored Energy

Power Level

An uncertainty of 2% is bounding for thermal power based upon plant performance data. This uncertainty is applied to the 2568 MWth rated power for Oconee to yield the initial power level assumed in these analyses (102% rated power or 2619 MWth). The high initial power level maximizes the initial steam generator inventory and feedwater flow rate, which results in conservative initial conditions on the secondary side. In addition, the higher power level maximizes energy generation prior to reactor trip and results in higher decay heat. Thus, the initial power level of 102% FP is conservative.

A steam line break accident from hot zero power (HZP) is not analyzed, since feedwater is aligned through the startup feedwater control valves, which results in a much lower feedwater flow rate than at full power. Thus, full power is the initial condition that provides a limiting mass and energy release.

RCS Temperature

The Integrated Control System (ICS) controls the average coolant temperature at a constant value of 579°F whenever power is greater than 15%. Because analyses indicated that a 2°F uncertainty allowance is bounding for the average coolant temperature, an initial value of 581°F is assumed.

RCS Pressure

An allowance of 30 psi has been determined to bound the pressure indication uncertainty. Since the nominal RCS pressure is controlled at 2155 psig, an initial pressure of 2185 psig is assumed.

SG Pressure

Turbine header pressure is normally controlled at a nominal pressure of 885 psig, yielding a pressure of approximately 910 psig at the steam generator outlet. With an uncertainty of 40 psi associated with the steam generator pressure indication, a pressure of 950 psig would appear to be the desired initial condition. However, this value is not used. Instead, the nominal full power steam generator pressure of 910 psig is used in this analysis. This initial condition is necessary to obtain the required heat transfer at full power.

Pressurizer Level

The pressurizer level control system normally maintains pressurizer level at the 220 inch setpoint. An allowance of 25 inches has been determined to bound the pressurizer level uncertainty. Thus, an initial pressurizer level of 245 inches is assumed.

Steam Generator Operating Level

Main feedwater flow in a once-through steam generator is not controlled based upon steam generator level during power operation. Instead, feedwater flow is controlled by the ICS and is based primarily upon the desired power output. As a consequence, level is not a true indication of the steam generator inventory because phenomena such as tube fouling and two-phase pressure drop variations will affect the level indications. A more reliable measure of steam generator inventory is the total mass of liquid and steam present. The nominal full power steady-state mass is [ ] lbm per steam generator. Thus, an upper bound on SG total inventory of [ ] lbm is assumed. This inventory includes an allowance of [ ] lbm to account for the impact of steam generator fouling on the overall primary-to-secondary heat transfer coefficient.

Core Parameters

See Section 5.4, Core Stored Energy and Fission Heat

Control Rod Position

The most reactive control rod is assumed to be in the fully withdrawn position.

Boron Concentration

See Section 5.4, Fission Heat

5.4 Boundary Conditions

This section presents the boundary conditions assumed for the steam line break mass and energy release analyses. The boundary conditions are selected in accordance with ANSI/ANS-56.4-1983 (Reference 5-6). A summary of the boundary conditions is shown in Table 5.4-1.

RCS Water and Metal

The volume of the RCS increases due to thermal expansion of the pressure boundary when the system is heated to operating conditions. This thermal expansion has been accounted for in the Oconee RETRAN model by increasing the calculated volume of the RCS piping by a nominal 1%, which is representative of the expansion resulting from a temperature change from cold to operating conditions. In addition, zero tube plugging is assumed, which maximizes RCS inventory.

Heat conductors are included in the RETRAN model for all structural metal which contacts the primary coolant. These heat conductors are assumed to be in equilibrium with the surrounding coolant and maintain an initial temperature distribution which is constant across each conductor to maximize the initial stored energy.

Steam Generator Secondary Water and Metal

The physical volume of the secondary system has not been increased to account for thermal expansion. However, the main feedwater flow boundary condition and initial steam generator inventory are chosen to ensure that secondary inventory is conservatively represented. Heat conductors are included in the RETRAN model for structural metal which contacts the secondary coolant. These heat conductors are assumed to be in equilibrium with the surrounding coolant and maintain an initial temperature distribution which is constant across each structure to maximize the initial stored energy. Zero tube plugging is assumed, which maximizes the primary-to-secondary heat transfer area.

Core Stored Energy

Stored energy in the core is a function of the initial average fuel temperature, which is a function of burnup. The maximum fuel temperature occurs near beginning of cycle. Core decay heat, on the other hand, is at a maximum at end of cycle. More importantly, the moderator temperature coefficient is most negative at end of cycle, which will result in a higher return to power. Therefore, end of cycle conditions are chosen to maximize the energy addition to the primary system. A high initial fuel temperature is used in the analysis to maximize the stored energy in the fuel. An upper bound on the end of cycle average fuel temperature is 1072°F.

Fission Heat

The RETRAN point kinetics model is used for the system thermal-hydraulic analysis. The particular option employed uses one prompt neutron group, six delayed neutron groups, eleven delayed gamma emitters, plus U-239 and Np-239. The point kinetics model is adequate for this application since special modeling has been included to ensure a conservative core response. The power distributions used in the system analysis are determined to be conservative as discussed below.

A low effective delayed neutron fraction and prompt neutron lifetime are chosen to maximize the reactivity addition rate.

In the RETRAN model, reactivity insertion curves are specified to model reactivity feedback. A table of reactivity as a function of moderator density is input to account for moderator reactivity effects, and a table of reactivity as a function of fuel temperature is input to model Doppler reactivity effects. The reactivity tables assume that moderator and Doppler reactivities are zero at hot full power. The consequences of a main steam line break accident are more severe at end of cycle due to the more negative moderator temperature coefficient. The most negative Doppler and moderator curves are used in the analysis. A most negative Doppler curve maximizes the reactivity addition during the cooldown. However, this curve will also result in the largest negative feedback during the return to power. Since the boron injected by the CFTs and HPI stops the power increase, and not the negative reactivity due to Doppler feedback, it is conservative to assume a most negative Doppler curve.

The control rods are inserted when the reactor trips. For this analysis, a top-peaked scram curve and a lower bound on the rod insertion time are assumed. This assumption is slightly non-conservative, as it minimizes post-trip energy addition into the RCS. However, sensitivity studies show that this assumption has a negligible impact on the mass and energy release analysis.

An initial boron concentration of 0 ppm is consistent with the assumption of end of cycle conditions. The negative reactivity inserted by boration is modeled by averaging the six core node boron concentrations to obtain an overall core boron concentration. This concentration is multiplied by a boron worth to determine the reactivity.

In addition to the physics data, a differential boron worth is needed to model the reactivity addition from the boron injected by the HPI System and core flood tanks. A high differential boron worth, in ppm / % $\Delta k/k$ is conservative in that it will minimize the negative reactivity added by these systems. A value of 120 ppm / % $\Delta k/k$ is selected as an upper bound on the differential boron worth over the range of temperatures expected during the steam line break accident.

Decay of Actinides and Fission Product Decay

The ANSI/ANS-5.1-1979 decay heat values for end-of-life conditions are used in this analysis.

Main Steam Lines

The turbine stop valves close on reactor trip, thereby isolating the unaffected steam generator from the affected steam generator. The turbine stop valve closure time assumed in the analysis is based on the closure time of $\leq$ 1.0 seconds per Technical Specification 3.7.2 (Reference 5-7).

Main Feedwater System

The model of the Condensate and Feedwater Systems is used to accurately predict the feedwater flow rate during the steam line break accident. Incorporation of this model in the RETRAN base deck allows the Feedwater System to be modeled from the discharge of the condensate booster pumps to the inlet of the steam generators. The condensate booster pumps and D heater drain pumps are modeled by fill tables of flow rate as a function of pump discharge pressure. The Condensate System model is initialized with two condensate booster pumps, two D heater drain pumps, and two main feedwater pumps operating. The third condensate booster pump is assumed to be in standby. This pump will automatically start if the suction pressure to the main feedwater pumps drops below 360 psig. This analysis does not take credit for low suction pressure trips of the hotwell, condensate booster, or main feedwater pumps. In addition, the steam generator high level trip of the main feedwater pumps is not credited in the analysis. However, the condensate booster pumps and D heater drain pumps will be tripped if their inventory sources are depleted.

The feedwater flow boundary condition is determined by the ICS modeling assumptions and the steam generator pressure transient. Thus, the feedwater flow rate is explicitly calculated during the accident based on the pump curves and the transient pressure response. When offsite power

is lost, it is assumed that the main feedwater pumps, condensate booster pumps, and D heater drain pumps trip due to the loss of power. A 10 second coastdown of the pumps is assumed.

The main feedwater heaters preheat the condensate from about 80°F to around 455°F during power operation. Extraction steam is used to preheat the condensate. An initial main feedwater temperature of 458.3°F is used. This temperature results in a secondary side heat removal equal to the power generated in the primary side, based on the assumed main feedwater flow rate. No uncertainty allowances are applied to main feedwater temperature.

When a reactor trip occurs, the steam source for feedwater heating is no longer available and the condensate temperature decreases. However, nominal full power heat addition by the main feedwater heaters is conservatively maintained until after main feedwater isolation.

Emergency Feedwater System

All three emergency feedwater (EFW) pumps automatically start upon a loss of both main feedwater pumps or a low steam generator level. Once actuated, the EFW flow rate is controlled to maintain steam generator level at 30 inches if the reactor coolant pumps are running or 240 inches if the reactor coolant pumps are tripped. The turbine-driven EFW pump and one motor-driven EFW pump supply flow to the steam generator with the broken steam line. The other motor-driven EFW pump supplies flow to the unaffected steam generator. A constant EFW temperature of 120°F is assumed.

RCS Flow Rate

A high RCS flow rate maximizes the primary-to-secondary heat transfer rate. The nominal best-estimate steady-state flow rate is approximately 112.5% of design flow (88,000 gpm per pump). The uncertainty associated with this value is approximately 2%. Therefore, an RCS flow rate of 115% design flow is assumed.

ECCS

The HPI System is modeled using a fill table with one HPI pump injecting through one train. The assumption of one HPI pump through one train is conservative as it results in a slightly higher return to power which will maximize the energy in the RCS that is available to be transferred to the secondary side. The HPI actuation setpoint is assumed to be 1480 psig. This

setpoint is based on a nominal plant setpoint of 1600 psig decreased by 120 psig to account for instrument uncertainty.

During the steam line break accident, the suction source for the HPI pump is the BWST. Both core flood tanks are independently modeled in this analysis. Purge volumes from the outlet of the borated water storage tank and the core flood tanks to the inlet of the RCS are separately calculated. These piping volumes are assumed to be initially at a concentration of 0 ppm. Borated water is assumed to reach the RCS only after an amount of unborated water equal to the purge volume has been injected.

The boron concentration of the injection water from HPI is assumed to be the minimum allowed boron concentration of 2210 ppm. An upper bound on BWST temperature of 120°F is assumed for this analysis. The boron concentration in the core flood tanks is assumed to be at the Technical Specification minimum of 1835 ppm. An upper bound on CFT temperature of 120°F is assumed.

The boron transport model used is described in Section VII.2.5 of Reference 5-3. The boron is assumed to be soluble in the transport medium and to have no direct effect on the fluid equation. The basic equation computes the time rate of change of boron mass in a control volume from the net inflow from connected volumes plus the net generation within that volume.

Single Failure

The failure of a 4160V switchgear is chosen as the single failure for this analysis. The failure of one of the three available switchgear will result in the loss of one train of LPI and one train of HPI.

Reactor Protective System Trips

A late reactor trip is conservative in that it maximizes energy generation in the primary system. Thus, the variable low pressure trip and low RCS pressure trip setpoints are adjusted by an uncertainty allowance to ensure that a late reactor trip occurs.

The nominal Technical Specification variable low pressure trip setpoint is defined by the following equation:

$$\text{Setpoint} = 11.14\ T_{hot} - 4696\ \text{psig}$$

A reactor trip signal is generated when the indicated RCS pressure decreases below this setpoint. The nominal trip setpoint is adjusted by an uncertainty allowance to the following equation:

$$\text{Setpoint} = 11.14\ T_{hot} - 4736\ \text{psig}$$

An upper bound on the delay time for this trip of 0.7 seconds is used. In addition, a conservative lag on the hot leg temperature indication of 5.04 seconds is used.

The low RCS pressure trip is also modeled in the analysis. The plant is calibrated to a trip setpoint of 1810 psig. The uncertainty allowance for this trip function is 30 psi. In order to delay reactor trip, the trip setpoint is decreased by 30 psi to a trip setpoint of 1780 psig. An upper bound on the delay time for this trip of 0.5 seconds is used.

ICS

Following the steam line break, the combined effect of decreasing turbine header pressure and T-ave will result in an increase in reactor demand to the high limit of about 103% FP. Since a reactor trip will occur within the first five seconds of the accident and the control rods move at a constant speed of 30 inches per minute, the rods would at most move about 3%. Thus, it is reasonable to make the simplifying assumption that the control rods are in manual control.

The steam line break causes a rapid decrease in steam generator pressure. Thus, the ICS will attempt to close the turbine control valves in order to restore turbine header pressure to its setpoint. Since the steam flow to the turbine is maximized if the turbine control valves remain open, it is conservative to assume that turbine control is in manual.

Since a reactor trip occurs within the first few seconds of the accident, changes in feedwater control over this time period will have a negligible impact on the accident. Following reactor trip, the BTU limits rapidly drive feedwater demand to zero. The decrease in the demand signal will cause the main feedwater and startup control valves to rapidly close and the main feedwater pump turbine speed to run back. If the ICS is functioning properly, the MFW control valves will not open again until steam generator level drops below the low level control setpoint of 25 inches.

Normally, steam generator level is controlled at 25 inches following reactor trip. A RETRAN control system used to simulate the low level controller. This control system is a simple proportional plus integral controller that adjusts the valve position to maintain level at the setpoint.

Loss of Offsite Power

If offsite power is lost, the hotwell pumps and condensate booster pumps will trip, resulting in a trip of the main feedwater pumps. Thus, in order to maximize the feedwater addition to the affected steam generator, it is conservative to assume that offsite power remains available. In addition, more primary-to-secondary heat transfer occurs when the reactor coolant pumps are running. However, during a loss of offsite power, power is not available to the main feedwater block valves to isolate main feedwater. In addition, emergency feedwater will actuate on the loss of the main feedwater pumps. Thus, a loss of offsite power coincident with reactor trip is also analyzed.

Containment Backpressure

A higher containment backpressure will result in a decrease in break flow. Thus, a conservatively low containment backpressure is used in this analysis.

Liquid Carryout and Superheat

The containment temperature response following a steam line break is heavily influenced by the amount of superheat of the steam exiting through the break. On the other hand, containment pressurization depends almost entirely on how much of the break fluid enters the primary containment atmosphere as steam. With a pure steam blowdown, all of the break flow enters the primary containment atmosphere. With a two-phase blowdown, part of the liquid in the break flow flashes in the primary containment, and the steam is added to the atmosphere while the remaining liquid falls to the sump without contributing significantly to primary containment pressurization. Therefore, liquid carryout and superheat effects are conservatively accounted for in determining the limiting mass and energy release.

From a peak containment pressure perspective, it is conservative to assume that all of the break flow is released in the form of steam. Assuming that primary-to-secondary heat transfer is

unchanged, an equivalent break flow that is all in the form of steam is calculated. However, by assuming that all break flow is steam, the amount of flow leaving through the break is reduced.

5.5 Results

The steam line break transient is analyzed both with offsite power maintained and with offsite power lost coincident with reactor trip. The event sequences for the two cases are presented in Tables 5.5-1 and 5.5-2. Figures 5.5-1 through 5.5-7 correspond to the case with offsite power maintained, and Figures 5.5-8 through 5.5-14 correspond to the case with offsite power lost. Tables 5.5-3 and 5.5-4 present the mass and energy release data for the two cases.
Note that all plots include a 5 second null transient for the initialization of the Condensate and Feedwater System model. However, all operator action times as well as the sequence of events for each case are referenced from the initiation of the steam line break at 5 seconds.

Offsite Power Maintained

This case assumes that the feedwater portion of the ICS is in automatic. Thus, main feedwater pump speed runs back after reactor trip, and the control valves adjust feedwater flow to maintain a minimum level of 75 inches in the steam generator. The single failure is assumed to be the loss of a 4160V switchgear resulting in only one train of HPI. Operator action is credited at 170 seconds to isolate feedwater to the affected steam generator using the main feedwater control valves.

Reactor coolant pressure (Figure 5.5-1) rapidly decreases after the steam line break occurs until the unaffected loop saturates at about 10 seconds. After this time, RCS pressure continues to decrease at a slower rate, eventually leading to HPI and CFT injection. After main feedwater is

isolated to the affected steam generator, primary system temperature begins to increase, which causes RCS pressure to gradually increase.

The cooldown in the unaffected loop is much less severe than in the affected loop (Figure 5.5-2). The cold leg temperature in the unaffected loop increases once the turbine stop valves close. The unaffected loop ΔT remains very small until this loop begins to void and the flow degrades. At about 65 seconds, the ΔT increases with the cold leg being at a higher temperature than the hot leg. This is a result of reverse heat transfer in the unaffected SG and flow reversal in the unaffected loop. A fairly large ΔT exists in the affected loop throughout the simulation. The cold leg temperature begins to stabilize at about 60 seconds as a result of the ICS controlling level in the affected steam generator to 75 inches.

Reactor power (Figure 5.5-3) rapidly decreases after reactor trip as the control rods are inserted into the core. As RCS temperatures decrease, reactivity feedback results in a return to power. Once borated water from HPI and the CFTs reaches the core, reactor power gradually decreases for the remainder of the simulation.

Initially, the steam line break causes steam generator pressure (Figure 5.5-4) to decrease in both generators. Once the turbine stop valves close, the unaffected steam generator repressurizes and opens the turbine bypass valves for a short period of time. After the turbine bypass valves close, the unaffected steam generator gradually depressurizes due to reverse heat transfer and extraction steam loads. The affected steam generator has depressurized to about 60 psig at the end of the simulation. The depressurization of the unaffected steam generator ends following the isolation of feedwater to the affected steam generator.

The ICS controls level in both steam generators to 75 inches until feedwater isolation to the affected generator occurs. Inventory in the affected steam generator (Figure 5.5-5) decreases throughout the transient until the generator is empty. Very little feedwater flow is delivered to the unaffected generator. Thus, inventory in the unaffected generator (shown on the same figure) will also decrease during the transient as inventory is gradually boiled off.

Break flow (Figure 5.5-6) initially decreases as steam generator pressure decreases. After the turbine stop valves close, flow from the unaffected steam generator stops. Beyond this point, break flow decreases with decreasing pressure. Following closure of the main feedwater control valves, the remaining inventory in the affected generator is released out the break. Break enthalpy (Figure 5.5-7) corresponds to superheated steam throughout the transient.

Loss of Offsite Power

This case assumes that a loss of offsite power occurs coincident with reactor trip. Operator action is credited to close the MFW and EFW control valves at 170 seconds. The main feedwater block valves are not available for feedwater isolation due to the loss of offsite power. The main feedwater, condensate booster, and D heater drain pumps trip on the loss of offsite power. EFW is initiated at the time of reactor trip due to the loss of both main feedwater pumps.

Although Figures 5.5-1 through 5.5-7 depict the case in which offsite power is maintained, the discussion is generally applicable to Figures 5.5-8 through 5.5-14, the case in which offsite power is lost at the time of reactor trip. Important exceptions are noted below.
RCS pressure (Figure 5.5-8) decreases more slowly with the loss of offsite power due to the increased voiding in the RCS associated with this case. The affected loop cold leg temperature response (Figure 5.5-9) is much more severe than with offsite power maintained. However, due to the low RCS flow rate, a loop ΔT of approximately 150°F is present. The cold leg and hot leg temperatures in the unaffected loop (shown on the same figure) trend with the hot leg temperature in the affected loop. The loss of forced reactor coolant flow results in a much smaller return to power (Figure 5.5-10).

Emergency feedwater is throttled to maintain steam generator level at the natural circulation setpoint of 240 inches. Inventory in the affected steam generator (Figure 5.5-12) increases until the boiling rate exceeds the rate of EFW addition. Operator action is credited at 170 seconds to isolate both main feedwater and emergency feedwater to the affected steam generator. Break flow (Figure 5.5-13) is sustained by EFW addition and flashing in the feedwater train until the time of feedwater isolation. The break enthalpy (Figure 5.5-14) is very similar to the case with offsite power maintained.

5.6 References

5-1 Oconee Nuclear Station Final Safety Analysis Report, Duke Power Company.

5-2 Deleted

5-3 RETRAN-02: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 4, EPRI, November 1988.

5-4 CAP-Containment Analysis Package (FATHOMS 2.4), Numerical Applications, Inc., October 10, 1989.

5-5 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Revision 3-PA, Duke Power Company, September 2004

5-6 ANSI/ANS-56.4-1983, "American National Standard Pressure and Temperature Transient Analysis for Light Water Reactor Containments," American Nuclear Society, December 23, 1983.

5-7 Oconee Nuclear Station Technical Specifications

TABLE 5.3-1

Principal Parameter Initial Conditions

Parameter	Value
Power Level	2619.4 MWth
T-ave	581°F
SG Pressure	910 psig
RCS Pressure	2185 psig
Pressurizer Level	245 inches
SG Total Mass	[ ] lbm/SG
RCS Flow Rate	115% design flow

TABLE 5.4-1

Boundary Conditions

Parameter	Value
RCS Flow Rate	115%
Average Fuel Temperature	[ ]°F
BWST Temperature	120°F
BWST Boron Concentration	2210 ppm
CFT Temperature	120°F
CFT Boron Concentration	1835 ppm
EFW Temperature	120°F
Single Failure	4160V Switchgear

TABLE 5.5-1

Sequence of Events

Steam Line Break - Offsite Power Maintained

Event	Time (seconds)
Break occurs	0.0
Reactor trip on variable low pressure	1.7
Third CBP starts	1.45
Turbine stop valves close	2.7
HPI actuates	20.1
HPI boron injection begins	90.4
CFT injection begins	134.7
Boron from CFT B starts	179.5
FW isolation of SG A begins	170
FW isolation of SG A complete	190
Simulation ends	247

TABLE 5.5-2

Sequence of Events

Steam Line Break - Loss of Offsite Power

Event	Time (seconds)
Break occurs	0.0
Reactor trip on variable low pressure	1.7
Third CBP starts	1.44
Turbine stop valves close	2.7
HPI actuates	24.0
HPI boron injection begins	97.9
CFT injection begins	250.7
Boron from CFT B starts	343.7
Boron from CFT A starts	359.4
MFW, CB, & DHD pumps trip	0
FW isolation of SG A begins	170
FW isolation of SG A complete	190
Simulation ends	570

TABLE 5.5-3

Steam Line Break Mass and Energy Release Data - Offsite Power Maintained

Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)	Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)	Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)	Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
0.0	0.0	1260.3	14.0	2931.3	1272.5	55.0	1651.9	1258.0	141.0	801.0	1255.6
0.2	13181.0	1270.1	15.0	2795.3	1273.4	58.0	1712.9	1256.5	149.0	820.2	1254.4
0.4	10456.3	1278.4	16.0	2404.7	1274.5	60.0	1691.4	1256.1	150.0	820.0	1253.8
0.6	10009.1	1282.9	21.0	2153.6	1271.5	62.0	1654.9	1253.7	152.0	818.8	1253.0
0.8	9811.6	1284.7	23.0	2131.8	1269.0	64.0	1691.7	1250.5	154.0	814.8	1254.1
1.0	9477.3	1286.4	25.0	1992.4	1266.2	67.0	1637.5	1250.0	162.0	783.5	1254.1
1.2	9084.9	1287.7	28.0	1482.2	1263.5	70.0	1677.4	1249.2	166.0	779.7	1252.1
1.6	8470.9	1289.0	29.0	1546.8	1262.9	72.0	1596.8	1250.0	169.0	765.8	1253.3
2.0	8055.8	1290.4	30.0	2036.0	1263.1	75.0	1573.2	1248.9	172.0	757.5	1252.8
2.4	7082.7	1292.5	31.0	1865.0	1263.8	78.0	1503.8	1246.3	177.0	738.2	1252.7
2.8	5442.1	1295.1	35.0	1709.5	1263.9	82.0	1385.1	1249.2	180.0	724.9	1251.2
3.2	4356.5	1297.3	36.0	1619.3	1263.9	86.0	1258.8	1250.2	183.0	709.4	1252.2
3.6	3635.4	1299.3	37.0	1448.3	1263.3	89.0	1157.8	1249.1	188.0	679.1	1252.7
4.0	3375.4	1300.3	38.0	1093.7	1262.0	92.0	1073.3	1252.7	192.0	631.7	1252.9
4.4	3254.7	1300.9	39.0	742.7	1259.8	95.0	1029.3	1253.7	200.0	442.6	1254.2
4.8	3219.9	1300.6	41.0	675.4	1256.2	99.0	1017.4	1251.4	205.0	300.7	1255.5
5.4	3053.3	1299.9	42.0	1452.8	1257.3	102.0	1017.8	1253.3	210.0	87.8	1257.0
5.8	2986.8	1299.3	43.0	929.8	1252.0	108.0	1034.8	1252.8	213.0	67.0	1257.6
6.0	2976.3	1298.8	45.0	831.8	1252.9	111.0	1044.4	1250.4	214.0	190.0	1257.4
7.0	3090.8	1294.2	48.0	829.5	1254.5	114.0	1037.8	1252.6	215.0	177.7	1257.7
8.0	3223.6	1289.5	49.0	1001.3	1254.9	120.0	996.6	1252.8	220.0	148.8	1258.1
9.0	3169.5	1285.8	51.0	984.8	1253.5	124.0	942.4	1251.1	230.0	81.8	1260.7
10.0	2991.3	1283.2	52.0	881.2	1253.2	128.0	879.7	1254.4	240.0	39.9	1261.9
12.0	2997.8	1275.1	53.0	1118.5	1251.4	134.0	809.9	1255.0	245.0	27.3	1262.7
13.0	3101.0	1272.8	54.0	1650.8	1253.3	137.0	796.0	1253.4	247.1	23.0	1262.9

TABLE 5.5-4

Steam Line Break Mass and Energy Release Data - Loss of Offsite Power

Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)	Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)	Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)	Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
0.0	0.0	1260.3	7.4	2912.4	1303.5	207.0	580.9	1251.5	427.0	29.3	1258.5
0.2	13139.5	1270.1	7.8	3127.8	1302.8	208.0	330.9	1254.1	432.0	6.6	1259.4
0.4	10455.2	1278.5	8.4	3259.7	1299.9	212.5	405.6	1257.9	444.0	28.9	1260.5
0.6	10009.4	1282.9	8.8	3130.9	1295.6	213.0	411.5	1258.1	453.0	3.7	1261.7
0.8	9807.5	1284.8	9.2	3054.5	1292.0	213.5	516.1	1258.2	459.0	33.4	1261.3
1.0	9467.3	1286.5	9.4	3042.0	1290.7	220.0	491.6	1248.4	463.0	6.3	1261.7
1.2	9067.5	1287.9	9.6	3055.1	1289.6	230.0	488.5	1250.3	471.0	5.9	1262.1
1.4	8708.3	1288.9	10.0	3246.1	1288.9	244.0	472.2	1247.5	476.0	25.1	1262.2
1.6	8459.6	1289.2	10.5	3306.1	1289.0	270.0	398.9	1245.7	482.0	3.8	1262.5
1.8	8239.2	1289.9	12.5	2981.6	1288.8	294.0	369.6	1245.0	486.0	10.2	1262.3
2.0	7996.0	1290.8	15.0	2437.4	1287.4	298.0	274.5	1246.3	488.0	3.0	1262.3
2.2	7652.0	1291.8	16.0	2496.1	1286.8	300.0	272.4	1247.1	494.0	33.1	1262.5
2.4	7022.4	1293.0	18.0	2246.5	1284.9	310.0	287.4	1249.0	498.0	3.6	1262.3
2.6	6183.3	1294.2	21.0	2052.7	1282.5	345.0	301.5	1253.4	509.0	3.6	1263.0
3.0	4692.9	1297.0	22.5	2047.3	1281.5	353.0	305.3	1258.8	513.0	29.3	1263.1
3.4	3870.5	1299.6	30.0	1643.8	1279.4	360.0	256.6	1263.8	514.0	0.8	1263.3
3.8	3519.4	1301.8	40.0	1326.2	1273.6	384.0	152.3	1260.7	515.0	10.0	1263.3
4.2	3318.5	1303.5	50.0	1163.8	1270.1	397.0	5.5	1259.7	517.0	2.0	1263.3
4.6	3126.0	1304.9	70.0	983.4	1267.0	400.0	26.5	1260.6	528.0	3.9	1262.9
5.0	3007.5	1305.7	100.0	788.4	1265.0	403.0	60.5	1261.0	533.0	30.7	1263.0
5.2	2975.0	1305.9	120.0	690.2	1263.1	405.0	56.4	1261.2	536.0	1.2	1263.0
5.4	2957.3	1306.0	180.0	527.2	1254.6	409.0	25.6	1260.9	548.0	2.1	1263.6
5.8	2964.3	1306.0	200.0	494.7	1251.9	410.0	5.8	1260.9	553.0	29.8	1263.4
6.4	3055.4	1305.2	205.5	515.9	1251.0	420.0	17.5	1259.8	556.0	3.9	1263.6
7.0	2937.4	1304.3	206.5	579.0	1252.2	425.0	45.2	1258.2	570.2	2.1	1263.8

FIGURE 5.2-1

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FIGURE 5.2-2

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FIGURE 5.2-3

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Oconee Steam Line Break Mass and Energy Release
Offsite Power Maintained

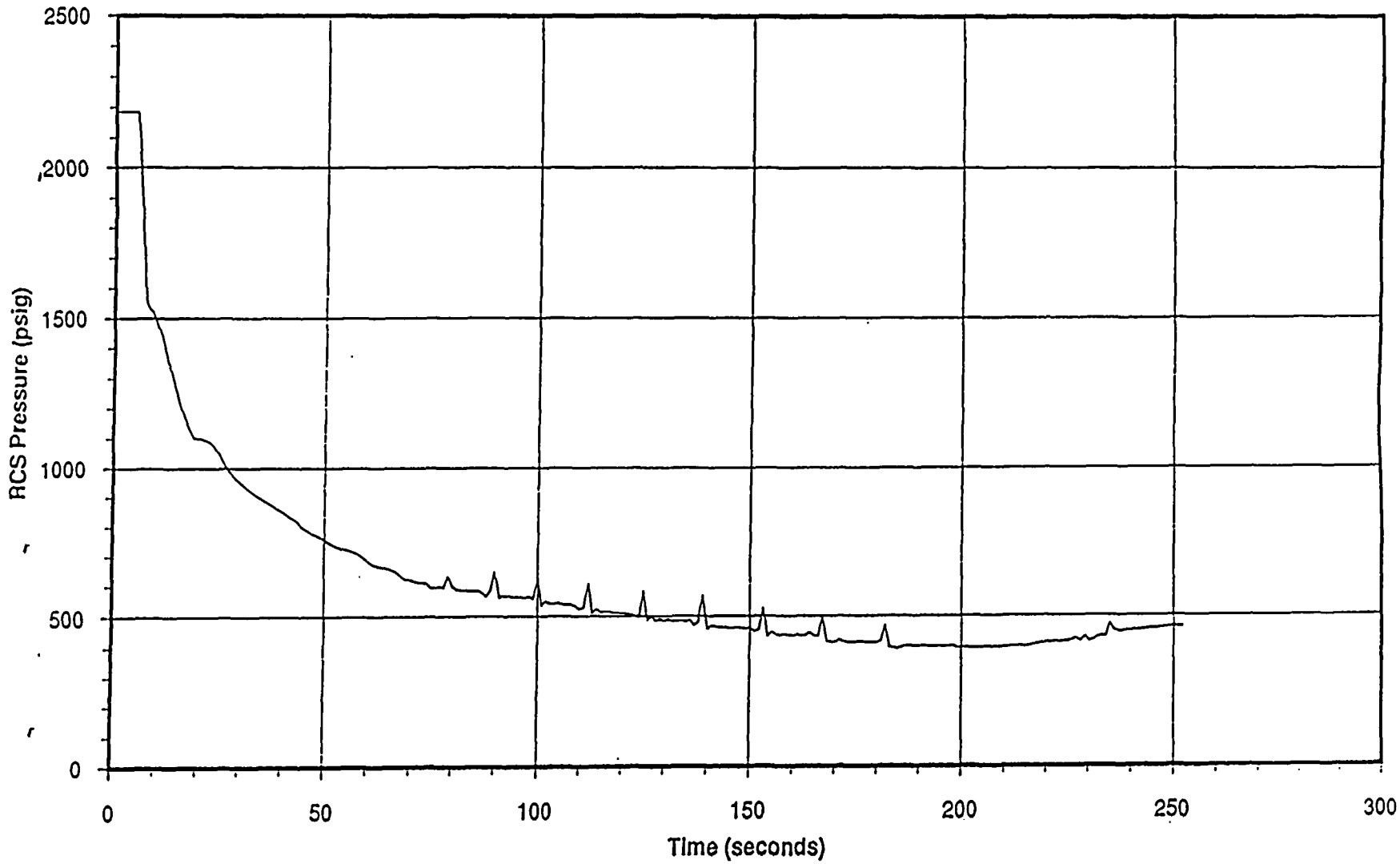

FIGURE 5.5-1

Oconee Steam Line Break Mass and Energy Release
Offsite Power Maintained

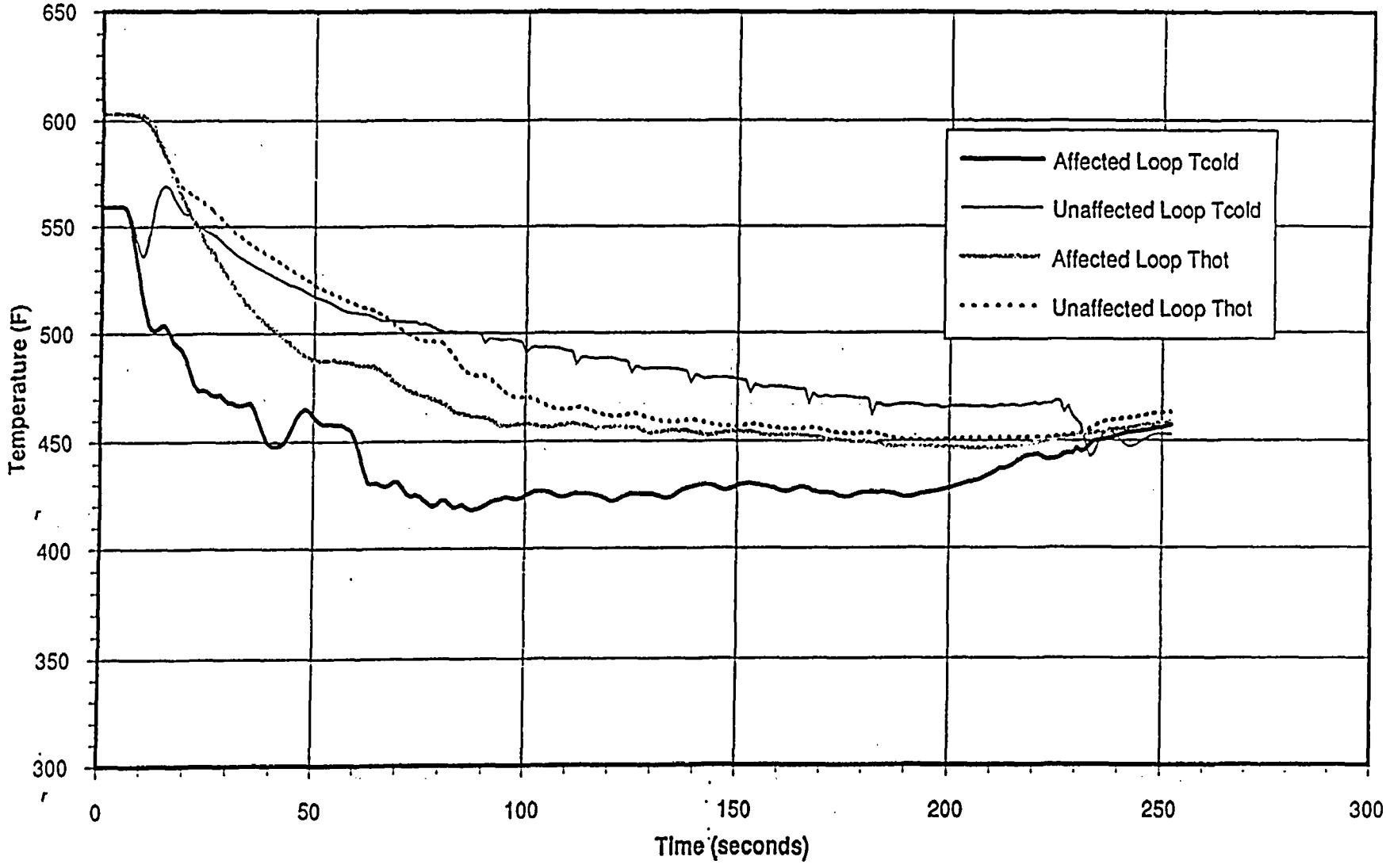

FIGURE 5.5-2

Oconee Steam Line Break Mass and Energy Release
Offsite Power Maintained

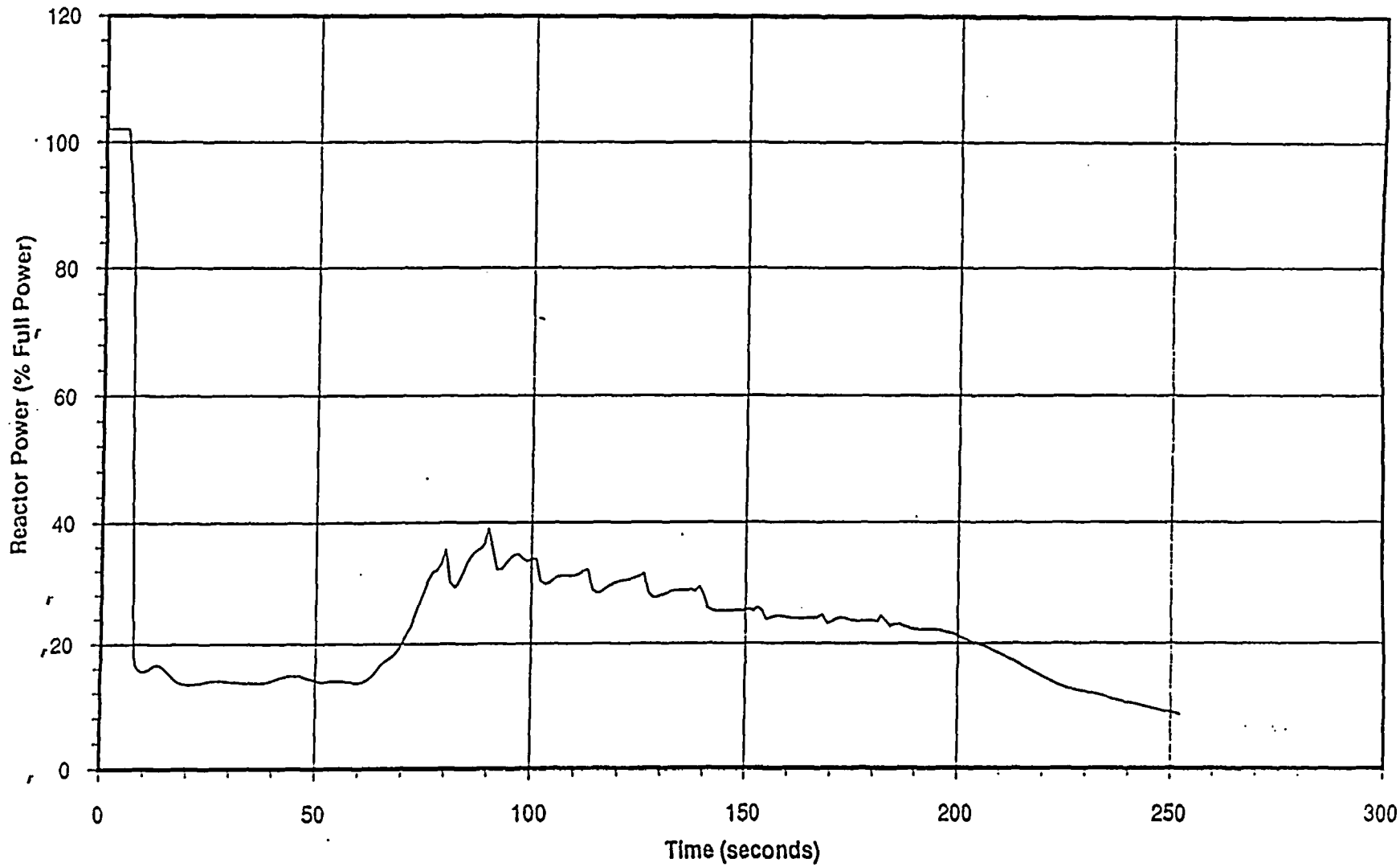

FIGURE 5.5-3

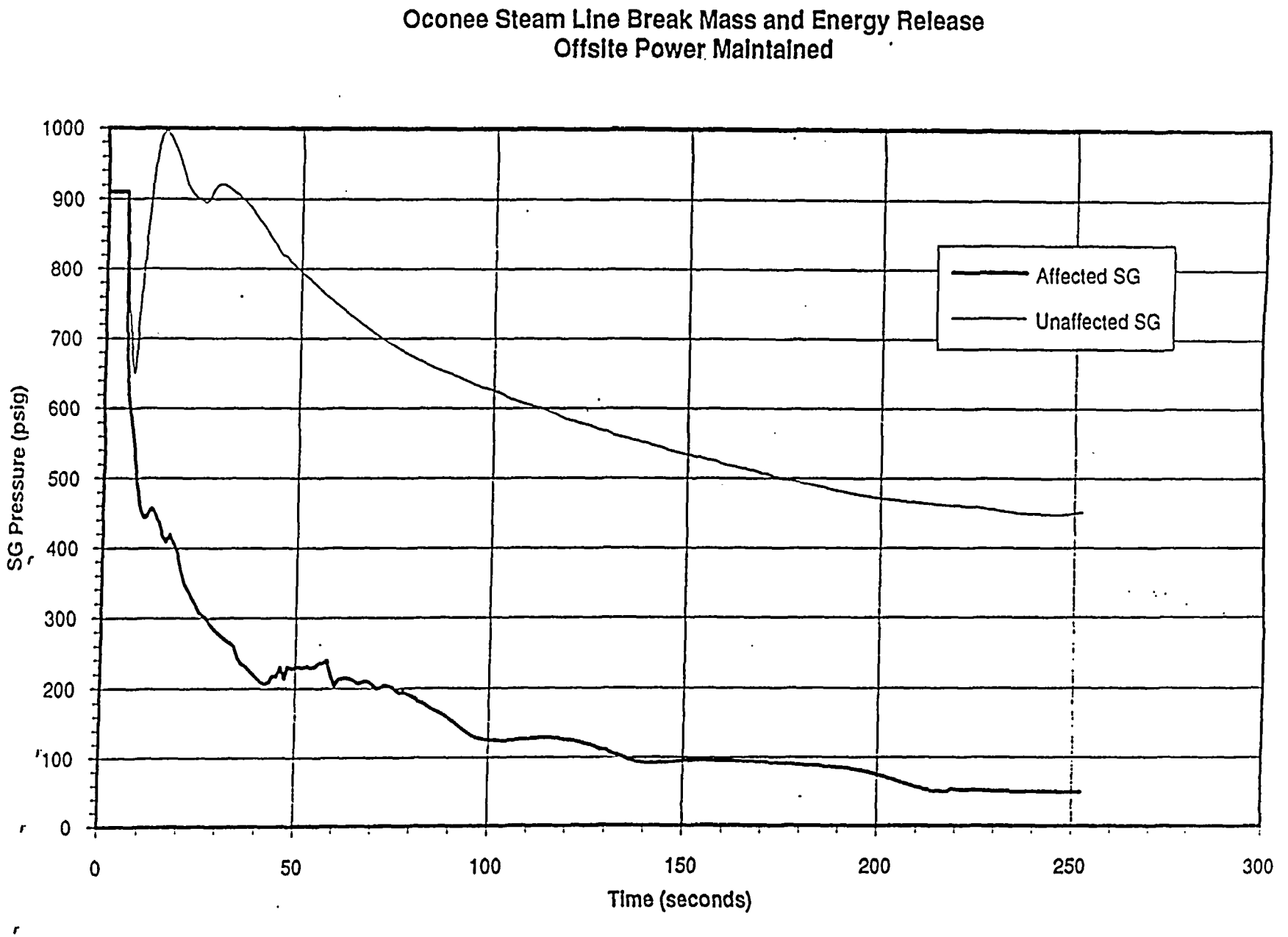
Oconee Steam Line Break Mass and Energy Release
Offsite Power Maintained
SG Pressure (psig)
1000
900
800
700
600
500
400
300
200
100
0
0
50
100
150
200
250
300
Time (seconds)
Affected SG
Unaffected SG

FIGURE 5.5-4

FIGURE 5.5-5

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Oconee Steam Line Break Mass and Energy Release
Offsite Power Maintained

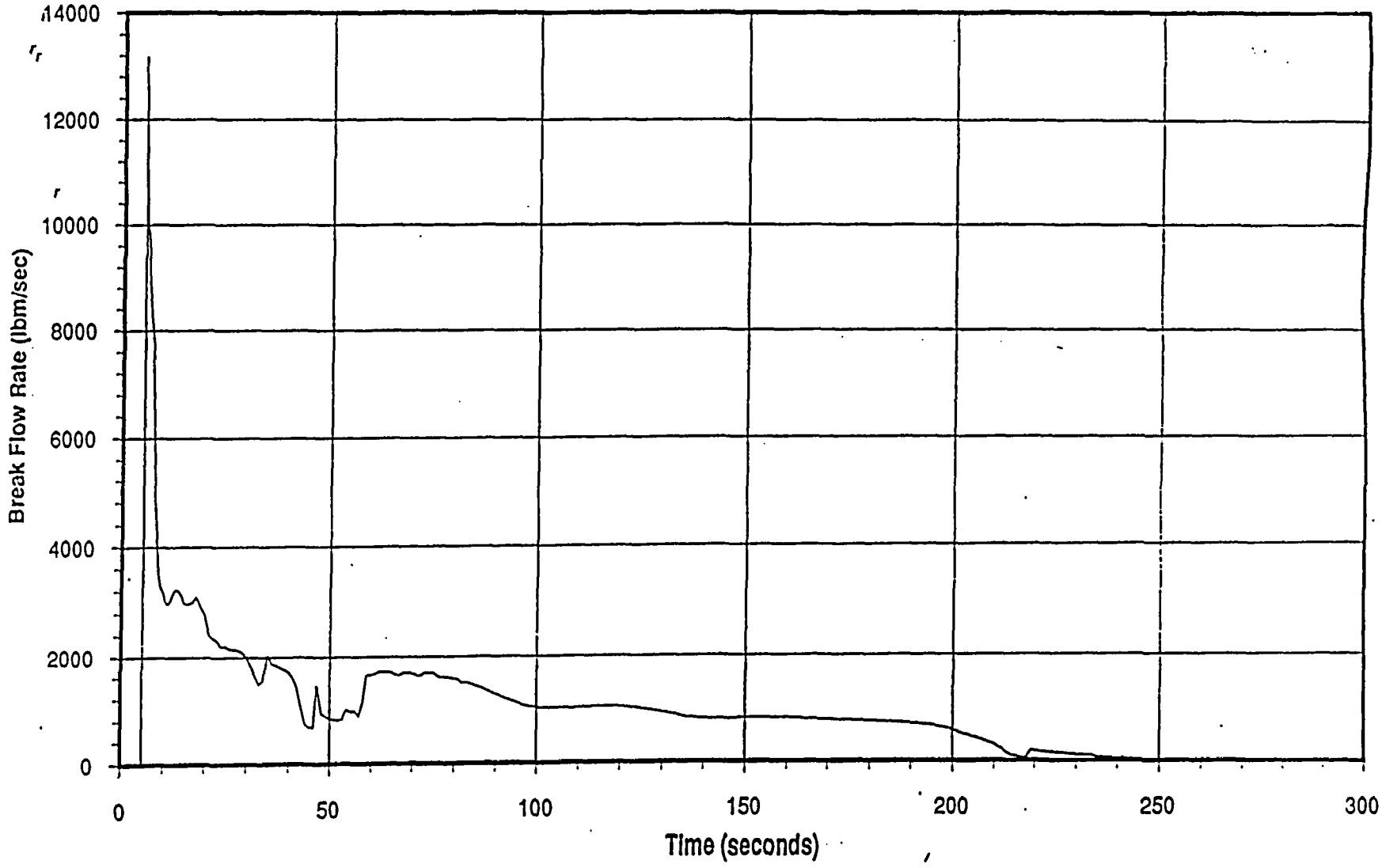

FIGURE 5.5-6

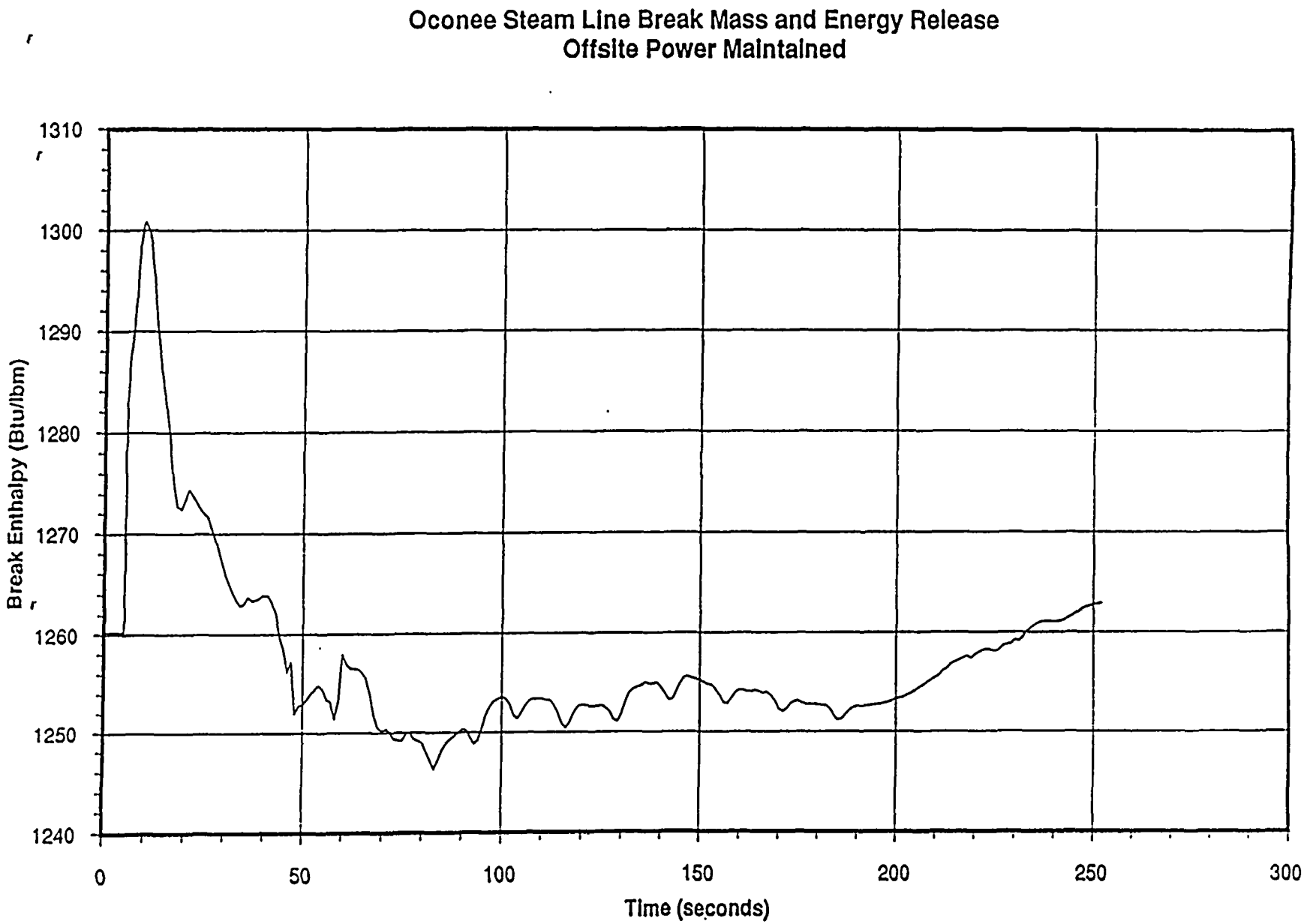
Oconee Steam Line Break Mass and Energy Release
Offsite Power Maintained
Break Enthalpy (Btu/lbm)
1310
1300
1290
1280
1270
1260
1250
1240
0
50
100
150
200
250
300
Time (seconds)

FIGURE 5.5-7

Oconee Steam Line Break Mass and Energy Release
Loss of Offsite Power

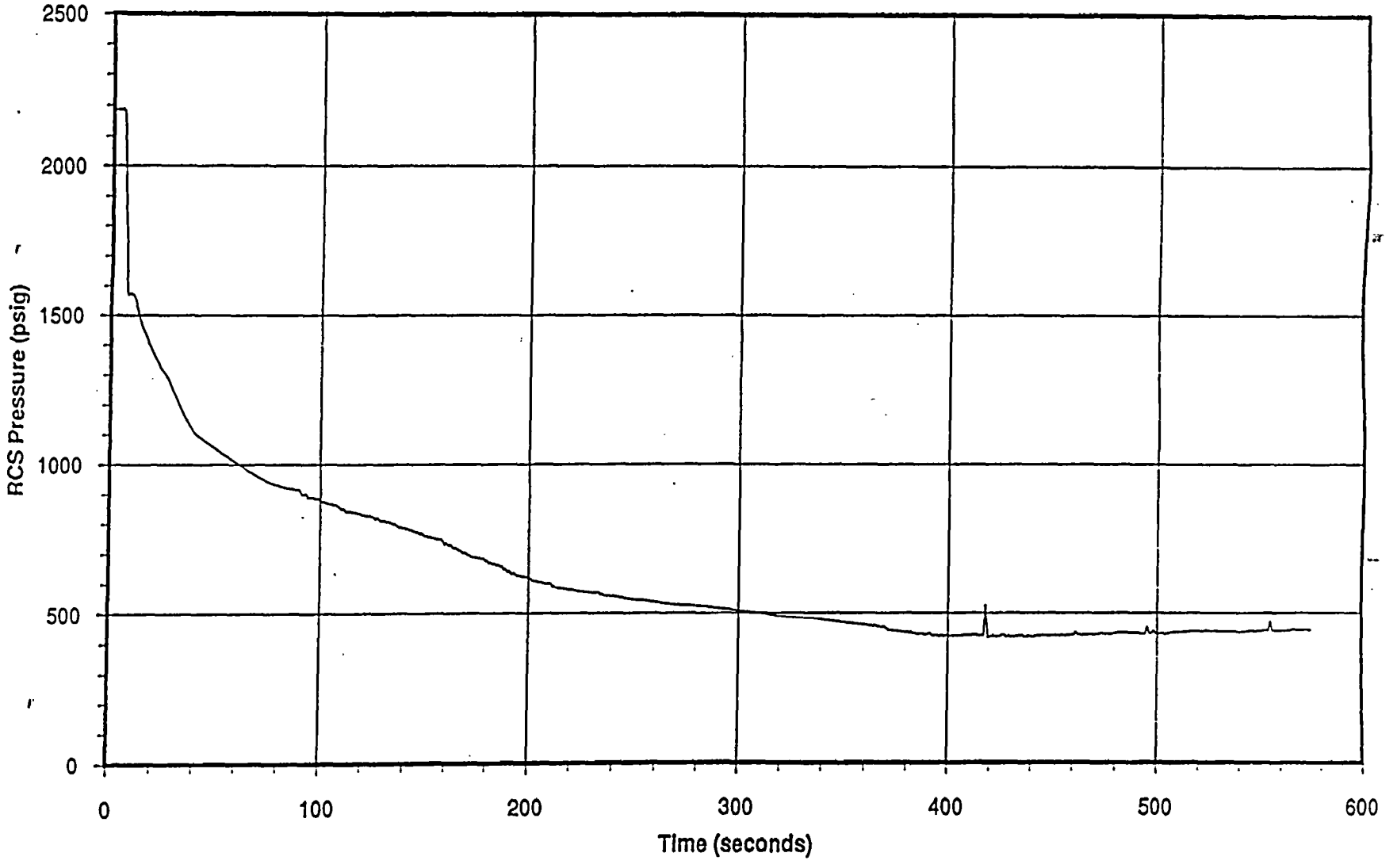

FIGURE 5.5-8

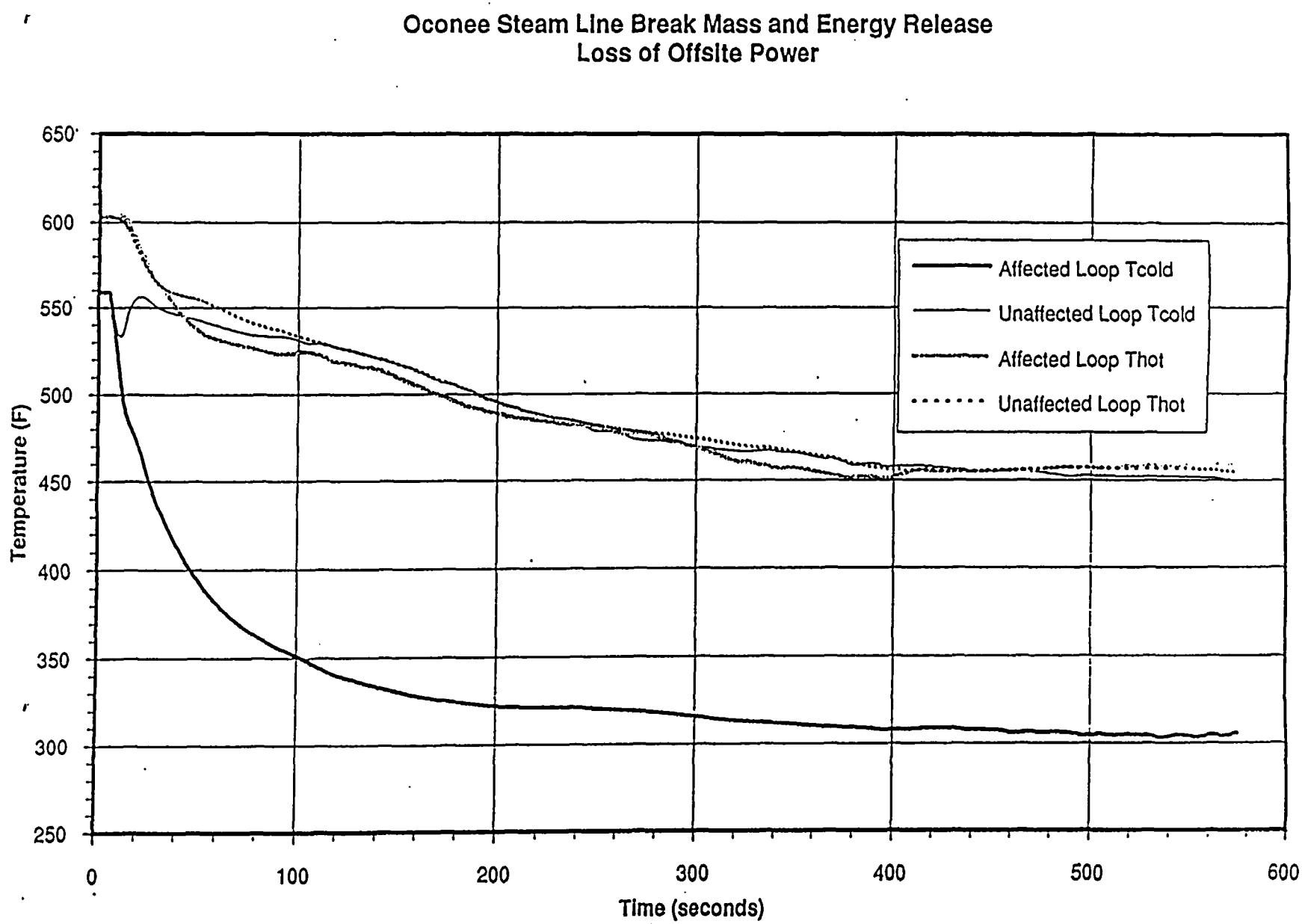
Oconee Steam Line Break Mass and Energy Release
Loss of Offsite Power
Temperature (F)
650
600
550
500
450
400
350
300
250
0
100
200
300
400
500
600
Time (seconds)
Affected Loop Tcold
Unaffected Loop Tcold
Affected Loop Thot
Unaffected Loop Thot

FIGURE 5.5-9

Oconee Steam Line Break Mass and Energy Release
Loss of Offsite Power

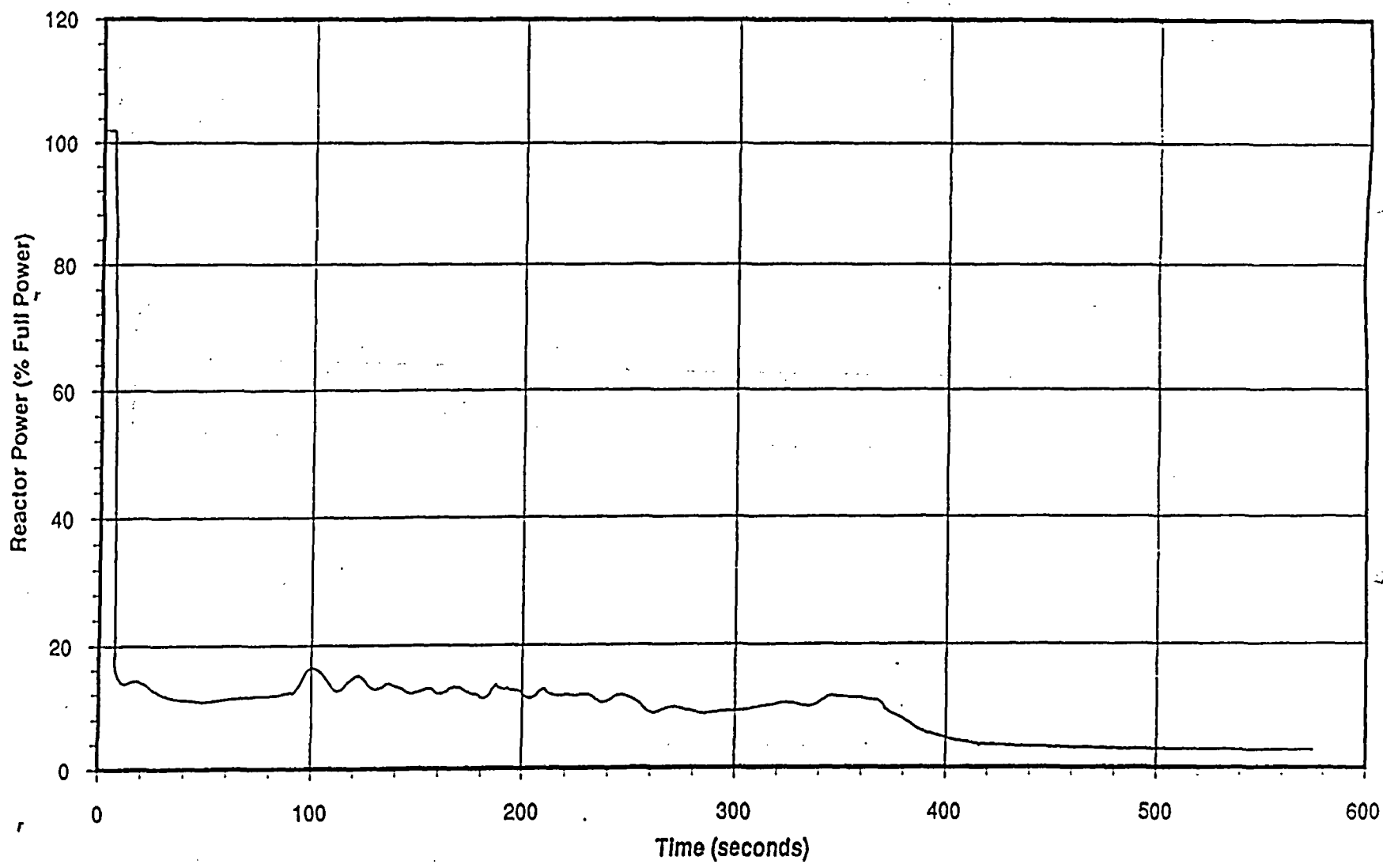

FIGURE 5.5-10

Oconee Steam Line Break Mass and Energy Release
Loss of Offsite Power

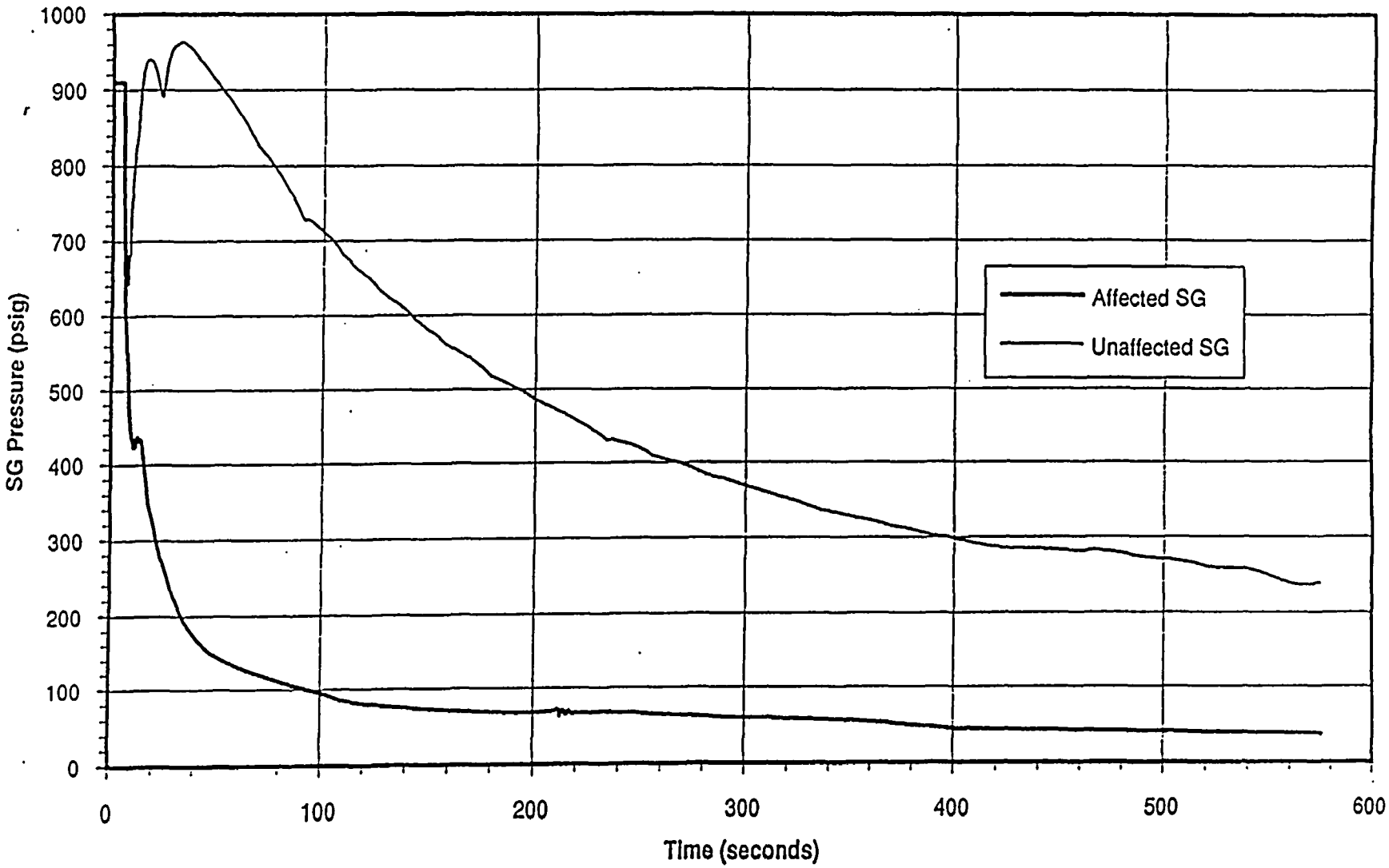

FIGURE 5.5-11

FIGURE 5.5-12

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Oconee Steam Line Break Mass and Energy Release
Loss of Offsite Power

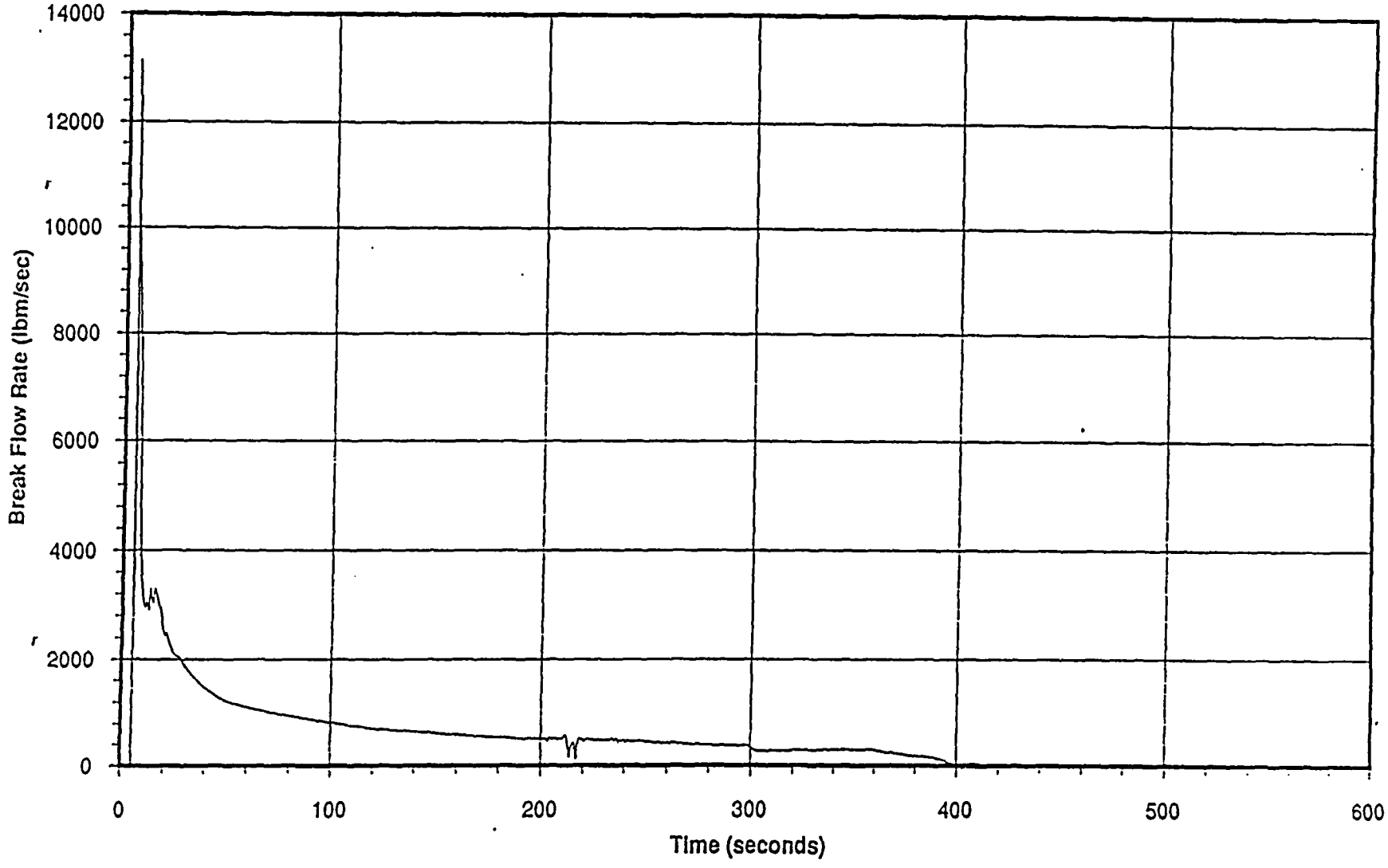

FIGURE 5.5-13

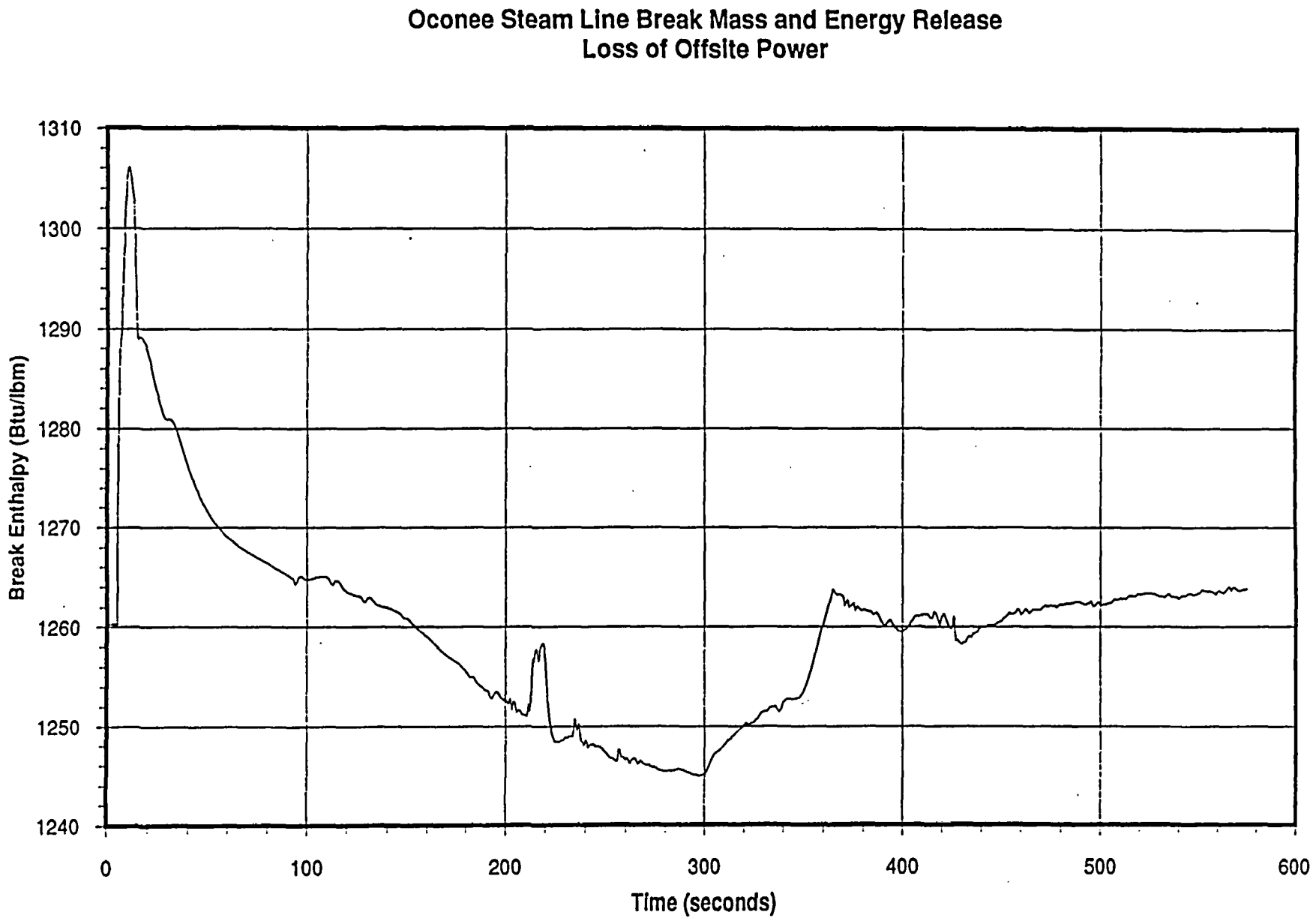
Oconee Steam Line Break Mass and Energy Release
Loss of Offsite Power
Break Enthalpy (Btu/lbm)
1310
1300
1290
1280
1270
1260
1250
1240
0
100
200
300
400
500
600
Time (seconds)

FIGURE 5.5-14

6. CONTAINMENT ANALYSES

6.1 Overview

Analyses are performed to determine the pressure and temperature response of the Reactor Building following a high-energy line break. The mass and energy release data generated by the RELAP5, BFLOW, and RETRAN-02 analyses detailed in Chapters 3, 4, and 5 is used as input to the FATHOMS/DUKE code. The RELAP5 mass and energy release data is given in tables, with mass flow rates given in lbm/sec and energies given as enthalpies, in Btu/lbm. [

] The containment model discussed in Section 2.4.2 is utilized for each case. Junctions are added to model the appropriate boundary conditions for each case. The initial and boundary conditions for each case are specified to conform to the ANSI/ANS-56.4-1983 standard (Reference 6-1).

Each containment analysis case is run in different time segments, depending on the transient being simulated. Typically, a different set of junctions is utilized in each segment to model the boundary conditions active in that particular segment. For the long-term large break LOCA, two segments are used to achieve the best accuracy in the [

] code. The small-break LOCA simulations are also divided into two segments to implement the models applicable for that type of analysis. The steam line break and peak pressure analyses are of short duration and may be performed in only one segment.

In the LOCA peak pressure analyses, the predicted pressure response is compared to the Reactor Building design pressure of 59 psig. This peak occurs during the blowdown period which spans the first 30 seconds for the large-break LOCA. The actuation of engineered safeguards and condensation of steam on structures is sufficient to keep building pressure from increasing again following the completion of the RCS inventory blowdown period.

For the long-term analyses which are the bases for the environmental qualification (EQ) of safety equipment, the EQ profiles for the limiting equipment are used as acceptance criteria for the reanalyses. The transient temperature and pressure in the Reactor Building must be bounded by the qualification testing of the safety equipment, or a specific evaluation is necessary. The pertinent recommendations from Appendix B of NUREG-0588 (Reference 6-2), regarding containment analysis for environmental qualification for electrical equipment, are applied for all

long-term analyses. The Uchida heat transfer correlation, however, is applied for all heat sinks instead of the Tagami correlation. This difference has no impact on the long-term results of this analysis since the Tagami heat transfer logic reverts to the Uchida correlation after the peak building pressure is reached (within 20 seconds for the large-break LOCA analyses). The long-term analyses are run out to a time of 15 days. The vapor temperature profile is then extrapolated to the 20 day point to examine the final point on the EQ envelope (which is actually at a time of 20 days, 16.4 hours). The rate of decrease for the final few days of each run is very stable, so a good estimate may be made for the 20-day point from the calculated temperatures for days 12 to 15.

The steam line break analyses result is compared to both the 59 psig Reactor Building design limit and the limiting EQ profiles. These analyses typically last less than 10 minutes, by which time the pressure and temperature trends are decreasing and the acceptance criteria may be evaluated.

Based on the results of the containment analyses, performance requirements for initial conditions, engineered safeguards, and operator actions can be established. These include Reactor Building pressure and temperature, LPI coolers, the Reactor Building Spray System, reactor building cooling units, feedwater isolation valves, and actuation setpoints. Parametric studies are used to assess the impact of such things as service water temperature.

6.2 Peak Pressure - Containment Response

6.2.1 Overview

Following a large break loss-of-coolant accident (LOCA), the rapid release of the Reactor Coolant System inventory leads to a very rapid increase in temperatures and pressures within the Reactor Building. The FATHOMS/DUKE-RS computer code is used to calculate the peak Reactor Building pressure reached following this accident. The mass and energy released from the RCS is calculated by the RELAP5/MOD2-B&W code as discussed in Section 3. This data is used as input to the FATHOMS/DUKE-RS (or FATHOMS) code. Several different large break LOCAs are simulated with RELAP5. Variations in break size, location, and boundary conditions are made in each simulation. The following cases are executed to determine the limiting scenarios.

Case	Description	Sensitivity
1A	14.1 ft^2 hot leg break at vessel outlet	Break size/location
1B	14.1 ft^2 hot leg break at S/G inlet	Break size/location
1C	8.55 ft^2 cold leg pump discharge break	Break size/location
1D	8.55 ft^2 cold leg pump suction break	Break size/location
2B	14.1 ft^2 hot leg break at vessel outlet w/ Ransom & Trapp critical flow model	Critical flow model
2D	14.1 ft^2 hot leg break at S/G inlet w/ Ransom & Trapp critical flow model	Critical flow model
3A	14.1 ft^2 hot leg break at vessel outlet w/o off-site power	Off-site power
3B	14.1 ft^2 hot leg break at S/G inlet w/o off-site power	Off-site power
3C	14.1 ft^2 hot leg break at vessel outlet with 30 psia containment back-pressure	Containment backpressure
3D	14.1 ft^2 hot leg break at vessel outlet with 50 psia containment back-pressure	Containment backpressure
3E	14.1 ft^2 hot leg break at vessel outlet with zero refill time	Refill time

6.2.2 Analytical Approach

To ensure conservatism, the initial and boundary conditions chosen for all containment analyses conform to ANSI/ANS-56.4-1983 (Reference 6-1). These assumptions result in a conservatively high calculated peak Reactor Building pressure. The required variations to the Oconee FATHOMS base model are discussed in Section 6.2.3 below.

Each case from above will be run to determine the peak containment pressure. It is expected that the peak pressure will occur within the first 20 seconds of each simulation. This varies greatly from the peak pressure results given in the original Oconee Final Safety Analysis Report (Reference 6-3). In the original UFSAR analyses, performed by Babcock and Wilcox prior to the initial startup of the station in 1973, two pressure peaks are present in most large break LOCA simulations, with one peak occurring during the blowdown period (at about 15-20 seconds), and

a second, higher peak occurring between 90-110 seconds. This is mainly due to differences in the passive heat structure data used in the respective analyses, rather than the use of different containment analysis codes. This is discussed in detail in Section 6.2.5.

6.2.3 Modifications to Base Containment Model

Containment Free Volume and Passive Heat Sink Data

The nominal containment free volume (CFV) from the base model is used, with a 2% reduction for conservatism. The containment volume assumption has a greater impact on the peak pressure value than any other single containment analysis assumption, so it is important to choose a conservative value for this parameter. This is implemented in the containment model by reducing the building cross-sectional area by 2%; the building height remains unchanged.

The surface areas of the passive heat sinks are used, with a 1% reduction. This assumption also reduces the volumes of the passive heat sinks, since the thicknesses of the heat sink slabs modeled remain unchanged. Since the peak pressure is expected to occur before the passive heat sinks can condense significant amounts of steam, it is not expected that this parameter will have a significant impact on the results of the analysis.

The Uchida heat transfer correlation for condensing heat transfer coefficients on the passive heat structures is utilized for all runs in this analysis. This varies from the original B&W analyses, in which an initial constant heat transfer coefficient of 110 Btu/hr-ft^2-°F is reduced to 40 Btu/hr-ft^2-°F once a certain total heat flux is achieved. Again, the use of the Uchida correlation is in conformance with Reference 6-1.

Droplet Size of Blowdown Mass

A 100 μm value for the size of the blowdown droplets is used in the Revision 1 methodology. Test data cited in Reference 6-3 suggests that the droplet sizes for high-pressure blowdowns range from 2 - 2000 μm, and Reference 6-8, suggests that 100 μm is a suitable average value to use with the FATHOMS code. A value of 20 μm value was used in the Revision 0 methodology based on Reference 6-4.

RBCUs and Building Spray Trains

No credit is taken for either of these safety systems. Since the peak pressure is expected to occur within the first 20 seconds of the transient, neither the RBCUs nor the Reactor Building Spray System will have any impact on the peak pressure, as it cannot be assumed that these systems will actuate and function in that time period.

6.2.4 Initial Conditions

Initial Pressure

An initial building pressure of 16.2 psia (1.5 psig) is assumed. Oconee Technical Specification 6.4 requires the Reactor Building to be below 1.2 psig during full power operation, so this assumption will be conservative.

Initial Temperature

It is conservative for this transient to assume a low initial building temperature to arrive at conservatively high peak building pressures. Cooler temperatures result in larger masses of air in a steam-air mixture at any given pressure. The contribution of the partial pressure of the air in the Reactor Building during the first 20 seconds results in a higher peak pressure than any high initial temperature case.

The combination of an initial Reactor Building pressure of 1.5 psig and temperatures below 110°F are unexpected during full power operation. Therefore, an initial building temperature of 110°F will be assumed.

Initial Relative Humidity

An initial relative humidity of 0% is assumed. This assumption is made to maximize the amount of air initially in the building. It is not expected that the variation in initial building humidity will have a significant impact on this analysis.

6.2.5 Boundary Conditions

The mass and energy release data described in Chapter 3 is supplied to the FATHOMS code as a boundary condition. The steam-water mixture is injected into the containment model through the FATHOMS junctions supplied for this purpose. They are injected at the proper enthalpy and pressure. All liquid is assumed to have an average droplet size of 20 μm, as described above. This impacts the amount of interfacial heat transfer, as larger droplets have less surface area and result in less heat transfer/condensation.

Since no Building Spray flow or RBCUs are taken credit for in the peak pressure analyses, the mass and energy release is the only boundary condition utilized. Since the duration of the analysis is only 180 seconds, sump recirculation mode is not entered, so none of the boundary conditions associated with this mode of operation are modeled.

A summary of important initial conditions, boundary conditions, and assumptions is given in Table 6.2-1.

6.2.6 Results

Case 1 Series

The Case 1 series is performed to examine the effect of break location on the containment peak pressure results. Figures 6.2-1 to 6.2-4 show the containment pressure response for Cases 1A to 1D. The building vapor temperature responses for these cases are shown in Figures 6.2-5 to 6.2-8. The results of these cases are summarized below.

Case	Time of Peak Pressure (sec)	Peak Pressure (psig)	Peak Temp. (°F)
1A	13.9	57.7	283.5
1B	14.9	58.4	284.4
1C	18.9	51.9	276.1
1D	23.2	54.4	279.3

These results demonstrate that a 14.1 ft^2 hot leg break at the steam generator inlet yields the highest peak building pressure. Figure 3.5-5 shows that Cases 1A and 1B have very

similar total energy release rates to containment; however, the higher release rate for Case 1B during the 10-15 second interval (coming from the S/G side of the break) causes the peak containment pressure for Case 1B to exceed that for Case 1A by about 0.7 psi.

Similarly, the peak pressure for Case 1D (8.55 ft^2 cold leg break) exceeds the peak pressure for 1C despite much higher energy release rates for Case 1C early in the transient. The higher steaming rates from the S/G side in Case 1D after 10 seconds again result in a higher peak pressure. The temperature difference between the concrete and steel heat structures in containment and the building atmosphere is much greater from 0 to 10 seconds than from 10 to 30 seconds. Therefore, the capacity of the structures to condense steam on the surfaces and lower the building pressure is much higher during this earlier interval. Transients which release large amounts of energy very rapidly may be less severe in terms of building pressurization than transients which release energy slower but over a longer period of time.

The pressure and temperature responses for Cases 1C and 1D show a slight increase in Reactor Building pressure and temperature towards the end of the analysis. This is a result of the lack of Reactor Building Spray or RBCUs in the peak pressure analysis. These systems were not taken credit for in this set of analyses. Had they been modeled, these increases would not be present in Cases 1C and 1D.

Case 2 Series

Cases 2B and 2D are analyzed to examine the different critical flow models at the various break locations. The containment pressure responses for these cases are shown in Figures 6.2-9 and 6.2-10, and the building temperature responses are shown on Figures 6.2-11 and 6.2-12. The results of these cases are summarized below.

Case	Time of Peak Pressure (sec)	Peak Pressure (psig)	Peak Temp. (°F)
2B	12.8	58.3	284.3
2D	13.8	58.9	285.0

Case 2D, in which a 14.1 ft^2 hot leg break at the S/G inlet is modeled using the Ransom & Trapp critical flow model, gives the highest peak pressure, 58.9 psig. This is still below the building design pressure of 59 psig.

Case 3 Series

The Case 3 series is performed to examine the sensitivity of off-site power availability, containment backpressure, and refill time assumptions on the peak building pressure. Only building pressure profiles are plotted for this series as shown in Figures 6.2-13 to 6.2-16. The containment responses for this series are summarized below.

Case	Time of Peak Pressure (sec)	Peak Pressure (psig)	Peak Temp. (°F)
3A	12.3	57.9	283.8
3B	14.5	58.9	285.0
3C	13.8	58.4	284.4
3D	13.5	58.4	284.4
3E	12.8	58.3	284.3

Cases 3A and 3B are revisions of Cases 2A and 2D, respectively, with varying loss-of-offsite-power assumptions. Cases 3C and 3D are revisions of Case 2B with different containment backpressure assumptions. Case 3E is a duplication of Case 2B, with a zero refill time used.

Cases 2D and 3B have the highest peak building pressures, each at about 58.9 psig. These are close to but still below the building design limit of 59 psig. Case 3B actually is higher than Case 2D by about 0.02 psi, and will be considered the worst-case LOCA for containment peak pressure.

The results from Cases 3C and 3D show that the different containment backpressure assumptions used in the RELAP5 mass and energy release calculations have no significant impact on the calculated peak pressure. The different backpressure assumptions result in a very small difference in peak pressures (< 0.1 psi).

The refill time sensitivity case performed (Case 3E) has identical results to the base case (Case 2D). This is because the zero refill time assumption only changes the mass and energy release after the blowdown period, and so the Reactor Building peak pressure is unaffected by this parameter. Therefore, no peak pressure plot is shown for this case.

Additional Sensitivities

The peak pressure in Case 3B occurs about 0.7 seconds later than Case 2D. Because of this difference, and the closeness of the peak pressures in the two cases, it is possible that Case 3B will not be the worst-case LOCA for all initial containment conditions. A second set of runs is performed for Case 2D and Case 3B. This time, the initial pressures and temperatures are modified from 1.5 psig and 110°F to 1.2 psig and 80°F. These are the limiting containment conditions for peak pressure calculation at temperatures below 110°F. If containment temperatures below 110°F were present at pressures above 1.2 psig during full power operation, the peak containment pressure for either Case 3B or Case 2D would be over the design limit of 59 psig.

The results of these runs are shown below. Again, the peak pressures are shown as identical. However, Case 2D has the higher peak pressure this time, by 0.01 psi.

Case	Time of Peak Pressure (sec)	Peak Pressure (psig)	Peak Temp. (°F)
2D - 80	13.7	58.8	284.0
3B - 80	14.3	58.8	284.0

The loss-of-offsite-power assumption, which is the difference in Cases 2D and 3B, causes the mass-and-energy releases to vary slightly. It is demonstrated that this assumption is conservative in low building temperatures and non-conservative for higher building temperatures. Since the Reactor Building temperature is expected to be closer to 110°F than 80°F during full power operation, Case 3B is considered to be the worst-case LOCA for peak Reactor Building pressure.

6.3 Long-Term Large Break LOCA Containment Response

6.3.1 Overview

Following the blowdown of the Reactor Coolant System (RCS) inventory during a large break loss-of-coolant accident (LOCA), the RCS refills with core flood tank and injection water from the borated water storage tank (BWST), with this water provided by the high and low pressure injection pumps (HPI, LPI). In the containment building, the rapid increase in pressure and temperature is followed by a gradual decrease as the systems designed to remove energy from the building activate. These systems are the Reactor Building Spray System (BS) and the reactor building cooling units (RBCUs). The RCS eventually depressurizes to equal the pressure in containment, so the RCS inventory will come to the approximate level of the pipe break. As discussed in Section 3.3.3.2, the pipe break site is never completely flooded, and a certain quantity of steam will exit the break throughout the transient for a cold leg break.

To meet the environmental qualification (EQ) requirements for the limiting equipment required to function following this accident, the Reactor Building temperature and pressure profiles following all high-energy line breaks must be enveloped by a prescribed set of curves developed for that equipment. Therefore, this accident must be simulated for a period approaching a 20-day duration, as the Reactor Building temperature is required to return to its initial value of 125°F at that point in time.

The FATHOMS/DUKE-RS computer code is used to calculate the Reactor Building pressure and temperature response following this accident. The mass and energy released from the RCS is calculated by the RELAP5/MOD2-B&W code as discussed in Chapter 3. This data is used as input to the FATHOMS/DUKE-RS (or FATHOMS) code. Since this data only lasts until the start of the recirculation mode, the BFLOW code is used to determine the quality of the flow exiting the break thereafter. As discussed in Section 3.4, a [

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The rates at which the Reactor Building temperature and pressure decrease are dependent on the capacity of the RBCUs and LPI coolers (which are used to cool the sump water once sump recirculation mode is entered). Another key parameter is the temperature of the Low Pressure Service Water (LPSW) water, which is used as the cooling water for both of these systems The cooler the LPSW water is, the greater the capacity of the RBCUs and LPI coolers to remove

energy from the Reactor Building. The ability of both these systems to remove energy from the Reactor Building has been affected by fouling and tube plugging.

Another important system which aids in cooling the building is the BS system. Once the BWST empties, the source of water for the BS pumps is switched to the building sump. This water is sprayed back into the building atmosphere to aid in condensation of the steam being released out the pipe break. The energy of this steam is then transferred to the building sump (in the form of condensate), where it may be removed from the building by the LPI coolers.

The goal of this analysis is to determine the minimum required capacity of the RBCUs as a function of LPSW temperature, LPI cooler capacity, and BS flow rate. The RBCU capacity is chosen as the primary variable because this capacity may vary based on operating data. The variation in LPI cooler capacity is less, also based on operating data. The LPSW temperature may vary with seasons, as this water is taken from Lake Keowee. The temperature may range from 55°F to a maximum of 90°F. As stated above, this has a significant impact on the RBCUs and LPI coolers.

6.3.2 Analytical Approach

A set of initial conditions and system flow rates has been chosen to give conservatively high containment pressures and temperatures, in accordance with ANSI/ANS-56.4-1983, Pressure and Temperature Transient Analysis for Light Water Reactor Containments (Reference 6-1). Many of these initial conditions are the same as those in the RELAP5 mass and energy release calculation. Where conservatism in the respective analyses dictate different flowrate assumptions, adjustments are made to correct these discrepancies.

Two different sets of flowpaths (junctions) are used in the analysis to implement the mass and energy release data supplied from the RELAP5 analysis and the data from the BFLOW code. Each set of flowpaths is active during one time segment of the analysis. The first segment ends at the end of the injection mode, which is the end of the mass and energy release data from the RELAP5 analysis. The second segment lasts from the start of the recirculation mode to the end of the analysis. These segments will be designated as Segments 1 and 2.

The required revisions to the Oconee FATHOMS base model are discussed in Section 6.3.3 below.

The environmental qualification profile for the limiting safety equipment is used for EQ acceptance criteria. It is expected that the only point on this EQ profile which could be challenged by changing the cooling capabilities addressed in this analysis is the 100,000 second point. Here, the EQ profile requires containment temperatures to be < 200°F. Another point on the EQ profile to be met is the final point at 1,787,200 seconds (20 days, 16.4 hours), when the containment temperature is required to have returned to 125°F. These two points are the only ones examined on each FATHOMS run, since no changes are made to the mass and energy release data (the same cold leg break is being analyzed in each run), and the changes in RBCU and BS capability are not enough to violate the EQ guidelines prior to aligning for the sump recirculation mode. Likewise, the first two step changes in the EQ profile after sump recirculation initiation (at 8657 and 33,338 seconds) will probably not be exceeded due to any change in cooling capability assumptions for this transient.

To conserve run time, each FATHOMS case is run for only 15 days. The rate of decrease in building pressures and temperatures is well established for a run of this duration. Therefore, the vapor temperature at 20 days is extrapolated from the data from the first 15 days. The vapor temperatures at 12, 13, 14, and 15 days are examined to determine an exact rate of decrease and provide a good estimate for the temperature at 20 days. Since the EQ requirement is for a point at 20 days + 16.4 hours, there is still a conservative margin left if the comparison is done for a 20-day extrapolation.

Likewise, the vapor temperature at the 100,000 second point is estimated in advance for all runs after the first segment (0 to 1 day, or 86,400 seconds) of each run. An extrapolation is done using the last few temperature data points in this segment to determine the approximate vapor temperature at 100,000 seconds. If it is apparent that the vapor temperature will not fall below 200°F by the 100,000 second point, the second and third segments of the analysis are not performed, since the EQ criteria have already been violated. For all other runs, a verification of the 100,000 second point is performed after the second segment (1 to 5 days) is run.

The set of runs is performed in a sequence to reduce the total number of runs required. The two primary parameters to be varied are RBCU capacity and LPSW temperature. The runs are performed to determine the minimum RBCU heat removal capacity required to meet the EQ criteria described above at a given LPSW temperature. The other two variables, LPI cooler capacity and Building Spray flow, are held constant. Therefore, varying the LPSW temperature

from maximum to minimum range and determining the minimum required RBCU capacity for each temperature will complete one set of results.

Once it has been established that a run has met the EQ criteria for a given RBCU heat removal capacity and LPSW temperature, all cases at the same RBCU capacity and lower LPSW temperature, or at the same LPSW temperature and higher RBCU capacity, also meet the EQ criteria (all other parameters being the same). Increasing the ability of any one (or more) of the containment heat removal systems to remove energy from the Reactor Building will never make the containment temperature higher at any point in the analyzed transient. This greatly decreases the number of runs required.

The LPI cooler capacity will initially be held at the minimum value, 93 x 10^6 Btu/hr, until a complete profile for the two primary variables is determined. Then, more runs are done at higher LPI cooler capacities to determine if a lower RBCU capacity is acceptable for these higher LPI cooler capacities.

A similar set of runs is performed with the same two primary variables (LPI cooler capability, RBCU heat removal capacity) to develop another minimum RBCU capacity profile. This time, the assumed Building Spray flow rate is increased from 600 to 800 gpm during sump recirculation. The minimum RBCU capacity needed to meet EQ requirements will be lower for this set of runs.

The above methodology can be implemented with any set of assumptions. The values assumed above are the currently valid values.

6.3.3 Modifications to Base Containment Model

Containment Free Volume and Passive Heat Sink Data

The containment free volume and passive heat sink values described in Section 6.2.3 are also used in this analysis.

BWST Volume and Temperature

During Segment 1, the ECCS injection flow coming from the BWST and the flow exiting the break are modeled in the RELAP5 analysis. After the switchover to sump recirculation mode

(during Segment 2), the source for the ECCS water will be the sump, so the BWST is not modeled.

Note: The following two paragraphs are superceded due to being applicable only to Revision 0 of the report. They are retained for understanding the Revision 0 results only.

The borated water storage tank (BWST) is modeled with a second calculational volume. It is only active during Segment 2 of the analysis. During Segment 1, the core injection flow coming from the BWST and the flow exiting the break are modeled in the RELAP5 analysis; they are accounted for in the FATHOMS analysis through the cold leg break boundary conditions. However, the RELAP5 analysis is performed for only 30 minutes, so the FATHOMS long-term mass and energy release model must be used during Segment 2. This requires the modeling of the source volume for the core injection water during this segment. After the switchover to sump recirculation mode (during Segment 3), the source for the core injection water will be the sump, so the BWST volume will again be inactive.

The BWST is initialized at the conditions at t=1800 seconds (the end of Segment 1) from the RELAP5 analysis. When the liquid volume fraction of the BWST volume drops below the appropriate setpoint, Segment 2 of the FATHOMS analysis is terminated and sump recirculation mode is entered (Segment 3). This occurs at about 2808 seconds.

The water in the BWST volume is initialized at 115°F. The upper limit on the BWST water temperature is 110°F, but 115°F is used in this analysis for conservatism.

Heat Transfer Correlations

The Uchida heat transfer correlation is chosen for the concrete and steel heat slabs, with the exception of the building base. The Direct Uchida model from FATHOMS is chosen for the analysis as this option is appropriate for a large-scale calculational mesh. Reference 6-1 states that the Tagami or Uchida condensation heat transfer correlations are acceptable for pressure and temperature transient analyses. Since the Tagami correlation reverts to the Uchida correlation after blowdown, the Uchida correlation is used throughout the entire FATHOMS analysis.

A constant heat transfer coefficient of 20 Btu/hr-ft^2-°F is assumed for the building base. Since the building base is in contact with both the sump water and the building atmosphere, it is not appropriate to choose a heat transfer coefficient applicable for pool-to-concrete or atmosphere-

to-concrete heat transfer only. The assumption of 20 Btu/hr-ft^2-°F represents an average between these two heat transfer modes. During Segment 2 the sump region is over 90% liquid, so that almost all of the heat transfer is pool-to-concrete (or concrete-to-pool, since the sump water is cooled faster than the building base). The choice of a large heat transfer coefficient during these segments is therefore conservative, since the sump water will be heated by the building base during recirculation. It should be noted that the heat fluxes calculated for the building base are always several orders of magnitude smaller than those for the other large heat structures since the temperature drop is much smaller than in the atmosphere region. The building base heat transfer choice therefore has very little effect on the analysis.

Droplet Size of Blowdown Mass

A 100 μm value for the size of the blowdown droplets is again used. This parameter has very little impact on the long-term containment response since the blowdown period is over within a few minutes.

LPI Flowpaths and Mixing Volume

The LPI System has two trains with two separate LPI coolers. For some failure scenarios it is possible for only one cooler to be supplied with cooling water, although both will have LPI flow. To model this asymmetric configuration, two LPI flowpaths from the sump are modeled with separate LPI coolers. Downstream of the LPI coolers the two flows are then combined in a mixing volume with one outlet flowpath.

LPI Cooler Data

A cross-tie modification has been installed that connects the two LPI trains at the pump discharge. This modification deletes the requirement for manual throttling of LPI flow to 3000 gpm. Conservative LPI flowrates are now used in the methodology consistent with this design change. The following original text is maintained so that the analysis results presented can be understood.

Heat removal rates for the LPI coolers vary with primary and secondary side flow rates, tube plugging (heat transfer area), fouling and flowstream temperatures. A spreadsheet algorithm is used extensively in this analysis to determine LPI cooler heat removal rates. This spreadsheet takes existing LPI cooler test data taken during plant outages and extrapolates cooler performance for different values of each of the variables listed above.

LPI cooler maximum performance is nominally identified by the heat removal rate at the following conditions: 3000 gpm LPI flow rate, 250°F LPI inlet temperature (primary), 5000 gpm LPSW flow rate, and 90°F LPSW temperature (secondary). The minimum heat removal capacity allowed per current tube plugging criteria is 93.0×10^6 Btu/hr. This is the minimum used for this analysis. Intermediate values of 97×10^6 and 102×10^6 Btu/hr are also selected.

The FATHOMS/DUKE-RS models require an overall heat transfer coefficient, U, as a function of LPI temperature (primary inlet). This heat transfer coefficient will also vary with the LPSW

temperature (secondary inlet). Therefore, to model each of the three LPI cooler capacities above, a set of spreadsheet data is generated by taking existing LPI cooler test data and varying the LPI and LPSW inlet temperatures. The LPI temperature is varied from 250 to 110°F (and lower, when lower inlet temperatures are anticipated), and the LPSW temperature is varied from 90 to 55°F in 5° increments. This extrapolated data is then inserted in tables in the FATHOMS input. No actual test data exists for the 93.0 x 10^6 and 97 x 10^6 Btu/hr rates, therefore data for these rates is generated by lowering the number of unplugged tubes (heat transfer area) and/or increasing the fouling factor in the spreadsheet calculation. It should be noted that the heat removal rates above are never reached in the analysis, since the LPI flow rate is conservatively lowered from 3000 to 2685 gpm to allow for flow measurement inaccuracies.

RBCU Data

The heat removal rates used for the RBCU model assume an initial temperature of 110°F. This is a conservative assumption, since there would be more steam, and therefore higher RBCU heat removal rates, in the building atmosphere at an assumed initial temperature of 125°F (see Initial Building Temperature below). The data used for the FATHOMS input represents the heat removal rate for the RBCUs as a function of containment temperature and humidity. Since the mass of air in the Reactor Building is constant, the total pressure is established by the containment temperature and humidity. Therefore, FATHOMS input tables for the RBCU heat removal rate vary with the temperature and humidity, but were constant with respect to variations in pressure. These tables represent a forcing function multiplier on a constant conversion factor which converts the heat removal rate to Btu/sec for the FATHOMS code. This conversion factor is adjusted for each run to model the reduced heat removal capacity of the RBCUs due to fouling. A 50% cooler, then, would have the referenced heat removal capacities times 0.5. In the long-term large break LOCA analyses, the number of RBCUs refers to the capacity of the 2 operating RBCUs in terms of the equivalent number of nominal RBCUs, from 2 down to 0 RBCUs. (See Table 6.3-2)

The heat removal rate is altered for lower LPSW temperatures. Each reference value, corresponding to a LPSW temperature of 90°F, is increased by the following ΔT ratio:

$$Q_{RBCU} = Q_{RBCU_{90}} \left(\frac{T_{atm} - T_{LPSW}}{T_{atm} - 90} \right)$$

This adjusts for the higher ΔT across the cooler tubes for cases with LPSW temperatures below 90°F.

The air flow rate for the RBCUs is 1800 ft^3/sec, or 108,000 cfm, the total for 2 RBCUs (fouling has no effect on this value). This remains constant for all cases. The initiation time for the coolers remains unchanged from the values given in Section 2.4.

Core Decay Heat

The core decay heat energy is implemented in the long-term FATHOMS/DUKE mass and energy release model by adding this energy to the flow stream in the appropriate junction. Data for end-of-cycle conditions is used, and is taken from the ANSI/ANS-5.1-1979 standard (Reference 6-6). For additional conservatism, uncertainties (2σ) are calculated and added to the decay heat values.

The energy from the Group 1 heat structures in the RELAP5 mass and energy release analysis is accounted for by adding this energy to the decay heat. This is because these Group 1 heat structures are in contact with the RCS water rather than the containment atmosphere.

6.3.4 Initial Conditions

Initial Building Temperature

The initial building temperature, as well as the surface temperature for all heat structures, is assumed to be 125°F. It has been determined that 125°F is a conservatively high value for the inlet temperature to the Reactor Building Cooling Units (RBCUs), and therefore for the overall building.

Initial Pressure and Humidity

The initial building pressure and humidity are set at 14.7 psia and 100% respectively. Changes in these parameters would have very little impact on this analysis due to its long duration.

6.3.5 Boundary Conditions

Boundary conditions are implemented in FATHOMS by using junctions to inject or remove flows from the calculational mesh. There are eighteen junctions in the long-term FATHOMS large break LOCA model, with each assigned to a particular boundary condition. The elevations of these junctions are assigned to place them in the proper location in the Reactor Building model. Junctions which add or remove water from the building sump are placed in the lower node, while all others are located in the upper node.

A summary of the function of each junction/boundary condition with its period of applicability is given below. Note that all flow out the pipe break is divided by phase and by which side of the break the flow is coming from. When the long-term mass and energy release model is used, all liquid flow out the break is sent directly to the sump.

1.	Cold leg break - side #1 (vap)	Seg. 1	Injection mode
2.	Cold leg break - side #1 (liq)	Seg. 1	Injection mode
3.	Cold leg break - side #2 (vap)	Seg. 1	Injection mode
4.	Cold leg break - side #2 (liq)	Seg. 1	Injection mode
10.	Sump drain - to LPI cooler #1	Seg. 2	Recirculation mode
11.	LPI cooler #1 - to mixing volume	Seg. 2	Recirculation mode
12.	Sump drain - to LPI cooler #2	Seg. 2	Recirculation mode
13.	LPI cooler #2 - to mixing volume	Seg. 2	Recirculation mode
14.	Sump drain - to Building Spray	Seg. 2	Recirculation mode
15.	Building Spray flow - from sump	Seg. 2	Recirculation mode
16.	Building Spray flow - initial	Seg. 1	Injection mode
17.	Nitrogen injection from CFTs	Seg. 1	Injection mode
19.	From mixing volume - to core	Seg. 2	Recirculation mode
20.	From core - to containment (vapor phase)	Seg. 2	Recirculation mode
21.	From core - to containment (liquid phase)	Seg. 2	Recirculation mode

"Note: The following table is applicable to the Revision 0 FATHOMS model only, and is included so that the results of the Revision 0 analysis can be understood."

1.	Cold leg break - side #1 (vap)	Seg. 1	0 - 1800 sec.
2.	Cold leg break - side #1 (liq)	Seg. 1	0 - 1800 sec.
3.	Cold leg break - side #2 (vap)	Seg. 1	0 - 1800 sec.
4.	Cold leg break - side #2 (liq)	Seg. 1	0 - 1800 sec.
5.	BWST - flow out	Seg. 2	1800 - 2808 sec.
6.	Cold leg break (vap)	Seg. 2	"
7.	Cold leg break (liq)	Seg. 2	"
8.	Building Spray flow	Seg. 2	"
9.	LPI/HPI/BS discrepancy w/ RELAP5	Seg. 2	"
10.	Sump drain - to LPI	Seg. 3	2808 - 1.3E6 sec.
11.	Cold leg break (vap)	Seg. 3	"
12.	Cold leg break (liq)	Seg. 3	"
13.	Building Spray flow - from LPI (not used)	Seg. 3	"
14.	Sump drain - to Building Spray	Seg. 3	"
15.	Building Spray flow - from sump	Seg. 3	"
16.	Building Spray flow - initial	Seg. 1	0 - 1800 sec.
17.	Nitrogen injection from CFTs	Seg. 1	0 - 45 sec.
18.	LPI/HPI/BS discrepancy w/ RELAP5	Seg. 1	0 - 1800 sec.

6.3.5.1 Segment 1 Flow Paths

The following three subsections give details of the use of the junctions in the above table. The number of the appropriate junction(s) is shown in the subsection heading.

Cold Leg Break Flow Paths (1-4)

The mass flow rates, enthalpies, and pressures for the cold leg pipe break used during Segment 1 are taken directly from Table 3.6-1. A total of ten tables used for four boundary conditions are required for this mass and energy release data (four mass flow tables, four enthalpy tables, two pressure tables).

Building Spray Flow (16)

The Reactor Building Spray (BS) flow during Segment 1 is modeled with the single boundary condition 16. The BS flow uncertainty is subtracted from the nominal flow rate to provide a conservatively low value.

This water is added to the containment atmosphere at 115°F (assumed BWST water temperature). The spray flow is initiated at t=96 seconds, the sum of the approximate time to reach the spray initiation setpoint plus the documented time delay. This flow is terminated after Segment 1 (the end of the RELAP5 mass and energy release data). Building Spray is modeled with other boundary conditions after this point.

The spray droplet size assumed for all phases of the long-term large break LOCA analyses is increased from the documented value by one order of magnitude to 7000 μm for conservatism This decreases the surface area available for condensation of steam.

Nitrogen Cover Gas Injection (17)

Boundary condition 17 is included for the injection of the nitrogen cover gas from the core flood tanks (CFTs). A constant nitrogen flow rate of 356 lbm/sec is assumed from t=37.6 to 44.5 seconds for the injection of this gas. The pressure and temperature (655 psig and 120°F) for the nitrogen is the same as that in the CFTs from the RELAP5 analysis.

6.3.5.2 Segment 2 Flow Paths

LPI Flowpaths (10-13, 19-21)

At the start of the recirculation phase water is taken from the sump drain through boundary conditions 10 and 12, and then cooled by the two LPI coolers. Downstream of the LPI coolers the two LPI flows are then combined in a mixing volume via boundary conditions 11 and 13, and then the flow is directed to the core via boundary conditions 19-21. The [

] analysis are used to determine the quality of the flow exiting the break. This exit flow is again separated by phase through separate boundary conditions (20 and 21). The LPI flow rate is reduced by the instrument uncertainty value for conservatism

Building Spray (14-15)

Another set of boundary conditions models the BS flowpaths during Segment 2. This set is completely separate from the LPI flowpaths described above. Water is taken from the sump floor through boundary condition 14 and sprayed into the Reactor Building atmosphere through boundary condition 15. No LPI cooler or decay heat is associated with this set of flowpaths.

During recirculation, the nominal BS flow rate is reduced from 1500 to 1000 gpm. This is done to prevent NPSH problems during the recirculation phase, per Reference 6-2. Two sets of runs are executed with different uncertainty allowances for BS flow rates during recirculation. The first set assumes a 600 gpm flow rate in this phase, and the second set assumes 800 gpm. This flow is not routed through the LPI cooler or through the long-term mass and energy release model in the code. However, this option is available to the station during this phase.

Important initial conditions, boundary conditions, and assumptions for the long-term large break LOCA analysis are summarized in Table 6.3-1.

6.3.6 Results

A summary of all runs executed for this analysis is shown in Table 6.3-2. The Reactor Building atmosphere temperatures for each run are shown at the 1 day and 15 day points. The runs may be divided into two sets of cases; the first assumes a BS flow rate of 600 gpm (Cases 1 through 11), and the second assumes an 800 gpm BS flow rate during recirculation (Cases 12 to 25). These two sets are further broken down into sets assuming different LPI cooler heat transfer capacities and different LPSW temperatures.

Figure 6.3-1 shows the minimum RBCU capacities (in increments of 0.05 RBCU) required to meet the environmental qualification requirements for the limiting large-break LOCA for the minimum LPI cooler heat transfer capacity of 93 x 10^6 Btu/hr. This figure shows a profile of all the runs performed with this LPI cooler capacity. These minimum RBCU capacities are shown as a function of LPSW temperature. The lower the LPSW temperature, the less RBCU capacity is required. Two curves are shown; the upper curve is for cases with 600 gpm recirculation spray, and the lower curve is for cases with 800 gpm recirculation spray.

Figures 6.3-2 and 6.3-3 show the minimum RBCU capacities required with the LPI cooler heat transfer capacity increased to 97 x 10^6 and 102 x 10^6 Btu/hr, respectively. For some of the LPSW temperatures, the RBCU requirement is the same at a given LPSW temperature as shown in Figure 6.3-1. For many of the cases, this increase in LPI cooler capacity did not significantly decrease the required RBCU capacity. Note that the curve in Figure 6.3-1 for the 600 gpm BS flow rate is identical to that shown in Figure 6.3-2. An increase from 93 x 10^6 to 97 x 10^6 Btu/hr in the LPI coolers does not decrease the RBCU requirements at this spray flow rate.

The results shown on Figures 6.3-1 through 6.3-3 demonstrate the effectiveness of the Building Spray system in enhancing the ability of the LPI coolers to remove energy from the Reactor Building following a large-break LOCA. An increase from 600 to 800 gpm in the assumed spray flow rate in the recirculation phase decreases the required RBCU capability to meet EQ requirements for the entire range of LPSW temperatures. Also, this same increase allows the LPI coolers to operate more efficiently by transferring more energy from the Reactor Building atmosphere to the sump.

The vapor temperature response for Cases 1 and 11 is shown in Figure 6.3-4, along with the EQ envelope to be met in all cases. These two cases represent the minimum RBCU requirements at the extremes of the LPSW temperature ranges, therefore these cases bound all cases which meet the EQ requirements at the 600 gpm BS flow rate. It should be noted that Case 1 (90°F LPSW water and 0.9 RBCU) is at least 10°F below the EQ envelope but is bounded by the 20 day point. Case 11 (55°F LPSW water and 0.3 RBCU) falls below 125°F well before the 20 day point but comes close to the EQ envelope shortly after recirculation phase is initiated, and is bounded by the 100,000 second requirement. The data for Case 1 does not go out to 20 days, but is expected to meet the 20 day requirement because of the rate of decrease in the vapor temperature approaching the 15 day point. Any run which fell below the 125°F level before 15 days (such as Case 11) was terminated to conserve run time.

The Reactor Building pressure profiles for Cases 1 and 11 are shown in Figure 6.3-5. The building pressure increases in all cases after the initiation of recirculation mode but falls off with the decrease in decay heat levels. This increase is more pronounced in the cases with cooler LPSW water and lower RBCU requirements. The pressure stays higher in the final stages of the transient for cases with warmer LPSW water and higher RBCU requirements. These trends match the trends for the vapor temperature profiles. In all cases, the Reactor Building pressure profile meets the EQ long-term pressure requirements given in Reference 6-7.

The sump temperature profiles for the same two cases are shown in Figure 6.3-7. The sump temperature falls below 150°F after 1 day in all cases, and remains at least 60° subcooled after this point.

Figures 6.3-7 to 6.3-9 show the analogous curves for the previous figures for the cases with 800 gpm of recirculation spray. Vapor temperatures for Cases 12 and 26 are shown in Figure 6.3-8, with the EQ envelope again shown. These two cases also represent the minimum RBCU requirements at the extremes of the LPSW temperature range, therefore bounding all cases at the 800 gpm BS flow rate. These cases are not directly comparable with those shown in Figure 6.3-4 since different BS flow rates and RBCU capabilities are assumed. Similarly, the building pressure profiles and sump temperatures are shown in Figures 6.3-8 and 6.3-9.

A Reactor Building Energy inventory for Cases 1 and 11 is shown in Figures 6.3-10 and 6.3-11. This inventory shows the relative effectiveness of the RBCUs and LPI coolers in each case. The burden of long-term heat removal is equally shared in Case 1, but the LPI coolers remove most of the energy from the building in Case 11. The Reactor Building heat structures remove very little energy after the first day of the transient in all cases, and as the building cools down the structures start to transfer heat back into the building. It is also clear that by the time the recirculation phase is entered most of the energy in the Reactor Building is in the sump water.

In conclusion, it has been shown that the RBCU performance requirements necessary to meet EQ limits varies greatly with LPSW temperature. There is some dependence on LPI cooler heat transfer capability and BS flow rate also, but this dependence is not as significant. Seasonal changes in LPSW temperature dramatically impact the heat transfer requirements on the RBCUs.

6.4 Small Break LOCA Containment Response

6.4.1 Overview

The blowdown of the RCS inventory during a small break LOCA is obviously much slower than the large break LOCA, so the Reactor Building heat-up and pressurization is slower. However, certain phenomena in some of the RELAP5 small break LOCA analyses causes problems in the long-term cooldown of the building. These problems result from a loss of primary-to-secondary heat transfer for some break sizes. When this occurs, cooling through the break is insufficient to enable the desired cooldown and depressurization of the RCS. Then, when sump recirculation mode is entered, the RCS has not depressurized below the pressure at which the LPI pumps may come on. Therefore, the only flow through the LPI coolers is the suction flow provided to the HPI pumps by the LPI pumps (high pressure recirculation or "piggyback" mode). Since this flow is on the order of 800 gpm, much lower than the nominal 3000 gpm flow rate of a single LPI pump, cooling of the RCS and heat rejection from the containment are significantly limited.

The Reactor Building Spray system (BS) and the Reactor Building Cooling Units (RBCUs) are still present to aid in the Reactor Building cooldown. However, since the RCS will not depressurize to the Reactor Building pressure and high RCS temperatures may exist for an extended period of time, some quantity of the water exiting the break will flash to steam. This steam must be condensed by these systems for many days.

The same environmental qualification (EQ) requirements discussed in Section 6.3.1 must be met. Therefore, this accident must also be simulated for a period approaching a 20-day duration, as the Reactor Building temperature is required to return to its initial value of 125°F at this point in time.

The FATHOMS/DUKE-RS computer code is used to calculate the Reactor Building pressure and temperature response following this accident. The mass and energy released from the RCS is calculated by the RELAP5/MOD2-B&W code as discussed in Section 3. This data is used as input to the FATHOMS/DUKE-RS (or FATHOMS) code.

The goal of this analysis is to determine whether any small break LOCA scenario may require a higher RBCU capacity than the limiting large break LOCA analyzed in Chapter 4. The same containment parameters are varied in this analysis as in the previous section.

6.4.2 Analytical Approach

A set of six break sizes are analyzed:

Case 1 - 0.0025 ft^2
Case 2 - 0.005 ft^2
Case 3 - 0.01 ft^2
Case 4 - 0.025 ft^2
Case 5 - 0.05 ft^2
Case 6 - 0.1 ft^2

From inspection of the RELAP5 analyses, it is not necessary to determine a containment response for Cases 1 and 2. From observing the mass and energy release data, it is obvious that these cases do not challenge any long-term containment pressure or temperature requirements and are limited by the large break LOCA release. Therefore, containment runs are performed with FATHOMS/DUKE-RS for Cases 3 to 6 only.

The initial conditions and system flow rates for the analysis are chosen to give conservatively high containment pressures and temperatures, in accordance with ANSI/ANS-56.4-1983, Pressure and Temperature Transient Analysis for Light Water Reactor Containments (Reference 6-1). Many of these initial conditions are the same as those in the RELAP5 mass and energy release calculation.

Several different sets of flowpaths (junctions) are used in the analysis to implement the mass and energy release data supplied from the RELAP5 analysis and the data from the BFLOW code. Since sump recirculation mode is modeled in Cases 3 and 4, but not in Cases 5 and 6, different sets of junctions are required in these cases to remove or inject water from the proper volumes. Some flowpaths are active in certain segments of each case, while some are not.

entering the RCS from the HPI/LPI injection lines approximately equaling the volume of water exiting the break.

Only 15 days of the transient are modeled for Cases 3 and 4, as in the LBLOCA analyses discussed in Section 6.3. The vapor temperature at 20 days is extrapolated from the data for the first 15 days. The EQ requirement of 200°F at the 100,000 second point in the transient is also checked, although it is not expected that the SBLOCA containment temperature response will challenge this requirement.

For Cases 5 and 6 only one day of the transient is modeled. Because the RCS pressure for these cases is expected to be well below the LPI shutoff pressure of 175 psig, greater flow through the LPI coolers will occur and the transient will be similar to the LBLOCA response. The containment response to these cases should be well within the profiles for the LBLOCA responses.

As in the LBLOCA analyses, once it has been established that a run has met the EQ criteria for a given RBCU heat removal capacity and LPSW temperature, all cases at the same RBCU capacity and lower LPSW temperature, or at the same LPSW temperature and higher RBCU capacity, also meet the EQ criteria (all other parameters being the same).

The LPI cooler capacity will initially be held at the minimum value, 93 x 10^6 Btu/hr. A few runs are performed at higher LPI cooler capacities to ensure that the RBCU requirements for each small break LOCA scenario are limited by the corresponding large break LOCA case.

6.4.3 Modifications to Base Containment Model

Containment Free Volume and Passive Heat Sink Data

The containment free volume and passive heat sink values described in Section 6.2.3 are also used in this analysis.

Reactor Coolant System Volumes

BWST Volume

During Segment 1, the ECCS flow coming from the BWST and the flow exiting the break are modeled in the RELAP5 analysis. After the switchover to sump recirculation mode (during Segment 2), the source for the ECCS injection water will be the sump, so the BWST is not modeled.

Note: The following paragraph is superceded due to being applicable only to Revision 0 of the report. The text is retained for understanding the Revision 0 results only.

For Cases 5 and 6, the borated water storage tank (BWST) is modeled with a second calculational volume. In Cases 3 and 4, sump recirculation mode is entered well before the RELAP5 mass and energy release data is exhausted, so this is unnecessary. However, in Cases 5 and 6, the BWST is still draining when the RELAP5 mass and energy release data is exhausted. Therefore, a BWST model is utilized to determine when sump recirculation mode is to be entered. The BWST is initialized at the level it reached at the end of the RELAP5 analysis with water at 115°F for these cases. Until the end of the RELAP5 mass and energy release data, the core injection flow coming from the BWST and the flow exiting the break are modeled in the RELAP5 analysis. They are accounted for in the FATHOMS/DUKE-RS analysis through the pipe break boundary conditions. However, the RELAP5 analysis did not model the Reactor Building Spray (BS) system. The only water taken from the BWST in the RELAP5 analysis was for the HPI system. Therefore, the FATHOMS analysis takes BS water directly from the BWST for Cases 5 and 6. After the switchover to sump recirculation mode, the source for the core injection water will be the sump, so the BWST volume will be inactive.

Junction Descriptions

Note: The boron dilution line is no longer opened in the Revision 1 methodology. The description of the boron dilution line modeling and associated Revision 0 results are retained for historical purposes.

All junctions in the model but three are associated with boundary conditions. The break junction and boron dilution line flow paths are modeled as junctions, with their flow rates and velocities based on the flow equations in the FATHOMS code. However, these junctions are not active throughout the analysis. [

]

Likewise, the boron dilution line is closed until the 54,000 second point in the transient. At this time, its loss coefficient is also lowered to open the path. This is only done in Cases 3 and 4 - no credit is taken for this flow path in Cases 5 and 6.

Heat Transfer Correlations

The same heat transfer correlations described in Section 6.2.3 for the passive heat structures in the Reactor Building are chosen for the small break LOCA analyses.

RCS and Secondary Heat Structures

Droplet Size of Blowdown Mass

A 100 μm value for the size of the blowdown droplets is used. This parameter has very little impact on the long-term containment response, since the break flow is two-phase for only a short length of time.

LPI Cooler Data

The same spreadsheet algorithm used in the large break LOCA analysis is again used to determine LPI cooler heat removal rates. Different flow rates are utilized for small break analyses; in particular, much lower LPI flow rates are required since the LPI cooler primary flow is supplied by the HPI pumps alone for some portions of the small break simulations.

Several different LPI flow rates are assumed to create three different sets of data giving the overall heat transfer coefficient as a function of LPI and LPSW temperature. This is because the HPI and LPI pumps are aligned in piggyback mode after recirculation mode is entered, and the pump flow rates vary with RCS pressure. The three flow rates and their times of applicability are:

- 830 gpm Cases 1 and 2, before boron dilution line is opened
- 1220 gpm Case 1, after boron dilution line is opened
- 1400 gpm Case 2, after boron dilution line is opened

In Cases 3 and 4, RCS pressure falls below the LPI shutoff head once the boron dilution line is opened. This effectively increases the size of the break and allows more flow to exit the RCS. This in turn lowers RCS pressure and allows the LPI pump to inject more flow into the RCS. (Not all of this flow reaches the RCS; see above.) The higher flow rates are the approximate equilibrium flow rates that the HPI/LPI pumps attain once the boron dilution flow path is opened. RCS pressure drops even lower later in the transient, allowing even more injection flow into the RCS, but the LPI cooler heat transfer capacities are based on the above constant flow rates for conservatism.

In Cases 5 and 6, it is assumed that only the LPI pumps are injecting to the RCS once recirculation mode is entered. The break size in these cases is large enough to allow the LPI pumps to inject at the design flow rates once recirculation mode is entered. Therefore, the LPI cooler heat transfer data for these cases is the same as that for the large break LOCA analyses.

RBCU Data

Same as Section 6.3.3.

6.4.4 Initial Conditions

Same as Section 6.3.4.

6.4.5 Boundary Conditions

There are 14 junctions in the Revision 1 model, whereas there were 17 junctions in the Revision 0 model. Junctions 10, 11, and 17 have been deleted from the methodology. The description of these three deleted junctions in the table and text that follows is retained so that the Revision 0 results can be understood. Also, Junctions 10, 11, and 17 have an asterisk added and the following footnote is inserted at the bottom of the table. Note: "*" indicates this junction is deleted in Revision 1. A summary of the function of each junction/boundary condition with its period of applicability is given below. Note that all flow out the pipe break is again separated by phase.

1.	Pressurizer surge line	All
2.	Pipe break junction	After RELAP5 data end
3.	Pipe break (vap) (1 of 2)	During RELAP5 data
4.	Pipe break (liq) (1 of 2)	During RELAP5 data
5.	Pipe break (vap) (2 of 2)*	During RELAP5 data
6.	Pipe break (liq) (2 of 2)*	During RELAP5 data
7.	Building spray flow (initial)	From spray initiation to recirculation
8.	HPI spill (before recirculation)	From 0 to recirculation
9.	HPI suction (injection)*	From recirculation to RELAP5 data end
10.	HPI suction (spill)*	From recirculation to RELAP5 data end
11.	HPI spill (after recirculation)*	From recirculation to RELAP5 data end
12.	HPI pump suction (final)	After RELAP5 data end
13.	HPI pump discharge (injected)	After RELAP5 data end
14.	HPI pump discharge (spilled)*	After RELAP5 data end
15.	Recirculation spray (suction)	During recirculation
16.	Recirculation spray (discharge)	During recirculation
17.	Boron dilution flow path*	After 54,000 sec

Note: " * " These junctions are not utilized in Cases 5 and 6. Only one pair of junctions is required to accommodate the RELAP5 mass and energy release data for these cases. The HPI suction and spill lines after recirculation initiation are not required because the HPI pumps are shut off at this time. The recirculation lines for these cases are for LPI flow only; therefore no spill is modeled during recirculation.

Cold Leg Break Flow Paths (3 to 6)

The mass flow rates, enthalpies, and pressures for the cold leg pipe break used are taken directly from Tables 4.5-1 through 4.5-4.

Building Spray Flow (initial) (7)

The Building Spray flow path during the RELAP5 mass and energy release data period is modeled with the single boundary condition 7. The flow rate used for this boundary condition is

determined by subtracting the BS flow uncertainty from the nominal flow rate of 1500 gpm, as in the large break LOCA simulations.

This water is added to the containment atmosphere at 115°F (assumed BWST water temperature). The spray flow is initiated at a different time for each case. This time is approximately equal to the time to reach the 20 psig point in the Reactor Building plus the required time delay. This flow is terminated at the end of the RELAP5 mass and energy release data. Building Spray is modeled with other boundary conditions after this point.

HPI Spill (before recirculation) (8)

The water spilling out of the broken HPI line is modeled with this junction. This water is added to the sump region - no credit is taken for any condensation of steam by this water.

HPI Suction, Injection, and Spill Junctions After Recirculation Initiation (9, 10, 11)

These junctions are only utilized in Cases 5 and 6. In Cases 3 and 4 the RELAP5 data is exhausted before recirculation phase is entered, and HPI flow is not modeled.

In Cases 3 and 4, several hours of the transient during recirculation phase are modeled by RELAP5. The only input to FATHOMS from this analysis is the flow exiting the break. Since the recirculation phase has been entered, core injection water is being drawn from the sump. This must be accounted for in the FATHOMS analysis. Junction 9 removes all of the water which is to be injected into the core. This water is not routed anywhere, but is simply removed from the FATHOMS calculations. Junction 10 removes all of the water which is to be spilled out of the broken HPI line back into the sump. This flow rate is taken from the tables used for Junction 8 above, since during recirculation this water is coming from the building sump instead of the BWST. This water comes back into the sump at a cooler temperature than it exited (having passed through the LPI cooler), so an adjustment must be made in the boundary condition.

A volumetric flow rate table is used to specify the amount of water to be removed in Junction 10. This same table is used for Junction 11, putting the water back into the sump at 121°F, the assumed LPI cooler exit temperature during this interval. Because volumetric flows are used, putting the same volume of water back into the sump at a cooler temperature would mean a net gain in the mass of water in the sump. Therefore, the flow of cooler water put back into the

sump through the broken HPI line (Junction 11) is lowered by the approximate density ratio of 190°F water/121°F water (190°F is assumed to be the sump water temperature in this phase of the transient.). This splitting of junctions between the HPI water being injected into the core and the HPI water being spilled is necessary due to this temperature difference.

HPI Suction, Injection, and Spill Junctions After M&E Data Exhausted (12, 13, 14)

Recirculation Building Spray Flow Paths (15, 16)

Another set of boundary conditions models the BS flow paths during Segment 3. This set is completely separate from the LPI/HPI flow paths described above. Water is taken from the sump floor through boundary condition 15 and sprayed into the Reactor Building atmosphere through boundary condition 16. A conservatively low flow rate of 600 gpm is assumed for the

recirculation building spray. No LPI cooler or decay heat is associated with this set of flow paths.

No credit is taken for recirculation building spray in Cases 5 and 6. The LPI coolers and RBCUs are sufficient to keep containment pressures and temperatures below EQ limits for these cases.

Important initial conditions, boundary conditions, and assumptions for the small break LOCA analyses are summarized in Table 6.4-1.

6.4.6 Results

A summary of all runs executed for this analysis is shown in Table 6.4-2. The building atmosphere temperatures for each run are shown at the 1 day and 15 day points. The runs are divided into four sets of cases, one set for each break size.

The following are general discussions of each transient response for the four different cases. Table 6.4-3 shows a summary of events for the cases executed.

6.4.6.1 Case 3 - 0.01 ft^2 SBLOCA

A series of runs is performed for Case 3, the 0.01 ft^2 break. The interruption in primary-to-secondary heat transfer apparent at the end of the RELAP5 analysis is assumed to continue throughout the FATHOMS containment analyses, as no secondary system modeling is done in the FATHOMS analyses. The variations in the FATHOMS executions of Case 3 are in RBCU capability and LPSW temperature; these variations do not greatly alter the progression of the transient, although the initiation of building spray would obviously be different with each case. The first run of Case 3 assumes 0.90 RBCU and 90°F LPSW water. The initiation time for the Building Spray system for this case is conservatively used in all Case 3 runs, since the containment pressurization is slower for this case than the others analyses (delaying the spray initiation is conservative). The RBCU initiation time does not vary with each run, since each run is identical until the RBCU setpoint is reached. The Building Spray initiation time must be entered into a table and is not varied with each case.

The containment pressure and temperatures (Figures 6.4-1 and -2) increase slowly as the RCS inventory discharges into containment. The RBCUs initiate at 1707 seconds, but this does not greatly slow the increase in building pressure and temperature. The building pressure passes the

20 psig point at about 6200 seconds, and the BS system actuates at 6980 seconds (again, this delay is conservative). This results in an immediate decrease in building pressure and temperature. The sump temperature, shown in Figure 6.4-3, starts to increase at this point. The BWST reaches the switchover setpoint at 12,504 seconds. At this time the HPI and LPI pumps are aligned in piggyback mode to supply the RCS with cool water via the LPI coolers. However, the RCS pressure is still too high for any LPI flow, since primary-to secondary heat removal has been interrupted and the only injection flow is supplied by the HPI pumps. All of this flow still does not get to the RCS due to the assumption of a broken HPI line.

The RELAP5 analysis has an abbreviated containment model to supply an estimate for the sump temperature from the recirculation initiation point to the end of the RELAP5 analysis. This is necessary to determine the approximate temperature at which the injection water enters the core after passing through the LPI coolers. Conservatively low LPI cooler data is used in the RELAP5 analysis, so the temperatures calculated for this injection water are conservatively high. Therefore, the RELAP5 cooldown calculations for the RCS during this interval are conservative.

Containment pressures and temperatures increase slowly after recirculation initiation. This is because decay heat is at a relatively high level compared with the amount of injection flow reaching the core. The spray water at this point is coming from the sump instead of the BWST from this point on, and this decrease in condensation also contributes to the building heatup.

The degraded RBCUs and LPI cooler (with only HPI flow) are able to match decay heat at about 31,000 seconds and start to bring containment pressures and temperatures back down. It should be noted that RCS temperatures have been decreasing since the start of the transient, but this does not mean that there is a corresponding drop in containment temperatures until much later. Coincidentally, the RELAP5 analysis terminates at 32,811 seconds, shortly after this turn in containment conditions. The FATHOMS analysis continues to demonstrate this decrease in containment conditions as decay heat drops off.

It would be beneficial to get more flow through the LPI coolers and increase the rate at which energy is removed from the Reactor Building at this point. If the RCS pressure decreases to 175 psig, some LPI flow could be used in addition to the HPI flow passing through the coolers to accomplish this heat removal. However, nothing in the analysis suggests that this point will be reached for several days, based on the slow drop-off in RCS temperature. Therefore, credit will be taken here for the opening of the boron dilution line. This effectively increases the size of the break and allows the RCS pressure to decrease more rapidly to the LPI shutoff pressure.

Once the boron dilution flow path is opened, there is a temporary surge of flow out the break until the RCS pressure reaches 175 psia (Fig. 6.4-4), at which time the LPI pumps start. The containment pressure and temperature decrease picks up after the boron dilution flow path is opened. This is due to the increase in cold water being injected into the RCS. Although more total flow is entering the containment building, the cooldown results because the break flow is cooler.

For the run executed with 90°F LPSW water and 0.90 RBCU capacity, both the 1-day and 15-day EQ temperature requirements are met, as well as all pressure requirements. The containment vapor temperature for this run has decreased to 129.93°F at the 15-day point. For the given rate of decrease, this case will meet the EQ requirement of 125°F at the final point of 20 days, 16 hrs. Since this RBCU requirement (0.90) is identical to that required for the limiting large-break LOCA for an LPSW temperature of 90°F (Figure 6.3-1), the expectation that the 0.01 ft^2 small break LOCA is limited by the large break LOCA is verified for this LPSW temperature.

Additional runs are performed to verify the expectation that the small break LOCA containment response is bounded by the large break LOCA for the entire range of LPSW temperatures. In each case, the RBCU requirements to achieve EQ requirements are equal to or less severe than those for a large break LOCA. In addition, two runs with nominal LPI cooler capacities of 102×10^6 Btu/hr are performed to verify that the EQ requirements for these cases match those for corresponding LBLOCA FATHOMS runs performed previously. Again, in each case, the EQ requirements for this SBLOCA simulation are equal to or less severe than those for the corresponding LBLOCA.

6.4.6-2 Case 4 - 0.025 ft^2 SBLOCA

A series of runs is performed using the mass and energy release data for Case 4 for a SBLOCA with size 0.025 ft^2. This break is very similar in many respects to the 0.01 ft^2 SBLOCA discussed above. Because of the larger size break, more flow may be injected into the RCS once the boron dilution flow path is opened. This increases core cooling and makes the containment response to this size break slightly less severe than the 0.01 ft^2 break.

The RELAP5 analysis for Case 6 showed that this size break exhibits the same tendency to interrupt primary-to-secondary heat transfer with no assurance of recovery. Therefore, the same FATHOMS model and assumptions are used to determine the containment response. The RBCU and BS System setpoints are reached earlier than in the preceding break (647 and 3772 seconds, respectively) due to the greater break size. Again, the containment pressure and vapor temperature (Figures 6.4-5 and -6) start to decrease with the initiation of the Building Spray system, and the sump temperature (Figure 6.4-7) starts to increase faster. The RCS pressure and temperature trends (Figure 6.4-8) are similar to those in Case 3, with the RCS pressure stabilizing about 100 psi lower due to the larger break size. The sump recirculation switchover point is reached at 11,088 seconds. At this time, containment pressures and temperatures again increase until decay heat drops to the point where it is matched by the RBCU/LPI cooler capacity to remove heat.

The RELAP5 mass and energy release data is exhausted at 32,932 seconds. Again, loss coefficients are selected to match the break flow and RCS temperature trends leading up to this point. Containment pressures and temperatures continue to drop until the boron dilution line is again opened at the 54,000 second point. Here, a brief surge again occurs as the flow out the break increases, and then the decrease continues as the LPI pump adds more injection flow. For this break size, the LPI cooler heat transfer coefficients for the period after the opening of the boron dilution line are based on a total flow rate through the cooler of 1400 gpm. This is higher than the 1220 gpm used in the 0.01 ft^2 analysis above, again due to the larger break size. The larger break allows the system to stabilize at a slightly lower pressure, allowing slightly more flow to be injected into the RCS.

As with the 0.01 ft^2 break analysis, a series of runs are done for this break size to verify that this size break is limited by the large break LOCA with respect to EQ requirements for the entire range of LPSW temperatures. This was accomplished in only a few runs. It is apparent that this

break requires less RBCU capacity to meet long-term EQ requirements than the Case 3 break analyzed in Section 6.4.6-1.

6.4.6-3 Case 5 - 0.05 ft^2 SBLOCA

Another case is run using the mass and energy release from the RELAP5 analysis of a 0.025 ft^2 SBLOCA. Only 1 hour of data is provided, so additional flow paths for core injection prior to recirculation initiation are required for this case, as well as the one following (discussed above). Cases 5 and 6 are more similar in the containment response to a large break LOCA than the SBLOCAs previously discussed. The RCS pressure is below the LPI shutoff pressure of 175 psia by the time recirculation is entered, so there is no need for HPI/LPI piggyback mode. Since the LPI pump provides full injection flow (minus instrument error), the RCS inventory cools fairly rapidly, and the containment response is fairly mild.

Because of the larger size break, the RBCU and BS system initiation setpoints are reached quicker than the previous two cases. The RBCUs start for this case in 323 seconds, and the BS system in 1295 seconds. Again, containment pressures and temperatures (Figures 6.4-9 and -10) turn around and start to decrease soon after the Building Spray system begins to spray water into the containment building (the turnaround is less dramatic in this case than the previous two because of the greater flow rate out the break). The sump temperature (Figure 6.4-11) again increases at this point.

Sump recirculation mode is entered at 4509 seconds. RCS pressure drops to a level just above containment pressure at this point, so the mass flow rate out the break increases. However, the high LPI flow cools the RCS inventory quickly, and containment temperature has decreased to <170°F by the 1-day point. This is well under the corresponding temperature for the LBLOCA case with same RBCU capacity and LPSW temperature. No recirculation spray is assumed in this SBLOCA case, for additional conservatism. It is clear that there are no long-term EQ problems presented by the 0.05 ft^2 SBLOCA simulated in Case 5. This transient is limited by previously analyzed LBLOCAs.

6.4.6-4 Case 6 - 0.1 ft^2 SBLOCA

Just as the containment response for Case 4 closely matched the general trends from Case 3 above, the response for the 0.01 ft^2 SBLOCA simulated in Case 6 is very similar to that described above for Case 5. The mass and energy release data extends for 300 in this case. Again, the RBCU and BS setpoints are reached quickly. The RBCUs initiate at 148 seconds, and the Building Spray system begins to inject at 640 seconds. The increase in containment temperature and pressure (Fig. 6.4-14 and -15) are abated by the initiation of Building Spray again. No recirculation spray is assumed in this case. The sump temperature (Figure 6.4-16) increase at the recirculation initiation point is not as dramatic as in previous SBLOCA simulations because of the much greater break size in Case 6.

The RCS temperature and pressure profiles are shown in Figures 6.4-17 and 6.4-18. The RCS pressure stabilizes at about 60 psia for this break size. It is clear that this case presents no long-term EQ problems when compared with the response to a large-break LOCA.

6.4.6.5 Summary

The long-term containment response for a range of small break sizes has been determined. Figure 6.4-19 shows a composite of the response for the four sizes analyzes with 90°F LPSW assumed. For Cases 3 and 4, the minimum RBCU requirements to meet EQ criteria match the requirements for the corresponding large-break LOCA case: 0.90 RBCUs for 90°F LPSW water.

The similarity in the rates of long-term temperature decrease for the LBLOCA and SBLOCA cases is explained by the similarities in heat being added and heat being removed from containment in these cases. The decay heat data and RBCU heat removal capacities are obviously identical in both cases. There is steam being released into containment in both cases. The steam release in the LBLOCA is direct since the break is assumed to be a cold-leg break, while the release out the break in the SBLOCA is liquid at high pressure, some of which flashes to steam upon entering containment. Both cases assume recirculation spray, although this does not help much in the SBLOCA cases since the temperature difference between the containment atmosphere and sump is much smaller than in the LBLOCA case. The LPI cooler heat removal capacities are the same, but the flow rate through the cooler is smaller in the SBLOCA cases due to the elevated RCS pressure. Despite this, the LPI cooler removes heat at about the same rate in the SBLOCA and LBLOCA cases. In the SBLOCA analysis, the lower flow rate is compensated by the higher sump temperature.

For cases with high LPSW temperature, the RBCU requirements to meet EQ criteria for the SBLOCA are just under or the same as those for the LBLOCA discussed in Section 6-3. This is because of the similarities described above in these cases. These RBCU requirements are only applicable for the range of break sizes which may not allow the steam generators to be used as a heat sink, or not drop enough in pressure to allow full LPI flow. This occurred in the RELAP5/ FATHOMS analyses for the 0.01 ft^2 and 0.025 ft^2 break sizes (Cases 3 and 4). For Case 2, described in Section 4, it is stated that for smaller break sizes, the high point vent valves may be used if necessary to effectively restore primary-to-secondary heat transfer. For larger break sizes, Cases 5 and 6 from this analysis show that enough LPI injection flow can be attained to effectively cool the core and prevent any long-term equipment problems in the containment building. These cases are clearly bounded by the large break LOCA.

Therefore, it is concluded that the entire range of small break LOCAs analyzed here are limited with respect to containment response by the long-term large break LOCA previously analyzed. The small break LOCAs do require a reduction in the BS actuation setpoint and opening of the boron dilution flowpath for acceptable containment analysis results.

6.5 Steam Line Break Containment Analysis

6.5.1 Overview

The steam line break (SLB) inside the containment would result in an increase in the internal reactor building pressure and temperature. The magnitude of the pressure and temperature increase is primarily dependent on the size of the break, the steam superheat, and the length of the period during which feedwater addition to the affected steam generator continues. The double ended break of the 34" main steam line is the limiting SLB case for both the containment pressure and temperature.

The three energy removal mechanisms that limit the increase of the containment pressure and temperature are the passive heat sinks of the containment structural materials, the Reactor Building Cooling Units (RBCUs), and the Reactor Building Spray System (BS).

The mass and energy releases for this accident are calculated with the RETRAN-02 code described in Section 5.0. The steam line break is analyzed both with offsite power maintained and with offsite power lost coincident with reactor trip. In both cases the operator manually isolates the feedwater to the affected steam generator at 170 seconds.

6.5.2 Analytical Approach

The FATHOMS/DUKE-RS (or FATHOMS) computer code is used to calculate the peak containment pressure and temperature following this accident. The base model described in Section 2.4 is used, with minor modifications as discussed in Section 6.5.3. The initial and boundary conditions are selected to conform with ANSI/ANS-56.4-1983 (Reference 6-1) to give conservative pressure and temperature results. The mass and energy release tables for the steam line break analyses described in Section 5.0 are used as input. The acceptance criteria for this analysis are discussed in Section 6.1. The containment pressure should not exceed 59 psig and the containment temperature should remain below the equipment qualification (EQ) limit curve.

6.5.3 Modifications to Base Containment Mode

Containment Free Volume and Passive Heat Sink Data

Same as Section 6.2.3

Heat Transfer Correlations

Same as Section 6.2.3

Reactor Building Cooling Units (RBCUs)

Two RBCUs with a reduced cooling capacity of 50% each are simulated in the model. The LPSW temperature is conservatively assumed 90°F, and the air flow through the coolers is 1800 ft^3/sec. The actuation setpoint for both units is 9 psig with a 78 second time delay. The heat removal capacity of the coolers is a function of the containment pressure, temperature and humidity. The total cooling capacity for the RBCUs for this analysis is 73.6 x 10^6 Btu/hr, which assumes an LPSW temperature of 90°F.

6.5.4 Initial Conditions

Initial Pressure

An initial building pressure of 16.2 psia (1.5 psig) is assumed. Technical Specification 6.4 requires the Reactor Building to be below 1.2 psig during full power operation. This assumption is conservative since a higher initial pressure increases the pressure peaks.

Initial Temperature

A high initial building temperature of 125°F is conservative for the steam line break. Although it is known from the LOCA analyses (Section 6.3.4) that a lower initial building temperature maximizes the containment peak pressures due to the higher initial air mass, for the steam line break the situation is different. The higher initial temperature inside the containment reduces the cooling capacity of the heat sinks, outweighing the impact of a higher initial air mass.

Initial Relative Humidity

An initial humidity of 0% is used throughout the analysis. This assumption maximizes the initial air mass and is conservative. The variation in initial building humidity does not have a significant impact on the results of this analysis.

6.5.5 Boundary Conditions

The mass and energy data described in Section 5.0 are supplied to FATHOMS as a boundary condition. The data consists of the steam mass flow, and the steam enthalpy and pressure. Another boundary condition in the simulation is the containment spray, which enters the containment through the second junction in the model. The containment spray flow from the base model is used:

Containment Spray Flow: 1500 gpm - 143 gpm = 1357 gpm,

The 1500 gpm flow rate is the design flow assuming one containment spray flow in operation and 143 gpm is a correction to account for instrument errors. The actuation setpoint for the spray is 30 psig + 92 seconds time delay. A droplet size of 700 μm is assumed for the spray droplets. Important initial conditions, boundary conditions, and assumptions are summarized in Table 6.5-1.

6.5.6 Results

The steam line break accident is analyzed for two cases. In the first case offsite power is maintained, and in the second case offsite power is lost coincident with the reactor trip. Figures 6.5-1 through 6.5-3 correspond to the case with offsite power maintained, and Figures 6.5-4 through 6.5-6 show the results for the case with loss of offsite power. The results of both cases are summarized in the following table.

Case	Time of Peak Pressure (sec)	Peak Pressure (psig)	Peak Temperature (°F)
Offsite Power Maintained	200	58.96	483
Loss of Offsite Power	366	58	494

Offsite Power Maintained

This case assumes that the ICS is in automatic before the initiation of the steam line break. After the reactor/turbine trip the feedwater control valves close and later reopen to control the steam generator level. The level setpoint in this case is assumed to be 75 inches to account for possible uncertainties in level control. The operator manually closes the feedwater control valve to the affected steam generator at 170 seconds with a stroke time of 20 seconds, terminating the feedwater.

The peak containment pressure (Figure 6.5-1) is 58.96 psig and occurs at 200 seconds, approximately 10 seconds after the feedwater valves are completely closed. The RBCU actuation signal occurs at 1.7 seconds and the units are running at 80 seconds. The containment spray actuation signal occurs at 17 sec and the spray flow starts at 109 seconds. The simulation is run for 247 seconds.

The containment temperature (Figure 6.5-2) exceeds the EQ limit for approximately 200 seconds in the early part of the transient. The peak temperature during this period is 483°F at 38 seconds. Although the nominal acceptance criterion for the containment temperature is exceeded, the impact must be evaluated on a case-by-case basis for each individual type of equipment. Most of the equipment located inside the containment is designed for higher temperatures than are calculated in this analysis. For other equipment, the brief period of time during which the temperature limit is exceeded is evaluated to confirm that the heat transfer between the containment atmosphere and the internal parts of the equipment would provide sufficient protection to ensure the intended design function of the equipment.

The temperature decreases later as the cooling effect of the containment passive heat sinks begins to have an impact. A second peak is reached at approximately 200 seconds, but the temperature turns around as a result of the feedwater isolation and never exceeds the EQ limit during the second peak.

Figure 6.5-3 depicts the energy addition and the energy removal rates for this transient. The heat removal rate of the passive heat sinks increases rapidly early in the transient as the temperature of the containment atmosphere rises. There is a mismatch between the energy addition and the energy removal rate for most of the transient. At approximately 205 seconds the energy removal rate exceeds the energy addition rate, marking the turnaround point for the containment pressure.

At this point the mass and energy release through the break diminish due to the isolation of the feedwater. The RBCUs and the building spray have no significant impact on the mitigation of the consequences of the accident, and the SLB imposes no requirements on these systems. The timely isolation of feedwater is the key action required to meet the acceptance criteria during a steam line break.

Loss of Offsite Power

This case assumes a loss of offsite power coincident with the reactor trip. The RCS pumps, the feedwater, the D heater drain and the condensate booster pumps trip on loss of power. EFW is automatically started on loss of main feedwater pumps at the time of reactor trip. The feedwater block valves are unavailable due to the load shed. The operator manually closes the MFW and EFW control valves to the affected steam generator at 170 seconds with a stroke time of 20 seconds.

The peak containment pressure (Figure 6.5-4) is 58 psig at 366 seconds. The containment temperature (Figure 6.5-5) exceeds the EQ limit for approximately 180 seconds in the early part of the transient and peaks at 494°F at 38 seconds. As discussed previously, the impact of the containment temperature on the selected equipment located inside the containment must be evaluated on a case-by-case basis. Between 350 - 400 seconds the containment temperature comes very close to the EQ limit, but a close examination shows that the limit is not exceeded.

The RBCUs setpoint is reached at 1.7 seconds and the units are activated at 80 seconds. The spray setpoint is reached at 17 seconds and the spray flow is initiated at 109 seconds.

The simulation is run for 570 seconds. The energy addition and removal rates are shown in Figure 6.5-6. The passive heat sinks constitute the most significant cooling mechanism throughout the transient. The energy removal rates of the RBCUs and the spray are not adequate to significantly affect the containment pressure and temperature during the steam line break. Both systems are intended for the long term cooling of the containment. The requirements imposed by the steam line break accident are met by timely isolation of feedwater to the affected steam generator.

Summary

The steam line break inside containment requires manual isolation of the feedwater to the affected steam generator within 170 seconds after the occurrence of the accident. Feedwater isolation in combination with the cooling capacity of the passive heat sinks assures that the containment pressure does not exceed 59 psig and the containment temperature does not impact the environmental qualification of equipment located inside the containment.

Two double-ended break cases of the main steam line are presented. In one case offsite power is maintained throughout the transient and in the second case offsite power is lost coincident with the reactor trip. In both cases the containment pressure remains below 59 psig and the containment temperature remains below the EQ limit, with the exception of an early peak during which the nominal limit is exceeded for approximately 200 seconds. The double ended break of the main steam line is the most limiting steam line break and is bounding for all other break sizes.

The steam line break differs significantly from the design basis LOCA. While less significant in terms of the initial blowdown, it has the potential for continued addition of feedwater to the affected steam generator and continued steaming for an extended period of time until feedwater is isolated.

The passive heat sinks play a major role in the cooling and the condensation of the steam released during the steam line break. Both the Reactor Building cooling units and the Reactor Building Spray System play insignificant roles in the mitigation of the consequences of the steam line break, and are primarily intended for the long term cooling of the containment. Even with the combined design cooling capacity of both systems, the containment pressure cannot be maintained below 59 psig without feedwater isolation occurring as described above.

6.6 References

6-1 ANSI/ANS-56.4-1983, "American National Standard Pressure and Temperature Transient Analysis for Light Water Reactor Containments", American Nuclear Society.

6-2 NUREG-0588, Interim Staff Position on Environment Qualification of Safety-Related Electrical Equipment, U. S. NRC, December 1979

6-3 NUREG/CR 3262, "COBRA-NC: A Thermal Hydraulics Code for Transient Analysis of Nuclear Reactor Components", Volumes 1 to 7, Wheeler, Thurgood, Guidotti, and DeBellis, Pacaific Northwest Laboratory, April 1986.

6-4 GOTHIC Containment Analysis Package - User's Manual, Numerical Applications, Inc., Version 3.4, April 1991.

6-5 NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", Nuclear Regulatory Commission, December 1979.

6-6 ANSI/ANS-5.1-1979, "Decay Heat Power in Light Water Reactors", American Nuclear Society.

6-7 Duke Power Company Response to I.E. Bulletin 79-01B, Revision 8, January 1988.

6-8 "Drop Behavior in GOTHIC," NAI-9301-7, Numerical Applications, Inc., January 10, 2001

(pages 6-49 through 6-51 intentionally deleted)

TABLE 6.2-1

Summary of Initial Conditions and Principal Input Assumptions
Post-LOCA Peak Pressure - Containment Analysis

Initial pressure:	1.5 psig
Initial building temperature:	110 °F
Initial relative humidity:	0%
Containment free volume:	Nominal - 2%
Passive heat sink volumes:	Nominal - 1%
No. of RBCUs operating:	NONE
Total RBCU cooling capacity:	N/A
RBCU setpoint:	N/A
LPSW temperature:	N/A
No. of containment spray trains:	NONE
Spray setpoint:	N/A
Spray flow :	N/A

(pages 6-50 and 6-51 intentionally deleted)

TABLE 6.3-1

Summary of Initial Conditions and Principal Input Assumptions
Long-Term Large Break LOCA - Containment Analysis

Initial pressure:	0.0 psig
Initial building temperature:	125 °F
Initial relative humidity:	0%
Containment free volume:	Nominal - 2%
Passive heat sink volumes:	Nominal - 1%
No. of RBCUs operating:	VARIABLE
Total RBCU cooling capacity:	VARIABLE
RBCU setpoint:	9 psig + 78 seconds delay time
LPSW temperature:	VARIABLE
No. of containment spray trains:	One
Spray setpoint:	30 psig + 92 seconds time delay
Spray flow :	1357 gpm (600 or 800 gpm during sump recirculation mode)

Case	INPUT					OUTPUT		
No.	LPI	RBCU	LPSW	BS		T(1 day)	T(15 days)	EQ Criteria
	(Btu/hr)		(F)	(gpm)		(F)	(F)	Met?
1	93 E6	0.9	90	600		187.67	130.47	Yes
2	102 E6	0.85	90	600		188.45	130.53	Yes
3	93 E6	0.7	85	600		195.05	130.72	Yes
4	102 E6	0.65	85	600		196.18	130.87	Yes
5	93 E6	0.55	80	600		201.97	130.74	Yes
6	93 E6	0.5	75	600		202.58	127.91	Yes
7	93 E6	0.45	70	600		203.58	125.03	Yes
8	102 E6	0.4	70	600		202.58	125.33	Yes
9	93 E6	0.4	65	600		205.03	<125.0	Yes
10	102 E6	0.35	60	600		203.92	125.29	Yes
11	93 E6	0.35	55	600		203.76	<125.0	Yes
12	93 E6	0.85	90	800		186.62	129.62	Yes
13	97 E6	0.8	90	800		188.61	130.07	Yes
14	102 E6	0.75	90	800		189.91	130.45	Yes
15	93 E6	0.65	85	800		193.66	129.56	Yes
16	97 E6	0.6	85	800		195.68	130.22	Yes
17	93 E6	0.5	80	800		200.17	129.20	Yes
18	97 E6	0.45	80	800		202.73	130.06	Yes
19	93 E6	0.45	75	800		200.59	126.02	Yes
20	97 E6	0.4	75	800		203.38	127.00	Yes
21	93 E6	0.4	70	800		201.38	< 125	Yes
22	102 E6	0.35	70	800		203.53	< 125	Yes
23	93 E6	0.35	65	800		203.51	< 125	Yes
24	102 E6	0.3	65	800		203.93	< 125	Yes
25	97 E6	0.3	60	800		202.67	< 125	Yes
26	93 E6	0.3	55	800		201.02	< 125	Yes
27	102 E6	0.25	55	800		202.55	< 125	Yes

TABLE 6.3-2 - FATHOMS Runs For Minimum Long-term RBCU Requirements

TABLE 6.4-1

Summary of Initial Conditions and Principal Input Assumptions
Small Break LOCA - Containment Analysis

Initial pressure:	0.0 psig
Initial building temperature:	125 °F
Initial relative humidity:	0%
Containment free volume:	Nominal - 2%
Passive heat sink volumes:	Nominal - 1%
No. of RBCUs operating:	VARIABLE
Total RBCU cooling capacity:	VARIABLE
RBCU setpoint:	9 psig + 78 seconds delay time
LPSW temperature:	VARIABLE
No. of containment spray trains:	One
Spray setpoint:	20 psig + 92 seconds time delay
Spray flow :	1357 gpm (600 gpm during sump recirculation mode)

Case	Run	INPUT					OUTPUT		
No.	No.	LPI	RBCU	LPSW	Recirc BS		T(1 day)	T(15 days)	EQ Criteria
		(Btu/hr)		(F)	(gpm)		(F)	(F)	Met?
1	1	93 E6	0.9	90	600		187.31	129.93	Yes
	2	102 E6	0.85	90	600		187.44	129.58	Yes
	3	93 E6	0.7	85	600		193.34	129.07	Yes
	4	102 E6	0.65	85	600		193.65	128.62	Yes
	5	93 E6	0.55	80	600		198.61	127.8	Yes
	6	93 E6	0.45	75	600		202.28	125.75	Yes
	7	93 E6	0.4	70	600		202.98	<125	Yes
	8	93 E6	0.35	60	600		200.71	<125	Yes
2	9	93 E6	0.85	90	600		187.34	129.35	Yes
	10	93 E6	0.65	85	600		193.18	128.09	Yes
	11	93 E6	0.4	80	600		198.69	126.67	Yes
	12	93 E6	0.3	70	600		200.72	<125	Yes
3	13	93 E6	0.8	90	0		152.03	-	Yes
4	14	93 E6	0.8	90	0		167.55	-	Yes

TABLE 6.4-2 - FATHOMS Runs For Small Break LOCA Analyses

TABLE 6.4-3

SBLOCA - Summary of Events

Case 1 (0.01 ft^2 break)	Case 2 (0.025 ft^2 break)	Case 3 (0.05 ft^2 break)	Case 4 (0.1 ft^2 break)	Event
0.0	0.0	0.0	0.0	Break initiated
212	~ 80	~ 35	~ 15	Time to 3 psig in Rx. Building
1707	647	323	148	RBCUs are activated
6980	3772	1295	640	RBS is activated
12,504	11,088	4509	4085	Switch to sump recirculation
32,811	31,851	3600	3000	(End of RELAP5 M&E data)
54,000	54,000	-	-	Boron dilution line opened
1,296,000	1,296,000	86,400	86,400	Job terminated

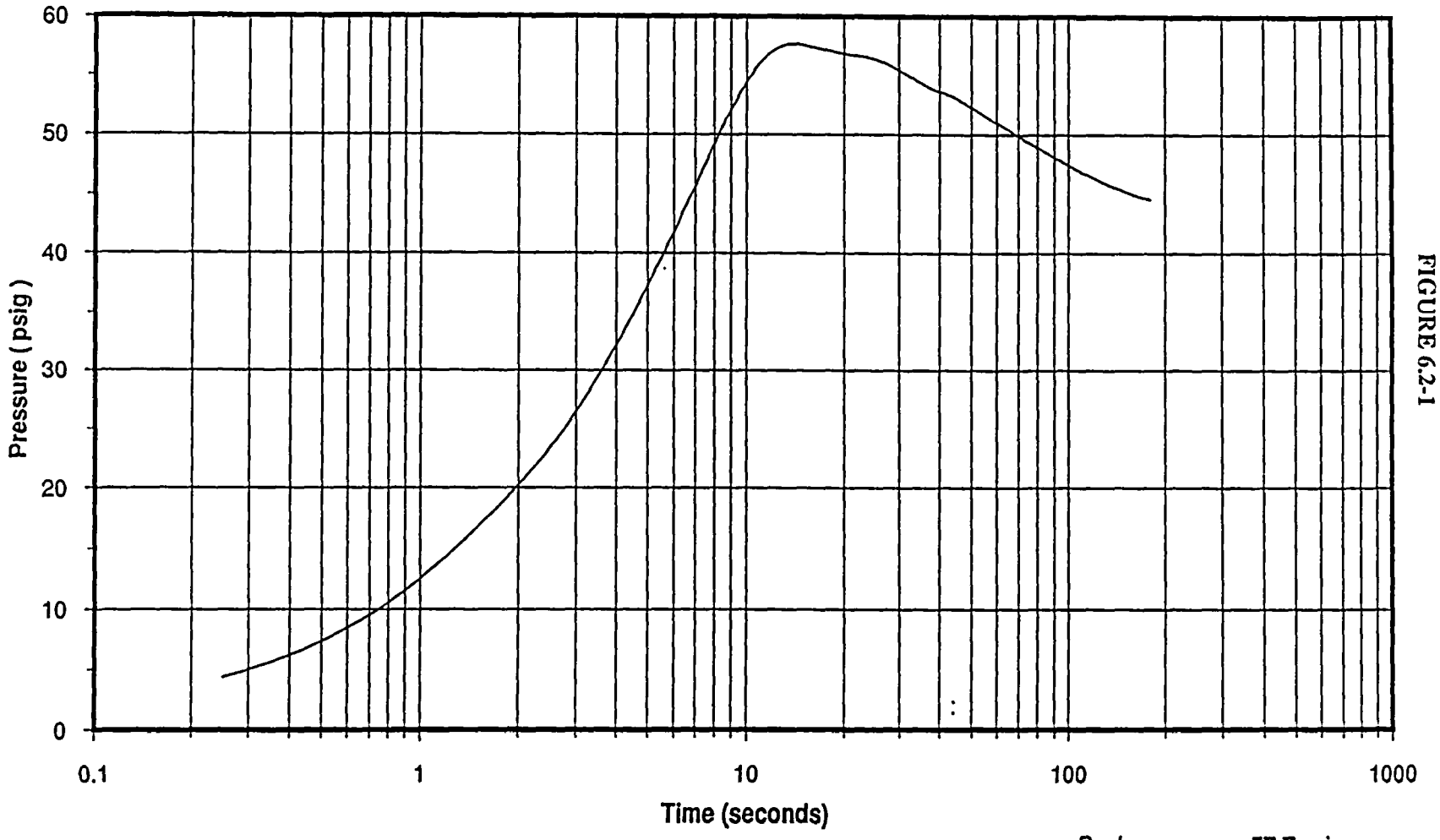
Note: All times in seconds

TABLE 6.5-1

Summary of Initial Conditions and Principal Input Assumptions
Steam Line Break - Containment Analysis

Initial pressure:	1.5 psig
Initial building temperature:	125°F
Initial relative humidity:	0%
Containment free volume:	Nominal - 2%
Passive heat sink volumes:	Nominal - 1%
No. of RBCUs operating:	Two at 50% capacity
Total RBCU cooling capacity:	73.6×10^6 Btu/hr
RBCU setpoint:	9 psig + 78 seconds delay time
LPSW temperature:	90°F
No. of containment spray trains:	One
Spray setpoint:	30 psig + 92 seconds time delay
Spray flow :	1357 gpm

Oconee Peak Pressure Analysis
Case 1A - Building Pressure Profile



Peak pressure = 57.7 psig

FIGURE 6.2-1

Oconee Peak Pressure Analysis
Case 1B - Building Pressure Profile

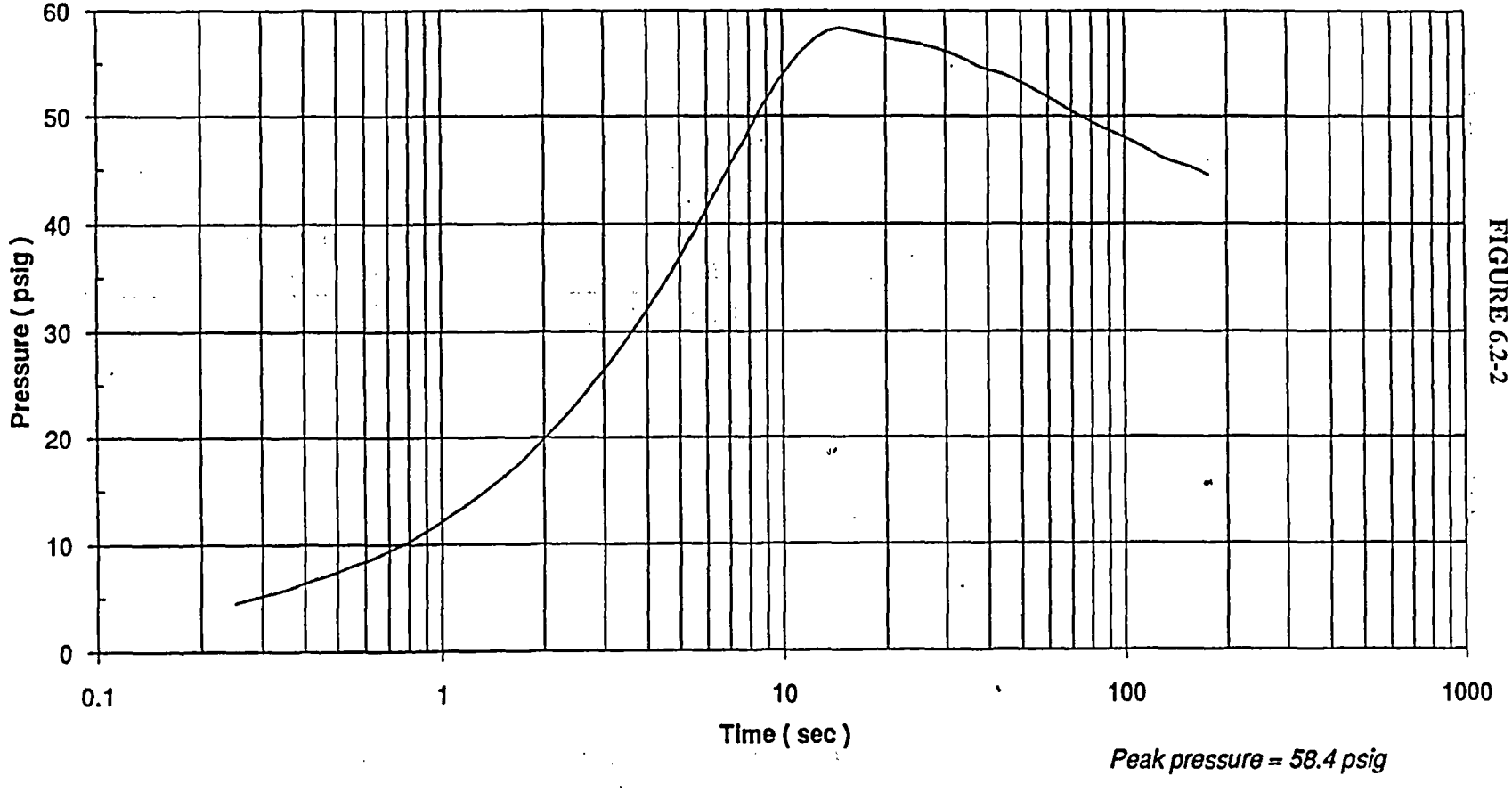


FIGURE 6.2-2

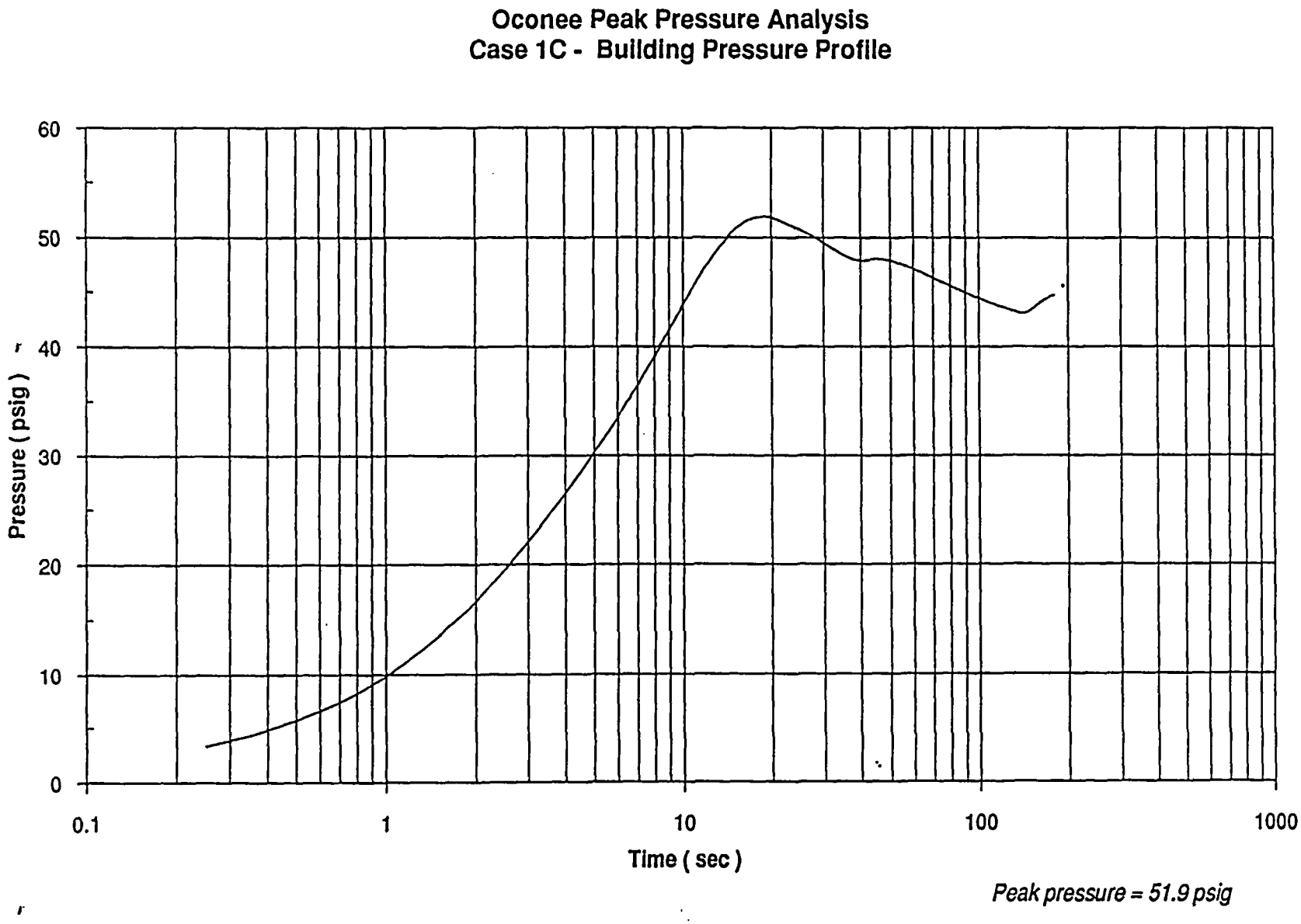
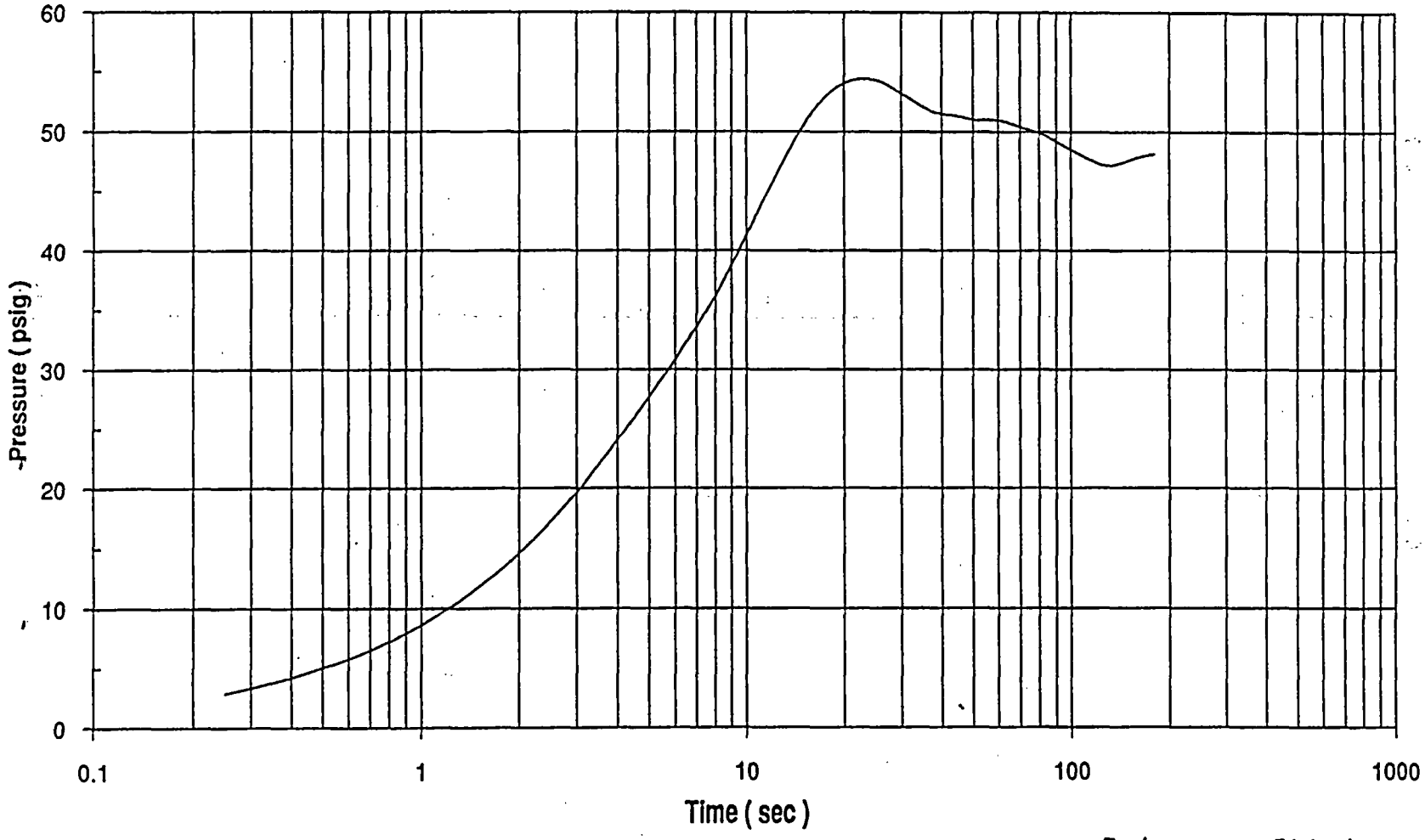

Oconee Peak Pressure Analysis
Case 1C - Building Pressure Profile
Pressure (psig)
60
50
40
30
20
10
0
0.1
1
10
100
1000
Time (sec)
Peak pressure = 51.9 psig

FIGURE 6.2-3

Oconee Peak Pressure Analysis
Case 1D - Building Pressure Profile



Peak pressure = 54.4 psig

FIGURE 6.2-4

Oconee Peak Pressure Analysis
Case 1A - Building Temperature Profile

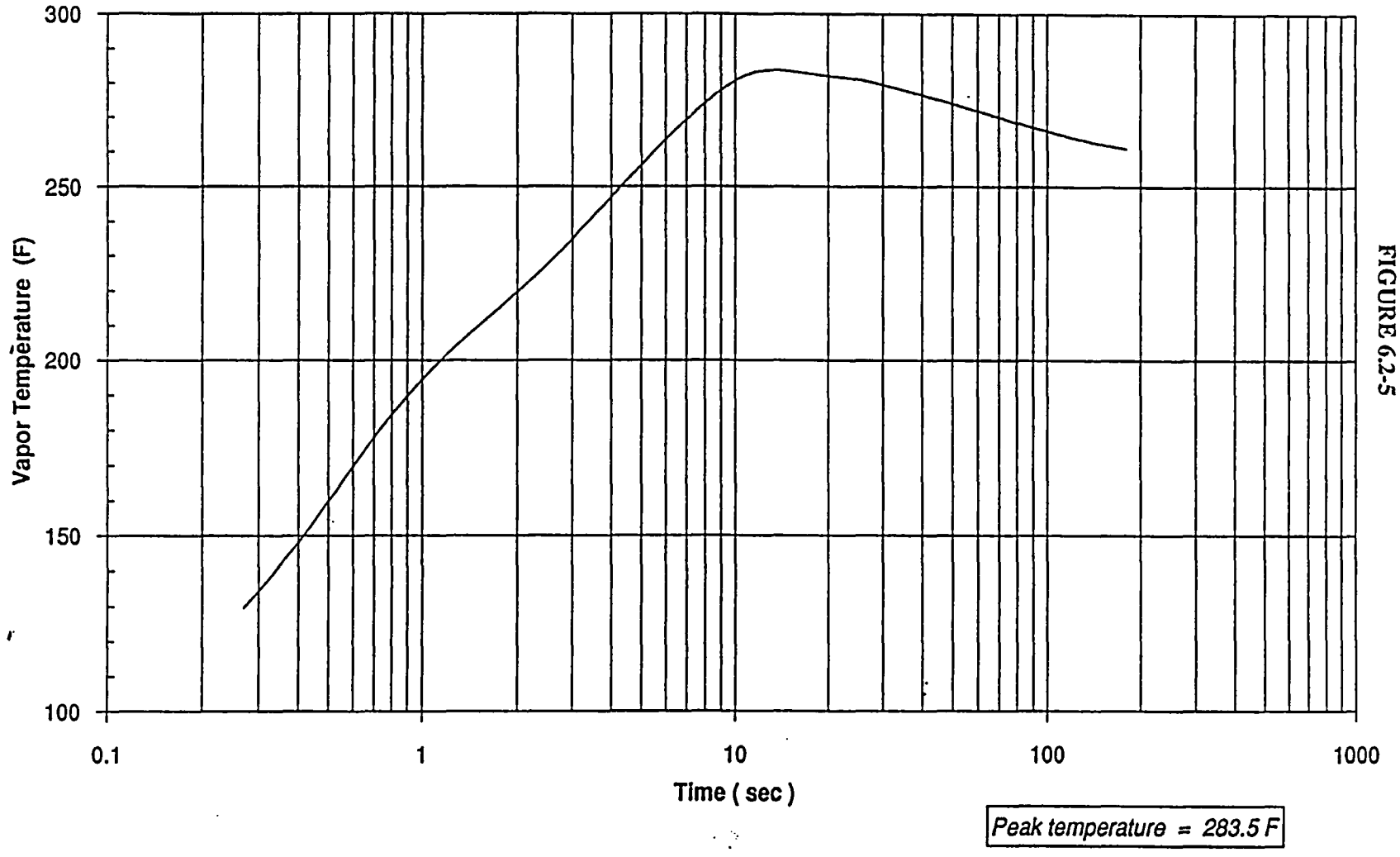


FIGURE 6.2-5

Oconee Peak Pressure Analysis
Case 1B - Building Temperature Profile

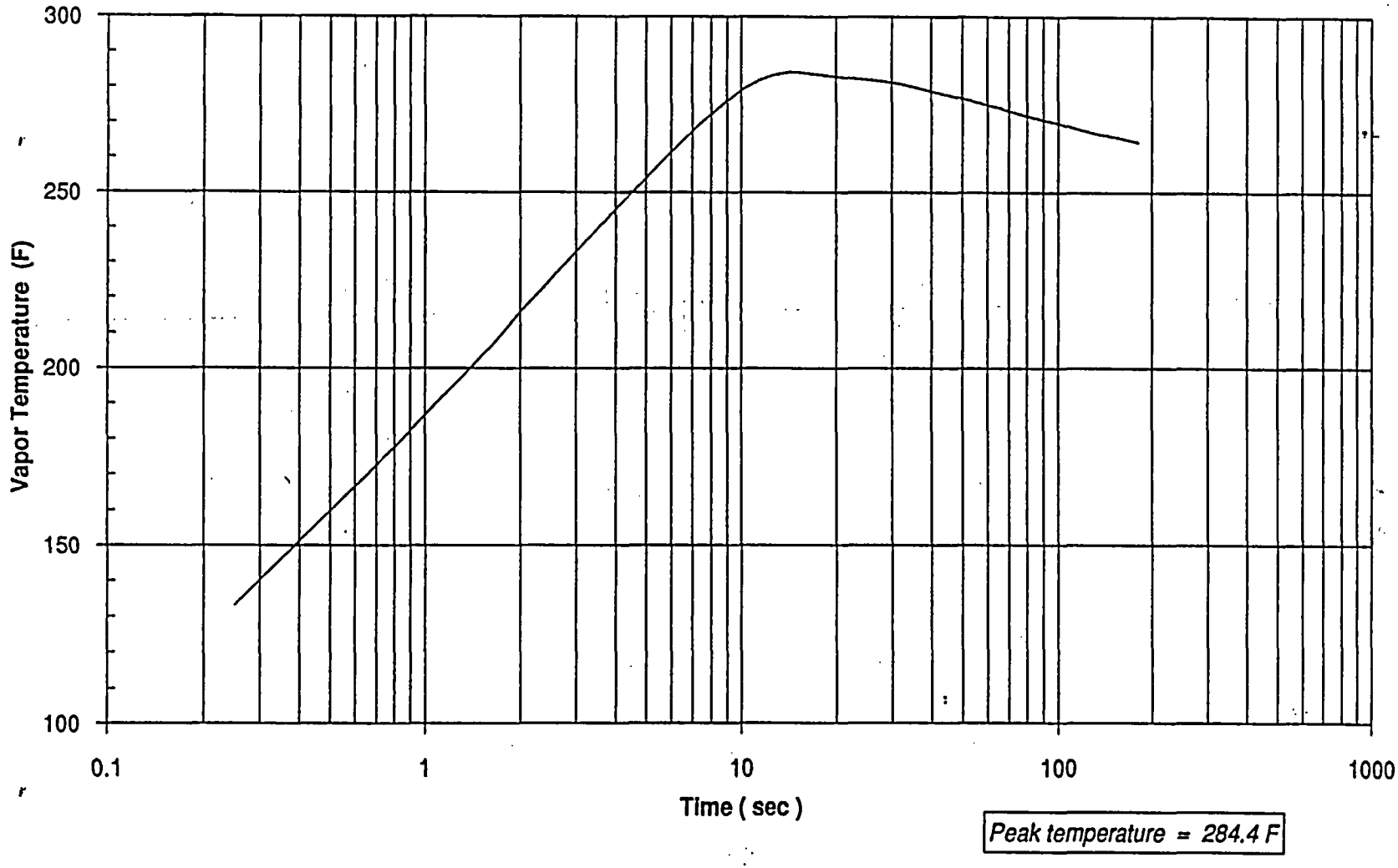


FIGURE 6.2-6

Oconee Peak Pressure Analysis
Case 1C - Building Temperature Profile

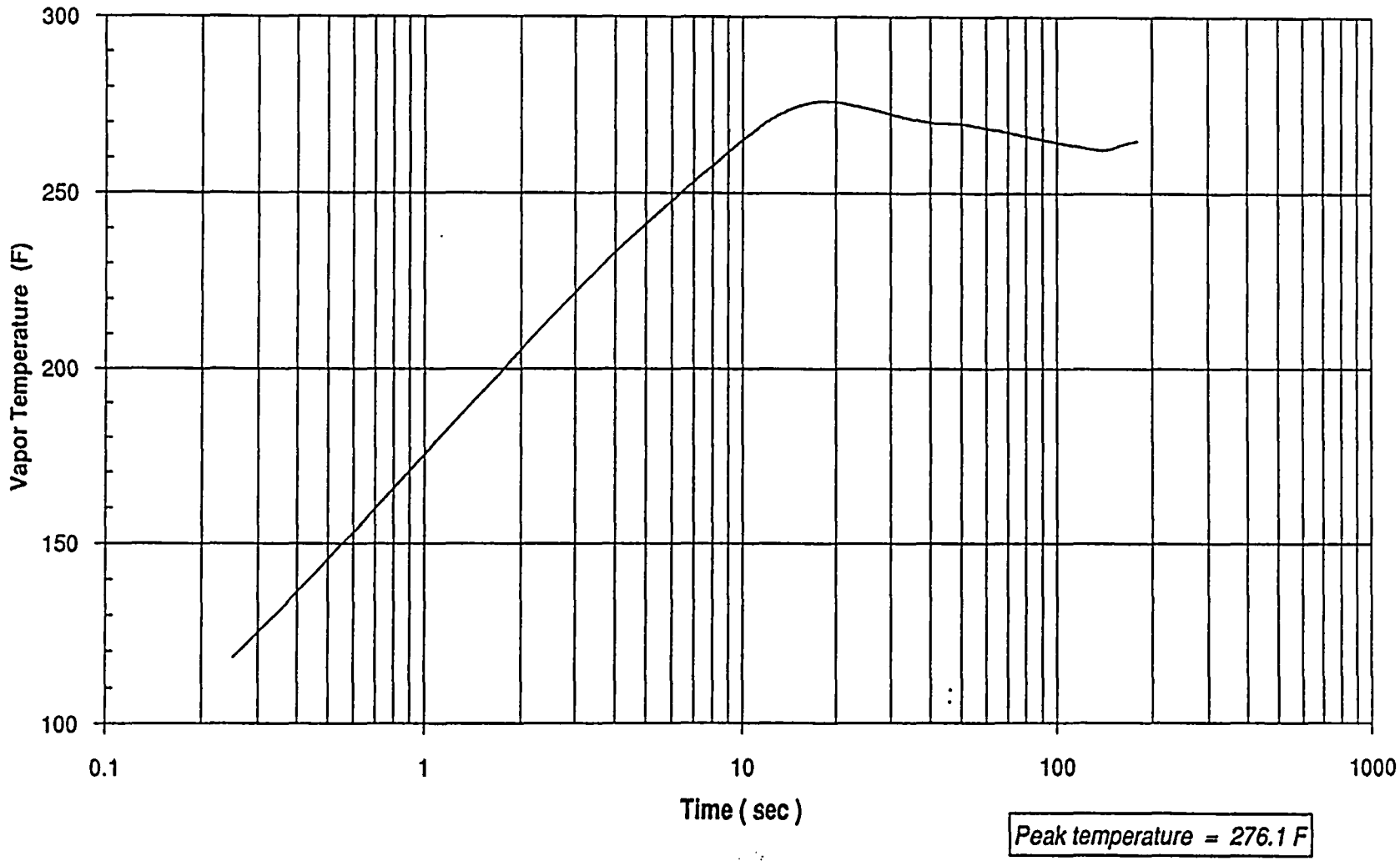


FIGURE 6.2-7

Oconee Peak Pressure Analysis
Case 1D - Building Temperature Profile

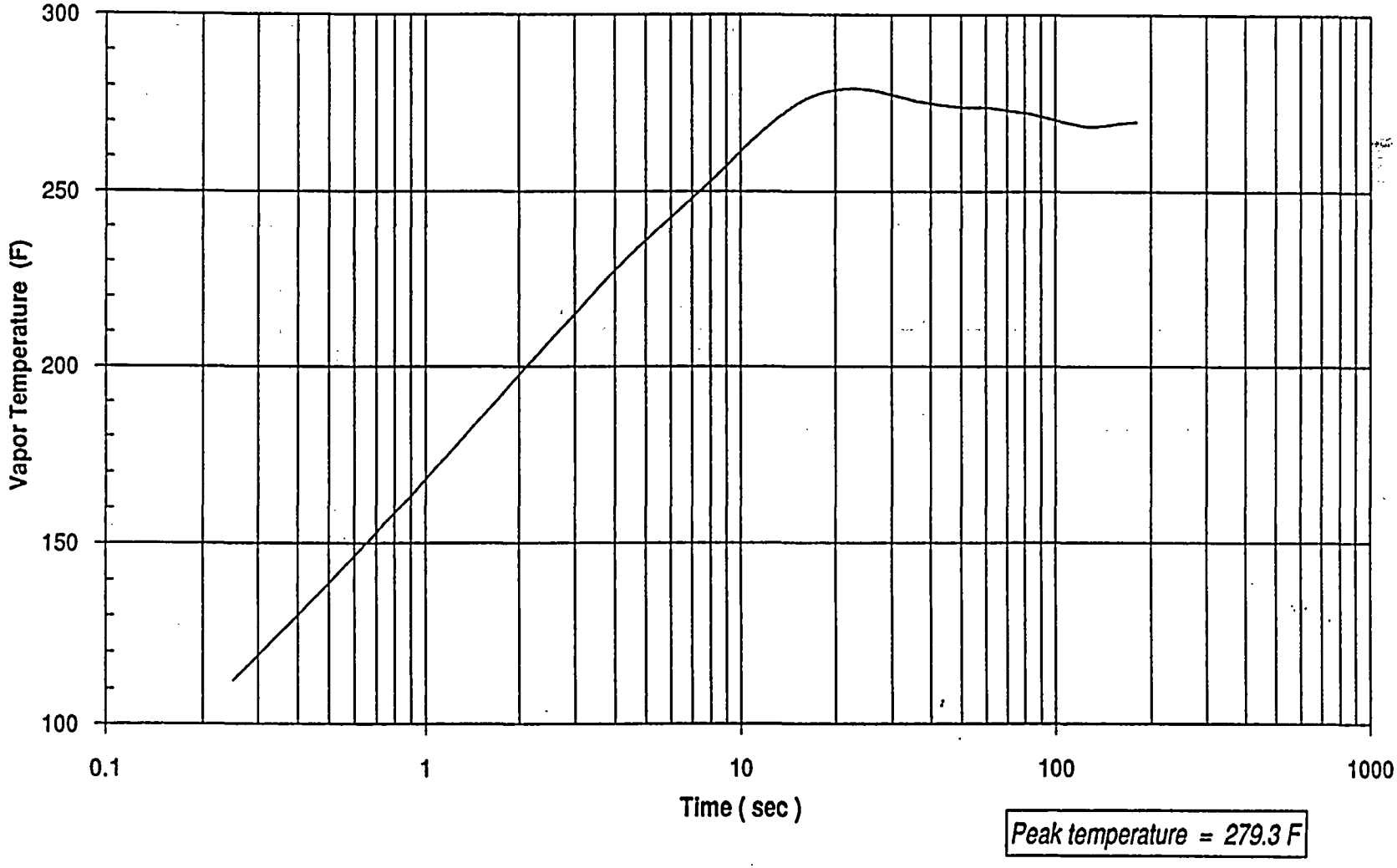


FIGURE 6.2-8

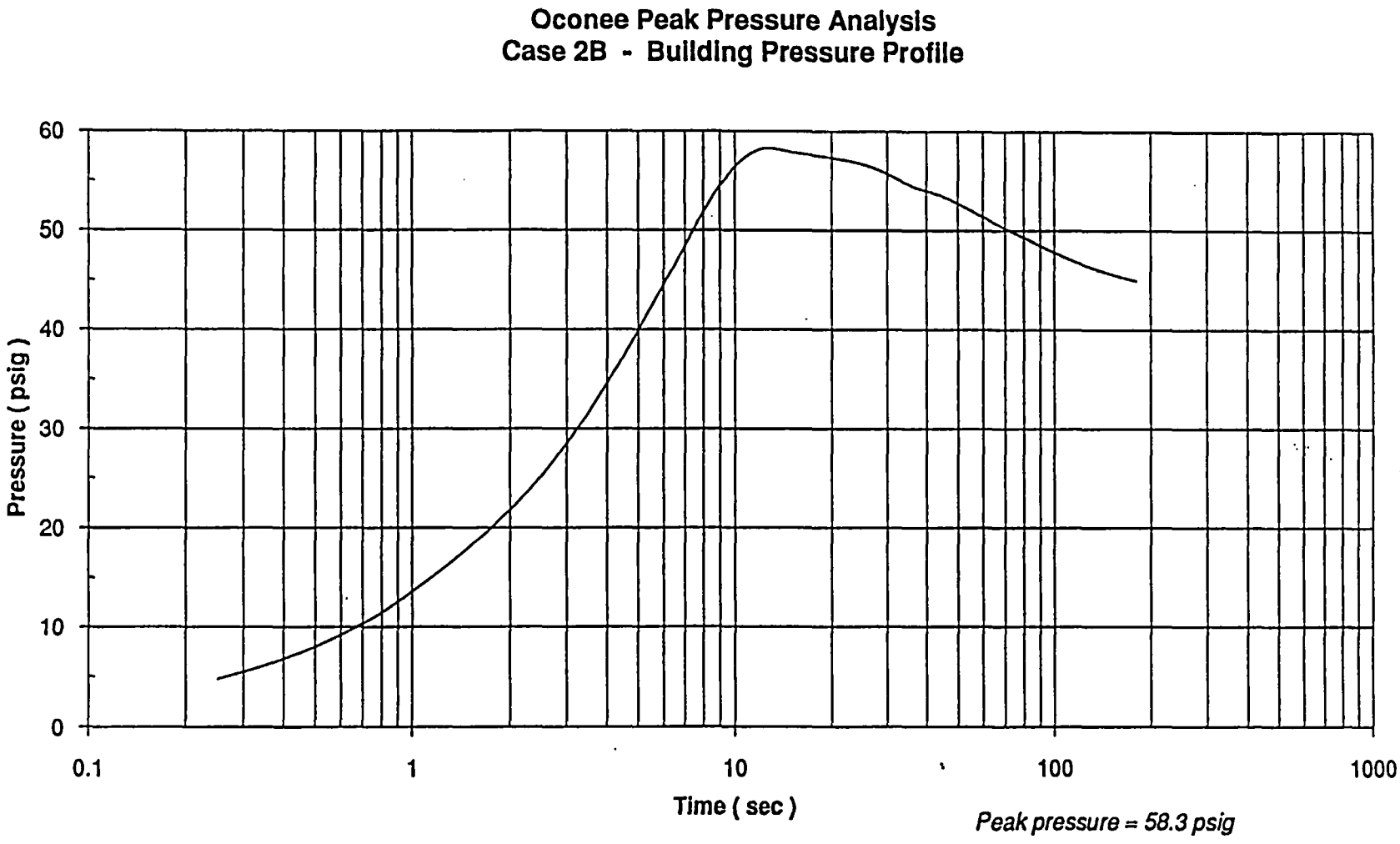

Oconee Peak Pressure Analysis
Case 2B - Building Pressure Profile
Pressure (psig)
60
50
40
30
20
10
0
0.1
1
10
100
1000
Time (sec)
Peak pressure = 58.3 psig

FIGURE 6.2-9

Oconee Peak Pressure Analysis
Case 2D - Building Pressure Profile

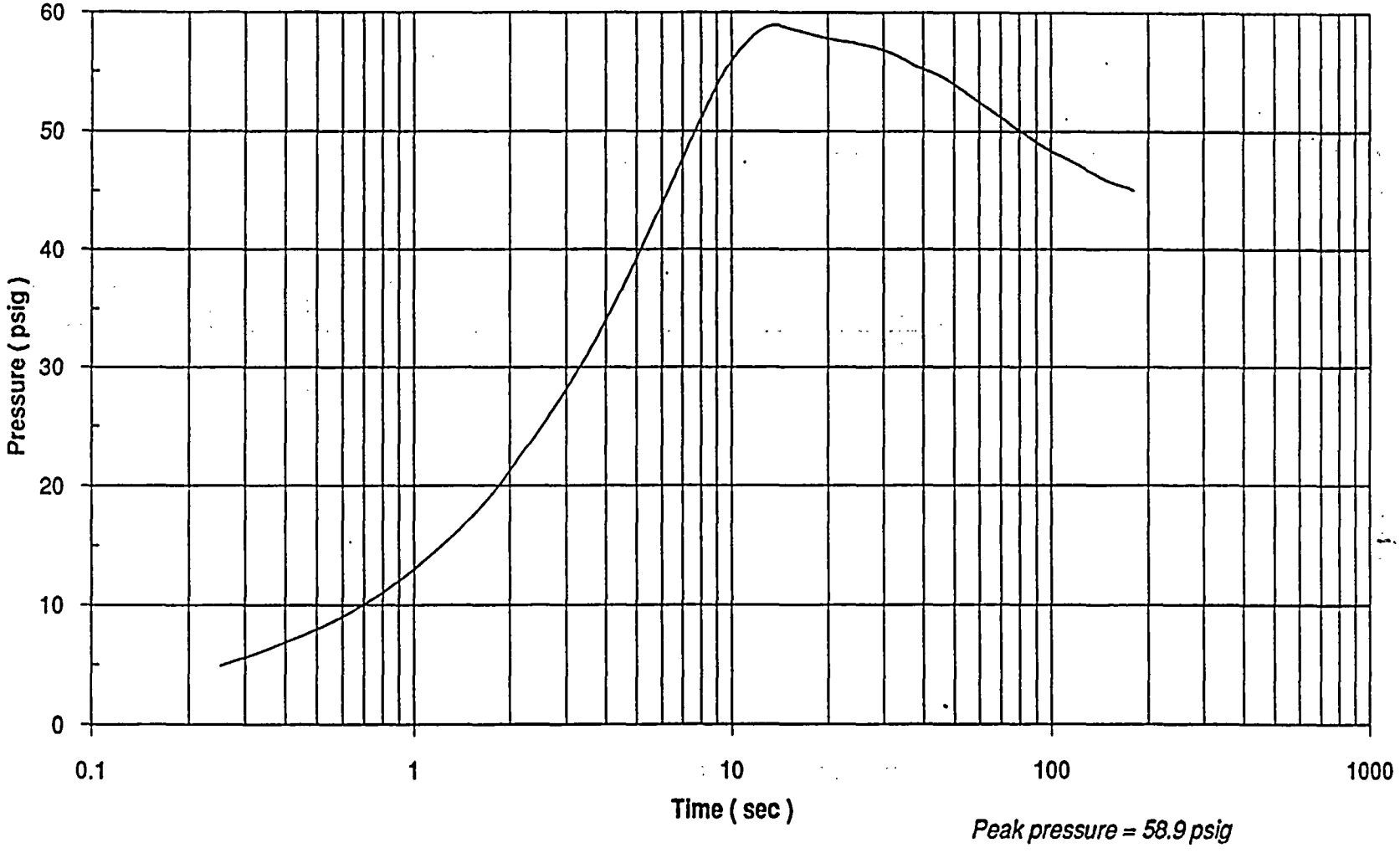


FIGURE 6.2-10

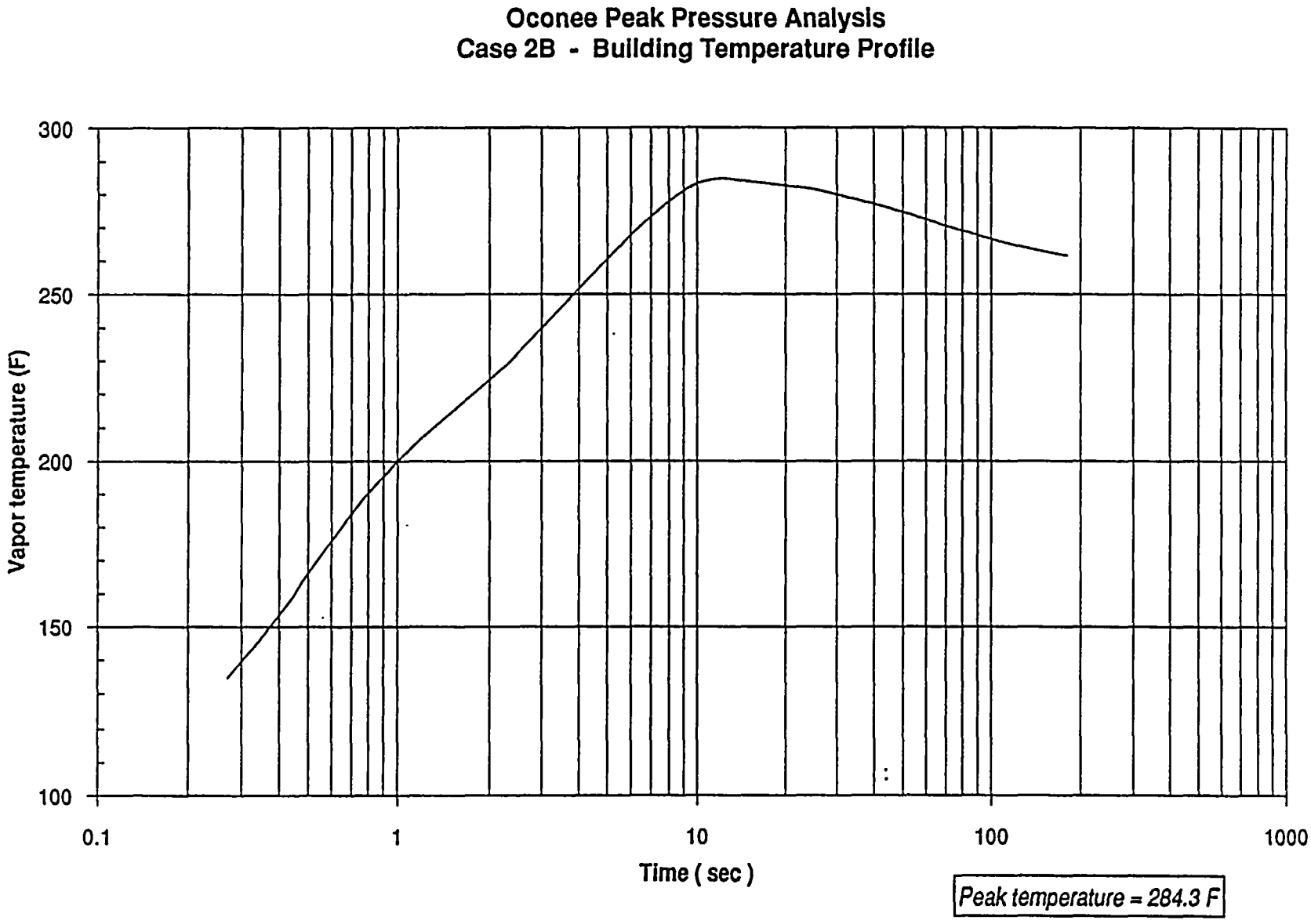

Oconee Peak Pressure Analysis
Case 2B - Building Temperature Profile
Vapor temperature (F)
300
250
200
150
100
0.1
1
10
100
1000
Time (sec)
Peak temperature = 284.3 F

FIGURE 6.2-11

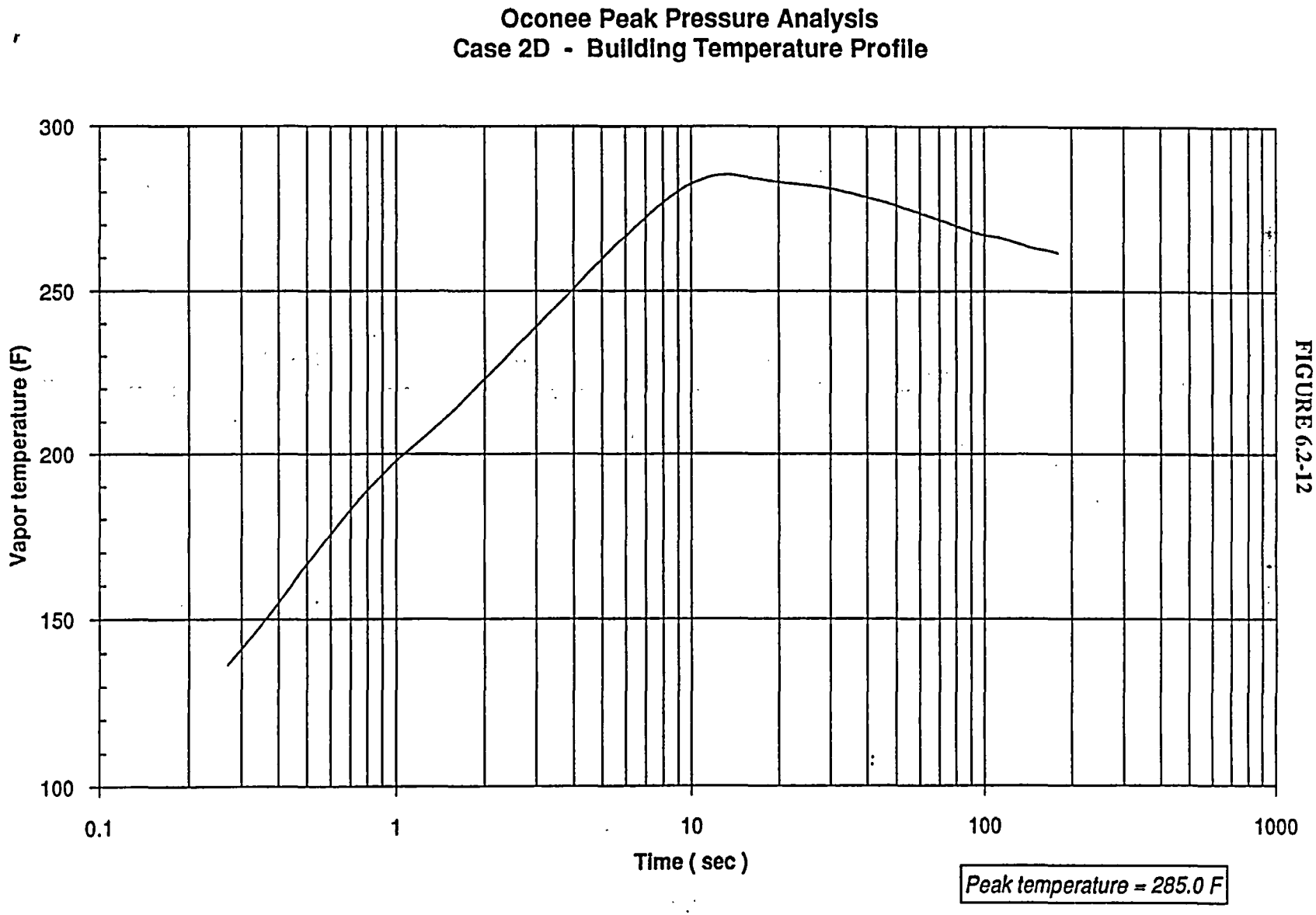
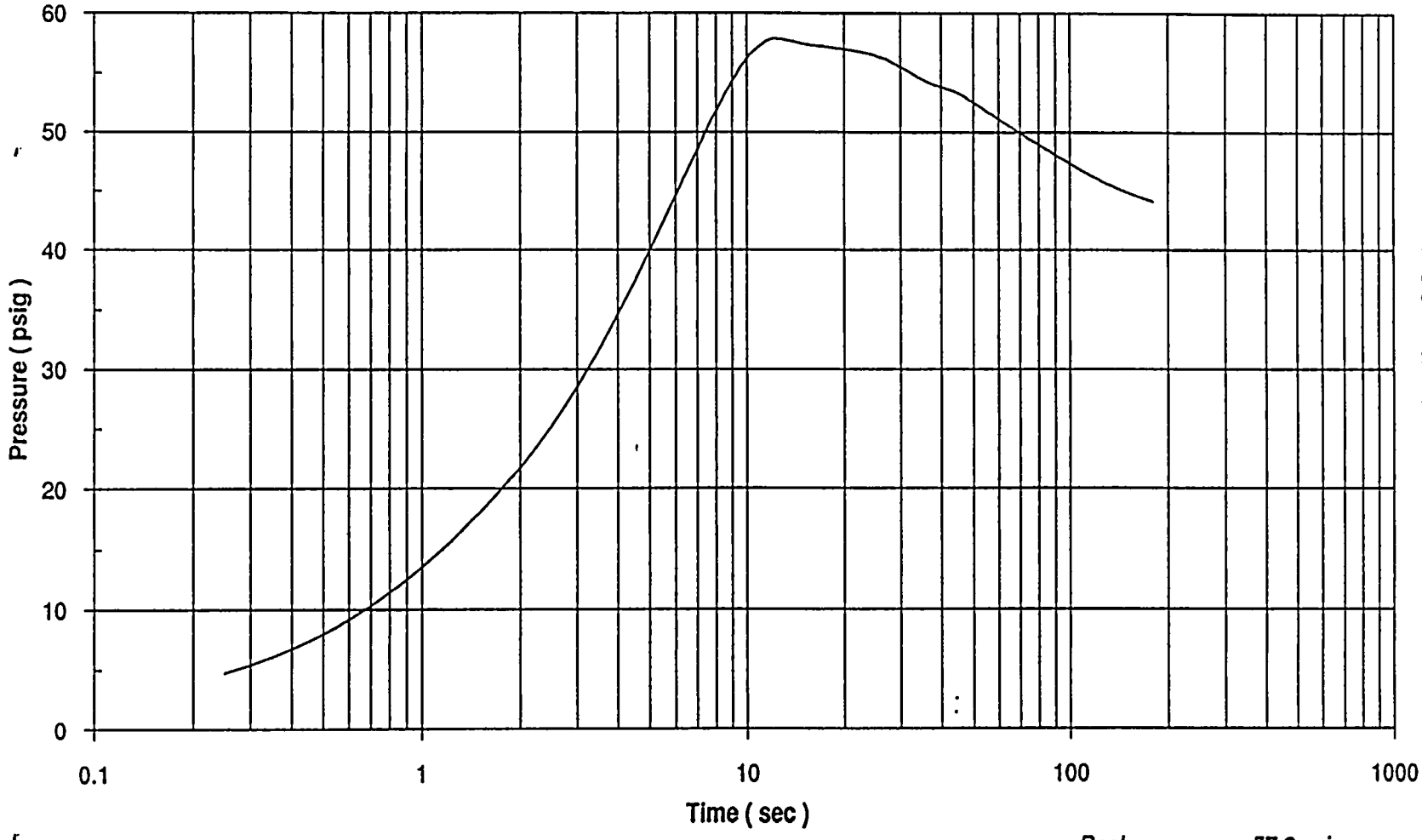

Oconee Peak Pressure Analysis
Case 2D - Building Temperature Profile
Vapor temperature (F)
300
250
200
150
100
0.1
1
10
100
1000
Time (sec)
Peak temperature = 285.0 F

FIGURE 6.2-12

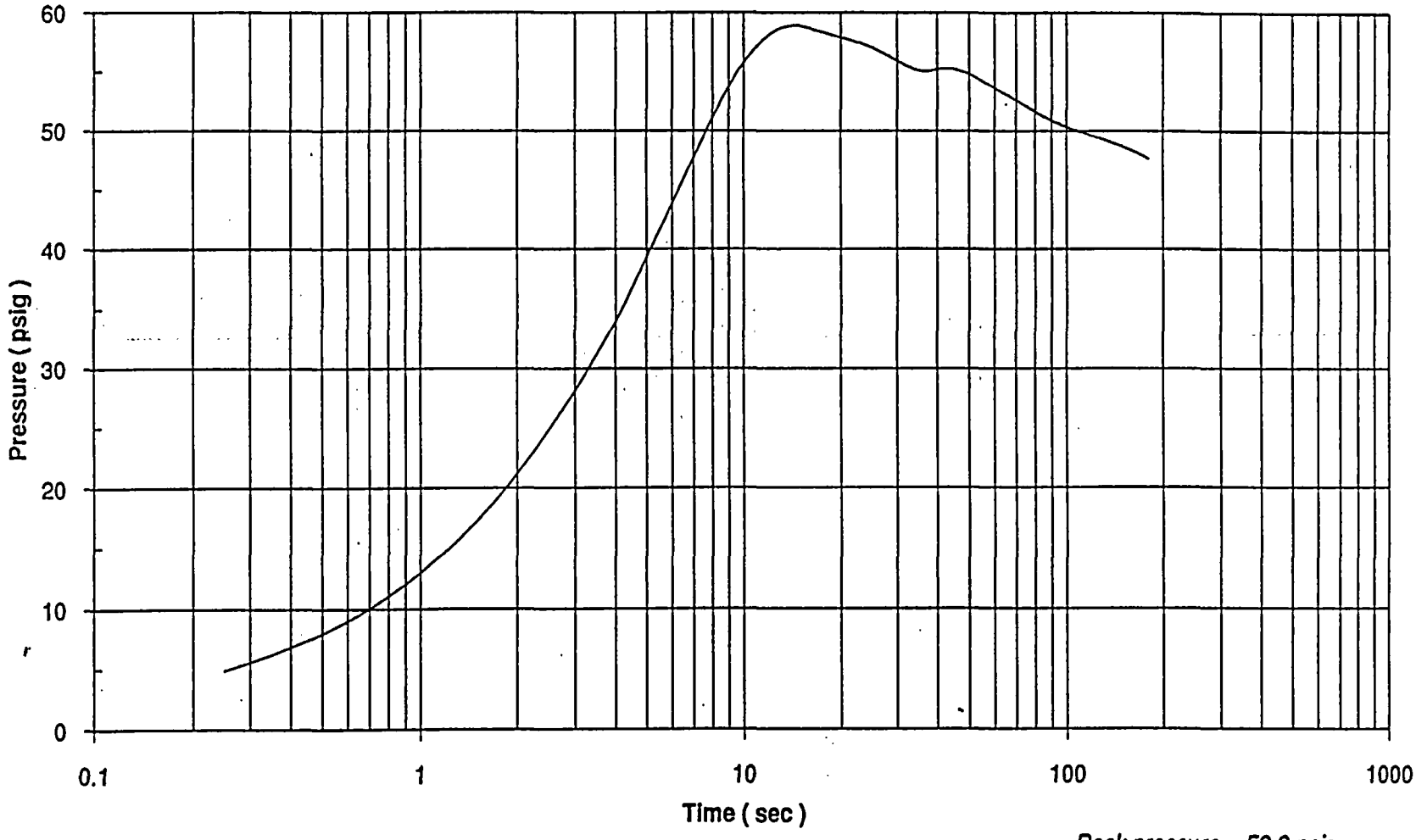
Oconee Peak Pressure Analysis
Case 3A - Building Pressure Profile



Peak pressure = 57.9 psig

FIGURE 6.2-13

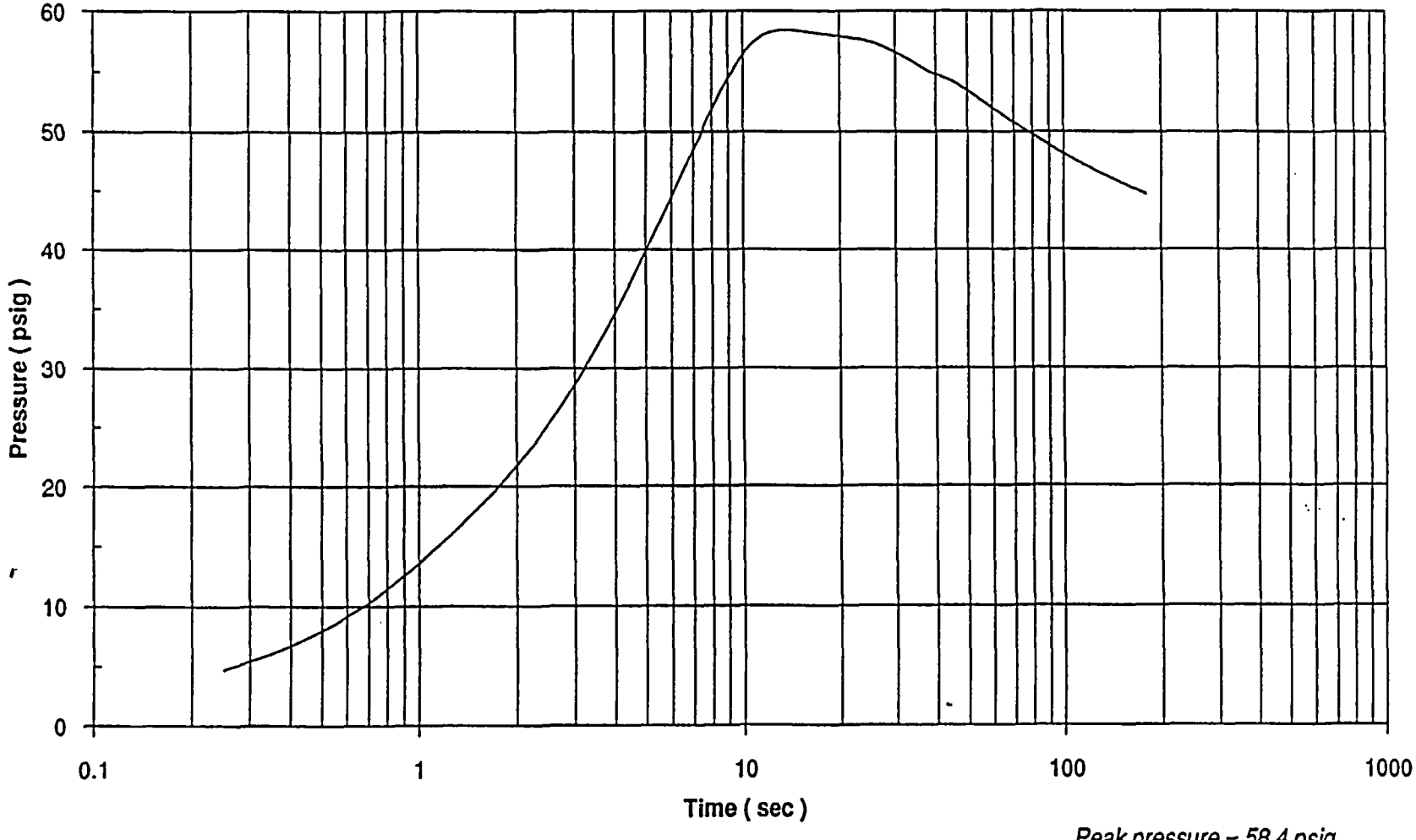
Oconee Peak Pressure Analysis
Case 3B - Building Pressure Profile



Peak pressure = 58.9 psig

FIGURE 6.2-14

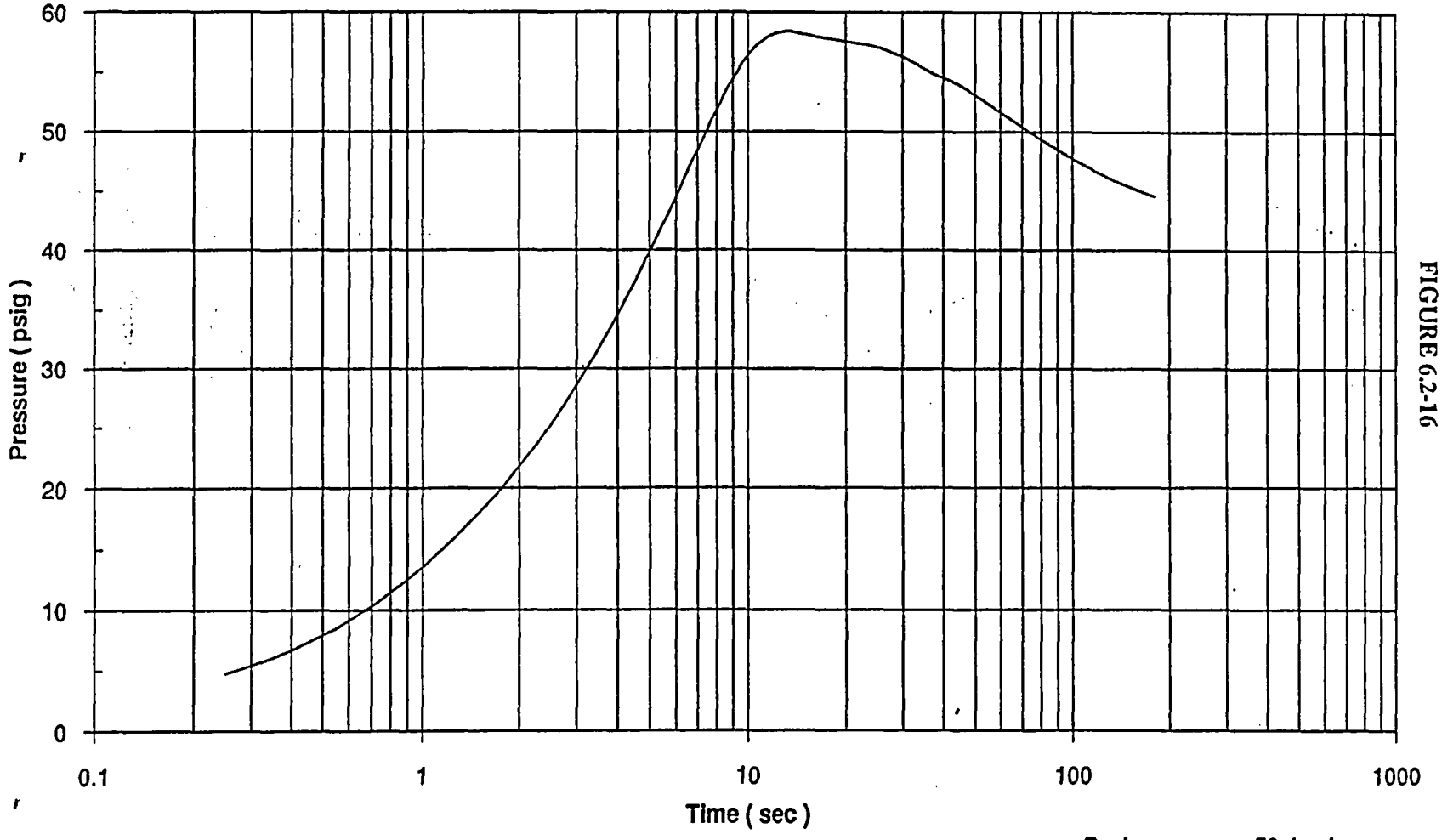
Oconee Peak Pressure Analysis
Case 3C - Building Pressure Profile



Peak pressure = 58.4 psig

FIGURE 6.2-15

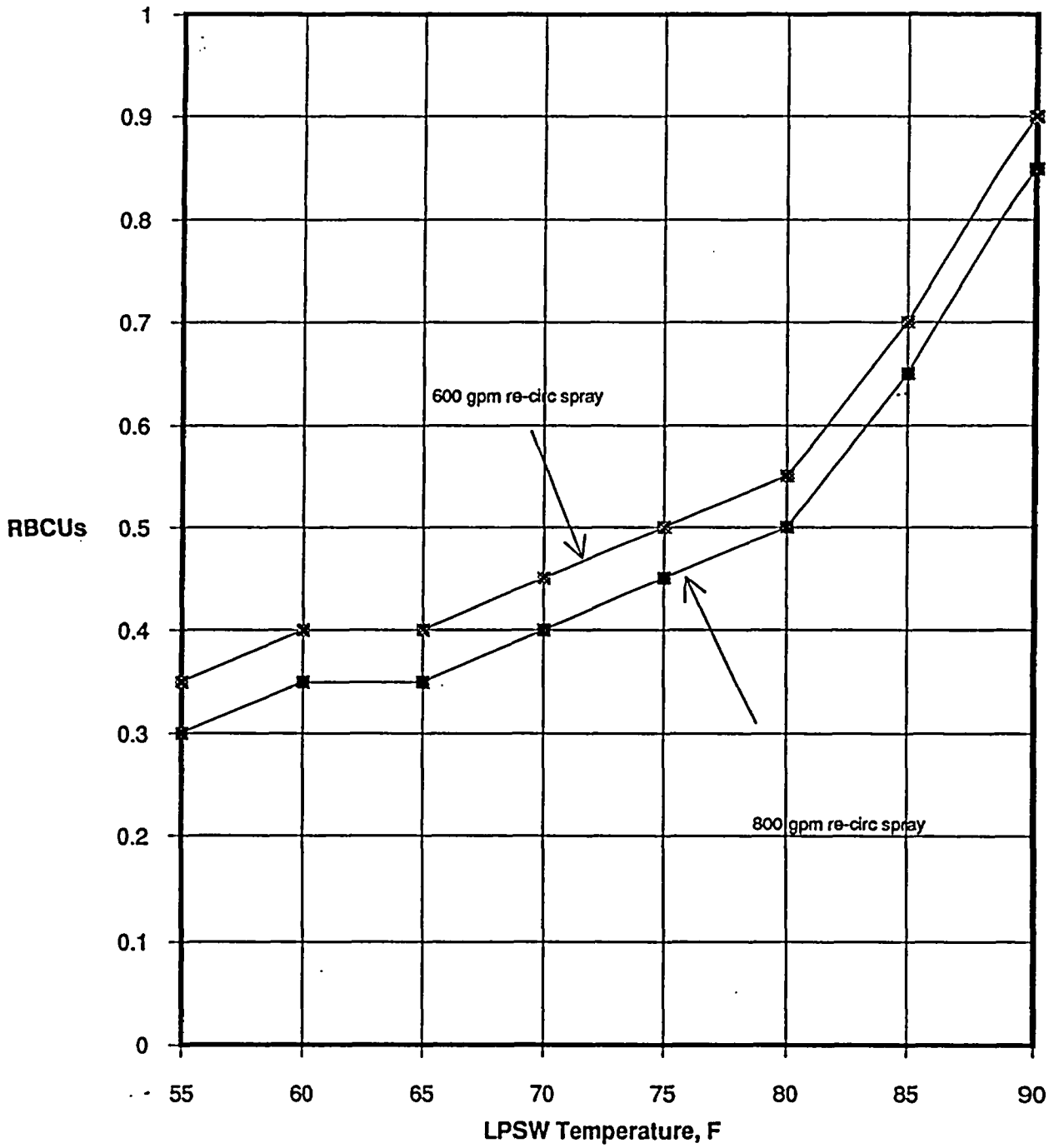
Oconee Peak Pressure Analysis
Case 3D - Building Pressure Profile



Peak pressure = 58.4 psig

FIGURE 6.2-16

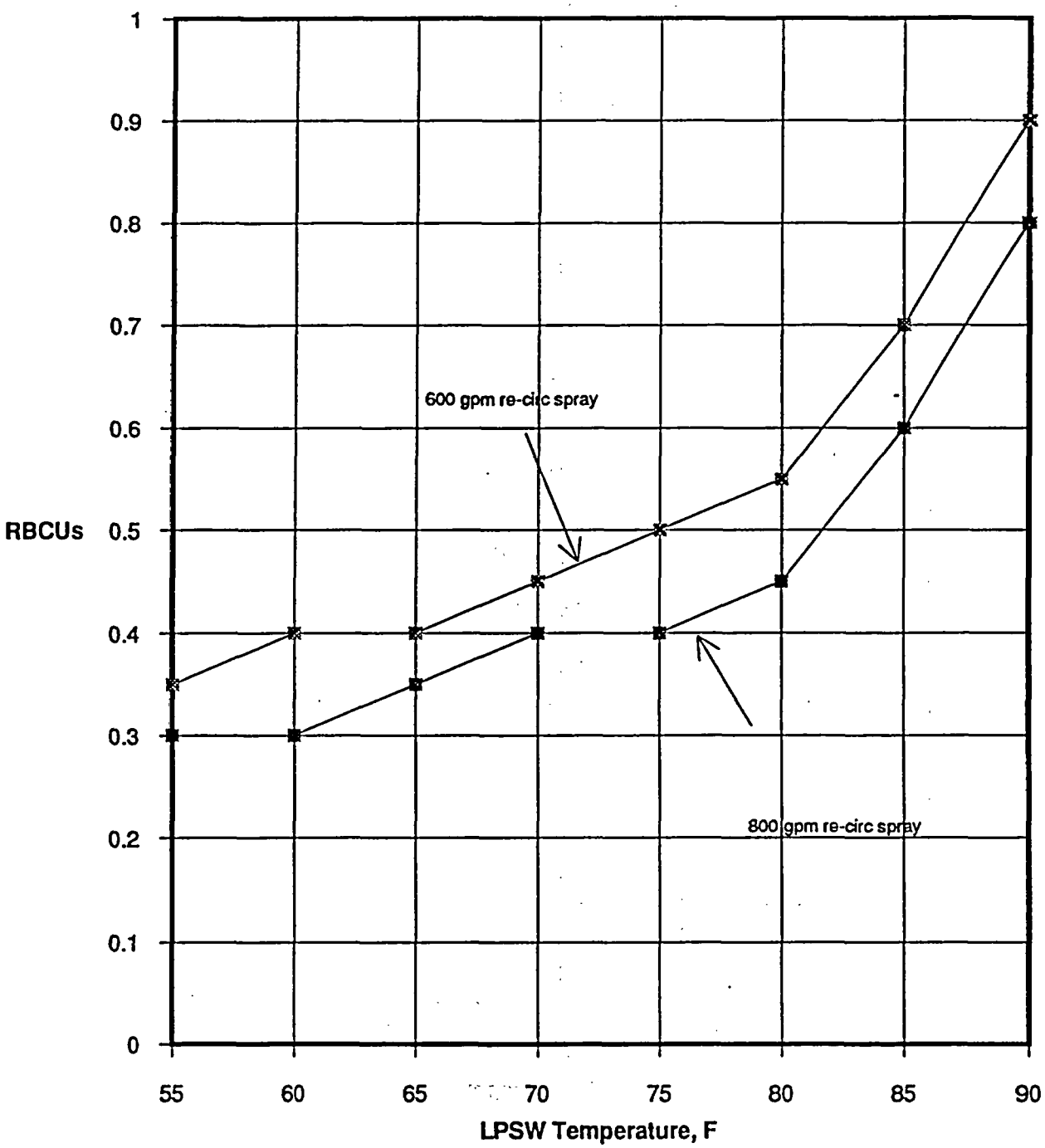
FIGURE 6.3-1



LPI cooler capacity = 93 E6 Btu/hr

Long-term Containment Cooling - RBCU Requirements

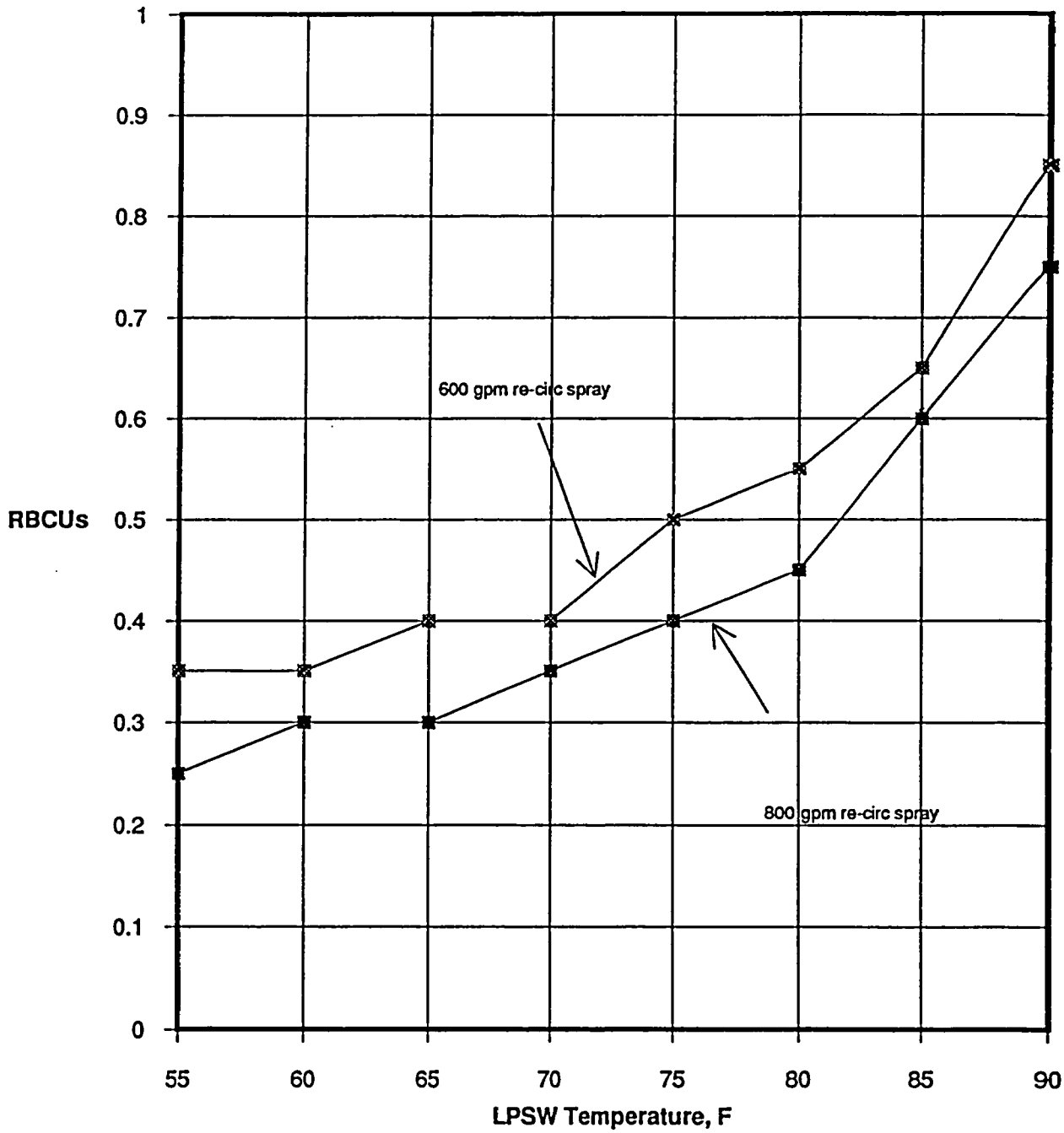
FIGURE 6.3-2



LPI cooler capacity = 97 E6 Btu/hr

Long-term Containment Cooling - RBCU Requirements

FIGURE 6.3-3



LPI cooler capacity = 102 E6 Btu/hr

Long-term Containment Cooling - RBCU Requirements

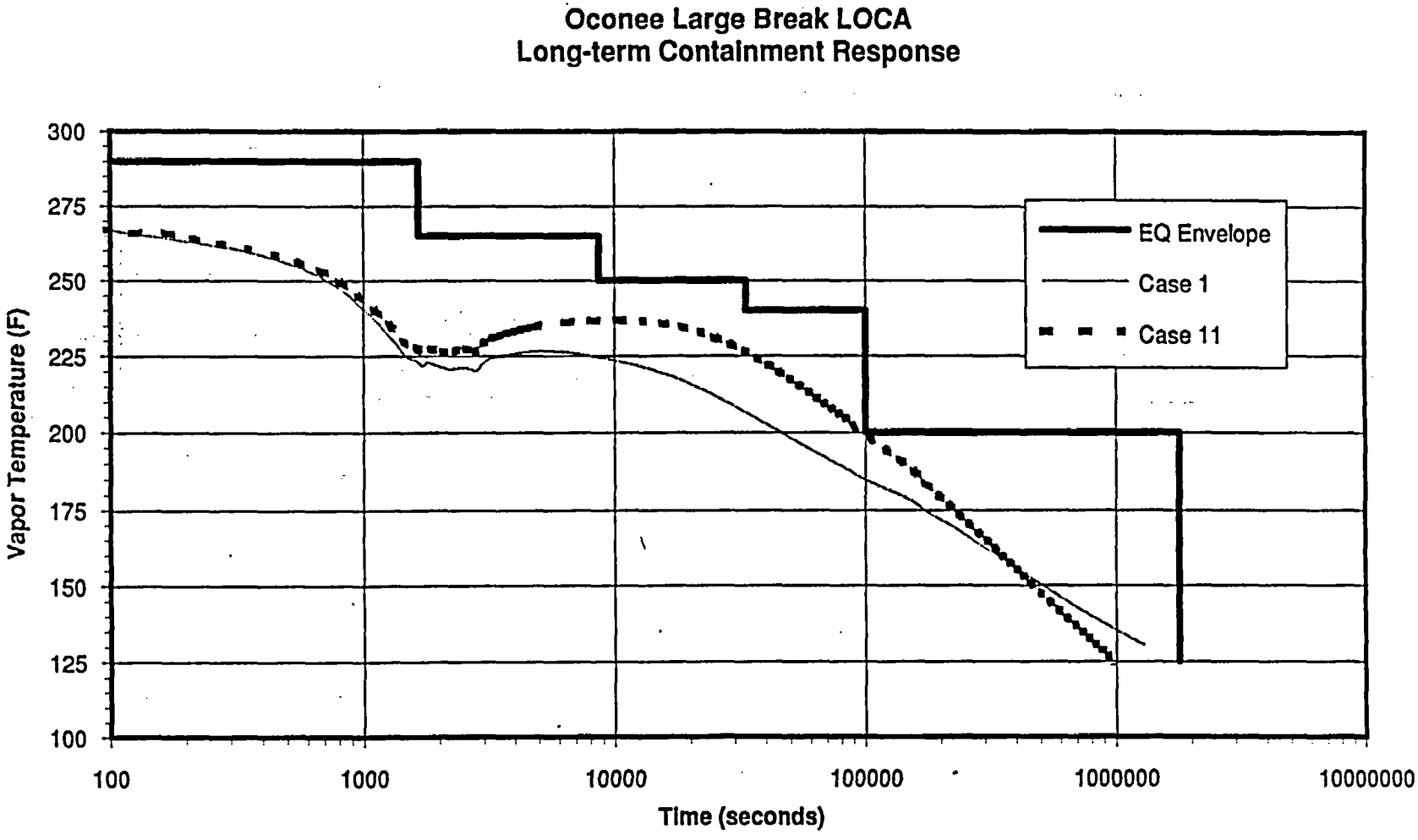


FIGURE 6.3-4

Vapor Temperature Profiles (600 gpm re-circ spray)

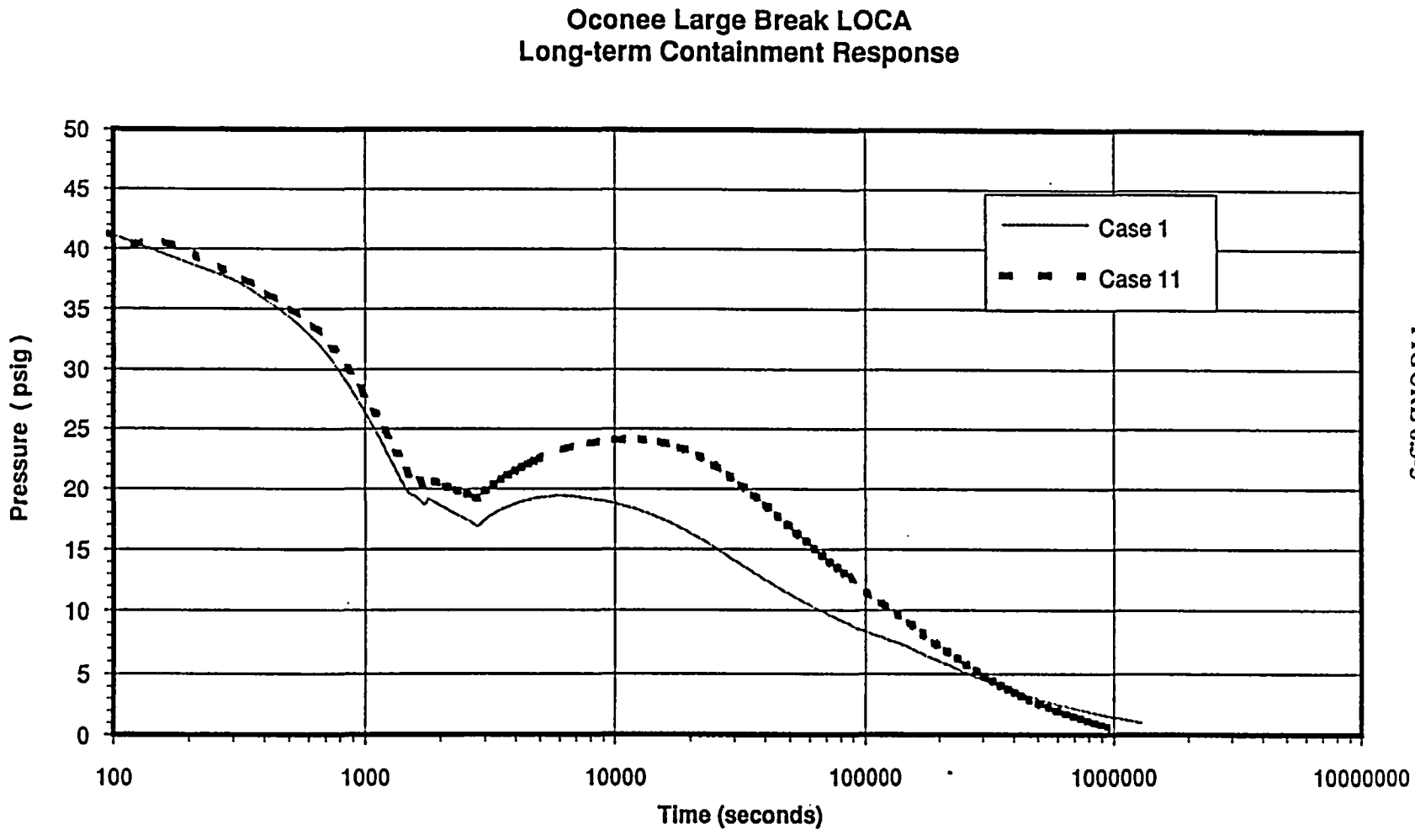


FIGURE 6.3-5

Building Pressure Profiles (600 gpm re-circ spray)

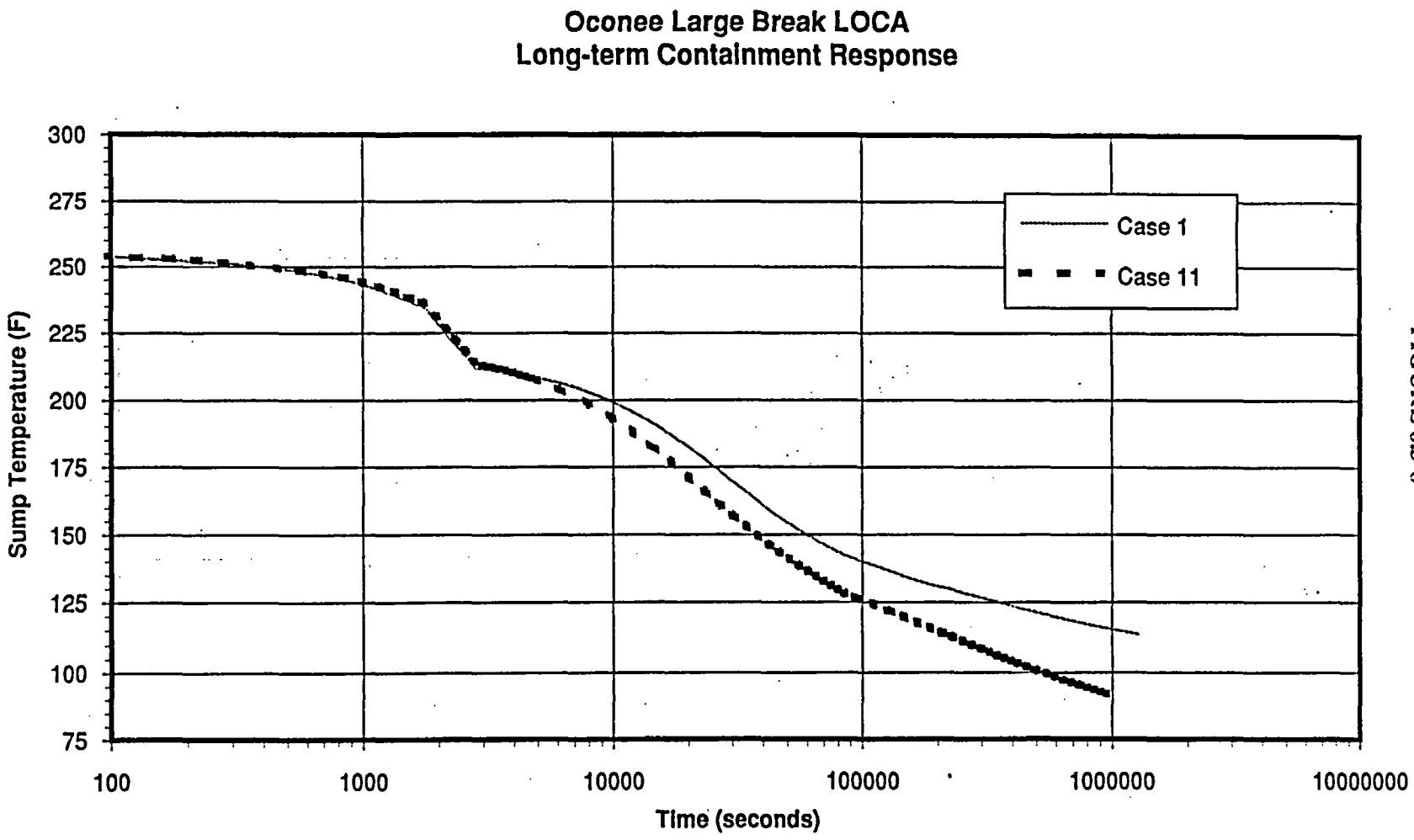


FIGURE 6.3-6

Sump Temperature Profiles (600 gpm re-circ spray)

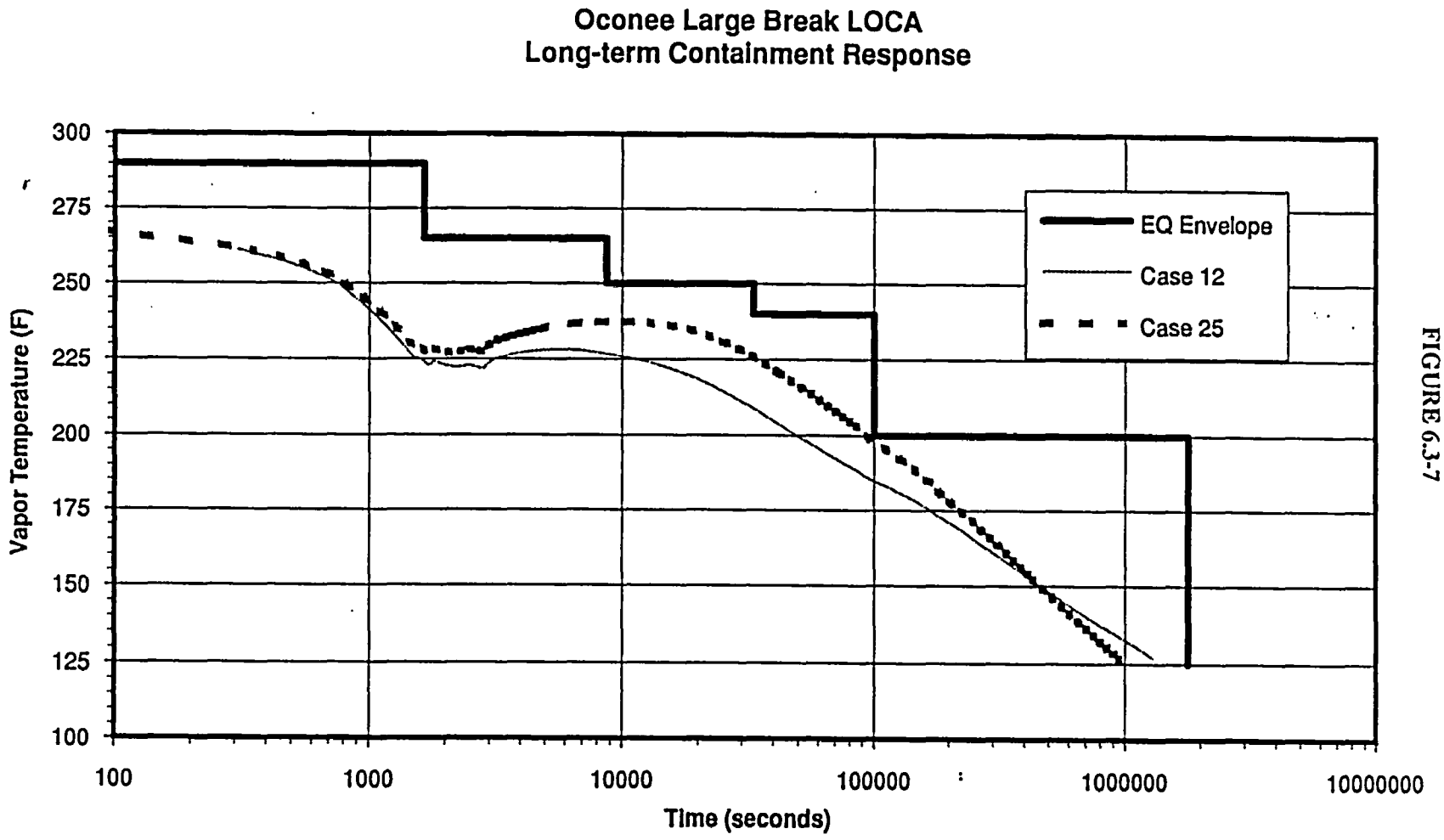


FIGURE 6.3-7

Vapor Temperature Profiles (800 gpm re-circ spray)

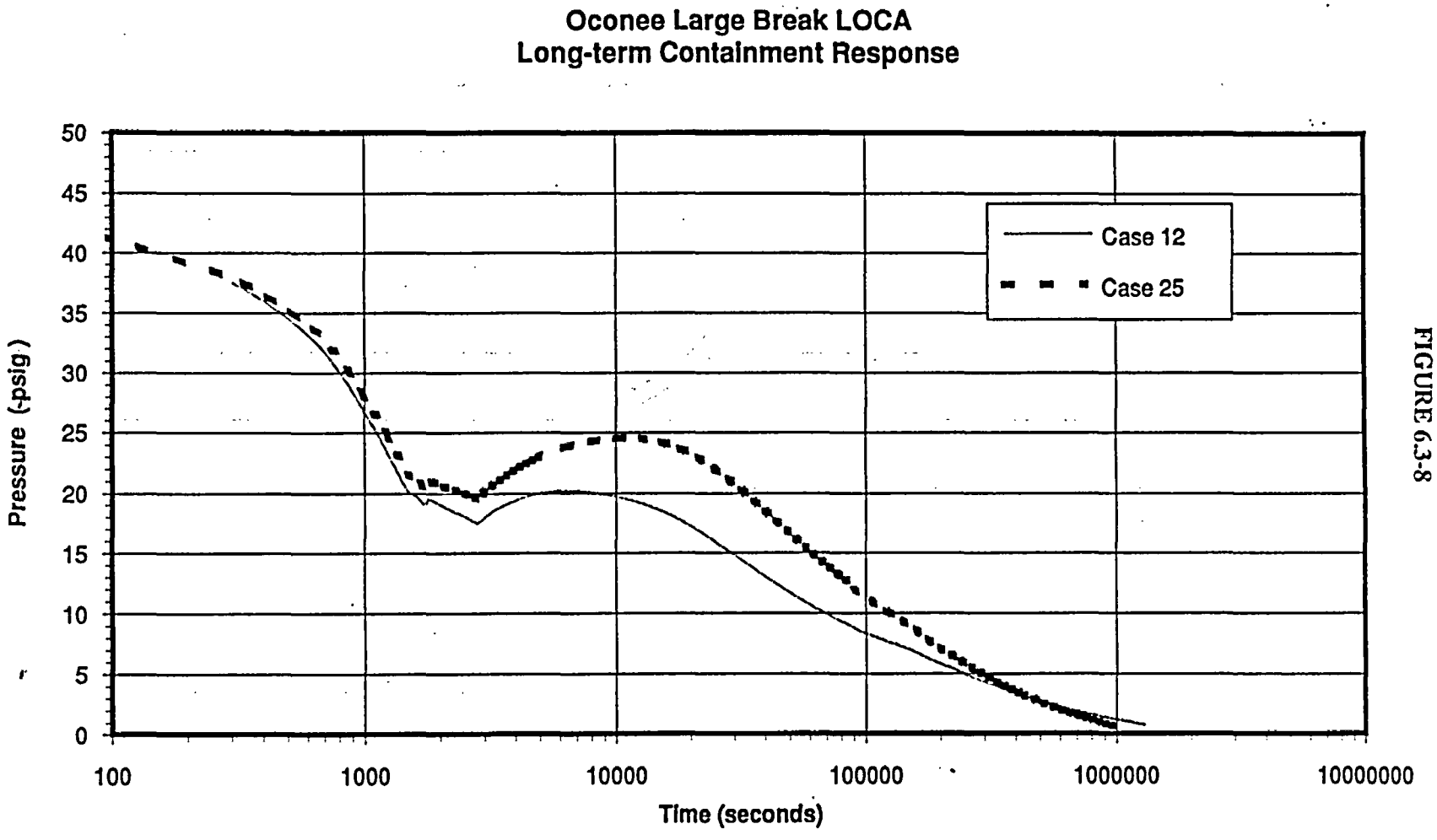


FIGURE 6.3-8

Building Pressure Profiles (800 gpm re-circ spray)

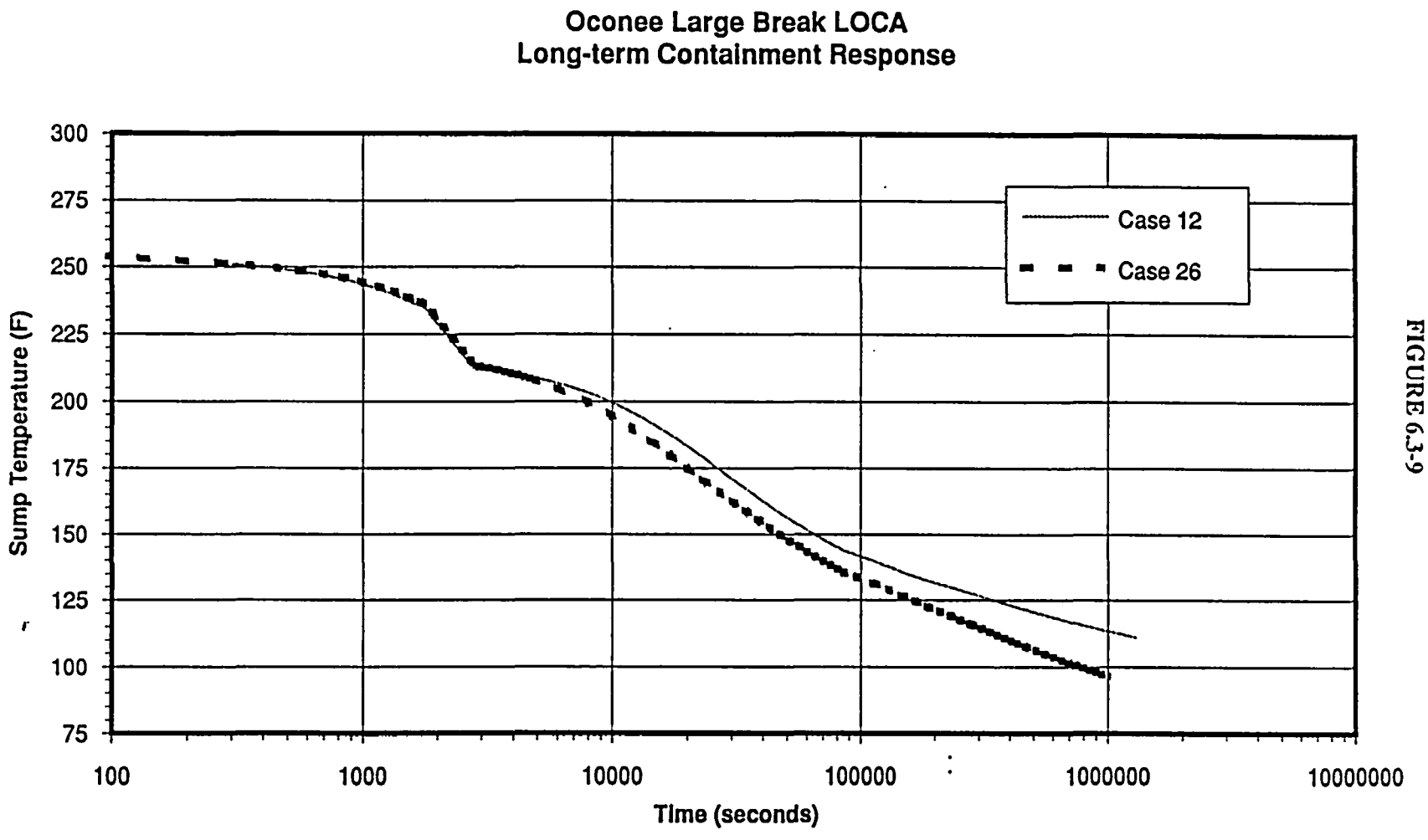


FIGURE 6.3-9

Sump Temperature Profiles (800 gpm re-circ spray)

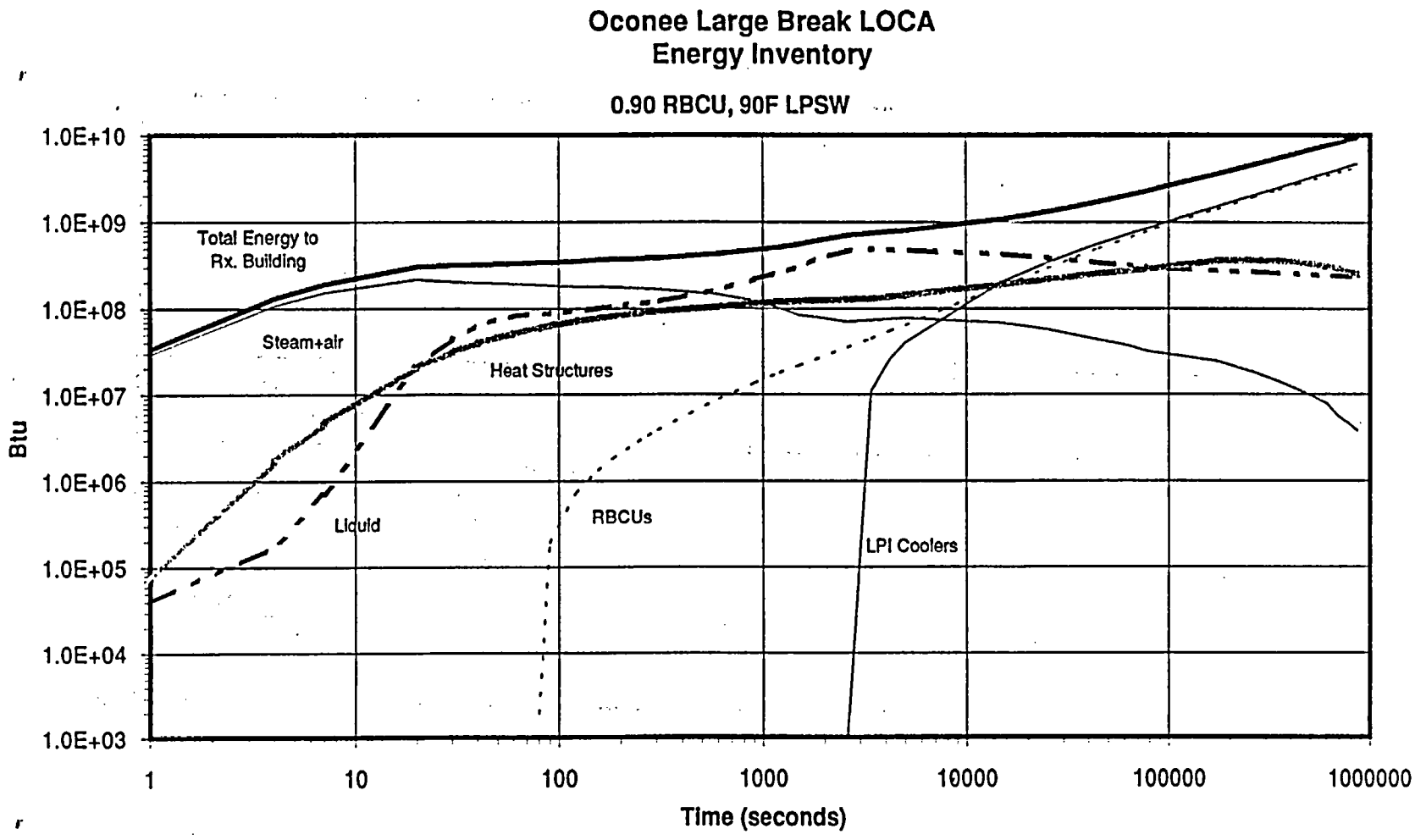


FIGURE 6.3-10

Reactor Building Energy Inventory - Case 1

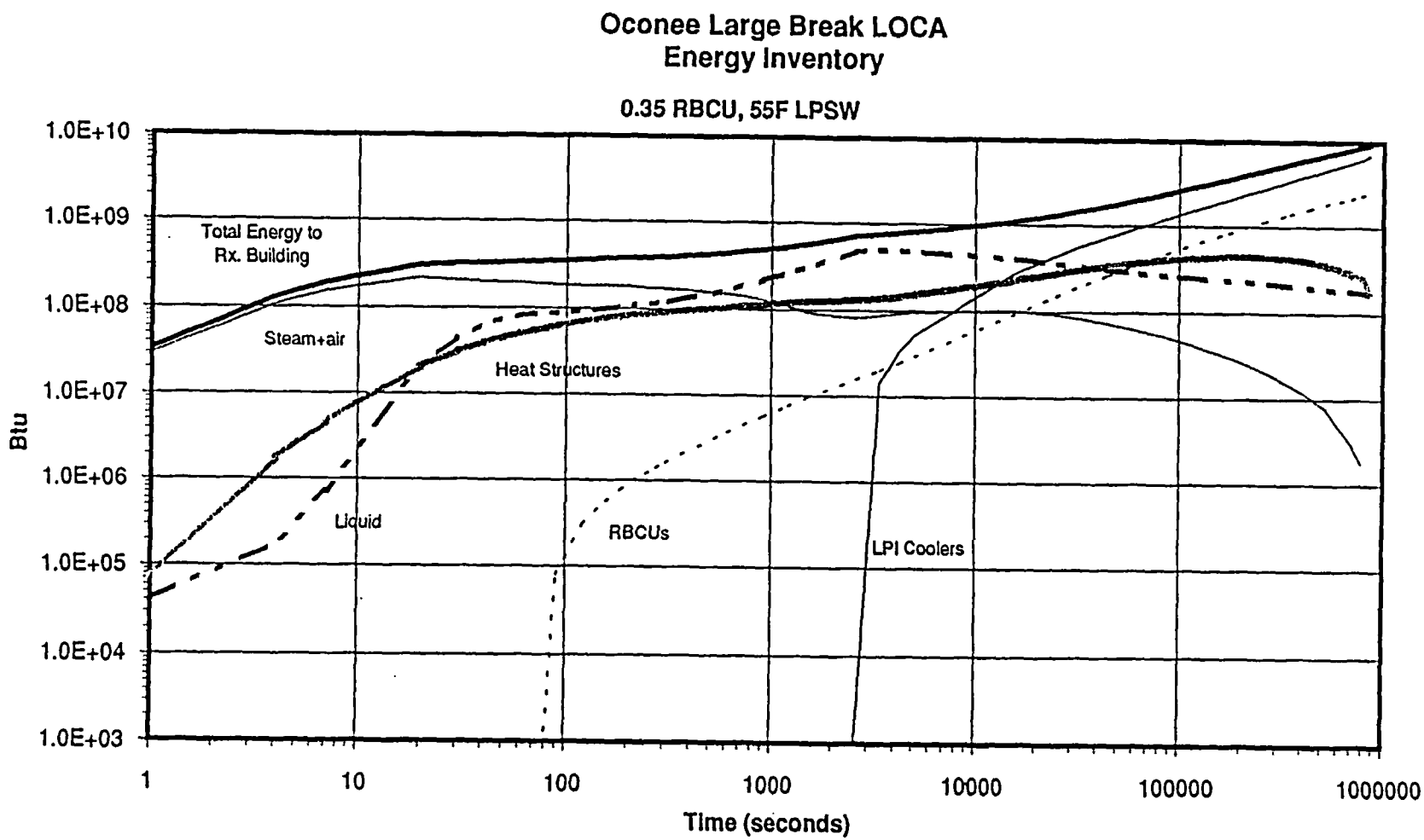


FIGURE 6.3-11

Reactor Building Energy Inventory - Case 11

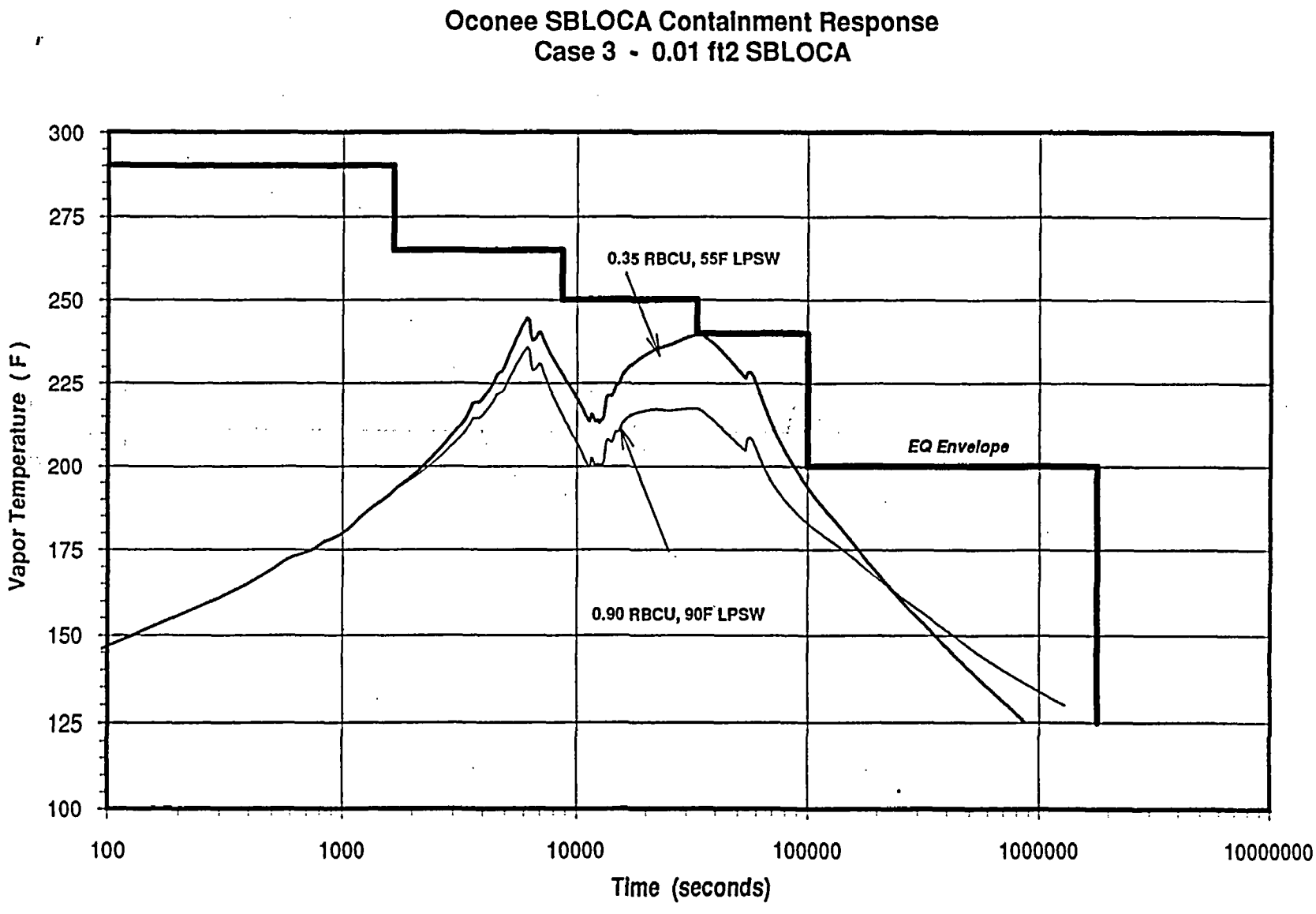


FIGURE 6.4-1

Vapor Temperature Profiles (Runs 1 and 8)

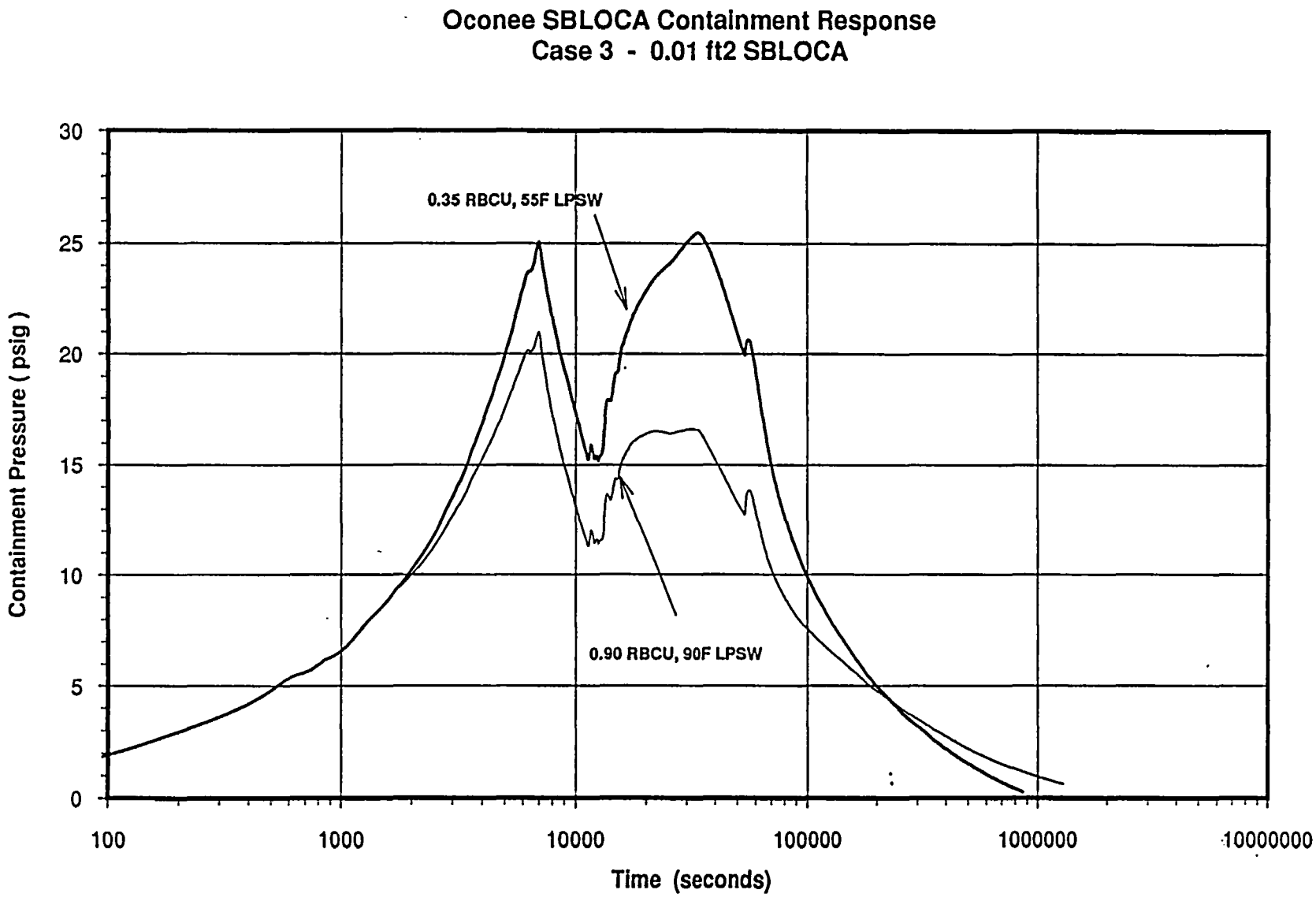


FIGURE 6.4-2

Building Pressure Profiles (Runs 1 and 8)

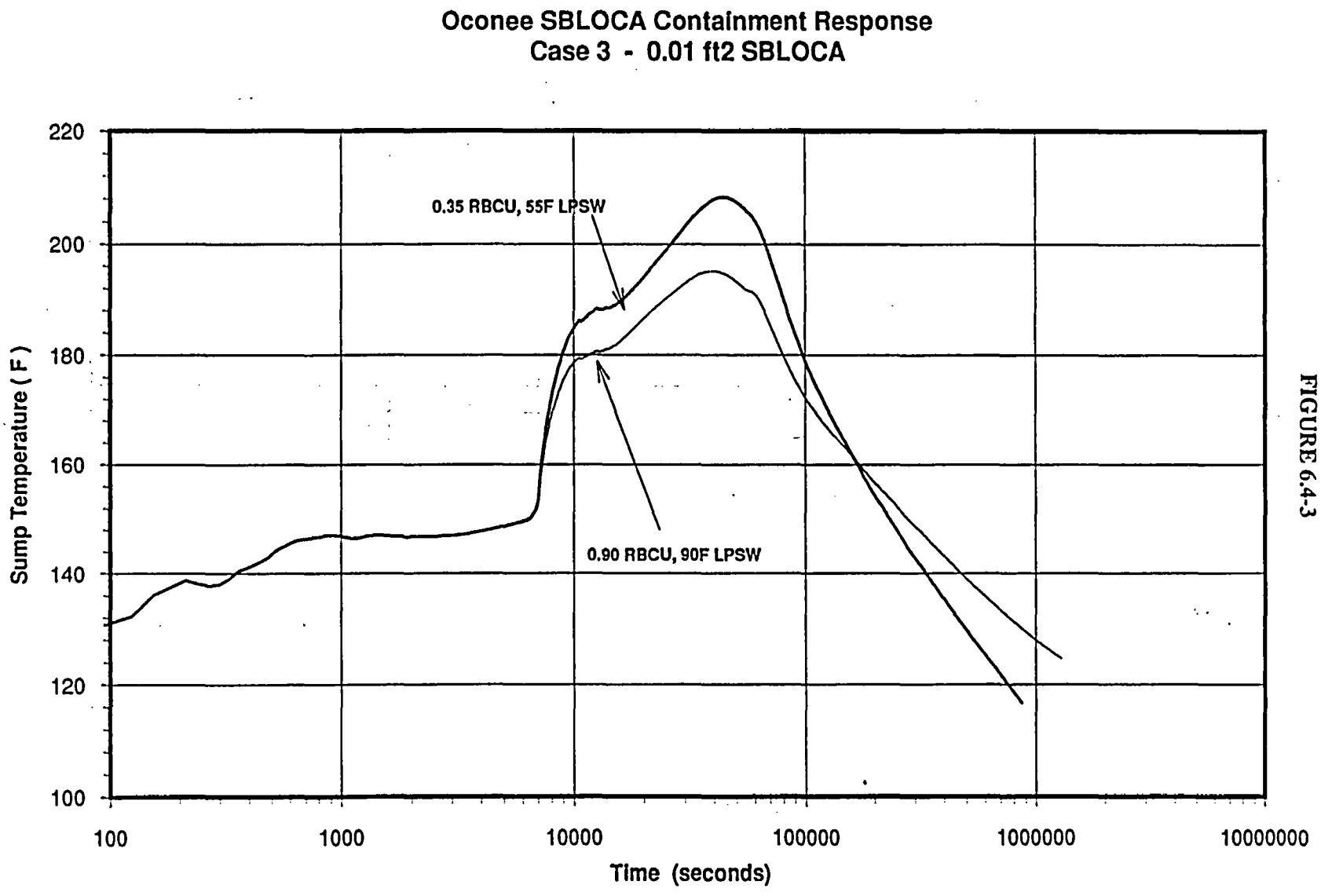


FIGURE 6.4-3

Sump Temperature Profiles (Runs 1 and 8)

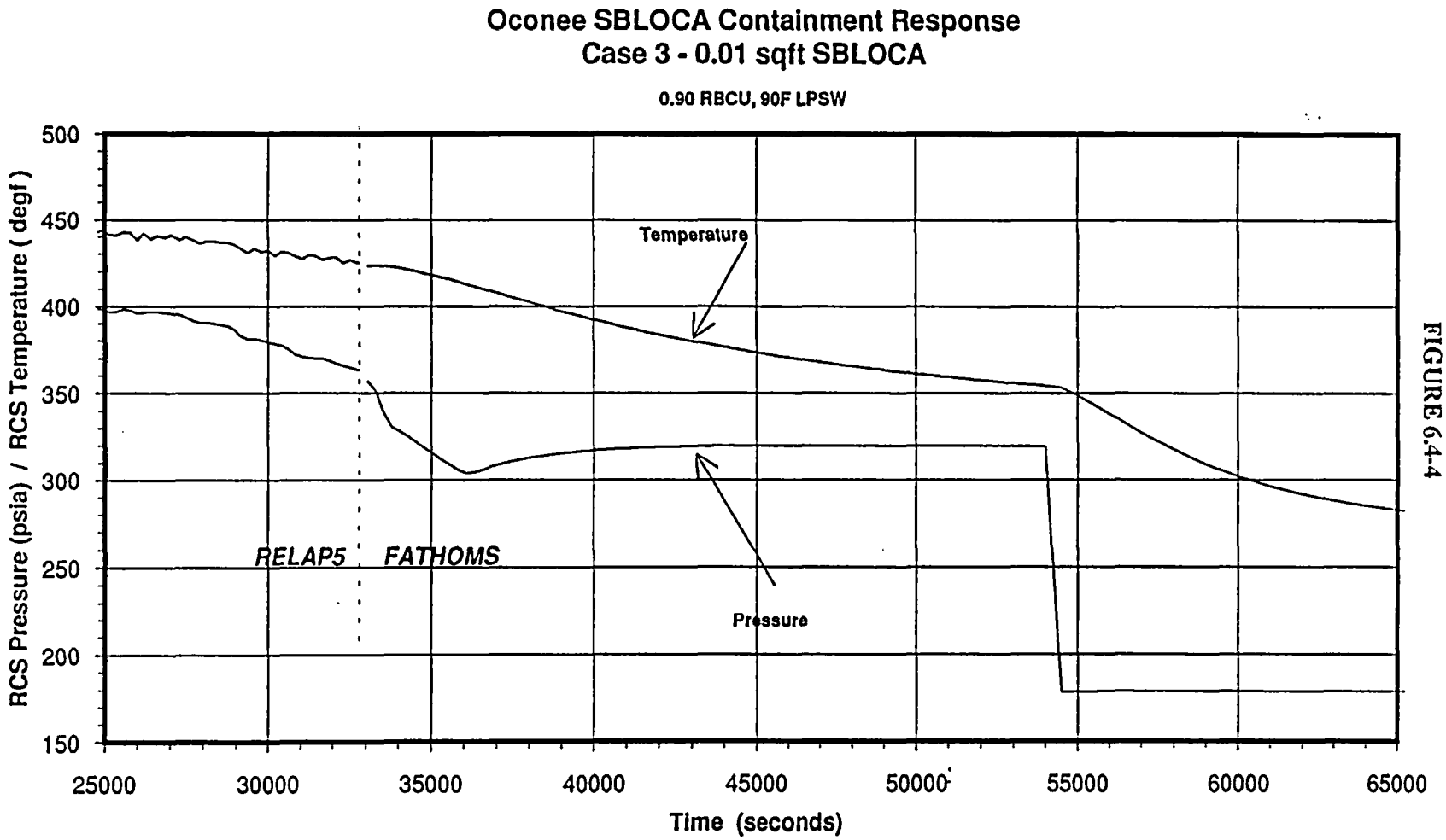


FIGURE 6.4-4

RCS Conditions

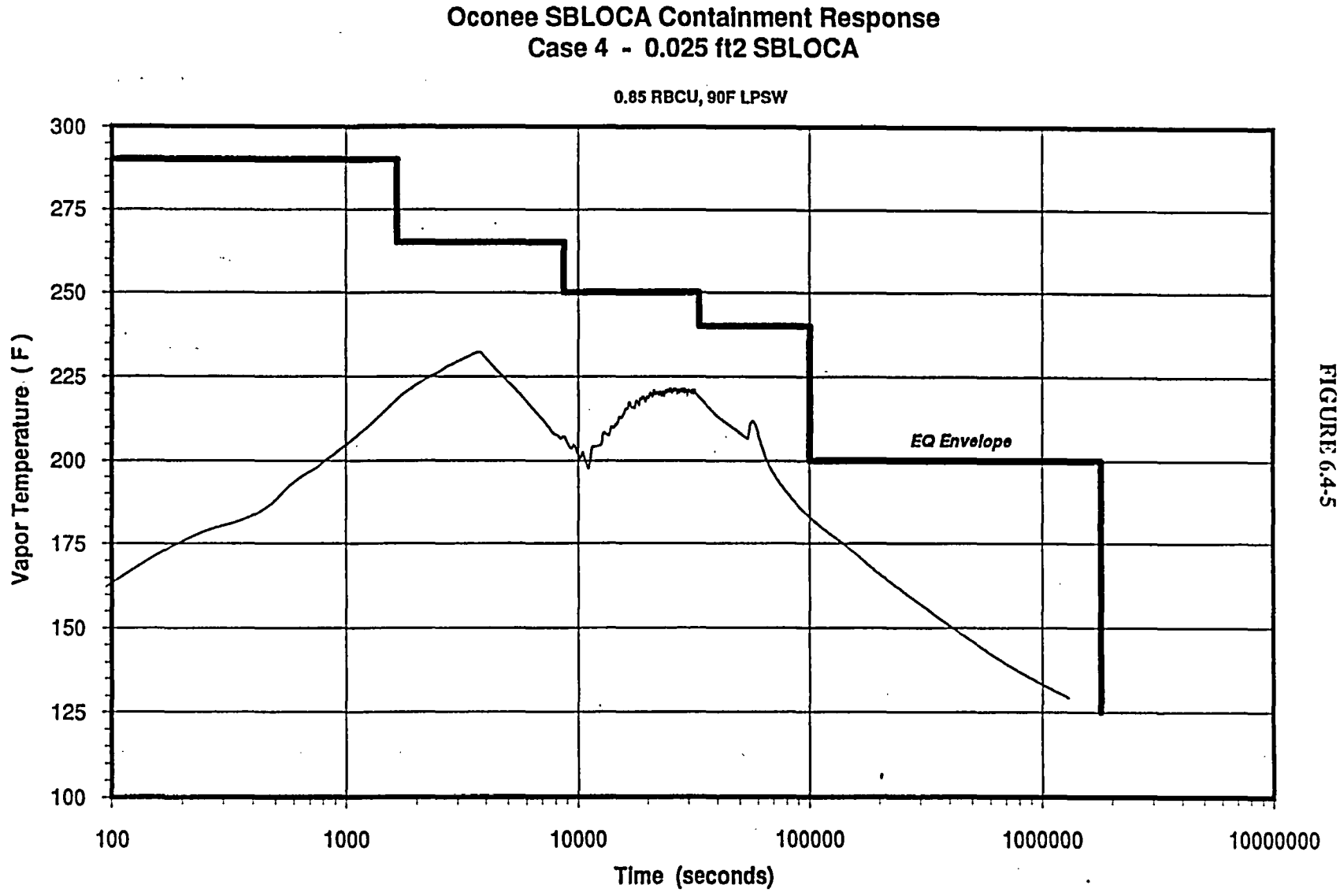


FIGURE 6.4-5

Vapor Temperature Profile

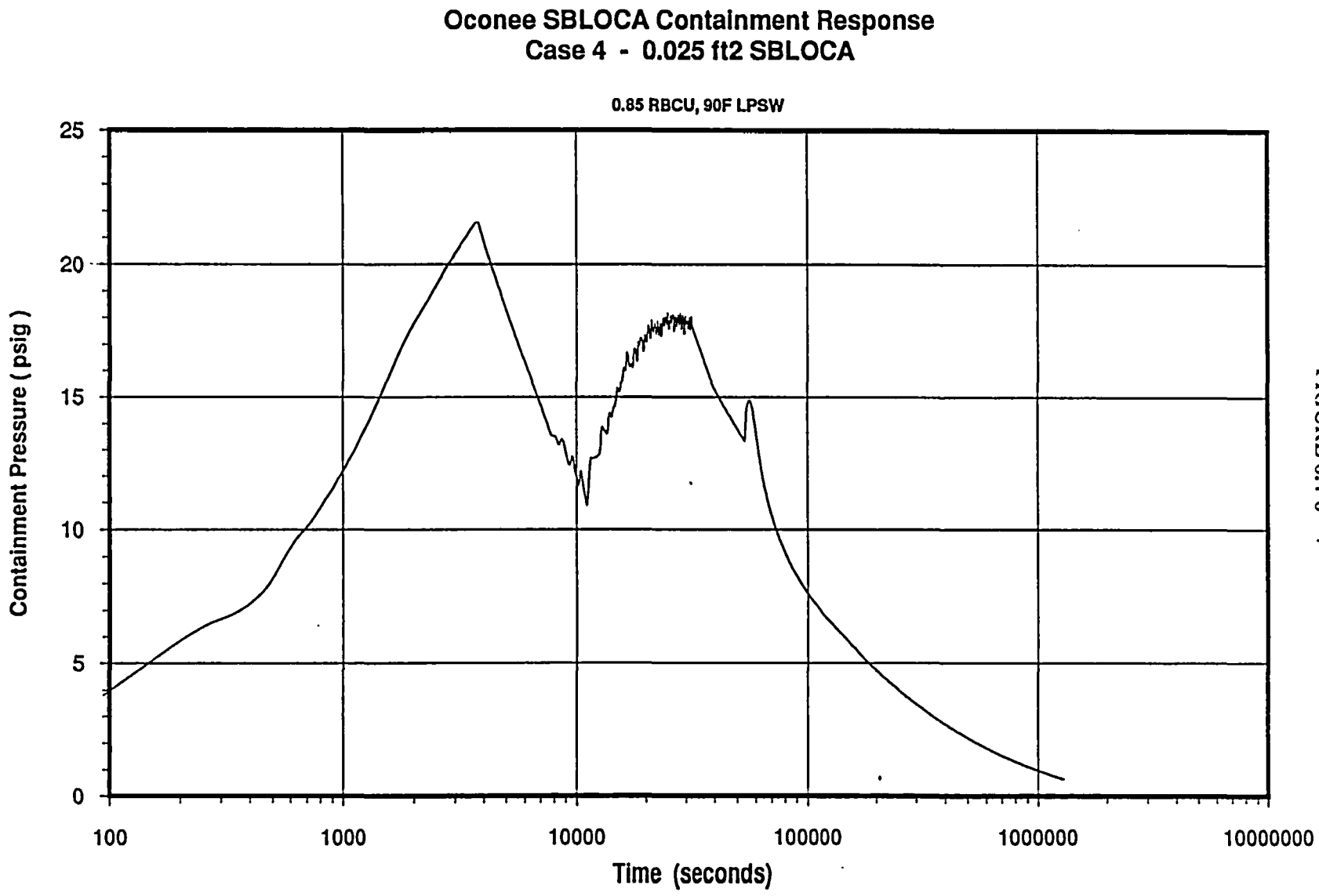
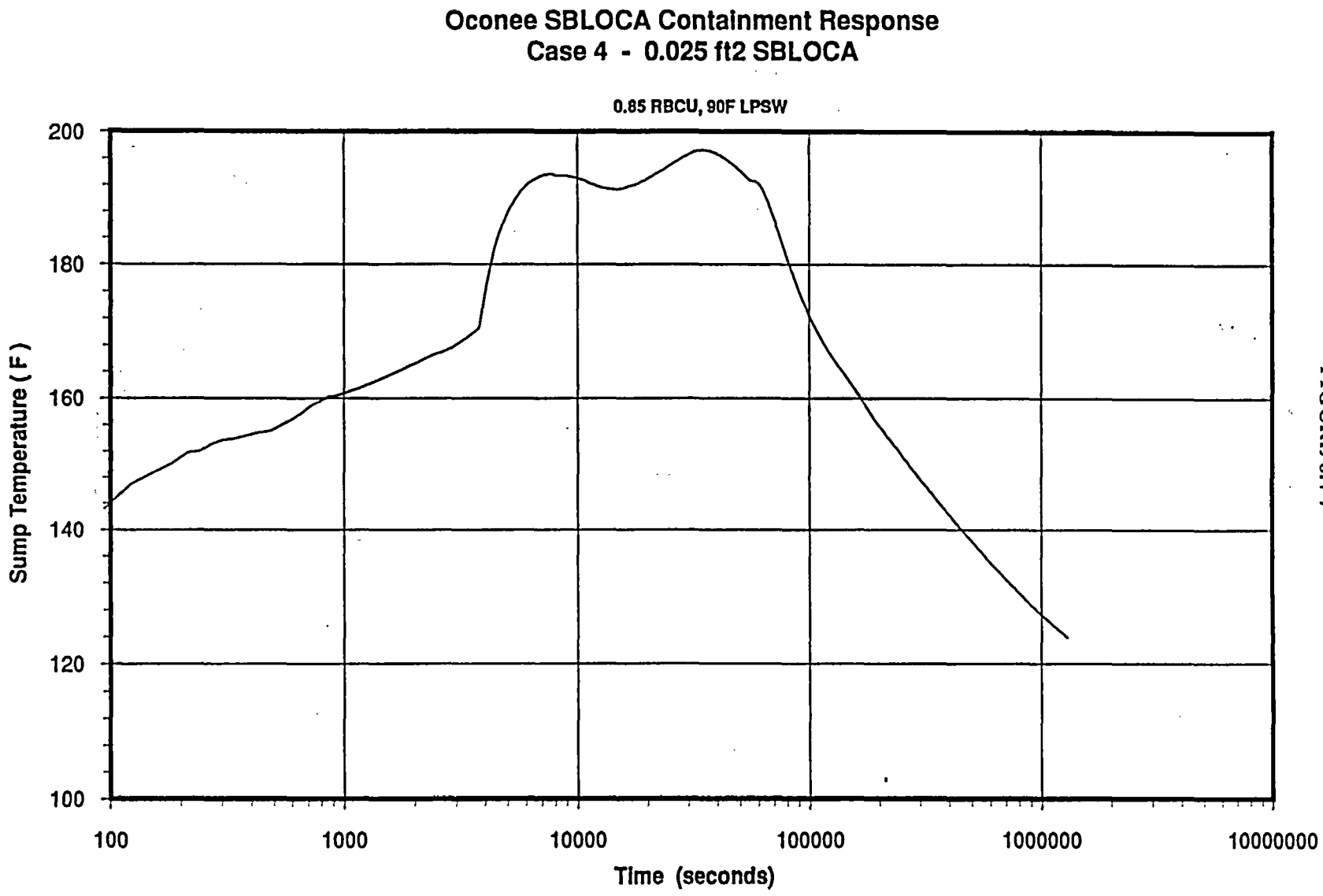


FIGURE 6.4-6

Building Pressure Profile



Sump Temperature Profile

FIGURE 6.4-7

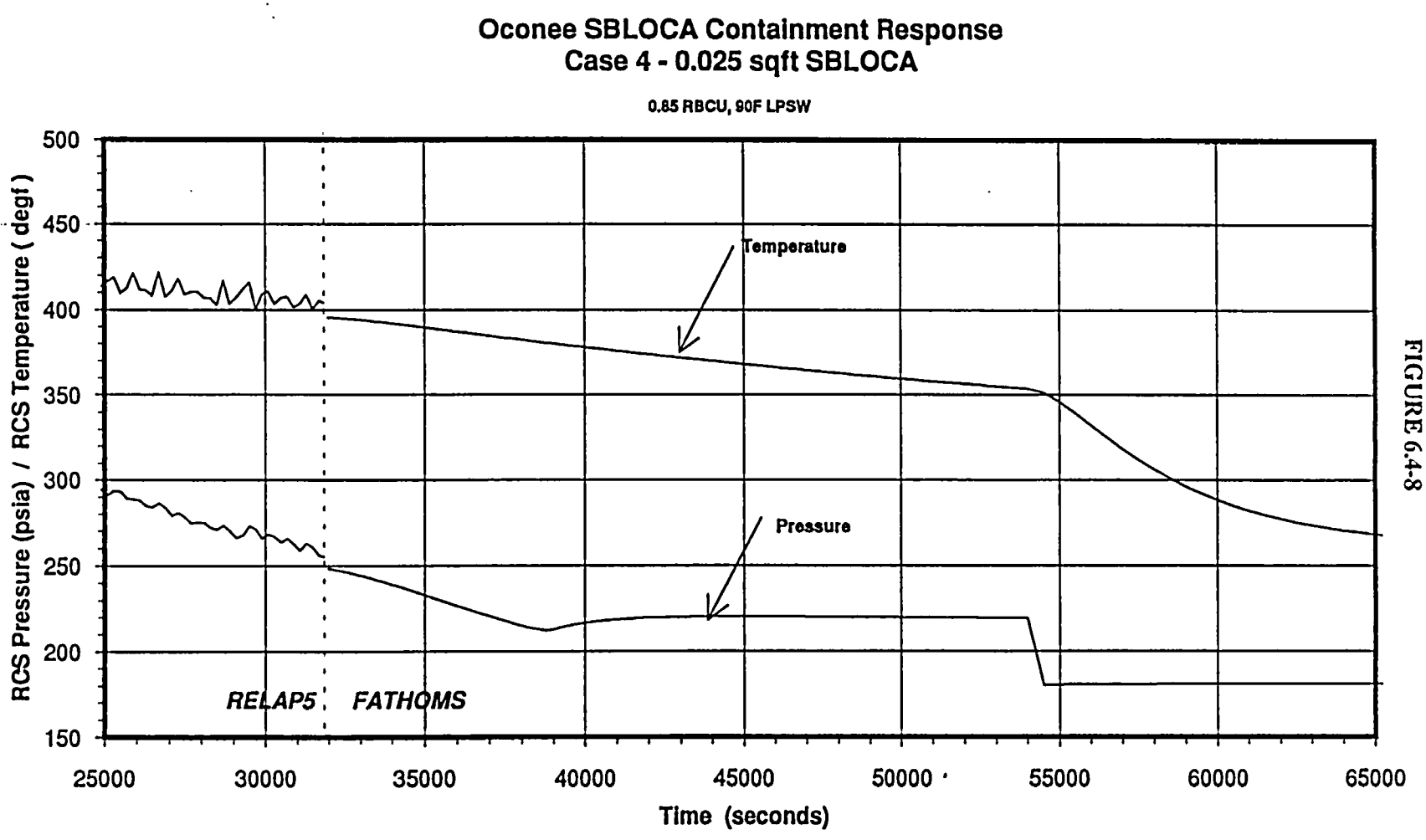


FIGURE 6.4-8

RCS Conditions

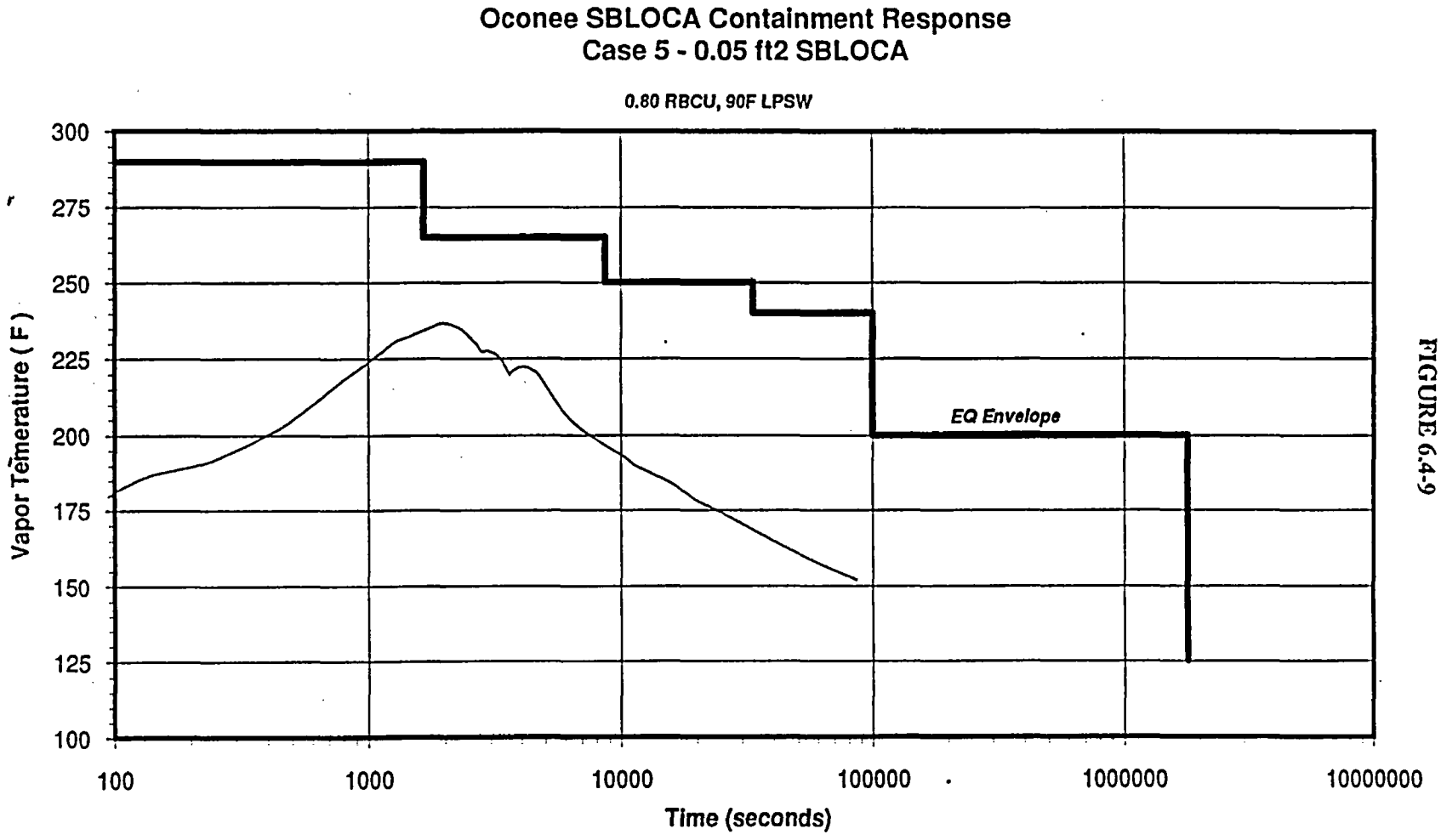


FIGURE 6.4-9

Vapor Temperature Profile

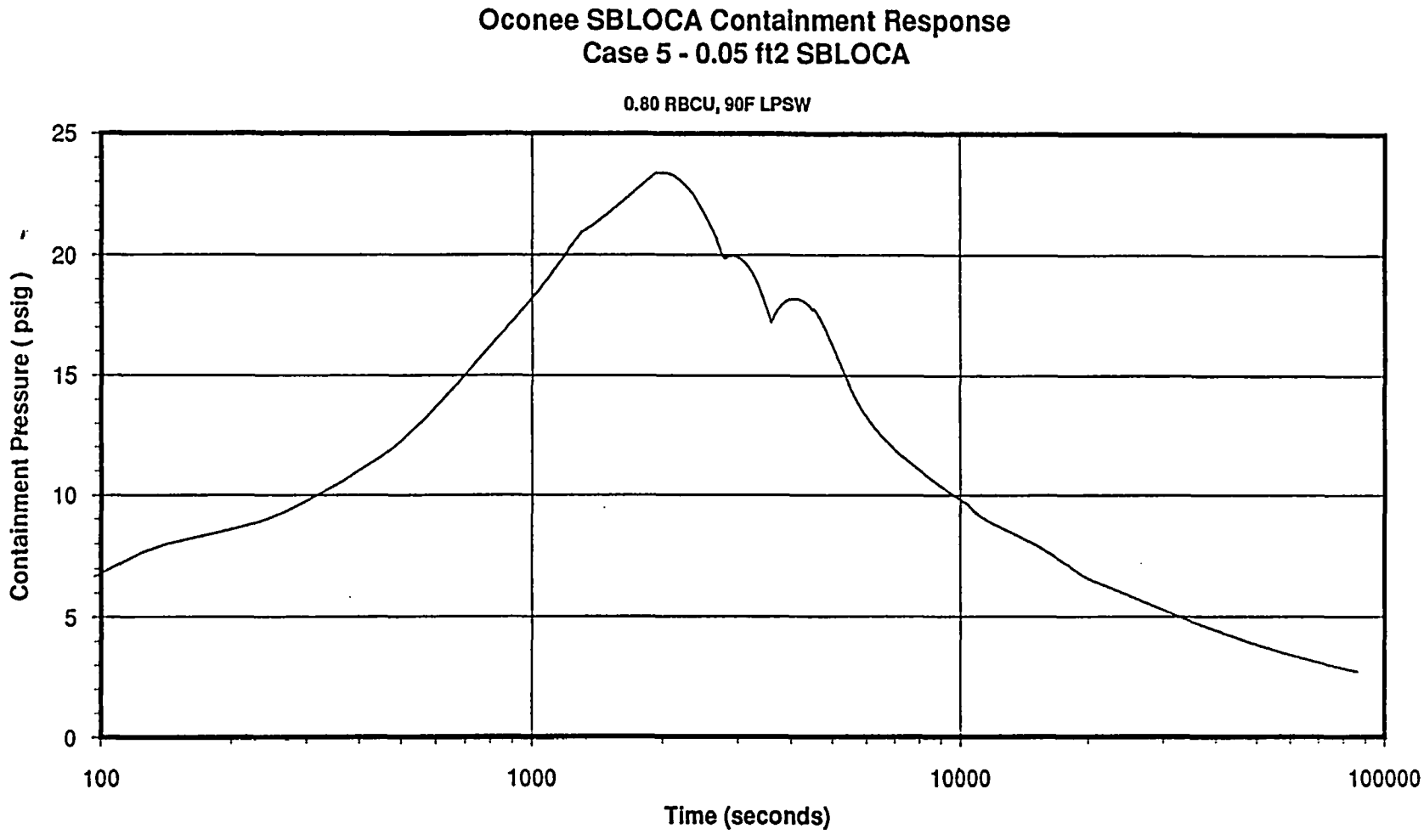


FIGURE 6.4-10

Building Pressure Profile

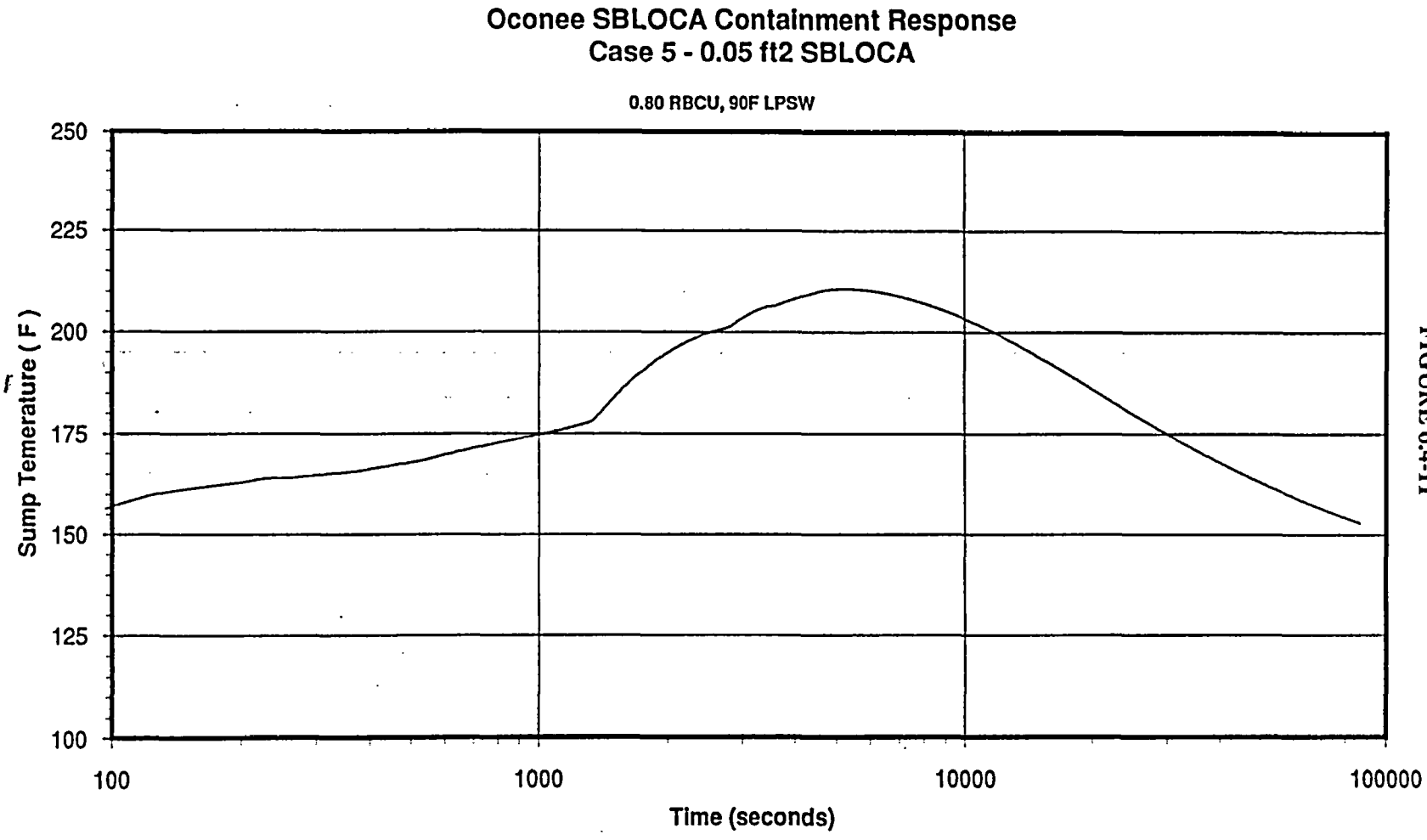


FIGURE 6.4-11

Sump Temperature Profile

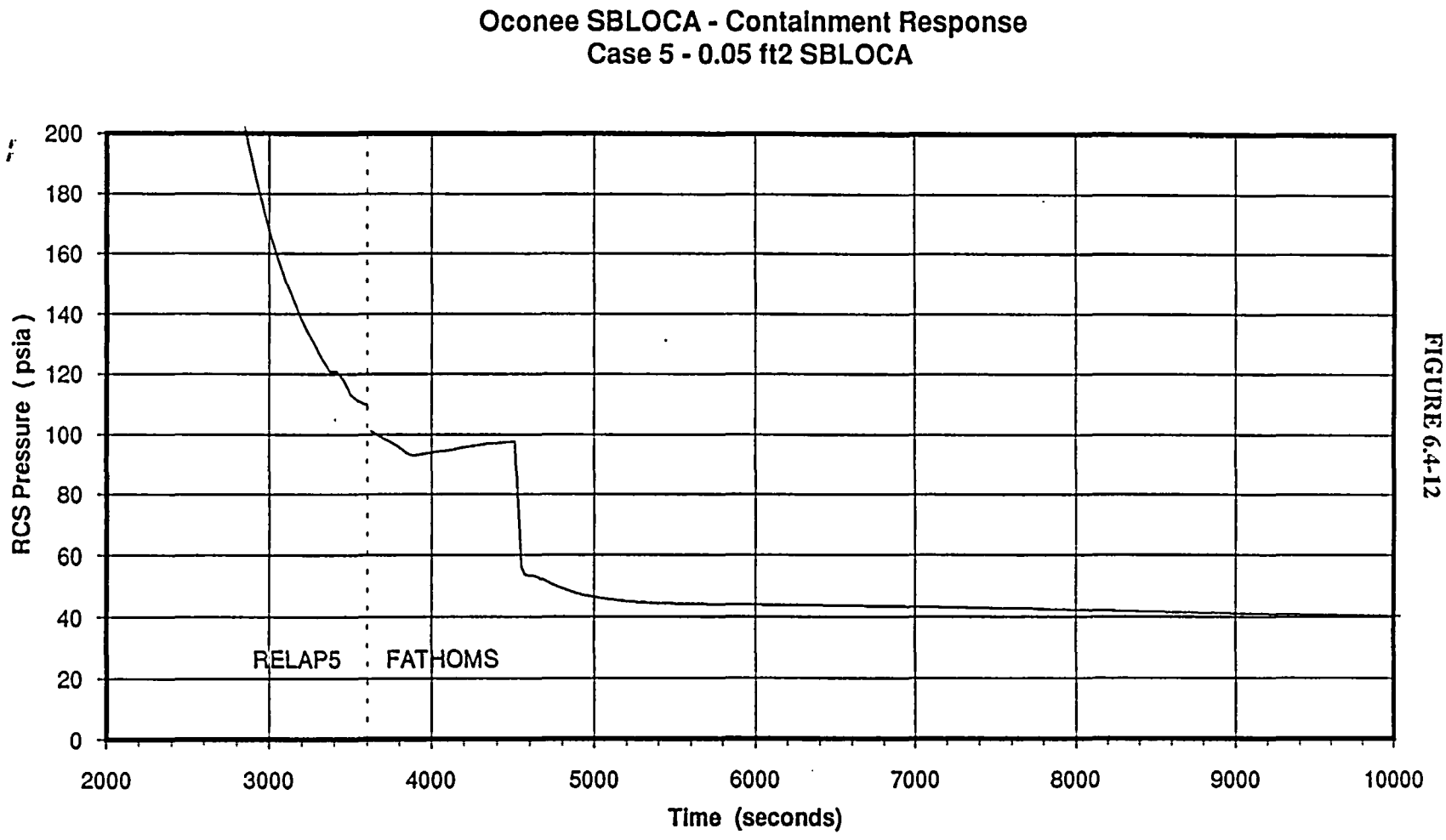


FIGURE 6.4-12

RCS Pressure

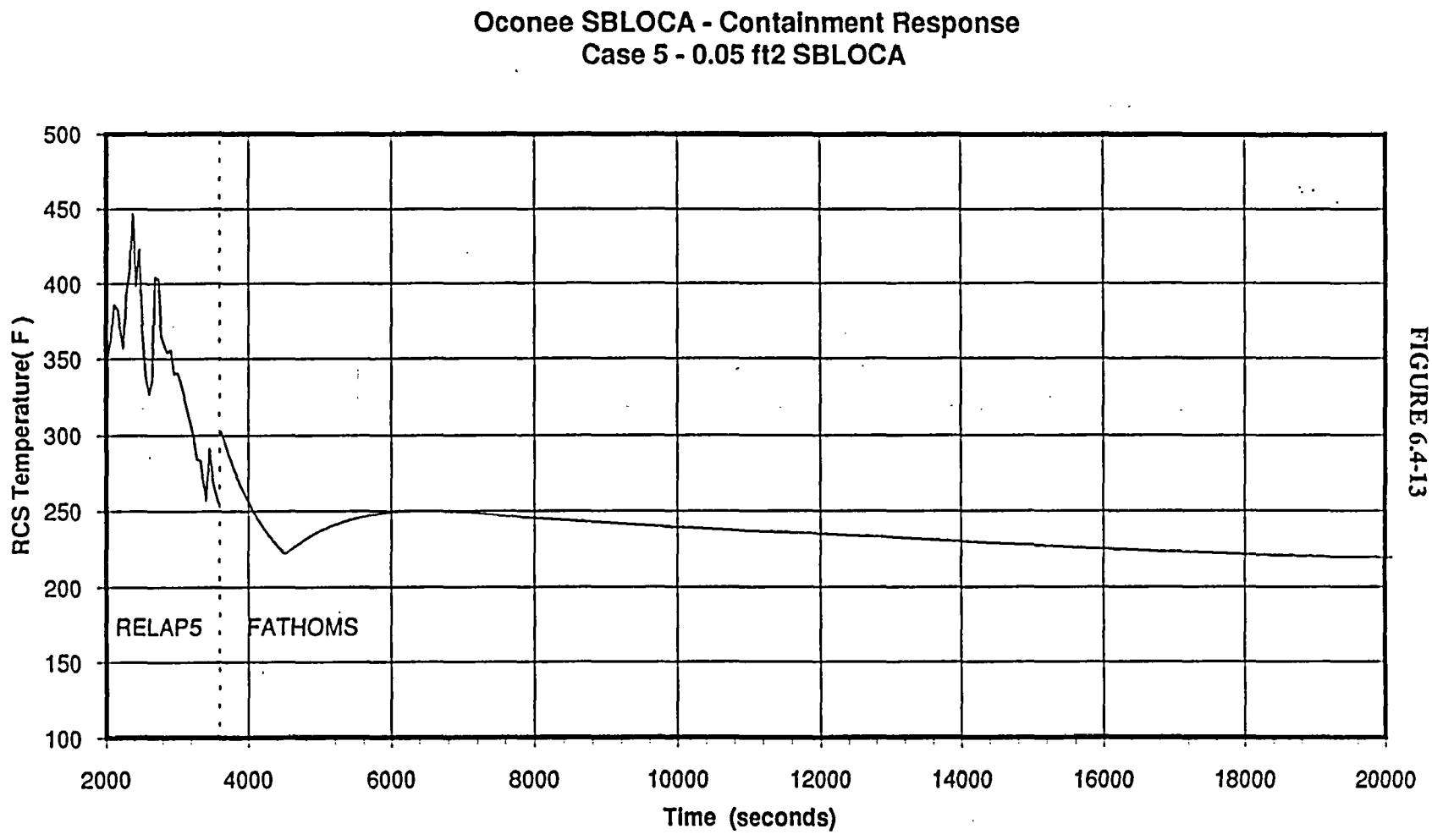


FIGURE 6.4-13

RCS Temperature

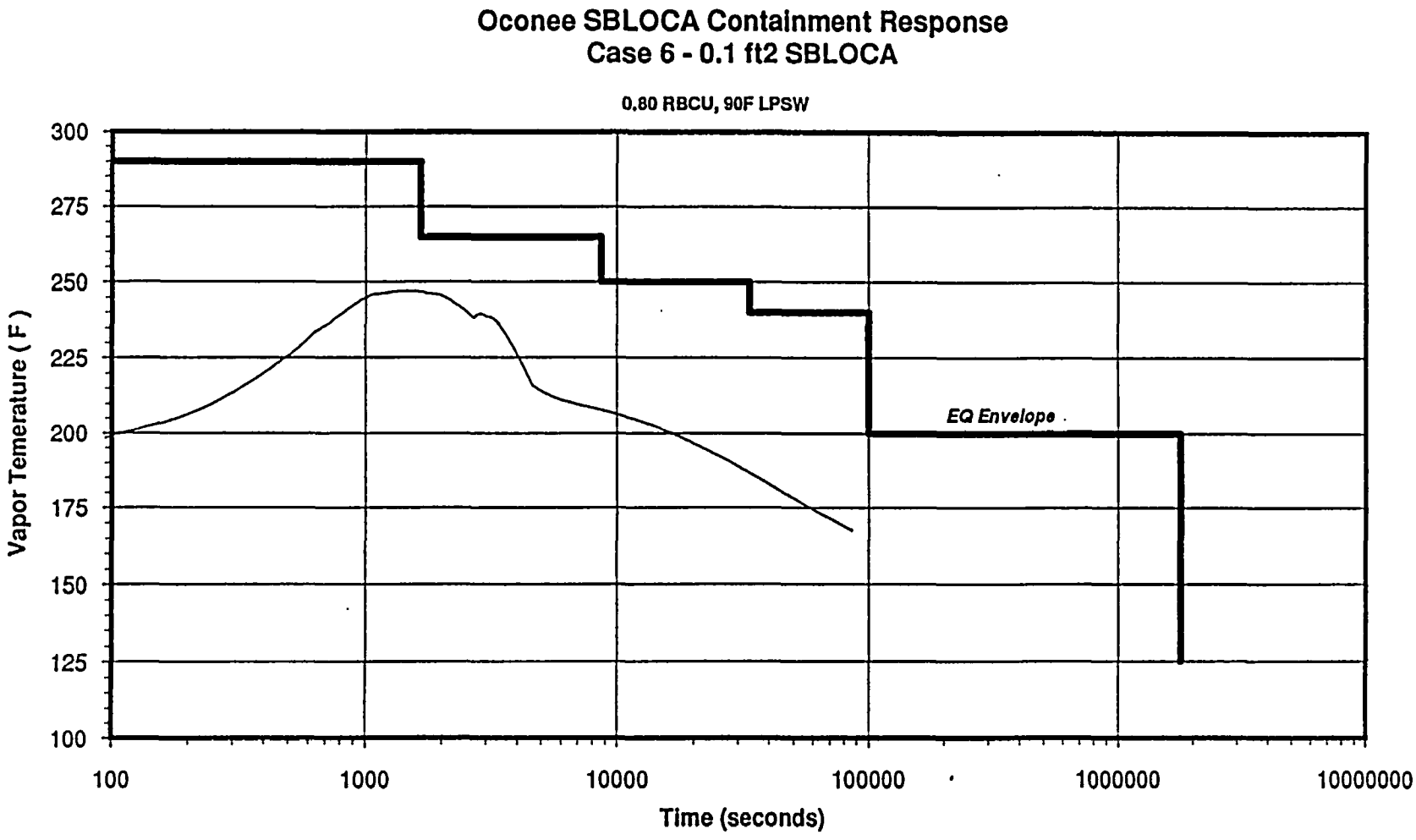


FIGURE 6.4-14

Vapor Temperature Profile

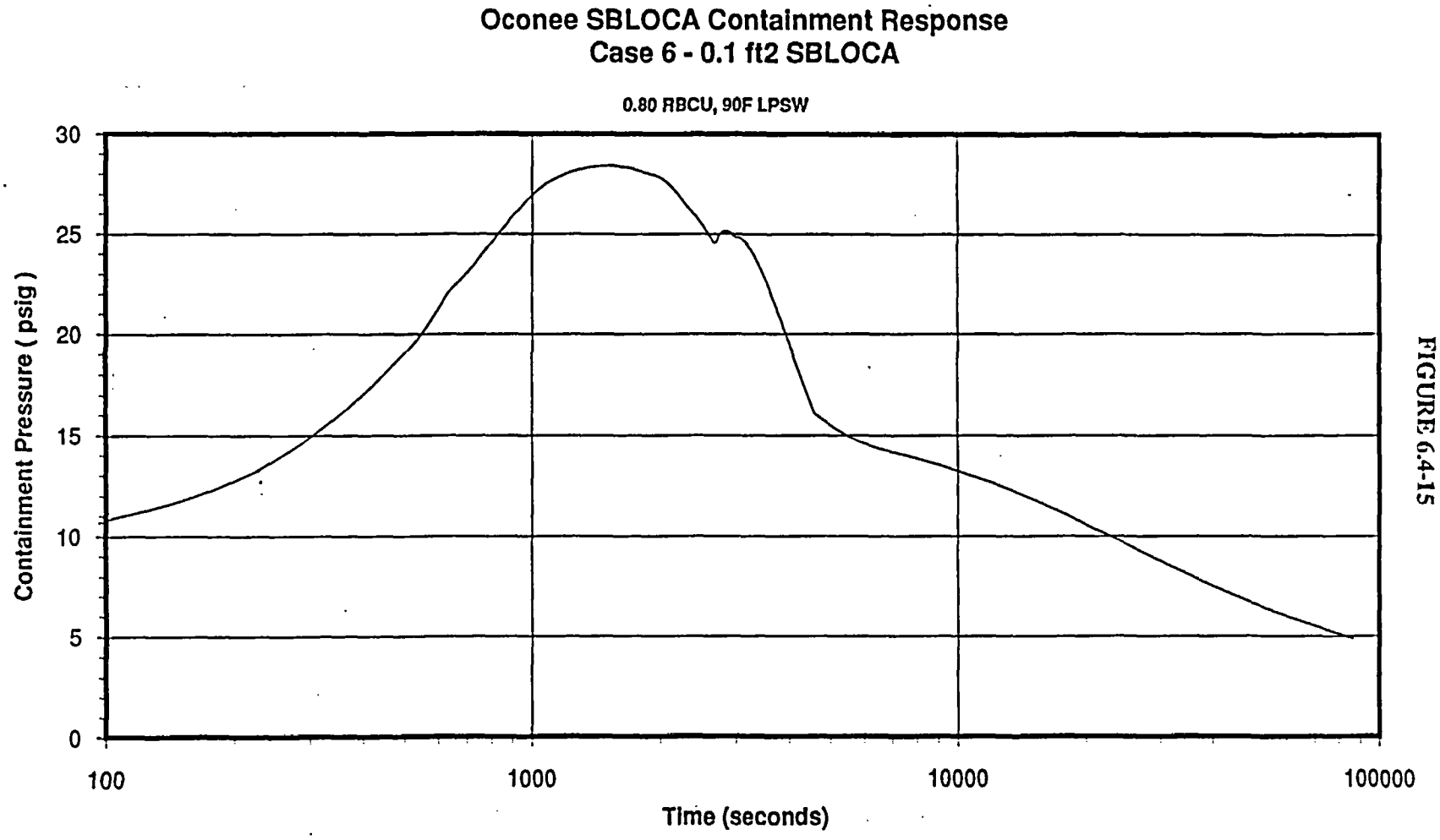


FIGURE 6.4-15

Building Pressure Profile

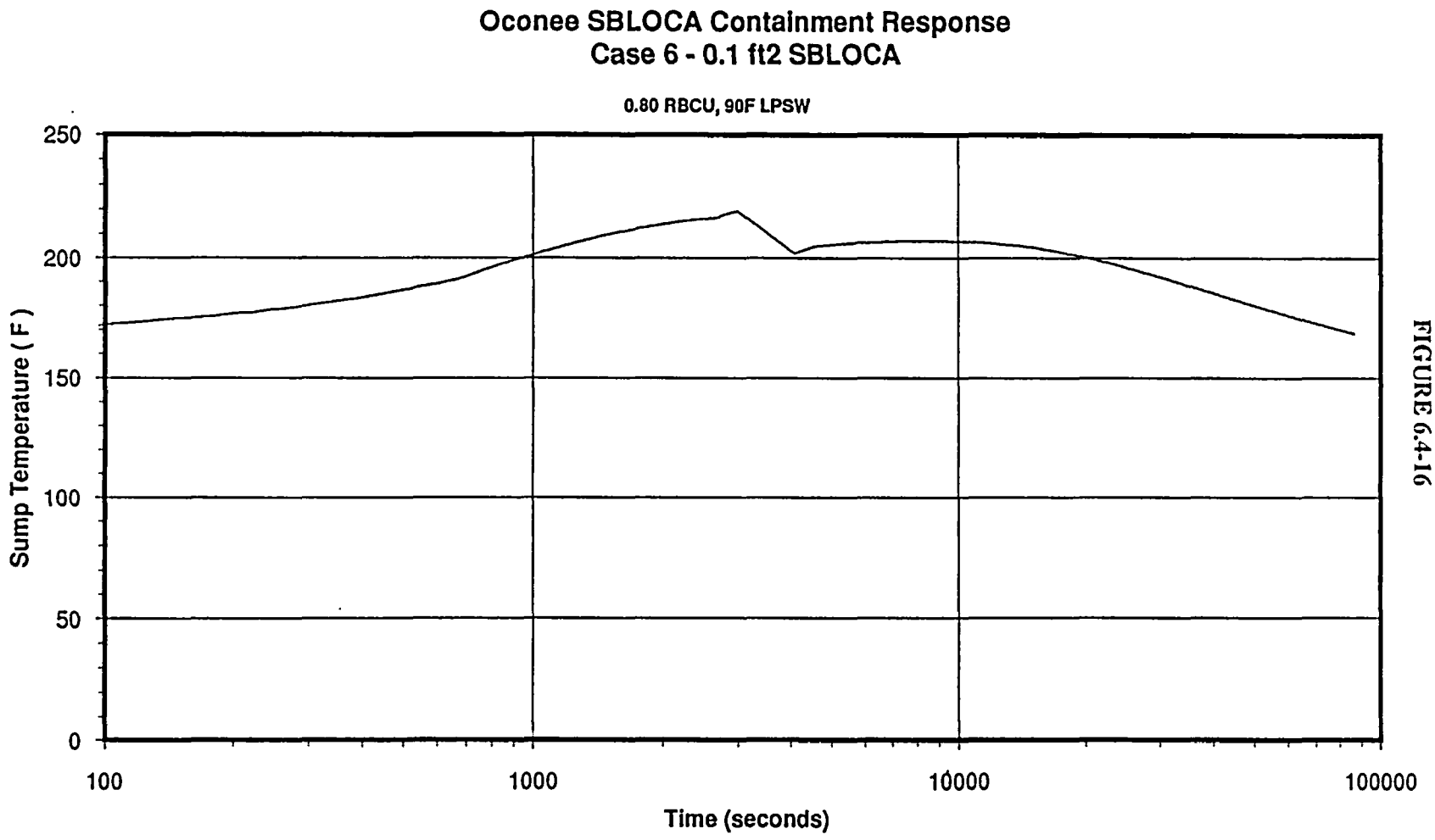


FIGURE 6.4-16

Sump Temperature Profile

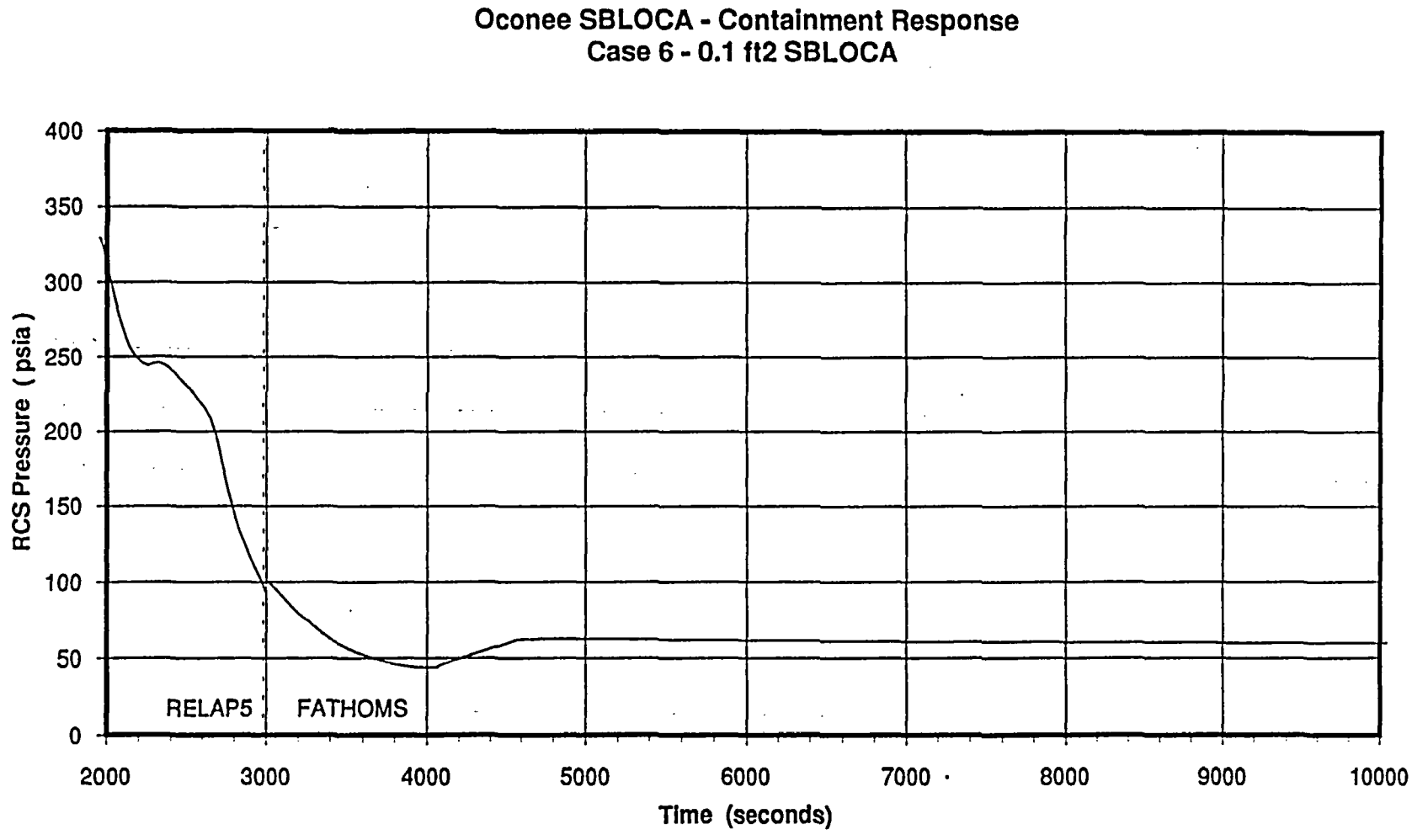


FIGURE 6.4-17

RCS Pressure

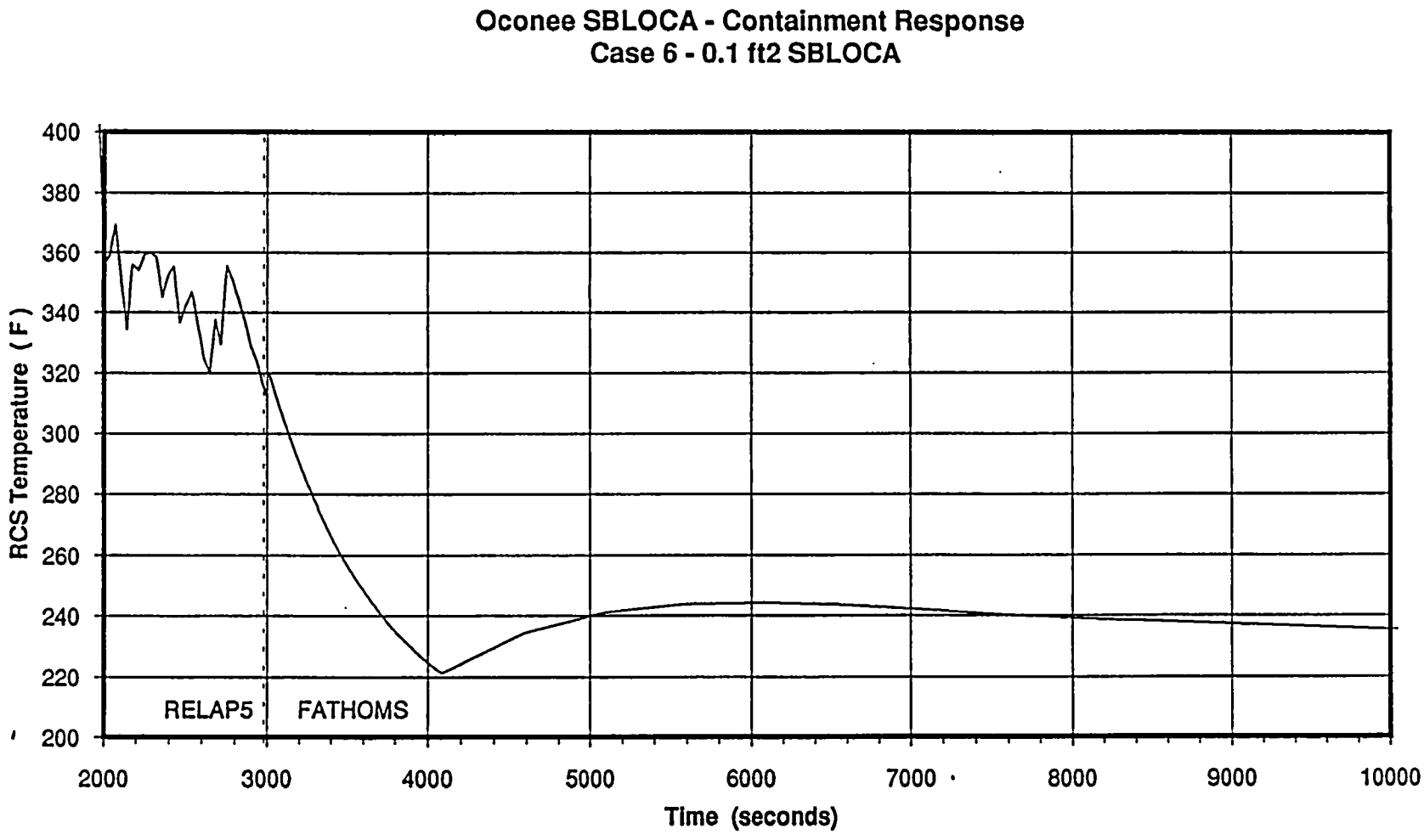


FIGURE 6.4-18

RCS Temperature

Oconee SBLOCA Containment Response
Cases 3-6

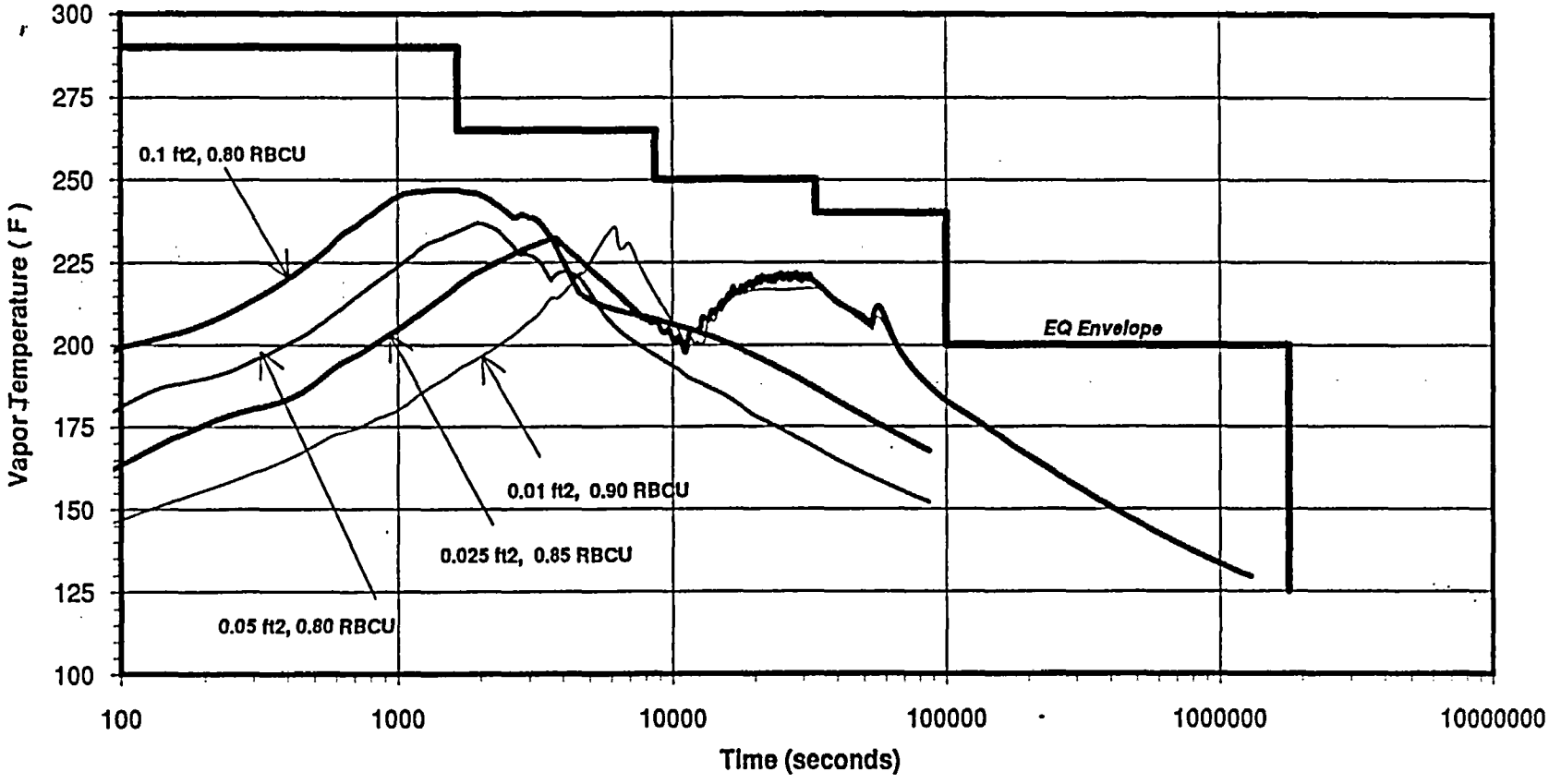


FIGURE 6.4-19

Oconee Steam Line Break Containment Pressure Response
Offsite Power Maintained

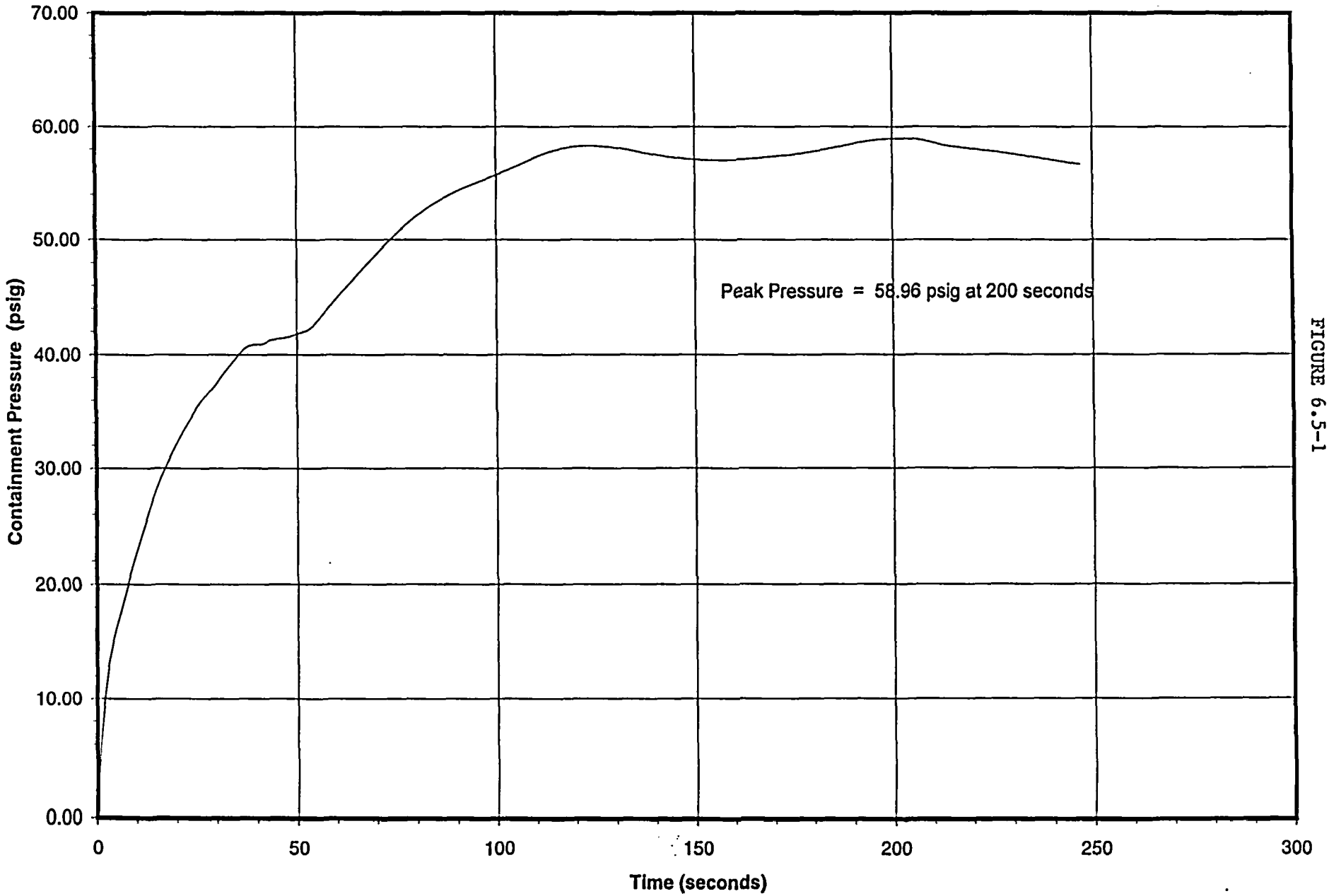


FIGURE 6.5-1

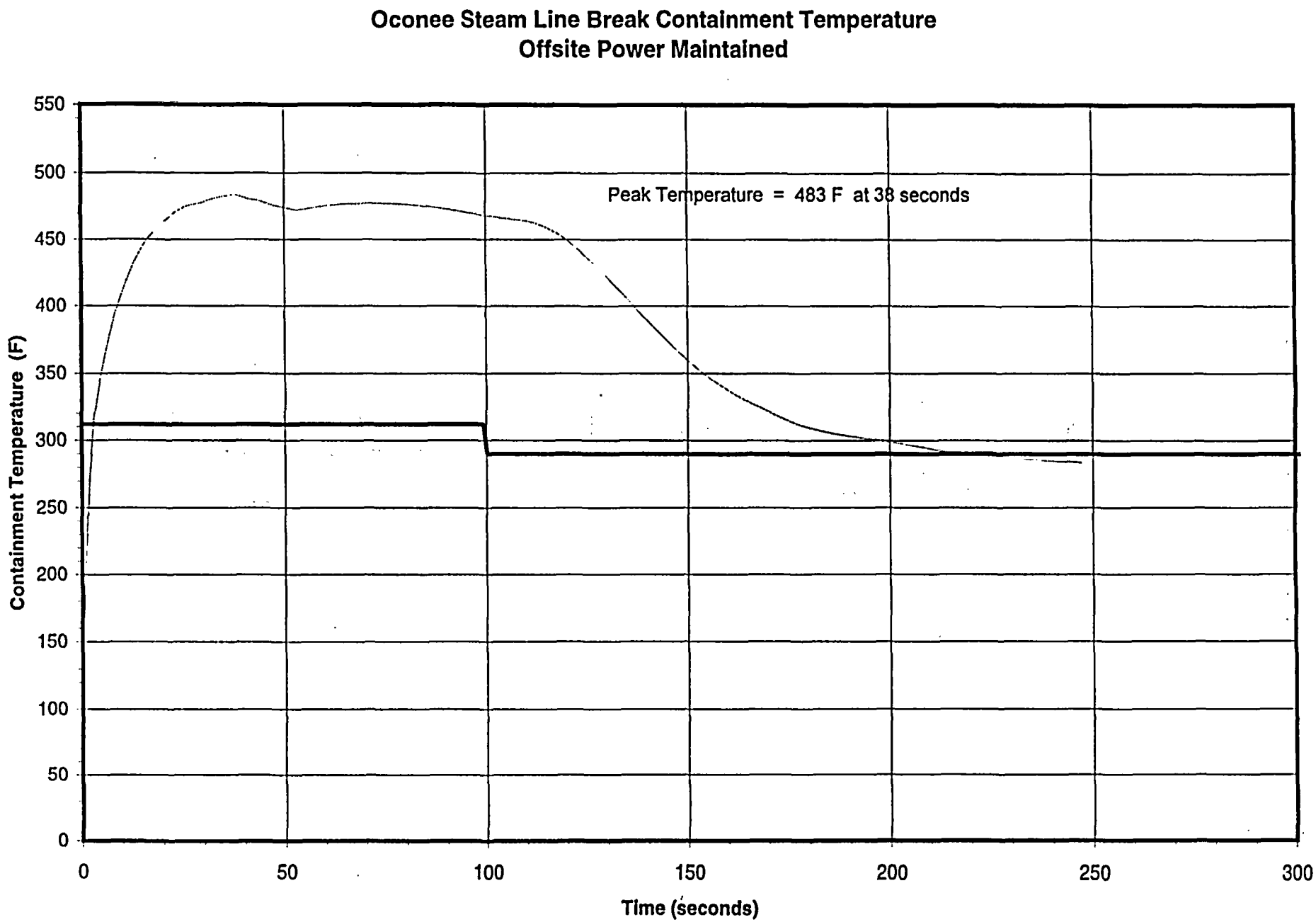

Oconee Steam Line Break Containment Temperature
Offsite Power Maintained
Peak Temperature = 483 F at 38 seconds
Containment Temperature (F)
550
500
450
400
350
300
250
200
150
100
50
0
0
50
100
150
200
250
300
Time (seconds)

FIGURE 6.5-2

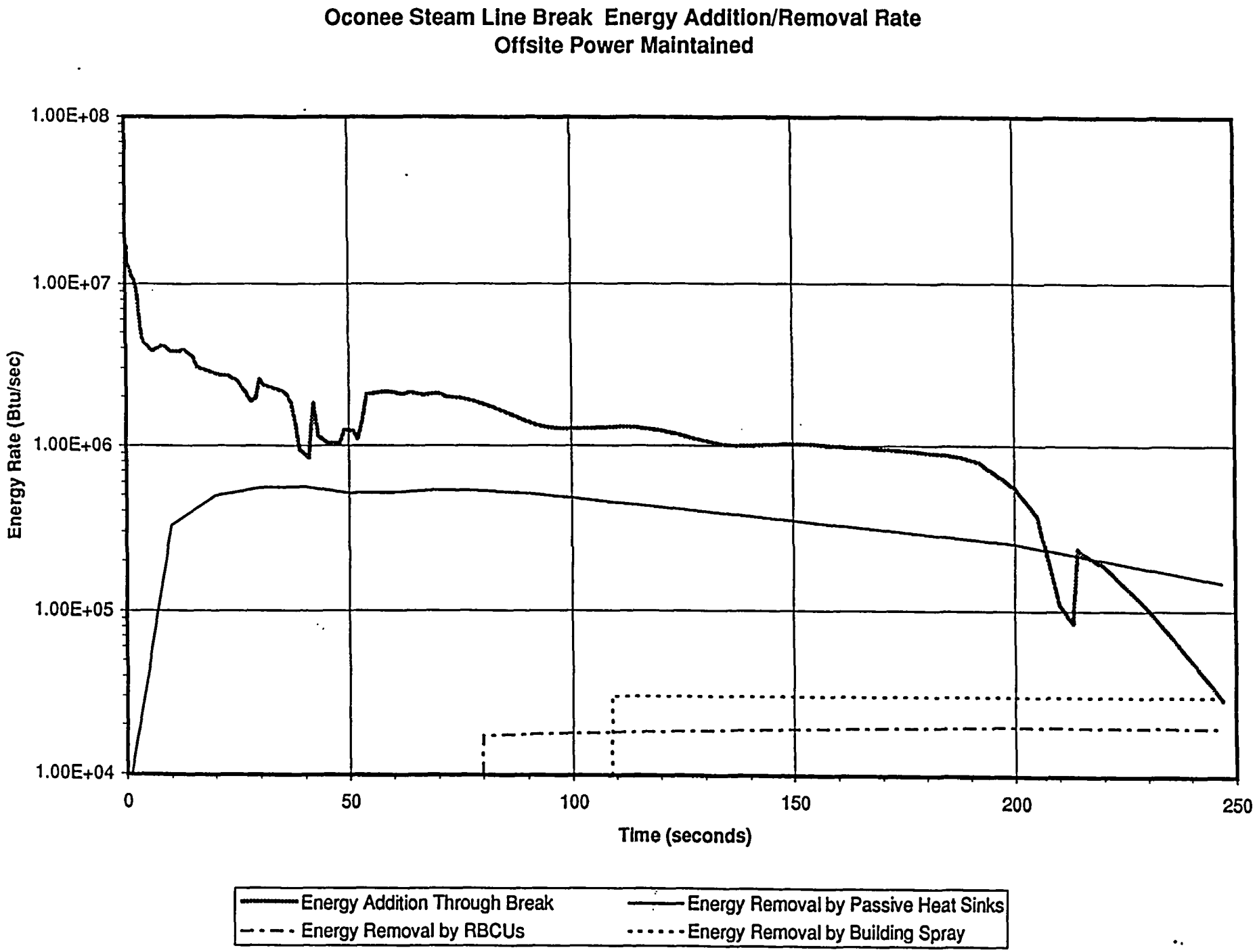

Oconee Steam Line Break Energy Addition/Removal Rate
Offsite Power Maintained
1.00E+08
1.00E+07
1.00E+06
1.00E+05
1.00E+04
Energy Rate (Btu/sec)
0
50
100
150
200
250
Time (seconds)
Energy Addition Through Break
Energy Removal by Passive Heat Sinks
Energy Removal by RBCUs
Energy Removal by Building Spray

FIGURE 6.5-3

Oconee Steam Line Break Containment Pressure Response
Loss of Offsite Power

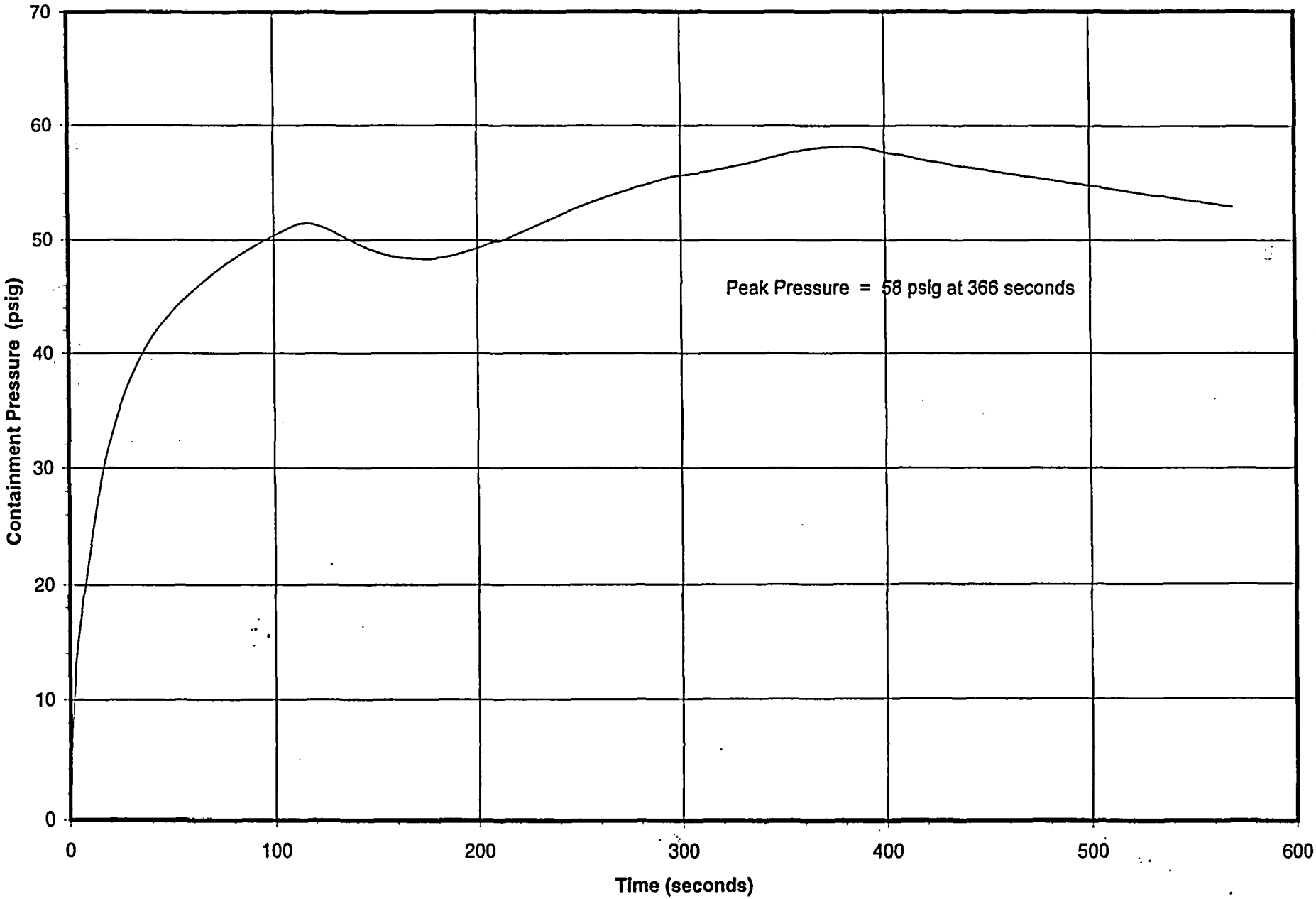


FIGURE 6.5-4

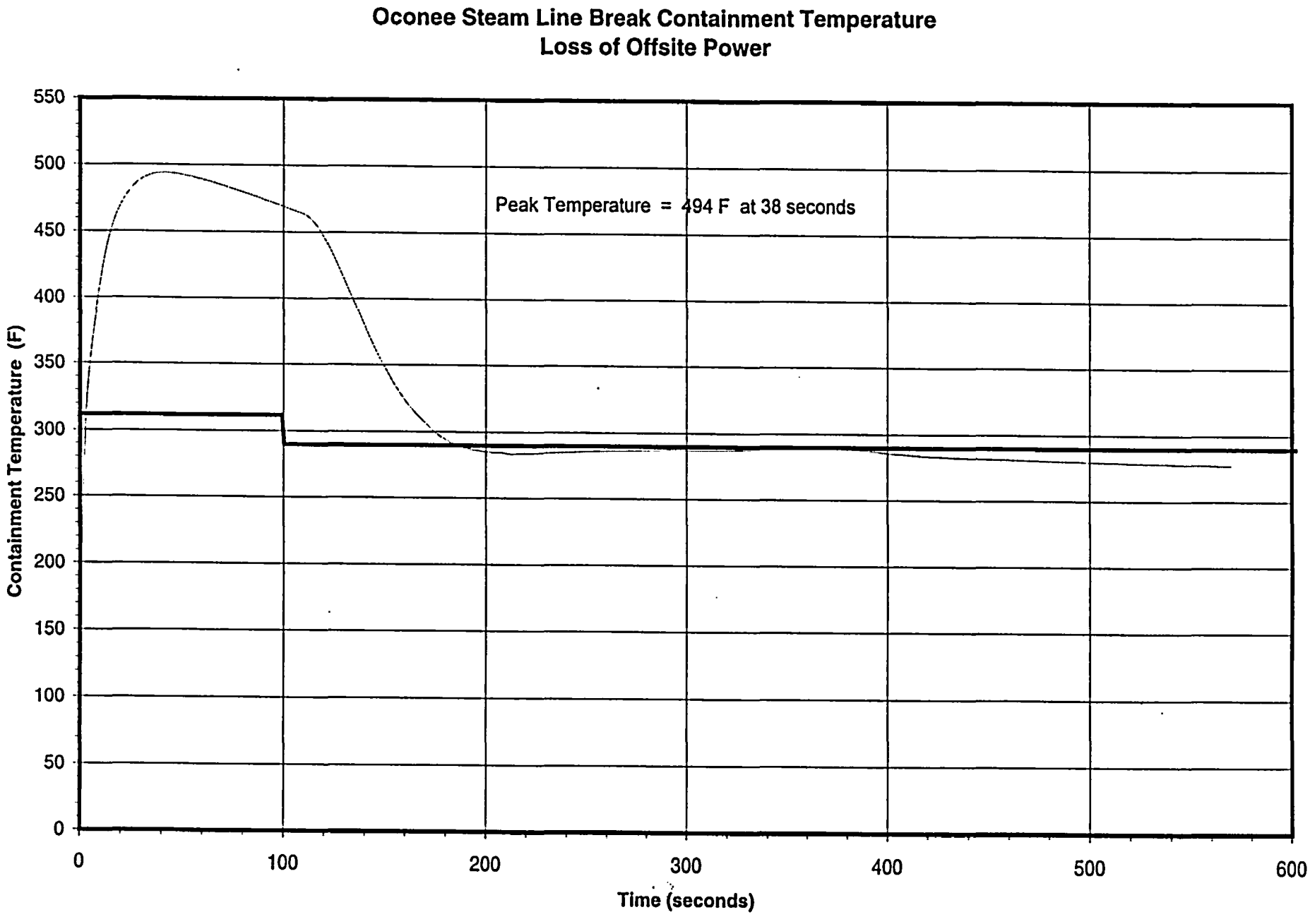

Oconee Steam Line Break Containment Temperature
Loss of Offsite Power
Peak Temperature = 494 F at 38 seconds
Containment Temperature (F)
550
500
450
400
350
300
250
200
150
100
50
0
0
100
200
300
400
500
600
Time (seconds)

FIGURE 6.5-5

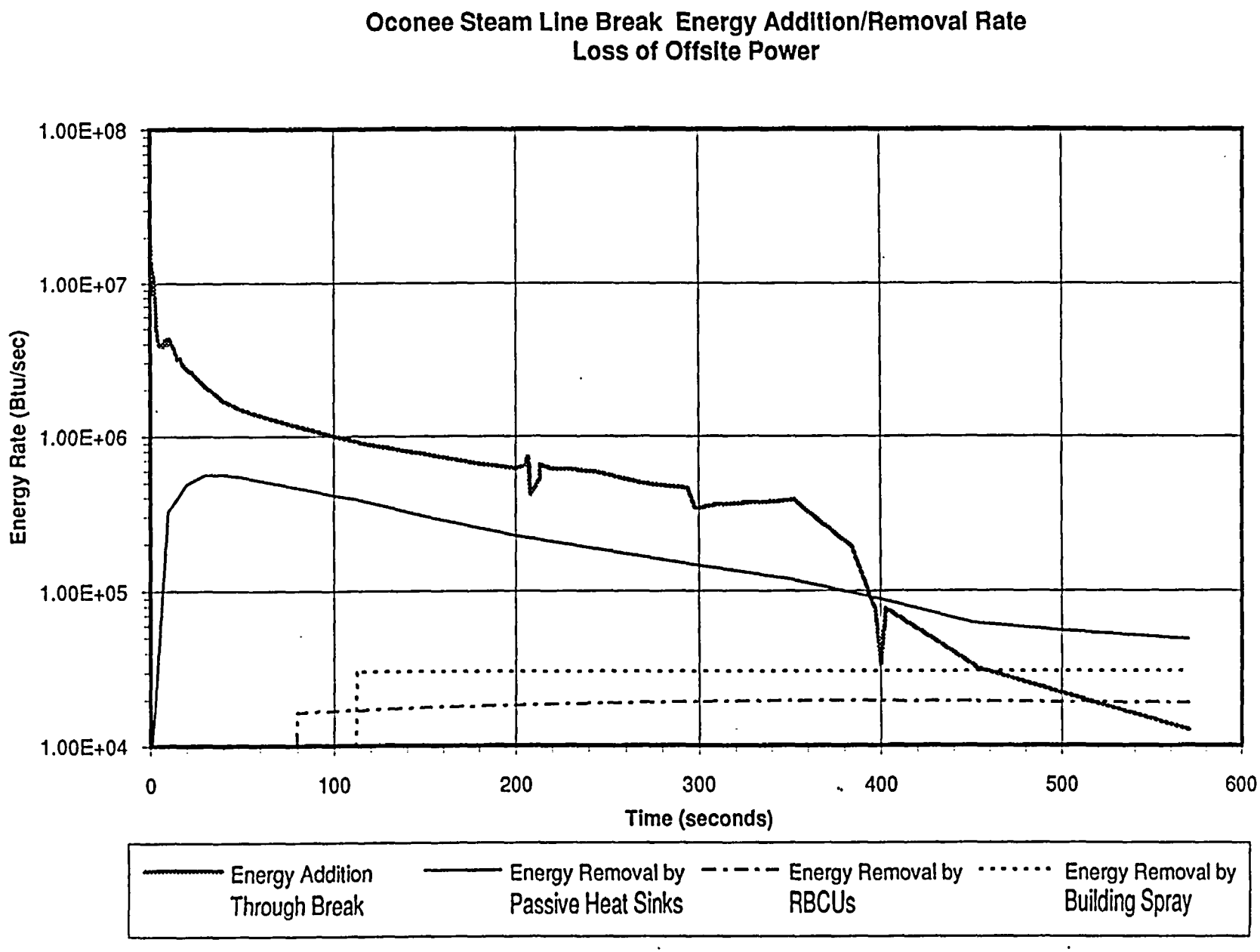

Oconee Steam Line Break Energy Addition/Removal Rate
Loss of Offsite Power
Energy Rate (Btu/sec)
1.00E+08
1.00E+07
1.00E+06
1.00E+05
1.00E+04
0
100
200
300
400
500
600
Time (seconds)
Energy Addition Through Break
Energy Removal by Passive Heat Sinks
Energy Removal by RBCUs
Energy Removal by Building Spray

FIGURE 6.5-6

7. SUMMARY AND CONCLUSIONS

This topical report describes the Duke Power Company methodology for simulating the mass and energy release and containment response to high energy line breaks for the Oconee Nuclear Station. The potentially limiting scenarios, which are large break LOCA, small break LOCA, and steam line break, have been reanalyzed with modern methods and updated initial and boundary conditions. These reanalyses have addressed all known issues regarding the validity of the existing UFSAR analyses, and benefit from twenty years of station operating experience. The results of the reanalyses have been shown to meet the acceptance criteria. Consequently, these reanalyses are suitably conservative and will replace the existing analyses. Based on these reanalyses the performance requirements for the affected engineered safeguards systems have been determined, as well as limits on the initial operating conditions of the plant.

Supplement 1 was included in the November 1997 publication of the report to provide the results of an application of the methodology with the Main Steam Line Break Detection and Main Feedwater Isolation System. This system was installed subsequent to the approval of the original topical report, and significantly changes the results by crediting automatic isolation of MFW and partial isolation of EFW. This application of the methodology was provided since it superceded the application in Chapters 5 and 6 of Revision 0.

Supplement 2 was added with the November 1997 version of Revision 0 of the report to present validation analyses comparing the FATHOMS prediction to data from the CVTR facility.

Appendix A was added in Revision 1 to describe the replacement steam generator (ROTSG) steam line break mass and energy release modeling with RETRAN-3D MOD003.1/DKE.

Appendix B was added in Revision 1 to address the RETRAN-3D SER conditions and limitations as relates to the replacement steam generator (ROTSG) steam line break mass and energy release modeling with RETRAN-3D MOD003.1/DKE.

Appendix C was added in Revision 1 to include the use of the GOTHIC 7.0/DUKE code for containment response analysis."

Supplement 1 to DPC-NE-3003-A

Main Steam Line Break Analysis With MSLB Detection and Feedwater Isolation System

Supplement 1 to DPC-NE-3003-A

Main Steam Line Break Analysis With MSLB Detection and Feedwater Isolation System

Introduction

This supplement describes the 1997 version of the UFSAR Section 6.2 main steam line break analysis for mass and energy release and containment response, which is the current licensing basis application of the steam line break analysis methodology approved per DPC-NE-3003-P. This reanalysis incorporates the recently designed and installed Oconee Main Steam Line Break (MSLB) Detection and Feedwater Isolation System. This plant modification was subsequent to the NRC review of the DPC-NE-3003-P topical report, and improves the automatic mitigation of a main steam line break by isolating main feedwater flow and reducing emergency feedwater flow. This analysis supercedes the analysis documented in Chapters 5&6 of DPC-NE-3003-PA. Chapters 5&6 are left intact in the body of the topical report since the methodology is the same. However, the analysis in Chapters 5&6 is no longer the current licensing basis analysis. It is also noted that the analysis in Chapter 6 was revised in DPC-NE-3003-PA and DPC-NE-3003-A, from what was originally in DPC-NE-3003-P, to correct an error in the FATHOMS code.

The function of the MSLB Detection and Feedwater Isolation System is to detect a main steam line break due to low steam generator pressure, and then to automatically respond by isolating all main feedwater to both steam generators and to stop the turbine-driven emergency feedwater pump. In the steam line break analysis methodology this system is modeled by changes in the main feedwater and emergency feedwater flow rate boundary conditions. These changes in feedwater flow boundary conditions result in different limiting single failures. All of these considerations and the revised results are discussed below. These analysis revisions are considered to be input changes to the methodology and do not constitute methodology revisions requiring NRC review.

This supplement is included with the approved versions of DPC-NE-3003 in order to clarify that the steam line break analysis applications in the original version of DPC-NE-3003 are no longer the current licensing basis and to provide the revised UFSAR analyses.

Revised Steam Line Break Analysis Boundary Conditions

Main Steam Line Break Detection and Feedwater Isolation System

The MSLB Detection and Feedwater Isolation System is designed to isolate main feedwater to both steam generators following a main steam line break. Steam generator outlet pressure and turbine header pressure are used as input signals to the circuitry for main steam line break detection and feedwater isolation. When a main steam line break is sensed (500 psig including a 50 psig uncertainty) or upon manual actuation, the main and startup feedwater control valves are closed to isolate the main feedwater flow paths to both steam generators. In addition, the main feedwater pumps are tripped. The turbine driven emergency feedwater pump is also stopped. The MSLB Detection and Feedwater Isolation circuitry is safety related and single failure proof. However, some of the equipment that is actuated by the circuitry is neither safety related nor single failure proof as described in Reference S1-1.

Loss of Offsite Power

A loss of offsite power does not impact the ability of the MSLB Detection and Feedwater Isolation System to perform its intended safety function. If a loss of offsite power occurs during a main steam line break, a sufficient inventory of air will continue to be available to drive the main and startup MFW control valves closed.

Single Failure

The single failure of the EFW control valve to the affected steam generator to close is the limiting single failure, since mass addition to the affected steam generator is maximized.

Main Feedwater Initial Conditions

The limiting assumption with respect to maximizing the feedwater addition to the affected steam generator is for the MFW control valves to be in manual control. A conservatively high MFW pump speed of 4400 rpm is assumed when the MFW control valves are in manual control.

Emergency Feedwater System

All three emergency feedwater (EFW) pumps automatically start upon a loss of both main feedwater pumps or a low steam generator level. As a conservative assumption, the analysis for the failure of the EFW control valve assumes that EFW is actuated when the steam line break occurs and that maximum EFW flow is delivered. The turbine-driven EFW pump and one motor-driven EFW pump supply flow to the steam generator with the broken steam line until flow from the turbine-driven EFW pump is stopped by the MSLB Detection and Feedwater Isolation System. The other motor-driven EFW pump supplies flow to the unaffected steam generator. A constant EFW temperature of 130°F is assumed. Operator action is credited at 10 minutes to isolate EFW flow to the affected steam generator.

ECCS

The HPI System is modeled using a fill table with two HPI pumps injecting through two trains. This is consistent with the limiting single failure.

Results

RETRAN-02 System Analysis

The transient response is presented in Table S1-1 and Figures S1-1 through S1-5. The mass and energy release data is shown in Table S1-2. Reactor coolant pressure (Figure S1-1) rapidly decreases until the unaffected loop saturates at about 10 seconds. After this time, RCS pressure continues to decrease at a slower rate, eventually leading to HPI and core flood tank injection. RCS pressure begins to slowly increase after 300 seconds due to core flood tank injection.

The cooldown in the unaffected loop is less severe than in the affected loop (Figure S1-2). The cold leg temperature in the unaffected loop increases once the turbine stop valves close. The unaffected loop ΔT remains very small until this loop begins to void and flow degrades. By approximately 350 seconds, the hot and cold leg temperatures of both loops are nearly identical.

Reactor power (Figure S1-3) rapidly decreases after reactor trip. The combination of control rod worth and negative reactivity from injected boron prevents any return to power.

Break flow (Figure S1-4) increases rapidly after initiation of the steam line break. After the turbine stop valves close, break flow from the unaffected steam generator is terminated. Beyond this point, break flow from the affected steam generator decreases with decreasing pressure. As

steam generator pressure rapidly decreases below the setpoint pressure, MFW flow is isolated, the MFW pumps are tripped, and the turbine driven EFW pump is stopped. The break flow is sustained by the motor driven EFW pump feeding the affected steam generator with its associated EFW control valve failed open. Operator action is credited at 10 minutes to isolate EFW flow to the affected steam generator and effectively terminate the mass and energy release. Break enthalpy (Figure S1-5) corresponds to superheated steam throughout the transient.

FATHOMS Containment Analysis

The peak containment pressure (Figure S1-6) is 54 psig and occurs at 70 seconds, well below the design limit of 59 psig. The RBCU actuation signal occurs at 2 seconds, and the units are running at 80 seconds. The containment spray actuation signal occurs at 17 sec, and the spray flow starts at 109 seconds. The simulation is run for 700 seconds. The containment temperature (Figure S1-7) exceeds the EQ limit for approximately 160 seconds in the early part of the transient. The peak temperature during this period is 490°F at 40 seconds. The equipment required for post-accident equipment qualification have been evaluated for the steam line break containment pressure and temperature and have been shown to be capable of performing their design functions.

References

S1-1 Letter from J.W. Hampton (Duke) to USNRC, June 14, 1995 (Supplemental Response to IE Bulletin 80-04, Delay of NRC Commitment Item).

Table S1-1

Sequence of Events
Steam Line Break - Offsite Power Maintained
MSLB Detection and Feedwater Isolation System Available

Event	Time (seconds)
Break occurs	0.0
Reactor trip on variable low pressure	1.7
Third CBP starts	2.05
Turbine stop valves close	2.7
MFW isolation setpoint reached	3.5
HPI actuates	20.3
MFW isolation complete	28.5
HPI boron injection begins	69.4
CFT injection begins	131.7
Boron from CFT B starts	221.1
EFW isolation of SG A begins	600.0
EFW isolation of SG A complete	613.0
Simulation ends	700.0

Table S1-2

Steam Line Break Mass and Energy Release Data - Offsite Power Maintained
MSLB Detection and Feedwater Isolation System Available

Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)	Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)	Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)	Time (sec)	Mass Flow (lbm/sec)	Enthalpy (Btu/lbm)
0.0	0.0	1260.3	15.0	3004.1	1266.0	43.0	2076.8	1244.9	170.0	92.3	1257.7
0.5	10157.6	1281.2	16.0	3126.7	1266.2	44.0	2040.1	1244.7	175.0	132.8	1256.6
1.0	9484.9	1286.3	17.0	3033.1	1266.0	45.0	2000.2	1244.6	180.0	68.7	1257.3
1.5	8593.8	1288.8	18.0	2958.4	1265.2	46.0	1957.6	1244.4	185.0	188.2	1257.0
2.0	8064.0	1290.3	19.0	2895.4	1263.8	47.0	1950.8	1242.8	195.0	140.6	1257.0
2.5	6719.1	1293.0	20.0	2835.4	1262.5	48.0	1913.2	1242.2	200.0	112.6	1257.3
3.0	4813.3	1296.0	21.0	2785.3	1261.9	49.0	1843.8	1242.0	210.0	177.3	1256.5
3.5	3838.4	1298.5	22.0	2743.9	1260.5	50.0	1770.1	1242.3	230.0	127.2	1257.0
4.0	3440.1	1299.8	23.0	2703.2	1258.6	55.0	1417.7	1244.0	250.0	38.3	1257.7
4.5	3282.4	1300.3	24.0	2664.7	1256.4	60.0	1105.7	1243.3	270.0	60.1	1258.7
5.0	3217.2	1300.2	25.0	2611.8	1254.4	65.0	904.5	1243.9	300.0	93.3	1259.5
5.5	3177.6	1299.6	26.0	2550.8	1252.5	70.0	749.5	1245.3	320.0	102.5	1259.6
6.0	3075.2	1298.8	27.0	2490.7	1250.9	75.0	499.8	1245.6	340.0	106.5	1259.1
6.5	3039.4	1297.7	28.0	2442.7	1249.6	80.0	212.5	1246.5	360.0	107.7	1258.3
7.0	3060.8	1295.7	29.0	2407.7	1248.6	85.0	62.3	1249.2	380.0	106.5	1257.5
7.5	3115.4	1293.4	30.0	2381.1	1247.8	95.0	4.6	1251.2	400.0	103.1	1256.7
8.0	3188.5	1290.9	31.0	2360.7	1247.2	110.0	79.4	1253.3	450.0	92.8	1254.5
8.5	3266.6	1288.3	32.0	2340.1	1246.7	120.0	39.1	1255.7	500.0	100.2	1252.0
9.0	3293.4	1285.8	33.0	2319.9	1246.3	125.0	99.9	1255.7	550.0	96.1	1249.2
9.5	3268.5	1283.4	34.0	2300.1	1245.8	130.0	46.9	1256.4	600.0	99.1	1246.6
10.0	3249.1	1281.3	36.0	2260.7	1245.0	140.0	120.2	1257.1	610.0	126.8	1246.0
11.0	3357.3	1277.4	38.0	2221.8	1244.8	145.0	35.9	1256.9	620.0	20.5	1245.7
12.0	3191.7	1274.8	40.0	2166.1	1245.0	155.0	135.5	1257.4	630.0	73.5	1246.1
13.0	3297.9	1272.8	41.0	2137.8	1245.1	160.0	29.9	1257.3	650.0	0.3	1246.5
14.0	3084.0	1268.5	42.0	2109.7	1245.0	165.0	161.0	1256.9	700.0	0.3	1248.7

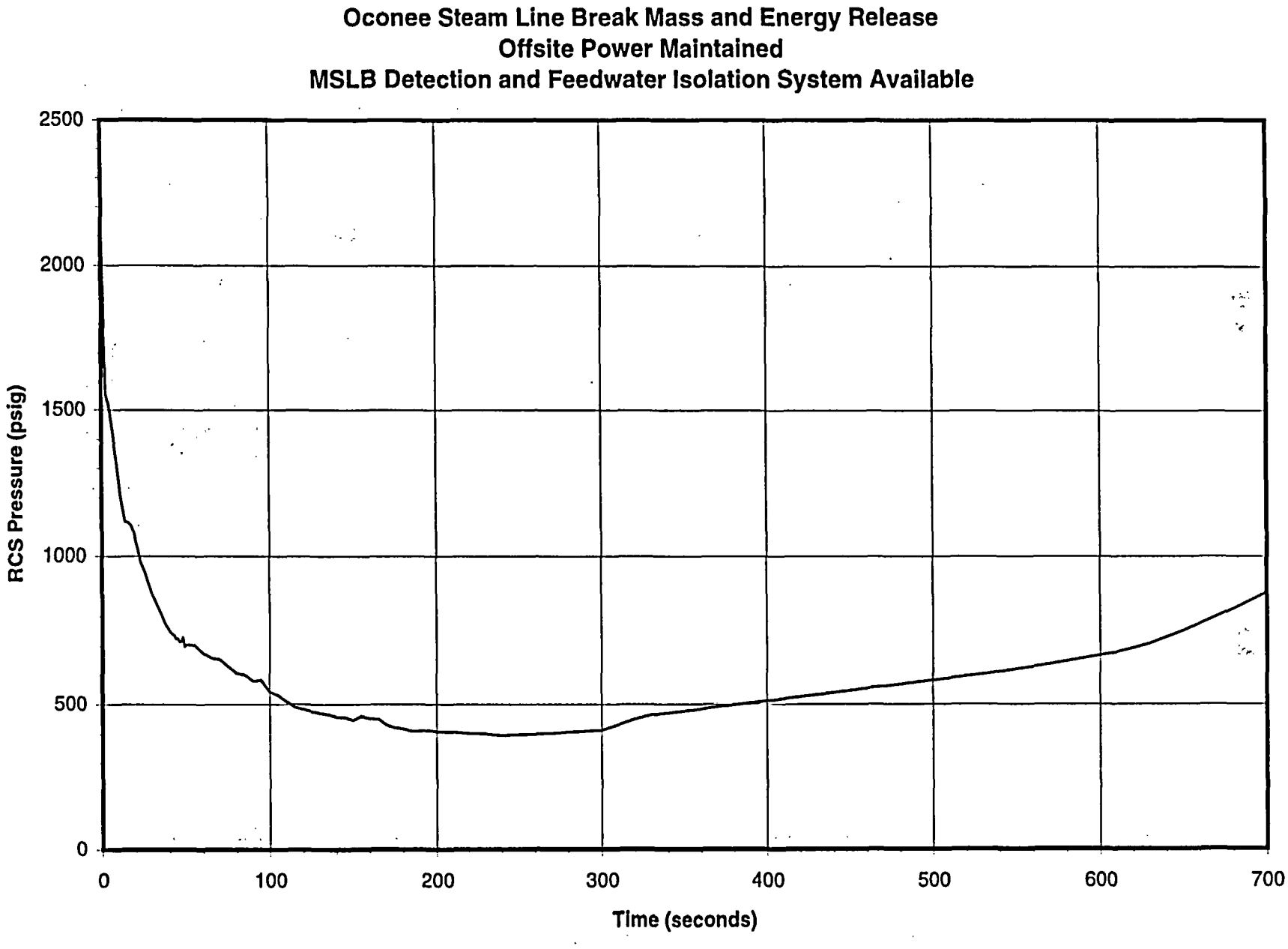

Oconee Steam Line Break Mass and Energy Release
Offsite Power Maintained
MSLB Detection and Feedwater Isolation System Available
RCS Pressure (psig)
2500
2000
1500
1000
500
0
0
100
200
300
400
500
600
700
Time (seconds)

Figure S1-1

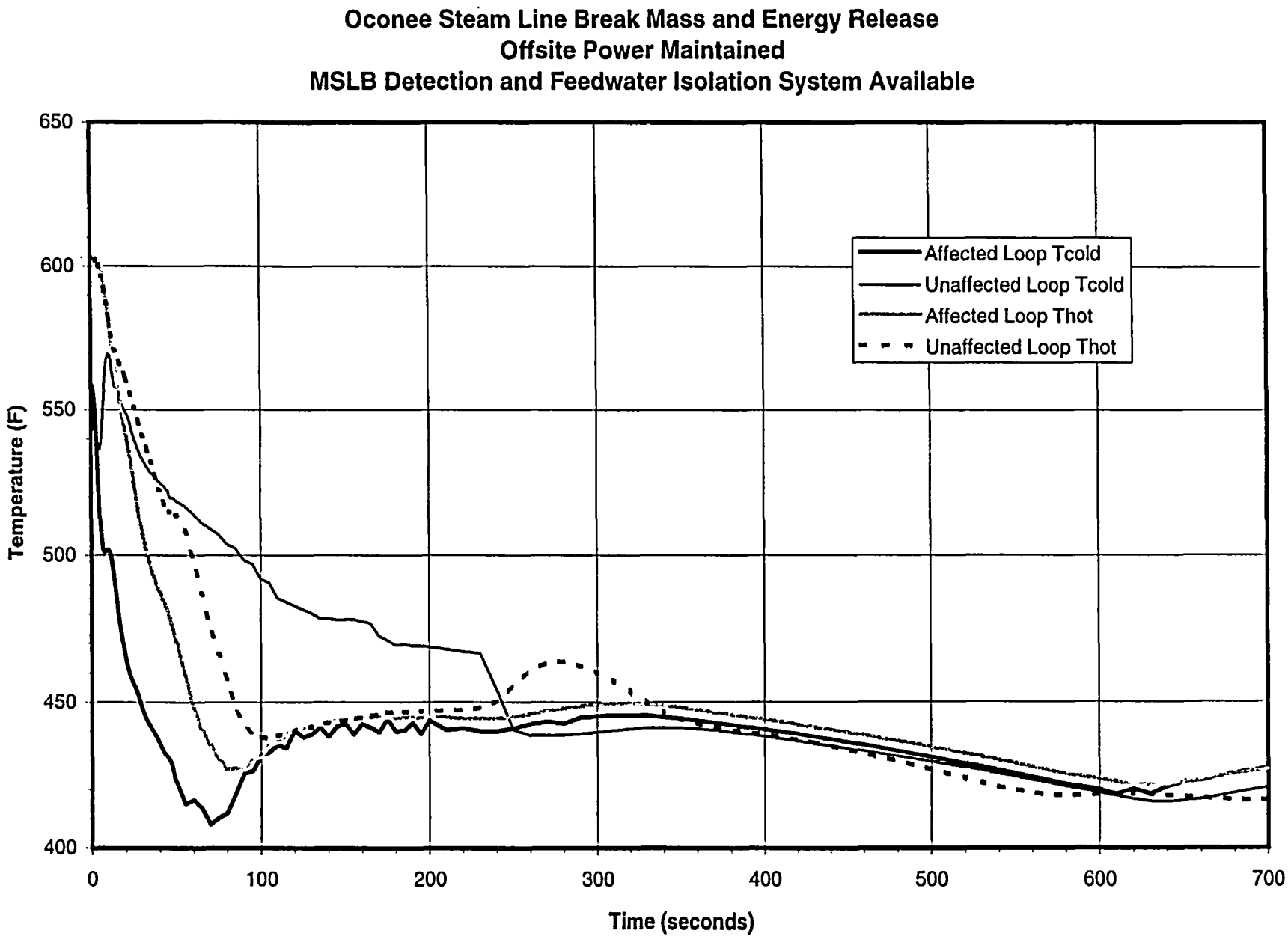

Oconee Steam Line Break Mass and Energy Release
Offsite Power Maintained
MSLB Detection and Feedwater Isolation System Available
Affected Loop Tcold
Unaffected Loop Tcold
Affected Loop Thot
Unaffected Loop Thot
Temperature (F)
650
600
550
500
450
400
0
100
200
300
400
500
600
700
Time (seconds)

Figure S1-2

Oconee Steam Line Break Mass and Energy Release
Offsite Power Maintained
MSLB Detection and Feedwater Isolation System Available

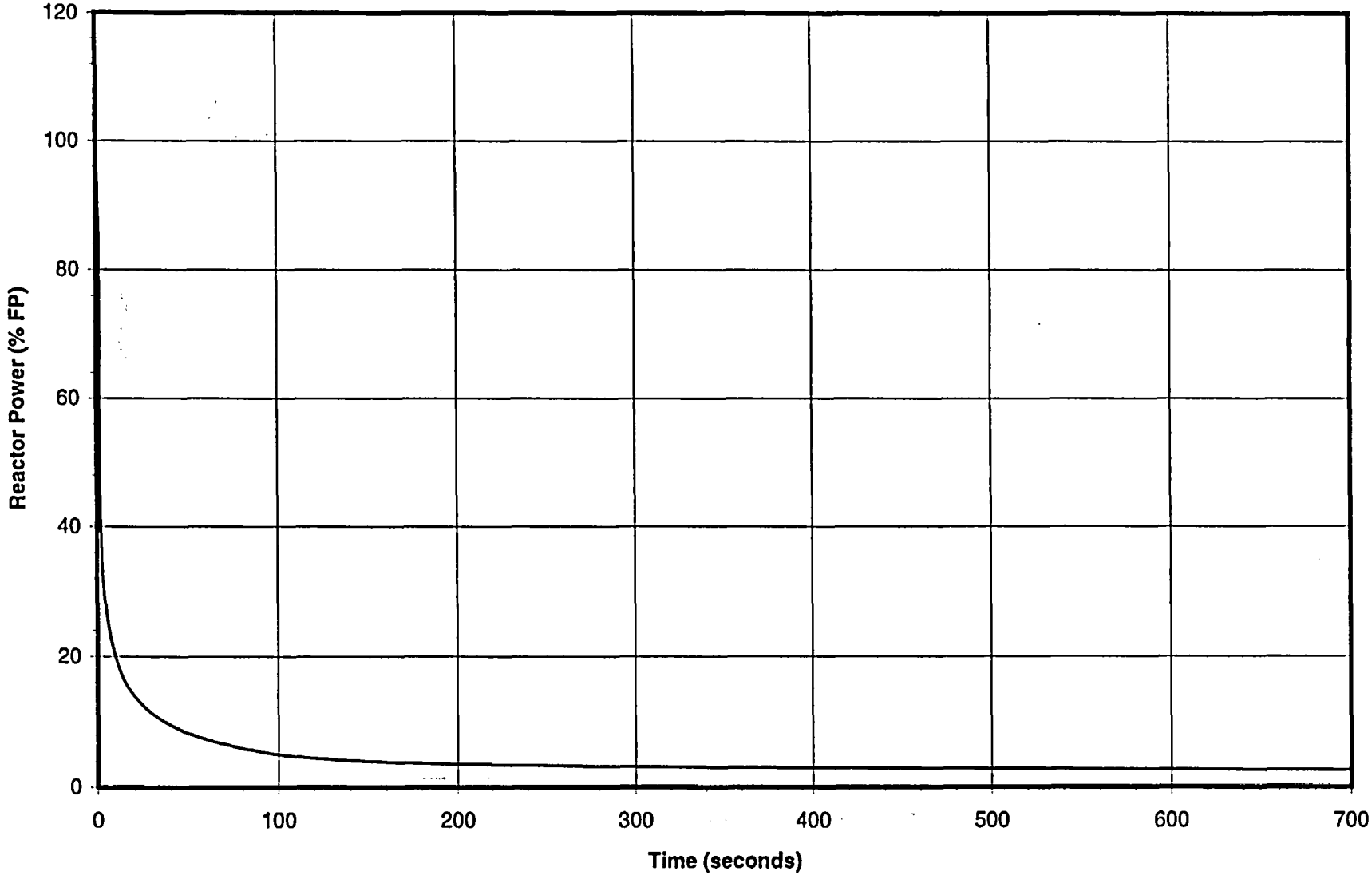


Figure S1-3

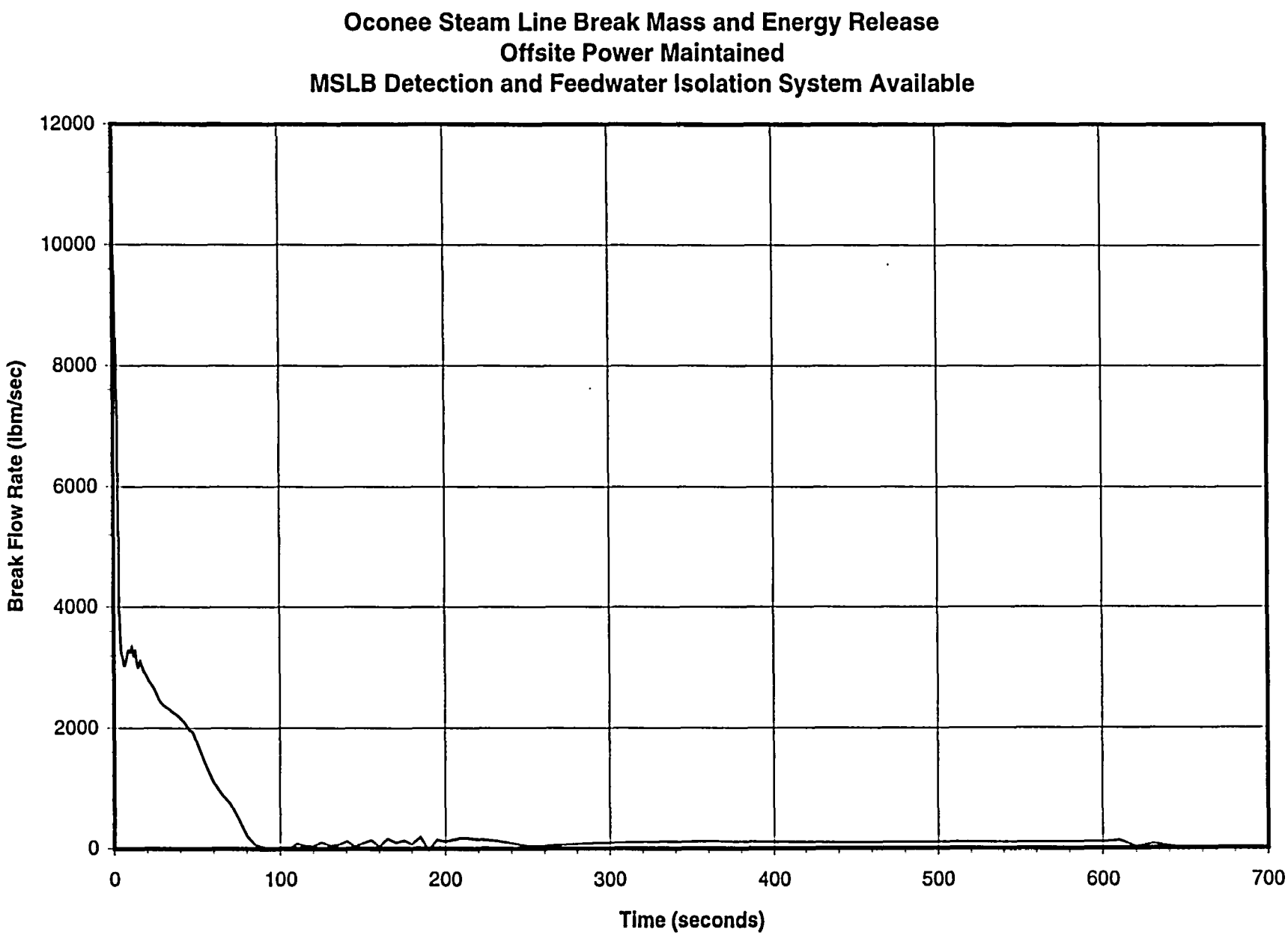

Oconee Steam Line Break Mass and Energy Release
Offsite Power Maintained
MSLB Detection and Feedwater Isolation System Available
Break Flow Rate (lbm/sec)
12000
10000
8000
6000
4000
2000
0
0
100
200
300
400
500
600
700
Time (seconds)

Figure S1-4

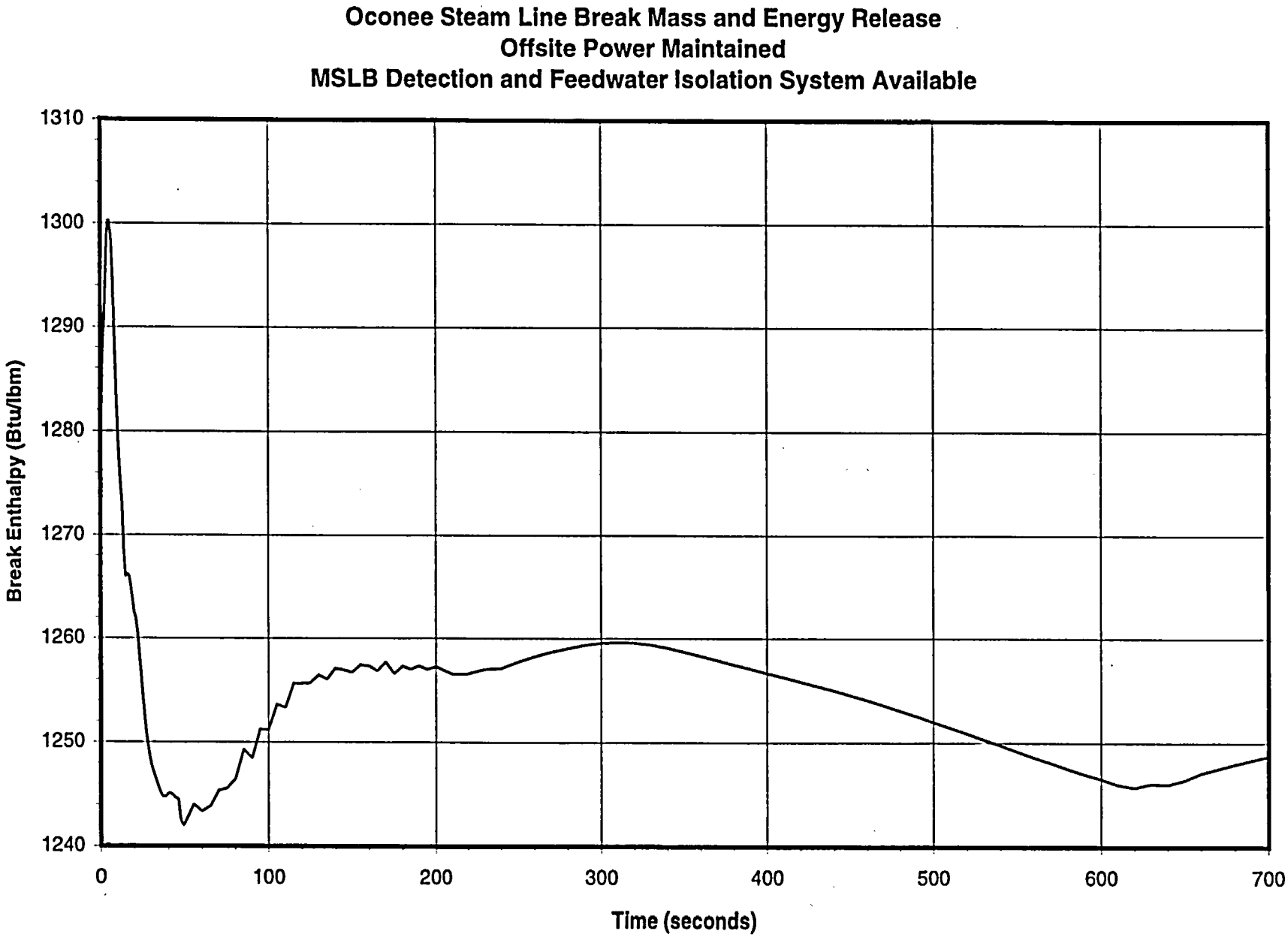

Oconee Steam Line Break Mass and Energy Release
Offsite Power Maintained
MSLB Detection and Feedwater Isolation System Available
Break Enthalpy (Btu/lbm)
1310
1300
1290
1280
1270
1260
1250
1240
0
100
200
300
400
500
600
700
Time (seconds)

Figure S1-5

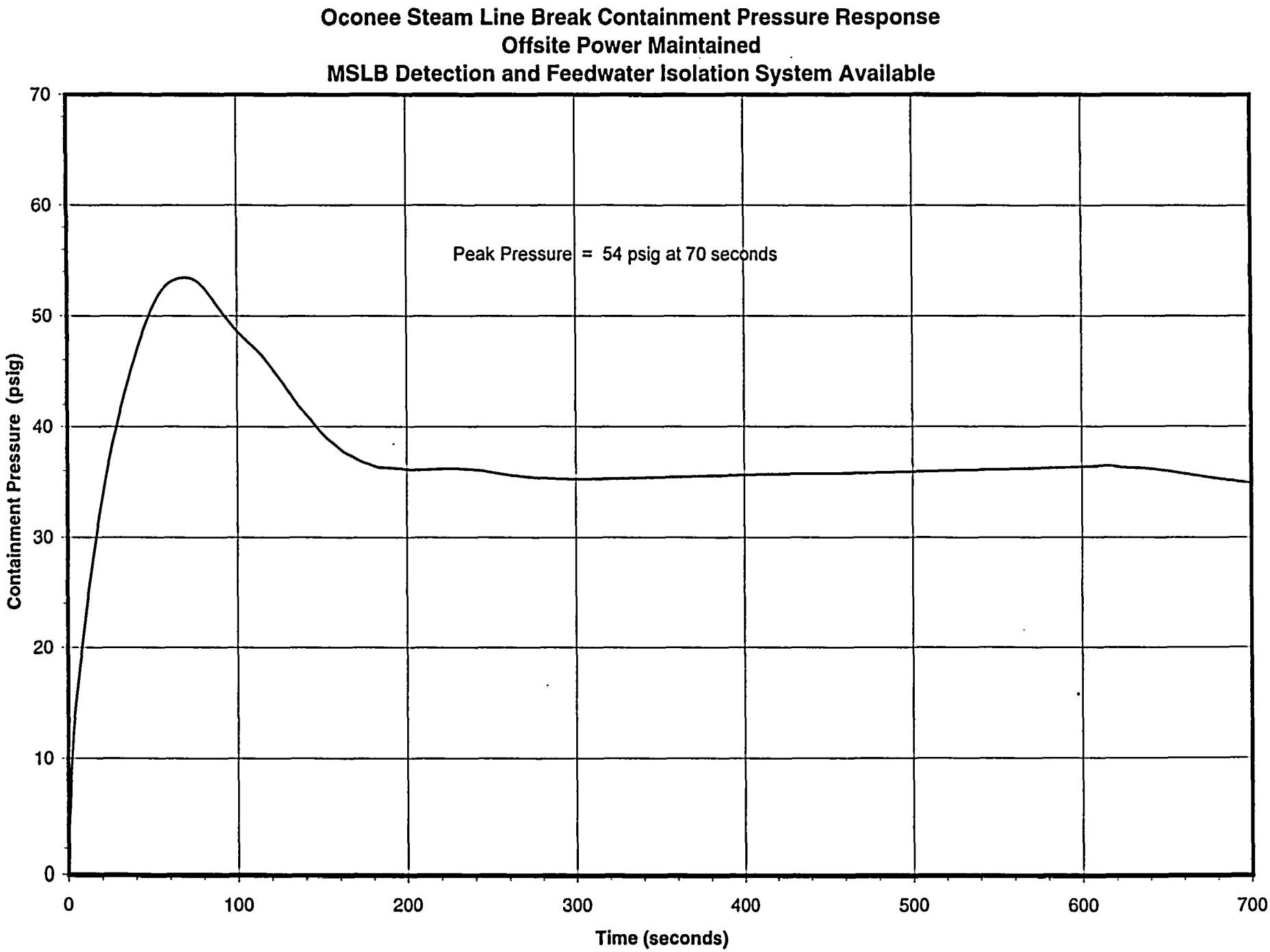

Oconee Steam Line Break Containment Pressure Response
Offsite Power Maintained
MSLB Detection and Feedwater Isolation System Available
Peak Pressure = 54 psig at 70 seconds
Containment Pressure (psig)
Time (seconds)
0
10
20
30
40
50
60
70
0
100
200
300
400
500
600
700

Figure S1-6

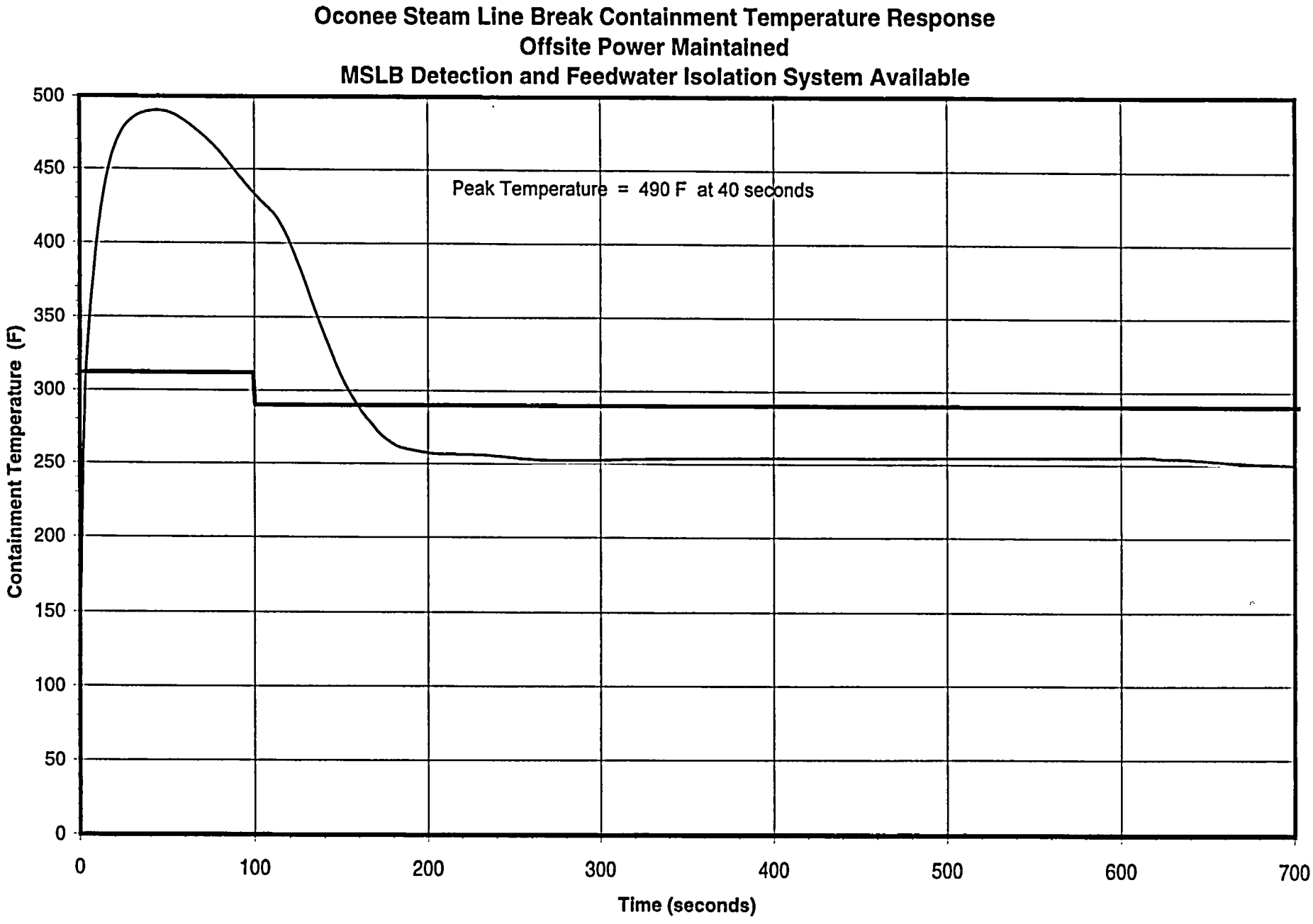

Oconee Steam Line Break Containment Temperature Response
Offsite Power Maintained
MSLB Detection and Feedwater Isolation System Available
Peak Temperature = 490 F at 40 seconds
Containment Temperature (F)
500
450
400
350
300
250
200
150
100
50
0
0
100
200
300
400
500
600
700
Time (seconds)

Figure S1-7

Supplement 2 to DPC-NE-3003-A

FATHOMS CVTR Benchmarking Study

FATHOMS CVTR BENCHMARKING STUDY

DESCRIPTION

A comparison was made between data taken from several steam blowdown tests at the Carolinas Virginia Tube Reactor (CVTR) and FATHOMS code results. This data was particularly applicable because a steam release into a large-scale dry containment was performed.

The CVTR was a heavy water- moderated and -cooled pressure tube prototype reactor operated in the 1960s. The reactor was decomissioned after four years of sucessful operation. The containment structure housed the reactor and primary system, the steam generator, and various auxiliary systems and components. The testing project described below was a joint effort of the plant's operating utilities group and the Atomic Energy Division of Phillips Petroleum Company.

The CVTR containment building was physically divided into three regions, referred to as the operating, intermediate, and basement regions. The building internal diameter was 57 ft 11-1/2 in., with an inside height of about 114 ft. The free volume of the CVTR structure was about 227,000 ft^3. Concrete and steel structures were present in all three regions. The FATHOMS model of the CVTR containment building, as well as the blowdown data for the three tests simulated, is documented below.

Five steam tests were performed during the CVTR test program. The first two were system checkout tests, and the final three were simulated DBA tests. Saturated steam was released to the building atmosphere at a rate of about 390,000 lbm/hr in all tests. It was released vertically above the operating floor. In the first of the three DBA tests, the building was allowed to depressurize through conduction of the steam energy into the concrete walls of the structure. In the other two tests, a pressure reduction spray system was utilized to depressurize the building, with differing spray flow rates in each test.

Pressure and temperature data was taken throughout the building in each test. Heat plugs imbedded in the concrete were also used to determine heat transfer rates into the passive building heat sinks.

CVTR FATHOMS CONTAINMENT MODEL

1. VOLUMES

	Volume designation	C.S.Area (ft^3)	Height (ft)	Baseline elev(ft)	wetted perim(ft)	Volume (ft^3)
Operating	(2,2)	2478.0	57.03	325.1	3477.0	141,320.
Intermediate	(1,3)	1711.8	32.0	-	212.7	54,778.
Basement	(1,2)	1711.8	18.0	275.0	387.4	30,812.

The general arrangement of the containment may be seen in Reference S2-1, Figure 1. The elevations of these volumes are taken from this figure. The intermediate and basement regions are actually one volume, arbitrarily divided at elevation 293'. The volume of each region is listed in Reference S2-1, pg. 89 (The volume of the operating region is documented as 8 ft^3 lower due to a rounding error). The actual heights of the intermediate and basement regions are used as dimensions for the calculational volumes, with the cross-sectional area of 1711.8 ft^2 calculated from volume/height. The actual height of the operating region is not used because the region is topped with a dome, and the c.s. area of this dome space is not constant. The c.s. area used for the operating region volume is calculated from the total volume divided by the area of the operating floor. The wetted perimeters of each volume are calculated as the sum of the perimeter of each volume plus the sum of the wetted perimeters of the heat structures in each volume, per Reference S2-2.

2. JUNCTIONS

Jctn.	End 1 loc.	Elev. (ft)	Ht. (ft)	End 2 loc.	Elev. (ft)	Ht. (ft)	Area (ft^2)	W.P. (ft)	Length (ft)	Loss Coeff.
1	(1,3)	320.	4.0	(2,2)	326.	10.	329.	500.	3.0	5.0
2	(2,2)	330.	5.0	-	-	-	-	-	-	-
3	(2,2)	370.	5.0	-	-	-	-	-	-	-

Junction 1 represents the flow path between the operating floor and intermediate regions. The elevations and spanning heights documented here for this junction are selected to establish a flow path for the steam-air mixture to interchange between these volumes.

Junctions 2 and 3 are established for the injection of the blowdown material and building spray, respectively, The areas, elevations and heights of these junctions are unimportant to the analysis; the junction only need be located within the appropriate volume. Dummy values are used for most of this input.

3. HEAT STRUCTURES

Heat Structure	Area (ft^2)	Thickness (ft)	Material	Location	Ref. 1, Table VI	W.P. (ft)
1	6370.	2.02083	.25" steel, 2' concrete	(2,2)	1	N/A
2	5370.	1.79167	.50" steel, 1.75' concrete	(2,2)	2	94.2
3	3880.	3.752583	concrete	(1,2)	3, 8, 10	215.6
4	5108.	3.433	concrete	(2,2)	5, 6, 7, 9, 11	133.0
5	1082.	0.05653	steel	(1,2)	12, 16, 18	60.1
6	3232.	0.15722	steel	(1,3)	13, 14, 15, 17, 20	101.0
7	500.	0.07833	steel	(2,2)	19	8.8
8	5824.	2.02083	same as 1	(1,3)	1	N/A
9	5824.	2.02083	same as 1	(1,2)	1	N/A
10	2478.	2.50	concrete	(2,2) and (1,3)	4	90.0*
11	30000.	0.3125	steel	(1,3)*	unlisted	500.0*

*Assumed.

The heat structures listed in Reference S2-1, Table VI have been re-arranged for the CVTR FATHOMS model. The cylindrical building shell has been partitioned among the three regions proportionally to the height of each region (heat structures 1,8,9). Heat structures 3-6 are lumped together from other structures, with an average thickness calculated for each structure. The wetted perimeter of each structure is calculated as the area of the structure divided by the height of the region that structure is located in, per Reference. 2.

An additional steel heat structure is included in all CVTR decks to include the smaller steel components not accounted for in the heat sinks given in Reference S2-1, Table VI. Only major steel compononets were counted in this table; large amounts of piping, ventilation ducts, gratings, tanks, and miscellaneous gear were neglected (Reference S2-1, pg. 94). This extra slab represents 30,000 ft^2 of 3/8" thick steel. No attempt was made to partition the steel among the three volumes in the FATHOMS model; it was placed entirely in region (1,3).

No two adjacent nodes in any heat structure have thickness ratios >2.0. Surface thicknesses for the concrete structure nodes are 0.025', and are slightly smaller for the steel structures. All structures are insulated on the second side except for the operating floor, which is exposed to region (2,2) on one side and (1,3) on the other.

Thermal properties data for concrete and steel, listed below, are taken from Reference S2-1, Table 5. No densities were given, so the nominal values for these densities are assumed.

Density (lbm/ft^3)	Thermal Cond. ($Btu/hr\text{-}ft^2\text{-}°F$)	Spec. heat ($Btu/lbm\text{-}°F$)	
490.	26.1	0.112	Carbon steel
150.	0.8	0.201	Concrete

The heat transfer correlations used for all heat structures are taken from the TAEH (thermal heat plug) data for all tests. The average of heat plugs 1 and 2 are used for Tests 3 and 5. Only data from heat plug 2 is given for Test 4 (Reference S2-1, Figs. 38-40).

Since the TAEH data ends shortly after the conclusion of the steam blowdown for all tests, some heat transfer coefficient must be assumed for the period after the blowdown. A constant heat transfer coefficient of 30 $Btu/hr\text{-}ft^2\text{-}°F$ is used in all these tests for this period. (See Results for discussion of the impact of this assumption.)

4. BOUNDARY CONDITIONS

Boundary conditions for all tests are given in Table S2-1. The mass and energy releases are taken from Reference S2-1, Figures 8-10 and Table 11. Start times for steam blowdown and average mass flow rates are selected to match the totals given on the referenced figures as closely as possible.

Test 3			Test 4			Test 5		
Time, sec	m, lbm/sec	h, Btu/lbm	Time, sec	m, lbm/sec	h, Btu/lbm	Time, sec	m, lbm/sec	h, Btu/lbm
0.0	0.0	1195.	0.0	40.0	1196	0.0	0.0	1197
4.0	55.56		10.0	100.0		10.0	106.0	
10.0	105.55		174.7	100.0		170.0	90.0	
60.0	100.0		177.0	33.33		173.1	0.0	
70.0	104.0		178.0	0.0				
100.0	101.0							
110.0	111.1							
160.0	105.55							
162.0	33.33							
166.4	0.0							

TABLE S2-1
CVTR Test Boundary Conditions

Avg. droplet size = 6.5e-4 ft (From ONS FATHOMS Containment base model)

Total mass = 16523 lbm 17352 lbm 16354 lbm
Total energy =19.745E6 Btu 20.753E6 Btu 19.576 Btu
(Cf. - Reference S2-1, Table 11)

Building spray:
(Reference S2-1, Figures 16-17)

Test 3	Test 4		Test 5	
	Spray flow:	290 gpm	Spray Flow:	500 gpm
-NO SPRAY-	Spray on:	210 sec	Spray on:	200 sec
	Spray off:	970 sec	Spray off:	900 sec
	Spray temp:	60°F	Spray temp:	60°F

Spray droplet size assumed: 2.3e-3 ft (700 μm). This is within the stated range of droplet sizes for the tests (Reference S2-1, pg. 5)

FATHOMS CVTR TEST COMPARISON - RESULTS

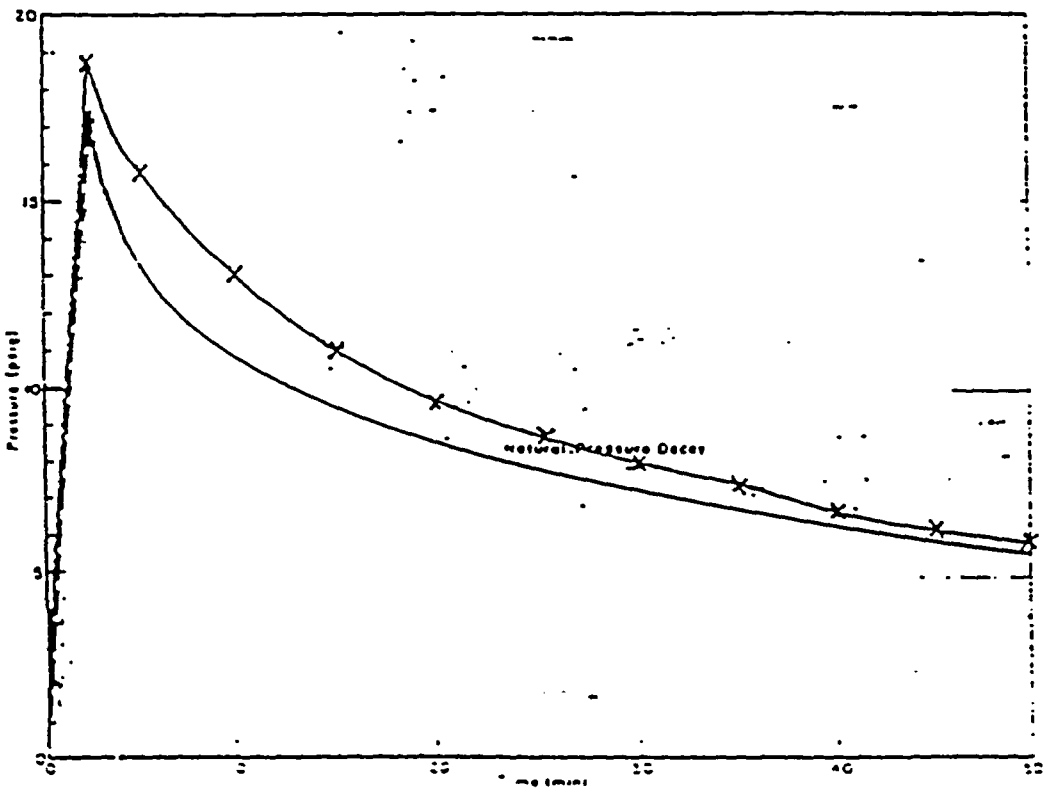
The results of the three CVTR test simulations are shown on Figures S2-1 to S2-3. The pressure response is predicted reasonably well for all three tests. The FATHOMS results show an over-prediction of the peak building pressure by between 1 - 2 psi in each test. The largest discrepancies between the simulation results and the data occur during the spray period, when the building pressure is decreasing. Heat transfer coefficients of 30 Btu/hr-ft^2-°F were used for all concrete and metal structures for all three tests during this portion of the transient. It is postulated that the steel heat structures may have been transferring heat to the atmosphere during this cooldown period at a much faster rate in the FATHOMS simulations than the actual tests. (No heat transfer coefficients were calculated in the CVTR analyses after the first three minutes or so. The value of 30 was assumed in the FATHOMS runs, as well as some of the original CVTR analyses, although the CVTR report states that the actual heat transfer coefficients were probably much lower during this phase, especially in the tests involving building spray.) The pressures at the conclusion of all three tests, however, match the data quite well; all heat structures in the building have nearly come to equilibrium with the atmosphere at this time.

The temperatures for CVTR Tests 3 and 5 are shown on Figure S2-4 with the corresponding FATHOMS data. No building spray was used in Test 3, while Test 5 had 500 gpm spray. This spray effect may be seen in the convergence of the operating region and basement temperatures for Test 5. The operating region temperatures (region (2,2)) compare very well for both tests. The basement temperature (region (1,2)) during the blowdown appears to be slightly underestimated by the FATHOMS simulations in both cases. The increase in basement temperature due to heated spray flow is accurately predicted, however, as are the long-term temperature profiles in both tests for both regions.

CONCLUSION

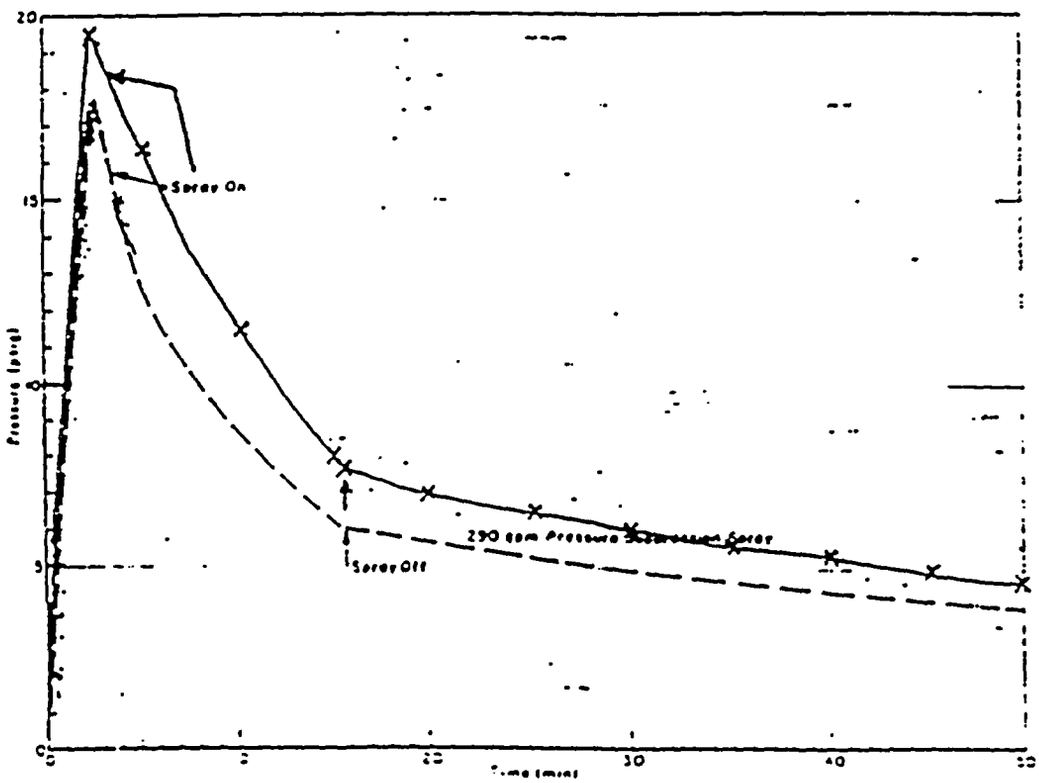
Figures S2-1 to S2-3 show that the CVTR data was predicted reasonably well by the FATHOMS code, with the calculated peak pressures slightly higher than the data in all tests. These conservative differences in peak pressures are reached despite a fairly substantial amount of steel placed in the CVTR FATHOMS model in an attempt to represent the small steel structures neglected in the CVTR documentation. Figure S2-4 shows that the CVTR temperatures were predicted fairly well in all regions of the building.

It is therefore concluded that FATHOMS can be used to determine the containment response to a steam line break accident with the assurance that conservative results can be obtained.



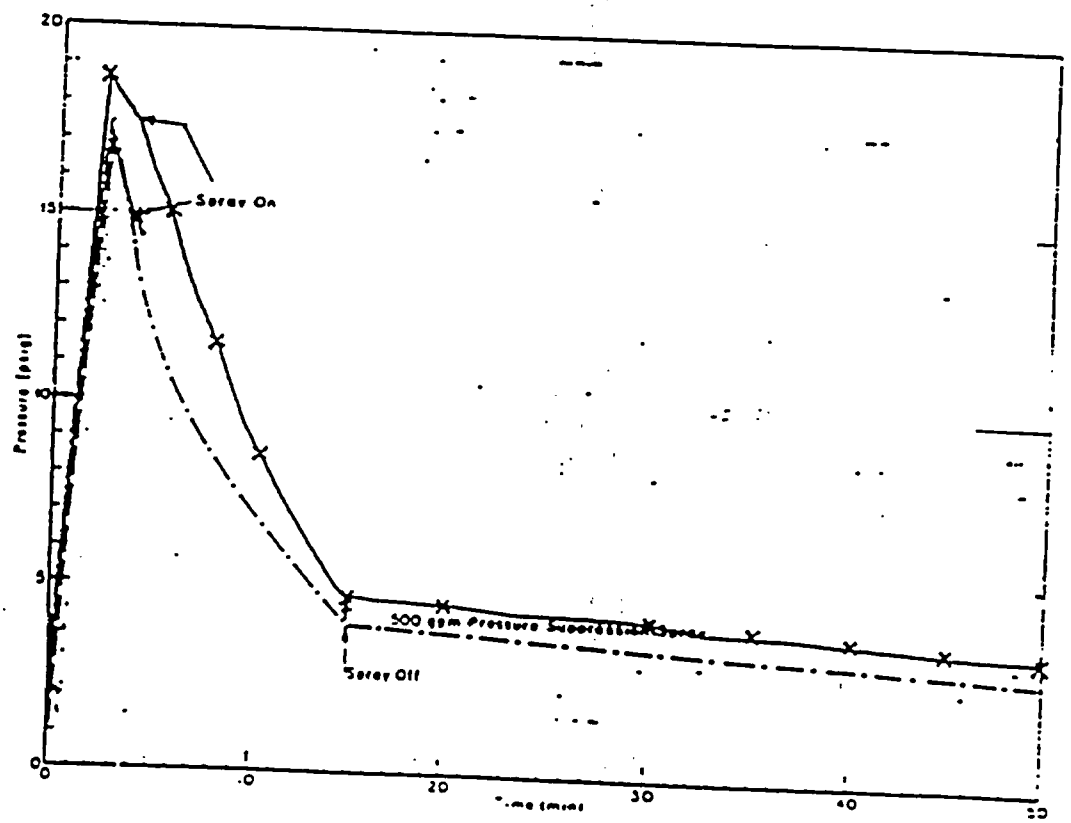
—— DATA —x— FATHOMS

Figure S2-1 - CVTR Test 3



—— DATA —×— FATHOMS

Figure S2-2 - CVTR Test 4



—— DATA —×— FATHOMS

Figure S2-3 - CVTR Test 5

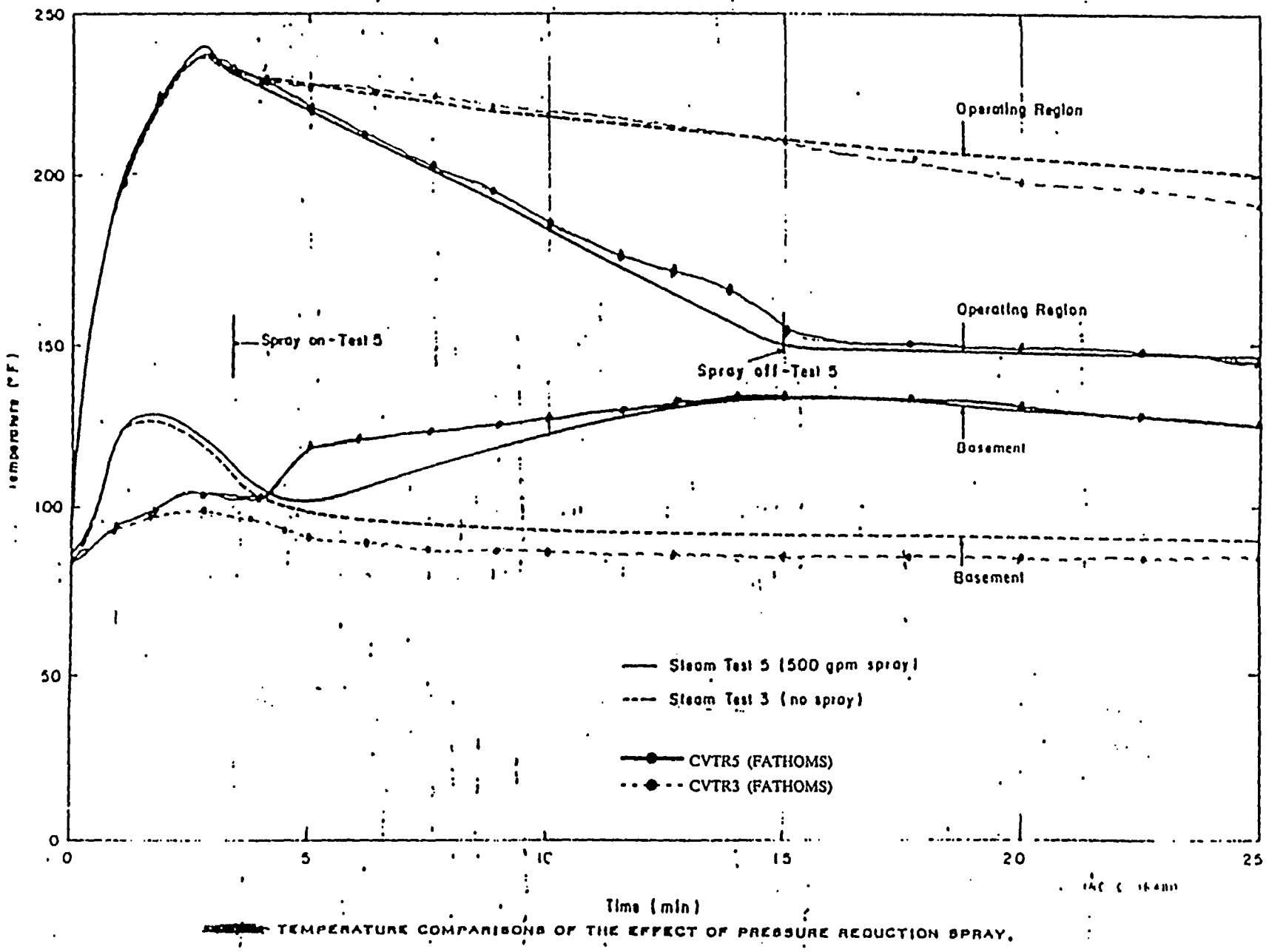


Figure S2-4 - FATHOMS CVTR Simulation vs. Data
Temperature Response

REFERENCES

S2-1. "Simulated DBA Tests of the Carolinas Virginia Tube Reactor Containment - Final Report", R.C. Schmitt, G.E. Bingham, J.A. Norberg, Idaho Nuclear Corp., Decenber 1970.

S2-2. FATHOMS / Containment Analysis Package User's Manual, Version 2.4, Numerical Applications, Inc., April 20, 1989.

APPENDIX A

METHODOLOGY REVISION FOR OCONEE REPLACEMENT STEAM GENERATORS

ROTSG Design Description

This appendix describes the Oconee RETRAN model revisions necessary for simulation of the steam line break mass and energy release analysis for the replacement steam generators (ROTSGs) being manufactured by Babcock & Wilcox Canada (BWC). The first pair of ROTSGs are scheduled for installation in Oconee Unit 1 in 2003. The other two Oconee units are scheduled for replacement outages in 2004. The design of the ROTSGs enables what is characterized as a "like-for-like" replacement due to the close similarity in the performance of the component. Due to the similarity of the ROTSGs with respect to the original OTSGs, the RETRAN modeling is also maintained very similar. The new RETRAN models for the ROTSGs are detailed in Revision 3 to Duke Power topical report DPC-NE-3003-PA, "Thermal-Hydraulic Transient Analysis Methodology" (Reference A-1). Additional new modeling details specific to the ROTSG steam line break mass and energy release analysis are included in this appendix. The significant design differences are as follows. Figure A-1 shows a comparison of the OTSG and ROTSG designs.

1) Flow-restricting orifices in the steam outlet nozzles
2) Inconel-690 tubes
3) 15,631 vs 15,531 tubes
4) Thinner pressure vessel / wider downcomer
5) Thinner tubesheets resulting in 3.625 inch longer heated tube length
6) 1.2% greater heat transfer area
7) More water in the steam generator.

Other than the ROTSG dimensional design data, Duke is also using steady-state thermal-hydraulic data provided by BWC as the reference data for the ROTSG initial conditions. These data included the pressure distribution, void fraction distribution, steam superheat profile, and water masses. The BWC simulations are three-dimensional, whereas RETRAN predictions are one-dimensional, so some approximation is necessary when comparing the results of the two codes.

RETRAN-3D Code

The Electric Power Research Institute's (EPRI) RETRAN-3D code (Reference A-2) is used for the Oconee RETRAN Model with ROTSGs. Previous modeling described in the body of this report with the original OTSGs used the EPRI RETRAN-02 code. RETRAN-3D, was approved

by the NRC in the SER dated January 25, 2001 (Reference A-3). The most significant change in the Oconee RETRAN methodology resulting from this RETRAN code version change is in the two-phase modeling of the secondary side of the steam generators. With RETRAN-02 the secondary side is modeled using the [

] All of the details of the transition to using RETRAN-3D instead of RETRAN-02 for Oconee modeling are presented. It is noted that the three-dimensional core model capability in RETRAN-3D is not included in this methodology, as well as some of the other new models. The conditions and limitations in the NRC's SER for RETRAN-3D are addressed in Appendix C.

ROTSG Nodalization

Details of RETRAN-3D Modeling

The selection of RETRAN-3D models, code options, and other input specification for the Oconee RETRAN Model with ROTSGs is presented and justified in Reference A-1, Appendix A. RETRAN-3D code modifications developed by Computer Simulation & Analysis, Inc, for Duke are also presented in detail in Reference A-1, Appendix A. The content of Reference A-1, Appendix A is not repeated here, but is incorporated in the methodology by reference. Additional methodology revisions specific to the steam line break mass and energy release analysis follow.

Revisions to Chapter 5 Methodology

Section 5.2.2 - Break Modeling
A spectrum of break sizes is analyzed in the Revision 1 methodology.

Technical Justification: The original methodology, although not excluding other than the double-ended maximum break size, did not investigate smaller break sizes. The application of the revised methodology covers a range of break sizes. This was determined to be necessary due to the ROTSG steam outlet nozzle flow orifices and the new Automatic Feedwater Isolation System (AFIS) station design modification.

Section 5.3 - Initial Conditions
Although initial conditions are considered inputs to the methodology rather than methodology, many of the initial conditions in Section 5.3 are not applicable to the ROTSG application. Changes to input values are not included.

Technical Justification: Changes to input values are not methodology changes and therefore do not need to be revised.

Section 5.4 Boundary Conditions - Main Feedwater System
This section is supplemented in Revision 1 by the Integrated Control System (ICS) and Automatic Feedwater Isolation System (AFIS) discussion below for additional modeling details regarding main feedwater control.

Section 5.4 Boundary Conditions - Emergency Feedwater System
This section is supplemented in Revision 1 by the Automatic Feedwater Isolation System (AFIS) discussion below for additional modeling details regarding emergency feedwater control.

Section 5.4 Boundary Conditions - Single Failure
This section is supplemented in Revision 1 to state that single failures are modeled in a consistent manner between the RETRAN-3D mass and energy release analysis and the FATHOMS or GOTHIC containment response analysis. For example, a single failure of a 4160V switchgear will have consistent results on HPI, EFW and RBS pumps and valves.

Technical Justification: Traditional transient and accident analyses assume a single failure in safety-related components or systems. This methodology clarification states that the single failure will be consistently applied to both the RETRAN and the FATHOMS or GOTHIC analysis. In other words, the consequences of the single failure will be realistic in terms of what components fail due to that specific single failure.

Section 5.4 Boundary Conditions - Reactor Protective System Trips
For the analysis of some smaller break sizes reactor trip on high Reactor Building pressure is credited in Revision 1. An uncertainty of 0.37 psi is added to the technical specification allowable value of 4.0 psig to ensure a late reactor trip. A conservatively long delay time of 0.4 seconds is assumed for this trip function. The time at which the RETRAN-3D analyses assume reactor trip on high Reactor Building pressure is dependent on break size and is determined by iteration with the FATHOMS or GOTHIC results.

Technical Justification: In the original methodology the focus was on double-ended maximum break sizes. In the application of the methodology to the ROTSGs a break spectrum was investigated. For some of the smaller break sizes the reactor trip on primary system trip functions such as variable low pressure-temperature (P/T) was delayed. A trip on high Reactor Building pressure would occur earlier than the P/T trip. Credit for the high Reactor Building pressure trip is included in the Revision 1 methodology. The FATHOMS or GOTHIC code is used to conservatively predict the timing of the reactor trip for each break size and set of analysis assumptions. A conservative high Reactor Building pressure trip setpoint including uncertainty, and a bounding trip delay time are included.

Section 5.4 Boundary Conditions - Integrated Control System (ICS)
The description of the modeling of the Integrated Control System (ICS) is expanded in Revision 1 to address the control of the Main Feedwater System (MFW) when the ICS is

initially in the manual control mode. The ICS switches to the automatic control mode upon reactor trip, and controls level to the appropriate post-trip level setpoint.

Technical Justification: Normally the ICS is in the automatic control mode. The predominant reason for the ICS to be in the manual control mode is if there has been some component failure in the ICS, or some other off-normal operational restriction exists. Repairs to the ICS to allow an expeditious return to the automatic control mode would be a high priority and would be completed in a timely manner. Therefore, operating with the ICS in manual is very infrequent and would be of short duration. With the ICS in manual the operations staff is highly focused on their responsibility for manual control. With the ICS in manual at the initiation of a transient, the MFW System will respond to the change in steam generator backpressure. If pressure decreases the MFW pump will shift along its head/flow curve and flow will increase. This is what will occur following a steam line break event. All steam line break events inside containment will cause a rapid pressurization of the Reactor Building and a reactor trip on high pressure at 4.0 psig plus uncertainty. In response to the reactor trip signal, the ICS will automatically switch from manual control to auto control, and will perform its normal post-trip design function of controlling steam generator level to setpoint. The setpoint is 25 inches on the startup level range with any reactor coolant pumps in operation, or 50% on the operating range level if all reactor coolant pumps are tripped. An allowance is made for the effect of a harsh containment environment on the steam generator level control setpoint. This ICS response is modeled in the steam line break analysis with the ICS initially in the manual control mode. A failure of the ICS to respond in this manner would constitute a failure in a control system. Failures in control systems are not assumed in conjunction with UFSAR transients and accidents in Oconee's licensing basis.

Section 5.4 Boundary Conditions - Liquid Carryout and Superheat Modeling

The original methodology for addressing the consequences of liquid carryout following a steam line break is revised in Revision 1. This modeling approach is necessary to conservatively correct the RETRAN-3D results that include liquid carryout since the carryout prediction cannot be validated due to an absence of data. Correcting for carryout is necessary to avoid potentially under-predicting the energy release. Computer Simulation & Applications, Inc., the RETRAN vendor, and Duke Power jointly developed the Revision 1 methodology that follows.

Card 08XXXY - Junction Data

Section 5.4 Boundary Conditions - Automatic Feedwater Isolation System (AFIS)

AFIS is designed to isolate MFW and EFW to a faulted steam generator during a steam line break event, thereby eliminating the need for prompt manual operator action to isolate EFW to a faulted steam generator. Detection of the steam line break event is accomplished by monitoring four pressure transmitters on each of the two main steam lines. When main steam

pressure falls below the low pressure setpoint (currently, 520 psig including a 30 psi uncertainty), the MFW pumps are tripped and the turbine-driven EFW pump is stopped. In addition, the MFW block valves, control valves, startup block valves, and startup control valves on the depressurized steam generator are closed. When a high rate of depressurization is sensed (currently, -3.3 psi/sec including a 0.3 psi/sec uncertainty) and pressure is below the low pressure setpoint, the motor-driven EFW pump(s) is tripped on the depressurized header(s). For break sizes where the rate of depressurization setpoint is not exceeded, manual operator action is credited at 10 minutes to isolate motor-driven EFW flow to the affected steam generator. Conservative modeling of the response of the components affected by an AFIS actuation is credited in the steam line break mass and energy release analyses.

Technical Justification: The above description of the AFIS design is accurately modeled in the RETRAN-3D ROTSG steam line break. This new plant modification is credited with automatically isolating MFW and EFW to a depressurizing steam generator. The AFIS design will replace the Main Steam Line Break Detection and Feedwater Isolation System design presented in Supplement 1.

References

A-1 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision 3, Duke Power Company, September 2004.

A-2 RETRAN-3D- A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI, NP-7450(A), Volumes 1-4, Revision 5, July 2001

A-3 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

Figure A-1

OTSG / ROTSG Comparison

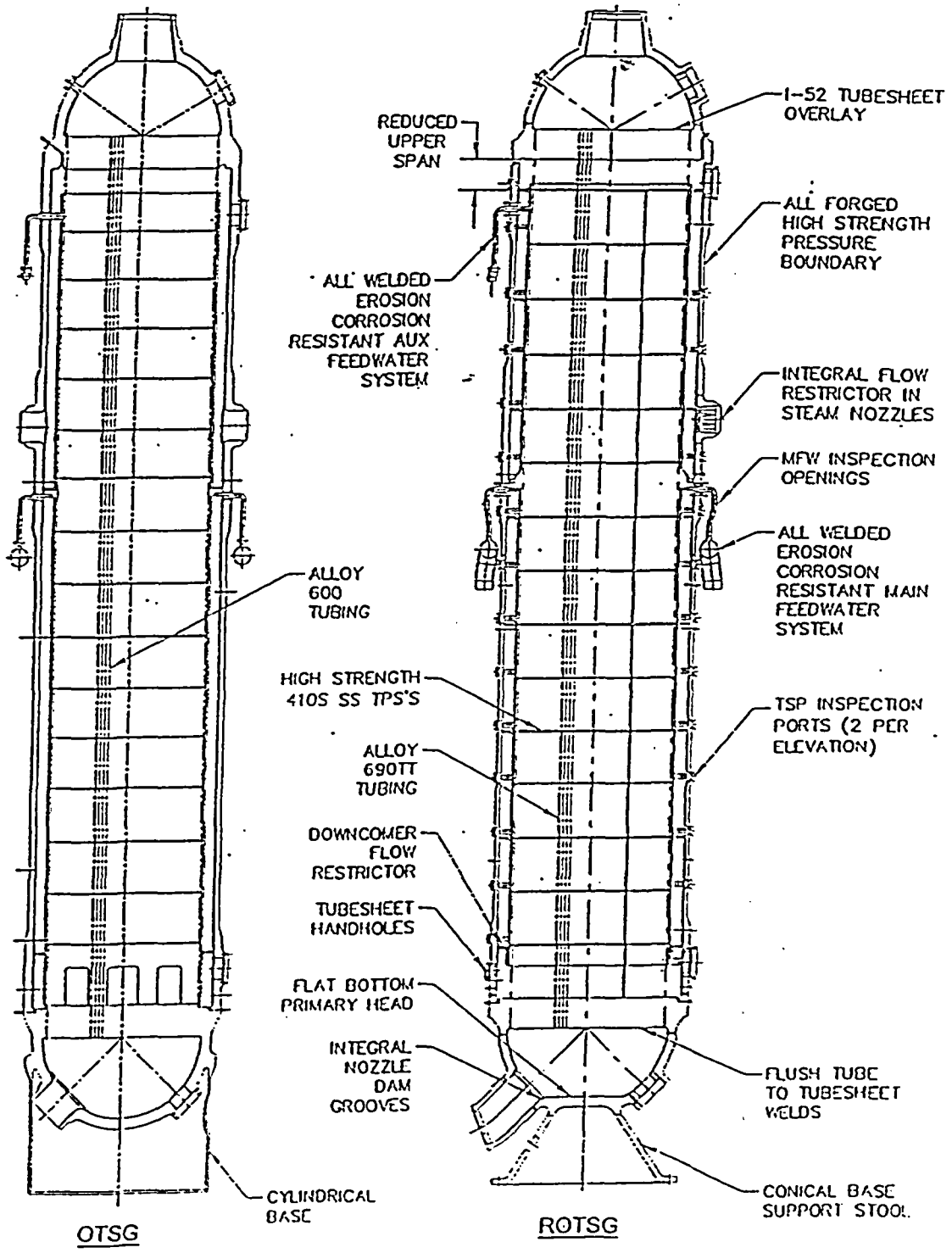


Figure A-2

ROTSG Nodalization

APPENDIX B

EVALUATION OF RETRAN-3D SER CONDITIONS AND LIMITATIONS FOR THE OCONEE RETRAN MODEL WITH ROTSGs

Purpose

This appendix evaluates the conditions and limitations in the RETRAN-3D SER (Reference B-1) for the application of RETRAN-3D to the Oconee Nuclear Station with replacement steam generators (ROTSGs). The results of this evaluation demonstrate that the use of the RETRAN-3D code for this application, as described in Appendix B, has been appropriately justified and is within the SER conditions and limitations. Therefore, the approval for use of RETRAN-3D as stated in the SER for the scope of approval specified in the SER can be credited. The Duke version of the RETRAN-3D code actually used is described first, including details on the custom code modifications that have been incorporated.

Description of the RETRAN-3D Code Version Used and Duke Code Modifications

RETRAN-3D MOD003.1DKE is the current Duke Power version of the standard Electric Power Research Institute (EPRI) RETRAN-3D MOD003.1 code, the NRC-approved code. This EPRI version includes the revisions agreed to during the SER review process. The Duke Power version, designated by the suffix "DKE", consists of two types of revisions to the standard EPRI version. The first type of revision is code error corrections. Duke periodically updates the code version in use to include error corrections obtained from Computer Simulation & Analysis, Inc. (CSA), the EPRI contractor for the RETRAN-3D code. The second type of revision is Duke custom code modifications purchased from CSA to address Duke-specific modeling needs. Each of these custom code changes are described in detail for NRC review and approval. The error corrections are not presented. All error corrections and code modifications are developed and controlled under CSA's Appendix B QA program. All Duke RETRAN-3D versions used for safety-related applications are certified and controlled under Duke's Appendix B software quality assurance program (Reference B-2). Any future modifications to the RETRAN-3D code versions used by Duke for safety-related applications that constitute significant model revisions or new models will be submitted for NRC review and approval. Code modifications that consist of error corrections or user features will be implemented under QA processes, but will not be submitted for NRC review.

Duke Code Modification #1 — Allow Access to the Condensation Heat Transfer Correlations With the Forced Convection Heat Transfer Map

Initialization of the ROTSGs using RETRAN-3D uses the forced convection heat transfer map. This is selected by setting variable IHTMAP on the 01000Y card to a value of zero. This standard forced convection option does not allow access to the condensation heat transfer correlations in RETRAN-3D. A code modification was implemented to allow access to the

condensation heat transfer correlations by setting IHTMAP to a value of 2. This option gives the forced heat transfer map, but allows condensation heat transfer to be modeled when appropriate for the local conditions present. This can be important in several transient conditions, such as when the primary water flowing through the steam generator tubes cools to below the secondary saturation temperature. In this situation an appropriate condensation heat transfer coefficient will be selected from the heat transfer correlation set. The technical justification for this code modification is that it allows a correct heat transfer correlation to be used for situations when condensation heat transfer occurs. This modification is functionally equivalent to a similar update made to RETRAN-02, which has been reviewed and approved for use by the NRC.

Duke Code Modification #2 — Allow the User to Specify the Dittus-Boelter Heat Transfer Correlation for a Specific Conductor

The RETRAN-3D heat transfer correlation package selects an appropriate heat transfer correlation for each conductor surface based on fluid conditions in adjacent volumes. Under some conditions it is useful to be able to select a specific correlation for a given conductor surface rather than using the code-selected correlation. This code modification allows the user to specify either the Dittus-Boelter liquid or vapor correlation for the left surface of a particular conductor. This will then override the code-selected correlation. Word IMCL on Card 15XXXY is set to a value of 41 to specify the liquid correlation, and to a value of 48 to specify the vapor correlation at the left surface of a conductor. The need for this modeling capability arose during analyses of the ROTSG upper shell heat transfer following a steam line break. Due to water carryout into the steam outlet annulus (the volume adjacent to the upper shell), the heat transfer was potentially too high for an analysis in which less heat transfer was conservative. This code modification allowed specifying use of the Dittus-Boelter correlation for vapor to conservatively model the heat transfer to the shell conductor. The code modification also allows specification of the Dittus-Boelter correlation for liquid as another modeling option. The technical justification for this code modification is that a capability to specify a heat transfer correlation for specific applications is appropriate.

Duke Code Modification #3

Duke Code Modification #4

Duke Code Modification #5

Evaluation of RETRAN-3D SER Conditions and Limitations

1. Multidimensional neutronic space-time effects cannot be simulated as the maximum number of dimensions is one. Conservative usage has to be demonstrated.

Staff Position: RETRAN-3D has been modified to include a 3-dimensional nodal kinetics model based on the analytic nodalization method similar to accepted codes. The code has been assessed by calculation of the response of the SPERT prompt-critical tests and has been confirmed by the staff by comparisons with calculations performed with the NESTLE and TORT codes. The staff concludes that the code can adequately predict the response to prompt-critical events such as the PWR rod ejection accident and the BWR rod drop accident. If void generation occurs from an initially un-voided case, the user will have to justify crediting this negative feedback in the analysis.

The code was used by a participant in the Nuclear Energy Agency's International Standard Problem calculation of a hypothetical main steam line break (MSLB) at the Three Mile Island Unit 1 plant. The results of the calculation comparison indicates that RETRAN-3D is comparable to any of the other participating codes.

RETRAN-3D is approved for main steam line break analyses subject to the following conditions. Thermal-hydraulic effects can have a large impact on the cross section evaluation and thus on the resulting power distribution and magnitude. Therefore, the licensee must justify the primary side nodalization for mixing in the vessel and core. The licensee must also evaluate the uncertainties in the modeling.

Duke Position: The RETRAN-3D three-dimensional kinetics model is not used.

2. There is no source term in the neutronics and the maximum number of energy groups is two. The space-time options assume an initially critical system. Initial conditions with zero fission power cannot be simulated by the kinetics. The neutronic models should not be started from subcritical or with zero fission power without further justification.

Staff Position: The basic models in RETPAN-3D are unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not analyze from subcritical or zero fission power initial conditions.

3. A boron transport model is unavailable. User input models will have to be reviewed on an individual basis.

Staff Position: As noted previously in this report, boron transport is handled as a "contaminant" by the "general transport model." This model uses first order accurate upwind difference scheme with an implicit temporal differencing. This approach is well known for being highly diffusive, especially if the Courant limit is exceeded. Since RETRAN-3D has the same model as RETRAN-02 MODOO3 and subsequent versions that have been approved for use, the RETRAN-3D model is also approved with the caveat that the potential to produce misleading results with this scheme necessitates careful review of the results for any case where boron transport/dilution is important.

Duke Position: Previously approved in RETRAN-02 and for Duke applications using RETRAN-02.

4. Moving control rod banks are assumed to travel together The BWR plant qualification work shows that this is an acceptable approximation.

Staff Position: The control bank limitation is applied only to the one-dimensional kinetics model. The staff agrees that the 3-dimensional kinetics model need not be restricted in this way.

Duke Position: Resolved per the Staff Position

5. The metal-water heat generation model is for slab geometry The reaction rate is therefore under-predicted for cylindrical cladding. Justification will have to be provided for specific analyses.

Staff Position: The basic models in RETRAN-3D are unchanged and, therefore, this condition of use applies. However, since RETRAN-3D is not being reviewed for loss-of-coolant accident analysis, where core uncovery and heatup are significant, this condition does not occur in the transients for which application of RETRAN-3D has been reviewed.

Duke Position: Duke does not use the metal-water heat generation model.

6. Equilibrium thermodynamics is assumed for the thermal-hydraulics field equations although there are nonequilibrium models for the pressurizer and the subcooled boiling region.

Staff Position: The RETRAN-3D five equation model permits thermal-hydraulic nonequilibrium between the liquid and vapor phases. While it allows subcooled liquid and saturated steam to be concurrently present, it does not account for subcooled liquid and superheated vapor being concurrently present. Use of the code is not approved for LOCA. Also, the user must be aware of this limitation and avoid conditions which will place subcooled liquid and superheated vapor in contact.

Duke Position: Duke does not use the RETRAN-3D five equation model.

7. While the vector momentum model allows the simulation of some vector momentum flux effects in complex geometry the thermal-hydraulics are basically one-dimensional.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this comment still applies.

Duke Position: As described in Appendix B, Duke is proposing to use the vector momentum option with junction angle input for certain junctions where the momentum flux terms are considered to be potentially important. It is acknowledged that the thermal-hydraulics are basically one-dimensional. Duke's use of this model is based on vendor recommendations.

8. Further justification is required for the use of the homogeneous slip options with BWRs.

Staff Position: RETRAN-3D has five slip equation options for the user to choose from, three of which are retained from RETRAN-02 for compatibility. The recommended model options are based on the Chexal-Lellouche drift flux correlation. The first is the algebraic slip model, which

is approved for use with BWR bundle geometry as given in condition (9). The second is a form of the dynamic slip model that uses the Chexal-Lellouche drift flux correlation to evaluate the interfacial friction approved in condition (10). The user must justify the use of any other slip options.

Duke Position: Duke is not modeling BWRs.

9. The drift flux correlation used was originally calibrated to BWR situations and the qualification work for both this option and for the dynamic slip option only cover BWRs. The drift flux option can be approved for BWR bundle geometry if the conditions of (16) are met.

Staff Position: The Chexal-Lellouche drift flux model has been used in comparisons with FRIGG-2 and FRIGG-4 void fraction data and is acceptable for use in BWR bundle geometry.

Duke Position: Duke is not modeling BWRs.

10. The profile effect on the interphase drag (among all the profile effects) is neglected in the dynamic slip option. Form loss is also neglected for the slip velocity. For the acceptability of these approximations refer to (17).

Staff Position: Form loss terms have been included in the RETRAN-3D dynamic slip model. The Taugl form of the dynamic slip equation also includes profile effects in the interphase drag model. These RETRAN-3D model improvements adequately address the concerns and the model is approved for use when the Chexal-Lellouche model is used to compute the interphase friction. Approval is subject to the conditions given in (16) for the Chexal-Lellouche drift flux correlation. Users must justify use of any other dynamic slip option.

Duke Position: Duke is not using the dynamic slip option.

11. Only one-dimensional heat conduction is modeled. The use of the optional gap linear thermal expansion model requires further justification.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use still applies.

Duke Position: Duke is not using the optional linear gap thermal expansion model.

12. Air is assumed to be an ideal gas with a constant specific heat representative of that at containment conditions. It is restricted to separated and single-phase vapor volumes. There are no other noncondensables.

Staff Position: RETRAN-3D has been extended to include a general noncondensable gas capability which resolves the original concern. However, the noncondensable gas flow model is approved for use subject to the following restriction.

As noted in Section 111.3.0 of the RETRAN-3D Theory Manual (Reference 4), none of the models available for calculating critical flow are appropriate when noncondensable gases are present. Consequently, the code automatically bypasses the critical flow model when noncondensable gases are present in a junction. Users must confirm that noncondensable flows do not exceed appropriate critical flow values or justify use of values that may exceed critical flow values.

Duke Position: Duke is not using the noncondensable gas flow model.

13. The use of the water properties polynomials should be restricted to the subcritical region. Further justification is required for other regions.

Staff Position: For enthalpies less than approximately 820 Btu/lbm, the difference between the ASME and RETRAN-3D curve fit values of the specific volume range from less than 0.2 percent to approximately 1.3 percent for pressures ranging from 0.1 to 6,000 psia. Further, for enthalpies greater than 820 Btu/lbm and pressures greater than 4200 psia, the differences in specific volume are also less than 1.0 percent. RETRAN-3D is approved for use with PWR ATWS analyses where the peak pressure resides in the regions described above.

For enthalpies greater than 820 Btu/lbm and pressures between 3200 and 4200 psia, the differences in specific volume increase as the enthalpy increases and the pressure decreases. The maximum error of approximately 3.8 percent occurs at the critical point. PWR ATWS analysis using RETRAN-3D in this region will require additional justification that the difference in specific volume does not adversely affect the calculation of the peak pressure.

Duke Position: Duke will address the above condition if an application encounters conditions in the region of concern.

14. A number of regime-dependent minimum and maximum heat fluxes are hardwired. The use of the heat transfer correlations should be restricted to situations where the pre-CHF heat transfer or single-phase heat transfer dominates.

Staff Position: RETRAN-3D contains both the "forced convection option" contained in RETRAN-02 which is the basis for this restriction, and a second option referred to as the "combination heat transfer map." If the first option is chosen, the "forced convection option," approval is granted only for use in pre-CHF and single-phase heat transfer regimes. If the second option is chosen, the "combination heat transfer map," then there are no discontinuities between successive heat transfer regimes and the appropriate heat transfer value should result. Therefore, the combination heat transfer option is approved for use.

Duke Position: Section 2.2.7.5 of Duke Power topical report DPC-NE-3000-PA, "Thermal-Hydraulic Transient Analysis Methodology,"[

]Note that Duke has implemented, and the NRC has approved, a RETRAN-02 code modification to enable accessing the condensation heat transfer

correlations with the forced convection heat transfer map. A similar code modification has been implemented for RETRAN-3D.

15. The Bennett flow map should only be used for vertical flow within the conditions of the data base and the Beattie two-phase multiplier option requires qualification work

Staff Position: The Beattie two-phase multiplier has been removed from RETRAN-3D. The Govier horizontal flow map has been added to supplement the Bennett map for vertical flow and is acceptable.

Duke Position: Resolved per the Staff Position.

16. No separate effects comparison have been presented for the algebraic slip option and it would be prudent to request comparisons with the FRIGG tests before the approval of the algebraic slip option.

Staff Position: The algebraic slip option has been modified to include the Chexal-Lellouche drift flux model. Use of the Chexal-Lellouche drift flux model for BWR and PWR applications within the range of conditions covered by the steam-water database used to develop and validate the model is approved. The model has been qualified with data from a number of steady-state and two-component tests. While the small dimensions of the fuel assembly are covered, as noted previously in this safety evaluation, the data for large pipe diameters, such as reactor coolant system pipes, are not extensive and use of the Chexal-Lellouche model will need justification. Assessment work indicates that the model tends to under-predict the void profile in the range of 12 to 17 MPa. In addition, the accuracy of the model in the range of 7.5 to 10 Mpa, which covers BWR ATWS conditions, has not been fully demonstrated. Results of analyses using the model in these ranges must be carefully reviewed.

The Chexal-Lellouche correlation cannot be used in situations where CCFL is important unless validation for appropriate geometry and expected flow conditions is provided.

Duke Position: [
]This use is consistent with the Staff Position.

17. While FRIGG test comparisons have been presented for the dynamic slip option the issues concerning the Schrock-Grossman round tube data comparisons should be resolved before the dynamic slip option is approved. Plant comparisons using the option should also be required.

Staff Position: Assessment analyses (Reference 4), have shown that "the issues concerning the Schrock-Grossman round tube data comparisons" (actually the Bennett round tube data) are due to early prediction of CHF, which is nearly independent of the slip model used. Since the issue raised in the limitation is not related to the dynamic slip model, the limitation is considered to be resolved. The dynamic slip model is approved for use as given in condition (10).

Duke Position: Resolved per the Staff Position.

18. The nonequilibrium pressurizer model has no fluid boundary heat losses, cannot treat thermal stratification in the liquid region and assumes instantaneous spray effectiveness and a constant rainout velocity. A constant UA is used and flow detail within the component cannot be simulated. There will be a numerical drift in energy due to the inconsistency between the two-region and the mixture energy equations but it should be small. No comparisons were presented involving a full or empty pressurizer. Specific application of this model should justify the lack of fluid boundary heat transfer on a conservative basis.

Staff Position: The concern raised in this limitation of use is partially resolved in RETRAN-3D. Wall heat transfer can be included in the RETRAN-3D pressurizer model. Including wall heat transfer resolves this concern.

While the model does not directly account for thermal stratification, its effects can be included by use of normal nodes below the pressurizer volume. The user will have to justify the lack of thermal stratification or the use of normal nodes below the pressurizer should there be an indication that it would be important in the analysis.

The mixture and two-region energy equations are consistent for the implicit solution method where the mixture energy equation is used with the vapor-region energy equation. This eliminates inconsistency between the two-region and mixture energy equations and the concern regarding a potential drift in the region energies.

The staff notes that when a pressurizer fills or drains, a single region exists for which the normal pressure equation of state is used. Lack of numerical discontinuities in validation analyses of filling and draining pressurizers indicates that the model is functioning properly. It is the responsibility of the code user to justify any numerical discontinuity in the pressurizer during a filling or draining event.

The pressurizer model has options that require user-supplied parameters. Users must provide justification for these model parameters.

Duke Position: Duke is not proposing any changes in modeling the pressurizer with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA. The good modeling practices in the Staff Position are noted.

19. The non-mechanistic separator model assumes quasi-statics (time constant approximately few tenths of seconds) and uses GE BWR6 carryover/carryunder curves for default values. Use of default curves has to be justified for specific applications. As with the pressurizer a constant L/A is used. The treatment in the off normal flow quadrant is limited and those quadrants should be avoided. Attenuation of pressure waves at low flow/low quality conditions are not simulated well. Specific applications to BWR pressurization transients under those conditions should be justified.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not use the separator model.

20. The centrifugal pump head is divided equally between the two junctions of the pump volume. Bingham pump and Westinghouse pump data are used for the default single-phase homologous curves. The SEMISCALE MOD-1 pump and Westinghouse Canada data are for the degradation multiplier approach in the two-phase regime. Use of the default curves has to be justified for specific applications. Pump simulation should be restricted to single-phase conditions.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: As noted in the Staff Position, the basic model in RETRAN-3D is unchanged from RETRAN-02. Duke is not proposing any changes in modeling the reactor coolant pumps with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA, Section 2.2.6.2.

21. The jet pump model should be restricted to the forward flow quadrant as the treatment in the other quadrants is conceptually not well founded. Specific modellng of the pump in terms of volumes and junctions is at the user's discretion and should therefore be reviewed with the specific application.

Staff Position: Subsequent revisions of RETRAN-02 addressed this limitation. Since RETRAN-3D has the same model as RETRAN-02 MODOO3, and subsequent versions, their acceptance applies to RETRAN-3D.

Duke Position: Duke does not model BWR jet pumps.

22. The non-mechanistic turbine model assumes symmetrical reaction staging, maximum stage efficiency at design conditions, a constant UA and a pressure behavior dictated by a constant loss coefficient It should only be used for quasi-static conditions and in the normal operating quadrant.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not use the turbine model.

23. The subcooled void model is a nonmechanistic profile fit using a modification of EPRI recommendations for the bubble departure point It is used only for the void reactivity computation and has no direct effect on the thermal-hydraulics. Comparisons have only been presented for BWR situations. The model should be restricted to the conditions of the qualification data base. Sensitivity studies should be requested for specific applications. The profile blending algorithm used will be reviewed when submitted as part of the new manual (MODOO3) modifications.

Staff Position: The profile blending algorithm approved for RETRAN-02 MODOO3 is used in RETRAN-3D therefore this condition has been satisfied.

Duke Position: Resolved per the Staff Position.

24. The bubble rise model assumes a linear void profile, a constant rise velocity (but adjustable through the control system), a constant UA, thermodynamic equilibrium, and makes no attempt to mitigate layering effects. The bubble mass equation assumes zero junction slip which is contrary to the dynamic and algebraic slip model. The model has limited application and each application must be separately justified.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies. However, the layering effects encountered in RETRAN-02 can be eliminated using the RETRAN-3D stack model. This partially resolves the concern by resolving the layering limitation through use of the stack model.

Duke Position: As noted in the Staff Position, the basic model in RETRAN-3D is unchanged from RETRAN-02. Duke is not proposing any changes in using the bubble rise model with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA, Section 2.2.6.4. Duke does not currently stack bubble rise volumes, but if future modeling does, the stack model will be used.

25. The transport delay model should be restricted to situations with a dominant flow direction.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies. The appropriate application of the model is for one-dimensional flow. The user will have to justify use of this option in the absence of a dominant flow direction.

Duke Position: As noted in the Staff Position, the basic model in RETRAN-3D is unchanged from RETRAN-02. Duke is not proposing any changes in using the transport delay model with RETRAN-3D relative to the previous NRC-approved RETRAN-02 modeling in DPC-NE-3000-PA, Section 2.2.7.4. The limitation with applying this model without a dominant flow direction is well known and is avoided.

26. The stand-alone auxiliary DNBR model is very approximate and is limited to solving a one-dimensional steady-state simplified HEM energy equation. It should be restricted to indicating trends.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke does not use the auxiliary DNBR model.

27. Phase separation and heat addition cannot be treated simultaneously in the enthalpy transport model For heat addition with multidirectional, multifunction volumes the enthalpy transport model should not be used without further justification. Approval of this model will require submittal of the new manual (MODOO3) modifications.

Staff Position: A number of the simplifying assumptions in the RETRAN-02 enthalpy transport model have been eliminated in RETRAN-3D which now allows multiple inlet and outlet flows

and eliminates the simplifying assumptions related to mass distribution and pressure change effects. This condition has been adequately addressed.

Duke Position: Resolved per the Staff Position

28. The local conditions heat transfer model assumes saturated fluid conditions, one-dimensional heat conduction and a linear void profile. If the heat transfer is from a local condition volume to another fluid volume, that fluid volume should be restricted to a non-separated volume. There is no qualification work for this model and its use will therefore require further justification.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position:

29. The initializer does not absolutely eliminate all ill-posed data and could have differences with the algorithm used for transient calculations. A null transient computation is recommended. A heat transfer surface area adjustment is made and biases are added to feedwater inlet enthalpies in order to justify steady-state heat balances. These adjustments should be reviewed on a specific application basis.

Staff Position: The over-specified condition is identified by the RETRAN-3D steady-state input checking, resolving the concern regarding ill-posed data. The user must still run null transients to ensure that unwanted control or trip actions are not affecting the transient solution.

RETRAN-3D has available a low power steady-state steam generator initialization option that eliminates the heat conductor area change used in the RETRAN-02 initialization scheme. When this option is used, no adjustments are made to the heat transfer area and this specific concern is resolved. However, either the pressure or temperature is adjusted on the secondary side. These adjustments should be reviewed by the user on a specific application basis. The low power steady-state initialization option is approved for use.

Duke Position: Resolved per the Staff Position.

30. Justification of the extrapolation of FRIGG data or other data to secondary-side conditions for PWRs should be provided. Transient analysis of the secondary side must be substantiated. For any transients in which two-phase flow is encountered in the primary all the two-phase flow models must be justified.

Staff Position: The Chexal-Lellouche correlation is approved for use with PWR applications as stated in conditions (10) and (16). The user must justify choosing any other two-phase flow correlation.

Duke Position: [ ]
Resolved per the Staff Position.

31. The pressurizer model requires model qualification work for the situations where the pressurizer either goes solid or completely empties.

Staff Position: The pressurizer model is approved for use with filling and draining events as given in condition (18).

Duke Position: Resolved per the Staff Position

32. Transients which involve three-dimensional space-time effects such as rod ejection transients would have to be justified on a conservative basis.

Staff Position: The 3-dimensional kinetics model, as noted in limitation 1 above, satisfies this limitation.

Duke Position: Resolved per the Staff Position. Duke is not using the three-dimensional model.

33. Transients from subcritical, such as those associated with reactivity anomalies should not be run.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke is not running any transients from subcritical.

34. Transients where boron injection is important, such as steamline break will require separate justification for the user-specified boron transport model.

Staff Position: The generalized transport model was added to RETRAN-3D to provide the capability to track materials such as boron. Specific application of the model to steam line break transients must be justified by the user. The model is approved for use as given in condition (3).

Duke Position: The generalized transport model is unchanged from RETRAN-02 relative to its use for modeling boron transport. The generalized transport model was approved in the SER for RETRAN-02 MOD005.0 dated November 1, 1991. Duke's use of this model for Oconee

emergency boron injection is documented in topical report DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," Revision 1, Section 15.3.1.1.3. This topical report was approved by the NRC with SERs dated October 1, 1998 and May 25, 1999. No further NRC review is necessary.

35. For transients where mixing and cross flow are important, the use of various cross flow loss coefficients has to be justified on a conservative basis.

Staff Position: The basic model in RETRAN-3D is unchanged and, therefore, this condition of use applies.

Duke Position: Duke models mixing and cross flow in the reactor vessel during transients and accidents in which loop asymmetry is important. Duke's modeling for Oconee is documented in topical report DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," Revision 1, Section 15.2.1.1. This topical report was approved by the NRC with SERs dated October 1, 1998 and May 25, 1999. No further NRC review is necessary.

36. ATWS events will require additional submittals.

Staff Position: RETRAN-3D is approved for PWR ATWS analyses as given in condition (13).

Duke Position: Resolved per the Staff Position.

37. For PWR transients where the pressurizer goes solid or completely drains, the pressurizer behavior will require comparison against real plant or appropriate experimental data.

Staff Position: The pressurizer model is approved for use with filling and draining events as noted in the discussion of conditions (18) and (31).

Duke Position: Resolved per the Staff Position

38. PWR transients, such as steam generator tube rupture, should not be analyzed for two-phase conditions beyond the point where significant voiding occurs on the primary side.

Staff Position: The use of slip models for PWR applications is approved for use as given in conditions (16) and (30).

Duke Position: In the Oconee steam generator tube rupture UFSAR Chapter 15 analysis significant voiding does not occur on the primary side. However, significant voiding can occur on the primary side for steam line break events. Duke's UFSAR Chapter 15 steam line break modeling for Oconee is documented in topical report DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," Revision 1, Chapter 15. This topical report was approved by the NRC with SERs dated October 1, 1998 and May 25, 1999. Duke's UFSAR Chapter 6 steam line break mass and energy release modeling for Oconee is documented in topical report DPC-NE-3003-PA, "Mass and Energy Release and Containment Response Methodology," Chapter 5. This topical report was approved by the NRC with SER dated March 15, 1995. Modeling of two-

[

] No further NRC review is necessary.

39. BWR transients were asymmetry leads to reverse jet pump flow such as the one recirculation pump trip, should be avoided.

Staff Position: As noted in the discussion of condition (21), this is resolved.

Duke Position: Duke does not model BWRs.

40. Organizations with NRC-approved RETRAN-02 methodologies can use the RETRAN-3D code in the RETRAN-02 mode without additional NRC approval, provided that none of the new RETRAN-3D models listed in the definition are used. Organizations with NRC-approved RETRAN-02 methodologies must obtain NRC approval prior to applying any of the new RETRAN-3D models listed above for UFSAR Chapter 15 licensing basis applications. Organizations without NRC-approved RETRAN-02 methodologies must obtain NRC approval for such methodologies or a specific application before applying the RETRAN-02 code or the RETRAN-3D code for UFSAR Chapter 15 licensing basis applications. Generic Letter 83-11 provides additional guidance in this area. Licensees who specifically reference RETRAN-02 in their technical specifications will have to request a Technical Specification change to use RETRAN-3D.

Duke Position: The submittal of DPC-NE-3000-P, Revision 3, includes the use of

The SER on p. 33 also states that use of the "new control blocks added to improve functionality" requires NRC review and approval. The new control blocks in RETRAN-3D are the following:

ABS - Absolute value
F2D - Two-dimensional interpolation
RAT - Rate
STF - Second-order transfer function

None of these new control block models have yet been incorporated into any of the Duke RETRAN models used for licensing basis applications. However, use of some of these new control block models in the future is likely to enhance and simplify applications. Since all of these new control blocks consist of well-founded arithmetic and mathematical formulas, similar to the control blocks included in RETRAN-02, it is not understood why NRC approval prior to their use is necessary. Duke requests NRC approval to use the new RETRAN-3D control blocks for future applications consistent with their formulation.

41. RETRAN may be used for BWR ATWS subject to the following restrictions: The licensee must validate the chosen void model over the range of pressure, channel inlet flow, and inlet subcooling encountered during the transient that are outside the range of conditions for which assessment is available. Furthermore, the licensee should validate the choice of steam separator model and evaluate its use relative to steam separator performance data relevant to the conditions

present during the ATWS simulation. The licensee must also evaluate the uncertainties in the modeling. See Condition (16) and the Staff Position for related information.

Duke Position: Duke does not model BWRs.

42. The RETRAN-3D five-equation, or nonequilibrium, model uses flow regime maps and flow pattern dependent heat transfer and interfacial area models to simulate the heat and mass transfer processes between phases. A licensee wishing to apply the five-equation model will have to justify its use outside areas of operation where assessment has been documented. This may include either separate effects or integral systems assessment that cover the range of conditions encountered by the application of interest. An assessment of the uncertainties must also be provided. The model is approved subject to these conditions.

Duke Position: Duke does not use the five-equation model for licensing basis applications.

43. Assessment performed in support of use of RETRAN-3D must also address consistency between the RETRAN-3D calculations and any auxiliary calculations that are a part of the overall methodology, such as, departure from nucleate boiling or critical power ratio.

Duke Position: Duke uses results from RETRAN-3D analyses for input to other codes to perform core power distribution analyses, detailed core thermal-hydraulic analysis of the departure from nucleate boiling phenomenon, fuel rod and pellet thermal and mechanical behavior analyses, and containment thermal and structural response to high-energy line breaks. The details of these other methodologies have been submitted and approved by the NRC as appropriate. Any revisions to these methodologies, including any changes due to the use of RETRAN-3D in place of RETRAN-02, will be submitted for NRC review prior to their use for licensing basis applications.

44. The staff concludes that the lack of a detailed RETRAN-3D specific user guideline document mandates a statement on the user's experience and qualification with the code when analyses are submitted in support of licensing actions. This statement is expected to be consistent with the guidance of Generic Letter 83-11.

Duke Position: It is noted that Volume 3 of the EPRI RETRAN-3D documentation set has been enhanced subsequent to the NRC SER to include a significant amount of user guidelines regarding modeling option selection, in particular for the new RETRAN-3D models and options. Revision 3 to DPC-NE-3000-P fully describes Duke's use of the RETRAN-3D code for simulating the Oconee Nuclear Station with replacement steam generators. This revision is submitted for NRC review with the intent of maintaining the documentation of the Duke RETRAN methodology current, along with the main purpose of obtaining NRC review and approval for the transition from RETRAN-02 to RETRAN-3D for Oconee. This topical report revision extends Duke's response to Generic Letter 83-11. Duke's current level of RETRAN user experience is 15 engineers with a total of 144 years of experience with RETRAN-02 and RETRAN-03/-3D.

45. Assessment of the RETRAN-3D code for the models not explicitly approved in this safety evaluation will be the responsibility of the licensee or applicant. In addition, application of the

RETRAN-02 or RETRAN-3D codes for best estimate analysis of UFSAR Chapter 15 licensing basis events may require additional code and model assessment, and an evaluation of uncertainties to assure accurate prediction of best estimate response. This condition is based on the absence, in the best estimate analysis approach, of the conservative assumptions in traditional UFSAR Chapter 15 licensing basis analyses. For each use of RETRAN-3D in a licensing calculation, it will be necessary for a valid approach to assessment to be submitted, which is expected to include a PIRT for each use of the code and the appropriate assessment cases and their results. The scope of the PIRT and validation/assessment will be commensurate with the complexity of the application.

Duke Position: Duke has previously received NRC review and approval for application of the RETRAN-02 code to the licensing basis applications for non-LOCA transients and accidents for the Oconee Nuclear Station. The three RETRAN-related topical reports and associated NRC SERs supporting Oconee are:

> DPC-NE-3000-PA, Revision 2, "Thermal-Hydraulic Transient Analysis Methodology", December 2000. SERs are dated 11/15/91 (Revision 0), 8/8/94 (Revision 1), and 12/27/95 (Revision 2)
>
> DPC-NE-3003-PA, "Mass and Energy Release and Containment Response Methodology", November 1997. SER is dated 3/15/95
>
> DPC-NE-3005-PA, Revision 1, "UFSAR Chapter 15 Transient Analysis Methodology, August 1999. SERs are dated 10/1/98 (Revision 0) and 5/25/99 (Revision 1)

These topical reports have all being revised and submitted for NRC review to address the Oconee replacement steam generators and use of the RETRAN-3D code for Oconee non-LOCA transient and accident analyses. Based on the close similarity of the replacement and original steam generators, the transient thermal-hydraulic behavior will be very similar. The only significant difference will be for the main steam line break analysis, in which the flow restricting orifices in the replacement steam generators steam outlet nozzles will effectively reduce the maximum break size and the blowdown rate.

That assessment

activity was presented and reviewed by the NRC during the review of earlier revisions of DPC-NE-3000. [ ] has been incorporated into the Duke version of the RETRAN-3D code as described in Revision 3 to DPC-NE-3000-P.

In summary, Duke has previously obtained NRC review for RETRAN-02 modeling of Oconee with the original steam generators. Substantial validation and assessments comparisons were associated with the previous revisions to DPC-NE-3000. The designs of the original and replacement steam generators are very similar, and the transient performance will be very similar except for the response to large steam line break accidents. Revision 3 describes the use of RETRAN-3D for modeling Oconee with replacement steam generators. [ ] Duke is not proposing to use this model for best-estimate licensing applications. The traditional conservative approach will continue to be used for licensing applications. A PIRT is not being submitted due to the previous NRC approval of the Duke RETRAN methodology topical reports, the limited scope of changes in the methodologies, the similarity of the designs of the new and replacement Oconee steam generators, the use of only one new RETRAN-3D model, and the assessment that has been performed to justify use of the one new model.

References

B-1 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

B-2 RETRAN-3D MOD003.1DKE, SDQA-30218-NGO, Duke Power, October 30, 2001

APPENDIX C

GOTHIC VERSION 7.0 CODE

C.1 Code Description

The GOTHIC Version 7.0 code (Reference C-1), developed by Numerical Applications, Inc. under contract to the Electric Power Research Institute (EPRI), is used for performing thermal-hydraulic analysis of nuclear power plant containment and auxiliary buildings. The GOTHIC 7.0 code was developed from the FATHOMS code (Reference C-2), which in turn was developed from the COBRA-NC code (Reference C-3). Due to the close similarities between GOTHIC 7.0 and FATHOMS, the content of this Appendix is very similar to Chapter 2.4. GOTHIC 7.0 solves the conservation equations for mass, energy, and momentum for multi-component, two-phase flow. These conservation equations are solved numerically on a finite-volume mesh made up of numerous computational cells. The code features a nodalization scheme in which lumped parameter, one-, two-, or three-dimensional analysis or any combination of these may be performed. Velocity fields are included for three phases:

- Vapor / non-condensable gas mixture
- Continuous liquid
- Liquid droplet

Interfacial heat, mass, and momentum transfer are modeled, and the effects of two-phase slip on pressure drop are also accounted for. Three temperature fields are included:

- Vapor / non-condensable gas mixture
- Continuous liquid
- Liquid droplet

These fields may be in thermal non-equilibrium within the same calculational volume. A total of eight non-condensable gases may be modeled, with mass transport equations solved for each component of the non-condensable gas mixture.

GOTHIC 7.0 includes finite-difference conduction models for passive thermal conductors. Concrete walls and structural steel within the Reactor Building are simulated with these models. These conductors may have the following geometrical shapes:

- Flat plate (wall)
- Cylindrical tube
- Solid rod

Heat is conducted through the thickness or radius of each of these conductor models. They may be thermally connected to fluid volumes on either surface. A variety of condensation heat transfer coefficient options are available for the surfaces of these thermal structures. Heaters and

coolers may also be modeled within calculational volumes in GOTHIC 7.0. Any volumetric heat sources or cooling equipment may be simulated with these models.

Each GOTHIC 7.0 analysis model consists of a network of volumes connected by flow paths. Typically, each volume represents a room or area within the Reactor Building. Any volume may be partitioned into a two- or three-dimensional mesh for which mass, energy, and momentum balance are maintained. This allows the fluid property distribution within a room to be calculated and flow patterns within a room to be predicted. The flow paths represent doorways, pipes or piping systems, vents, etc. Within each subdivided volume, the momentum equations for the vapor/gas mixture, droplets, and liquid are solved for all flow paths, with each phase having its own velocity. A variety of mechanical components to simulate various plant equipment can be modeled along these flow paths. These include:

- Valves
- Pumps
- Heat exchangers
- Spray nozzles
- Volumetric fans

Boundary conditions for the containment model are implemented using these flow paths. The user may specify mass and energy sources or sinks with these boundary conditions. The boundary conditions may be coupled in groups to model flowstreams which are sent to different calculational volumes. The user has the ability to model cooling and spray systems associated with containment safety features using these options.

GOTHIC 7.0 features the use of control variables that are not available in FATHOMS. The control variables provide enhanced capability and flexibility to input boundary conditions or forcing functions into the code. These are essentially user convenience features and do not constitute elements of the methodology.

The long-term mass and energy release boundary condition modeling in FATHOMS has been similarly installed in GOTHIC 7.0. This modeling approach, which enables interpolation of the BFLOW code results as an input boundary condition, are described in Sections 2.2.1.1, 2.4.1.2, and 6.3 of DPC-NE-3003. GOTHIC 7.0/DUKE is the Duke-specific version that includes these code revisions.

The GOTHIC 7.0 code was released by EPRI in 2001 to provide EPRI members with an efficient, state-of-the-art containment analysis code. The code had undergone extensive testing and benchmarking by Numerical Applications, Inc. prior to release.

C.2 Simulation Model

This section gives a brief description of the simulation models used for the GOTHIC 7.0 containment analyses. The nodalization diagram is the same as for the FATHOMS base model shown in Figure 2.4-1.

Nodalization

Large dry containments have traditionally been modeled with one large volume, with all pressures and temperatures being considered volume-averaged quantities. The Oconee GOTHIC

7.0 base model uses a similar modeling approach. As in the FATHOMS model, a separate sump volume is represented in the GOTHIC 7.0 base model. This consists of the lowest 8.6 feet of the containment space, and the upper 195 feet representing the remainder of the containment space. These two nodes are referred to here as the atmosphere region and the sump region. The sump temperature refers to the liquid temperature in the sump volume, and the vapor temperature refers to the temperature of the vapor phase of the atmosphere volume. There is a small vapor space in the sump region, but it is so small that it does not affect the results of applications of the model.

The containment free volume and passive heat sink data used for the base model include a 1% conservative allowance for uncertainty.

Initial Conditions

The initial pressure and temperature assumed in the Reactor Building varies with each analysis, depending on the purpose of the analysis. For the peak pressure analysis, it was determined that variations in the initial air mass had a large impact on the calculated peak pressure. A higher initial air mass leads to higher calculated peak pressures. The initial air mass can be increased by assuming a low initial building temperature The early time at which the peak pressure occurs (within 30 seconds for each limiting large break LOCA case) means that the increased initial air mass had a larger effect than the decreased heat transfer into the passive heat structures at a higher initial temperature. Therefore, conservatively low initial temperatures are assumed in the peak pressure analyses. In all long-term analyses, as well as the steam line break analysis, a high initial building temperature is assumed.

Likewise, the initial Reactor Building humidity is assumed to be at 0% to maximize the air mass in the peak pressure analyses, and at 100% in the long-term analyses. The initial building humidity has very little impact on the results of either type of analysis.

A high initial Reactor Building pressure is conservatively assumed for all peak pressure analyses. This maximizes the calculated peak building pressure. In the long-term analyses, this initial pressure has no significant impact on results.

Heat Structures

The concrete and steel heat structures present in the Oconee containment building are modeled using the heat slab models in GOTHIC 7.0. They are modeled with an insulated boundary condition for one surface. For the building walls, dome, and base, it would be many days before significant quantities of heat would conduct through the thickness of the structure, so this insulated boundary condition is used on each outer surface. The remaining structures, all internal to the Reactor Building, are exposed to the containment atmosphere on all sides, but the entire surface area is combined onto one surface for the GOTHIC 7.0 heat slab model. The zero heat fluxes on the second surface here represent a symmetry condition at the midplane of the structure. The slabs are modeled in full thickness, however, so that the total volume will be conserved. All structures are initially at the initial Reactor Building temperature.

All structures are modeled as walls, with one-dimensional heat transfer occurring in the direction perpendicular to the structure surface. The noding divisions are finer closer to the surface, where larger temperature gradients are expected.

The Uchida heat transfer correlation is used for heat transfer between the structure subcooled surfaces and the containment atmosphere. The standard heat transfer logic present in GOTHIC

7.0 is used for other heat transfer modeling. For the Reactor Building floor, which would be covered with a pool of water, a constant heat transfer coefficient of 20 Btu/hr-ft^2-^{o}F is assumed.

Reactor Building Cooling Units

The Reactor Building Cooling Units (RBCUs) are safety-grade fan coolers modeled using the standard cooler model in GOTHIC 7.0. Input parameters are chosen from design values for the coolers. The heat removal characteristics of the coolers varies with each analysis. The coolers are typically actuated at 9 psig Reactor Building pressure, plus a 300 second delay time. These setpoints include allowances for instrument uncertainties and all other associated delays.

Reactor Building Spray System

The Reactor Building Spray System (RBS) is modeled with a single junction for injection from the borated water storage tank (BWST), or with a set of junctions for sump recirculation mode. The RBS flow rate varies depending on the purpose of the analysis. Uncertainty in the RBS flow rate is included.

All spray is assumed to enter the containment atmosphere with an average droplet size of 700 μm. No spray efficiency parameter is necessary in the GOTHIC 7.0 flow models. The spray is typically actuated at 30 psig Reactor Building pressure, with a delay time of 92 seconds for the spray header fill time. These setpoints include allowances for instrument uncertainties and all other associated delays.

C.3 Code Validation

The GOTHIC 7.0 code authors, Numerical Applications, Inc., subjected the code to extensive benchmarking efforts throughout the process of refining and updating the code. The code has been compared against analytic solutions as well as experimental data (Reference C-1). This set of tests includes comparisons with data from the Battelle Frankfurt HDR containment experimental facility, the Hanford Engineering Development Laboratory (HEDL) facility, and many other tests. The validation effort included a comparison against a number of separate effect test, such as fan coolers, pump head vs. flow tests, jet breakup tests and other specially designed analytic problems to test specific components and GOTHIC 7.0 code capabilities.

Of special importance is the comparison by NAI of GOTHIC 7.0 predictions with the Carolina Virginia Tube Reactor (CVTR) results. This set of data is particularly appropriate for this purpose because it applies to the pressurization of a large, dry containment building similar to that of Oconee. These experiments studied the pressurization of the CVTR building due to a saturated steam blowdown, with passive heat structures and building spray aiding in the depressurization. A GOTHIC 7.0 model was developed for the CVTR containment building, and simulations of three steam blowdown tests were performed. The pressure and temperature responses for all three tests are predicted reasonably well.

The GOTHIC 7.0 code is an extension of the COBRA-NC code and retains all of the modeling capabilities of COBRA-NC. The COBRA-NC code was also benchmarked against analytic solutions and experimental data, including steam and steam/water blowdowns in the Battelle Frankfurt Model Containment, Containment Analysis Standard Problems 1, 2, and 3, blowdown tests in the HDR facility and hydrogen distribution tests in the Battelle Frankfurt Model Containment (Volume 7, Reference C-3). These verification results can be reasonably assumed to valid for the GOTHIC 7.0 code as well. All efforts have validated the code's ability to accurately predict pressure and temperature transients within containment structures.

3.4 Model Validation

After a GOTHIC 7.0 model of the Oconee containment was developed, efforts to compare the results obtained from that model were undertaken. Since no test data exists for the pressurization rate of the Oconee Reactor Building following a high-energy line break, an available means of validation of the GOTHIC 7.0 model for the Oconee design is a code-to-code comparison to the results of a FATHOMS LOCA analysis. The FATHOMS analyses was compared previously (Chapter 2.4) with the CONTEMPT predictions performed by Babcock and Wilcox (B&W) for the original Oconee FSAR. An analysis using mass and energy release data for a 14.1 ft^2 hot leg break, representing a double-ended guillotine break of the hot leg, is used. This is the bounding peak containment pressure transient for Oconee according to the existing FATHOMS licensing basis analyses in the UFSAR.

The Oconee GOTHIC 7.0 model, input data, and code option selection, as well as the boundary conditions, are identical to the corresponding FATHOMS analysis to the extent possible. This approach is intended to allow a code-to-code comparison.

The results of this comparison are shown in Figures C-1 and C-2. The containment pressure curves (Figure C-1) show an identical trend for both codes. The peak pressure calculated by GOTHIC 7.0 is approximately 1 psi lower than the FATHOMS result. The containment temperature profile (Figure C-2) is identical for both codes. These results show that the GOTHIC 7.0 code and the Oconee GOTHIC model provide essentially identical results when compared to the FATHOMS code and model in the DPC-NE-3003 methodology. Use of either code and model in the future is technically equivalent. The intent is to begin a transition from FATHOMS to GOTHIC 7.0 upon approval of DPC-NE-3003-P, Revision 1.

C.5 References

C-1 NAI 8907-02 Rev 13, "GOTHIC Containment Analysis Package – User Manual", Version 7.0, July 2001

C-2 CAP-Containment Analysis Package (FATHOMS 2.4), Numerical Applications, Inc., October 10, 1989

C-3 NUREG/CR 3262, "COBRA-NC: A Thermal Hydraulics Code for Transient Analysis of Nuclear Reactor Components", Volumes 1 to 7, Wheeler, et.al., Pacific Northwest Laboratory, April 1986.

GOTHIC 7.0 - FATHOMS Comparison For
Oconee Large Break LOCA (14.1 ft^2 break - S/G Inlet)

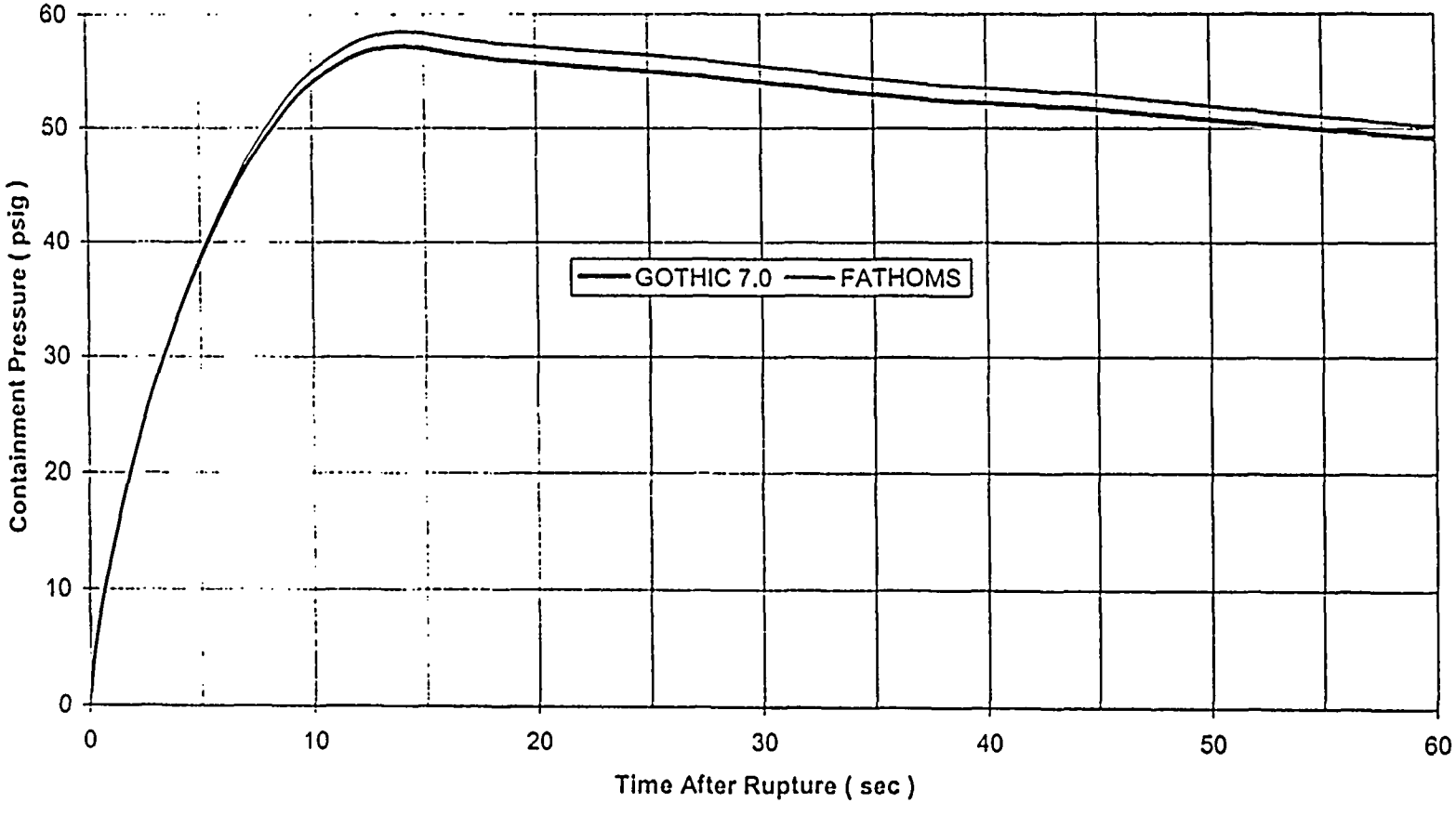


FIGURE C-1

GOTHIC 7.0 - FATHOMS Comparison For
Oconee Large Break LOCA (14.1 ft2 break - S/G Inlet)

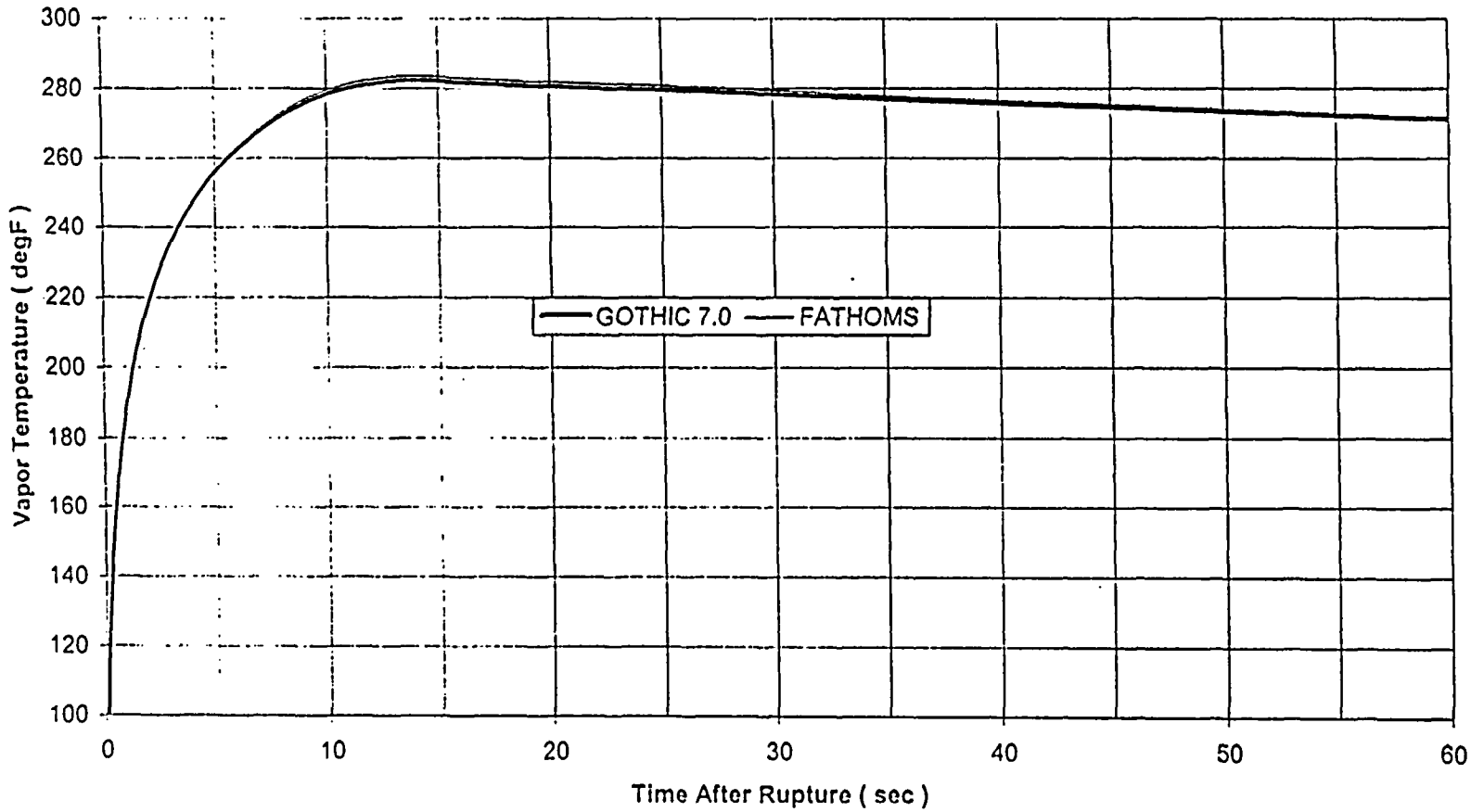


FIGURE C-2

DPC-NE-3003-A
Revision 1

List of Attached Docketed Correspondence

1. 8/11/93 original submittal letter, M. S. Tuckman to NRC
2. 1/27/94 NRC RAI letter, L. A. Wiens (NRC) to J. W. Hampton
3. 6/1/94 response to RIA letter, M. S. Tuckman to NRC
4. 6/9/94 response to NRC telecon questions related to EQ
5. 6/13/02 letter submitting Revision 1, M. S. Tuckman to NRC
6. 7/7/03 response to RAI, K. S. Canady to NRC
7. 7/28/03 response to RAI, K. S. Canady to NRC

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DUKE POWER

August 11, 1993

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Subject: Duke Power Company
Oconee Nuclear Station
Docket Numbers 50-269, -270, and -287
Topical Report DPC-3003-P, "Mass and Energy Release and Containment Response Methodology"

Enclosed for your review is Duke Power Company's Topical Report DPC-NE-3003-P, "Mass and Energy Release and Containment Response Methodology." This report describes Duke's methodology for simulating the mass and energy release from high-energy line breaks, and the resulting containment response for the Oconee Nuclear Station. Also enclosed is a copy of the report entitled "CAP - Containment Analysis Package," which is referenced by DPC-NE-3003, and is included to facilitate the review.

The objective of this report is to describe new methods used to reanalyze the FSAR containment response to loss-of-coolant accidents and steam line breaks, in order to address numerous identified limitations in the existing analyses which date to the early 1970s. The methods and analyses in this report represent a new design basis for which the performance requirements on engineered safeguards systems and the limiting conditions for operation are well established. This modernization of the design basis will improve future evaluations of abnormal conditions and plant design and testing issues.

In accordance with 10CFR 2.790, Duke Power Company requests that this report be considered proprietary. Information supporting this request is included in the attached affidavit. A non-proprietary version of DPC-NE-3003 will be submitted following receipt of the Safety Evaluation Report.

If you have any questions, or need more information, please call Scott Gewehr at (704) 382-7581.

M S. Tuckman

M. S. Tuckman

cc: Mr. L. A. Wiens, Project Manager
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Mr. S. D. Ebneter, Regional Administrator
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R. C. Jones (15 copies)
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Mail Stop 8 E23, OWFN
Washington, D. C. 20555

bxc: (w/o Attachments)
G. A. Copp
K. S. Canady
G. B. Swindlehurst
File: OS-801.01

AFFIDAVIT OF M. S. TUCKMAN

1. I am Senior Vice President, Nuclear Generation Department, Duke Power Company ("Duke"), and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing, and am authorized to apply for its witholding on behalf of Duke.

2. I am making this affidavit in conformance with the provisions of 10 CFR 2.790 of the regulations of the Nuclear Regulatory Commission ("NRC") and in conjunction with Duke's application for withholding which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10 CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke and and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs, relative to a method of analysis that provides a competitive advantage to Duke.

   (iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.790, it is to be received in confidence by the NRC.

   (iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

   (v) The proprietary information sought to be withheld in this submittal is that which is marked in the proprietary version of the report DPC-NE-3003, "Mass and Energy Release and Containment Response Methodology" and supporting documentation, and omitted from the non-proprietary versions.

M. S. Tuckman

M. S. Tuckman

(continued)

This information enables Duke to:

(a) Simulate the mass and energy release rates from loss-of-coolant accidents and steam line break accidents in pressurizer water reactors of the Babcock and Wilcox design.

(b) Simulate the response of a conventional dry containment design to a high-energy line break inside containment.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring a commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman

M. S. Tuckman

(continued)

AFFIDAVIT OF M. S. TUCKMAN (Page 3)

M. S. Tuckman, being duly sworn, on his oath deposes and says that he is the person who subscribed his name to the foregoing statement, and that the matters and facts set forth in the statement are true.

M. S. Tuckman

M. S. Tuckman

Sworn to and subscribed before me this 17th day of August, 1993. Witness my hand and official seal.

Mary P. Dolms

Notary Public

My commission expires January 22, 1996.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 27, 1994

Docket Nos. 50-269, 50-270
and 50-287

Mr. J. W. Hampton
Vice President, Oconee Site
Duke Power Company
P. O. Box 1439
Seneca, South Carolina 29679

Dear Mr. Hampton:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING TOPICAL REPORT DPC-NE-3003-P "MASS AND ENERGY RELEASE AND CONTAINMENT RESPONSE METHODOLOGY"(TAC NOS. M87258, M87259, AND M87260)

By letter dated August 11, 1993, Duke Power Company (DPC) submitted Topical Report DPC-NE-3003-P for review. This report describes the DPC methodology for the calculation of containment temperature and pressure responses to high energy line breaks inside containment for the Oconee facilities. In order to complete our review, additional information, as indicated in the enclosure, is required. Your response to the enclosed questions is requested within 45 days of the date of this letter. If you have questions regarding this matter, please contact me at (301) 504-1495.

This requirement affects fewer than ten respondents and, therefore, it is not subject to Office of Management and Budget review under P.L. 96-511.

Sincerely,

L. A. Wiens, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Request for Additional
Information

cc w/enclosure:
See next page

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Duke Power Company

Oconee Nuclear Station

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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION
OCONEE TOPICAL REPORT DPC-NE-3003-P

1. Figure 2.4-2 (page 2-52) of the topical report presents comparative results of the FATHOMS and CONTEMPT large break LOCA analysis. Please clarify whether the CONTEMPT analysis is a new confirmatory analysis performed using the same input data and assumptions as the FATHOMS analysis, or is the older FSAR analysis discussed in Section 2.4.3.

2. As described in Section 4.4.2 (page 4-15), high point vents are opened at 7200 seconds into the 0.005 ft^2 SB-LOCA analysis. Is the HPV effluent included or accounted for in the break flow for purposes of containment mass and energy release analysis?

3. ANS-56.4-1983, paragraph 3.3, states that "a spectrum of break areas shall be analyzed to assure that the highest primary containment peak pressure and temperature have been determined." Your analyses encompass a single break size. Provide a rationale for concluding that the 34-inch MSL-DEGB containment response is limiting.

4. Referring to Section 5.3 "SG Pressure" (page 5-5), the lower OTSG pressure is non-conservative for break flow and enthalpy. Have any sensitivity studies been performed to examine its effect, or any compensating bias applied to the results? Explain the reason for the modeling problem.

5. Referring to Section 5.3 "Steam Generator Operating Level" (page 5-5), explain how the numbers add-up to 55,000 lbm and why the numbers are inconsistent with FSAR 15.13.4.

6. Regarding "Fission Heat," in Section 5.4 (page 5-7), are *all* or *n minus 1* rods assumed to insert?

7. Regarding "Limiting Single Failure," in Section 5.4 (page 5-11), identify what other single-failures were considered. Indicate whether the proposed plant modifications intended to eliminate the operator action requirement to terminate FW addition involve or could create new single-failure concerns.

8. Regarding "ICS" (page 5.11) in Section 5.4, please provide additional justification or rationale for neglecting the effect of rod motion.

9. Regarding MSLB containment analyses, explain if and how revaporization is considered?

10. Section 6.4.6.5 (page 6-44) of the topical report states that SB-LOCAs require a reduction in the containment spray actuation setpoint and opening of the boron dilution flowpath for acceptable containment response. Please explain the extent to which these requirements have been implemented.

11. Section 6.4.5 (page 6-35) of the report indicates a spray initiation setpoint of 20 psig (plus delay). Section 6.5.5. (page 6-47) indicates 30 psig (plus delay). Please clarify the spray initiation setpoint.

12. The EQ envelope depicted in the report for MSLB (e.g., figures 6.5-2, 6.5-5) is different than that depicted for LOCAs (e.g., figures 6.3-4, 6.4-1). Explain the discrepancy. Also, indicate what "case-by-case" analyses have been performed to confirm the acceptability of the MSLB responses with respect to EQ requirements.

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DUKE POWER

June 1, 1994

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Subject: Duke Power Company
Oconee Nuclear Station
Docket Numbers 50-269, -270, and -287
Topical Report DPC-NE-3003; "Mass and Energy Release and Containment Response Methodology"; Non-proprietary Version of Responses to RAI

By letter dated February 16, 1994, responses were provided to the NRC staff's request for additional information regarding the subject topical report. These responses contained information that Duke considers proprietary, and therefore it was requested that the information be withheld from public disclosure pursuant to 10 CFR 2.790. In accordance with the provisions of ¶ 2.790, attached please find a non-proprietary version of the responses provided by the February 16, 1994 letter.

If there are any questions, please call Scott Gewehr at (704) 382-7581.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

cc: Mr. L. A. Wiens, Project Manager
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Duke Power Responses To The NRC Request For
Additional Information Dated January 27, 1994

Oconee Topical Report DPC-NE-3003-P

Question 1

Figure 2.4-2 (page 2-52) of the topical report presents comparative results of the FATHOMS and CONTEMPT large break LOCA analysis. Please clarify whether the CONTEMPT analysis is a new confirmatory analysis performed using the same input data and assumptions as the FATHOMS analysis, or is the older FSAR analysis dicsussed in Section 2.4.3.

Response

The CONTEMPT analysis shown in Figure 2.4-2 is the old FSAR analysis discussed in Section 2.4.3. The CONTEMPT-generated pressure response is taken from Figure 15-56 in the Oconee FSAR. The FATHOMS result is plotted on the FSAR figure to facilitate comparison.

Question 2

As described in Section 4.4.2 (page 4-15), high point vents are opened at 7200 seconds into the 0.005 ft^2 SB-LOCA analysis. Is the HPV effluent included or accounted for in the break flow for purposes of containment mass and energy release analysis?

Response

The HPV effluent for the 0.005 ft^2 SBLOCA case described in Section 4.4.2 is included in the mass and energy release results. However, since primary-to-secondary heat transfer is recovered for this case, the long-term mass and energy release will be non-limiting. Consequently, the containment response is not specifically analyzed for this case (see pp. 4-16 and 6-28).

Question 3

ANS-56.4-1983, paragraph 3.3, states that "a spectrum of break areas shall be analyzed to assure that the highest primary containment peak pressure and temperature have been determined." Your analyses encompass a single break size. Provide a rationale for concluding that the 34-inch MSL-DEGB containment response is limiting.

Response

Given that the double-ended guillotine break results in the highest break flow rate and there is no liquid carryout credited in this analysis, this break size provides the peak containment pressure. Sensitivity studies with smaller break sizes were performed to determine the limiting case with respect to peak containment temperature. For these smaller break sizes, reactor trip is delayed due to the less severe blowdown. The amount of superheat greatly impacts containment temperature and is a function of hot leg temperature and steam generator pressure. The sensitivity studies show that although the smaller break sizes result in higher hot leg temperatures, break enthalpy is lower than that seen during the first few seconds of the double-ended guillotine break. For a short period of time after initiation of the double-ended guillotine break, the combination of hot leg temperatures near their full power values and rapidly decreasing steam generator pressure results in much higher break enthalpies than the smaller break cases. Thus, the 34 inch double-ended guillotine break is the limiting break size with respect to both peak containment pressure and temperature.

Question 4

Referring to Section 5.3 "SG Pressure" (page 5-5), the lower OTSG pressure is non-conservative for break flow and enthalpy. Have any sensitivity studies been performed to examine its effect, or any compensating bias applied to the results? Explain the reason for the modeling problem.

Response

When initializing a RETRAN model to the desired initial conditions, the user must choose the parameters which are considered fixed and allow other parameters to vary in order to obtain a converged steady-state solution. This approach permits introducing conservative allowances in many initial conditions, but also is constrained by physical reality in that an overall energy balance must exist at time zero. The fixed parameters chosen are usually the ones which are most significant for the analysis (e.g. primary coolant average temperature, coolant flow rate, etc.). Steam generator pressure is one of the parameters that is allowed to vary during initialization in order to achieve the required steady-state energy balance. The value resulting from this initialization process for steam generator initial pressure is 910 psig, which is near the realistic pressure. Therefore, the initial secondary system conditions are consistent with the heat transfer required at this power level. Additional conservatism in this parameter cannot by accommodated by RETRAN and would be non-physical.

Question 5

Referring to Section 5.3 "Steam Generator Operating Level" (page 5-5), explain how the numbers add-up to [ ] lbm and why the numbers are inconsistent with FSAR 15.13.4.

Response

FSAR Section 15.13.4 states that an initial steam generator inventory of 62,600 lbm was used to evaluate the steam line break mass and energy release. However, more current information based on plant operating experience indicates that this assumption is overly conservative. The steam generator inventory of [ ] lbm is based on the assumption that the steam generator downcomer is filled with saturated liquid. Given the volume of the downcomer and the density of saturated liquid at the nominal full power steam generator pressure, a downcomer full of saturated liquid would contain [ ] lbm of water. The mass of water and steam in the tube region of a clean generator at power operating conditions is about [ ] lbm. This would increase the total inventory to about [ ] lbm. However, steam generator fouling will also increase the required tube region inventory. Thus, the assumption of [ ] lbm includes an allowance of [ ] lbm to account for the impact of steam generator fouling on the overall primary-to-secondary heat transfer coefficient.

Steam generator fouling impacts inventory in two ways. First, steam generator fouling increases the frictional pressure drop in the tube region of the steam generator. Thus, a higher level in the downcomer is needed to offset this increased frictional pressure drop. The assumption of a downcomer full of saturated liquid clearly bounds the impact of steam generator fouling on the downcomer inventory. Second, steam generator fouling decreases the overall primary-to-secondary heat transfer coefficient. In order to remove the same amount of energy, the heat transfer surface area, or boiling length, must be increased. An increase in the boiling length increases the tube region inventory. Discussions with B&W steam generator experts indicate that this effect does not appreciably impact the total steam generator inventory. Thus, the total inventory of [ ] lbm should be adequate to conservatively account for the impact of steam generator fouling.

Question 6

Regarding "Fission Heat," in Section 5.4 (page 5-7), are *all* or *n minus 1* rods assumed to insert?

Response

The most reactive control rod is assumed to remain in the fully withdrawn position to maximize the return to power due to the cooldown of the RCS (i.e., n-1 rods are assumed to insert on reactor trip).

Question 7

Regarding "Limiting Single Failure," in Section 5.4 (page 5-11), identify what other single-failures were considered. Indicate whether the proposed plant modifications intended to eliminate the operator action requirement to terminate FW addition involve or could create new single-failure concerns.

Response

The two cases presented in DPC-NE-3003-P assume that the ICS is in automatic control with the MFW control valve functioning properly. These cases are presented primarily to demonstrate the analysis methodology. In order to alleviate the reliance on operator action, a plant modification will be made to automatically initiate feedwater isolation during a steam line break. The design of the feedwater isolation system has not yet been completed. Thus, at the present time the limiting single failure cannot be identified, nor is it known if any new single failure concerns will be created by the feedwater isolation system. A single failure analysis will be performed and these concerns will be addressed as part of the design of the feedwater isolation system.

Question 8

Regarding "ICS" (page 5-11) in Section 5.4, please provide additional justification or rationale for neglecting the effect of rod motion.

Response

Following larger steam line breaks but prior to the resulting reactor trip signal, the combined effect of decreasing turbine header pressure and T-ave results in the ICS increasing reactor demand to the high limit of about 103% FP. Since reactor trip occurs within the first five seconds of the accident and the control rods move at a constant speed of 30 inches per minute, the rods would at most move about 3%. This amount of rod movement would result in a negligible reactivity addition. Thus, it is reasonable to assume that the control rods are in manual control.

For the sensitivity studies with smaller break sizes, the combined effect of decreasing turbine header pressure and T-ave would result in an increase in reactor demand to the high limit of about 103% FP. The nuclear instrumentation (NI) flux error is the difference between actual and indicated power level due to the effect of the reactor vessel downcomer temperature on the excore NI flux detectors. When downcomer temperature decreases, the indicated power level will be less than the actual power level. Examination of the actual power level response and NI flux error for the smaller break cases shows that the indicated power level would increase well above 103% FP due to the cooldown of the Reactor Coolant System. Thus, control rod insertion would occur with the ICS in automatic. Therefore, it is conservative to assume that the control rods are in manual. The NI flux error is conservatively accounted for in the Reactor Protective System by increasing the high power and flux/flow trip setpoints.

Question 9

Regarding MSLB containment analyses, explain if and how revaporization is considered?

Response

The standard interfacial mass and heat transfer equations for the superheated vapor phase in FATHOMS are used to determine the amount of condensation remaining in the droplet phase (that is, remaining in the vapor region as droplets). The heat transfer rates are dependent on drop concentrations, relative velocities between the phases, and several other factors. These equations are given on p. 34 of NUREG/CR-3262, Vol. 1 (Reference 2-20 of DPC-NE-3003-P), which is the COBRA-NC code equations manual. The COBRA-NC equations manual serves as the basis for the FATHOMS code, and therefore these heat transfer equations are the same as in FATHOMS. As the liquid film thickness on the surfaces of the containment walls and heat structures increases, some of this film may be vaporized by the superheated atmosphere. This is all calculated in FATHOMS using the equations referenced above. No additional algorithms to model this heat transfer are applied in any of the FATHOMS analyses.

In the FATHOMS code, separate mass and momentum equations exist for the droplet and vapor phases, but only one set of energy equations exists for the liquid/droplet phases. Therefore, the droplets and continuous liquid phases are always at the same temperature within a single calculational volume. For this reason, the Oconee FATHOMS model contains a separate node for the sump region. Within the atmosphere region, very little liquid will exist in the continuous liquid phase. The temperatures of this liquid film and the droplets in the atmosphere should be very close, so this modeling technique is applied.

Question 10

Section 6.4.6.5 (page 6-44) of the topical report states that SB-LOCAs require a reduction in the containment spray actuation setpoint and opening of the boron dilution flowpath for acceptable containment response. Please explain the extent to which these requirements have been implemented.

Response

A reduction in the Technical Specification value for the spray initiation setpoint is required to keep the Reactor Building temperature within EQ requirements for some SBLOCA scenarios. This reduction is from 30 psig to 20 psig. Even though the Technical Specification setpoint is 30 spig, the actual setoint used at Oconee is 10 psig. Therefore, the impact of the reanalyses on the spray actuation setpoint does not have any real safety impact on current operations. It was decided not to formally request approval of a Technical Specification change until the methodologies in DPC-NE-3003 are approved by the NRC.

Current emergency operating procedures (EOPs) require the opening of the boron dilution flowpath within 9 hours following a large break LOCA. For conservatism, this flowpath was not taken credit for in the large break LOCA containment analyses until 24 hours. In the SBLOCA analyses opening of the boron dilution flowpath is credited at 15 hours. Current EOPs do not instruct the

operator to open the boron dilution line following a SBLOCA. This guidance will be added to the EOPs following approval of DPC-NE-3003-P. In the interim it is concluded that the time available to perform this action is sufficient to enable the engineering staff in the emergency response organization to assume the responsibility for this action.

Question 11

Section 6.4.5 (page 6-35) of the report indicates a spray initiation setpoint of 20 psig (plus delay). Section 6.5.5 (page 6-47) indicates 30 psig (plus delay). Please clarify the spray initiation setpoint.

Response

As mentioned above in the above response to Question #10, the Technical Specification spray initiation setpoint must be lowered from 30 to 20 psig for some SBLOCA scenarios. In the LBLOCA and MSLB containment analyses, the existing 30 psig setpoint was assumed since this value had no unacceptable impact on the results of either analysis.

Question 12

The EQ envelope depicted in the report for MSLB (e.g., figures 6.5-2, 6.5-5) is different than that depicted for LOCAs (e.g. figures 6.3-4, 6.4-1). Explain the discrepancy. Also, indicate what "case-by-case" analyses have been performed to confirm the acceptability of the MSLB responses with respect to EQ requirements.

Response

The EQ envelopes for the MSLB and LOCA figures of Chapter 6 of DPC-NE-3003 are the same. From 0 to 100 seconds, the EQ requirement is 312°F, and from 100 to 1500 seconds, the requirement is 290° F. The may appear different due to the LOCA figures not including the first 100 seconds. During the time period from 0-100 seconds, the EQ envelope is not challenged by the LOCA and was not plotted in order to provide better resolution after 100 seconds. Figure 6.2-12 shows that the peak LOCA temperature in the 0-100 second time period is 285°F. This is well below the 312°F EQ envelope limit during that period of time.

Upon the completion of DPC-NE-3003-P, Duke Power performed an engineering calculation to demonstrate the impact of the new MSLB peak containment temperature analysis result on the safety-related equipment located inside containment. All equipment required to mitigate and/or monitor MSLB was included in this evaluation. To demonstrate the adequacy of the equipment to perform its safety function, two "worst-case" pieces of equipment were selected. The "worst-case" equipment are the Viking penetration and BIW cable. "Worst-case" is defined as the equipment exposed to the lowest test temperature for the shortest period of time during the LOCA test program. The test temperatures for these cases were plotted and compared to the containment temperatures obtained for the MSLB to demonstrate the insignificance of those time periods for which the MSLB curve is above the tested curve when compared to the temperatures and duration of the testing. The calculation utilizes the results and conclusions of an analysis performed by

Babcock & Wilcox using a two dimensional finite element model. This model showed that even for containment temperature reaching almost 500°F during MSLB, the temperature of the equipment internals was significantly lower than for a LOCA due to the brief period of time during which the equipment was exposed to elevated temperatures. These analyses were done for transmitters, motor operators, electrical penetrations, instrument enclosures, and cable jackets. Based on these engineering calculations, it was concluded that the MSLB temperature response did not have an unacceptable impact on safety-related equipment inside containment.

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DUKE POWER

June 9, 1994

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555
Attention: Document Control Desk

Subject: Oconee Nuclear Station
Docket Numbers 50-269, -270, and 287
Topical Report DPC-NE-3003, "Mass and Energy Release and Containment Response Methodology"; Response to RAI

In a telephone call on June 1, 1994 between Len Wiens of the NRC and Scott Gewehr of Duke, regarding the subject topical report, a question was put forth regarding main steam line break temperature and equipment qualification in containment. Attached please find Duke's response to this question.

If there are any questions, please call Scott Gewehr at (704) 382-7581.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

cc: Mr. L. A. Wiens, Project Manager
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File

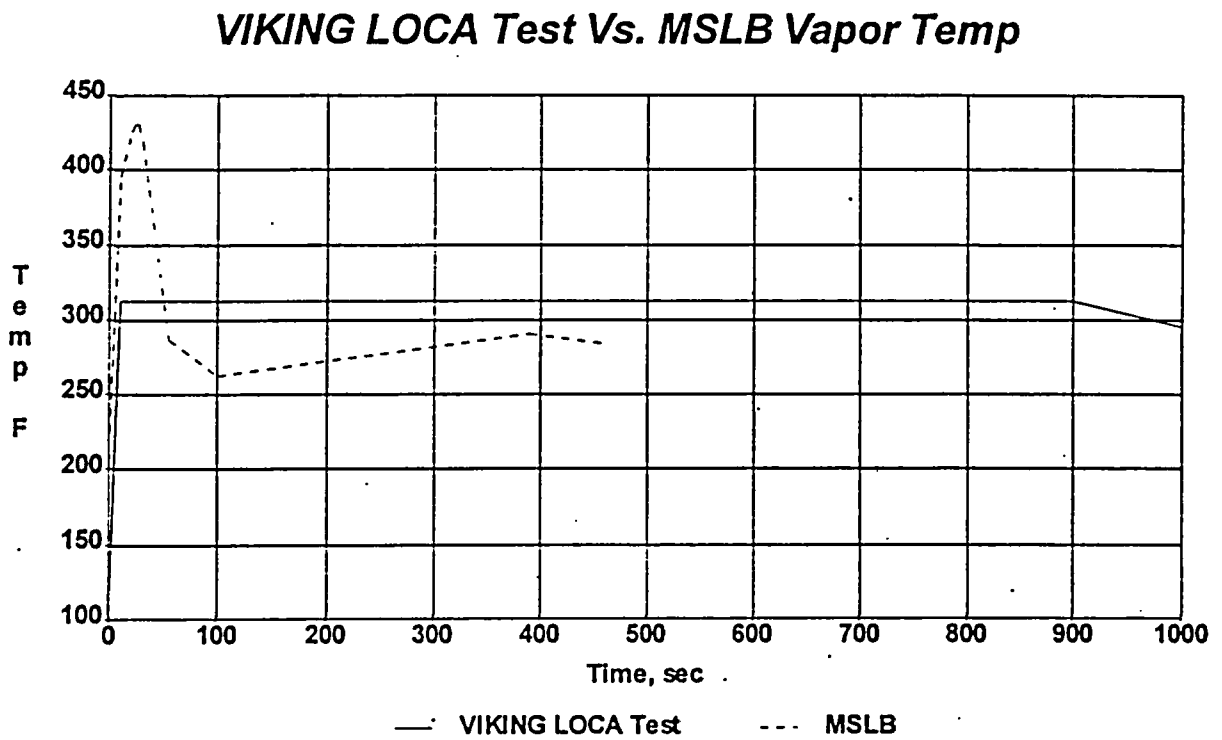
OCONEE MSLB/EQ ANALYSIS

An analysis was conducted under Oconee Calculation OSC-5460. The following is a summary of that calculation:

The attached table lists all equipment located inside containment required to mitigate and/or monitor a MSLB. This table also provides the required operability times and qualification parameters (temperature and pressure) to which the equipment was tested. For the purpose of this calculation, the "worse-case" test parameters have been selected to demonstrate adequacy of the equipment to perform its safety function. "Worse-case" is defined as the lowest test peak temperature over the shortest period of time tested. Review of the LOCA test profile indicates that the two "worse-case" pieces of equipment are the Viking penetration and BIW cable. For conservatism, LOOP 180 (highest temperature/longest period) , was used for comparison to the equipment test profiles.

VIKING PENETRATION :

The Viking penetration was tested at a peak temperature of 312°F for the first 15 minutes, ramped down to 295°F for the next 45 minutes, and then held at 290°F for a further 23 hours. The Viking profile compared to the LOOP180 MSLB is shown below:

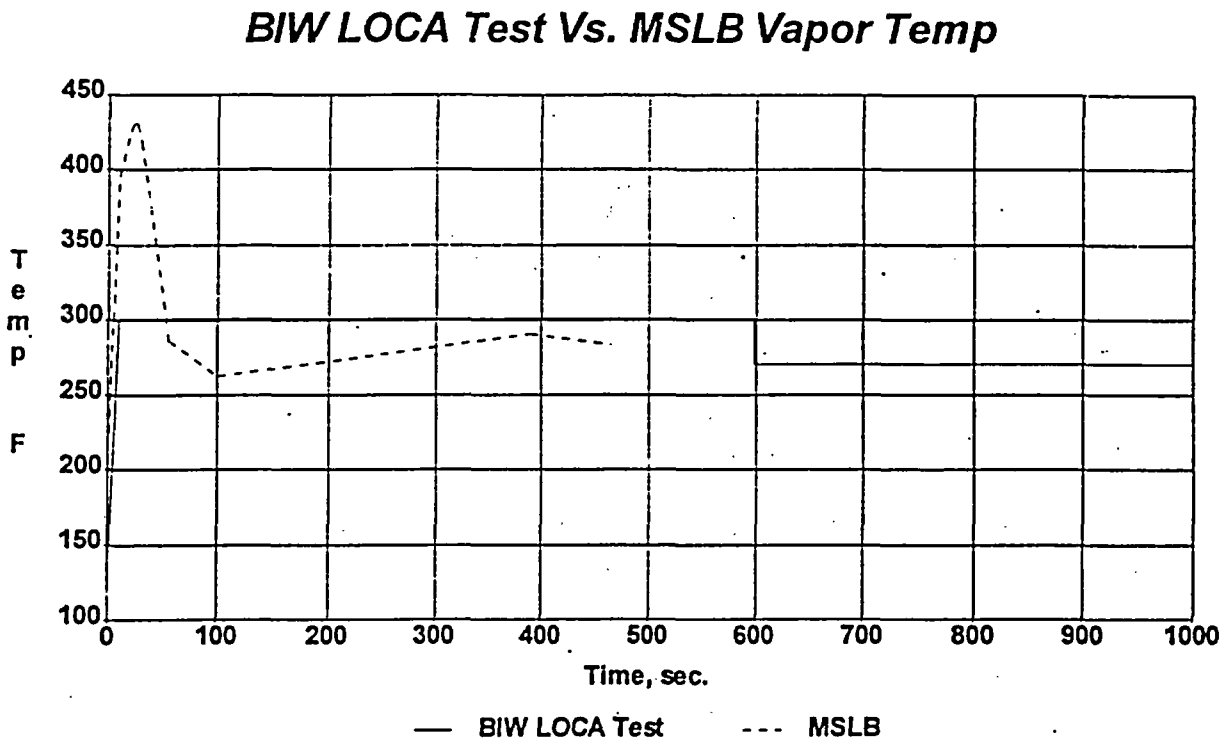


Additionally, the Viking penetration was thermally aged and irradiated prior to LOCA testing. The component parts were thermally aged to well over 60 years using Arrhenius methodology and irradiated to postulated LOCA doses (>10^8 rads) which are not present during a MSLB.

OCONEE MSLB/EQ ANALYSIS

BIW CABLE:

The BIW cable was tested at a peak temperature of 300°F for the first 10 minutes, ramped down to 270°F from 10 minutes to 2 hours, ramped down to 240°F from 2 hours to 4 hours, ramped down to 210°F from 4 hours to 8 hours, and finally ramped down to 181°F from 8 hours to 30 days. The BIW profile compared to the LOOP180 MSLB is shown below:



The BIW cable was also thermally pre-aged and irradiated prior to LOCA testing. The cable was thermally aged at 121°C (250°F) for 375 hours and irradiated to $1.1X10^8$ rads.

These curves are only taken out to 1000 seconds because the MSLB profile begins to ramp down at approximately 400 seconds. Additionally, these curves are intended to illustrate the insignificance of those time periods for which the MSLB curve is above the tested curve when compared to the temperatures and duration of the testing.

Typical LOCA test profiles last for significantly longer periods of time as compared to MSLB profiles, minutes and hours versus seconds. As a result of thermal lag the internals of equipment will not experience the peak temperature associated with the MSLB. Because the LOCA test profiles are considerable longer and "soak" the equipment, the internal temperatures of the equipment reaches a higher temperature than would be expected during a MSLB. Therefore the LOCA testing subjects the equipment to a more severe environment than that of a MSLB.

OCONEE MSLB/EQ ANALYSIS

This conclusion is supported by B&W Engineering Analysis 51-1158880-00. This analysis was conducted to demonstrate the effect of high SLB temperatures versus LOCA temperatures on equipment.

For example, the internal electronics temperature response for a transmitter to both a SLB and LOCA was examined. The SLB temperatures for the internal electronics was shown to be significantly lower compared to the LOCA response. This is due to the longer "soak time" at peak temperature for a longer period of time associated with the LOCA profile. Additionally, the same phenomena for a motor operator was also demonstrated. Similar analyses has been conducted modeling electrical penetrations, instrument enclosures, and cable jackets with the same results.

The conclusion of the B&W analysis indicates that "...the brief temperature spike due to super-heated steam following a Steam Line Break does not affect the internals of containment mounted equipment significantly. In every case examined, the long duration temperatures associated with a LOCA were more severe to equipment than the higher SLB spike."

This conclusion is also applicable to the installed Oconee equipment and additional margin can be realized due to the following:

- All containment cable listed is armored cable which is not taken into consideration in the B&W analysis. Additionally, the cable was tested without armor.

- The Viking penetration installed configuration utilizes a galvanized enclosure cover at the interface of the connectors to the penetrations

CONCLUSION: This analysis, in conjunction with the B&W analysis, demonstrates that equipment internals do not experience the peak temperatures associate with the MSLB vapor temperature spike. Even though the MSLB temperature may be higher for a short period of time, the equipment internal temperature is higher for each point in time when subjected to a LOCA. Additionally, LOCA testing of equipment assures that equipment has been tested to more severe conditions than would be experienced by the equipment during a MSLB. Therefore, the equipment required to mitigate the consequences of the MSLB is qualified and would perform its safety function.

OCONEE NUCLEAR STATION

MSLB EQUIPMENT/EQ PARAMETERS

TAG #	FUNCTION	MANU/MODEL	BUILDING	OP. TIME	QUAL. PARAMETERS TEMP.	QUAL. PARAMETERS PRESS.	COMMENTS
PT-17P, 18P, 19P, 20P	RCS Pressure NR	Rosemount 1152 (U1), 1154 (U2&3)	RB	2 min.	350°F 420°F	70 psig 110 psig	
NI-1, 3	Power Range NI's	Gamma Metrics	RB	2 min.	420°F	70 psig	
PT-21P, 22P, 23P	RCS Pressure WR	Rosemount 1153D	RB	15 min.	460°F	88 psig	
CF-1, 2, 5, 6	CF Iso. Valves		RB	10 days			These Valves are administratively controlled - Power is racked out during normal operation.
HP-3, 4, 20	HPI Active Valves	Limitorque	RB	10 days	340°F	105 psig	
	RBCU Fan Motors	Joy/Reliance	RB	10 days	330°F	78 psig	
MSPT-0277, 0278, 0279, 0280	OTSG Pressure Transmitters	Rosemount 1154	RB	10 days	420°F	110 psig	
LT-80, 81, 82, 83	S/G Level Transmitters	Rosemount 1154	RB	10 days	420°F	110 psig	
LPSW-565, 566	Aux. RB Coolers	Rotork NA-1	RB	30 min.	385°F	75 psig	
CABLES:							
1PSX16H.3	Pressure Transmitters	BIW	RB	15 min.	300°F	60 psig	
12XJ12G1 3XJ250G.2	Valves RBCU Motors	OKONITE.	RB	10 days	324°F	80 psig	
19XJ12G.1	Valves	ANACONDA	RB	30 min.	346°F	113 psig	
1SPX16G.3	Level Transmitters	SAMUAL MOORE	RB	10 days	340°F	105 psig	
PENETRATIONS							
		VIKING	RB	10 days	312°F	65 psig	
		CONAX	RB	2 min.	390°F	80 psig	
MISCELLANEOUS							
	Sealing Material	Scotchcast 9	RB	10 days	398°F	61 psig	
	Splice Material	Raychem	RB	10 days	442°F	132 psig	

Duke Energy.

M. S. Tuckman
Executive Vice President
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June 13, 2002

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington D. C. 20555-0001

ATTENTION: Chief, Information Management Branch
Division of Program Management
Policy Development and Analysis Staff

Subject: Duke Energy Corporation
Oconee Nuclear Station - Units 1, 2, and 3
Docket Nos. 50-269, 50-270, and 50-287

Revisions to Topical Reports DPC-NE-3000, -3003, and 3005 In Support of Steam Generator Replacement

Enclosed herein, please find DPC-NE-3000-P, Revision 3, *"Thermal-Hydraulic Transient Analysis Methodology,"* DPC-NE-3003-P, Revision 1, *"Mass and Energy Release and Containment Response Methodology,"* and DPC-NE-3005-PA, Revision 2, *"UFSAR Chapter 15 Transient Analysis Methodology."* These reports are submitted for NRC review and acceptance as licensing Topical Reports under the NRC licensing topical report program for referencing in other license applications. Both the proprietary and non-proprietary versions of these Topical Reports are enclosed.

Please note that there is information enclosed that Duke Energy Corporation (Duke) considers proprietary. In accordance with 10CFR2.790, Duke requests that this information be withheld from public disclosure. An affidavit which attests to the proprietary nature of this information is included in this letter.

The replacement of steam generators at the Oconee Nuclear Station requires revision of three Duke Energy topical reports that detail the methodologies for analysis of transients and accidents. In addition, minor revisions to these reports are required periodically to maintain the technical content of these

reports current with the plant design and for consistency with other Duke topical reports. These reports and a summary of the revisions are as follows:

DPC-NE-3000-PA, Revision 2, *"Thermal-Hydraulic Transient Analysis Methodology"*, describes the RETRAN-02 system transient thermal-hydraulic analysis models, and the VIPRE-01 core thermal-hydraulic analysis models for analyzing non-LOCA transients and accidents for the Oconee, McGuire, and Catawba Nuclear Stations. Revision 2 was approved by the NRC in the Safety Evaluation Report (SER) enclosed with the letter dated October 14, 1998. Revision 3 includes the new RETRAN-3D model for the Oconee replacement steam generators. RETRAN-3D was reviewed and approved by the NRC by SER dated January 25, 2001. Compliance with the conditions and limitations in the NRC's RETRAN-3D SER are included in the attachments. The other revisions that are included in Revision 3 are minor technical or editorial changes necessary to maintain the topical report current since the last submittal in 1997. A description of each of the revisions and the associated technical justification are provided in the attachments. Attachments 1 and 2 are the proprietary and non-proprietary versions, respectively.

DPC-NE-3003-PA, *"Mass and Energy Release and Containment Response Methodology"*, describes the models for analyzing the mass and energy release from high-energy line breaks inside containment, and the resulting containment pressure and temperature response for the Oconee Nuclear Station. This report was approved by the NRC in the SER enclosed with the letter dated March 15, 1995. Revision 1 includes the new RELAP5 model for the replacement steam generator LOCA mass and energy release analyses, and the new RETRAN-3D model for the steam line break mass and energy release analyses. Revision 1 also includes a new GOTHIC 7.0 containment analysis model. Other minor revisions that are included in Revision 1 are minor technical or editorial changes necessary to maintain the topical report current since the last submittal in 1993. A description of each of the revisions and the associated technical justification are provided in the attachments. Attachments 3 and 4 are the proprietary and non-proprietary versions, respectively.

DPC-NE-3005-PA, Revision 1, *"UFSAR Chapter 15 Transient Analysis Methodology"*, describes the methodology for analyzing the UFSAR

Chapter 15 non-LOCA transients and accidents for the Oconee Nuclear Station. Revision 1 was approved by the NRC in the SER enclosed with the letter dated May 25, 1999. Revision 2 includes changes associated with the replacement steam generators and use of RETRAN-3D. Other minor revisions that are included in Revision 2 are minor technical or editorial changes necessary to maintain the topical report current since the last submittal in 1999. A description of each of the revisions and the associated technical justification are provided in the attachments. Attachments 5 and 6 are the proprietary and non-proprietary versions, respectively.

Approval of these topical report revisions is requested by June 3, 2003 to support startup of Oconee Unit 1 with the replacement steam generators currently scheduled for September, 2003.

If there are any questions or if additional information is needed on this matter, please call J. A. Effinger at (704) 382-8688.

Very truly yours,

M. S. Tuckman

M. S. Tuckman
ATTACHMENTS

xc: with Non-Proprietary Attachments

L. A. Reyes, Regional Administrator

U.S. Nuclear Regulatory Commission, Region II
Atlanta Federal Center
61 Forsyth Street, SWW, Suite 23T85
Atlanta, GA 30303

M. C. Shannon, NRC Senior Resident Inspector (ONS)

xc: with Proprietary and Non-Proprietary Attachments
L. N. Olshan, NRC Project Manager (ONS)
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Mail Stop 08-H12
Washington, DC 20555

AFFIDAVIT

1) I am Executive Vice President of Duke Energy Corporation; and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant licensing; and am authorized on the part of said Corporation (Duke) to apply for this withholding.

2) I am making this affidavit in conformance with the provisions of 10CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding, which accompanies this affidavit.

3) I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4) Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld:

   a) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

   b) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs relative to a method of analysis that provides a competitive advantage to Duke.

   c) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, it is to be received in confidence by the NRC.

   d) The information sought to be protected is not available in public to the best of our knowledge and belief.

M. S. Tuckman

M. S. Tuckman

(Continued)

e) The proprietary information sought to be withheld in this submittal is that which is that which is marked in Attachments 1, 3, and 5 to Duke Energy Corporation letter dated June 13, 2002; Subject: Revisions to Topical Reports DPC-NE-3000, -3003, and 3005 in Support of Steam Generator Replacement. This information enables Duke to:

   i) Respond to NRC requests for information regarding transient response of Babcock & Wilcox Pressurized Water Reactors.

   ii) Simulate UFSAR Chapter 15 transients and accidents for Oconee Nuclear Station.

   iii) Perform safety evaluations per 10CFR50.59.

   iv) Support Facility Operating License/Technical Specifications amendments for Oconee Nuclear Station.

f) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

   i) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

   ii) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

   iii) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman

M. S. Tuckman

(Continued)

M. S. Tuckman affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to on this 13TH day of

JUNE, 2002

Mary P. Nelms
Notary Public

My Commission Expires:

JAN 22, 2006

SEAL

Attachment 4

Mass and Energy Release and Containment Response Methodology

DPC-NE-3003 (Non-Proprietary)
Revision 1

June 2002

Nuclear Engineering Division
Nuclear Generation Department
Duke Power Company

Description and Technical Justification

Revision 1 to DPC-NE-3003 consists of changes necessary to model the replacement steam generators with RETRAN-3D and RELAP5, a new approach to address water carryout during main steam line break, an addition of the GOTHIC 7.0 code for containment analysis, modeling enhancements, technical and editorial changes to modernize and update the methodology since Revision 0 was submitted in 1993, and an evaluation of compliance with the NRC's SER on RETRAN-3D. Each of the revisions is described in detail, and model revisions are supported by technical justification. The RETRAN-3D modeling has been reviewed by Computer Simulation & Analysis (CSA), Inc., the developer of the RETRAN-3D codes. There are currently no other organizations using RETRAN-3D for modeling B&W-designed reactors, so a peer review other than by the code vendor was not possible. The GOTHIC 7.0 content of the report has been reviewed by Numerical Applications, Inc., the developer of the FATHOMS and GOTHIC codes.

The Oconee replacement steam generators (ROTSGs) are being manufactured in Canada by Babcock & Wilcox Canada (BWC). The ROTSG are quite similar functionally to the original OTSGs (Figure 1), and are characterized as a like-for-like replacement. The design differences that are important for thermal-hydraulic modeling are 1) flow-restricting orifices in the steam outlet nozzles, 2) Inconel-690 tubes, 3) 15,631 vs 15,531 tubes, 4) thinner pressure vessel / wider downcomer, 5) thinner tubesheets resulting in 3.625 inch longer heated tube length, 6) 1.2% greater heat transfer area, and 7) more water in the steam generator. Of these design changes only the flow-restricting steam exit nozzles have a significant impact on the UFSAR transient and accident analyses. These nozzles reduce the effective size of steam line breaks during blowdown of a steam generator to 1.804 ft^2. This design feature significantly changes the steam line break mass and energy release. The other design changes will not cause a significant change in plant response during transients and accidents. For that reason the modeling of the ROTSGs is similar to the modeling of the OTSGs, with changes limited to those described herein.

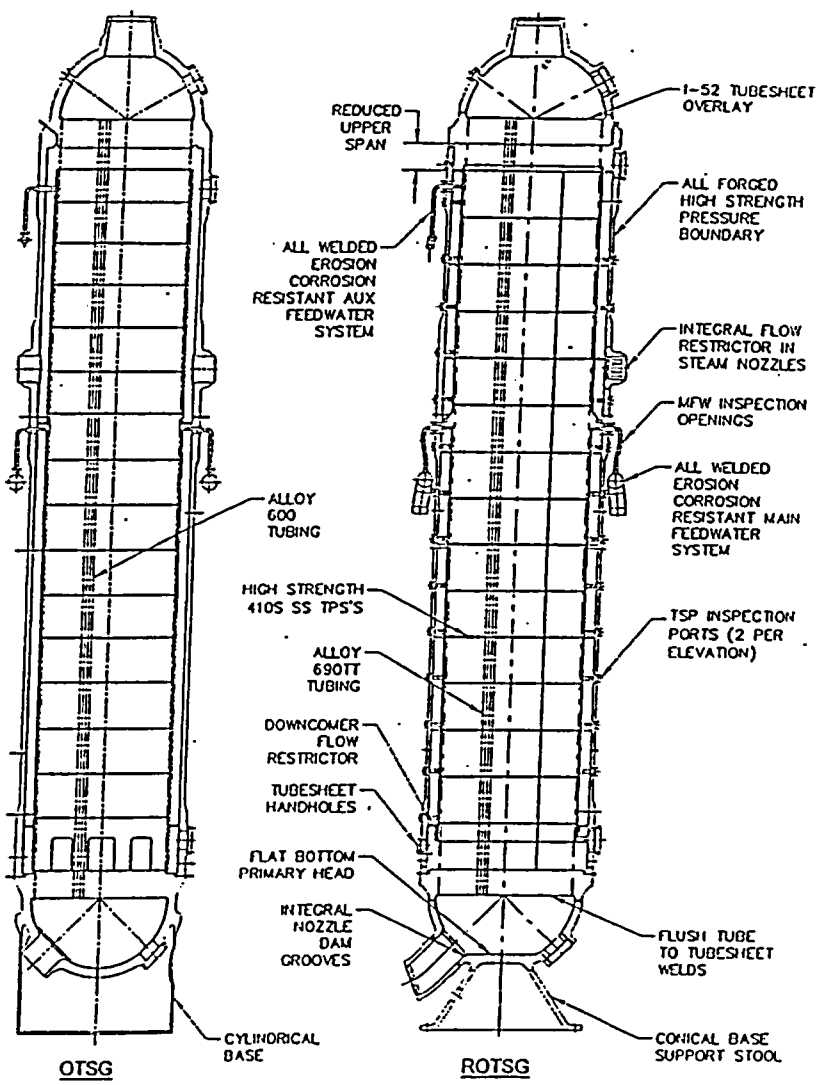
The original steam line break mass and energy release analysis methodology was based on the Electric Power Research Institute's RETRAN-02 code. The next generation in the RETRAN family of codes, RETRAN-3D, was approved by the NRC in the SER dated January 25, 2001. Revision 1 to DPC-NE-3003 describes the modeling for the Oconee steam line break mass and energy release analysis with RETRAN-3D replacing RETRAN-02. [

] In addition, a new modeling approach was developed by Duke and CSA to conservatively account for water carryout during steam line breaks. All of the details of the transition to using RETRAN-3D instead of RETRAN-02 for Oconee modeling are presented. The conditions and limitations in the NRC's SER for RETRAN-3D are also addressed.

The modeling enhancements and technical and editorial revisions that modernize and update the methodology are not necessarily associated with either the ROTSGs or the transition to

RETRAN-3D. These changes are technical enhancements or are necessary to maintain the content of the report consistent with the current plant design, as well as to correct errors. No technical justification is necessary for editorial revisions and corrections. The changes are presented in the order that they appear in DPC-NE-3003. The analysis inputs and results presented in Chapters 3-6 are a demonstration of the Revision 0 methodology. New Appendix A describes the Oconee ROTSG steam line break mass and energy release methodology. New Appendix B addresses the limitations and conditions in the RETRAN-3D SER as relates to the application to steam line break mass and energy release. New Appendix C details the use of GOTHIC 7.0 for the Oconee containment response analysis methodology.

Figure 1 OTSG / ROTSG Comparison



Note

Appendices A, B, and C have been inserted into the main body of DPC-NE-3003-A with Revision 1, and have been deleted from this attachment.

Changes and Technical Justification

Cover Page and Frontal Pages

1. The revision and date will be revised

2. The table of contents and the lists of table, figures , and acronyms will be updated

Chapter 1

3. Page 1-1, second paragraph, revise to indicate the correct location in the UFSAR of the analyses associated with this topical report (editorial)

   Change: "Chapter 15 of the"

   To: "Chapter 15 (now Section 6.2)"

4. Page 1-3, new text at bottom, add the following descriptions of the existing supplements and the new Revision 1 appendices (editorial)

   Supplement 1 was added with the November 1997 version of Revision 0 of the report to present updated steam line break mass and energy release and containment response analysis results that supercede the results in Chapters 5 and 6 of the original report.

   Supplement 2 was added with the November 1997 version of Revision 0 of the report to present validation analyses comparing the FATHOMS prediction to data from the CVTR facility.

   Appendix A was added in Revision 1 to describe the replacement steam generator (ROTSG) steam line break mass and energy release modeling with RETRAN-3D (Reference 1-10).

   Appendix B was added in Revision 1 to address the RETRAN-3D SER (Reference 1-11) conditions and limitations as relates to the replacement steam generator (ROTSG) steam line break mass and energy release modeling with RETRAN-3D.

   Appendix C was added in Revision 1 to include the use of the GOTHIC 7.0 code (Reference 1-12) for containment response analysis.

5. Section 1.1: New references added and existing references updated to current revisions. (editorial)

   1-5 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

   1-10 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI, NP-7450(A), Volumes 1-4, Revision 5, July 2001

1-11 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

1-12 GOTHIC Containment Analysis Package Version 7.0, NAI-8907-02, Revision 13, July 2001

Chapter 2

6. Section 2.1.2, p. 2-6, Simulation Model, is revised to include a new subsection "Main Feedwater Piping" to describe the inclusion of additional nodes in the RELAP5 model to include main feedwater piping. (model revision)

   Insert the following new subsection at the bottom:

   "Main Feedwater Piping

   The RELAP5 nodalization is revised to include the main feedwater piping between the last check valve and the steam generator. This enables modeling flashing of the main feedwater in the piping if the steam generator pressure decreases low enough for the flashing to occur. Should this occur additional hot water will be expelled into the steam generator with the potential to increase secondary-to-primary heat transfer. This additional modeling detail is necessary to accurately model the main feedwater boundary condition."

   Technical Justification: Steam generator depressurization to below the saturation pressure of the hot water in the main feedwater piping will cause some of the water to flash to steam and expel water into the steam generators. For LOCA mass and energy releases it is possible that the steam generators will depressurize sufficiently for this flashing phenomenon to occur. Therefore it is necessary for the RELAP5 model to include the main feedwater piping that may flash. The results of the analyses have shown this to be a very minor effect, but it is modeled for completeness.

7. Section 2.1.2, p. 2-6, Simulation Model, is revised to include a new subsection "ROTSG Model" to describe the changes in the RELAP5 model to represent the replacement steam generators. (model revision)

   Insert the following new subsection at the bottom:

   "ROTSG Model

   The replacement steam generators (ROTSGs) are modeled in RELAP5 with the same nodalization as the original steam generators. All of the RELAP5 model input data have been recalculated for the ROTSGs based on data supplied by B&W Canada, the manufacturer. Refer to Appendix A for a description of the design differences."

   Technical Justification: The Oconee replacement steam generators are very similar to the original steam generators. The main differences are dimensional or material, such as the steam outlet nozzles being equipped with flow-restricting orifices, Inconel-690 tube

alloy, slightly longer tubes, thinner pressure vessel, etc. Based on these limited design changes the existing RELAP5 nodalization was judged to be suitable for the ROTSGs also. All of the input data were recalculated based on data supplied by B&W Canada (BWC), the manufacturer. The initial conditions provided by BWC were used to determine the conservative initial conditions for the applications in this report. This model building activity and the integration with the original RELAP5 model have achieved the desired results.

8. Page 2-22 was inadvertently replaced by Page 3-22 in the 1997 published version. The corrected page 2-22 is the next page in this attachment (editorial)

9. New Section 2.5 "RETRAN-3D Code Description". (model revision)

"2.5 RETRAN-3D Code Description

RETRAN-3D (Reference 2-23) was developed by Computer Simulation & Analysis, Inc. for EPRI to enhance and extend the simulation capabilities of the RETRAN-02 code. RETRAN-3D has many new and enhanced capabilities relative to RETRAN-02, in particular, a 3-D kinetics core model, improved two-phase models, an improved heat transfer correlation package, and an implicit numerical solution method. Most of the capabilities of the RETRAN-02 code have been retained within RETRAN-3D as options, except for a limited number of models and correlations that were not in use. The NRC SER for RETRAN-3D is dated January 25, 2001 (Reference 2-24). The SER includes limitations and conditions on the use of the code for licensing applications. Refer to Appendix A for details on the application of RETRAN-3D for steam line break mass and energy release analyses. Appendix B is an evaluation of the conditions and limitations in the RETRAN-3D SER."

10. New Section 2.6 "GOTHIC 7.0 Code Description". (model revision)

2.6 GOTHIC 7.0 Code Description

The GOTHIC 7.0 code (Reference 2-25) was developed by Numerical Applications, Inc. for EPRI as the next version of the GOTHIC family of codes for thermal-hydraulic analysis of nuclear power plant containment and auxiliary buildings. The GOTHIC code was developed from the FATHOMS code (Reference 2-19), which in turn was developed from the COBRA-NC code (Reference 2-20). Similar to FATHOMS, GOTHIC 7.0 solves the conservation equations for mass, energy, and momentum for multi-component two-phase flow. Refer to Appendix C for details on the application of GOTHIC 7.0 for LOCA and steam line break containment response analyses.

11. Section 2.4 Referenced renumbered to 2.7 and new references and revisions to existing references added (editorial)

2-18 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, June 2002

2-23 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI, NP-7450(A), Volumes 1-4, Revision 5, July 2001

Refer to Change #8

2-24 Letter, S. A. Richards (NRC), to G. L. Vine (EPRI), Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", January 25, 2001

2-25 GOTHIC Containment Analysis Package Version 7.0, NAI-8907-02, Revision 13, July 2001

Chapter 3

12. Section 3.1, p. 3-2, second paragraph, change the termination of the RELAP5 analysis from 30 minutes to the end of the BWST injection phase (model revision)

    Change: "quasi-steady-state conditions have evolved in the RCS ($\approx$30 minutes)."

    To: "the end of the cold leg injection phase."

    Technical Justification: With further experience in applying the methodology it has become apparent that the appropriate time for terminating the RELAP5 phase of the analysis is at the transition from the BWST injection phase to the sump recirculation phase, not necessarily at 30 minutes. At that point in the LOCA scenario the system has evolved into a nearly steady-state boiling pot mode of core cooling, and the transition to the use of the BFLOW code for the continuation of the mass and energy release calculation can be made. The transition to the sump recirculation phase also involves a step change in the ECCS suction temperature. Making the code transition at this point in time is consistent with a transition in the cooling mode.

13. Section 3.1, p. 3-2, second paragraph, clarify that the BFLOW code is only used for cold leg break LOCA analysis, and to describe what is used for hot leg break LOCAs. (model revision)

    Change: "The BFLOW code is used to calculate the mass and energy release for the remainder of the analysis."

    To: "The BFLOW code is used to calculate the mass and energy release for the remainder of the analysis for cold leg break LOCAs. RELAP5 is run for hot leg LOCAs until the break flow has reached a quasi-steady condition. The remainder of the mass and energy release is performed in FATHOMS using an input boundary condition based on decay heat."

    Technical Justification: The BFLOW code is designed to calculate the long-term mass and energy release for cold leg break LOCAs. This was not mentioned in the original report since the focus of the report was modeling cold leg LOCAs. This change is made for clarification purposes. The hot leg break long-term mass and energy release is calculated in the FATHOMS code. This method consists of adding decay heat through boundary condition data to heat the ECCS water originating in the sump after it has been cooled by the LPI heat exchangers. This simple approach is appropriate because hot leg break LOCAs evolve into a simple process by which water is pumped into the reactor vessel where it flows up through the core and absorbs decay and sensible heat before flowing out the break.

14. Section 3.2, p. 3-4, Pressurizer Level, the basis for the initial pressurizer level has been updated for consistency with technical specifications. (model revision)

    Change: "The level control system normally maintains pressurizer level at the 220 inch setpoint. However, the only operational restriction imposed on pressurizer level during power operation is to trip the reactor manually if level reaches 375 inches. High level and high-high level alarms exists at 260 and 315 inches to warn the operator of a high level condition. Although an allowance of 25 inches has been determined to bound the level uncertainty, an initial level of 245 inches (220 + 25) is less than the setpoint value for both high level alarms. The initial pressurizer level is assumed to be 315 inches since it is reasonable to believe that the operators will respond to the high level alarms and act to maintain pressurizer level below this value."

    To: "The level control system normally maintains pressurizer level at the 220 inch setpoint. High level and high-high level alarms exists at 260 and 315 inches to warn the operator of a high level condition. Technical specifications require pressurizer level to be <285 inches. The uncertainty on the level instrument is 25 inches. Therefore, an initial condition of 310 inches, corresponding the technical specification value plus uncertainty is assumed in the analyses."

    Technical Justification: The technical specifications include limits on the ranges of parameters for operation to ensure that the licensing basis safety analyses remain bounding. Assuming the maximum allowed technical specification value for initial pressurizer level plus instrument loop uncertainty is appropriate for this application.

15. Section 3.2, p. 3-5, Steam Generator Operating Level, is changed to indicate that the current mass values are only applicable to the original steam generators, and that a conservative high mass is assumed for the replacement steam generators. (model revision)

    Change: "The nominal full power steady-state mass is [ ] per steam generator."

    To: "The nominal full power steady-state mass is [ ] per steam generator for the original steam generators."

    Change: "An initial steam generator mass of approximately [ ] . . ."

    To: "An initial steam generator mass of approximately [ ] for the original steam generators . . ."

    Insert at the bottom: "A conservative high initial mass is assumed for the replacement steam generators."

    Technical Justification: The methodology for both the original and the replacement steam generators assumes a high initial mass. This maximizes the energy transferred from the secondary to the primary during the LOCA mass and energy release, and is a conservative assumption. This revision clarifies that the mass numbers in the report are applicable to the original steam generators only.

16. Section 3.2, p. 3-6, Safety Injection Tanks, is revised to "Core Flood Tanks", and to change the water temperature from 120°F to 130°F to be consistent with the bounding tank water temperature. (editorial)

    Change: "Safety Injection Tanks"

    To: "Core Flood Tanks"

    Change: "120°F"

    To: "130°F"

    Technical Justification: The ECCS accumulators at Oconee are referred to as the core flood tanks, and so the title of this section is revised. Although the temperature of the core flood tank water is considered an input value, since the 120°F value was stated in the topical report it is updated to 130°F, the value currently assumed to bound plant operating conditions.

17. Section 3.2, p. 3-7, RCS Flow, is changed to indicate that the current flow values are only applicable to the original steam generators, and that high and low RCS flow are analyzed as a sensitivity parameter for the replacement steam generators. (model revision)

    Insert new paragraph at the bottom:

    "The above discussion is applicable to the methodology used in the analysis of the original steam generators. For the replacement steam generator analyses RCS flow is a sensitivity parameter and both bounding high and low initial RCS flow values are analyzed to determine the limiting case."

    Technical Justification: The methodology for the original steam generators assumed low initial flow was conservative based on the justification presented. The flow numbers are associated with the original analysis. The application of the methodology for the replacement steam generators has been revised to consider RCS flow as a sensitivity parameter, and analysis are performed with both bounding high and bounding low RCS flow values to determine the limiting case. Although RCS flow is not an extremely important initial condition, this approach ensures that the limiting case is analyzed.

18. Section 3.3.1.1, p. 3-8, RCS and Steam Generator Level, the pressurizer level input value is changed from 315 inches to 310 inches. (editorial)

    Change: "315 inches"

    To: "310 inches"

    Technical Justification: See Section 3.2 item above for details.

19. Section 3.3.1.1, p. 3-9, Core Stored Energy, the fuel temperature values are those assumed in the original analysis. New analyses use bounding values for the current fuel design. (editorial)

    Insert new paragraph at the bottom:

"The above fuel temperatures were assumed in the original analyses. New analyses use bounding values for the current fuel design."

Technical Justification: Fuel temperature is considered an input value in the methodology. The values presented in the report were bounding values for the fuel designs used at that time. The methodology requires bounding high fuel temperatures for conservatism. New analyses use bounding high values for the fuel designs currently in use.

20. Section 3.3.1.1, p. 3-10, Main Feedwater Line requires revision to update the modeling when the flow is realigned to the upper auxiliary header following trip of the reactor coolant pumps. Also, the subsection title "Main Feedwater Line" is revised to "Main Feedwater System". (model revision)

    Change: "Main Feedwater Line" to "Main Feedwater System"

    Insert the following at the bottom: "The Main Feedwater System is realigned to feed through the upper auxiliary feedwater header following the trip of all reactor coolant pumps. Flow through this alignment is restricted to much lower flowrates to prevent exceeding the steam generator tube flow-induced vibration limit."

    Technical Justification: The Main Feedwater System is realigned by the Integrated Control System to feed through the upper auxiliary header whenever all four reactor coolant pumps are tripped. This realignment is intended to promote the natural circulation mode of heat transfer. Flow through the upper header is restricted to not exceed the steam generator tube flow-induced vibration limit. This results in less feedwater flow following the realignment. This modeling approach is consistent with the plant design.

21. Section 3.3.1.2, p. 3-12, Borated Water Storage Tank (BWST) Level and Temperature, the last sentence in the first paragraph is deleted since it is no longer correct. (editorial)

    Delete: "With a maximum BWST level uncertainty of 20.2 inches, a minimum initial level of 44.3 feet is assumed."

    Technical Justification: The technical specifications surveillance on BWST level has been revised so that the 46 feet level requirement must be exceeded by the instrument uncertainty. Therefore, the minimum actual level is 46 feet, and that is what is now assumed in the analysis.

22. Section 3.3.1.2, p. 3-13, Steam Generator Level, the 10.5% value for level instrument error is deleted. (editorial)

    Delete: "(10.5%)"

    Technical Justification: The 10.5% steam generator level instrument error is an input to the analysis. Specifying a value is not required in the topical report. A bounding instrument error value is assumed in the analyses.

23. Section 3.3.2.1, p. 3-15, Fission Heat, clarify that ECCS boron concentrations are actually in the COLR, and delete a sentence that has extraneous information on BWST boron concentrations. (editorial)

    Change: "The minimum ECCS boron concentration required by Technical Specifications ensures . . . "

    To: "The minimum ECCS boron concentration required by technical specifications and specified in the COLR ensures . . . "

    Delete: A minimum BWST boron concentration of 1950 ppm is required by the COLR as referenced by Technical Specifications (Note: Higher boron concentrations than 1950 ppm exist in current COLRs).

    Technical Justification: These changes are editorial in nature and simply clean up the text by deleting extraneous content and making clarifications.

24. Section 3.3.2.1, p. 3-18, first paragraph, Metal-Water Reaction Rate, the methodology has been changed to assumed a whole core oxidation rate of 1.0% rather than an analysis-specific result. (model revision)

    Change: "0.557% which is presented in Reference 3-6, Section 8. This fraction corresponds to a total energy release of 3.134×10^6 Btu."

    To: "1.0%, consistent with the regulatory limit."

    Technical Justification: The original text states a metal-water reaction rate of 0.557% based on the then-current LOCA analysis of record. The methodology has been revised to assume a value of 1.0%, which is the regulatory limit. This simplifies the methodology and avoids the possibility of an invalid input to the methodology in the future. The sentence also give the reference for the 0.557% value and a conversion to Btu units. Neither of these are needed with the new approach and they are deleted.

25. Section 3.3.2.2, p. 3-20, ECCS Injection, second paragraph, the modeling of LPI flow is revised to be consistent with current design and operating procedures (plant design change)

    Change: " Injection flow will be available from two HPI pumps and one LPI pump following the failure of a 4160V switchgear. The injected HPI flow is represented as a function of RCS back pressure. LPI flow is assumed to be constant at 3000 gpm based upon throttling guidance in the Emergency Operating Procedure. This value is biased by +311 gpm to account for instrument uncertainties. Instrument error is added to the assumed LPI flow since the greater flow will increase the core reflood rate and release energy to the containment sooner than a lesser flow would."

    To: "Injection flow will be available from two HPI pumps and one LPI pump following the failure of a 4160V switchgear. The injected HPI flow is represented as a function of RCS back pressure. LPI flow is assumed to be a function of RCS backpressure until operator action is assumed based upon throttling guidance in the emergency operating procedure. The throttled flow value is biased to account for instrument uncertainty."

Technical Justification: Throttling of LPI flow as directed by the Oconee Emergency Operating Procedure has changed since the original methodology was submitted. The above change reflects the current procedure guidance. These changes in the methodology are necessary to maintain consistency between the plant design and procedures and the UFSAR analyses.

26. Section 3.3.2.2, p. 3-21, Reactor Coolant Pumps, the two-phase head degradation model has been changed from the RELAP5 Semiscale pump model, [ ] model. (model revision)

    Change: " The RELAP5 two-phase pump head degradation model is used to model RCP performance in a saturated system. Two-phase head degradation characteristics are assumed to be similar to the Semiscale pump."

    To:

    [ ]

    Technical Justification:

    [ ]

27. Section 3.3.3.1, p. 3-22, Fission Heat, the fission power history from the B&W CRAFT2 LOCA analysis is no longer used. Also, the Metal-Water Reaction modeling is now the same as in the short-term analysis methodology. The methodology of Section 3.3.2.1 is used instead. (model revision)

    Delete: All text in Section 3.3.3.1.

    Insert: "Refer to Section 3.3.2.1."

    Technical Justification: The original long-term LOCA mass and energy release methodology used fission heat results from a B&W CRAFT2 LOCA simulation for Oconee as an input rather than actually calculating the post-LOCA core power response due to changes in reactivity during the event. In Revision 1 of DPC-NE-3003 the CRAFT2 results are no longer used. The RELAP5 modeling described in Section 3.3.2.1 is used instead, since it will reflect the initial and boundary conditions, as well as the transient prediction of the RELAP5 code. Also, the metal-water reaction of 1.0% as described in Section 3.3.2.1 is also used. Rather than repeat the text, a reference to Section 3.3.2.1 suffices. This is an improvement in the methodology.

28. Section 3.3.3.2, p. 3-23, Assumptions, is revised to include discussion of hot leg break locations. (model revision)

    Change: " The long-term large break containment analysis considers only a single break size and location: a double-ended guillotine break located at the A1 cold leg pump discharge."

    To: "The long-term break containment analysis considers both cold leg pump discharge and hot leg breaks. "

    Technical Justification: Additional analysis experience since the submittal of the original topical report has shown a need to perform hot leg break analyses for other design purposes. The methodology is revised to include hot leg breaks in addition to cold leg breaks.

29. Section 3.3.3.2, p. 3-24, Break Location, is revised to include discussion of hot leg break locations. (model revision)

    Change: "As described previously, the break size and location chosen for this analysis is a double-ended guillotine break (8.55 ft^2) located at the A1 cold leg pump discharge. The basis for choosing a cold leg break as opposed to a hot leg break is obvious once the characteristics of each break are considered."

    To: "The break size and location is dependent upon the analysis result of interest. For the limiting containment EQ response, the limiting break location is a double-ended guillotine break (8.55 ft^2) at the reactor vessel outlet nozzle. For the maximum containment sump water temperature analysis, the limiting break location is a double-ended guillotine break (14.1 ft^2) located at the reactor vessel outlet nozzle."

    Change: "Naturally, when the break site is flooded, decay heat will be transferred to containment in the liquid phase resulting in a less severe containment response."

    To: "Naturally, when the break site is flooded, decay heat will be transferred to containment in the liquid phase resulting in a less severe containment pressure response, but a higher sump water temperature."

    Technical Justification: These revisions are necessary to expand the discussion to include hot leg breaks and explain associated LOCA phenomena.

30. Section 3.3.3.2, p. 3-24, BWST Depletion - ECCS and Containment System Flow Rates, is revised to add information on the sensitivity of the results to BWST depletion rates. (model revision)

    Change: " Assumptions which deplete the BWST and cause an early switch to sump recirculation are therefore conservative with respect to long term mass and energy releases. BFLOW analyses substantiate this presumption. The break steam flow rate calculated by BFLOW is much more sensitive to ECCS subcooling than it is ECCS flow rate. Consequently, depleting the BWST rapidly more than compensates for the slight

reduction in break steaming rate which will occur with the higher ECCS flow rates assumed to deplete the BWST.

The objective of this boundary condition is to then deplete the BWST as quickly as possible to maximize the time that sump recirculation is aligned. This is accomplished by assuming conservatively high flow rates "

To: "BFLOW analyses substantiate that the break steam flow rate is much more sensitive to ECCS subcooling than it is ECCS flowrate. Consequently, depleting the BWST rapidly more than compensates for the slight reduction in break steaming rate which will occur with the higher ECCS flow rates assumed to deplete the BWST. However, higher ECCS flow rates tend to enhance LPI cooler performance. These considerations are included in the selection of the ECCS boundary conditions."

The objective of this boundary condition is to then deplete the BWST to determine the time that sump recirculation is aligned. This is accomplished by assuming conservatively high or low flow rates "

Technical Justification: These revisions are necessary to expand the text to include hot leg break locations and the maximum sump temperature analysis, which can involve different assumptions regarding the BWST depletion rate in order to ensure a conservative analysis.

31. Section 3.3.3.2, p. 3-26, BWST Depletion - ECCS and Containment System Flow Rates, is revised to include methodology changes to reflect the current station design and procedures. (model revision)

Change: " The assumed single failure of a 4160V switchgear will eliminate all but one LPI and one RBS pump. The constant LPI flow assumed is identical to the peak pressure case: 3000 gpm +311 gpm to account for instrument errors. RBS flow is modeled in this analysis to calculate the BWST depletion rate. The Emergency Operating Procedure instructs the operator to limit RBS flow to ≤1500 gpm per train. When the post-accident uncertainty associated with the spray instrument is considered, it is possible for the spray pump to deliver 1643 gpm while the instrumentation reads 1500 gpm. The RBS System is assumed to deplete the BWST at a constant rate of 1643 gpm. It should be noted that an inconsistency is introduced here between the spray flow assumed in the BWST depletion calculation and the flow assumed in the FATHOMS analysis. Instrument errors are subtracted from 1500 gpm to yield the assumed spray flow in the FATHOMS analysis because this is more conservative with respect to the long-term containment response. This 286 gpm discrepancy is accounted for in the FATHOMS analysis by assuming that the fluid is spilled directly to the sump in order to conserve mass inventory in containment.

The switch to sump recirculation will occur at a nominal BWST level of 6 feet. An uncertainty is 20.2 inches is associated with the BWST level indication. Although this error is applied in the negative sense when determining the initial BWST level (see Section 3.3.1.2, BWST Level and Temperature), it is applied in the positive sense when determining the setpoint to align sump recirculation. When the assumed setpoint of 7.68 feet is reached in the analysis, the switch to sump recirculation is assumed to occur instantaneously (i.e. no delay for valve stroke times and operator actions)."

To: "The assumed failure of a 4160V switchgear will eliminate all but one LPI and one RBS pump. LPI flow is assumed to be a function of RCS backpressure until operator action is assumed based upon throttling guidance in the Emergency Operating Procedure. The throttled flow value is biased to account for instrument uncertainties. RBS flow is modeled in this analysis to calculate the BWST depletion rate. The RBS System is assumed to deplete the BWST at a constant rate using a conservative flow rate. It should be noted that consistent assumptions are made between the spray flow assumed in the BWST depletion calculation and the flow assumed in the FATHOMS analysis. BWST water that is assumed to be spilled directly to the sump is accounted for in the FATHOMS analysis in order to conserve mass inventory in containment.

The emergency operating procedure instructs the operator to switch to sump recirculation at a nominal BWST level of 6 feet. An uncertainty adjustment associated with the BWST level indication is made to conservatively minimize the amount of available BWST inventory. Although this uncertainty adjustment is applied in the negative sense when determining the initial BWST level (see Section 3.3.1.2, BWST Level and Temperature), it is applied in the positive sense when determining the setpoint to align sump recirculation. When the assumed setpoint is reached in the analysis, the switch to sump recirculation is assumed to occur instantaneously (i.e. no delay for valve stroke times and operator actions)."

Technical Justification: Since the original methodology was submitted the station design and procedures have been revised. This has necessitated revising the methodology for the depletion of the BWST and the associated operator actions and setpoints. The revisions stated above integrate the original methodology that remains valid and the new aspects of the methodology that reflect the changes at the plant. In addition, the revised methodology is focused on maintaining consistency between the mass and energy release elements and the containment response elements of the methodology.

32. Section 3.3.3.4, p. 3-31, Overview, the modeling transition no longer occurs at 30 minutes but rather at the end of the RELAP5 analysis. (model revision)

Change: "Because RELAP5 is not used to predict the mass and energy release beyond 30 minutes, a conservative method must be derived to release this stored energy during the FATHOMS analysis."

To: "At the end of the RELAP5 analysis a conservative method must be derived to release this stored energy during the FATHOMS analysis."

Technical Justification: With further experience in applying the methodology it has become apparent that the appropriate time for terminating the RELAP5 phase of the analysis is at the transition from the BWST injection phase to the sump recirculation phase, not necessarily at 30 minutes. At that point in the LOCA scenario the system has evolved into a nearly steady-state boiling pot mode of core cooling, and the transition to the use of the BFLOW code for the continuation of the mass and energy release calculation can be made. The transition to the sump recirculation phase also involves a step change in the ECCS suction temperature. Making the code transition at this point in time is consistent with a transition in the cooling mode.

33. Section 3.7: Two references deleted and one new reference added. (editorial)

Reference 3-8 - Deleted

Reference 3-9 - Deleted

Reference 3-10 BWNT LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants, BAW-10192-PA, Framatome Advance Nuclear Products, Revision 0, June 1998

Chapter 4

34. Section 4.1, p. 4-2, Analysis Overview, second paragraph, the containment backpressure is now modeled as a boundary condition rather than with a simple containment model within the RELAP5 model. (model revision)

Change: " A model of the Reactor Building is added to each small break analysis to provide a means for estimating these boundary conditions."

To: "Containment pressure and sump water temperature are input to RELAP5 as boundary conditions based on FATHOMS analysis results."

Technical Justification: For small break LOCA mass and energy release analyses the simulation can last many hours. Consequently the containment backpressure on the break and the containment sump water temperature are important boundary conditions. In particular the containment sump water temperature is much hotter than the BWST temperature, and has a direct impact on the mass and energy release. In the original methodology a simple containment model was develop within the RELAP5 model. This was only an approximate approach but was suitable for the intended purpose. In Revision 1 this approach is replaced by using boundary conditions (pressure and sump water temperature) based on results from FATHOMS analyses. The outputs from FATHOMS analyses for similar break sizes are used, with attention to ensuring applicability to the RELAP5 case.

35. Section 4.1, Long-Term Mass and Energy Release, first paragraph on p. 3, is revised to update the description of the modeling of the stored energy during the transition from RELAP5 to FATHOMS. (model revision)

Change: "The initial fluid conditions in each node is determined by averaging fluid conditions in the corresponding RELAP5 nodes. The RELAP5 heat structures are also collapsed into three heat slabs consistent with the simplified node boundaries. The average temperature in each heat slab is similarly determined by averaging the stored energy in all corresponding RELAP5 heat structures. "

To: "The initial fluid conditions in each node are determined by averaging fluid conditions in the corresponding RELAP5 nodes. The stored energy in the RELAP5 primary system heat structures are modeled by three FATHOMS heater components consistent with the simplified node boundaries. The stored energy in the RELAP5 secondary fluid and secondary heat structures are modeled by two additional FATHOMS heater components. The FATHOMS heater component models the stored energy in the

RELAP5 components as an average heat flux for a specified time period boundary condition."

Technical Justification: The transition from RELAP5 to FATHOMS for the small break LOCA mass and energy release modeling requires consideration of the stored energy at the time of the transition. This modeling has been revised as follows in Revision 1. The stored energy in the secondary system fluid and structural metal is now included in Revision 1. This is done for completeness in modeling all possible energy sources. All of the stored energy modeling now uses the FATHOMS heater component model. The various stored energy sources in RELAP5 are converted into average heat flux for a specified time period boundary conditions. This is an improvement and a simplification relative to the original methodology that modeled the stored energy as heat slabs and required an appropriate heat transfer coefficient to be modeled.

36. Section 4.3.1, Fission Heat, p. 4-3, is replaced in its entirety to use simpler modeling that has the same result. (model revision)

"Fission Heat

The RELAP5 point kinetics model is used to calculate delayed neutron power as a function of time. Upon reactor trip a negative reactivity insertion equal to the technical specification minimum shutdown margin, currently 1.0% $\Delta k/k$, using a normalized rod worth vs. position curve is modeled. The reactor is assumed to trip on the first RPS setpoint encountered. This setpoint is generally the variable low RCS pressure trip which is modeled with the equation given in the technical specifications. Instrument uncertainties are also applied to this equation. The rod insertion rate is based upon Technical Specification values. For conservatism, a delay is added so that the rods begin to fall 0.7 seconds after the trip signal is received. No thermal feedback is modeled. This approach is consistent with the core reload design process which ensures that the minimum shutdown margin is maintained post-trip for LOCA-type events. This is a conservative model since greater than the minimum shutdown margin always exists."

Technical Justification: Both the original and the Revision 1 methodology conservatively limit the negative reactivity insertion on reactor trip to the technical specification shutdown margin. In the original methodology thermal feedback was calculated, but then the total reactivity was limited to the minimum shutdown margin. In effect, modeling the thermal feedback was not actually contributing to the analysis result. In Revision 1 the thermal feedback has been deleted.

37. Section 4.3.1, Total Rod Worth, p. 4-4, is deleted in its entirety and replaced with text to indicate that the total rod worth is equal to the technical specification minimum shutdown margin. (model revision)

"Total Rod Worth

The total rod worth is set to the technical specification minimum shutdown margin, currently 1.0% $\Delta k/k$ converted to dollar units by diving by the value for beta effective."

Technical Justification: The simplification of the modeling of the reactivity insertion following reactor trip as described in the revision to "Fission Heat" described above, also simplifies the total rod worth assumed in the analysis. Since no thermal feedback is

modeled, the total rod worth is simply set to the technical specification shutdown margin value, currently 1.0%Δk/k. This is converted to dollar units by dividing by the value for beta effective, the delayed neutron fraction. This modeling revision is only a simplification relative to the original methodology and does not change the results.

38. Section 4.3.1, p. 4-4, add new subsection "Fission Products and Actinides Decay" at the bottom of Section 4.3.1 to include a description of the decay heat modeling. This text was inadvertently not included in the original methodology, and is unchanged in Revision 1. The text from Section 3.3.1.1 on p. 3-10 is used without revision. (editorial)

    "Fission Product and Actinides Decay

    The heat produced from the radioactive decay of fission products and actinides is calculated for end-of-cycle conditions. This calculation determines decay heat as a function of time after trip for Oconee Units 1, 2, and 3 using the ANSI/ANS-5.1-1979 standard (Reference 3-5). For additional conservatism, uncertainties (2σ) are calculated and added to the mean decay heat values."

39. Section 4.3.2, p. 4-6, Reactor Building Model, is deleted in its entirety and replaced with a new subsection titled "Containment Boundary Conditions", since this model is no longer used in the Revision 1 methodology. (model revision)

    Delete: Section 4.3.2, Subsection "Reactor Building Model" in its entirety.

    Insert: New Subsection "Containment Boundary Conditions":

    "Containment Boundary Conditions

    Results from FATHOMS analyses are used as inputs to the RELAP5 analyses to model two containment boundary conditions. One boundary condition is the containment backpressure that the LOCA break flow discharges into. The second boundary condition is the containment sump water temperature used for the ECCS suction supply after the transition to the sump recirculation mode. These are selected based on FATHOMS cases that are consistent with the RELAP5 analysis in terms of break size and intended application."

    Technical Justification: Refer to the above Section 4.1, Analysis Overview, p. 4-2 revision technical justification for this related revision.

40. Section 4.3.2, pp. 4-7, 8, and 9, BWST Depletion - ECCS and Containment System Flow Rates, is revised in three places to no longer use RELAP5 for modeling the containment boundary conditions. FATHOMS results are used instead. (model revision)

    Change: "As discussed above, one of the primary purposes of the RELAP5 containment model is to provide an estimate of when this system is activated."

    To: "Containment analysis inputs required for the RELAP5 analysis are obtained from FATHOMS analysis results."

    Change: "The spilled HPI flow is accounted for in the RELAP5 containment model and the FATHOMS analysis as a flow boundary condition."

To: "The spilled HPI flow is modeled in the RELAP5 analysis and is used in the FATHOMS analysis as a flow boundary condition."

Change: "The RBS System is assumed to initiate when pressure in the RELAP5 containment model reaches 20 psig."

To: "The RBS System is assumed to initiate consistent with the FATHOMS results."

Technical Justification: Refer to the above Section 4.1, Analysis Overview, p. 4-2 revision technical justification for this related revision.

41. Section 4.3.2, p. 4-9, Steam Generator Level, is revised in the modeling of EFW actuation and control for consistency with the current station procedures and the UFSAR Chapter 15 SBLOCA peak cladding temperature analysis. (model revision)

    Change: "However, a ten minute delay from reactor trip is assumed in this analysis. Beyond 10 minutes, steam generator level is controlled at the LSCM setpoint minus instrument errors (56 inches) with the EFW System."

    To: "However, a 20 minute delay from reactor trip is assumed in the methodology. Furthermore, only one motor-driven EFW pump is credited to feed one steam generator to the LSCM setpoint minus an allowance for level instrument uncertainty."

42. Section 4.3.2, p. 4-10, Steam Generator Pressure Control, is revised to change the time at which the atmospheric steam dump valves are credited and the target cooldown rate. This change is necessary to be consistent with current station procedures. (model revision)

    Change: "An operator is assumed to begin manipulating these valves locally beginning 30 minutes after break initiation in an attempt to maintain a target cooldown rate of 80°F/hr, based upon the core exit temperature."

    To: An operator is assumed to begin manipulating these valves locally beginning 60 minutes after break initiation in an attempt to maintain a cooldown rate >50°F/hr, based upon the core exit temperature."

    Technical Justification: Current station procedures initiate the opening of the atmospheric steam dump valves to initiate the post-LOCA cooldown by no later than 60 minutes. The cooldown rate is targeted for the technical specification maximum of 100°F/hr. An actual cooldown rate of no less than 50°F/hr will be achieved. The revised methodology is updated to be consistent with the current station procedures.

43. Section 4.3.2, p. 4-10, Emergency Feedwater System, the modeling is revised to credit only one motor-driven emergency feedwater pump at 20 minutes. This change is necessary to be consistent with current station design and procedures. (model revision)

    Change: " The single failure of a 4160V switchgear will disable one motor-driven emergency feedwater (MDEFW) pump leaving one MDEFW pump and the turbine-driven emergency feedwater (TDEFW) pump. The operator is assumed to trip the one remaining MDEFW pump and control levels with the TDEFW pump feeding both steam generators. The maximum possible EFW flow is limited by the flow capacity of a single

TDEFW pump. A RELAP5 control system is used to control the actual EFW flow based upon the desired steam generator setpoint. The B&W EFW spreading model is used to model the phenomena of EFW spraying through the upper header onto the tubes in the outer periphery of the steam generators (see Section 2.1.1.2)."

To: " The single failure of a 4160V switchgear will disable one motor-driven emergency feedwater (MDEFW) pump, leaving one MDEFW pump and the turbine-driven emergency feedwater (TDEFW) pump available. No credit is taken for the TDEFW pump, leaving one MDEFW pump feeding one steam generator. A RELAP5 control system is used to control the actual EFW flow based upon the desired steam generator setpoint. The B&W EFW spreading model is used to model the phenomena of EFW spraying through the upper header onto the tubes in the outer periphery of the steam generators (see Section 2.1.1.2). Since MFW is conservatively used to feed to the natural circulation level setpoint, EFW is only used to raise the steam generator levels to the loss of subcooled margin setpoint. This is assumed to begin at 20 minutes based on manual operator action consistent with procedures. This is also consistent with the UFSAR Chapter 15 SBLOCA peak cladding temperature analysis."

Technical Justification: The original methodology for modeling EFW actuation and control for SBLOCA mass and energy release has been significantly revised to reflect current station operating procedures and for consistency with the SBLOCA peak cladding temperature analyses in UFSAR Chapter 15. This involves crediting only one MDEFW pump beginning at 20 minutes with flow delivered to only one steam generator. Furthermore, the EFW is only used to raise the steam generator level from the natural circulation setpoint to the loss of subcooled margin setpoint. This manual action is necessary to establish the boiler-condenser mode of heat transfer following SBLOCAs that require primary-to-secondary heat transfer for core cooling and unit cooldown.

44. Section 4.4.2, p. 4-14, third paragraph, third sentence, change "increase" to "decrease". (editorial)

    Change: " This causes the void fraction in the break volume to increase which shifts the break flow from two-phase to single phase liquid."

    To: " This causes the void fraction in the break volume to decrease which shifts the break flow from two-phase to single phase liquid."

45. Section 4.4.4, p. 4-18, first sentence, change "five times larger than" to "250% of". (editorial)

    Change: " The break area assumed in this case is five times larger than the area assumed in the previous case."

    To: " The break area assumed in this case is 250% the area assumed in the previous case."

46. Table 4.4-3, p. 4-25, change "Case 2" and "Case 3" to "Case 5" and "Case 6" in the table headings, respectively. (editorial)

    Change: "Case 2"

    To: "Case 5"

Change: "Case 3"

To: "Case 6"

Chapter 5

47. Section 5.1.1, Description of Steam Line Break Accident, p. 5-1, change "UFSAR Section 15.13" to "UFSAR Section 6.2 (Reference 5-1)" for consistency with the current UFSAR section for this content. (editorial)

    Change: "UFSAR Section 15.13"

    To: "UFSAR Section 6.2 (Reference 5-1)"

48. Section 5.1.2, Acceptance Criteria, p. 5-1, revise the text to simplify the acceptance criteria discussion so that it is applicable for the steam line break mass and energy release and containment response analysis. (editorial)

    Change: "The criteria for unit protection and the release of fission products to the environment for the steam line break accident as listed in UFSAR Section 15.13 (Reference 5-1) are as follows:

    1. The core will remain intact for effective core cooling, assuming minimum tripped rod worth with a stuck rod.
    2. No steam generator tube loss of primary boundary integrity will occur due to the loss of secondary side pressure and resultant temperature gradients.
    3. Doses will be within 10CFR100 limits.

    The stated UFSAR acceptance criteria for steam line break do not include containment design pressure as an acceptance criterion. However, a containment pressure response analysis was performed and is presented in the UFSAR with a comparison of the results to the design pressure limit. In addition, IE Bulletin 80-04 (Reference 5-2) requested a review of the containment pressure response analysis to determine the impact of runout flow from the Emergency Feedwater System and the impact of other energy sources, such as continuation of feedwater or condensate flow. Thus, containment pressure must remain below the design limit of 59 psig during a steam line break accident."
    "

    To: "The acceptance criteria for the steam line break mass and energy release and containment response analyses are that the containment pressure will not exceed the design pressure of 59 psig, and that safety-related equipment inside containment will survive the harsh environment resulting from the transient (equipment qualification)."

49. Section 5.4, p. 5-8, Main Steam Lines, change "Technical Specification 4.8" to "Technical Specification 3.7.2" for consistency with the current technical specifications. (editorial)

    Change: " Technical Specification 4.8"

To: " Technical Specification 3.7.2"

50. Section 5.4, p. 5-8, Main Feedwater Lines, change subsection title to "Main Feedwater System" to more accurately reflect the content. (editorial)

    Change: " Main Feedwater Lines"

    To: "Main Feedwater System"

51. Section 5.5 , p. 5-14, Offsite Power Maintained, second paragraph, change "affected loop saturates" to "unaffected loop saturates" to correct an error. (editorial)

    Change: "Reactor coolant pressure (Figure 5.5-1) rapidly decreases after the steam line break occurs until the affected loop saturates at about 10 seconds."

    To: " Reactor coolant pressure (Figure 5.5-1) rapidly decreases after the steam line break occurs until the unaffected loop saturates at about 10 seconds "

52. Section 5.6, References: One reference deleted and two references revised. (editorial)

    Reference 5-2 - Deleted

    Reference 5-5 - Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-P, Revision 3, Duke Power Company, May 2002

    Reference 5-7 - Oconee Nuclear Station Technical Specifications

    To: " Technical Specification 3.7.2"

Chapter 6

53. Section 6.1 , p. 6-1, Overview, second paragraph, the methodology is revised due to the large break LOCA and small break LOCA analyses being divided into two time segments instead of three. (model revision)

    Change: "For the long-term large break LOCA, three segments are used to achieve the best accuracy in the [ ] code. The small-break LOCA simulations are also divided into three segments . . ."

    To: "For the long-term large break LOCA, two segments are used to achieve the best accuracy in the [ ] code. The small-break LOCA simulations are also divided into two segments . . ."

    Technical Justification: The FATHOMS modeling is divided into time segments with common post-LOCA plant operating configurations, and therefore can be modeled with the same boundary conditions. Based on current modeling, the long-term large break LOCA can be modeled with two time segments instead of three time segments as stated

in the original methodology. The small-break LOCA can now be modeled with two time segments instead of three also.

54. Section 6.2.3, Droplet Size of Blowdown Mass, p. 6-4, is revised in Revision 1 to change the droplet size to 100 microns and a new reference is provided. (model revision)

    Change: "A 20 μm value for the size of the blowdown droplets is used. Test data cited in Reference 6-3 suggests that the droplet sizes for high-pressure blowdowns range from 2 - 2000 μm, and Reference 6-4 suggests that 20 μm is a suitable average value to use with the FATHOMS code."

    To: "A 100 μm value for the size of the blowdown droplets is used in the Revision 1 methodology. Test data cited in Reference 6-3 suggests that the droplet sizes for high-pressure blowdowns range from 2 - 2000 μm, and Reference 6-8, suggests that 100 μm is a suitable average value to use with the FATHOMS code. A value of 20 μm value was used in the Revision 0 methodology based on Reference 6-4."

    Technical Justification: Reference 6-8 recommends use of a droplet size of 100 μm in the FATHOMS code. This change is made to update the methodology to a more current value.

55. Section 6.3.1, Overview, p. 6-11, top paragraph, the RELAP5 analysis now runs through to the end of the injection phase instead of stopping at 1800 seconds. (model revision)

    Change: ". . . only lasts for 1800 seconds, the BFLOW code is used to determine the quality of the flow exiting the break following the first 1800 seconds."

    To: ". . . only lasts until the start of the recirculation mode, the BFLOW code is used to determine the quality of the flow exiting the break thereafter."

    Technical Justification: In the Revision 0 methodology the RELAP5 analysis was stopped at 1800 seconds. In the Revision 1 methodology it was decided that a more appropriate time for terminating the RELAP5 analysis was at the end of the injection mode (beginning of the recirculation mode). This choice is a more natural time for transitioning from RELAP5 to the BFLOW code, since there is a step change in the ECCS suction water temperature at that time.

56. Section 6.3.2, Analytical Approach, p. 6-12, top paragraph, the RELAP5 analysis now runs through to the end of the injection phase instead of stopping at 1800 seconds. Also, there are two time segments instead of three. (model revision)

    Change: "Three different sets of flowpaths (junctions) are used in the analysis to implement the mass and energy release data supplied from the RELAP5 analysis and the data from the BFLOW code. Each set of flowpaths is active during one time segment of the analysis. The first segment ends at 1800 seconds, which is the end of the mass and energy release data from the RELAP5 analysis. The second segment lasts from 1800 seconds to 2808 seconds, which is the switchover point to sump recirculation mode. The third segment lasts from 2808 seconds to the end of the analysis, at 15 days. These segments will be designated as Segments 1, 2, and 3."

To: " Two different sets of flowpaths (junctions) are used in the analysis to implement the mass and energy release data supplied from the RELAP5 analysis and the data from the BFLOW code. Each set of flowpaths is active during one time segment of the analysis. The first segment ends at the end of the injection mode, which is the end of the mass and energy release data from the RELAP5 analysis. The second segment lasts from the start of the recirculation mode to the end of the analysis. These segments will be designated as Segments 1 and 2."

Technical Justification: In the Revision 0 methodology the RELAP5 analysis was stopped at 1800 seconds. In the Revision 1 methodology it was decided that a more appropriate time for terminating the RELAP5 analysis was at the end of the injection mode (beginning of the recirculation mode). This choice is a more natural time for transitioning from RELAP5 to the BFLOW code, since there is a step change in the ECCS suction water temperature at that time.

57. Section 6.3.3 , p. 6-14, BWST Volume and Temperature, the volume modeling is revised due to the large break LOCA analysis being divided into two time segments instead of three. The Revision 1 methodology is inserted in front of the Revision 0 methodology, and the Revision 0 text is only retained for historical purposes. (model revision)

    Insert the following new text at the beginning of this subsection:

    "During Segment 1, the ECCS injection flow coming from the BWST and the flow exiting the break are modeled in the RELAP5 analysis. After the switchover to sump recirculation mode (during Segment 2), the source for the ECCS water will be the sump, so the BWST is not modeled.

    Note: The following two paragraphs are superceded due to being applicable only to Revision 0 of the report. They are retained for comprehension of the Revision 0 results only."

    Technical Justification: The FATHOMS modeling is divided into time segments with common post-LOCA plant operating configurations, and therefore each time segment can be modeled with the same boundary conditions. Based on current experience, the long-term large break LOCA can be modeled with two time segments instead of three time segments as stated in the original methodology. The original methodology is retained only so that the results presented can be understood.

58. Section 6.3.3 , p. 6-15, Heat Transfer Correlation, first paragraph, delete "1800 seconds". Also, in the second paragraph, the number of time segments is changed from 3 to 2. (model revision)

    Change: ". . . start of Segment 2 (1800 seconds) . . . "

    To: ". . . start of Segment 2 . . ."

    Change: "During Segments 2 and 3, the sump region is over . . . "

    To: "During Segment 2 the sump region is over . . . "

Technical Justification: The FATHOMS modeling is divided into time segments with common post-LOCA plant operating configurations, and therefore can be modeled with the same boundary conditions. Based on current modeling, the long-term large break LOCA can be modeled with two time segments instead of three time segments as stated in the original methodology. In the Revision 1 methodology the end of Segment 1 is at the end of the injection mode, rather than at 1800 seconds. This choice is a more natural time for transitioning from RELAP5 to the BFLOW code, since there is a step change in the ECCS suction water temperature at that time.

59. Section 6.3.3 , p. 6-15, Group 2 and 3 Heat Structures, second paragraph, the number of time segments is changed from 3 to 2. (model revision)

    Change: "A constant heat transfer coefficient is assumed for these heat structures during Segments 2 and 3."

    To: " A constant heat transfer coefficient is assumed for these heat structures during Segment 2."

    Technical Justification: The FATHOMS modeling is divided into time segments with common post-LOCA plant operating configurations, and therefore can be modeled with the same boundary conditions. Based on current modeling, the long-term large break LOCA can be modeled with two time segments instead of three time segments as stated in the original methodology.

60. Section 6.3.3, Droplet Size of Blowdown Mass, p. 6-16, is revised in Revision 1 to change the droplet size to 100 microns. (model revision)

    Change: "A 20 μm value for the size of the blowdown droplets is again used."

    To: "A 100 μm value for the size of the blowdown droplets is again used."

    Technical Justification: Reference 6-8 recommends use of a droplet size of 100 μm in the FATHOMS code. This change is made to update the methodology to a more current value.

61. Section 6.3.3, Modifications to Base Containment Model, p. 6-16, insert a new subsection titled "LPI Flowpaths and Mixing Volume" prior to subsection "LPI Cooler Data". (model revision)

    Insert the following new text:

    "<u>LPI Flowpaths and Mixing Volume</u> The LPI System has two trains with two separate LPI coolers. For some failure scenarios it is possible for only one cooler to be supplied with cooling water, although both will have LPI flow. To model this asymmetric configuration, two LPI flowpaths from the sump are modeled with separate LPI coolers. Downstream of the LPI coolers the two flows are then combined in a mixing volume with one outlet flowpath."

    Technical Justification: The Revision 0 model combined both LPI trains into one equivalent train for modeling purposes. This was thought to be an adequate model for the

intended applications. Subsequently, it was determined that some failure scenarios existed for which cooling water would not be supplied to one of the two LPI coolers. However, both LPI trains would still have LPI flow. This would result in one train of cooled LPI flow, and one train of LPI pumping sump water back into the reactor vessel without cooling. This is accommodated in the Revision 1 model by separately modeling both LPI trains from the sump to the LPI cooler outlet, and then combining the flows exiting the LPI coolers into a mixing volume. This mixing volume is non-physical since the two trains are not necessarily headered in the actual valve alignment. However, this modeling approach is functionally correct for the intended simulation purposes and is used due to the modeling of the long-term decay heat in the FATHOMS model.

62. Section 6.3.5, Boundary Conditions, p. 6-19, a new table of junctions and boundary conditions is inserted for Revision 1, and the existing table is noted as Revision 0. This table is only a summary of revisions explained in other items. (editorial)

Insert the following new table and text prior to the existing table:

1.	Cold leg break - side #1 (vap)	Seg. 1	Injection mode
2.	Cold leg break - side #1 (liq)	Seg. 1	Injection mode
3.	Cold leg break - side #2 (vap)	Seg. 1	Injection mode
4.	Cold leg break - side #2 (liq)	Seg. 1	Injection mode
10.	Sump drain - to LPI cooler #1	Seg. 2	Recirculation mode
11.	LPI cooler #1 - to mixing volume	Seg. 2	Recirculation mode
12.	Sump drain - to LPI cooler #2	Seg. 2	Recirculation mode
13.	LPI cooler #2 - to mixing volume	Seg. 2	Recirculation mode
14.	Sump drain - to Building Spray	Seg. 2	Recirculation mode
15.	Building Spray flow - from sump	Seg. 2	Recirculation mode
16.	Building Spray flow - initial	Seg. 1	Injection mode
17.	Nitrogen injection from CFTs	Seg. 1	Injection mode
19.	From mixing volume - to core	Seg. 2	Recirculation mode
20.	From core - to containment (vapor phase)	Seg. 2	Recirculation mode
21.	From core - to containment (liquid phase)	Seg. 2	Recirculation mode

"Note: The following table is applicable to the Revision 0 FATHOMS model only, and is included so that the results of the Revision 0 analysis can be understood."

63. Section 6.3.5.1, LPI/BS Discrepancy Flowpath (18), p. 6-21, can be deleted since it is no longer part of the methodology. (model revision)

Delete: Subsection "LPI/BS Discrepancy Flowpath"

Technical Justification: The LPI/BS discrepancy flowpath was part of the original methodology based on the transition from RELAP5 to FATHOMS. This approach to address an inconsistency is no longer applicable or necessary with Revision 1. It is therefore deleted.

64. Section 6.3.5.2, Segment 2 Flow Paths, pp. 6-21 to 6-23 can be deleted since Segment 2 is no longer part of the methodology. Section 6.3.5.3 will be resequenced. (model revision)

    Delete: The text of Section 6.3.5.2 "Segment 2 Flow Paths" except for the title.

    Technical Justification: Segment 2 was part of the original methodology but is not part of the Revision 1 methodology. Therefore the text can be deleted. The title is retained with new text from the subsequent section (see below).

65. Section 6.3.5.3 (actually 6.5.4.3 in the report due to an error), Segment 3 Flow Paths, p. 6-23, delete the section title and move the text to Section 6.3.5.2 to reflect the deletion of the old Segment 2. The dual LPI flowpaths and the mixing volume are added. Text discussing the Reactor Building Spray System is no longer applicable and is deleted. (model revision)

    Delete: The title of Section 6.3.5.3 (actually 6.5.4.3 in the report due to an error)

    Change: "<u>LPI Flowpaths (10-13)</u> At 2808 seconds, the recirculation phase is initiated. Water is taken from the sump drain through boundary condition 10, cooled by the LPI coolers, and then injected into the core. As during Segment 2, the quality matrices from the BFLOW analysis are used to determine the quality of the flow exiting the break. This exit flow is again separated by phase through separate boundary conditions (11 and 12). Also as in Segment 2, the LPI flow rate is reduced by the instrument uncertainty value for conservatism. Boundary condition 13, which takes water from the LPI cooler outlet to Building Spray, is not used; all Building Spray comes directly from the sump during recirculation for this analysis. However, the option of using the LPI coolers for cooling Building Spray water is still available at Oconee."

    To: "LPI Flowpaths (10-13, 19-21) At the start of the recirculation phase water is taken from the sump drain through boundary conditions 10 and 12, and then cooled by the two LPI coolers. Downstream of the LPI coolers the two LPI flows are then combined in a mixing volume via boundary conditions 11 and 13, and then the flow is directed to the core via boundary conditions 19-21. The[ ]BFLOW analysis are used to determine the quality of the flow exiting the break. This exit flow is again separated by phase through separate boundary conditions (20 and 21). The LPI flow rate is reduced by the instrument uncertainty value for conservatism."

    Technical Justification: Revision 0 included only one equivalent LPI train, which did not allow for asymmetric LPI train performance. The Revision 1 model has been changed to allow both LPI trains to be modeled from the sump to the LPI cooler outlet. At that point the two trains are combined into a mixing volume before returning the flow to the core. Various flowpaths and boundary conditions are changed to make these modeling revisions. Also, the modeling of cooled spray water via the LPI coolers is now longer an option at Oconee and has been deleted from the model.

66. Section 6.5.4.3 , Building Spray (14-15), p. 6-23, first paragraph, the number of time segments is changed from 3 to 2. (model revision)

Change: "Another set of boundary conditions models the BS flowpaths during Segment 3."

To: "Another set of boundary conditions models the BS flowpaths during Segment 2."

Technical Justification: The FATHOMS modeling is divided into time segments with common post-LOCA plant operating configurations, and therefore can be modeled with the same boundary conditions. Based on current modeling, the long-term large break LOCA can be modeled with two time segments instead of three time segments as stated in the original methodology.

67. Section 6.4.2, Analytical Approach, p. 6-29, first paragraph, deleted the boron dilution line since its use for this purpose has been removed from the methodology due to design issues. (plant design change)

    Delete: "The boron dilution line is also modeled for Cases 3 and 4. As described in Section 2.2.1.3, the boron dilution line allows saturated liquid to exit the reactor vessel to the Reactor Building sump. Whenever this flow path is opened, the size of the RCS break is effectively increased. Taking credit for this flow path from the RCS to the Reactor Building sump allows more flow to exit the RCS, therefore allowing more cool water to be injected into the RCS and aid in the system cooldown."

    Technical Basis: The use of the boron dilution line for this purpose has been removed from station procedures due to design issues.

68. Section 6.4.3, BWST Volume, p. 6-30, is not included in the Revision 1 methodology, since the BWST is no longer modeled in the FATHOMS analysis. Text is inserted in front of the Revision 0 methodology to discuss this change, and the Revision 0 text is only retained for historical purposes. (model revision)

    Insert the following new text at the beginning of this subsection:

    "During Segment 1, the ECCS flow coming from the BWST and the flow exiting the break are modeled in the RELAP5 analysis. After the switchover to sump recirculation mode (during Segment 2), the source for the ECCS injection water will be the sump, so the BWST is not modeled.

    Note: The following paragraph is superceded due to being applicable only to Revision 0 of the report. The text is retained for comprehension of the Revision 0 results only."

    Technical Basis: In the original methodology the RELAP5 analysis would be terminated and a transition to FATHOMS would be made to continue the analysis. This was based on time at 30 minutes, and may be with the ECCS still aligned for injection from the BWST. With the Revision 1 methodology the ECCS will be aligned to the sump at the time of the transition to FATHOMS. Therefore, modeling the BWST in FATHOMS is deleted from the methodology.

69. Section 6.4.3, Junctions Descriptions, p. 6-31, the second paragraph is revised to indicate that opening the boron dilution line for this purpose has been deleted in Revision 1 from the

methodology due to design issues. The existing text is retained for historical purposes.(plant design change)

Insert the following new paragraph before the second paragraph on p. 6-31:

"Note: The boron dilution line is no longer opened in the Revision 1 methodology. The following paragraph regarding Revision 0 results is retained for historical purposes."

Technical Basis: The use of the boron dilution line has been removed from the methodology in Revision 1 due to design issues associated with its use for this purpose. This change is consistent with current station procedures.

70. Section 6.4.3, RCS Heat Structures, p. 6-31, is re-titled "RCS and Secondary Heat Structures" and revisions are made to include the modeling of steam generator secondary side stored energy. (model revision)

Change:

"RCS Heat Structures

[ ]

To:

"RCS and Secondary Heat Structures

[ ]

[ ]

Technical Justification:

[ ]

71. Section 6.4.3, Droplet Size of Blowdown Mass, p. 6-32, is revised in Revision 1 to change the droplet size to 100 microns. (model revision)

    Change: "A 20 μm value for the size of the blowdown droplets is again used."

    To: "A 100 μm value for the size of the blowdown droplets is again used."

    Technical Justification: Reference 6-8 recommends use of a droplet size of 100 μm in the FATHOMS code. This change is made to update the methodology to a more current value.

72. Section 6.4.5, Boundary Condition, p. 6-33, is revised to indicate that the Revision 1 methodology has 14 instead of 17 junctions. A statement is added to identify which three are no longer in the methodology. The Revision 0 text is retained for the three deleted junctions for historical purposes. (model revision)

    Change: "There are 17 junctions in the model, with all but three assigned to a particular boundary condition."

    To: "There are 14 junctions in the Revision 1 model, whereas there were 17 junctions in the Revision 0 model. Junctions 10, 11, and 17 have been deleted from the methodology. The description of these three deleted junctions in the table and text that follows is retained so that the Revision 0 results can be understood." Also, Junction 10,11, and 17 have an asterisk added and the following footnote is inserted at the bottom of the table on p. 6-33. "Note: "*" indicates this junction is deleted in Revision 1".

    Technical Justification: Junctions 10, 11, and 17 have been deleted from the methodology for two reasons. Junctions 10 and 11 have been deleted since the transition from RELAP5 to BFLOW has been shifted until after the alignment for sump recirculation. Therefore these two junctions are no longer applicable. Junction 17 has been deleted since the use of the boron dilution line for this purpose has been removed due to station design issues.

73. Section 6.4.5, HPI Spill (before recirculation) (8), p. 6-34, is revised to delete the second sentence regarding the FATHOMS BWST model, which has been deleted in Revision 1. (model revision)

Delete: " The inventory from the BWST is compensated for in the RELAP5 analyses, so this water is not drawn from the FATHOMS BWST model until the RELAP5 mass and energy release data is exhausted.."

Technical Justification: The BWST is no longer modeled in FATHOMS in the Revision 1 methodology. The remaining text with this revision correctly describes the current methodology.

74. Section 6.6, References: Reference 6-2 corrected and new Reference 6-8 added. (editorial)

Reference 6-2 - NUREG-0588, Interim Staff Position on Environment Qualification of Safety-Related Electrical Equipment, U.S. Nuclear Regulatory Commission, December 1979

Reference 6-8 - "Drop Behavior in GOTHIC," NAI-9301-7, Numerical Application, Inc., January 10, 2001

Chapter 7

75. Chapter 7, p. 7-1 is revised to add the following paragraphs at the end to summarize the content of the supplements and appendices. (editorial)

"Supplement 1 was included in the November 1997 publication of the report to provide the results of an application of the methodology with the Main Steam Line Break Detection and Main Feedwater Isolation System. This system was installed subsequent to the approval of the original topical report, and significantly changes the results by crediting automatic isolation of MFW and partial isolation of EFW. This application of the methodology was provided since it superceded the application in Chapters 5 and 6 of Revision 0.

Supplement 2 was added with the November 1997 version of Revision 0 of the report to present validation analyses comparing the FATHOMS prediction to data from the CVTR facility.

Appendix A was added in Revision 1 to describe the replacement steam generator (ROTSG) steam line break mass and energy release modeling with RETRAN-3D.

Appendix B was added in Revision 1 to address the RETRAN-3D SER conditions and limitations as relates to the replacement steam generator (ROTSG) steam line break mass and energy release modeling with RETRAN-3D.

Appendix C was added in Revision 1 to include the use of the GOTHIC 7.0 code for containment response analysis."

Supplement 1

76. Subsection RETRAN-02 System Analysis, p. S1-4, first paragraph, change "affected loop saturates" to "unaffected loop saturates" to correct an error. (editorial)

    Change: "Reactor coolant pressure (Figure S1-1) rapidly decreases after the steam line break occurs until the affected loop saturates at about 10 seconds."

    To: " Reactor coolant pressure (Figure S1-1) rapidly decreases after the steam line break occurs until the unaffected loop saturates at about 10 seconds "

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July 7, 2003

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ATTENTION: Chief, Information Management Branch
Division of Program Management
Policy Development and Analysis Staff

Subject: Duke Energy Corporation
Oconee Nuclear Station - Units 1, 2, and 3
Docket Nos. 50-269, 50-270, and 50-287

Revisions to Topical Report DPC-NE-3003 In Support of Steam Generator Replacement
Response to NRC Staff Request for Additional Information

Reference: Duke Submittal Dated June 13, 2002

Enclosed herein, please find the Duke Energy Corporation (Duke) response to the May 12, 2003 NRC staff's request for additional information concerning topical report DPC-NE-3003, Revision 1, "Mass and Energy Release and Containment Response Methodology."

Attachment 1 to this letter constitutes Duke's response to questions 1 - 12, and 14 - 17. Duke's response to question 13 will be submitted by July 31, 2003.

During the preparation of the responses, two (2) additional revisions to the methodology were identified as necessary. These new revisions are detailed in Attachment 2. Duke requests that the NRC include these revisions within the scope of review of the original June 13, 2002 submittal.

If there are any questions or if additional information is needed on this matter, please call J. A. Effinger at (704) 382-8688.

Very truly yours,

K. S. Canady

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Attachment 1

Duke Power Response To Request For Additional Information

DPC-NE-3003 Revision 1 "Mass and Energy Release and Containment Response Methodology"

General Questions

1. Discuss the major differences between FATHOMS and GOTHIC 7.0 and the effect these differences will have on the containment analyses.

Response: FATHOMS Version 2.4 was released in April of 1989. Shortly thereafter, FATHOMS became the GOTHIC code under EPRI sponsorship. Since then there have been many versions of GOTHIC released as part of the ongoing GOTHIC Enhancement Project. Each new version of GOTHIC adds new modeling features, user conveniences and improves and extends existing modeling capabilities. The list of differences between FATHOMS 2.4 and GOTHIC 7.0 is extensive but only a few of the differences are of significance for Duke's application for the analysis of the temperature and pressure response in containment for the Oconee Nuclear Station. These differences are described below.

Drop Energy Equation

FATHOMS 2.4 solves one energy equation for the vapor phase (air and steam) and a second energy equation for the combined drop and liquid phases. This means that within each computational control volume the drops and the liquid phases are at the same temperature. One of the first modifications under the GOTHIC project was the addition of a separate energy equation for the drop field. With this improvement, the drops and liquid phases within a computational control volume can be at different temperatures. If drops are present, the combined drop/liquid energy equation in FATHOMS 2.4 generally tends to promote thermal equilibrium in the liquid and vapor phases. The large interface area of the drops effectively extends the interface area for the water pool or films, enhancing the interphase heat and mass transfer.

Drop Deposition

Both FATHOMS 2.4 and GOTHIC 7.0 include models for the drop deposition. However, for the Oconee applications, the FATHOMS model is limited to gravitational settling using a constant drag coefficient appropriate for large drops. This makes the settling rate independent of the drop size. The drop deposition model in GOTHIC 7.0 includes gravitational settling and impaction. Deposition due to impaction is small for this application because the bulk average containment velocity is low. The settling deposition in GOTHIC 7.0 depends on the drop diameter because the drag coefficient is a function of the drop size. Small drops, typical of the size created by a LOCA, will fall slower in GOTHIC 7.0 due to the larger drag coefficient. Therefore, the water retained as drops in the atmosphere will typically be larger with GOTHIC 7.0 than with FATHOMS 2.4. GOTHIC 7.0 also includes a drop agglomeration model whereby smaller drops combine to form larger drops that are subject to higher fall out rates.

Drop Entrainment

Both FATHOMS 2.4 and GOTHIC 7.0 include models for drop entrainment. Typically drop entrainment occurs at high vapor velocity across a liquid surface. The bulk average velocities calculated for Oconee applications are not high enough to result in significant entrainment. However, GOTHIC 7.0 includes another entrainment mechanism that is not in FATHOMS. In GOTHIC 7.0, a portion ($1/6^{th}$) of the condensation rate is assumed to be converted to drops to simulate dripping from ceilings and equipment. The model was incorporated to improve comparison with experimental results from HDR and other large-scale facilities. The drops created by this mechanism are large and quickly fall from the atmosphere but they do provide the potential for increased interface heat and mass transfer.

Interphase Heat and Mass Transfer

In FATHOMS 2.4, the interface heat and mass transfer at the drop and liquid surfaces is calculated using effective heat transfer coefficients for each side of the interface that incorporates mass transfer as appropriate. The particular correlation used for the effective heat transfer coefficients depends on whether the phase is superheated or subcooled relative to the local saturation temperature. The energy balance at the interface used to calculate the rate of phase change is

$$\Gamma \Delta h_{\Gamma} = A_I H_{l,eff}(T_l - T_s) + A_I H_{v,eff}(T_v - T_s)$$

where Γ is the rate of phase change, Δh_{Γ} is the heat of vaporization, A_I is the interface area, $H_{l,eff}$ and $H_{v,eff}$ are the heat transfer coefficients, including phase change effects, for the liquid and vapor sides of the interface respectively. T_l, T_v and T_s are the liquid, vapor and saturation temperatures, respectively.

In GOTHIC 7.0, the interfacial heat and mass transfer are calculated using a heat and mass transfer analogy. The energy balance for the interface is

$$\Gamma \Delta h_{\Gamma} = A_I H_l (T_l - T_I) + A_I H_v (T_v - T_I)$$

where H_l and H_v are the sensible heat transfer coefficients for the liquid and vapor sides of the interface respectively and T_I is the interface temperature. The unknown interface temperature is obtained by simultaneously solving the interface energy balance with the mass transfer equation

$$\Gamma = -A_I H_m (\varphi_v - \varphi_I)$$

and the assumption that the steam pressure at the interface is the saturation pressure at the interface temperature. H_m is the mass transfer coefficient and φ_v and φ_I are the steam concentrations in the vapor phase and at the interface, respectively.

The heat transfer coefficients are from correlations for free and forced convection as appropriate for the local conditions. The mass transfer coefficient is obtained from the heat transfer correlations using a heat and mass transfer analogy. The formulation for the interface heat and mass transfer in FATHOMS tends to keep the phases closer to thermal

equilibrium than the GOTHIC formulation. The interface area is largely dependent on the flow regime that the code determines is appropriate for the given situation. Although there have been changes to the flow regime selection logic in GOTHIC, for this application the interface area would be based on an assumed film regime configuration and should be approximately the same for the two codes.

Observed Differences in Code Predictions

A comparison of the code results from FATHOMS 2.4 and GOTHIC 7.0 for the Oconee cases shows that for LOCA:

1. GOTHIC 7.0 predicts peak pressures that are generally slightly lower (~0.5 psia)
2. GOTHIC 7.0 predicts peak temperatures that are generally slightly lower (~0.5° F)

For MSLB:

1. GOTHIC 7.0 predicts peak pressures that are generally slightly higher (~0.5 psia)
2. GOTHIC 7.0 predicts peak temperatures that are generally lower (~10° F)

For MSLB and LOCA:

1. GOTHIC 7.0 predicts sump water volumes that are generally lower.
2. GOTHIC 7.0 predicts sump water temperatures that are generally lower.

These differences are consistent with the code differences. For the LOCA events both codes predict a saturated containment atmosphere due to the large amount of drops injected. The slightly lower pressure and temperature from GOTHIC 7.0 may be attributed to additional drop hold up in the atmosphere. This effectively increases the heat capacity of the atmosphere and reduces the temperature rise for a given energy addition.

For the MSLB events, the higher pressure predicted by GOTHIC 7.0 may be attributed to additional steam generation from the improved interface heat and mass transfer models and from the drops that are generated due to dripping. The additional vaporization takes heat from the atmosphere and reduces the atmosphere temperature.

The lower sump liquid volume predicted by GOTHIC 7.0 may be attributed to higher drop concentrations in the atmosphere due to larger drag on small drops and drops generated by the dripping model. The lower sump temperature predicted by GOTHIC 7.0 may be attributed to the addition of the second energy equation for the drops and the improved heat and mass transfer models in GOTHIC 7.0.

The code changes discussed above were all directed toward providing a more realistic and physically based solution for accident events. All of the model improvements were made in conjunction with code and model validation. The validation base for GOTHIC 7.0 is significantly greater than for FATHOMS 2.4 and code results are generally in good agreement with experimental data.

2. How is the nitrogen released from the accumulators accounted for in the containment LOCA calculations?

Response: The nitrogen cover gas injection boundary condition in the long-term large-break LOCA containment response is described on pg. 6-20. A constant nitrogen flow rate of 356 lbm/sec is assumed from t=37.6 to 44.5 seconds for the injection of this gas. This boundary condition is also included in the peak containment pressure analyses. Because the peak containment pressure is reached prior to the time at which this gas is injected into containment for all cases analyzed, it has no impact on the calculated peak containment pressure.

The determination of the core flood tank initial conditions to maximize the containment pressure is discussed on pgs. 3-5 and 3-6.

3. Please provide the results using the GOTHIC 7.0 code of peak pressure and temperature calculations for the sensitivity cases listed in Table 3.1-1 of DPC-NE-3003 in tabular and graphical form. If the results or conclusions are significantly different from those presented in DPC-NE-3003P, explain the difference.

Response: Table 3.1-1 of DPC-NE-3003 describes a spectrum of LOCA peak pressure cases including a range of break locations and sensitivity analyses for several important modeling assumptions. The staff's request is understood to focus on the difference in the predicted results from the FATHOMS and GOTHIC 7.0 code versions. Rather than show the comparison for the cases in Table 3.1-1, a different set of cases has been selected which is more focused on the current knowledge of the key sensitivity parameters. Also, the results presented are for the ROTSGs, which is the subject of Revision 1 to the topical report.

The results of the 14 code comparison analyses are shown in Table 3-1 below and in the figures that follow. In general, the pressure and vapor temperature response predictions were very similar in both magnitude and trend. GOTHIC 7.0 predictions for peak pressures and vapor temperature were slightly lower (~0.5 psig and 0.5°F). These differences are consistent with the code model differences described in the response to Question #1. Both codes predict a saturated containment atmosphere due to the large amount of drops injected. The slightly lower pressure, vapor temperature, and liquid temperature from GOTHIC 7.0 may be attributed to additional drop hold up in the atmosphere. This effectively increases the heat capacity of the atmosphere and reduces the temperature rise for a given energy addition. A comparison of the total liquid and drop masses, and liquid temperatures, show the predictions of the two codes to be essentially the same. This is consistent with the changes in the drop deposition and entrainment models discussed in Question #1. These results show that the two code versions give essentially similar results for the same break location and initial and boundary conditions.

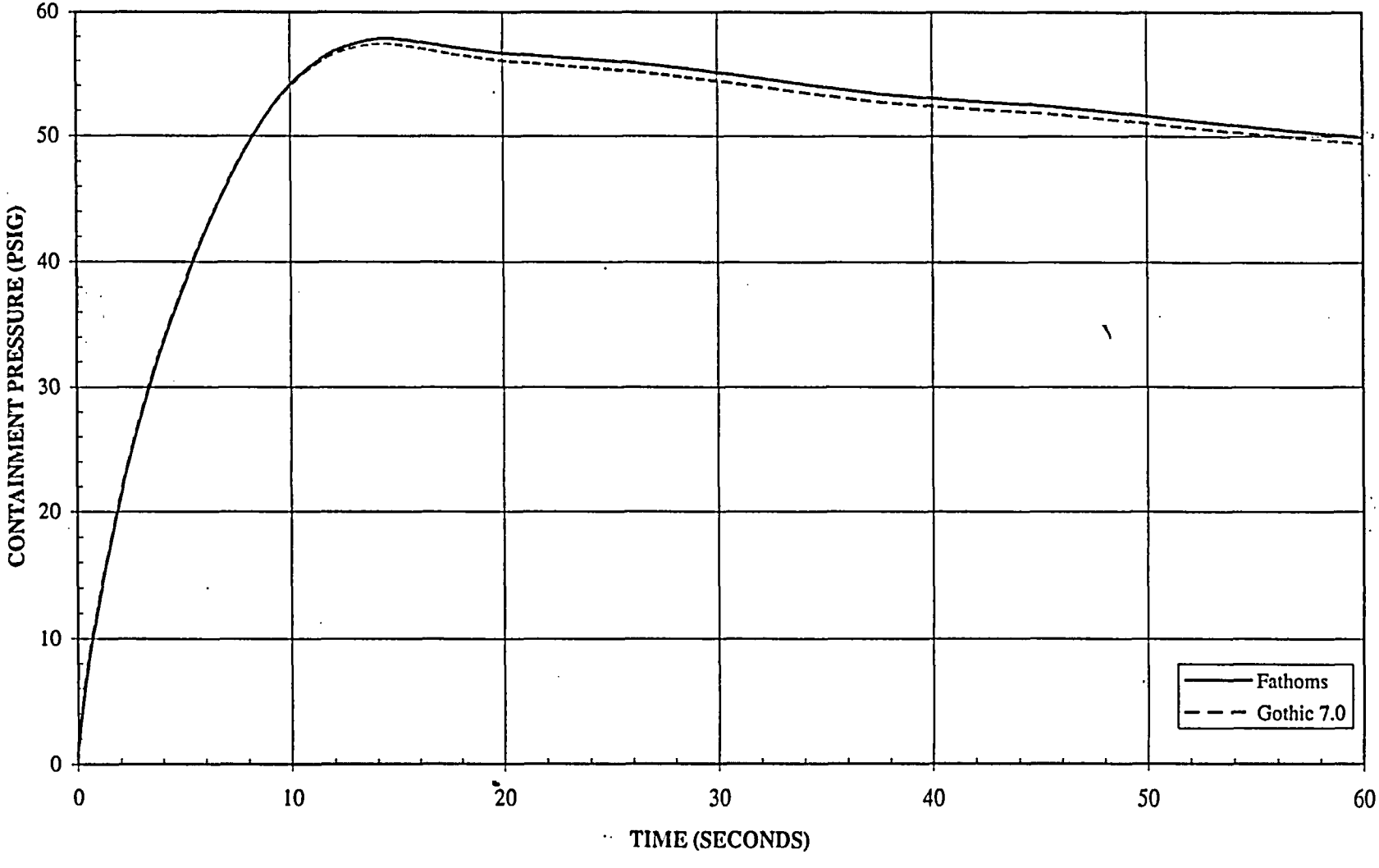
Table 3-1

ROTSG Case Description	Peak Containment Pressure (psig)		Peak Containment Temperature (°F)	
	Fathoms	Gothic 7.0	Fathoms	Gothic 7.0
Hot Leg SG Inlet Break w/ High RCS Flow Cases				
14.1 ft^2 Hot Leg Break w/ M1 Pump Degradation Model	57.83	57.41	282.77	282.41
14.1 ft^2 Hot Leg Break w/ M3 Pump Degradation Model	58.03	57.63	283.02	282.67
14.1 ft^2 Hot Leg Break w/ Sensitivity to Steamline Superheat	57.99	57.60	282.98	282.64
14.1 ft^2 Hot Leg Break w/ Sensitivity to Main Feedwater	58.48	58.09	283.56	283.22
Hot Leg SG Inlet Break on a 5° ΔTcold Loop Cases				
14.1 ft^2 Hot Leg Break 5° ΔTcold Loop Case 1	57.68	57.28	282.60	282.25
14.1 ft^2 Hot Leg Break 5° ΔTcold Loop Case 2	57.88	57.48	282.84	282.49
Hot Leg SG Inlet Break w/ Low RCS Flow Cases				
14.1 ft^2 Hot Leg Break w/ M1 Pump Degradation Model	58.02	57.59	282.99	282.62
14.1 ft^2 Hot Leg Break w/ M3 Pump Degradation Model	57.76	57.37	282.70	282.35
Hot Leg RV Outlet Break w/ Low RCS Flow Cases				
14.1 ft^2 Hot Leg Break w/ M1 Pump Degradation Model	57.14	56.84	281.97	281.71
14.1 ft^2 Hot Leg Break w/ M3 Pump Degradation Model	56.85	56.58	281.64	281.40
Hot Leg RV Outlet Break w/ High RCS Flow Cases				
14.1 ft^2 Hot Leg Break w/ M1 Pump Degradation Model	57.16	56.87	282.00	281.75
14.1 ft^2 Hot Leg Break w/ M3 Pump Degradation Model	56.87	56.59	281.65	281.41
Cold Leg Pump Discharge Break w/ High RCS Flow Case				
8.55 ft^2 Cold Leg Break @ Pump Discharge	53.03	52.44	276.68	276.15
Cold Leg Pump Suction Break w/ High RCS Flow Case				
8.55 ft^2 Cold Leg Break @ Pump Suction	54.30	53.45	278.26	277.46

GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

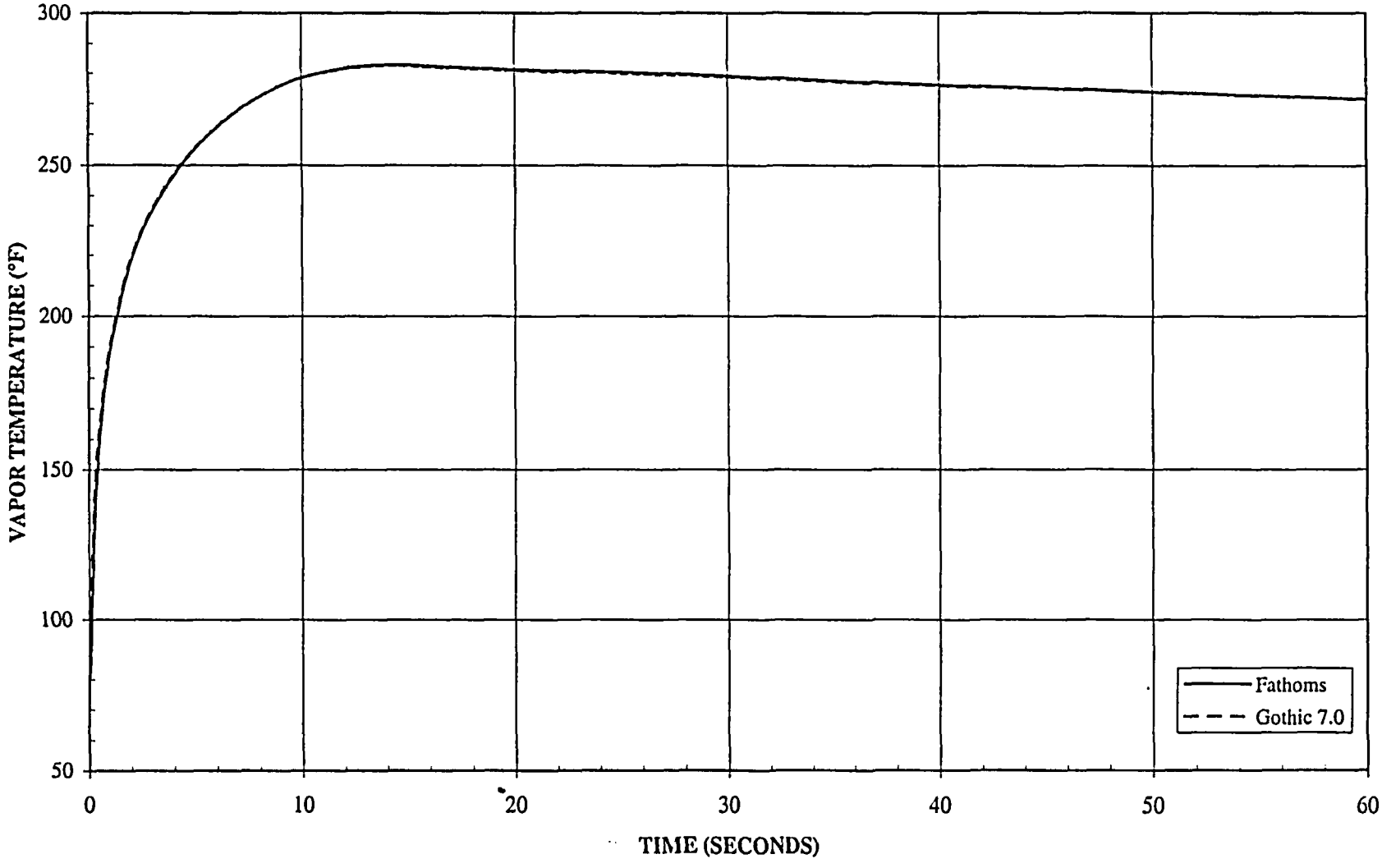
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

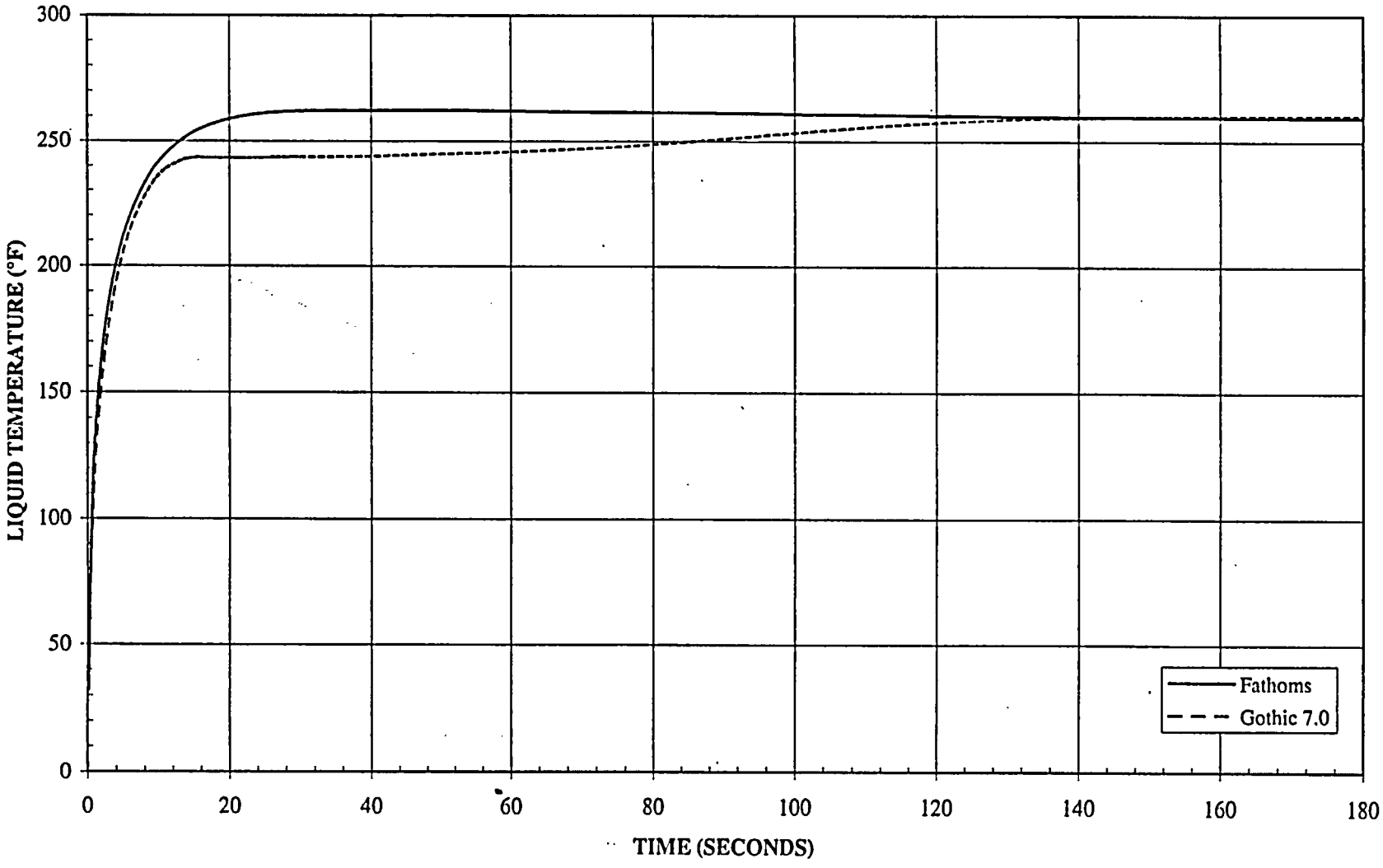
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

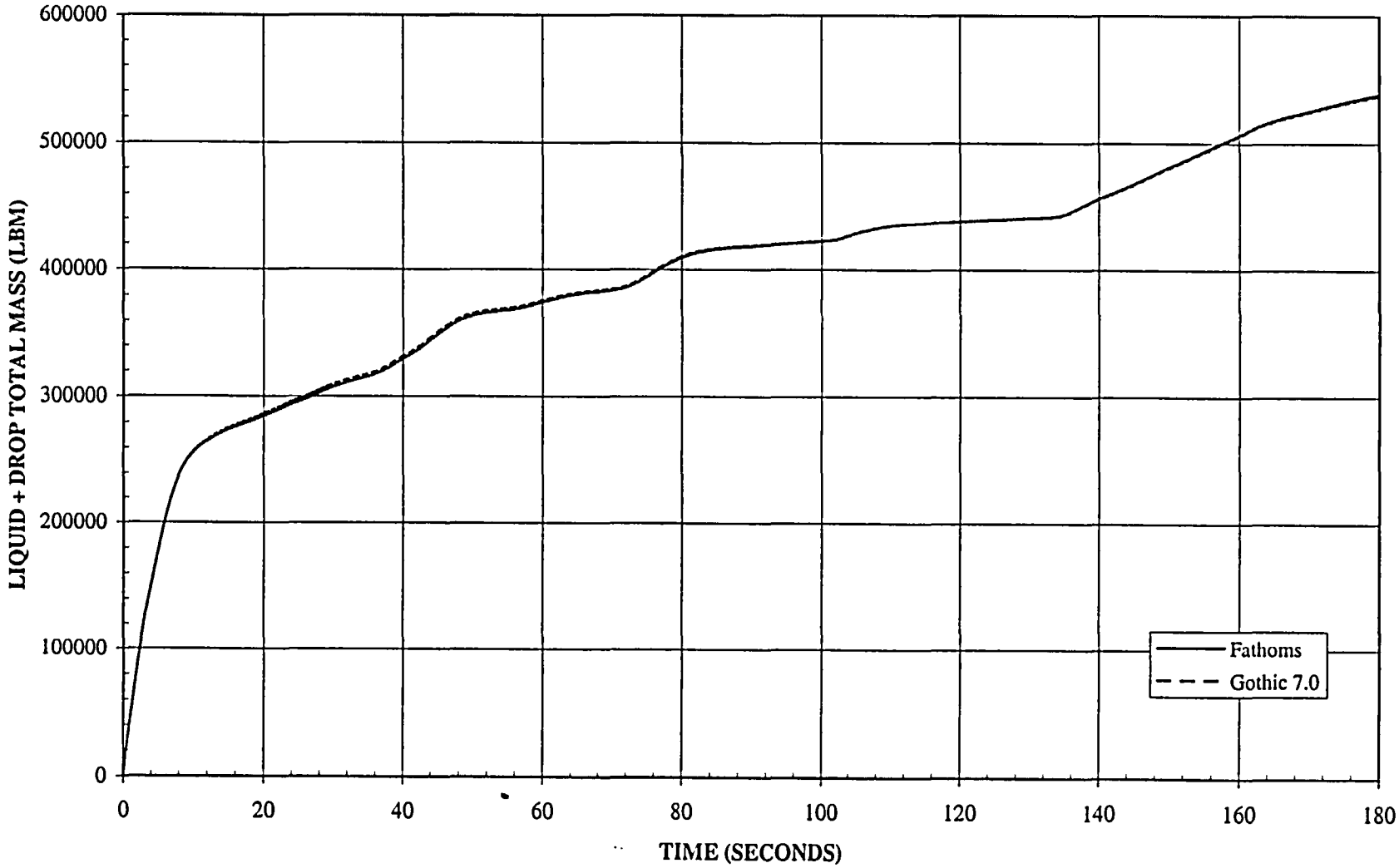
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

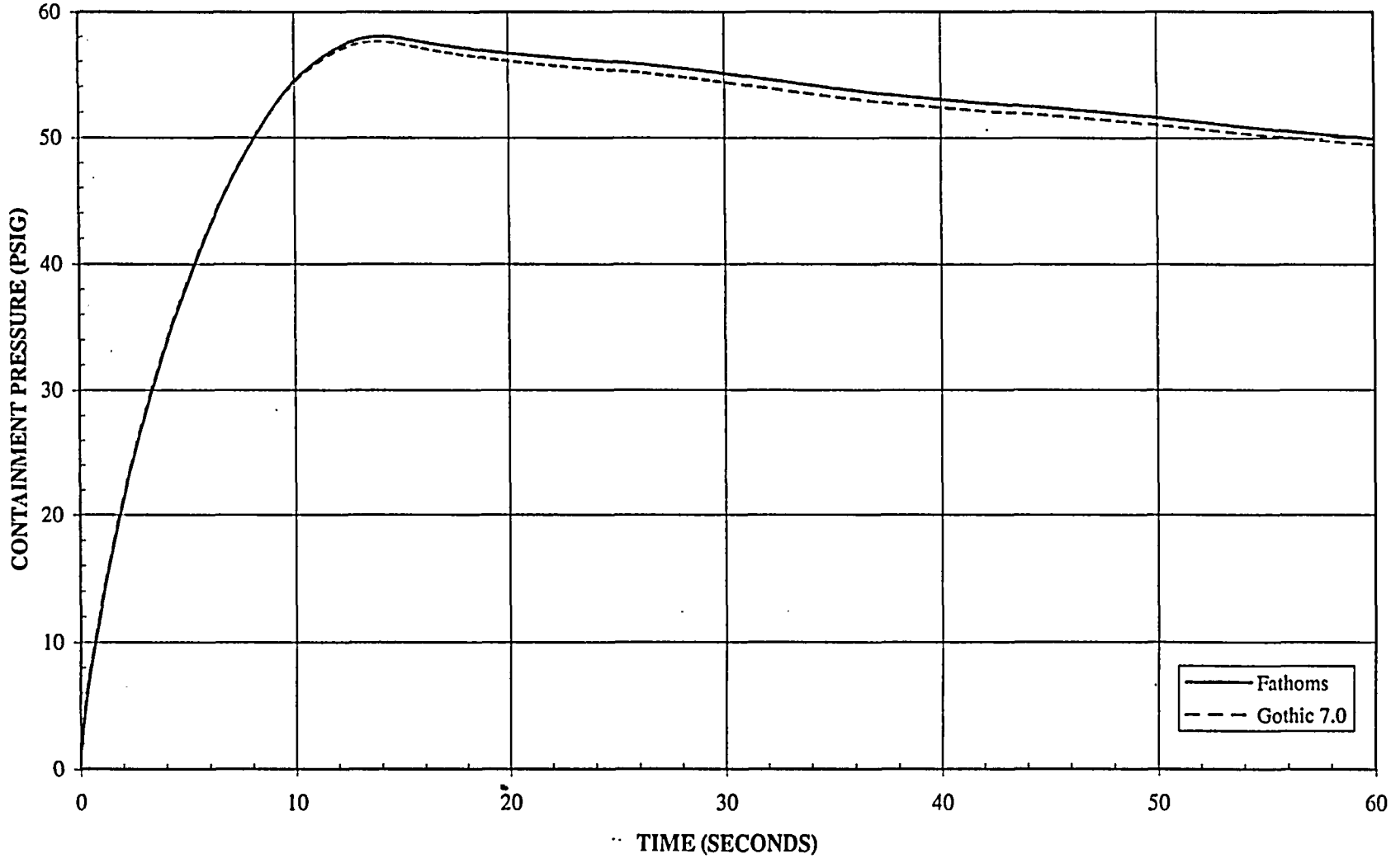
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

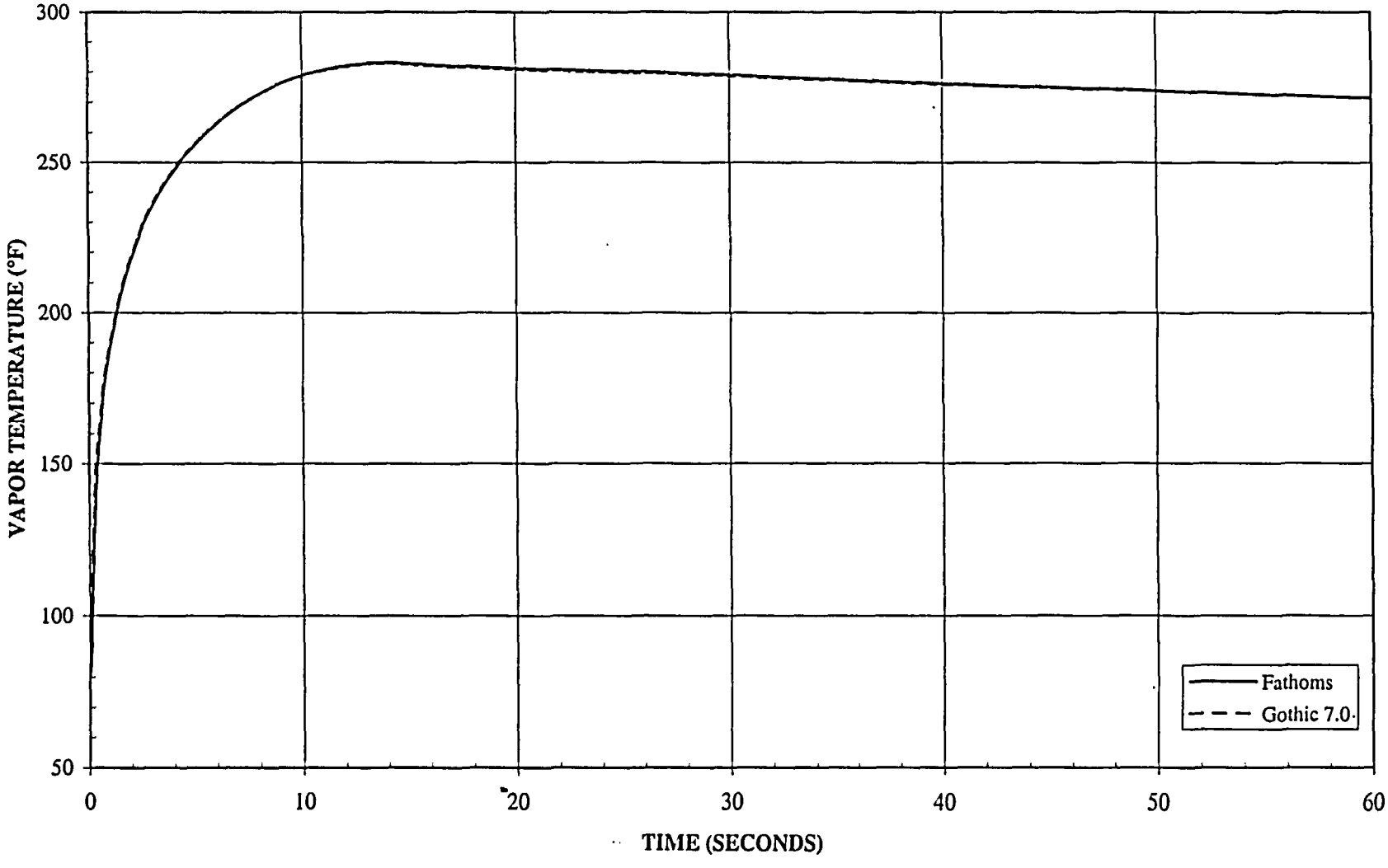
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ M3 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

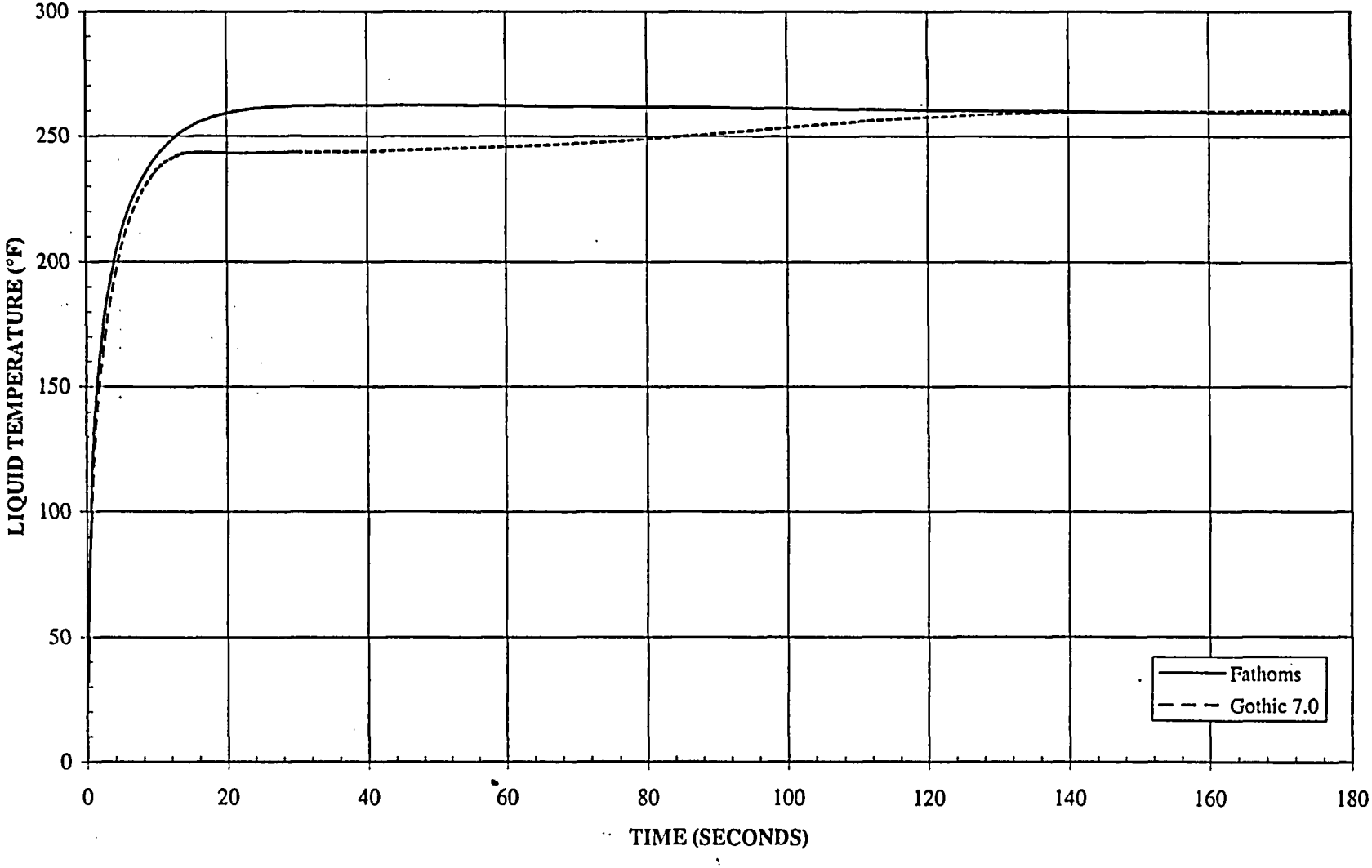
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ M3 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

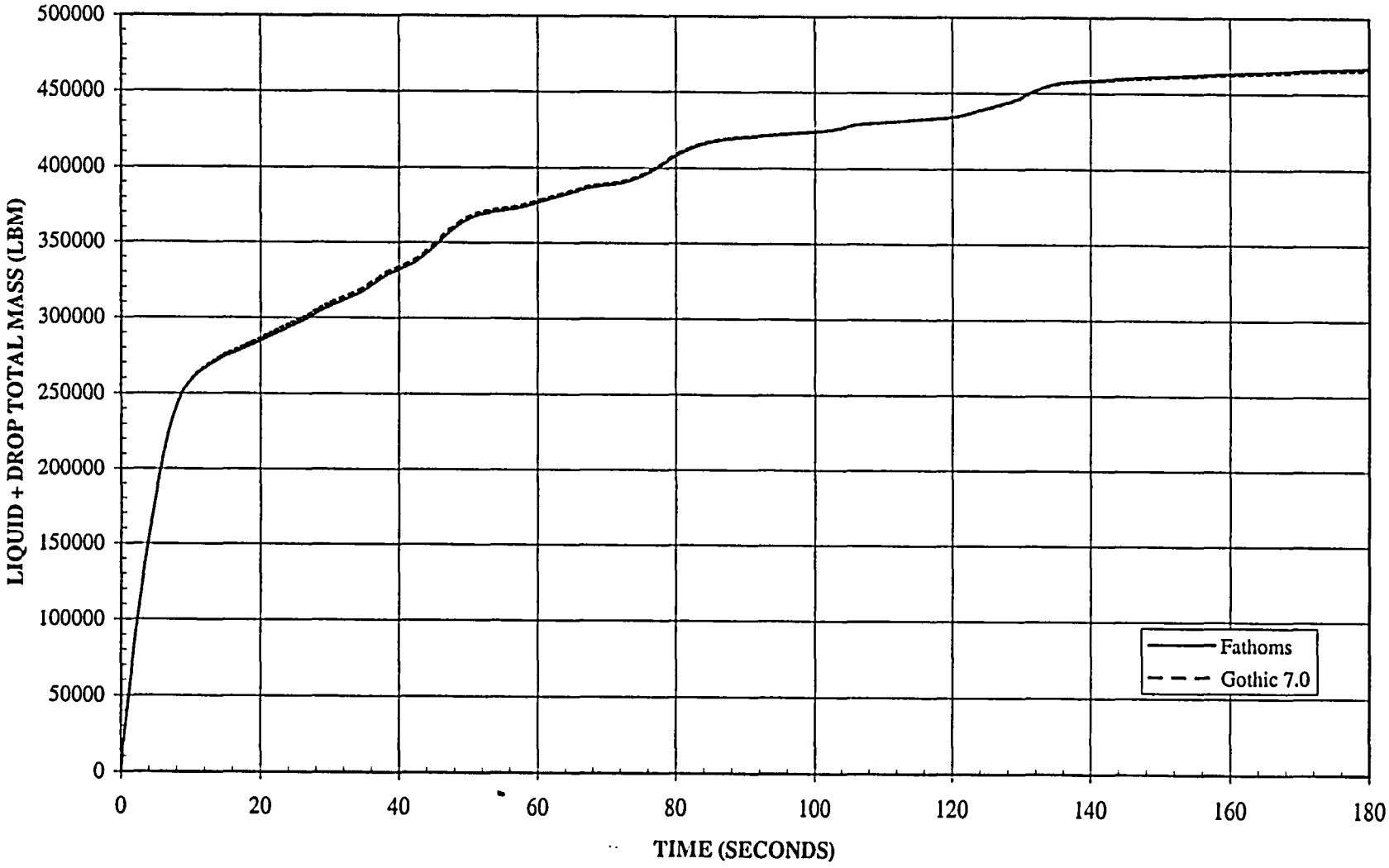
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ M3 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

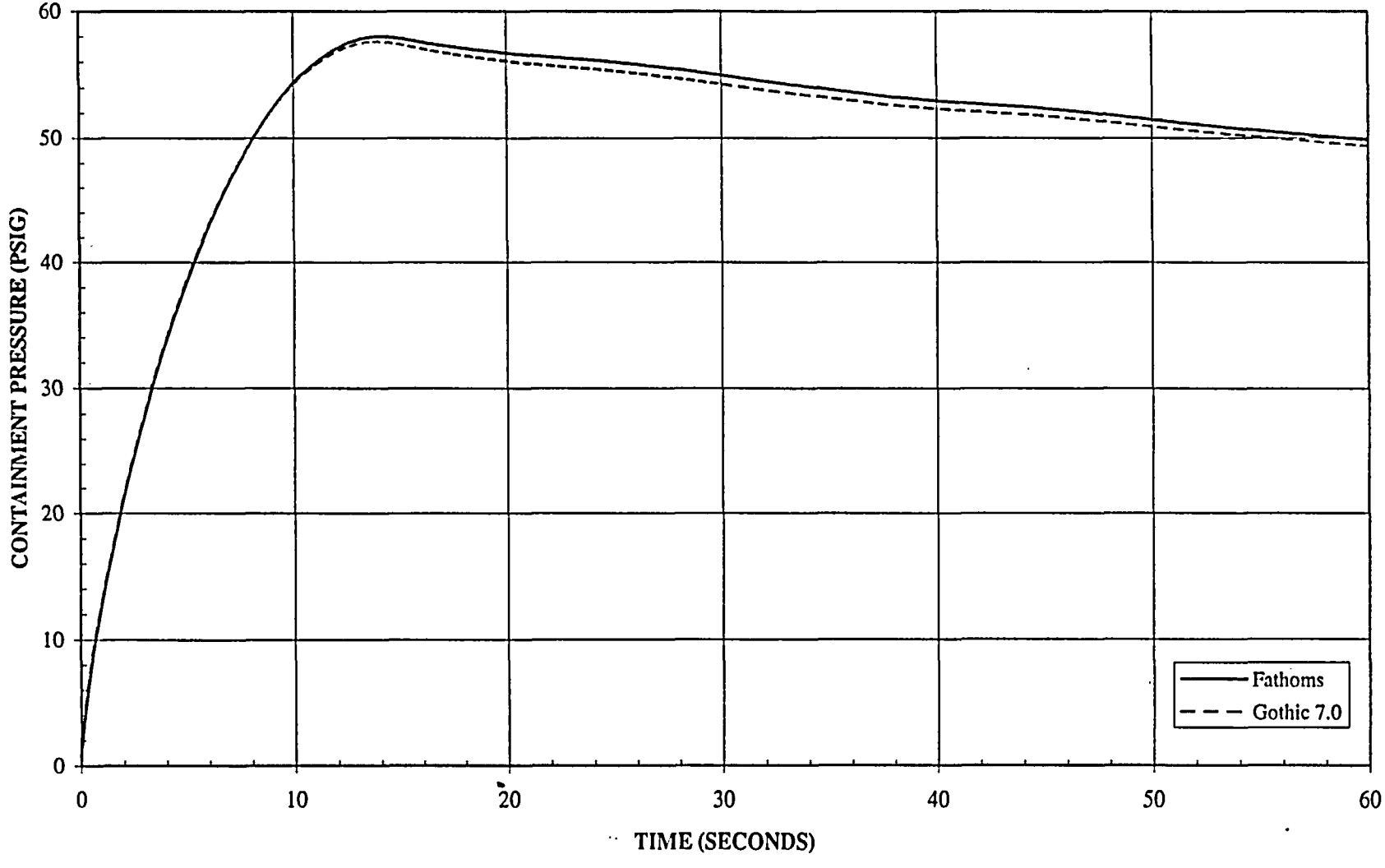
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ M3 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

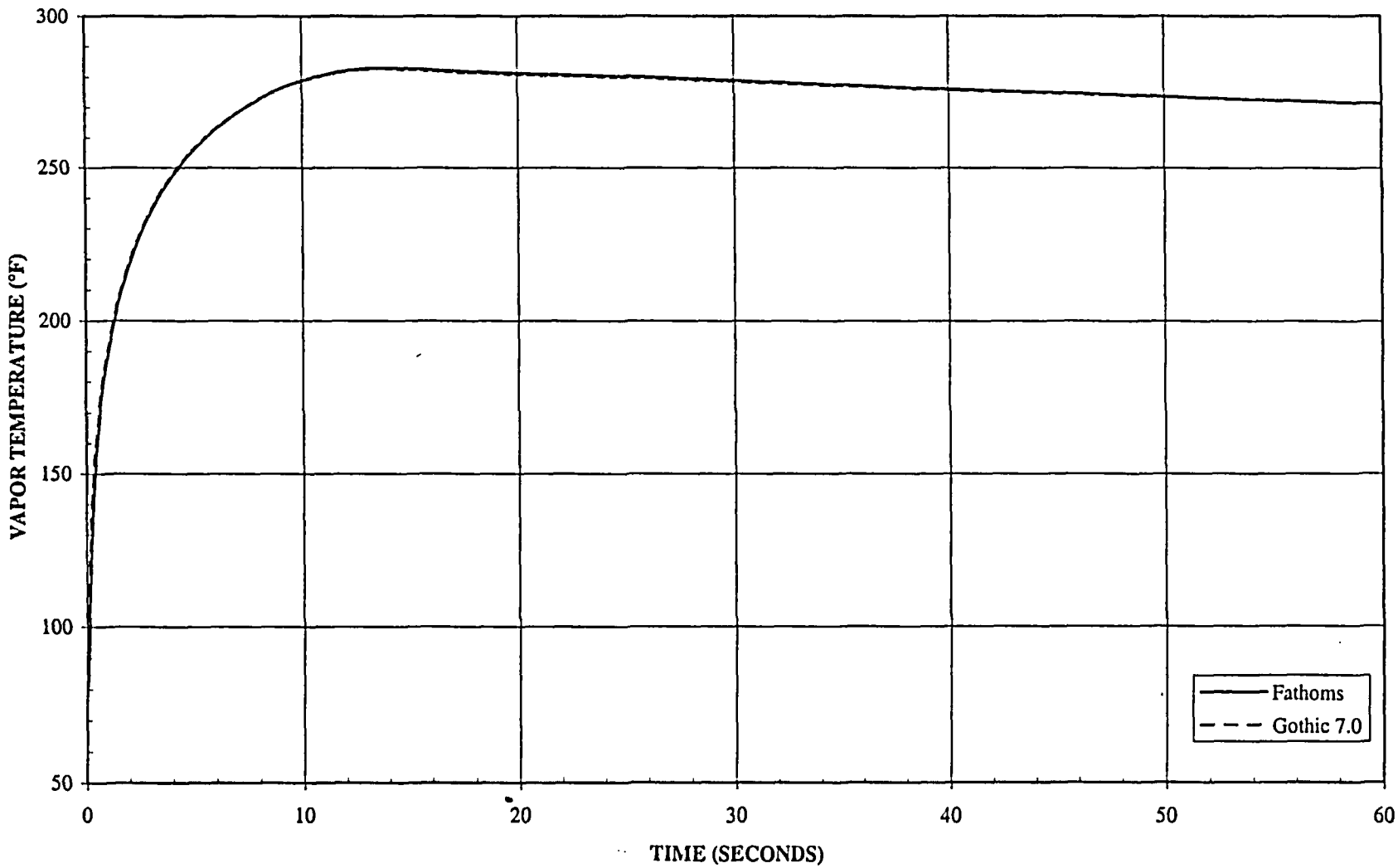
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ Sensitivity to Steamline Superheat)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

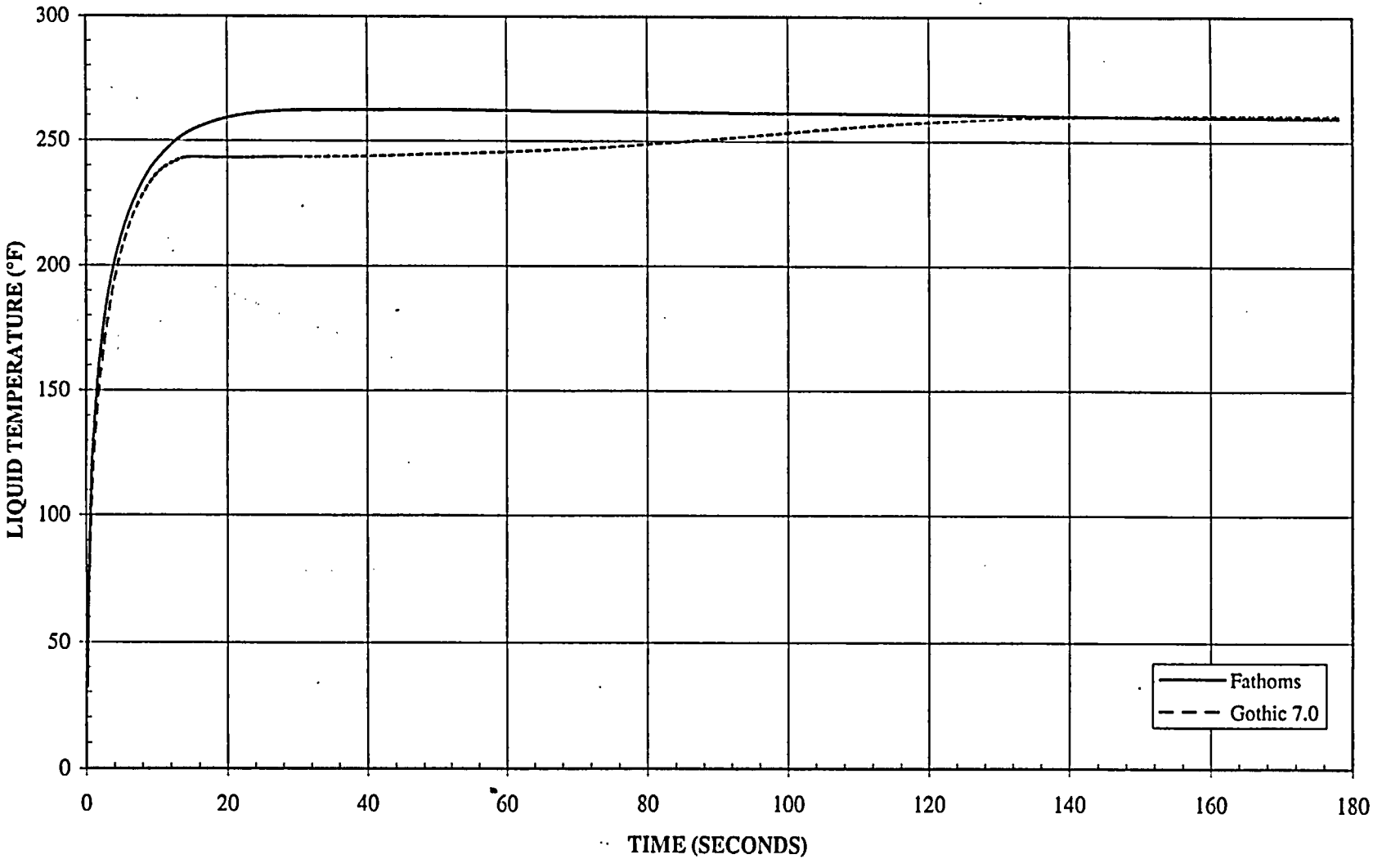
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ Sensitivity to Steamline Superheat)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

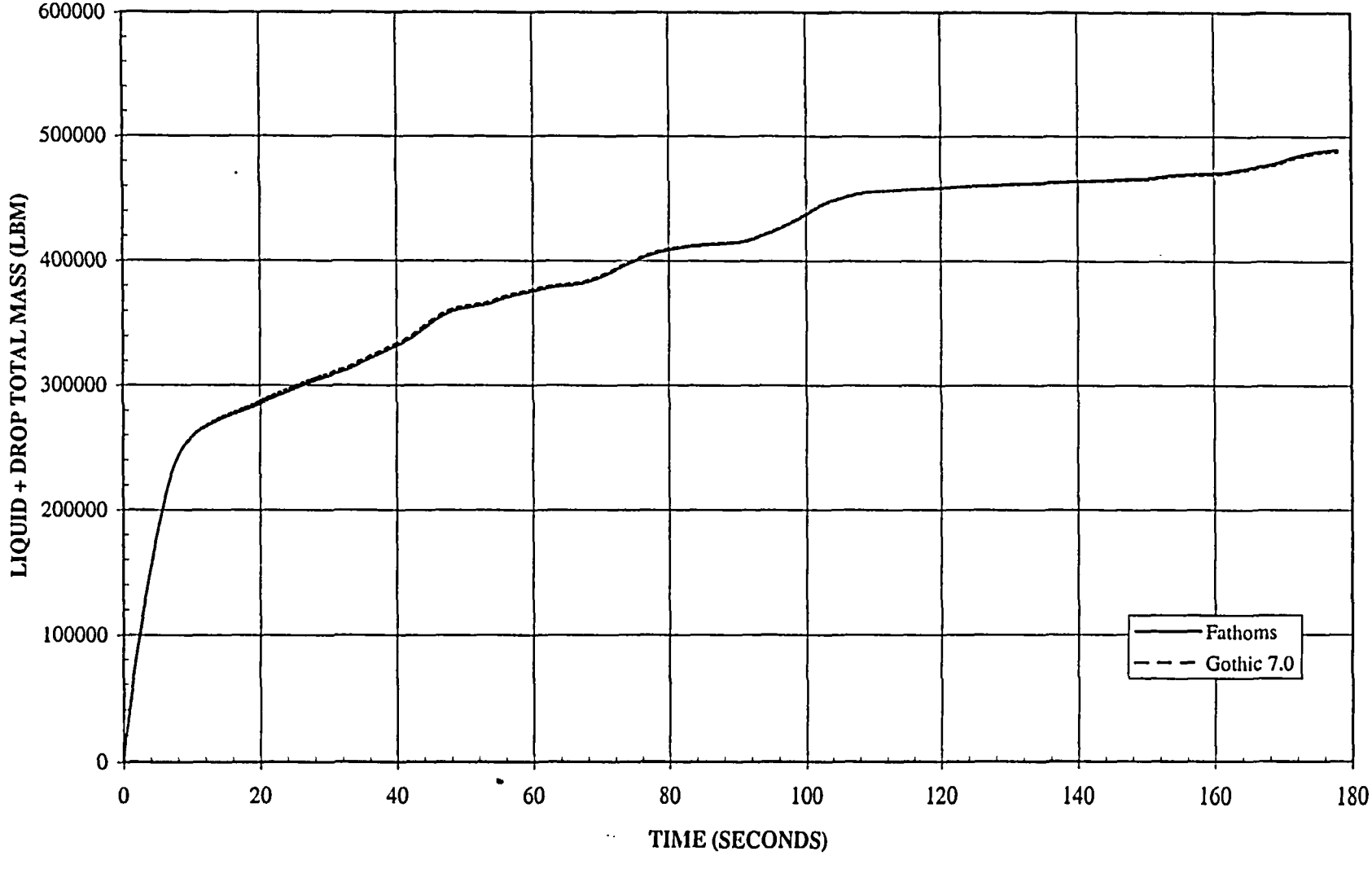
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ Sensitivity to Steamline Superheat)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

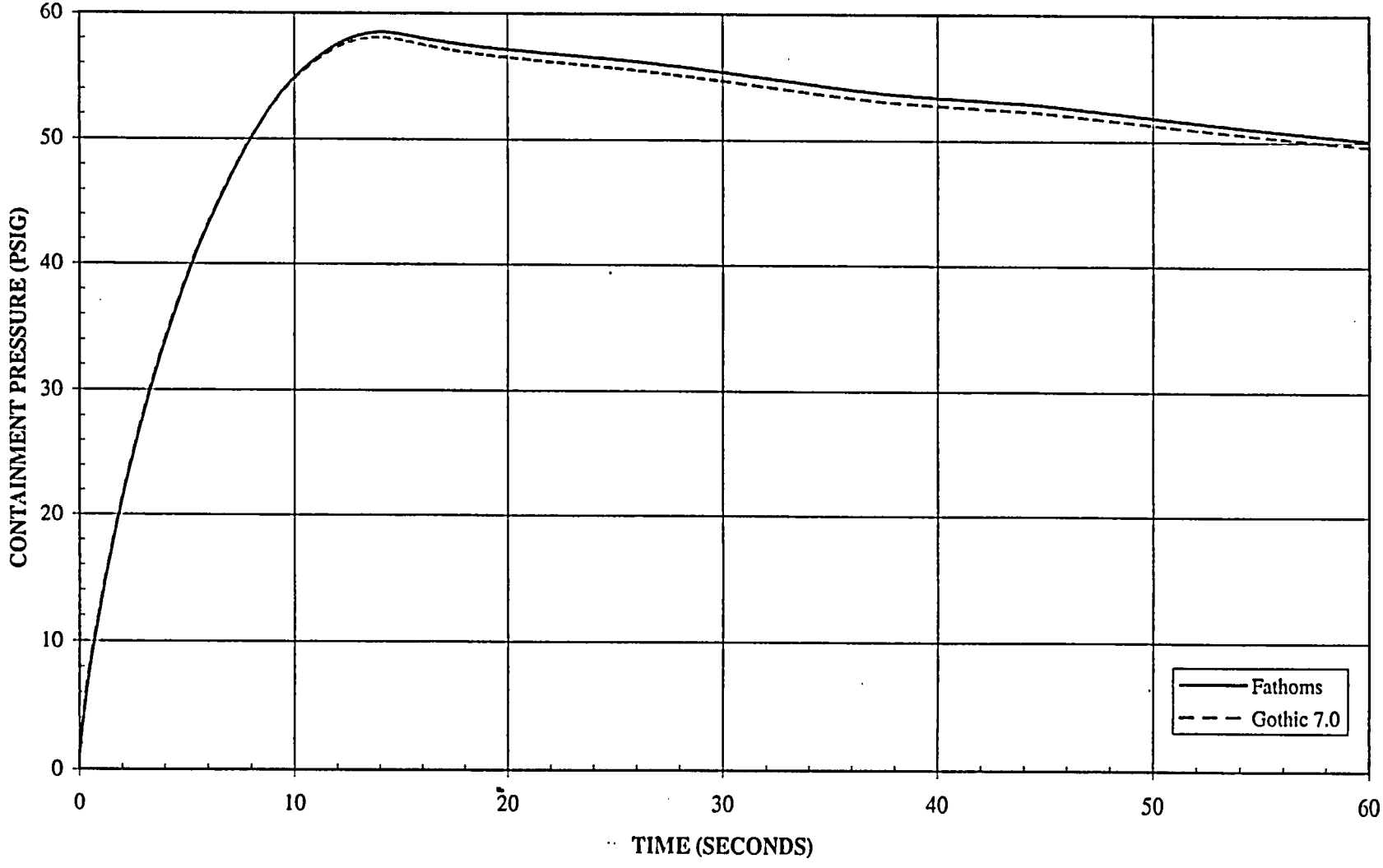
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ Sensitivity to Steamline Superheat)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

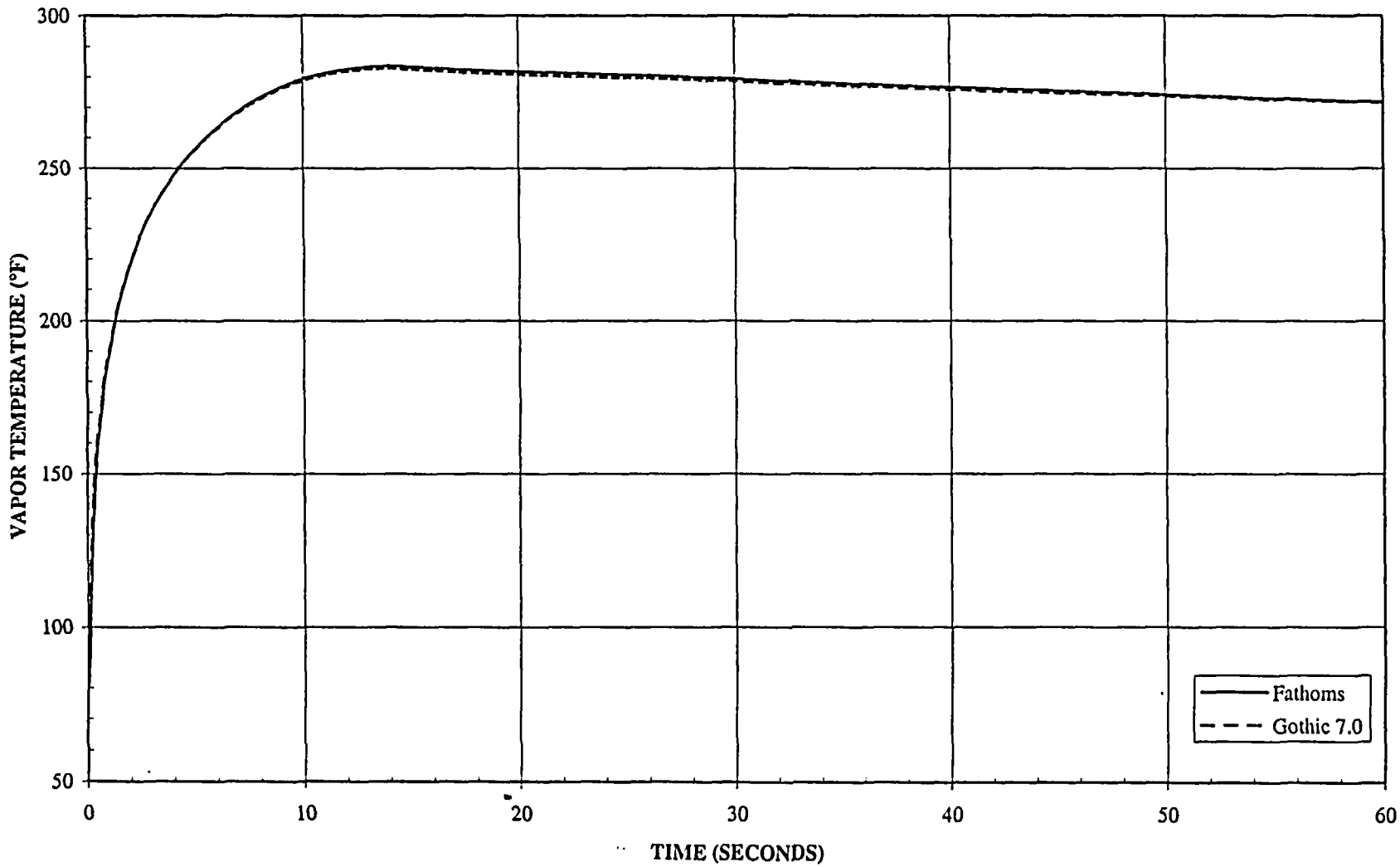
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ Sensitivity to Main Feedwater)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

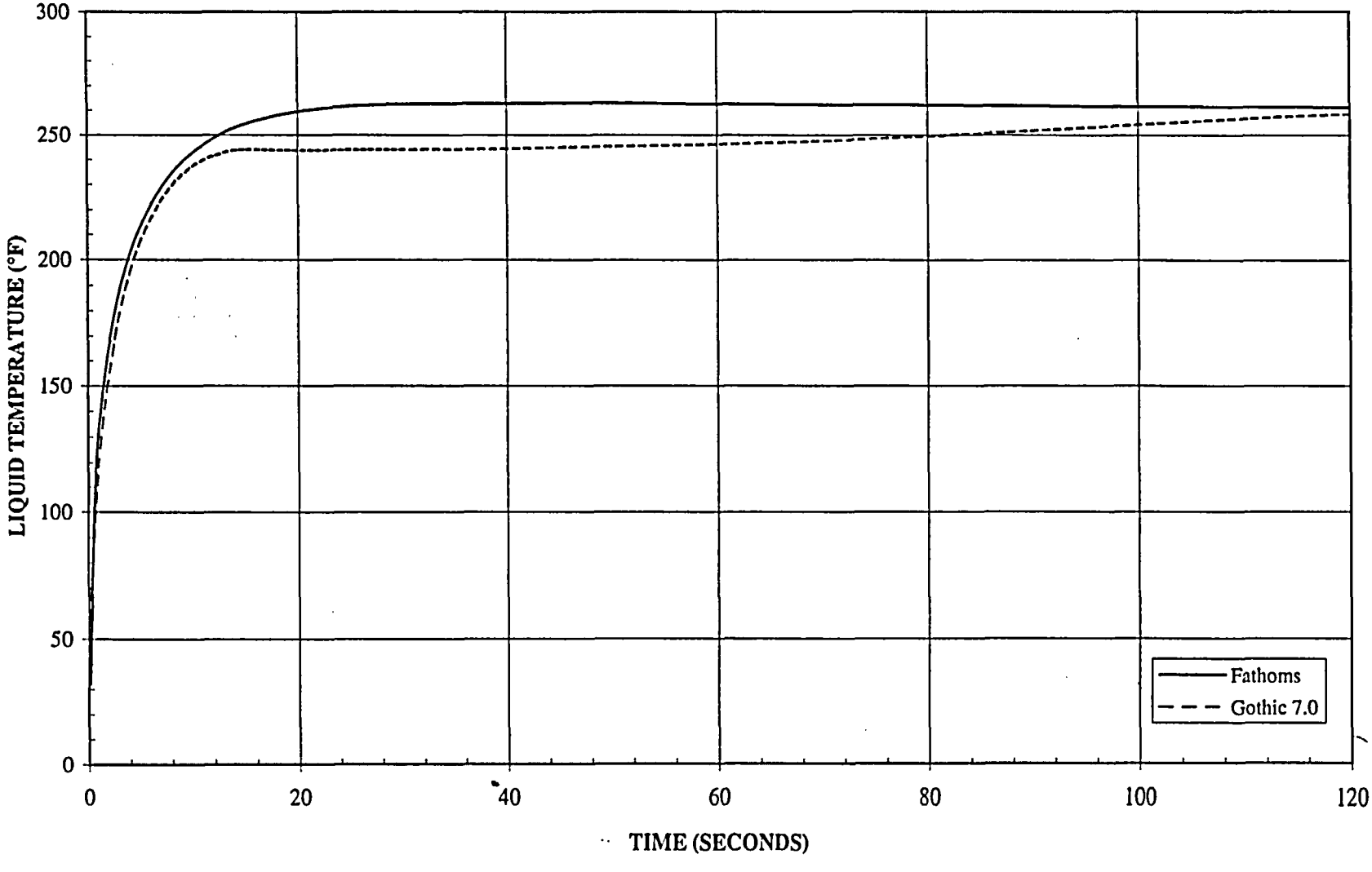
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ Sensitivity to Main Feedwater)

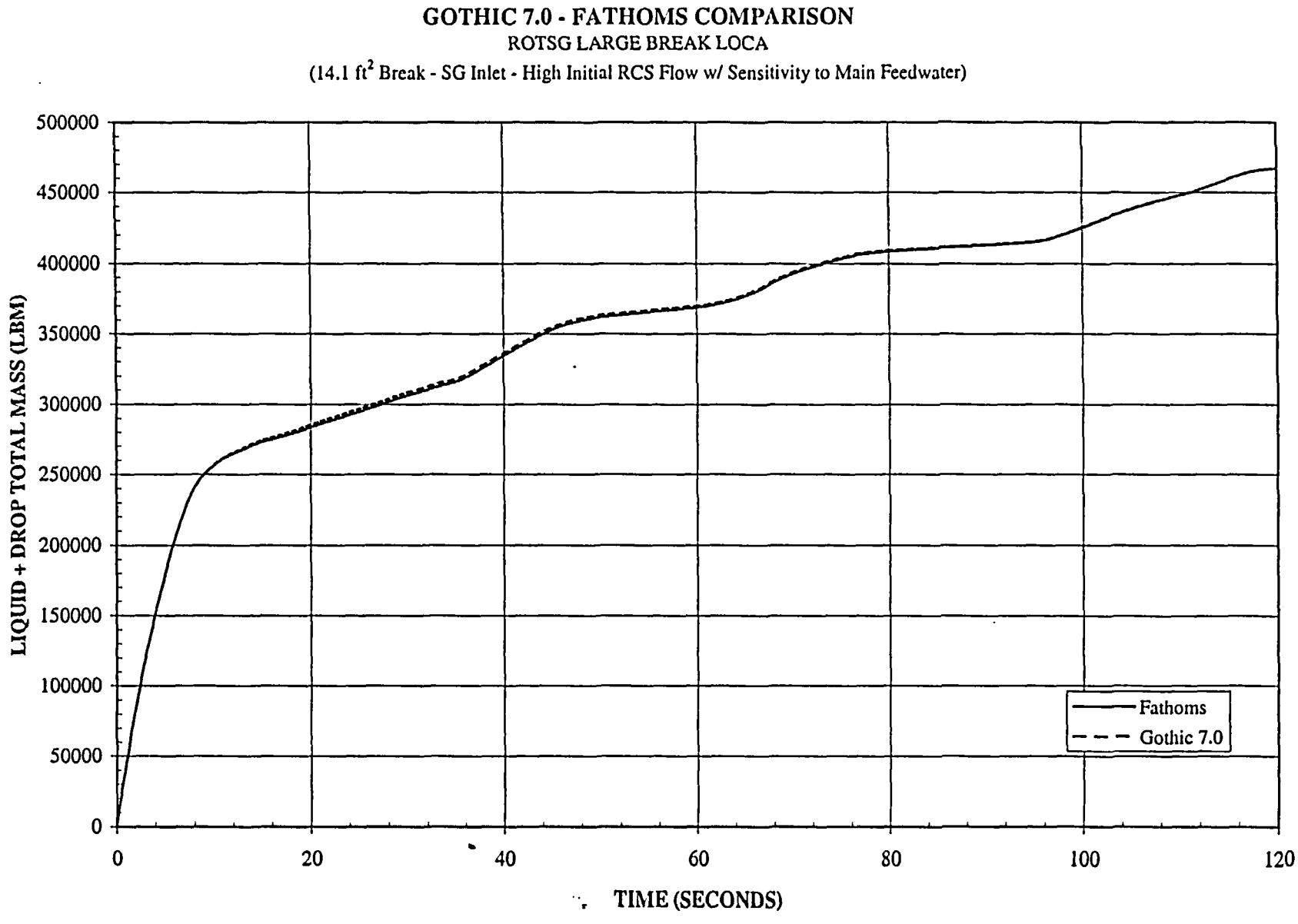


GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ Sensitivity to Main Feedwater)

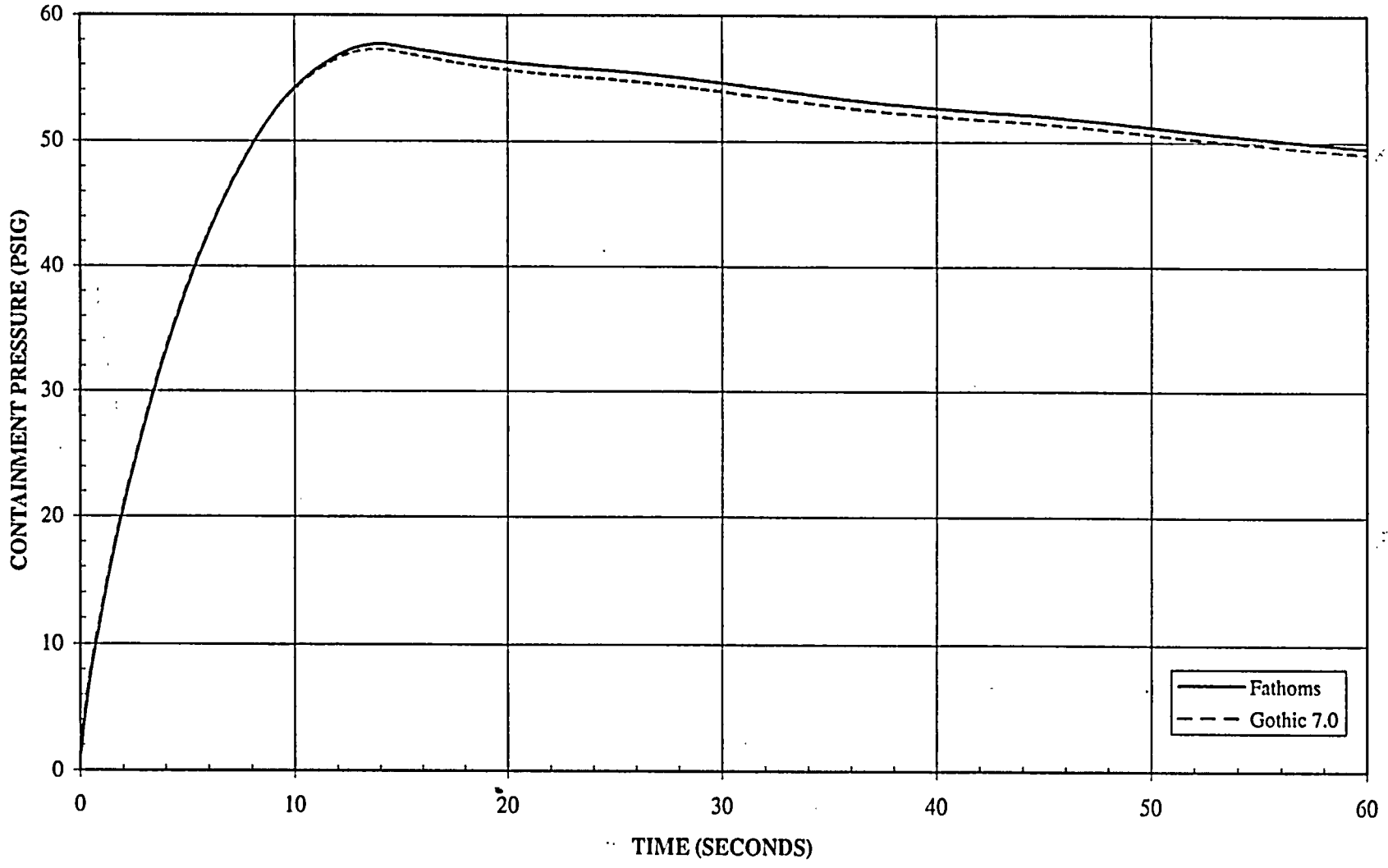



GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG LARGE BREAK LOCA
(14.1 ft² Break - SG Inlet - High Initial RCS Flow w/ Sensitivity to Main Feedwater)
LIQUID + DROP TOTAL MASS (LBM)
500000
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
20
40
60
80
100
120
TIME (SECONDS)
Fathoms
Gothic 7.0

GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

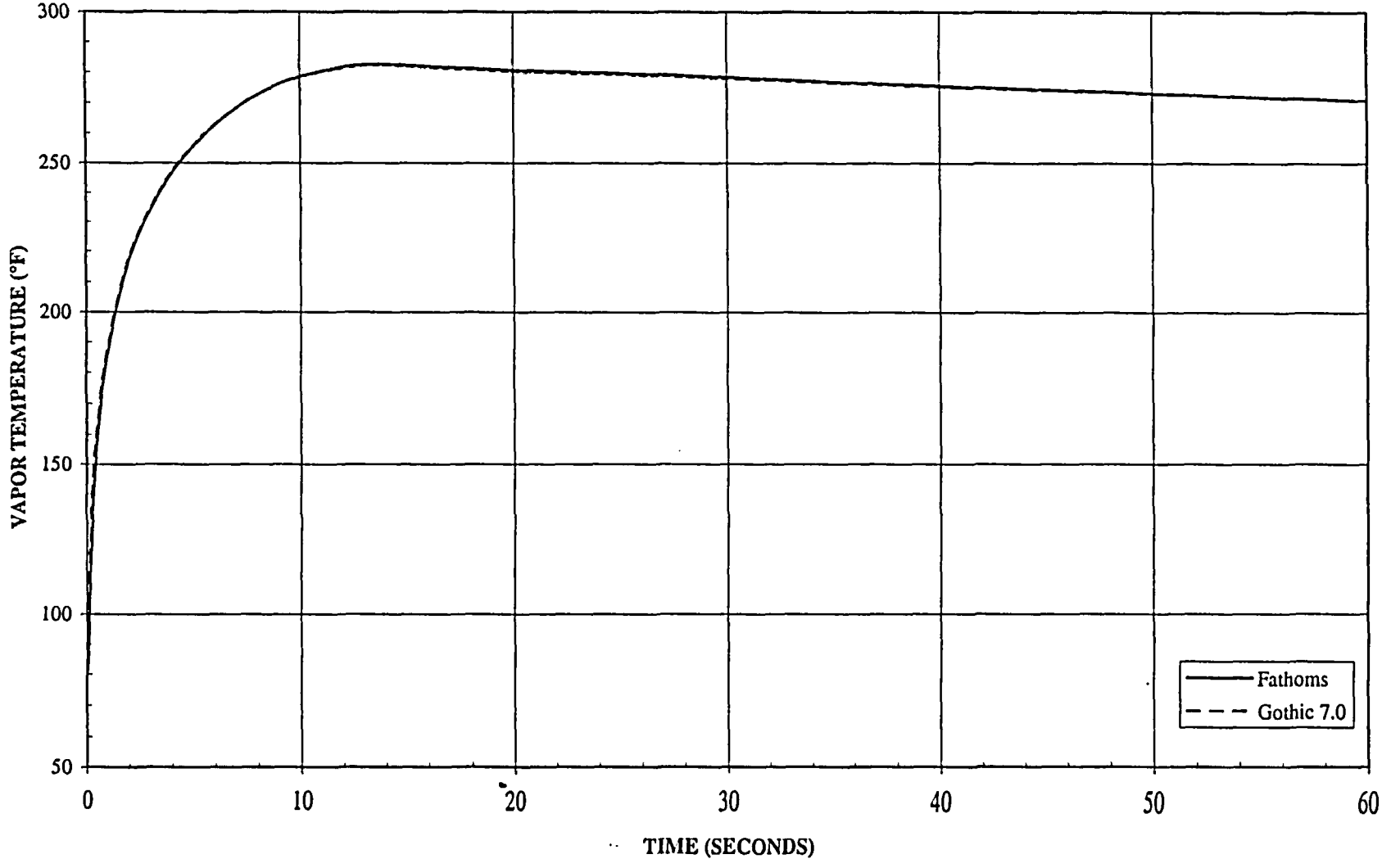
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ 5° ΔTcold Loop Case 1)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

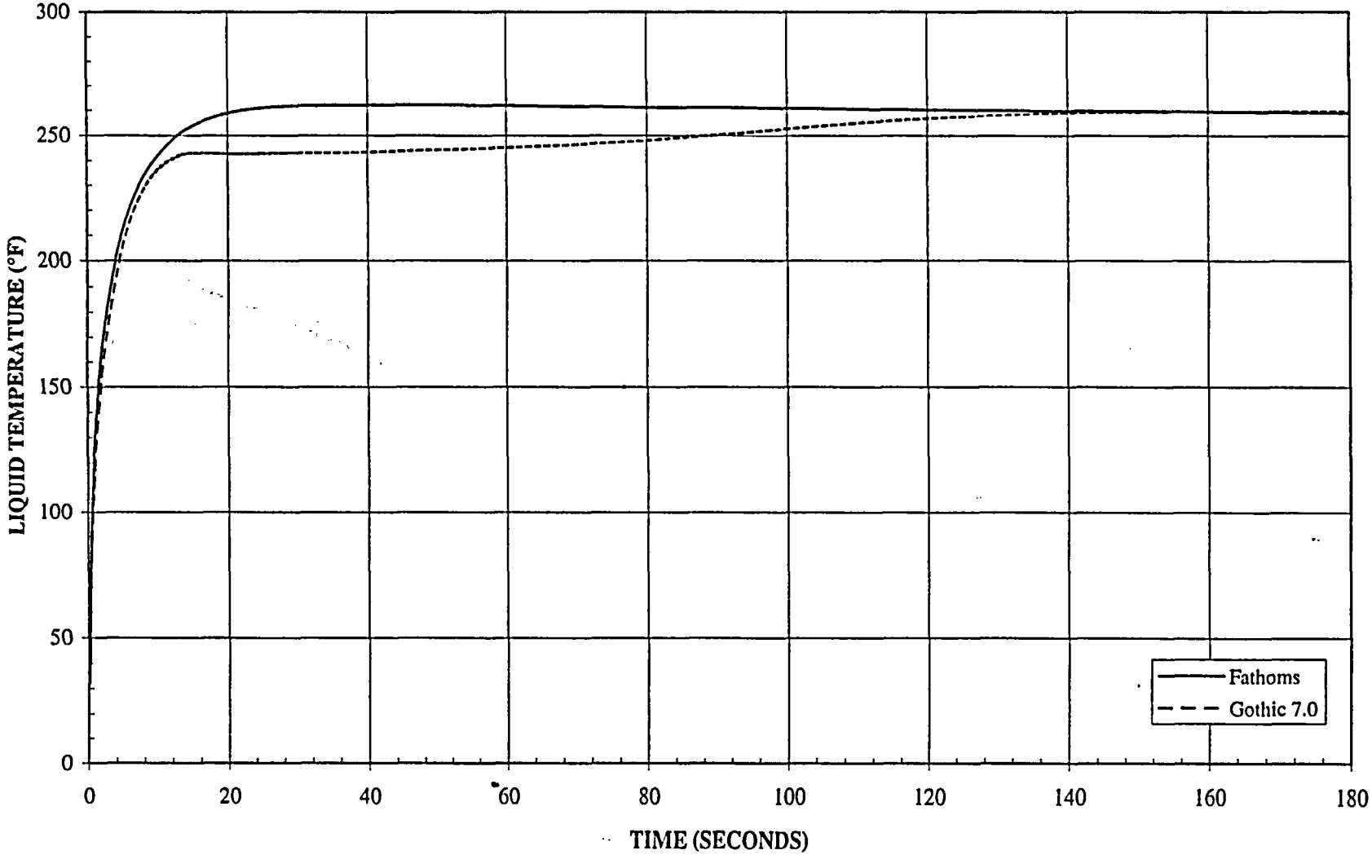
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ 5° ΔTcold Loop Case 1)

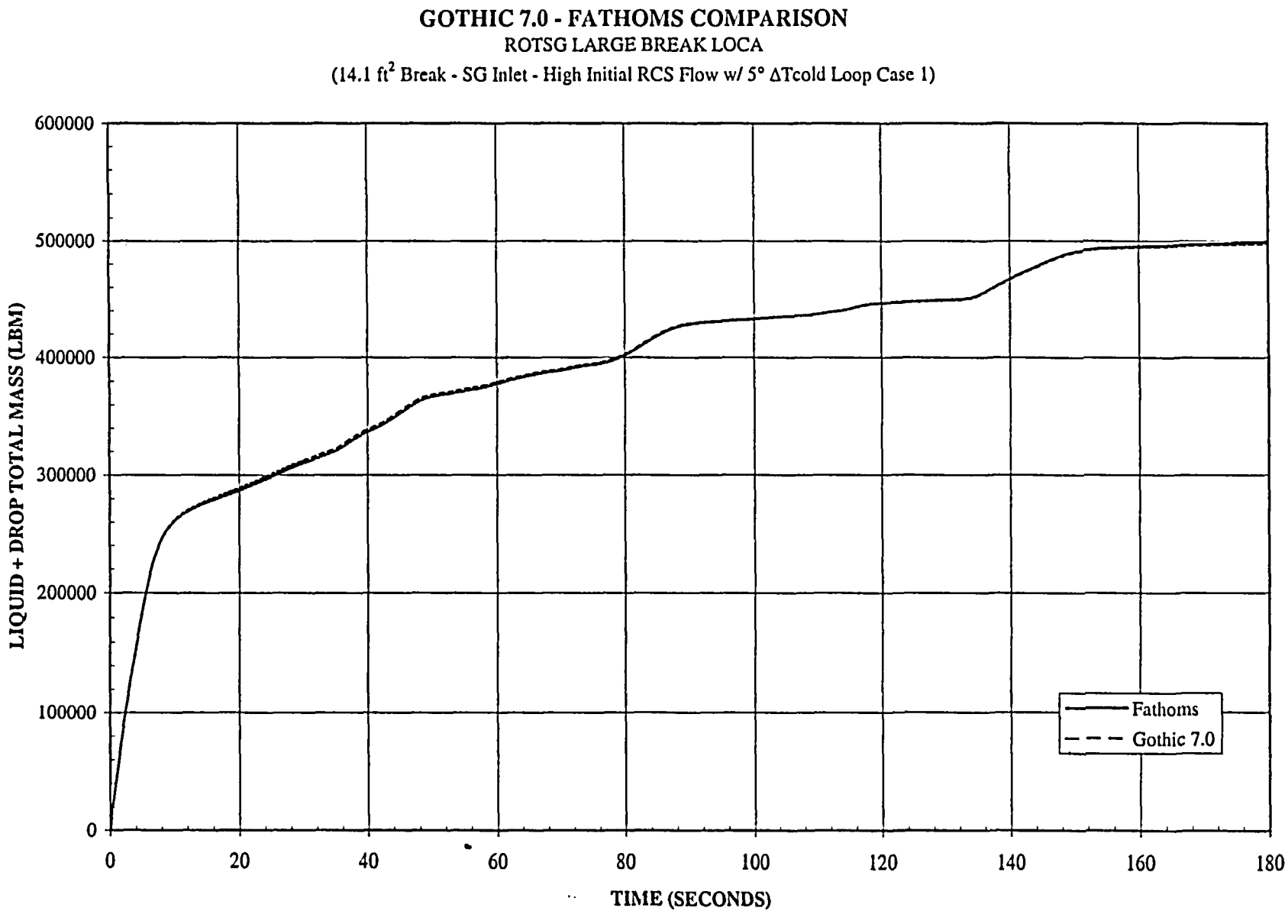


GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ 5° ΔTcold Loop Case 1)

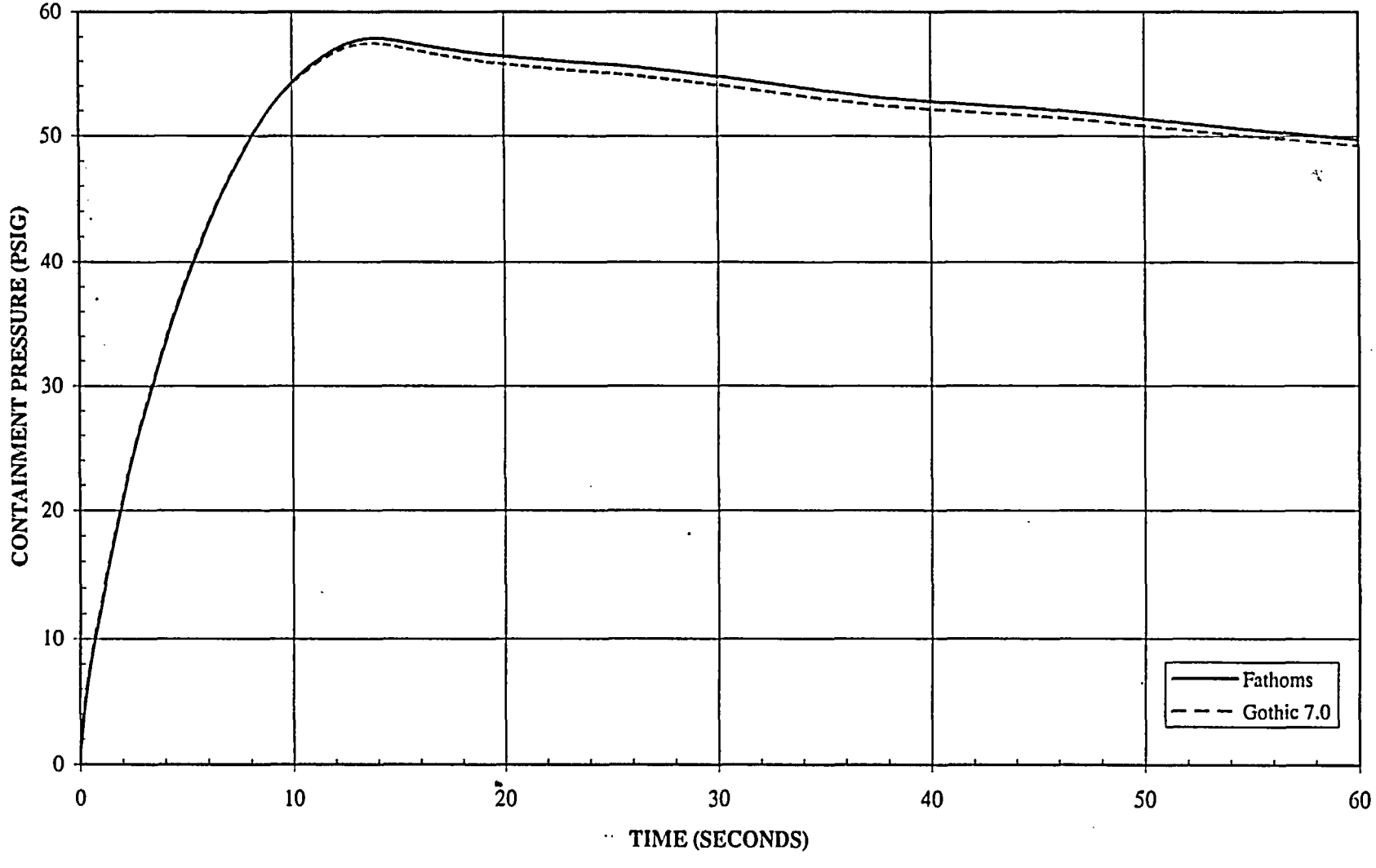



GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG LARGE BREAK LOCA
(14.1 ft² Break - SG Inlet - High Initial RCS Flow w/ 5° ΔTcold Loop Case 1)
LIQUID + DROP TOTAL MASS (LBM)
600000
500000
400000
300000
200000
100000
0
0
20
40
60
80
100
120
140
160
180
TIME (SECONDS)
Fathoms
Gothic 7.0

GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

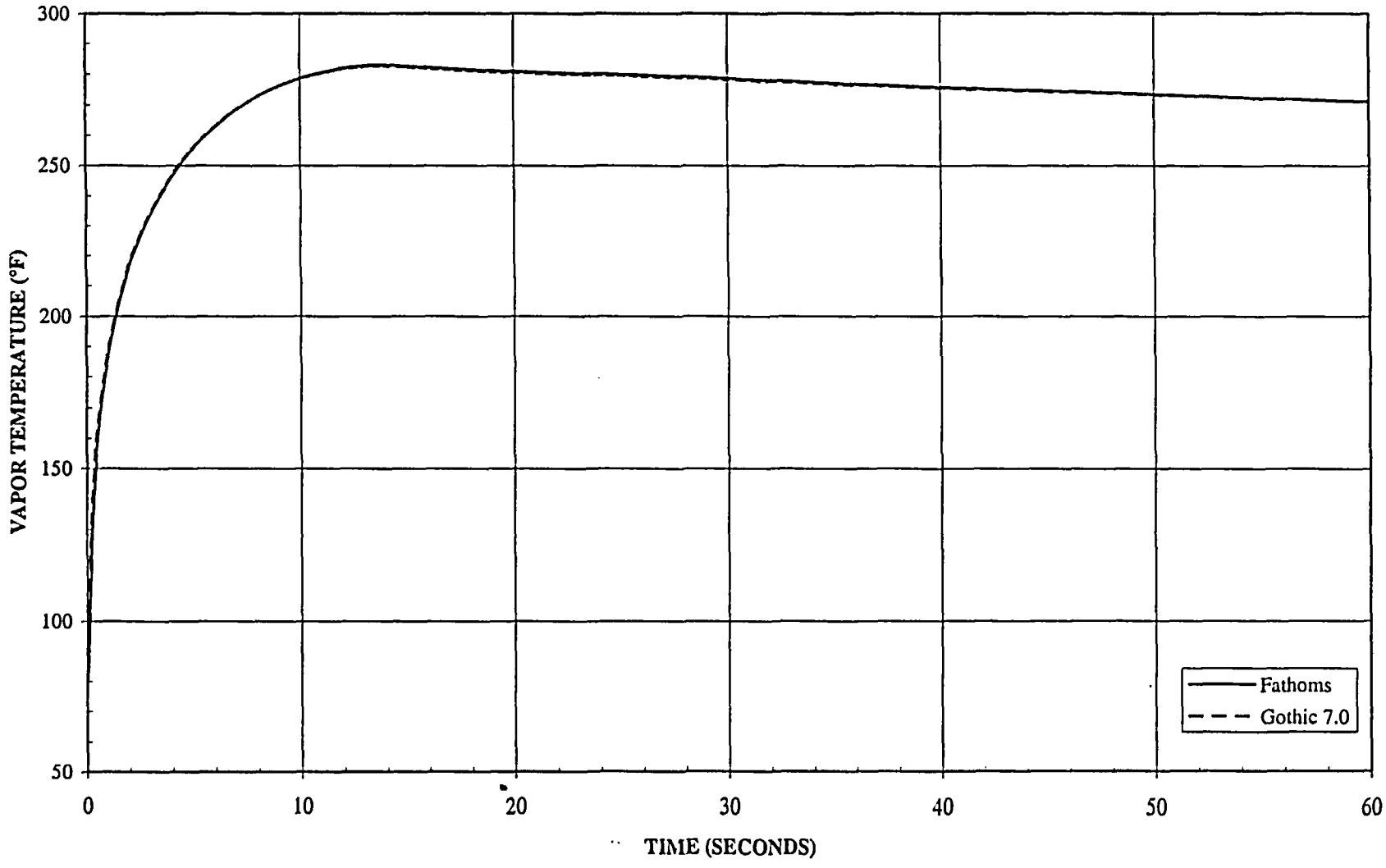
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ 5° ΔTcold Loop Case 2)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

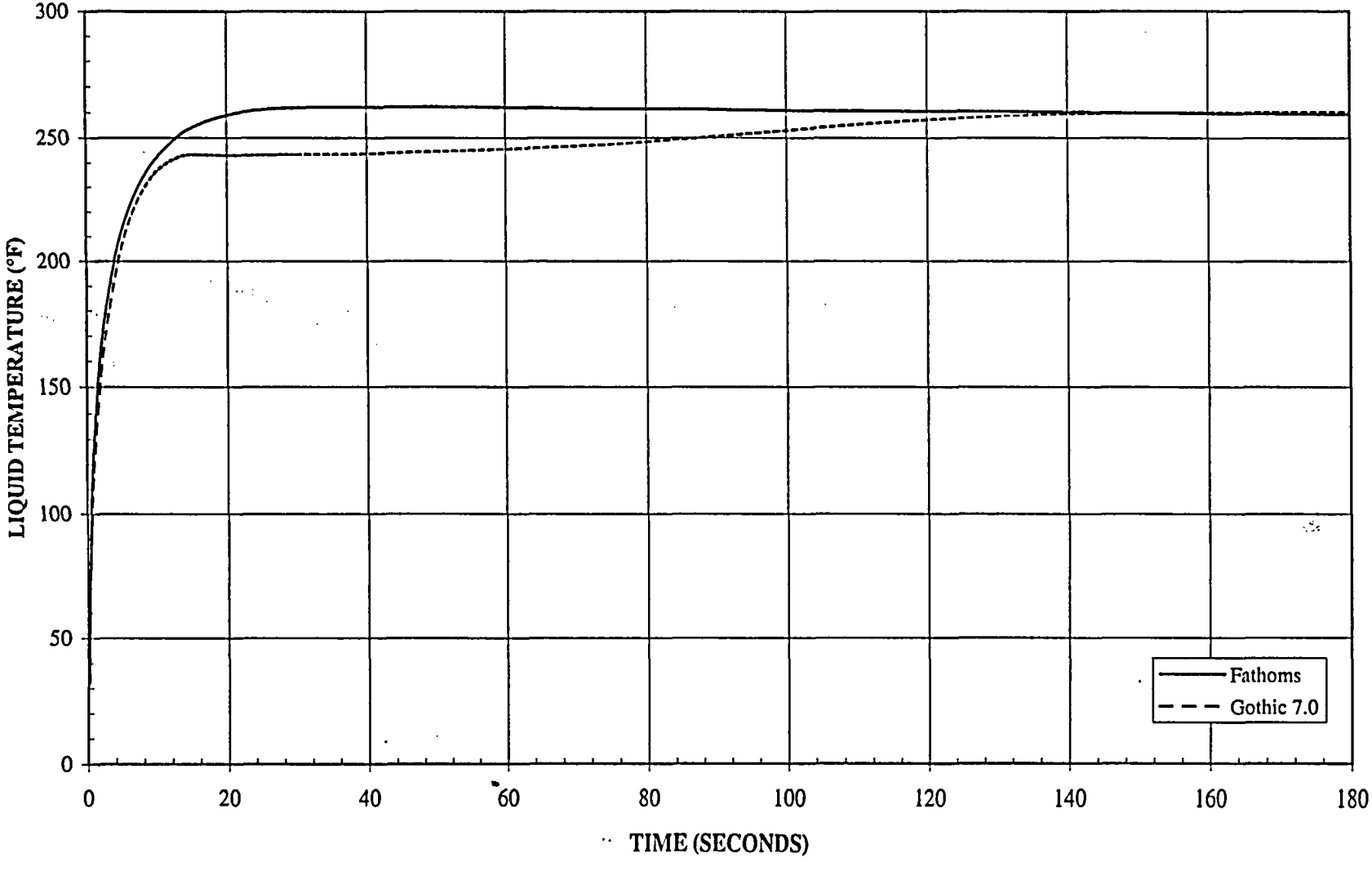
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ 5° ΔTcold Loop Case 2)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

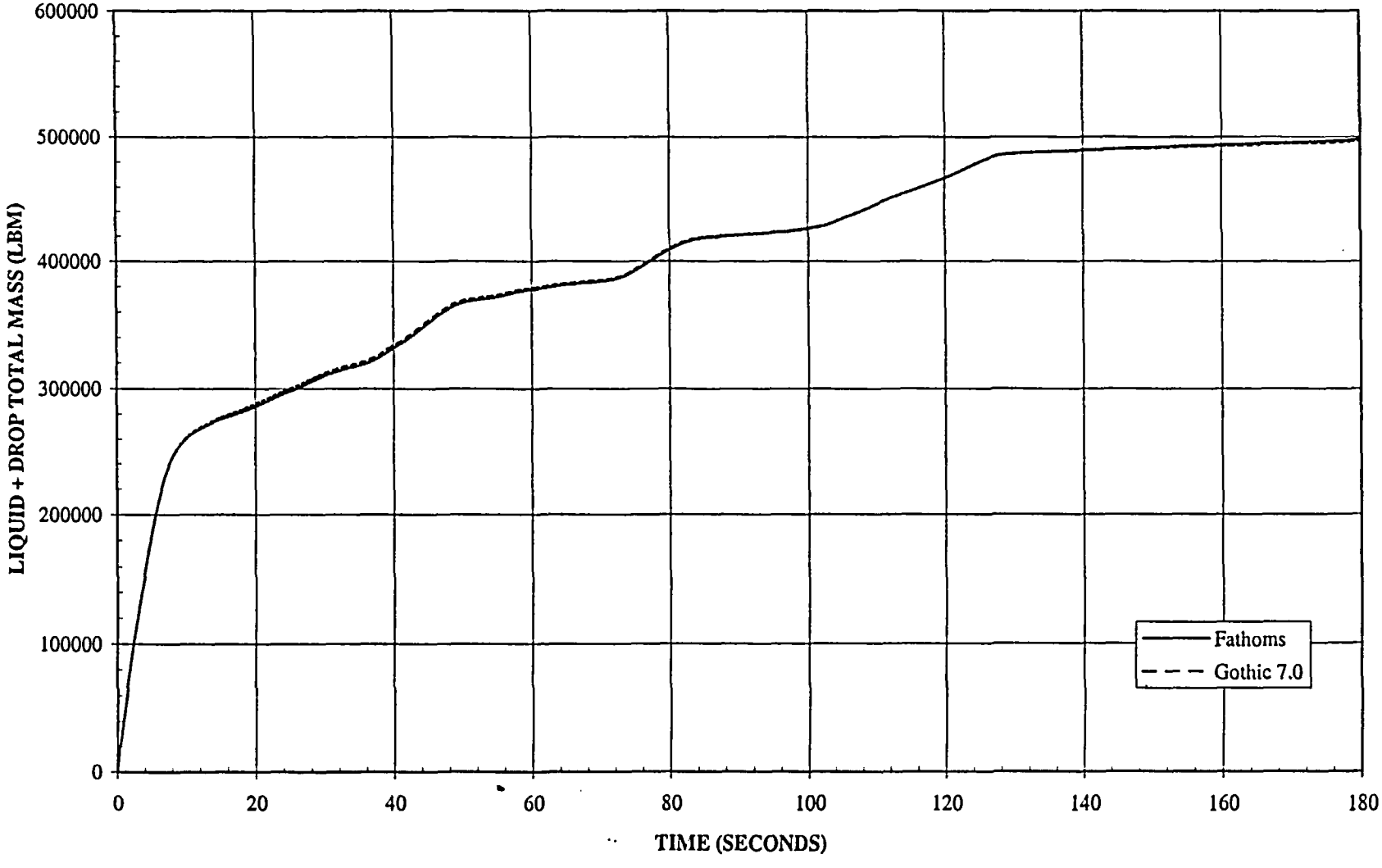
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ 5° ΔTcold Loop Case 2)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

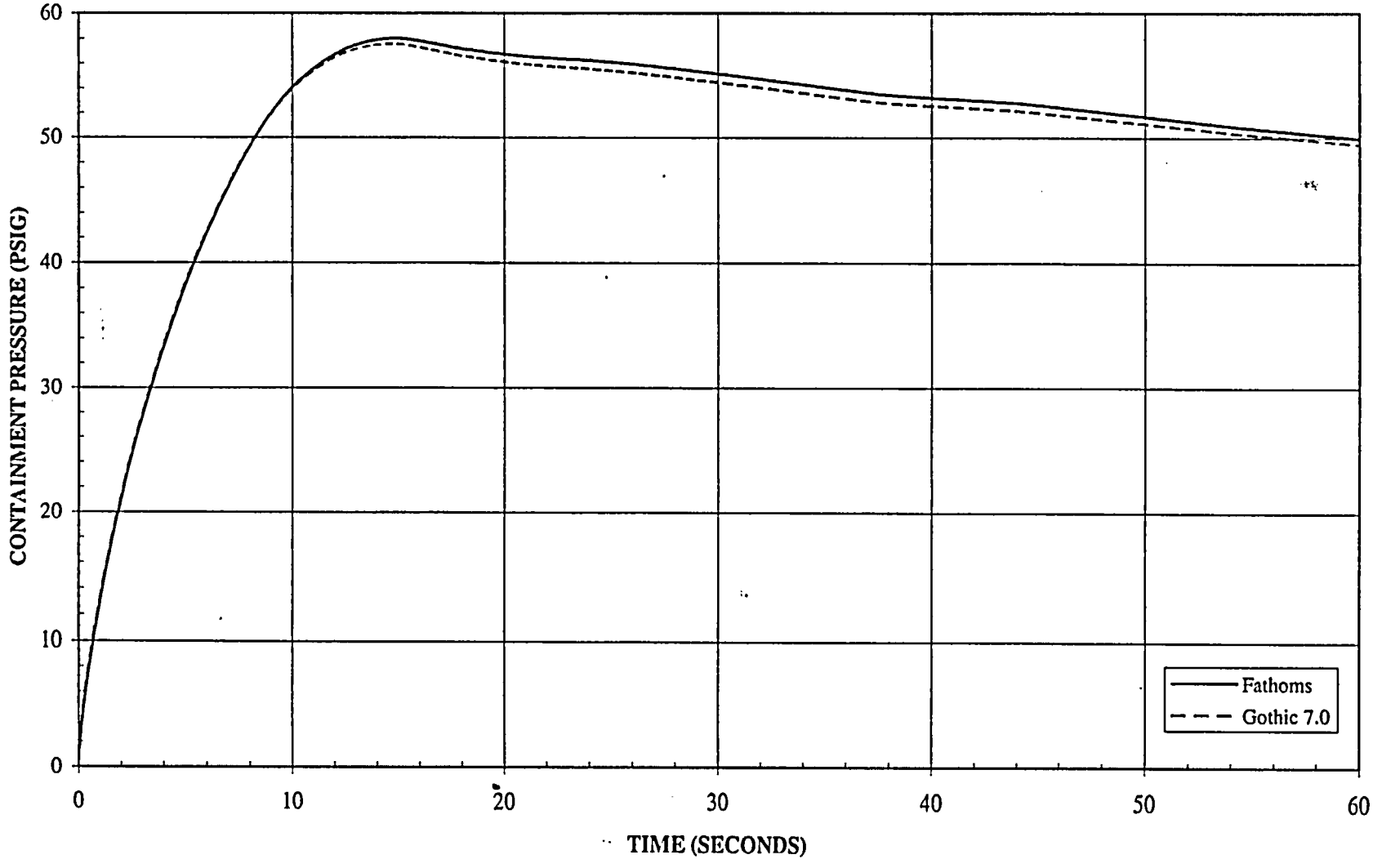
(14.1 ft^2 Break - SG Inlet - High Initial RCS Flow w/ 5° ΔTcold Loop Case 2)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

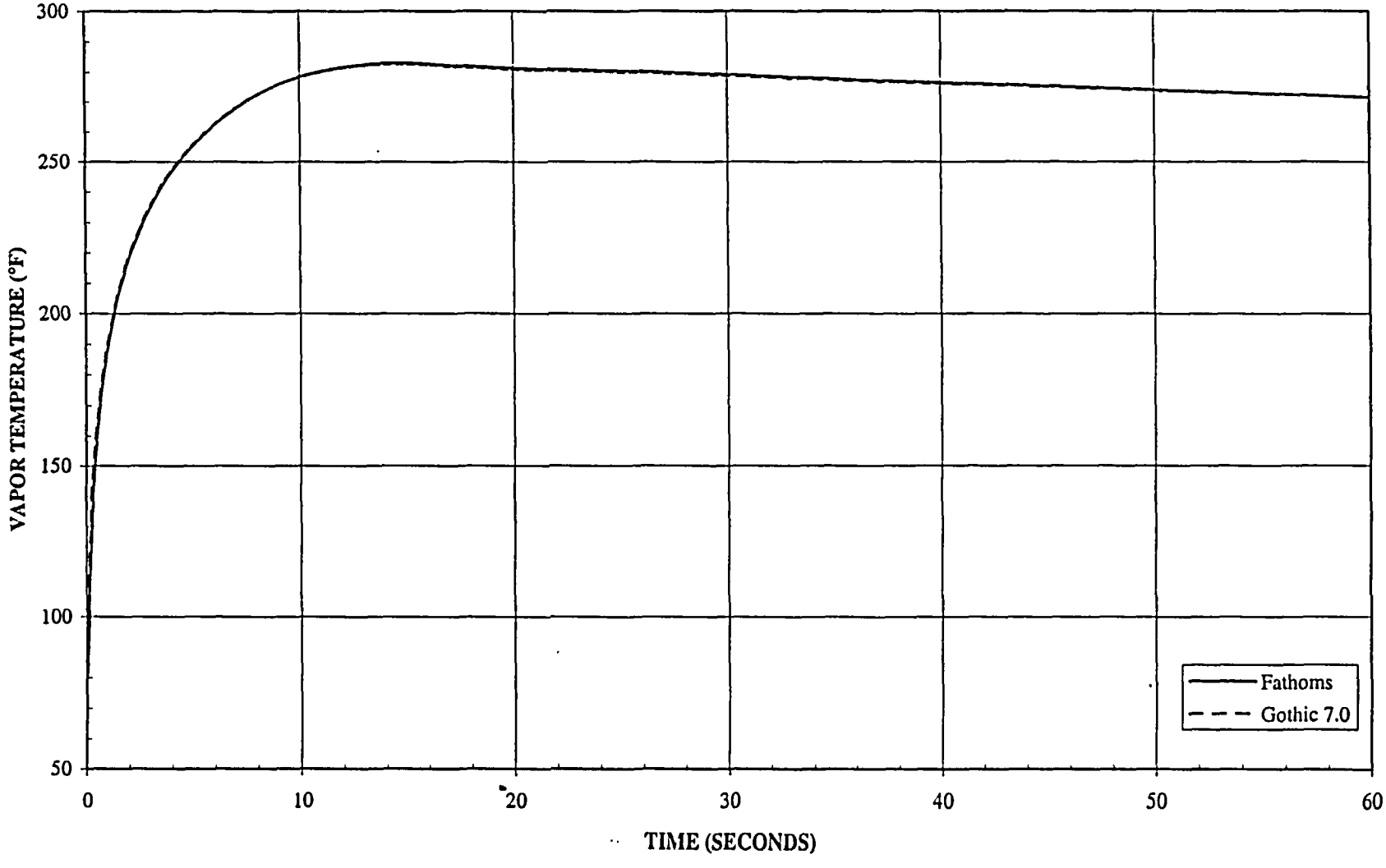
(14.1 ft^2 Break - SG Inlet - Low Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

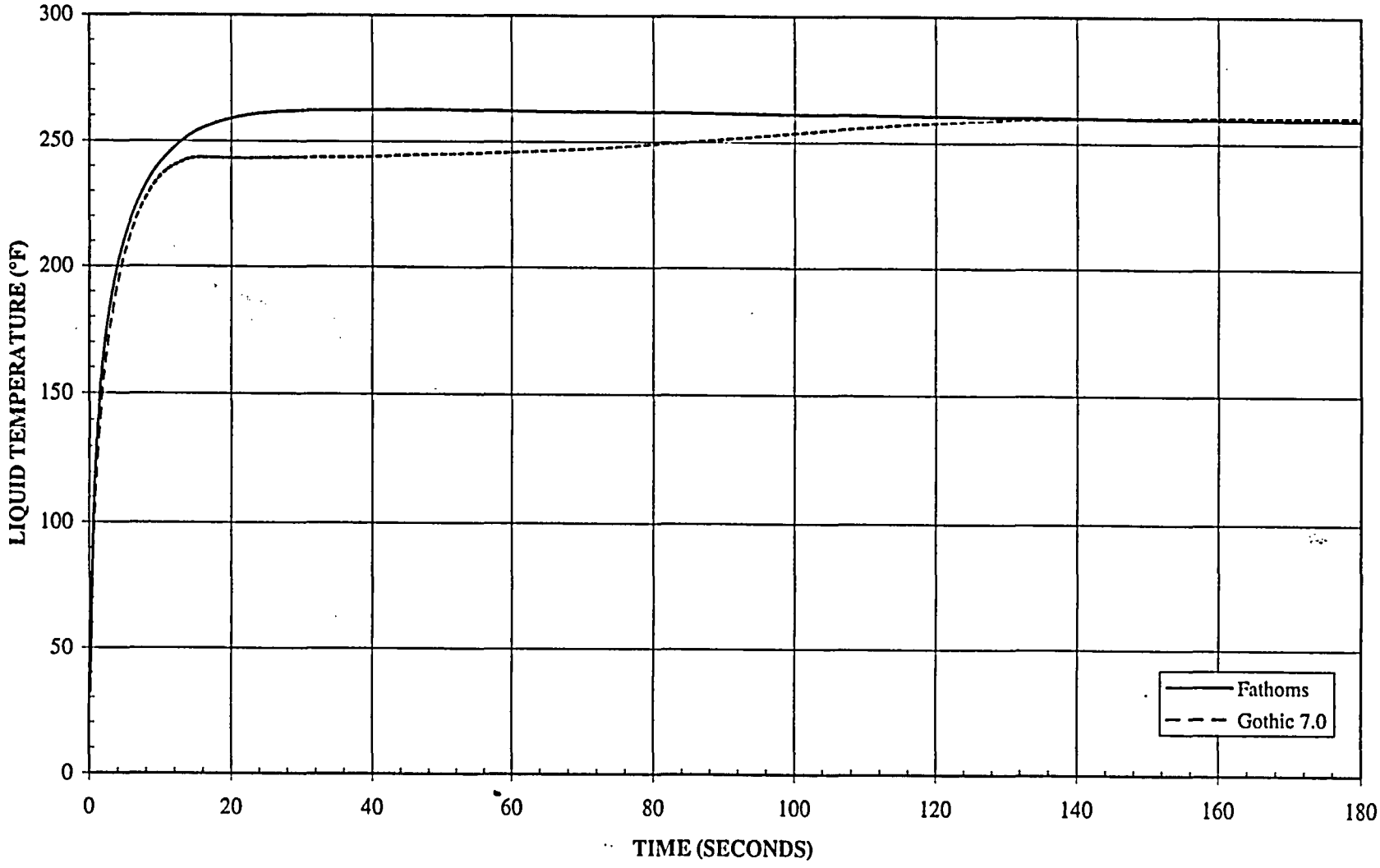
(14.1 ft^2 Break - SG Inlet - Low Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

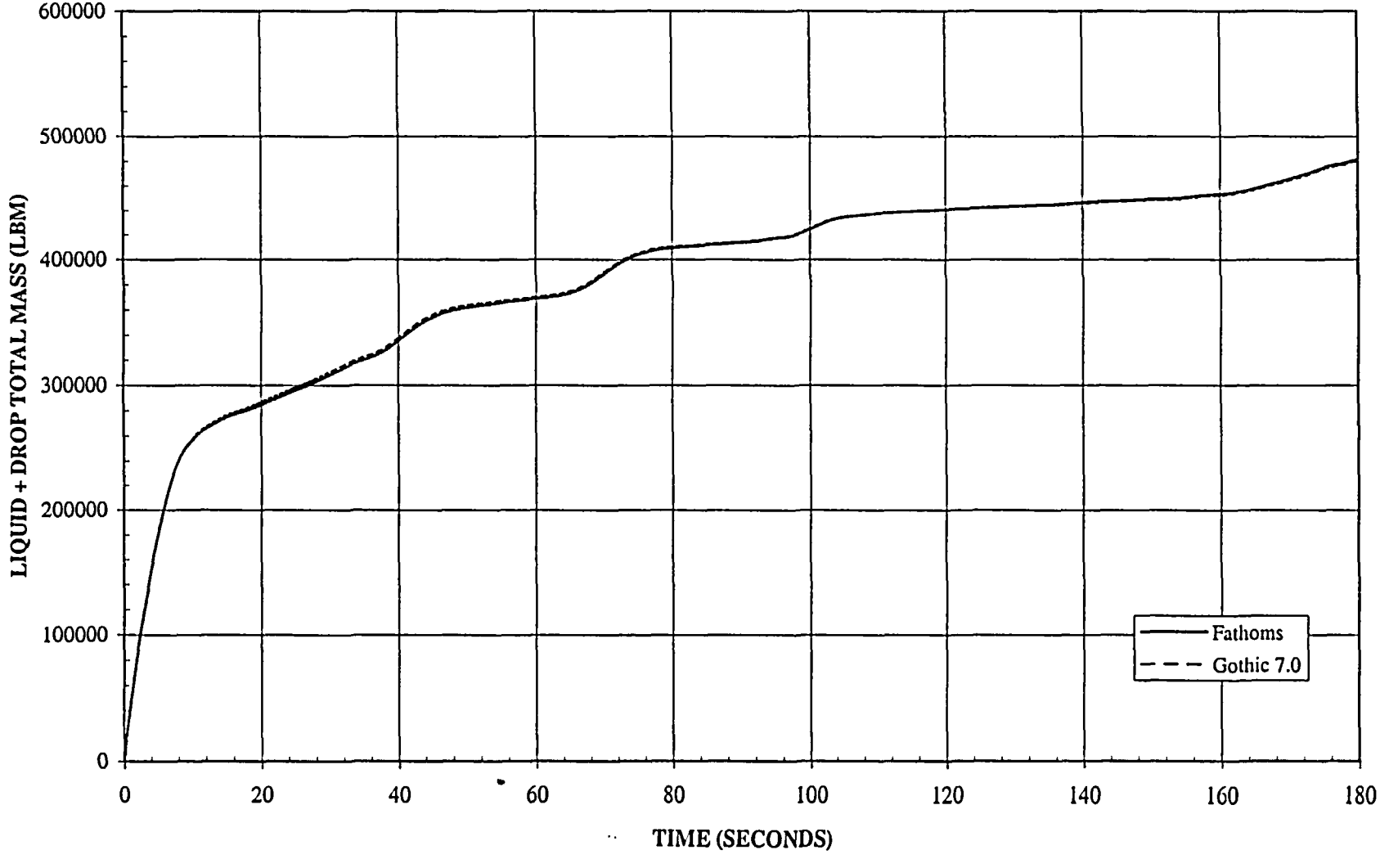
(14.1 ft^2 Break - SG Inlet - Low Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

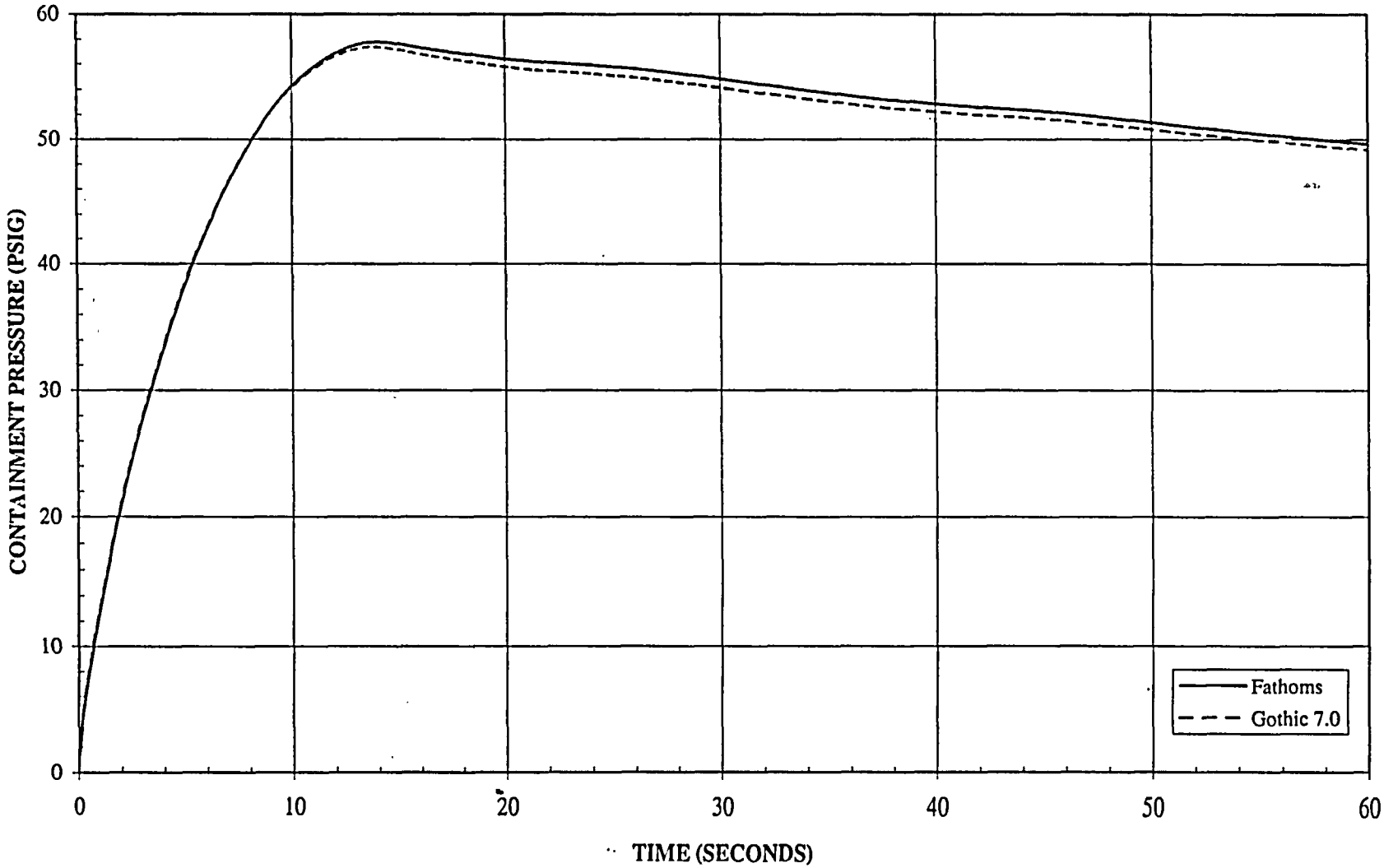
(14.1 ft^2 Break - SG Inlet - Low Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

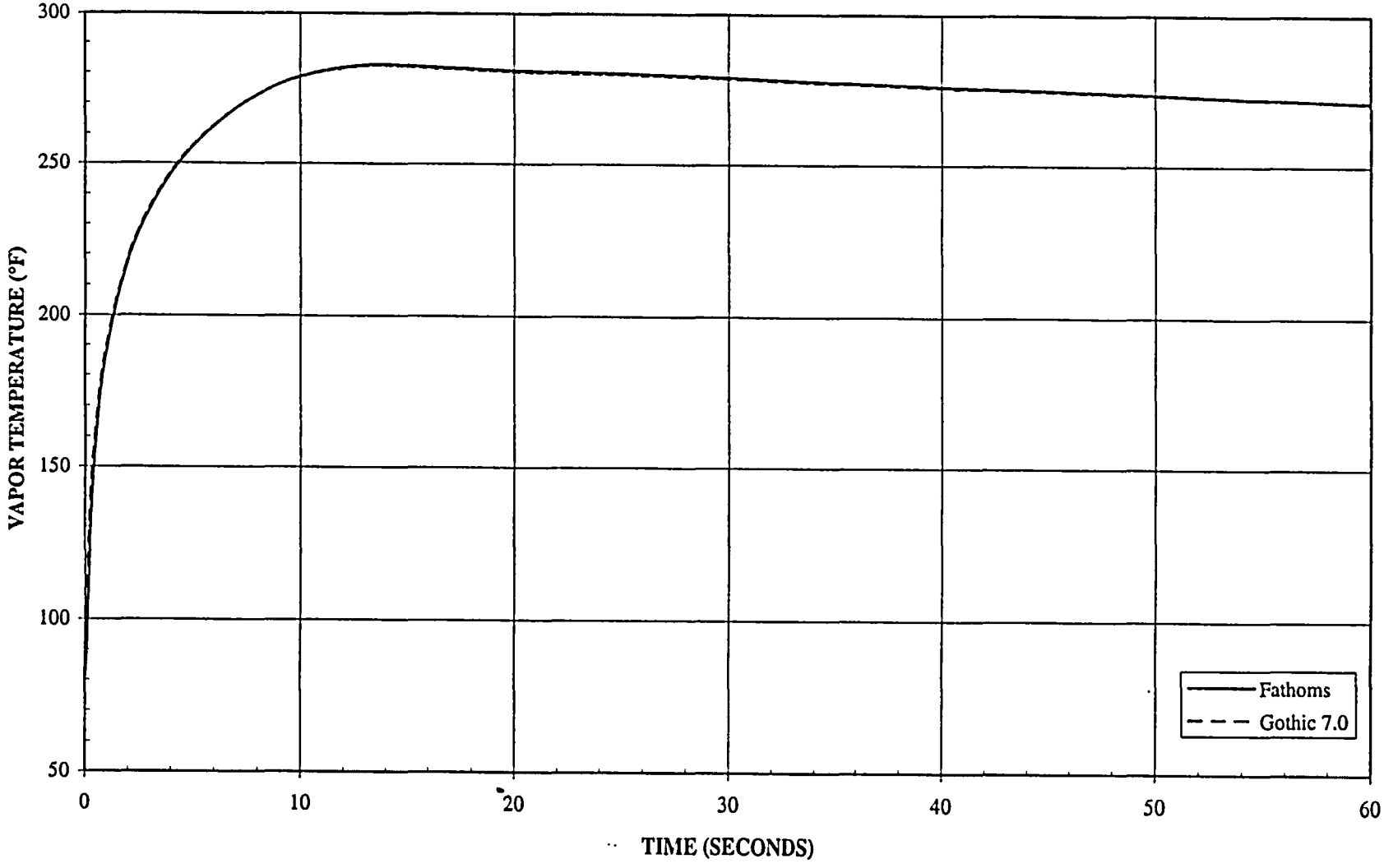
(14.1 ft^2 Break - SG Inlet - Low Initial RCS Flow w/ M3 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

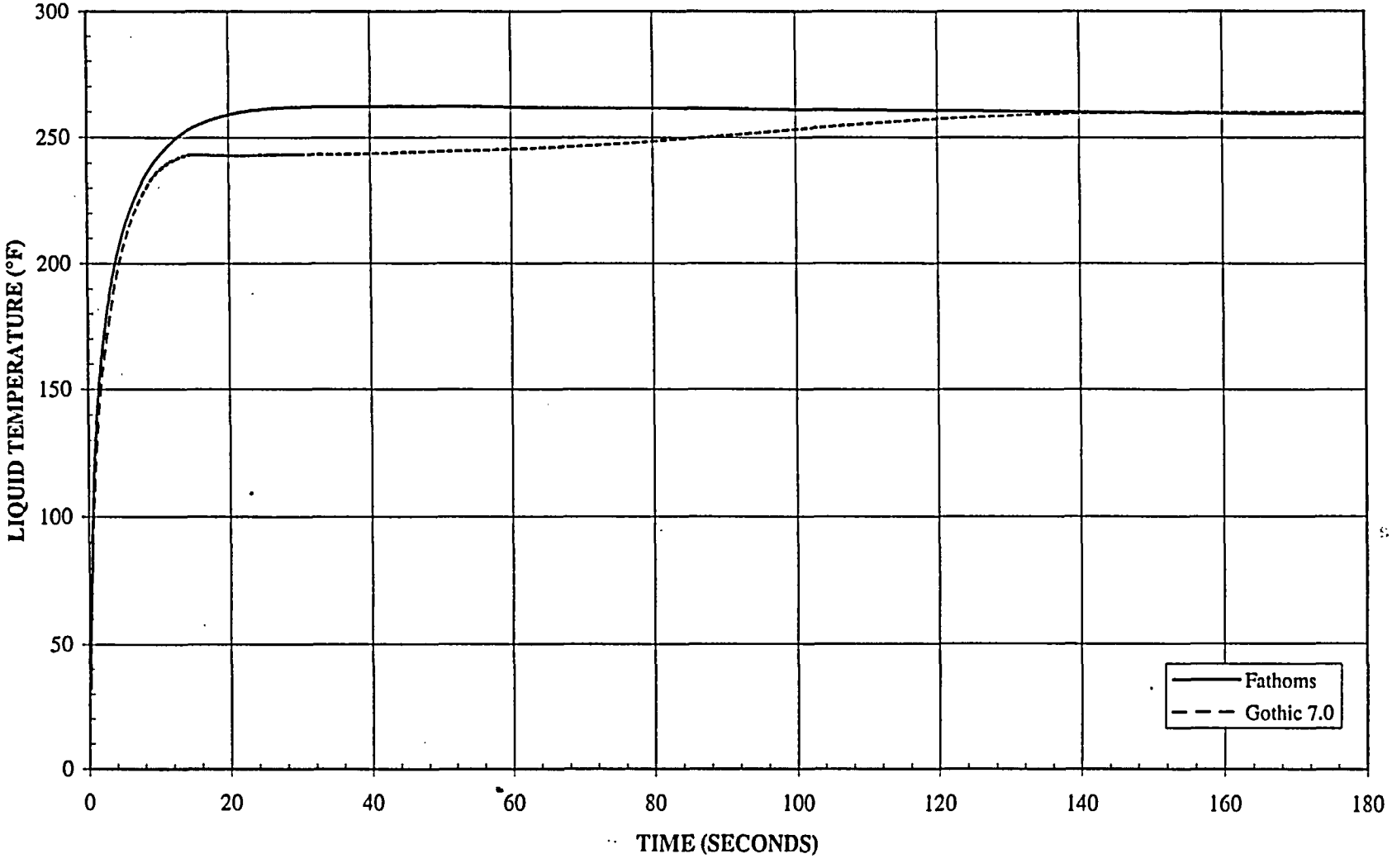
(14.1 ft^2 Break - SG Inlet - Low Initial RCS Flow w/ M3 Pump Degradation Model)

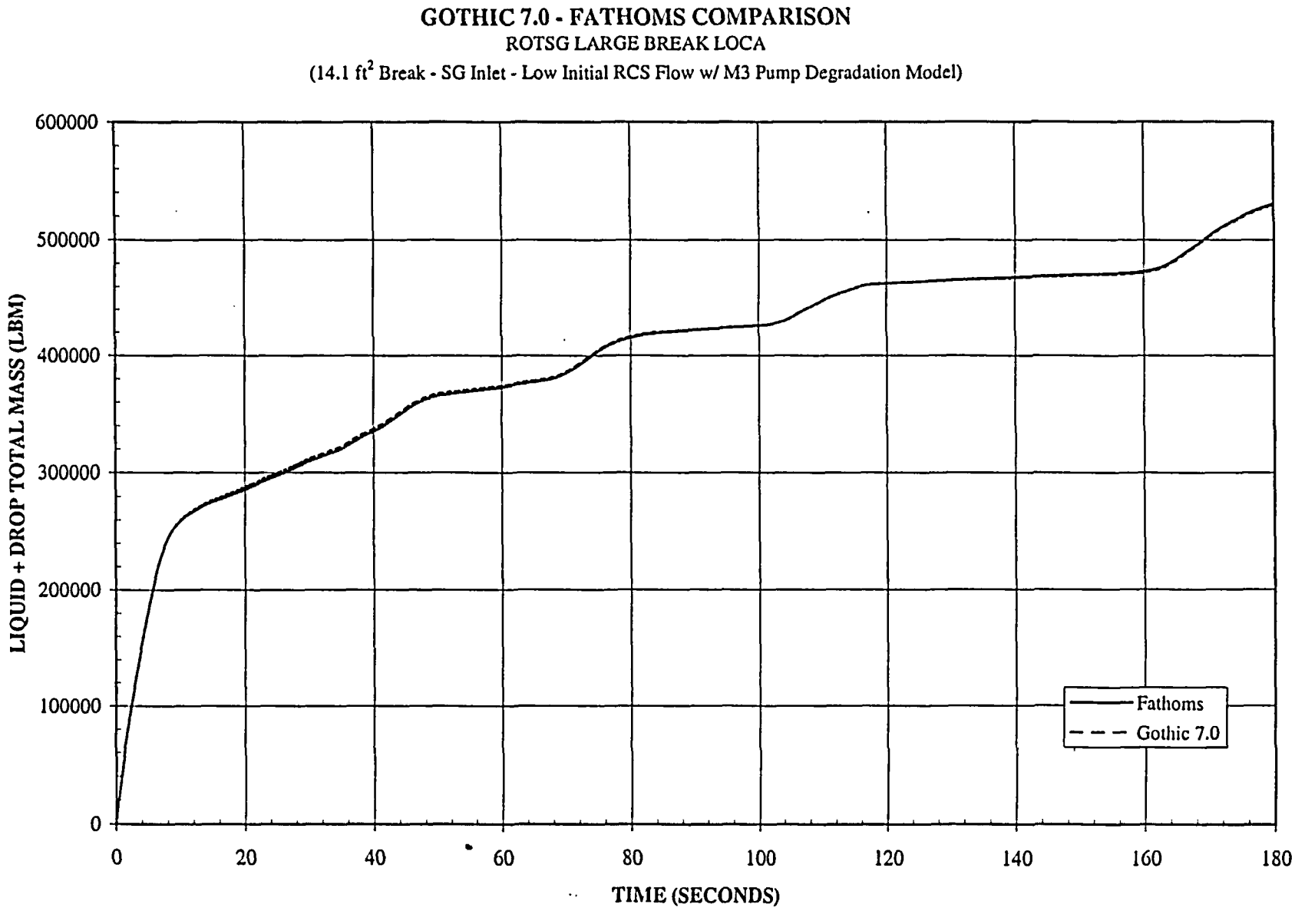


GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

(14.1 ft^2 Break - SG Inlet - Low Initial RCS Flow w/ M3 Pump Degradation Model)

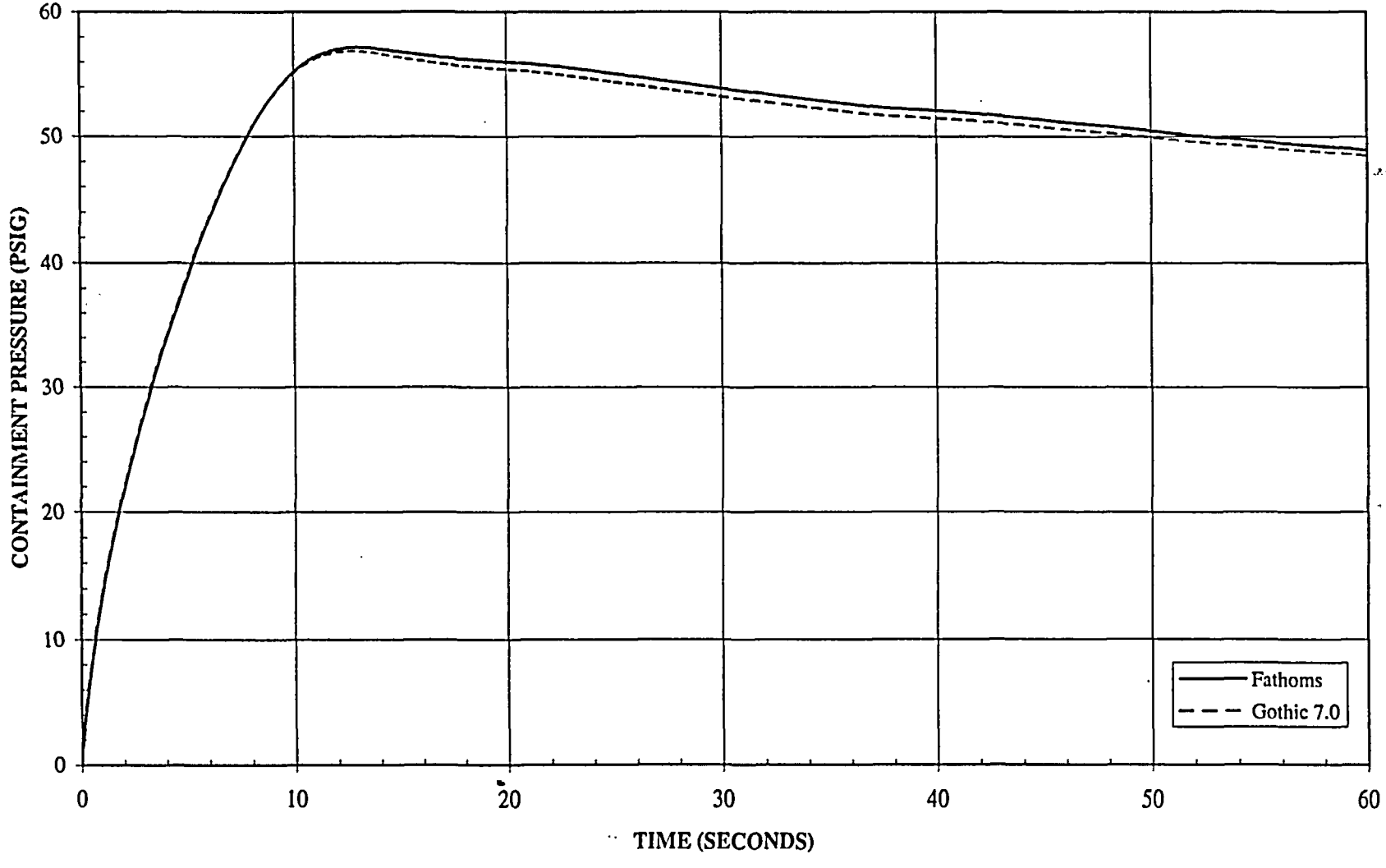



GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG LARGE BREAK LOCA
(14.1 ft² Break - SG Inlet - Low Initial RCS Flow w/ M3 Pump Degradation Model)
LIQUID + DROP TOTAL MASS (LBM)
600000
500000
400000
300000
200000
100000
0
0
20
40
60
80
100
120
140
160
180
TIME (SECONDS)
Fathoms
Gothic 7.0

GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

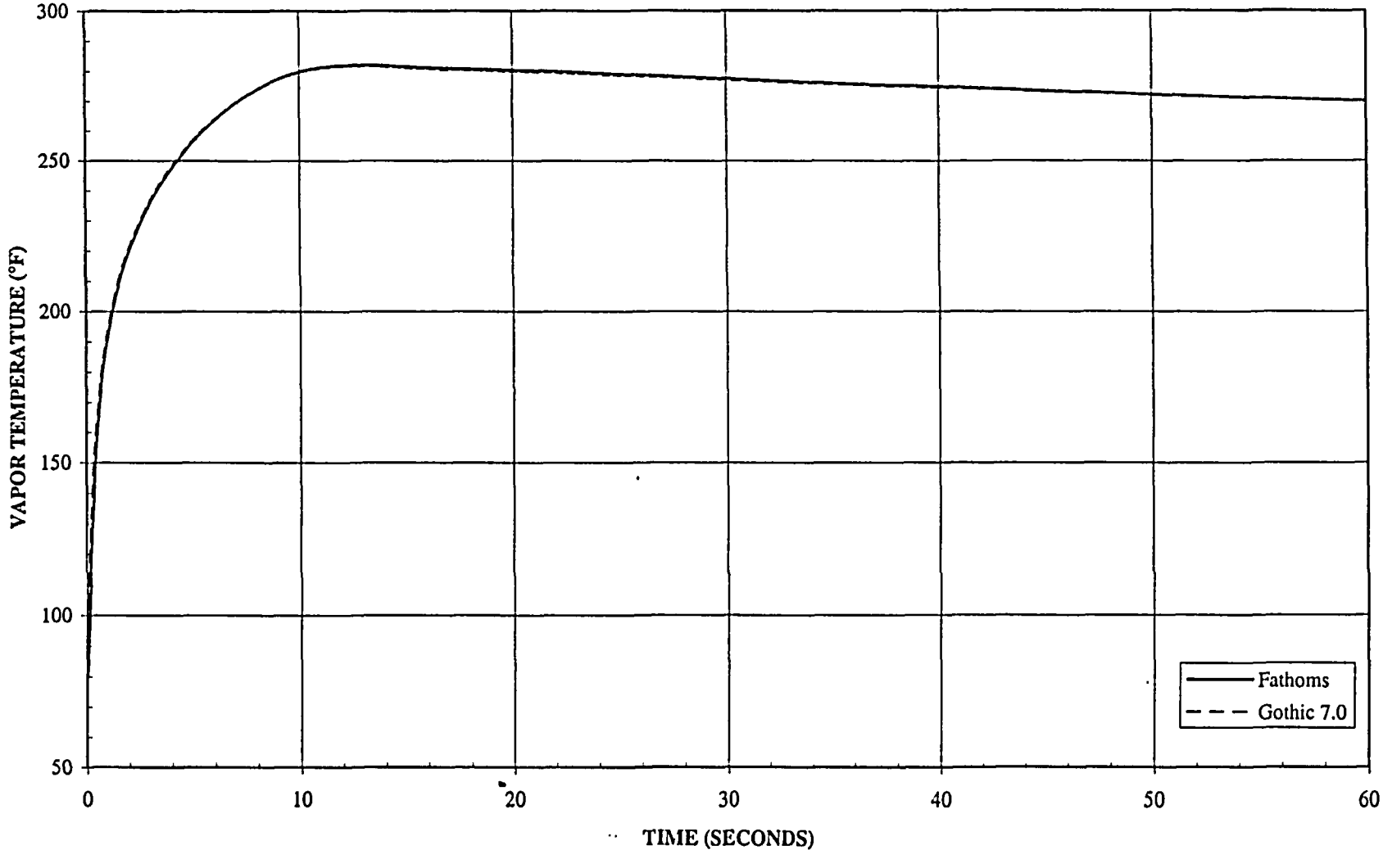
(14.1 ft^2 Break - RV Outlet - Low Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

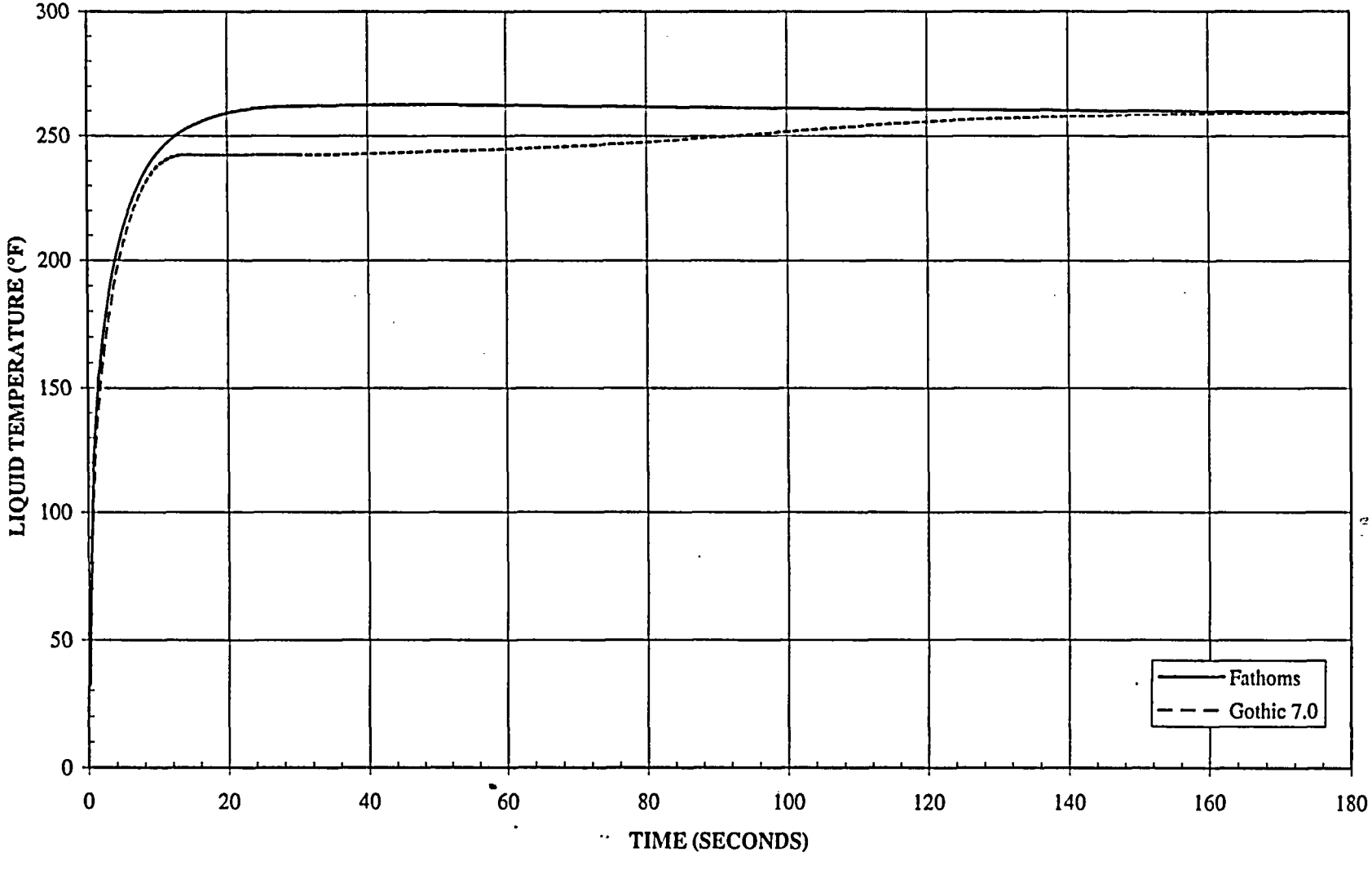
(14.1 ft^2 Break - RV Outlet - Low Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

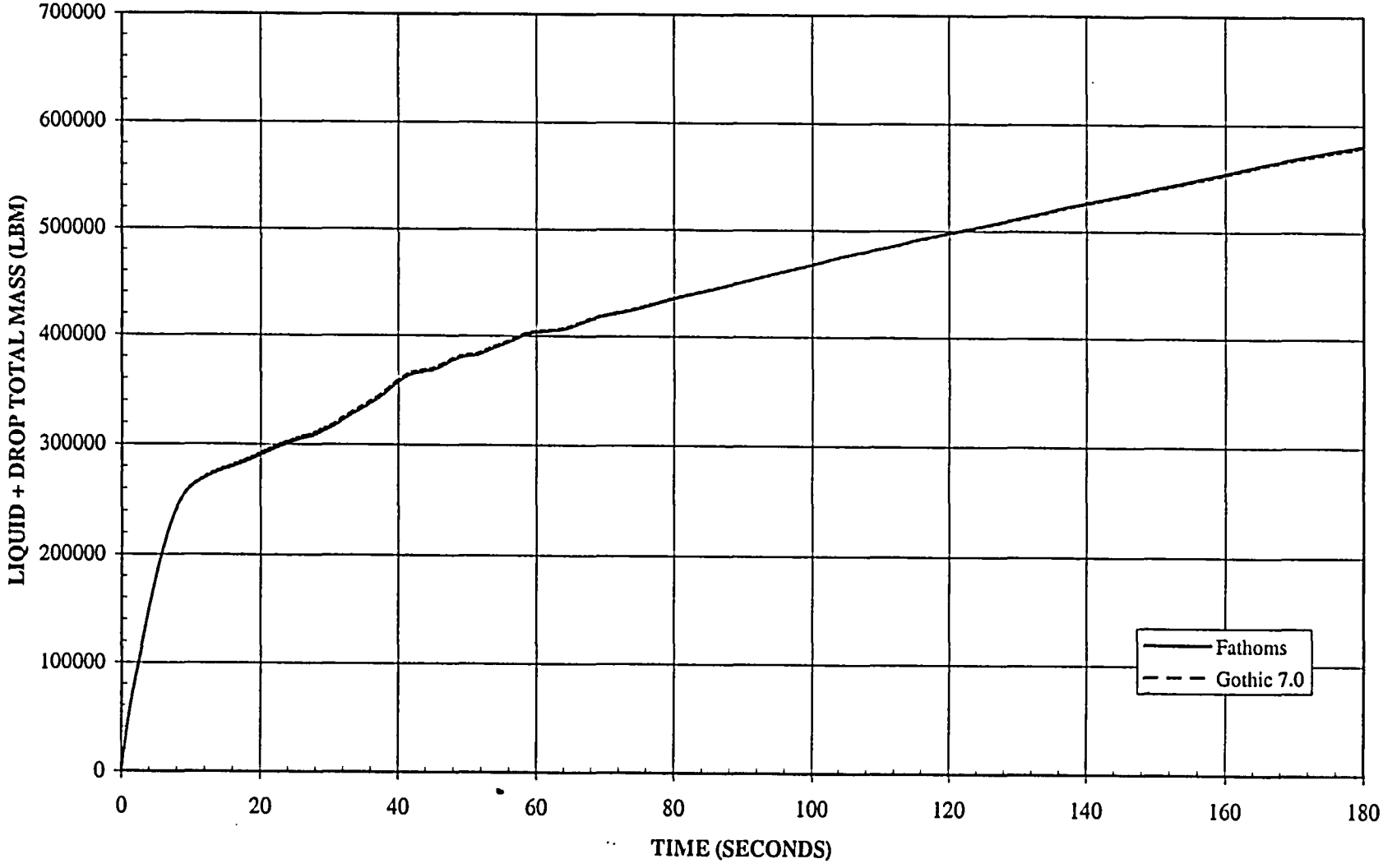
(14.1 ft^2 Break - RV Outlet - Low Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

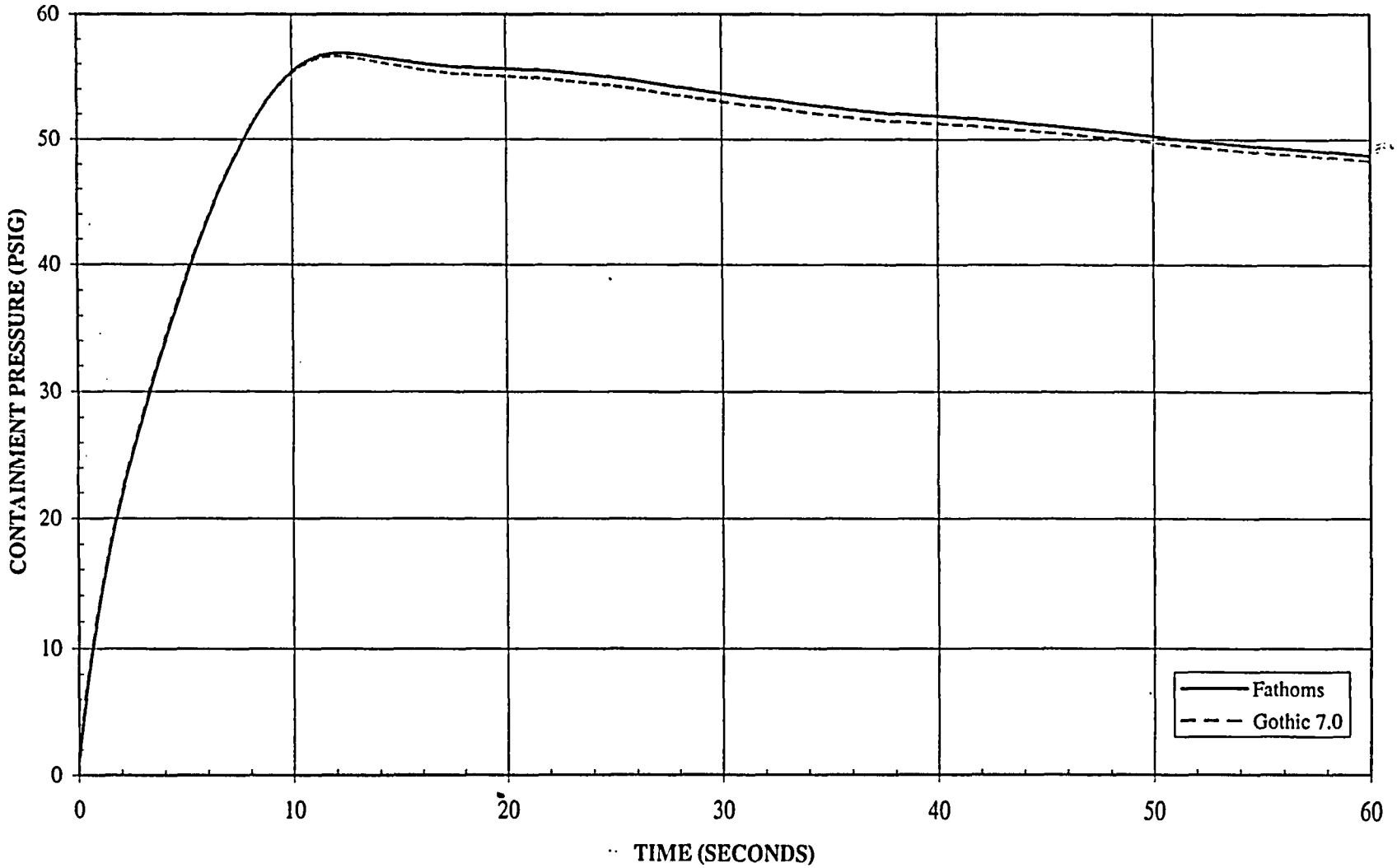
(14.1 ft^2 Break - RV Outlet - Low Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

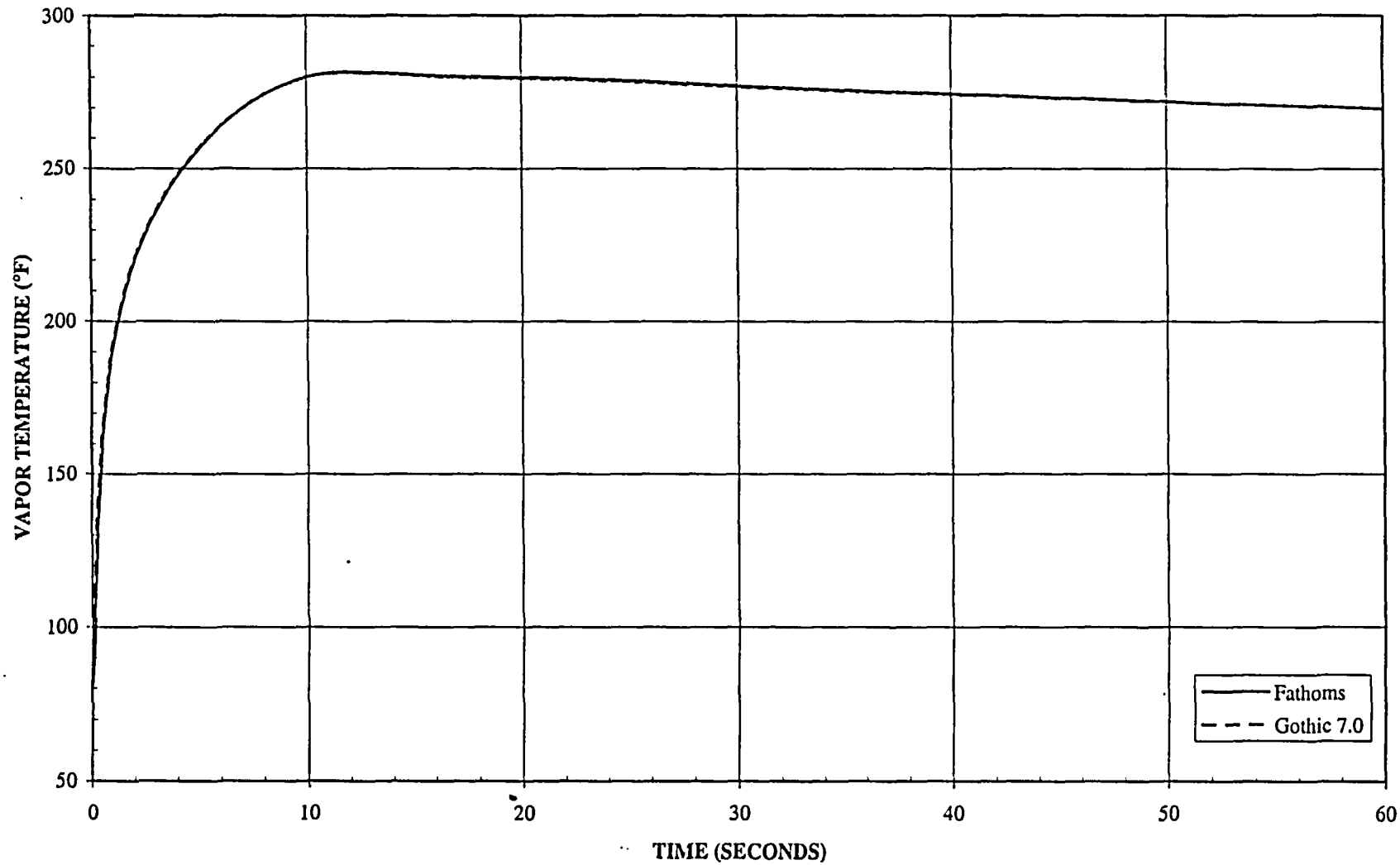
(14.1 ft^2 Break - RV Outlet - Low Initial RCS Flow w/ M3 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

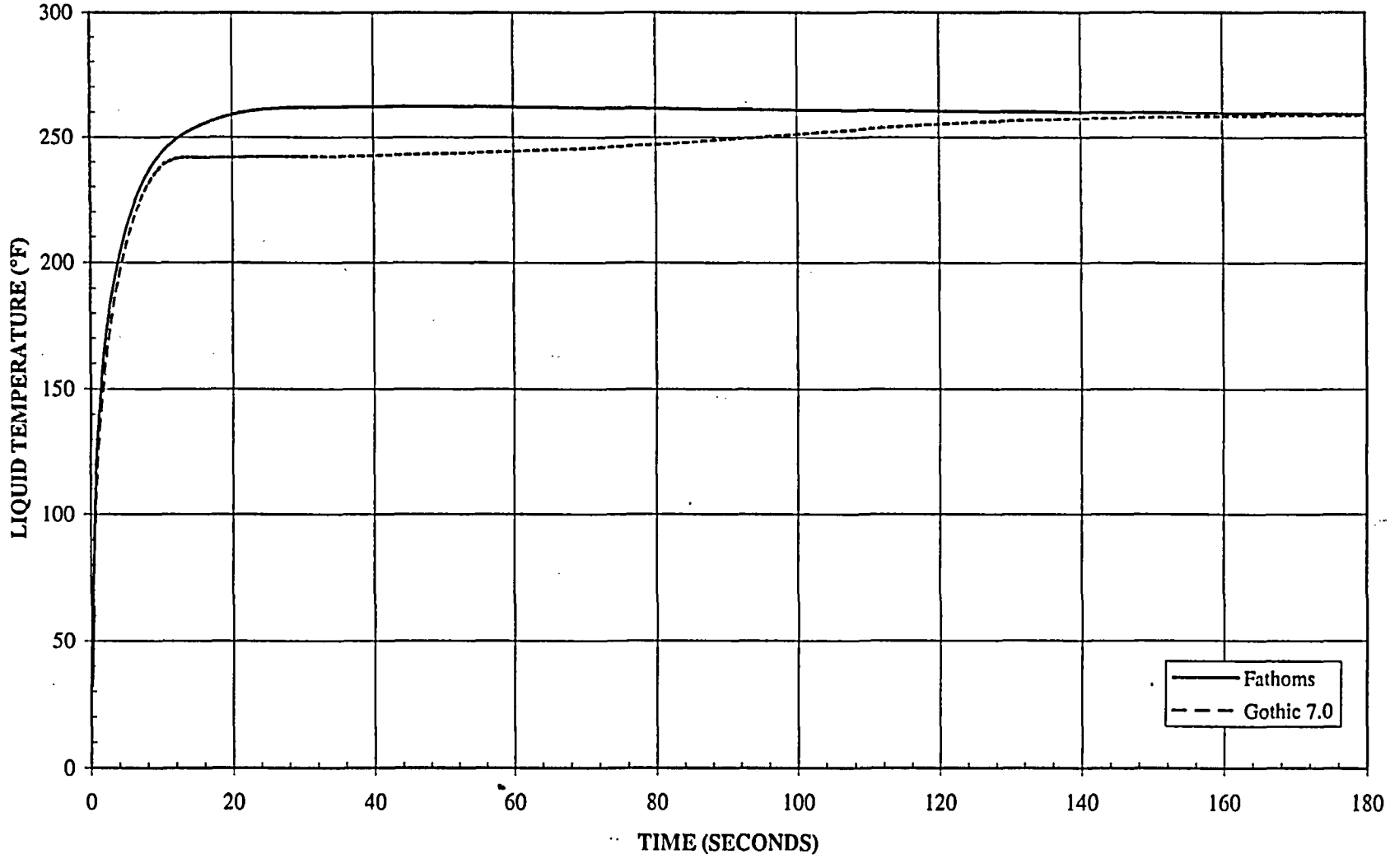
(14.1 ft^2 Break - RV Outlet - Low Initial RCS Flow w/ M3 Pump Degradation Model)

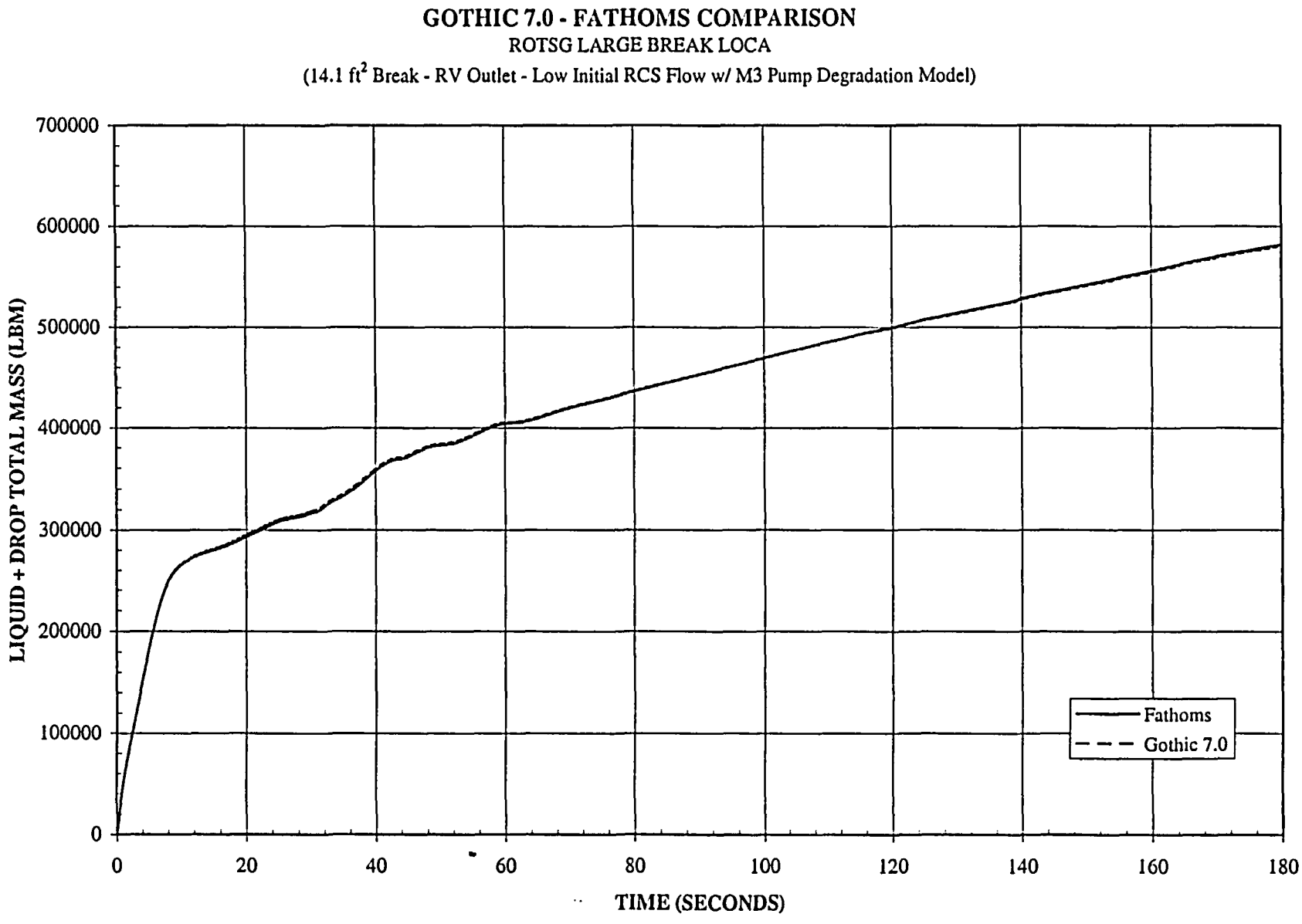


GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

(14.1 ft^2 Break - RV Outlet - Low Initial RCS Flow w/ M3 Pump Degradation Model)

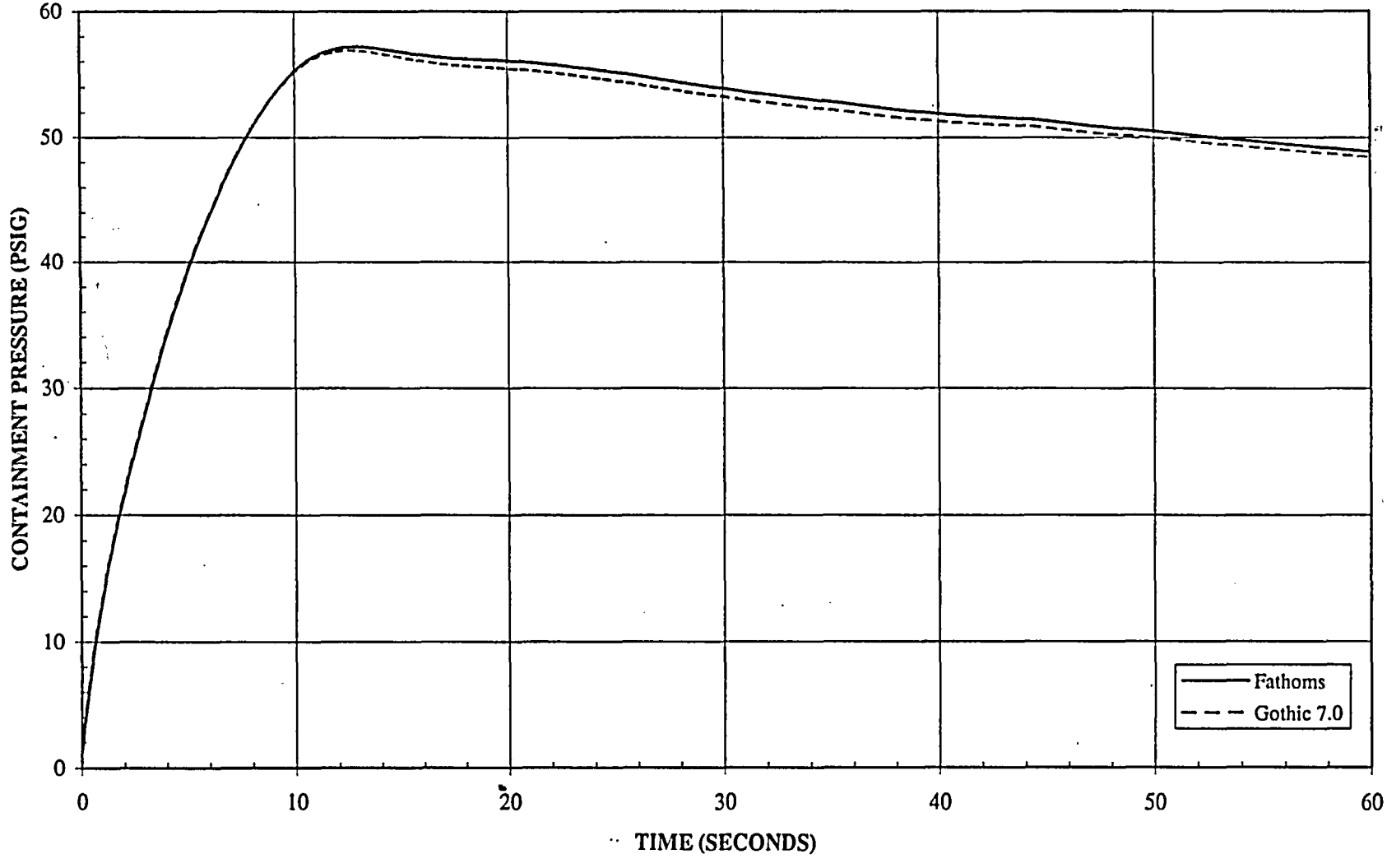



GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG LARGE BREAK LOCA
(14.1 ft² Break - RV Outlet - Low Initial RCS Flow w/ M3 Pump Degradation Model)
LIQUID + DROP TOTAL MASS (LBM)
700000
600000
500000
400000
300000
200000
100000
0
0
20
40
60
80
100
120
140
160
180
TIME (SECONDS)
Fathoms
Gothic 7.0

GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

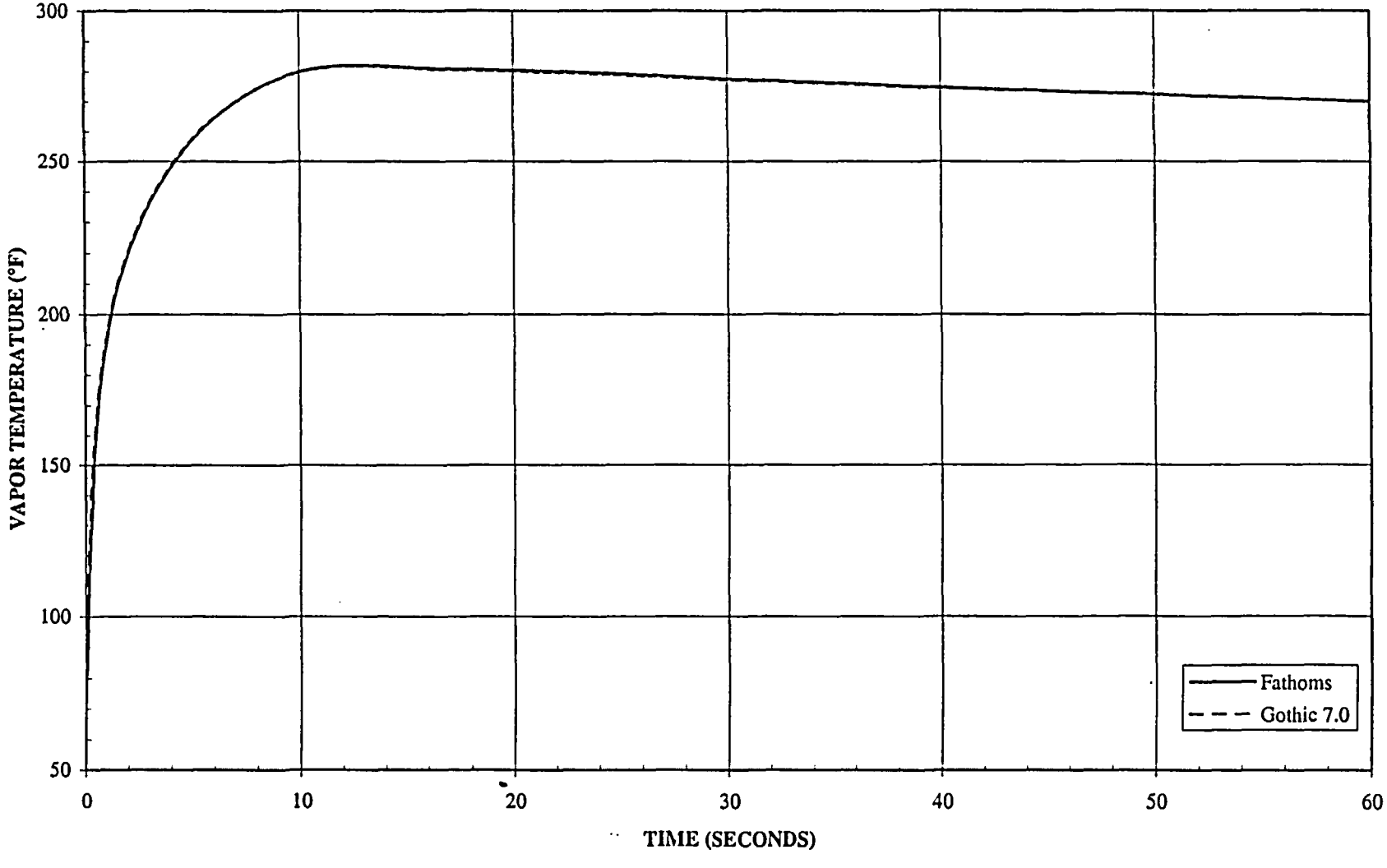
(14.1 ft^2 Break - RV Outlet - High Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

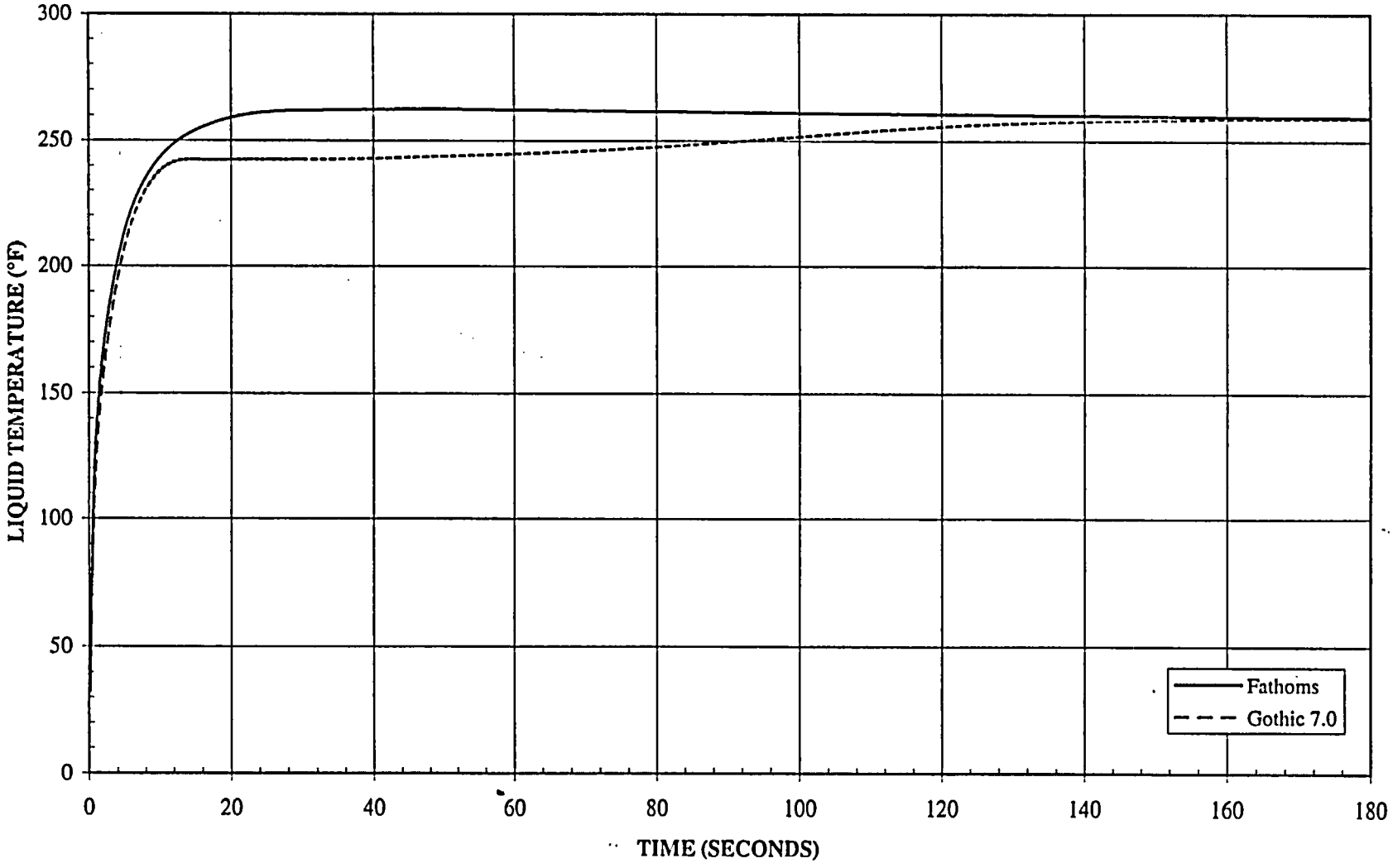
(14.1 ft^2 Break - RV Outlet - High Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

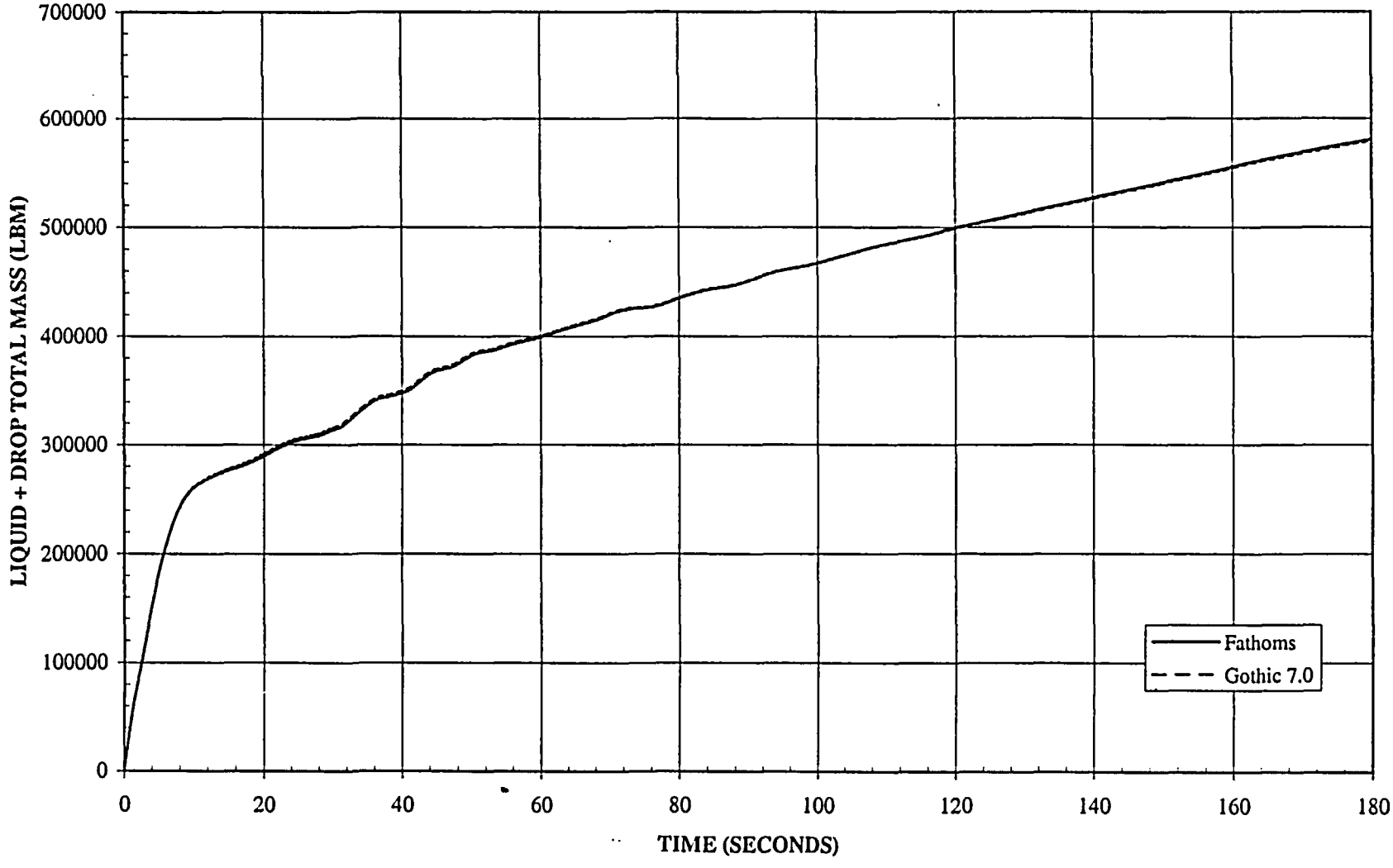
(14.1 ft^2 Break - RV Outlet - High Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

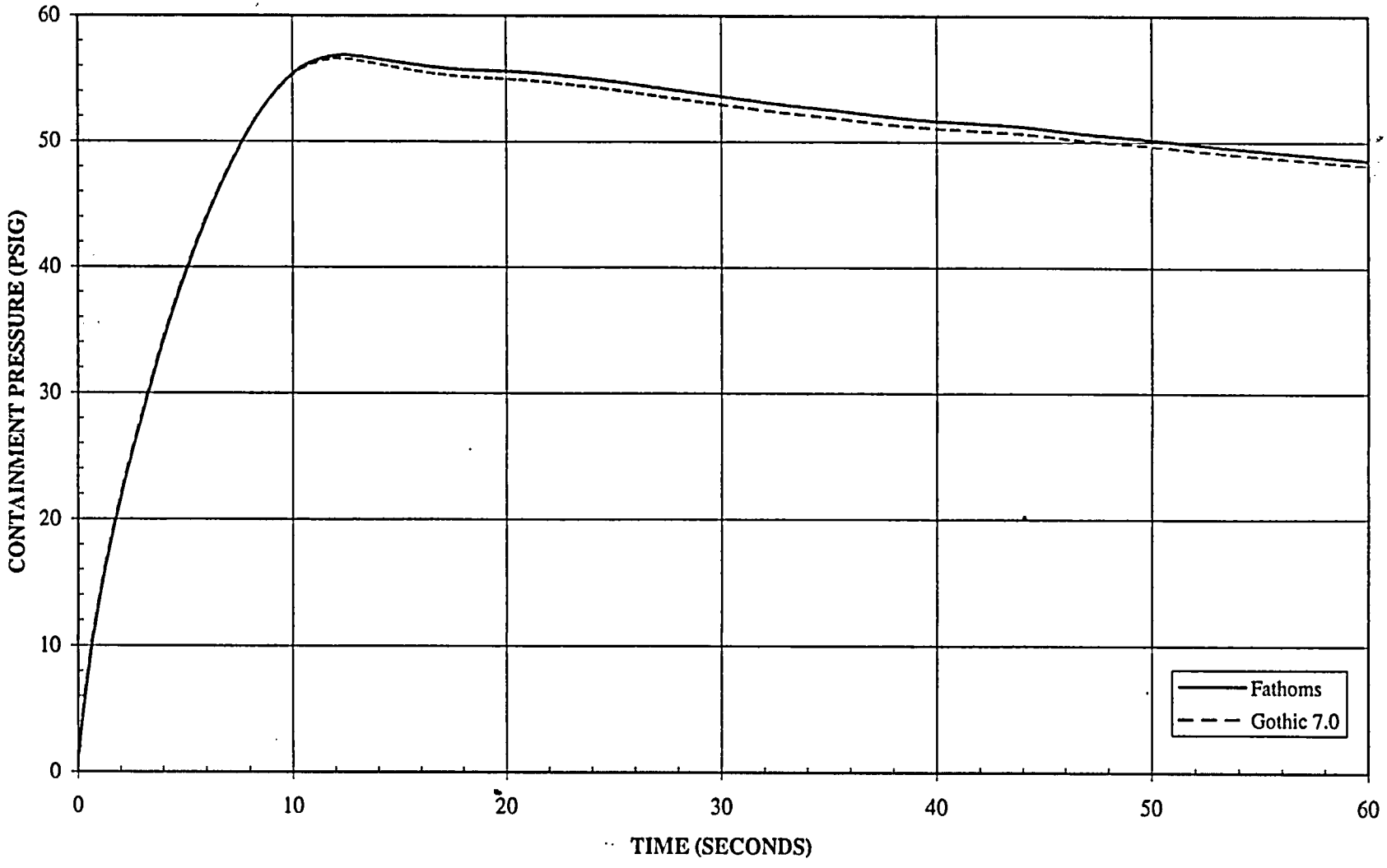
(14.1 ft^2 Break - RV Outlet - High Initial RCS Flow w/ M1 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

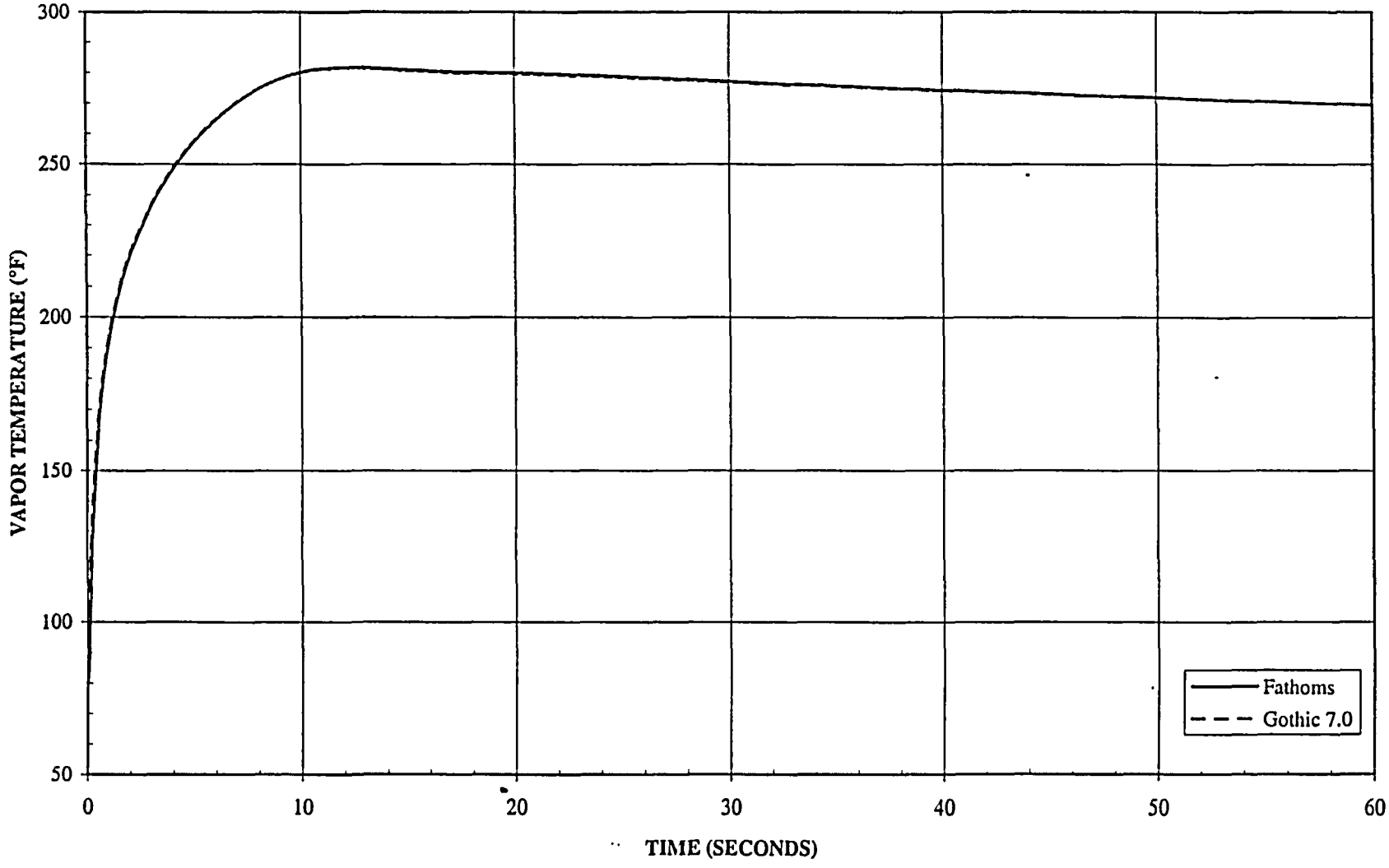
(14.1 ft^2 Break - RV Outlet - High Initial RCS Flow w/ M3 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

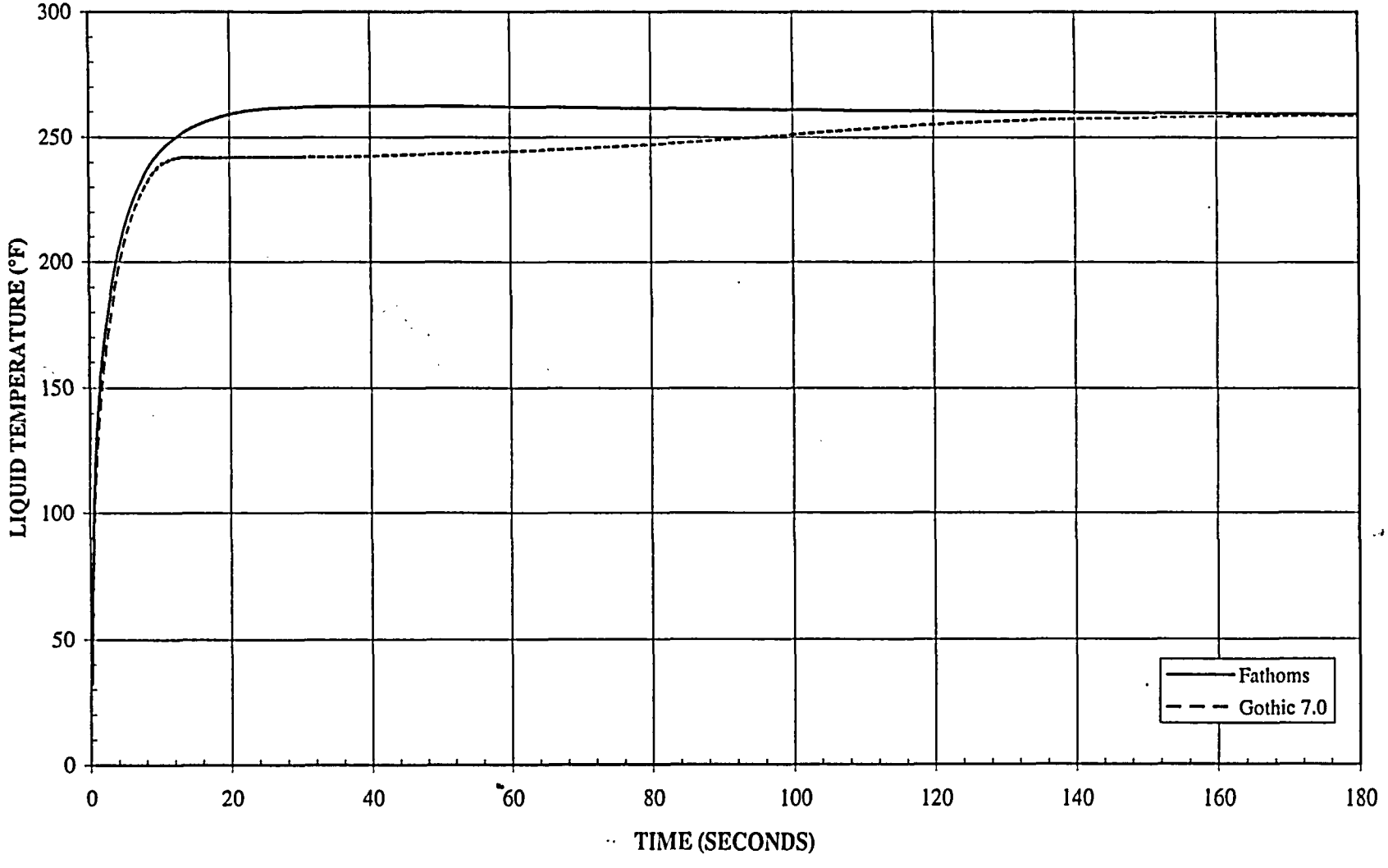
(14.1 ft^2 Break - RV Outlet - High Initial RCS Flow w/ M3 Pump Degradation Model)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

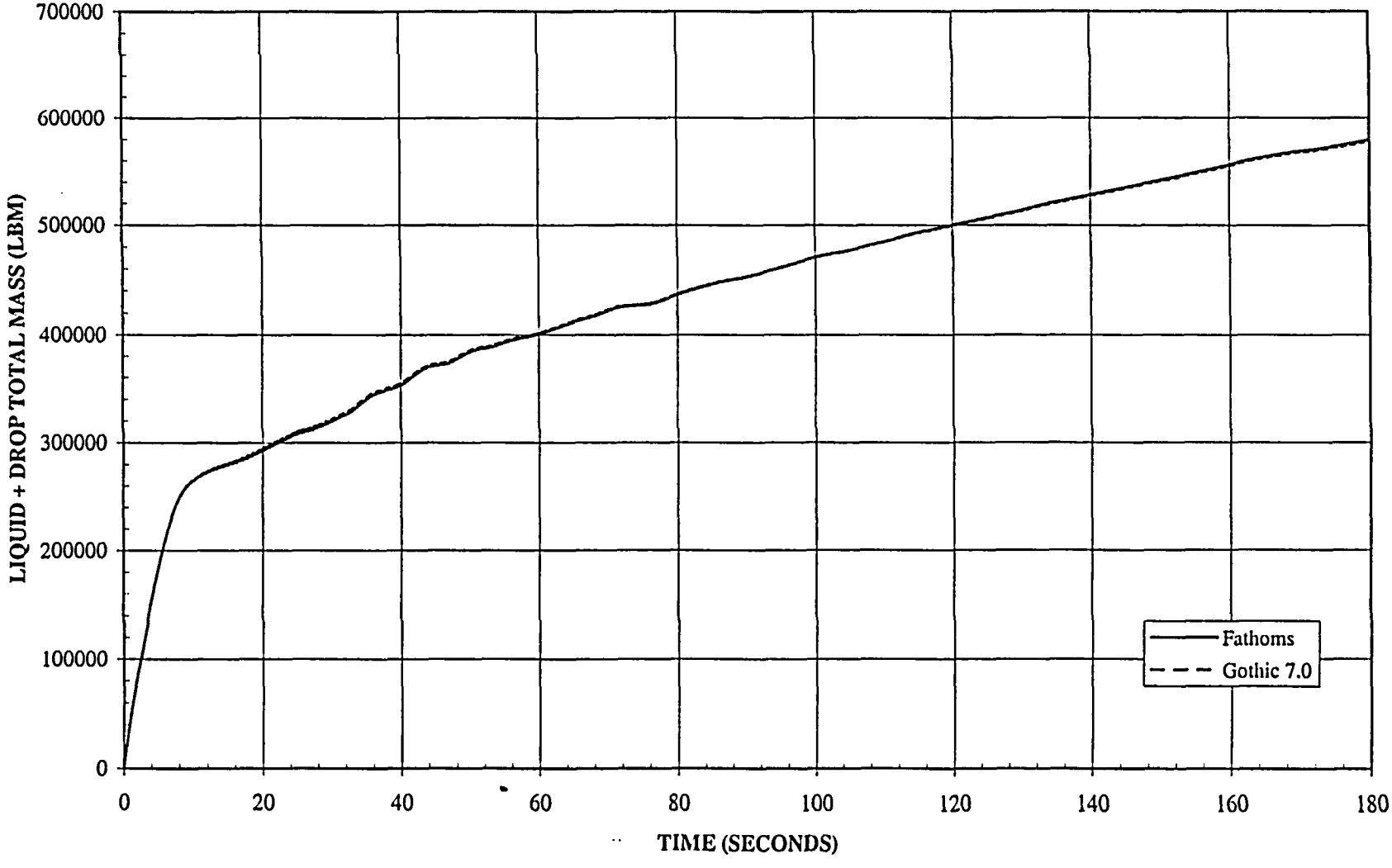
(14.1 ft^2 Break - RV Outlet - High Initial RCS Flow w/ M3 Pump Degradation Model)

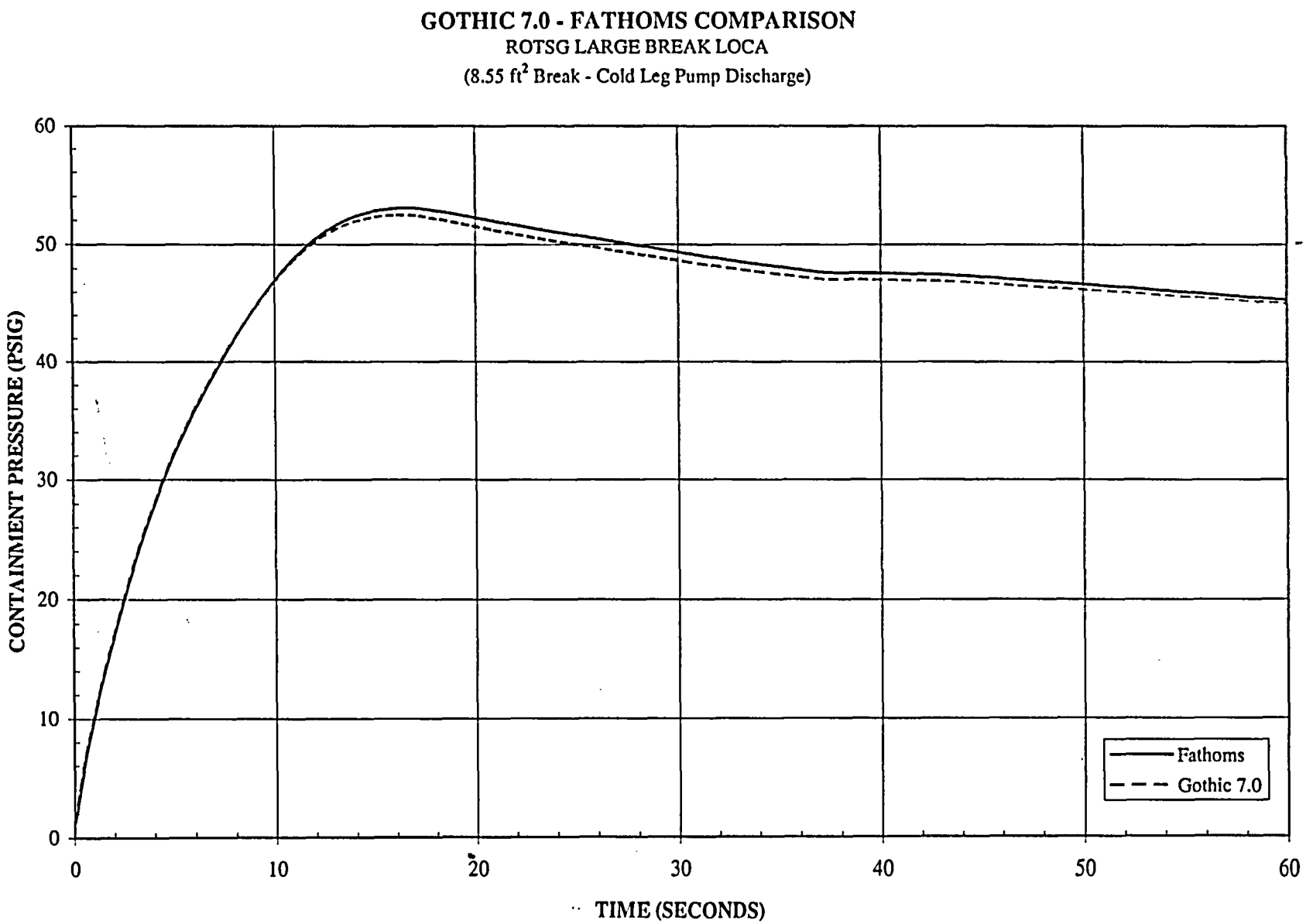


GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

(14.1 ft^2 Break - RV Outlet - High Initial RCS Flow w/ M3 Pump Degradation Model)

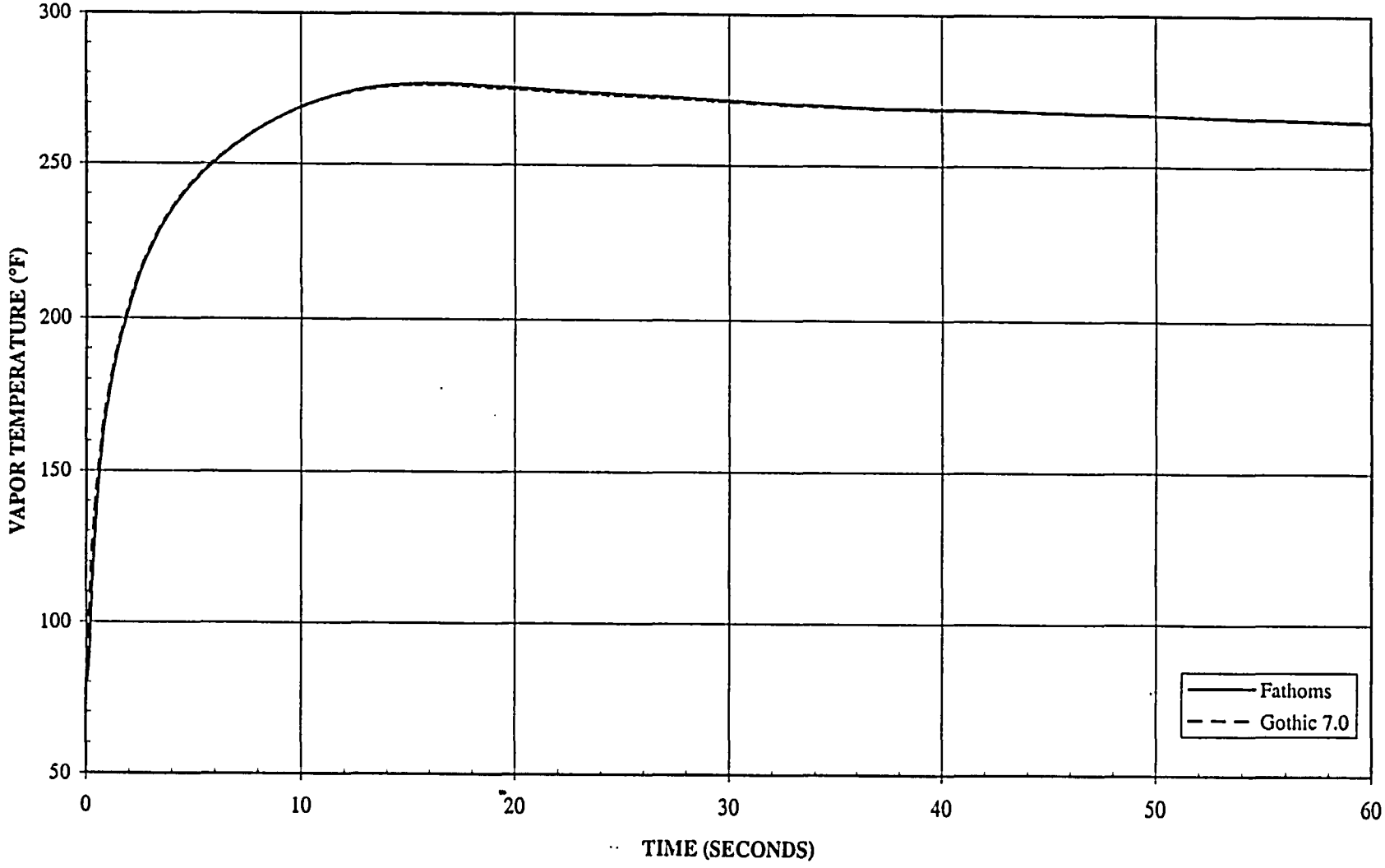



GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG LARGE BREAK LOCA
(8.55 ft² Break - Cold Leg Pump Discharge)
CONTAINMENT PRESSURE (PSIG)
TIME (SECONDS)
0
10
20
30
40
50
60
Fathoms
Gothic 7.0

GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

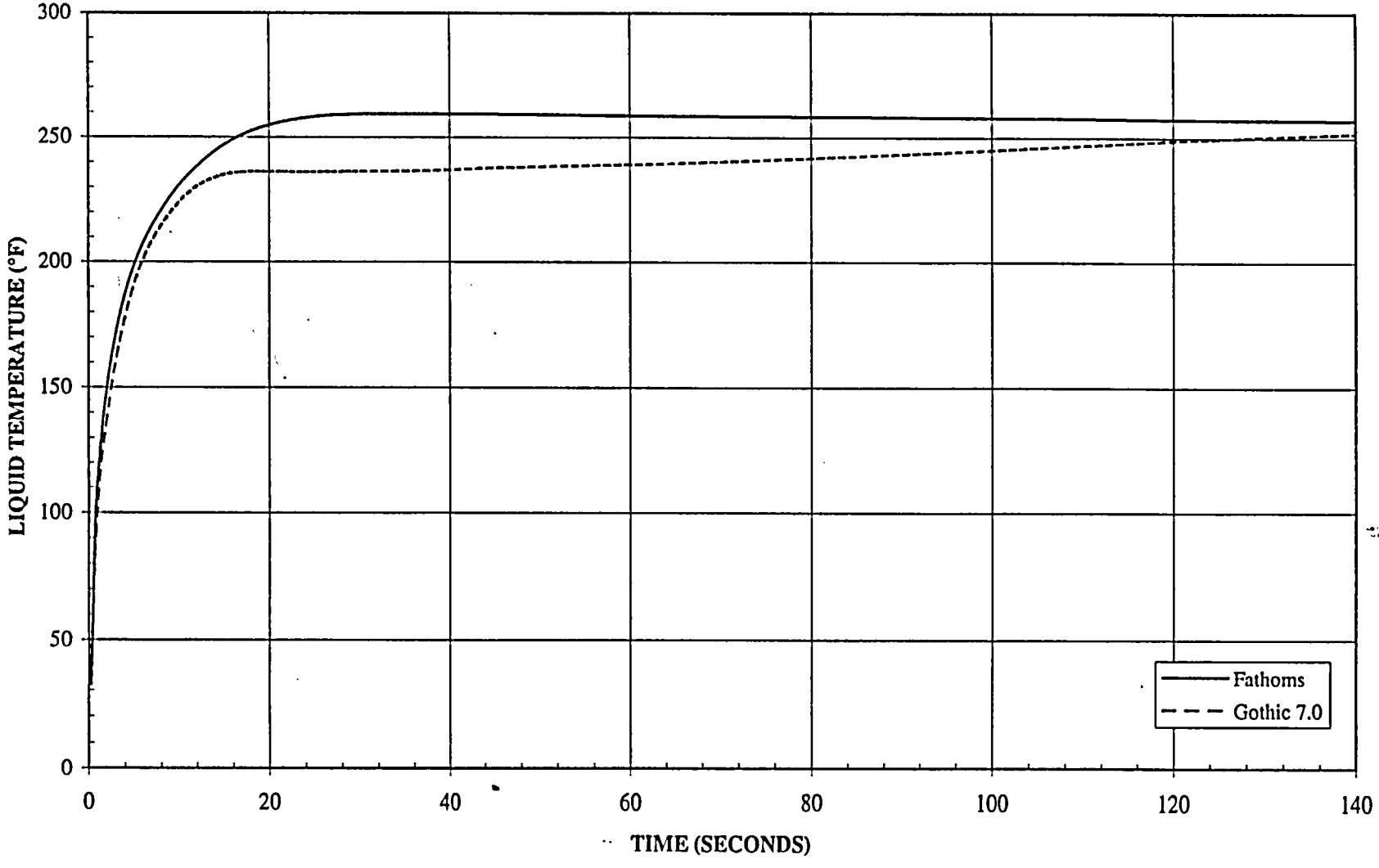
(8.55 ft^2 Break - Cold Leg Pump Discharge)

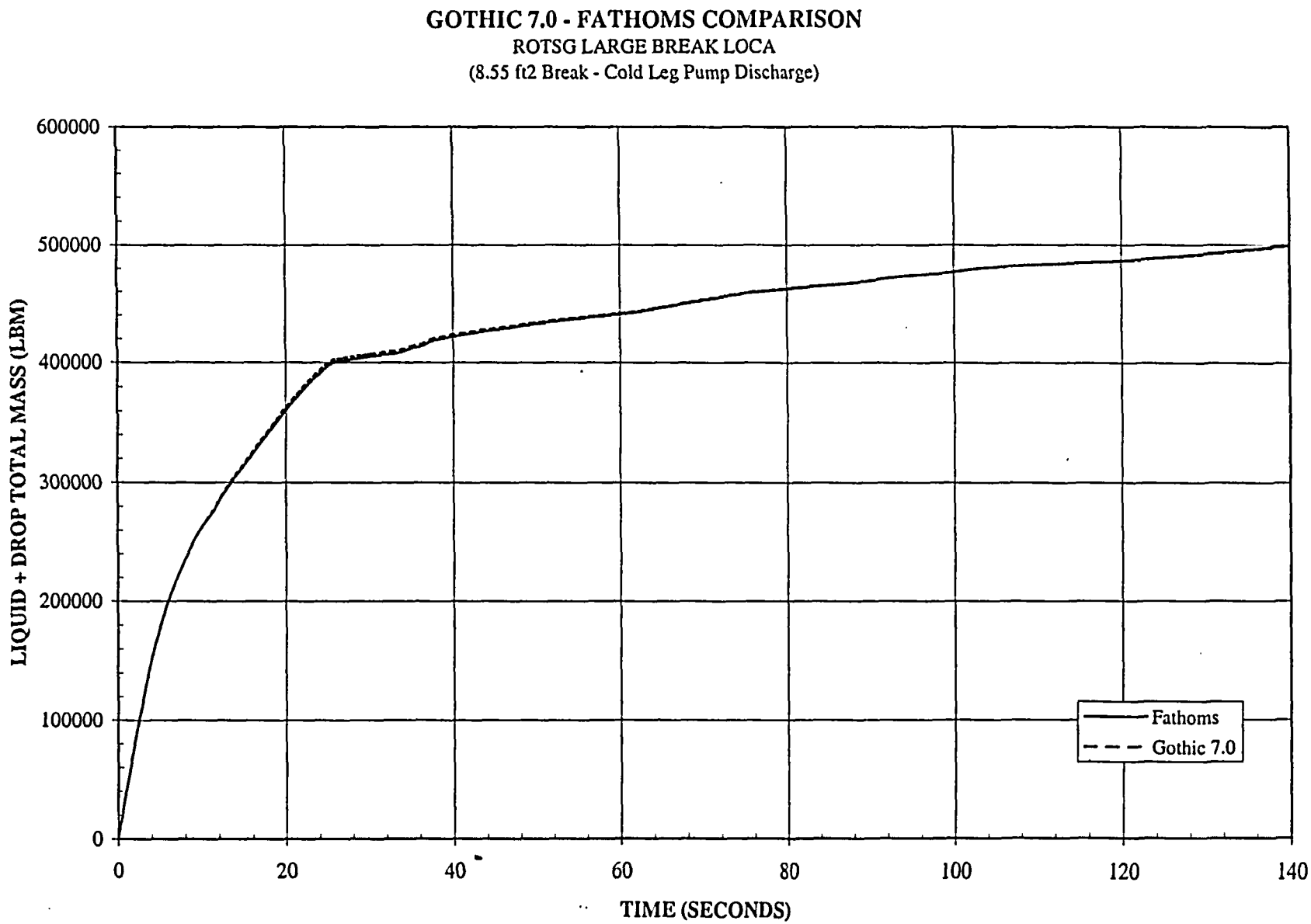


GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

(8.55 ft2 Break - Cold Leg Pump Discharge)

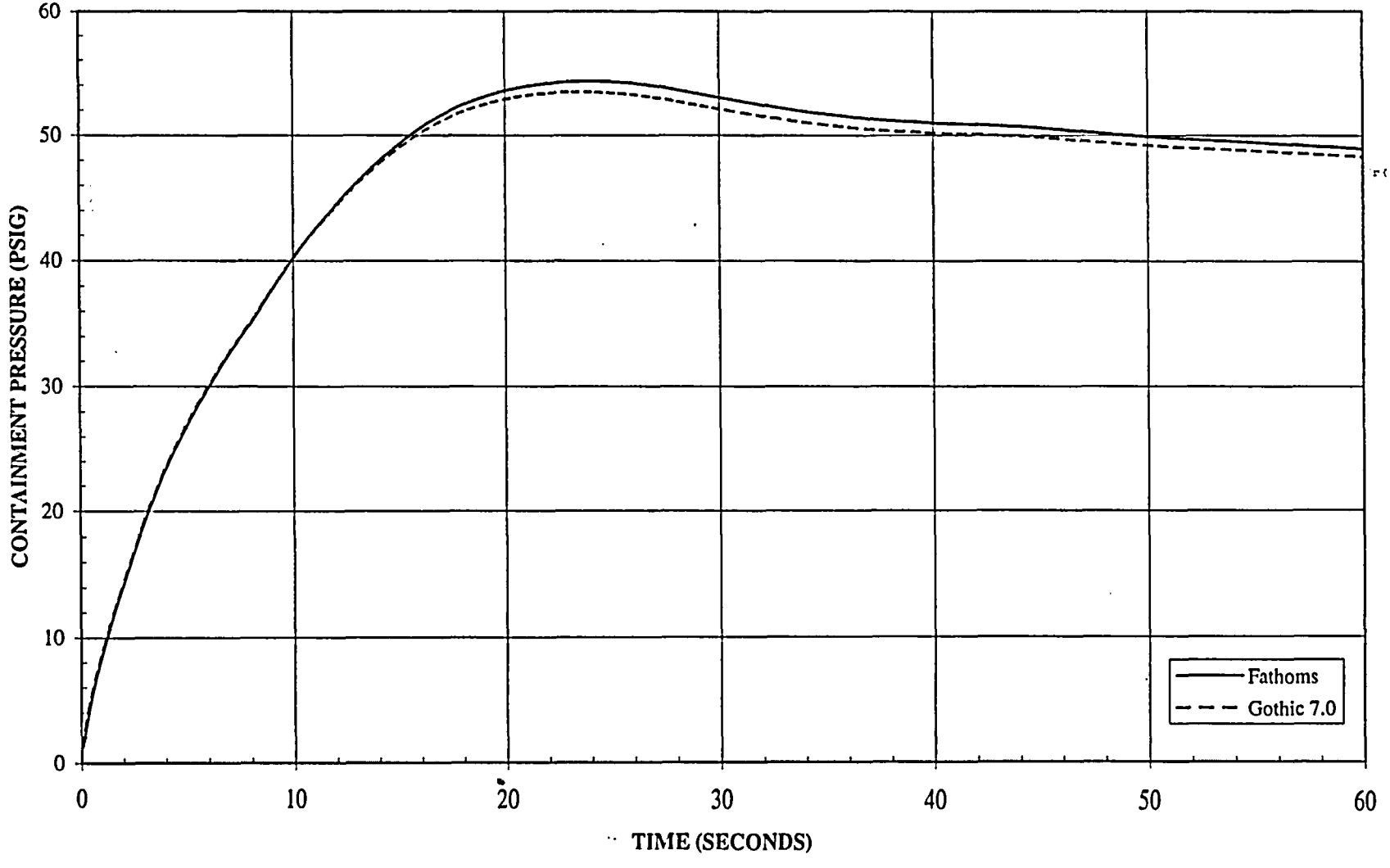


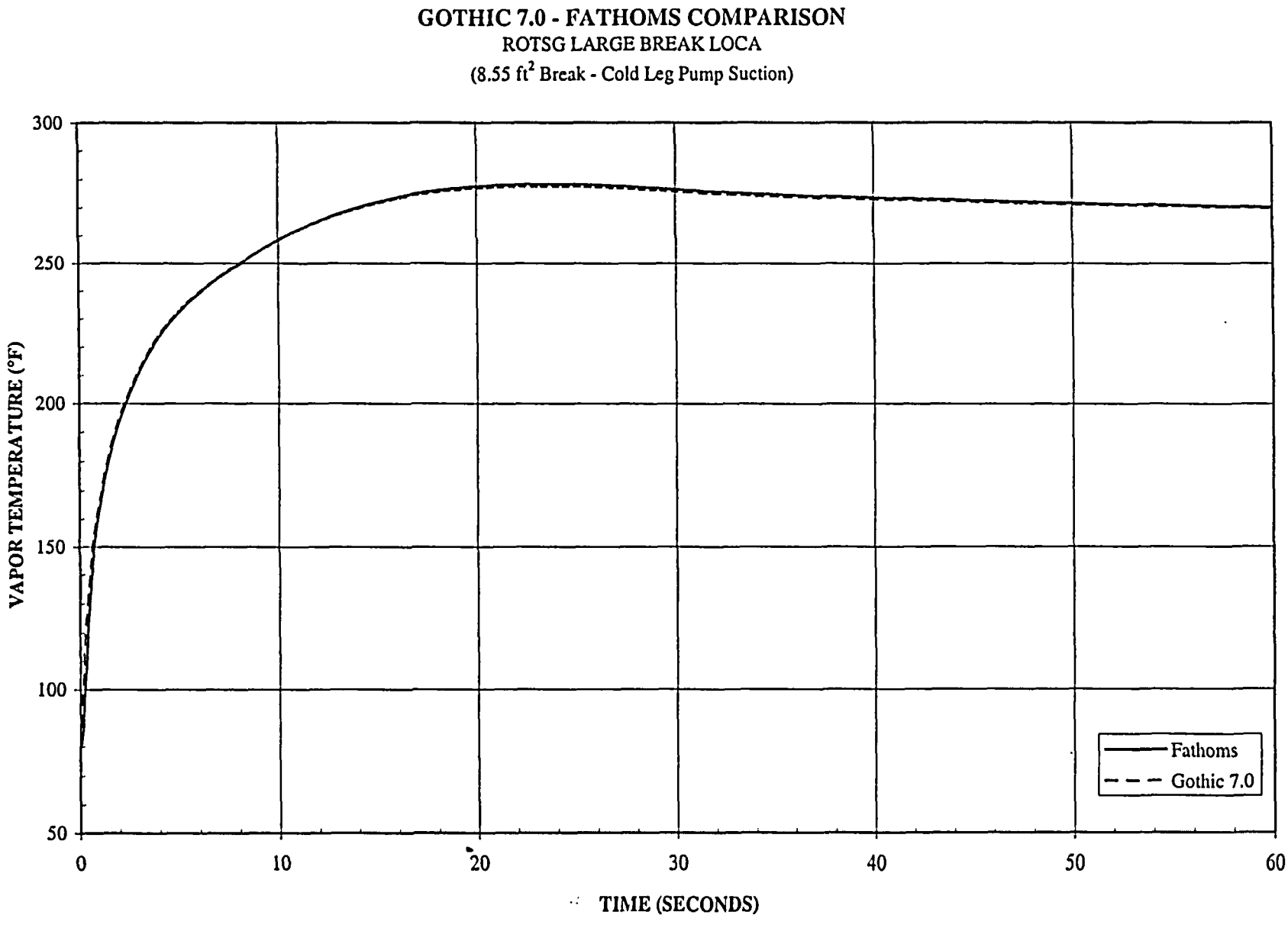

GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG LARGE BREAK LOCA
(8.55 ft2 Break - Cold Leg Pump Discharge)
LIQUID + DROP TOTAL MASS (LBM)
600000
500000
400000
300000
200000
100000
0
0
20
40
60
80
100
120
140
TIME (SECONDS)
Fathoms
Gothic 7.0

GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

(8.55 ft^2 Break - Cold Leg Pump Discharge)

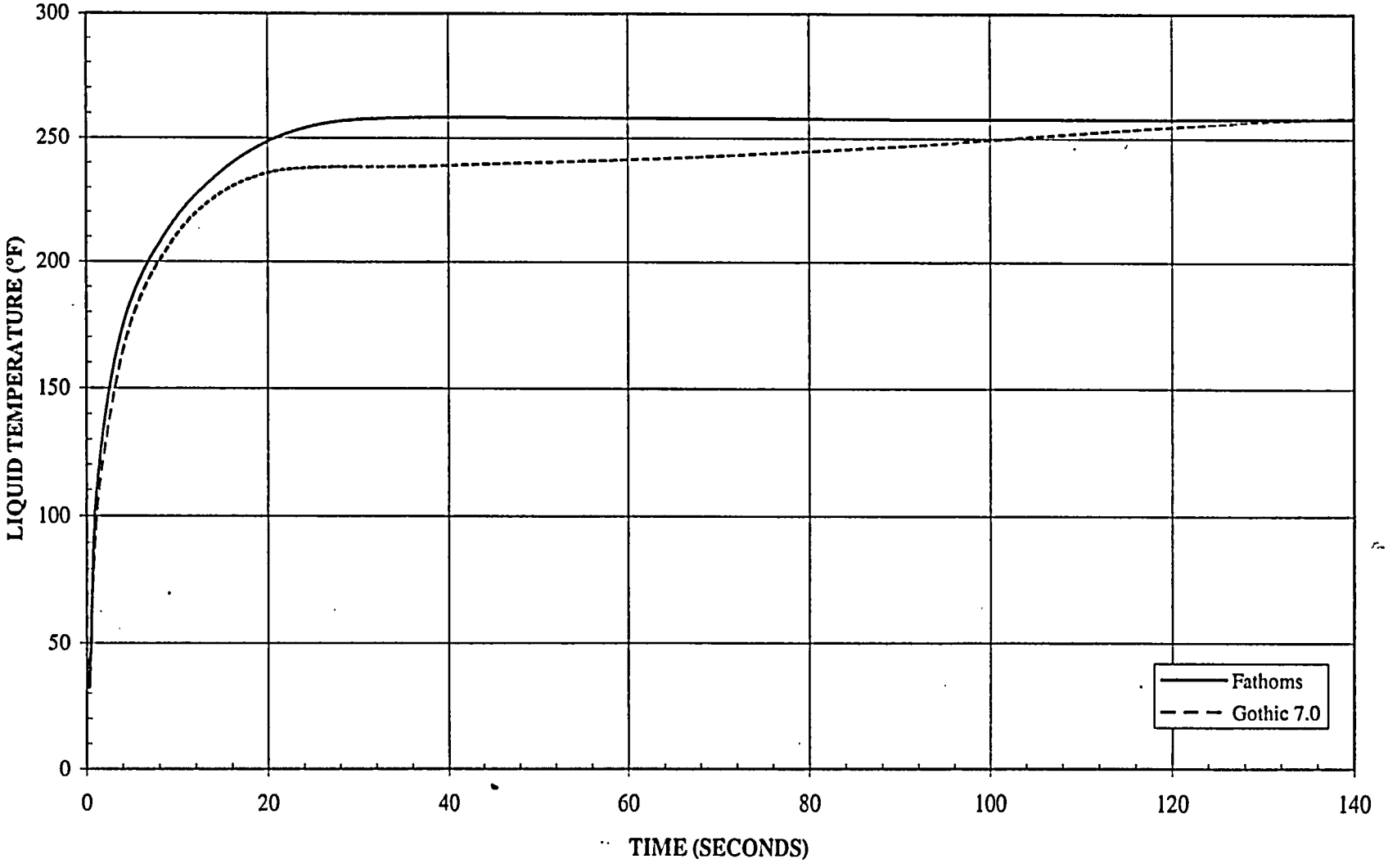


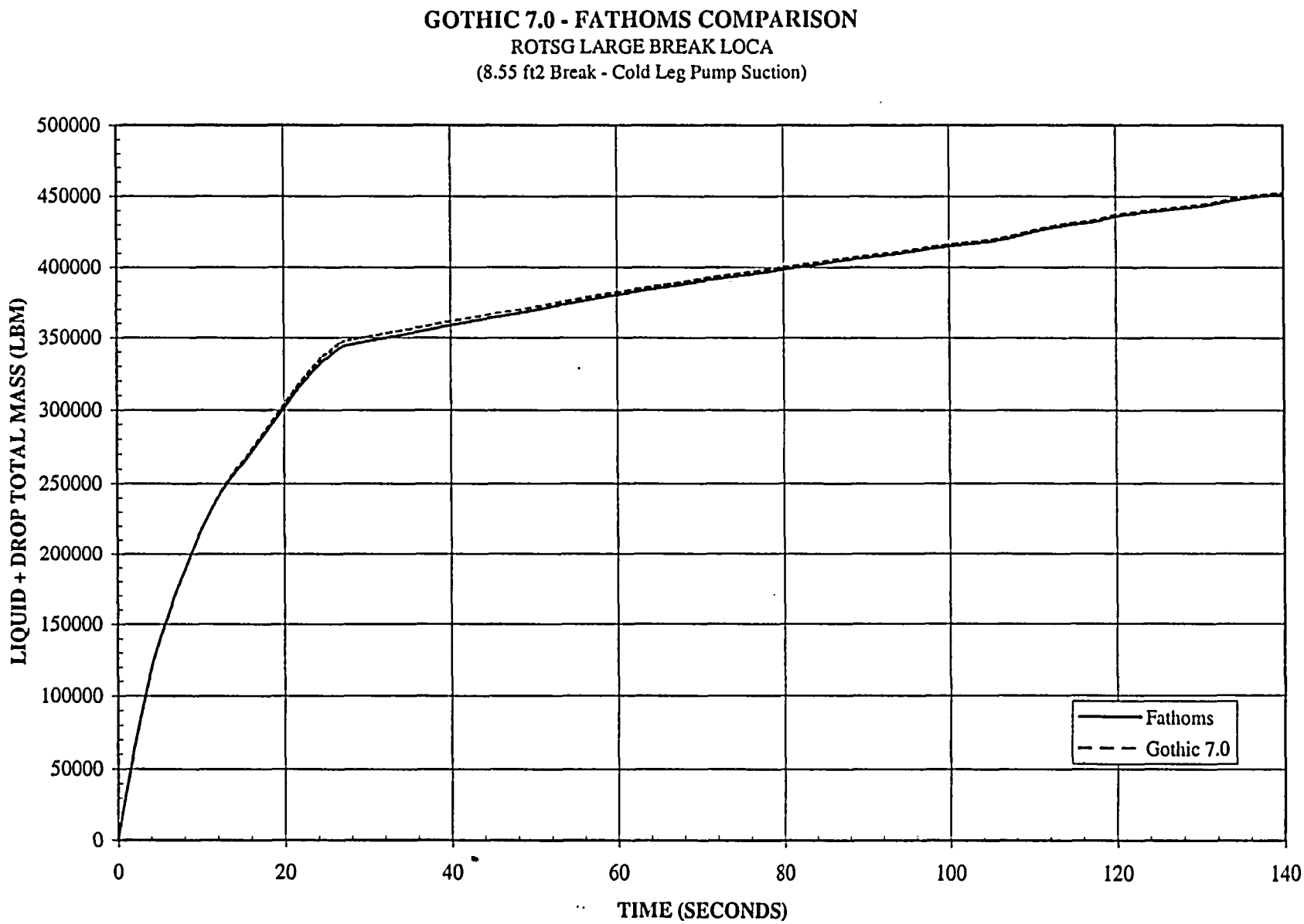

GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG LARGE BREAK LOCA
(8.55 ft² Break - Cold Leg Pump Suction)
VAPOR TEMPERATURE (°F)
300
250
200
150
100
50
0
10
20
30
40
50
60
TIME (SECONDS)
Fathoms
Gothic 7.0

GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG LARGE BREAK LOCA

(8.55 ft2 Break - Cold Leg Pump Suction)




GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG LARGE BREAK LOCA
(8.55 ft2 Break - Cold Leg Pump Suction)
LIQUID + DROP TOTAL MASS (LBM)
500000
450000
400000
350000
300000
250000
200000
150000
100000
50000
0
0
20
40
60
80
100
120
140
TIME (SECONDS)
Fathoms
Gothic 7.0

Questions On DPC-NE-3003-P and DPC-NE-3003-P Revision 1

Chapter 1

4. Page 1-2 of DPC-NE-3003-P: The last paragraph discusses the FATHOMS code. This should be modified to include discussion of the GOTHIC code.

Response: Revision 4 on p. 3-4 of Attachment 3 to the June 13, 2002 submittal includes a new sentence on p. 1-3 of the topical report that mentions the new Appendix C for the GOTHIC 7.0 code. This is the preferred text to include referral to the GOTHIC 7.0 content in Chapter 1 of the report. Also, Revision 10 on p. 3-6 includes more discussion of GOTHIC 7.0 in Chapter 2.

5. Page 1-3: (a) Why was NRC review of Supplement 1, the November 1997 version of DPC-NE-3003-P not required? (b) Please provide Supplement 2 of DPC-NE-3003-P.

Response: (a) Supplement 1 is a revised application of the NRC-approved methodology for the steam line break mass and energy release and containment response methodology. The main difference in the Supplement 1 application compared to the Chapters 5 & 6 application is crediting main feedwater isolation via the Main Steam Line Break Detection and Main Feedwater Isolation System. The Main Steam Line Break Detection and Main Feedwater Isolation System is a station modification that was installed subsequent to the submittal of the topical report. This station modification and associated analyses were reviewed and approved by the NRC by letter dated December 7, 1998 (License Amendments 234, 234, and 233). Supplement 1, the current UFSAR Chapter 6 analysis, was added since it replaced the analysis in Chapters 5 & 6 as the licensing basis analysis. The intent was to avoid future use of the superceded analysis results in Chapters 5 & 6. The Chapters 5 & 6 analyses were retained for historical purposes. No new methodology was included in Supplement 1, and so submittal for NRC review was not required.

(b) Supplement 2 of DPC-NE-3003-P is already included at the back of the November 1997 published version.

Chapter 3

6. Pages 3-9 and 3-10: Conservative mass and energy calculations depend not only on bounding high fuel stored energy but the rapid transfer of this energy to the coolant. Please describe the modeling of this energy transfer and why it is conservative.

Response: The gap thermal conductivity required to obtain the desired conservative initial fuel temperature is held constant during the peak containment pressure analysis. The time of peak containment pressure roughly corresponds with the end of the primary coolant system blowdown as the mass release is the primary driving factor for the containment pressure response.

During the blowdown the core average fuel temperature decreases from an initial value of approximately 1390°F, to approximately 590°F. At the end of blowdown the volume average core fluid temperature is about 310°F with an average vapor void fraction

exceeding 98%. With the current modeling approach, approximately three quarters of the core stored energy and all of the heat generated in the fuel is released during the primary coolant system blowdown. This modeling approach is conservative since the gap thermal conductivity is held constant. As the fuel cools during blowdown, the gap would actually open causing the gap thermal conductivity to decrease. This phenomenon is not credited in the methodology.

7. Page 3-13: (a) Please explain why the sump water temperature and the EQ temperatures are bounding for different break sizes. (b) It appears that the report is stating that a double-ended guillotine break at the same location (reactor vessel outlet nozzle) has two different values (8.55 ft^2 and 14.1 ft^2). Please clarify.

Response: The double-ended guillotine flow area for a cold leg pump discharge break is 8.55 ft^2. The double-ended guillotine flow area for a hot leg break is 14.1 ft^2. The cold leg pump discharge break is the limiting break location for the EQ response. A typographical error is included in this change. The sentence that includes the 8.55 ft^2 flow area should have identified the break location as "*reactor vessel inlet nozzle*" instead of "*reactor vessel outlet nozzle*".

8. Page 3-15: Why was it necessary to change the RBS flow rate assumptions in the BWST depletion and the FATHOMS analysis to be consistent when they were previously conservative? What is the effect of this loss of conservatism in terms of peak containment pressure and temperature, and EQ temperatures?

Response: The original topical report text included specific flow values which are changing due to modifications being made to the reactor building spray headers. Specifically, some of the spray nozzles are being plugged and flow orifices are being installed to reduce the total spray flow. Previously RBS flow was manually throttled to a setpoint. The modifications described will eliminate the need for throttling. The spray values assumed in the analysis are conservative for the plant following the modifications.

There is no impact to the short-term LOCA peak containment pressure analysis as RBS operation is not credited. The assumed spray flowrates assumed in the long-term LOCA analysis and the steam line break analysis are conservative values. Use of conservative values for inputs to the methodology is considered appropriate. The reduction is spray flow is offset in the steam line break analysis that determines the EQ temperature profile by increasing the required capacity of the RBCUs (the fan coolers).

Chapter 4

9. Page 3-16: (a) Rather than using a simple containment model built into the RELAP5 calculation, input to RELAP5 from the FATHOMS code is used. (a) Is there a feedback to ensure consistency between the RELAP5 and the FATHOMS calculations? (b) The description of this phase of the calculation says that "similar break sizes" are used. What criteria are used to determine that the break sizes are "similar" and why do calculations from "similar" break sizes give results that are better than those from a simple model of (presumably) the same break size? (c) Provide results from sensitivity calculations to

demonstrate that the start times for the RBCUs and the RBS are close enough to each other and that the containment peak conditions are acceptably close.

Response: (a) There is no feedback between the FATHOMS and RELAP5 calculations. However, it should be noted that for all SBLOCA cases the break remains choked and therefore the containment pressure does not directly impact the RELAP5 calculation. There are two FATHOMS results that do impact the RELAP5 calculation 1) the RBS actuation time (actuates on high containment pressure), which effects the timing of the swap to sump recirculation and 2) the containment sump water temperature, which affects the injection water temperature during sump recirculation. Iterations are performed until the results converge or conservative agreement is achieved between the FATHOMS calculation and the RELAP5 assumptions. These parameters are compared in the answer to item (c) below.

(b) The wording in the technical justification could be modified to be more precise. For all of the cases that were reanalyzed for the ROTSG, the FATHOMS results used in each of the RELAP5 calculations were taken from the cases with the same size break. In general a larger break could be used as a conservative boundary condition for a smaller break since spray actuation will occur earlier and the sump temperature will generally be hotter. The wording should have stated "for a similar case" instead of "for a similar break size" since there are other assumption in the FATHOMS analysis (i.e. RBCU performance, cooling water temperature, initial conditions) which can impact the containment results.

(c) The RBCUs are not modeled in the RELAP5 calculation. The same RBS actuation time is assumed in the RELAP5 and the FATHOMS calculations. The assumed actuation time was determined from preliminary calculations. In the final FATHOMS calculation, a check is made of the predicted FATHOMS containment pressure at RBS actuation and the analysis setpoint (20 psig). The assumed actuation times and the FATHOMS calculated pressures at the time of RBS actuation are provided in Table 9-1. The containment sump water temperatures calculated by FATHOMS and assumed in the RELAP5 calculation are also provided in Table 9-1.

Table 9-1
Comparison of FATHOMS calculated parameters with the RELAP5 assumptions

Parameter	Break Size		
	0.01 ft^2	0.025 ft^2	0.05 ft^2
RBCU actuation - FATHOMS	1895 sec	898 sec	573 sec
RBCU actuation - RELAP5	not modeled	not modeled	not modeled
RBS actuation - FATHOMS	9000 sec	4000 sec	1200 sec
RBS actuation - RELAP5 (for BWST depletion only)	9000 sec	4000 sec	1200 sec
FATHOMS calculated containment pressure at RBS actuation	21 psig	25 psig	22 psig
FATHOMS calculated sump temperature - At start of sump recirculation - Peak	 184 °F 201 °F	 190 °F 200 °F	 200 °F 215 °F
RELAP5 assumed sump water temperature - At start of sump recirculation - Peak	 185 °F 205 °F	 200 °F 200 °F	recirculation phase not modeled

10. Page 3-16 to 3-17: In the transition from RELAP5 to FATHOMS for small break calculations the heat structures will be modeled as heaters with a prescribed heat flux. They were previously modeled as heat slabs. (a) What is the advantage of this change? (b) How is the heat flux of the heater component determined as a function of time? (c) Provide a sensitivity study of the effect of the heat flux vs. heat structure on the long- term containment response.

Response: The original version of DPC-NE-3003-PA contained a discrepancy between the RELAP5 and FATHOMS treatments of the primary and secondary system fluid and metal energy content and the manner in which it was passed from RELAP5 to FATHOMS. In Section 4.1, Long-Term Mass and Energy Release, first paragraph on p.4-3, the original documentation discussed the RELAP5 heat structures being collapsed into heat slabs consistent with the simplified FATHOMS node boundaries. In Section 6.4.3, pg. 6-31, the original documentation stated that the RCS heat structures are modeled with the FATHOMS heater model. The changes to Section 4.1 of DPC-NE-3003 were made to remove this discrepancy in the documentation, and represent no changes to the mass and energy release or containment response methodology.

The technical justification to Change #35 in the DPC-NE-3003, Revision 1 change package incorrectly stated that there was a change in the stored energy modeling for the FATHOMS small break LOCA containment analyses. The FATHOMS heater component model has always been used for the small beak LOCA analyses; there were no changes to this methodology introduced, other than the inclusion of the stored energy in the secondary system fluid and structural metal (which had been left out in the original analyses). The final sentence in the Technical Justification to Change #35 also incorrectly discusses the impact of a modeling change from the heat slab to heater component model in the FATHOMS small break LOCA analyses.

In summary, there was no change to the methodology used to transfer the energy from the primary and secondary system fluid/metal from the RELAP5 to the FATHOMS small break LOCA analyses (other than the inclusion of the secondary system energy content). The discussion in the Technical Justification to Change #35 regarding such a change was in error. The heat flux is calculated by dividing the total energy content of each component at the end of the RELAP5 analysis by the remaining transient time in the FATHOMS analysis (a constant heat flux is assumed for each component for the duration of the FATHOMS analysis).

11. Section 6.1 Page 6-1 of DPC-NE-3003-P: Some words are missing between the last sentence on page 6-1 and the first words on page 6-2 of the original report. The proposed revision did not fix this. Please provide the missing words.

Response: The November 1997 version added the missing paragraph to the bottom of p. 6-1.

12. Section 6.2.1 Page 6-9 First paragraph: If Case 3B is considered the worst case LOCA for containment peak pressure, isn't the loss of offsite power, rather than loss of a 4160-V bus, the more limiting case? Please explain.

Response: The with or without offsite power assumption, as discussed on pp. 3-2 and 3-21, is mainly of interest for the LOCA peak pressure analyses relative to the continued

operation of the reactor coolant pumps. Both with offsite power (RCPs continue to run) and without offsite power (RCPs trip at time zero) cases are analyzed. The limiting single failure assumption, as discussed beginning on p. 3-12, involves the loss of emergency power to a 4160V bus, which causes a loss of one train of safeguards. This single failure is irrelevant for the peak pressure cases since the peak containment pressure is reached prior to the actuation of any of the safeguards powered by the 4160V bus regardless of whether or not the single failure is assumed. The loss of offsite power assumption has only a minor impact on the peak pressure results as discussed in the first paragraph on p. 6-9.

Questions Concerning GOTHIC

13. Provide a temperature envelope curve for EQ purposes calculated with GOTHIC and with FATHOMS. Discuss the differences in EQ envelopes calculated with FATHOMS and GOTHIC due to the use of different models in the two codes, especially those models which affect moisture in the atmosphere.

Response: To be provided at a later date.

14. Show that GOTHIC and FATHOMS calculations result in the same pressures, temperatures, sump temperatures, etc. for the design basis LOCA and main steam line break events. If they do not, explain differences in terms of the differences in models between the two codes. (See Question 1.)

Response: Refer to the response to Question #3 for LOCA. The results of the 6 code comparison analyses for the main steam line break are given in Table 14-1 and the figures that follow. The containment response following a main steam line break event for the ROTSG was performed to compare the results from FATHOMS 2.4 and GOTHIC 7.0. In general, the pressure and vapor temperature responses are very similar. GOTHIC 7.0 predictions for peak pressures are slightly lower on the smaller break sizes (~0.3 psi) and slightly higher for larger break sizes (~0.95 psi). The peak vapor temperature responses predicted by GOTHIC 7.0 overall are slightly lower (~12°F). For larger break sizes, the higher pressure predicted by GOTHIC 7.0 may be attributed to additional steam generation from the improved interface heat and mass transfer models and from the drops that are generated due to dripping. The slightly lower pressure for smaller break sizes may be attributed to additional droplets in the atmosphere. The slightly lower vapor temperature and liquid temperature in GOTHIC 7.0 for all break sizes may be attributed to additional droplets in the atmosphere also. This effectively increases the heat capacity of the atmosphere and reduces the temperature rise for a given energy addition. As discussed in the responses to Questions #1 and #3, more liquid is predicted to remain in droplet form in GOTHIC 7.0 which results in less liquid mass in the sump. In addition, the lower sump temperature predicted by GOTHIC 7.0 may be attributed to the addition of the second energy equation for the drops and the improved heat and mass transfer models in GOTHIC 7.0.

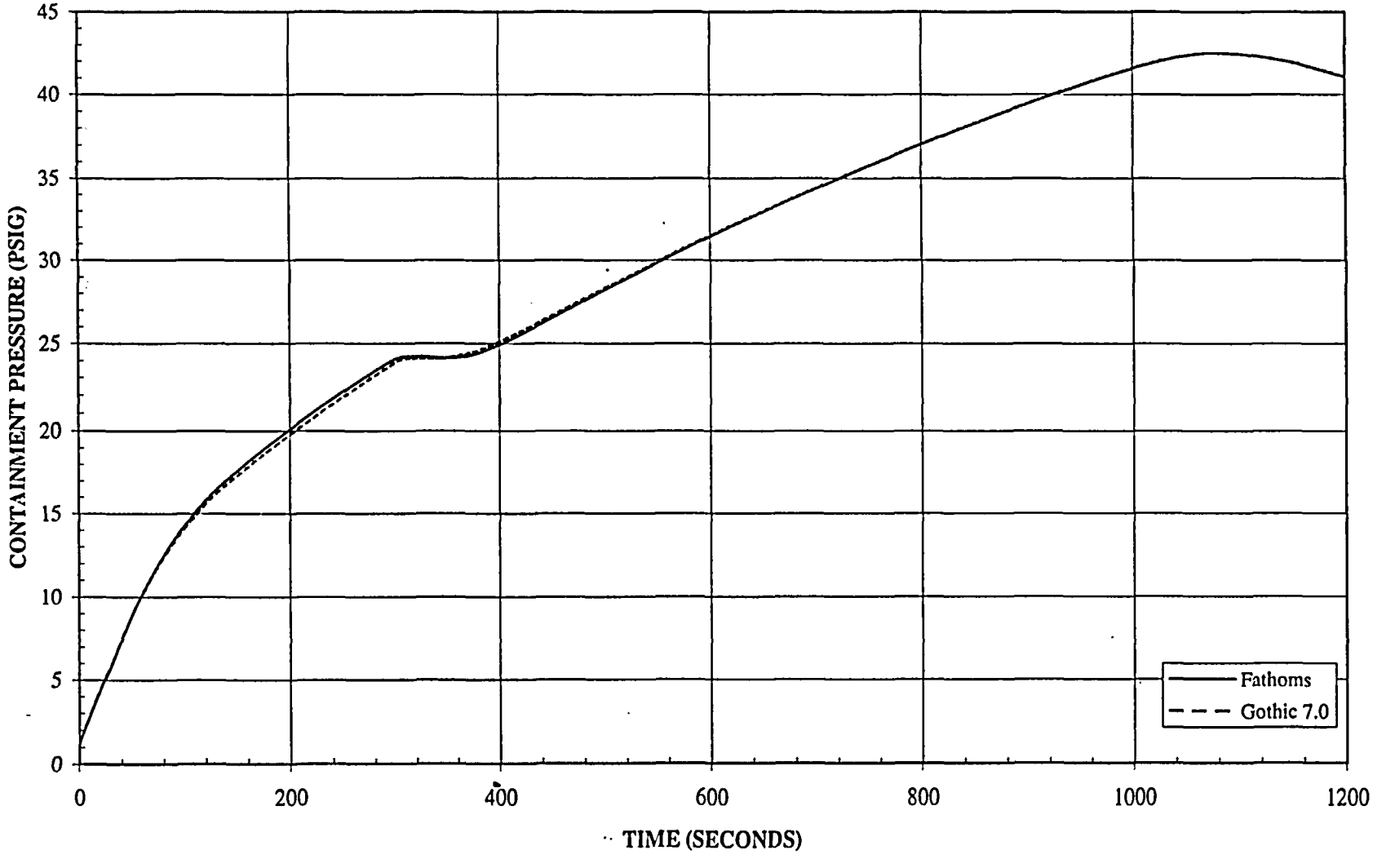
Table 14-1

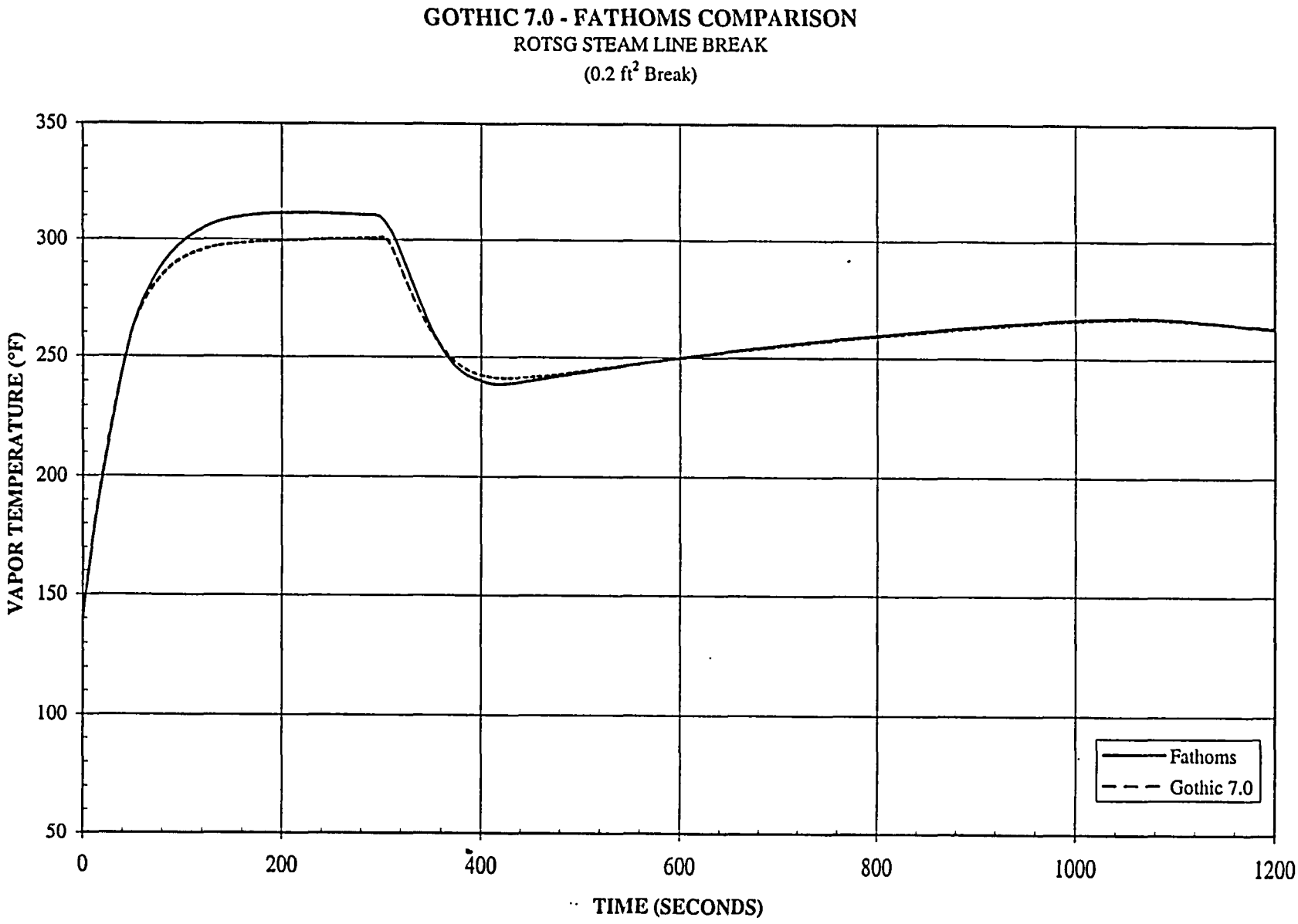
ROTSG Case Description	Peak Containment Pressure (psig)		Peak Containment Temperature (°F)	
	Fathoms	Gothic 7.0	Fathoms	Gothic 7.0
0.2 ft^2 Steam Line Break	42.48	42.48	311.38	301.04
0.6 ft^2 Steam Line Break	54.76	54.53	384.69	372.05
1.0 ft^2 Steam Line Break	56.60	56.32	417.82	405.07
1.0 ft^2 Steam Line Break with 90" SG Level	43.57	43.74	417.23	404.63
1.4 ft^2 Steam Line Break	52.23	53.18	438.80	426.02
Double Ended Guillotine Steam Line Break	55.23	55.63	463.08	450.11

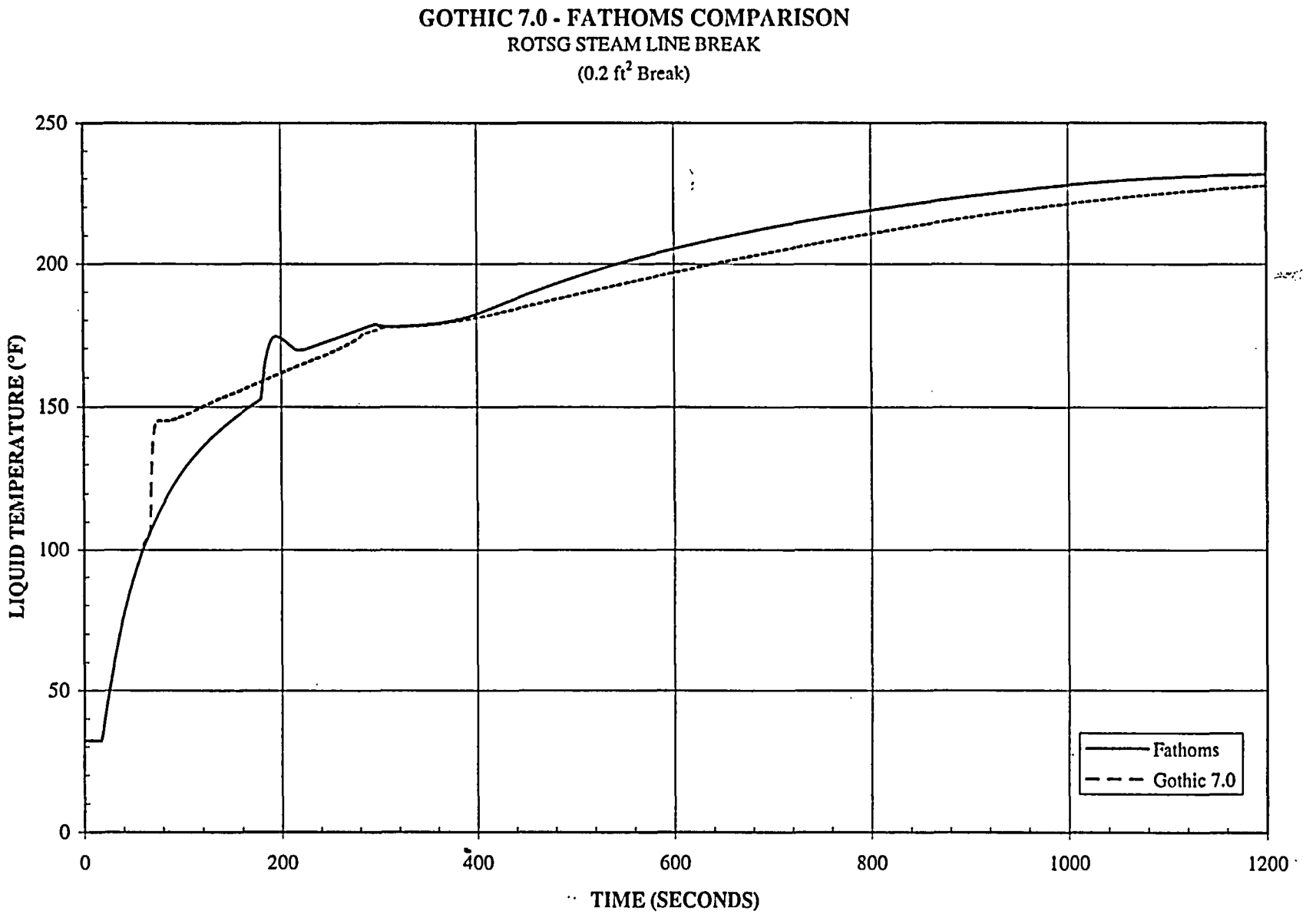
GOTHIC 7.0 - FATHOMS COMPARISON

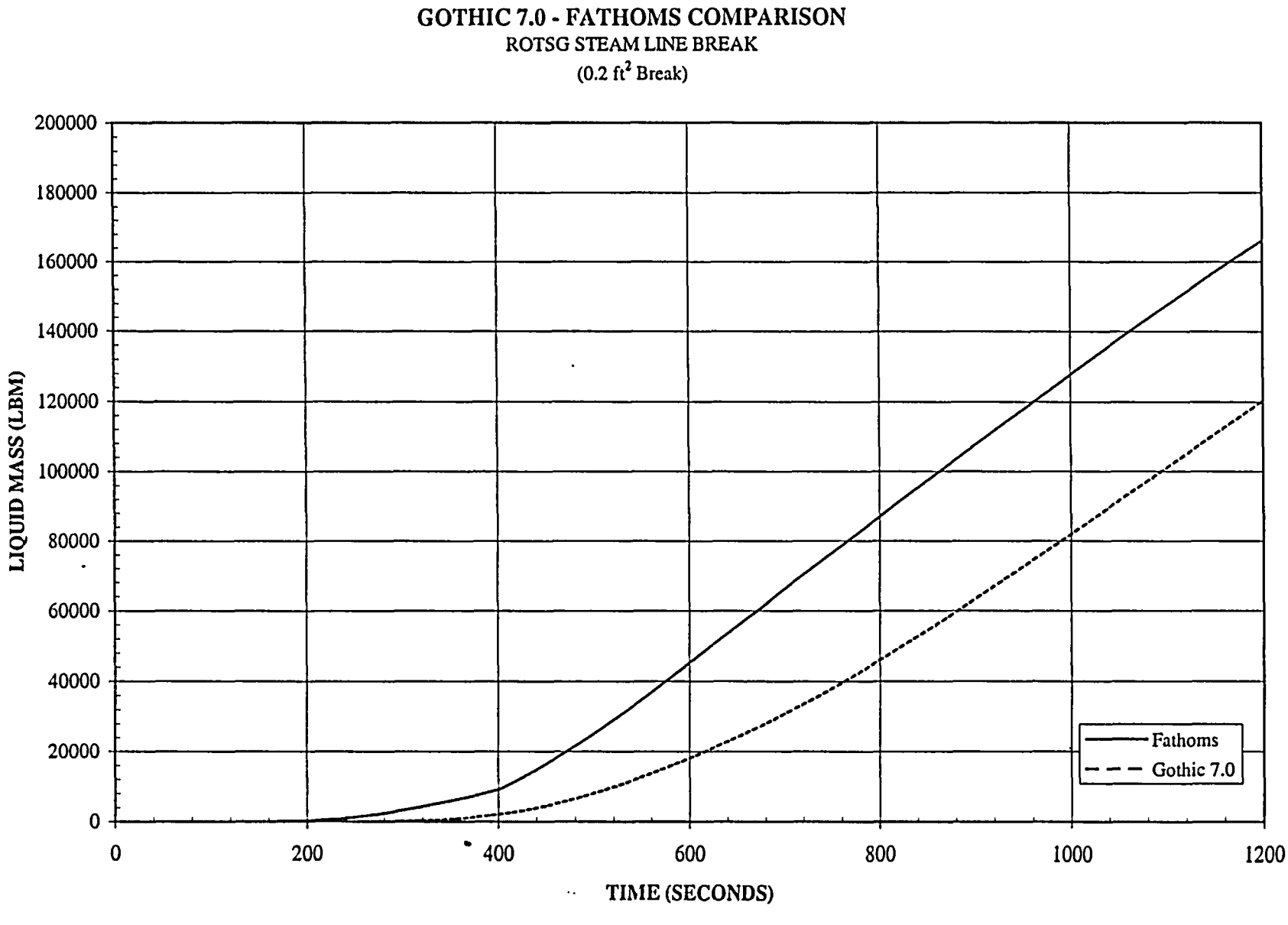
ROTSG STEAM LINE BREAK

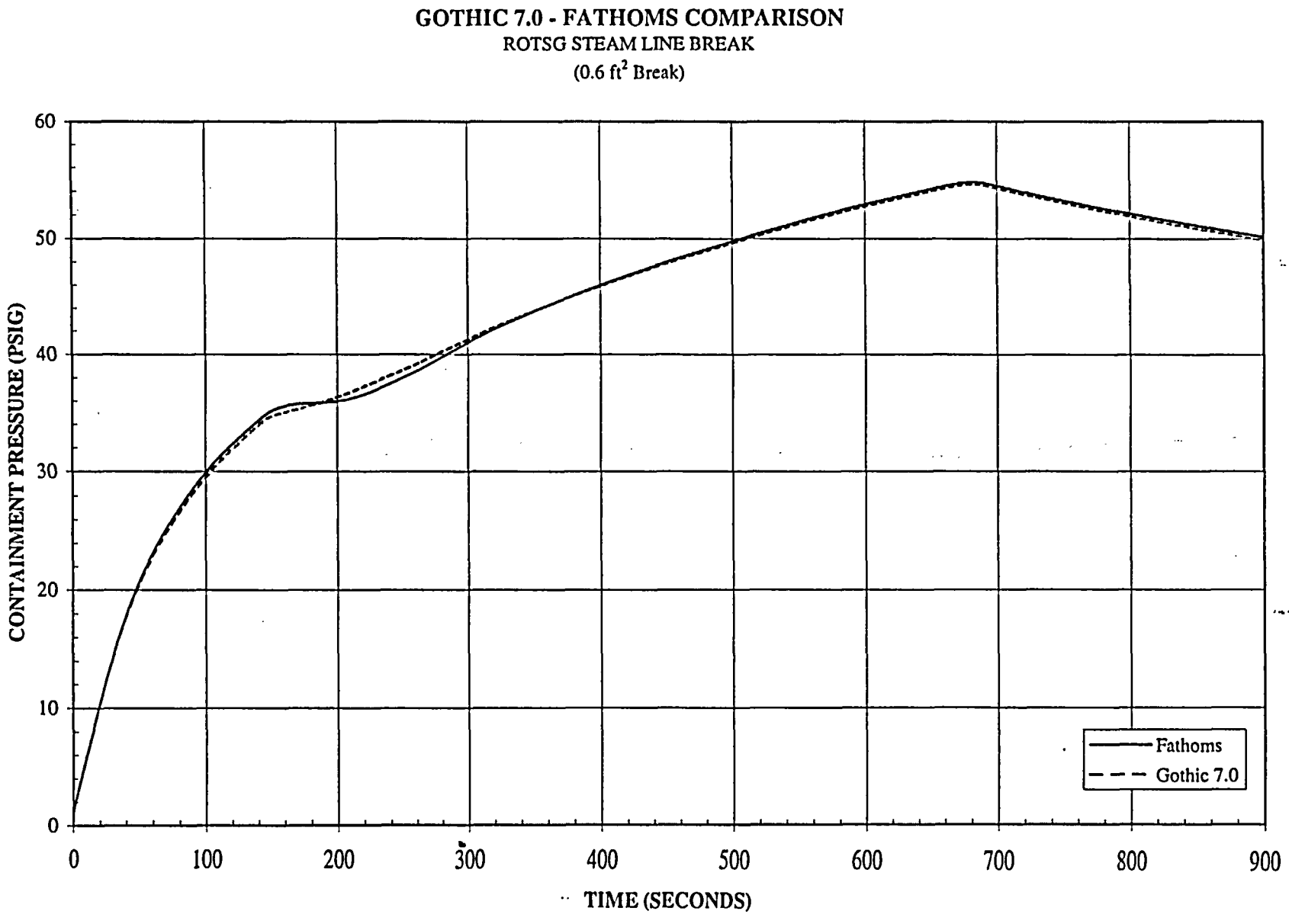
(0.2 ft^2 Break)

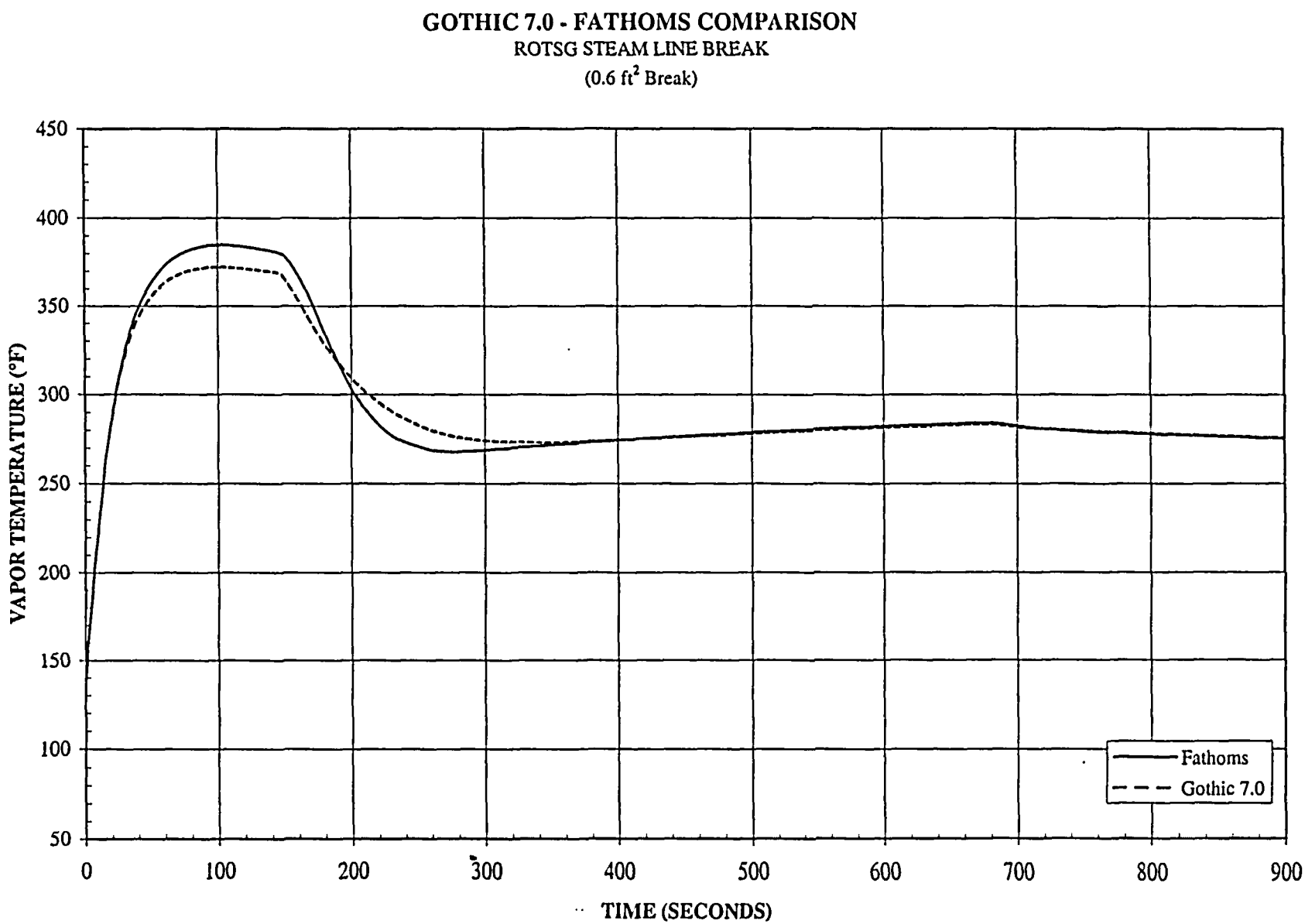


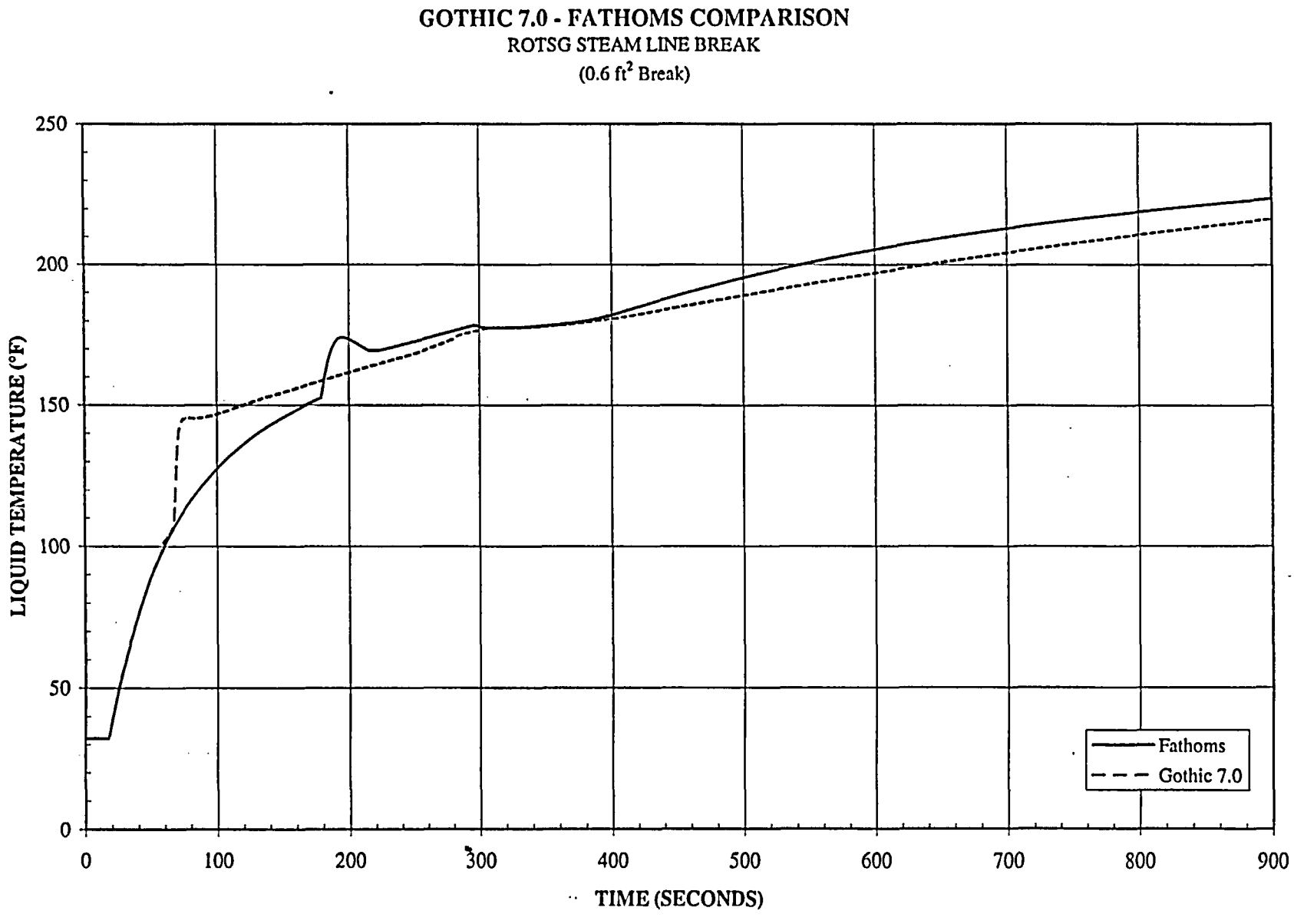

GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(0.2 ft² Break)
VAPOR TEMPERATURE (°F)
350
300
250
200
150
100
50
0
200
400
600
800
1000
1200
TIME (SECONDS)
Fathoms
Gothic 7.0


GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(0.2 ft² Break)
LIQUID TEMPERATURE (°F)
250
200
150
100
50
0
0
200
400
600
800
1000
1200
TIME (SECONDS)
Fathoms
Gothic 7.0


GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(0.2 ft² Break)
LIQUID MASS (LBM)
200000
180000
160000
140000
120000
100000
80000
60000
40000
20000
0
0
200
400
600
800
1000
1200
TIME (SECONDS)
Fathoms
Gothic 7.0


GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(0.6 ft² Break)
CONTAINMENT PRESSURE (PSIG)
0
10
20
30
40
50
60
0
100
200
300
400
500
600
700
800
900
TIME (SECONDS)
Fathoms
Gothic 7.0

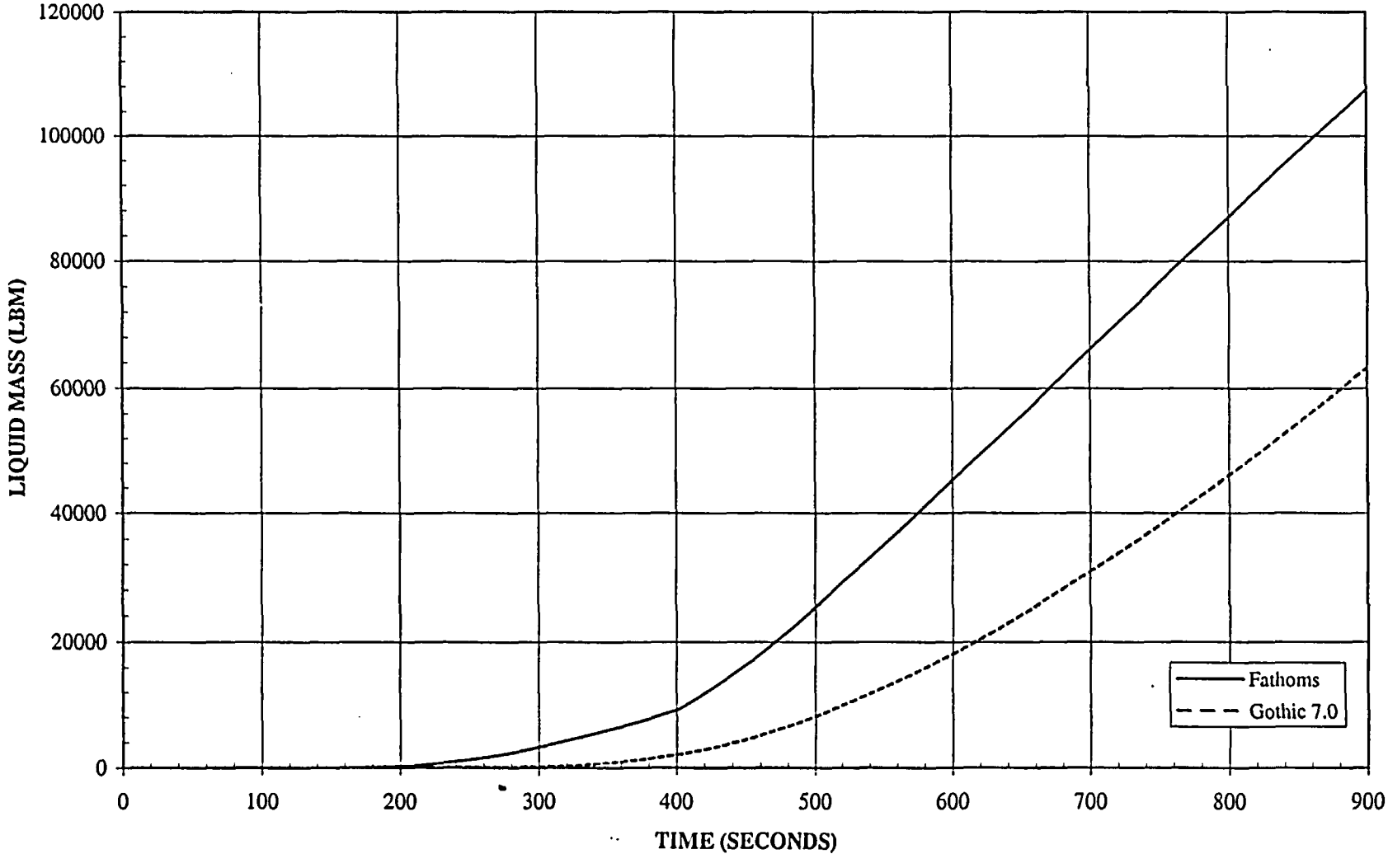

GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(0.6 ft² Break)
VAPOR TEMPERATURE (°F)
450
400
350
300
250
200
150
100
50
0
100
200
300
400
500
600
700
800
900
TIME (SECONDS)
Fathoms
Gothic 7.0


GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(0.6 ft² Break)
LIQUID TEMPERATURE (°F)
250
200
150
100
50
0
0
100
200
300
400
500
600
700
800
900
TIME (SECONDS)
Fathoms
Gothic 7.0

GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG STEAM LINE BREAK

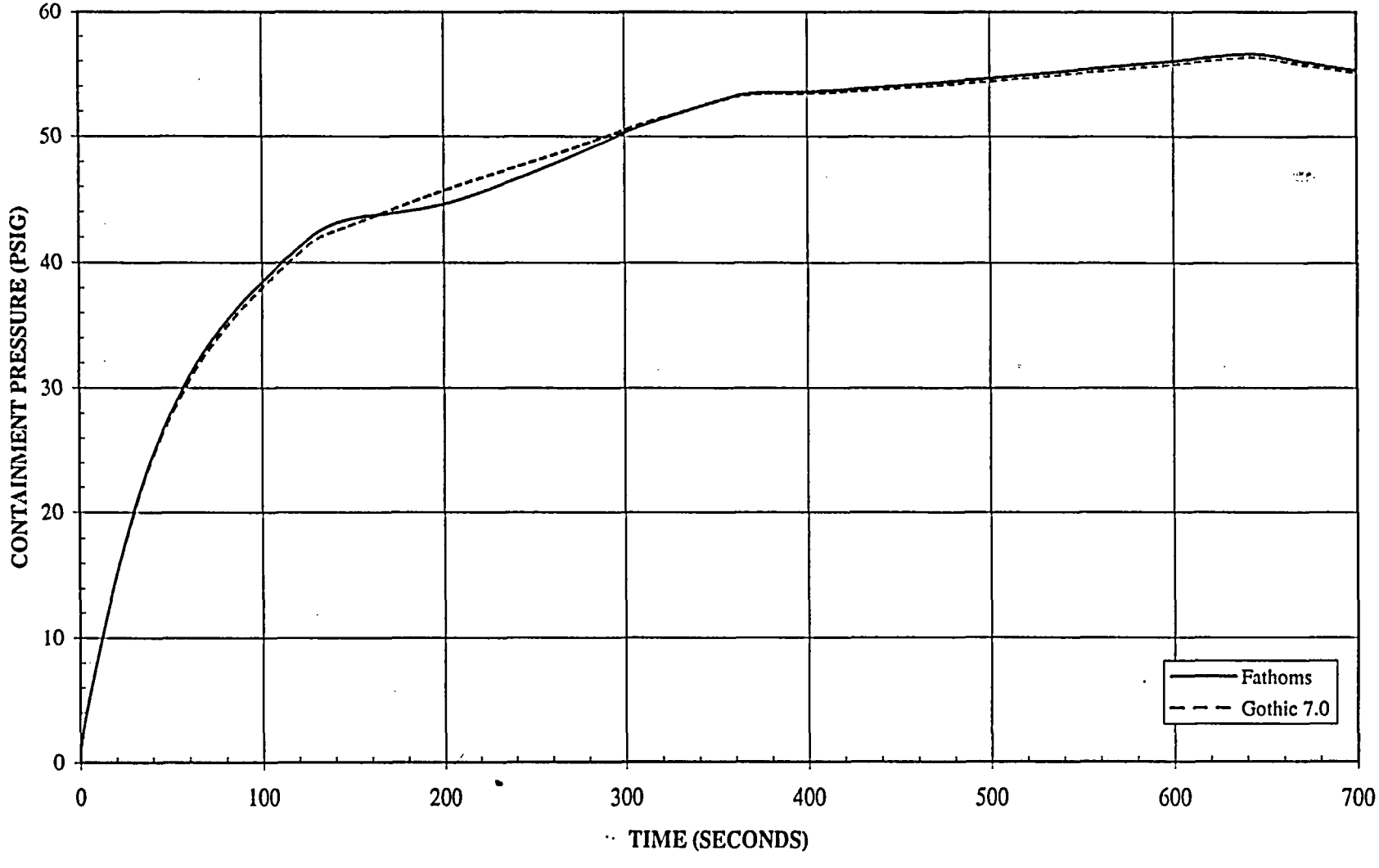
(0.6 ft^2 Break)

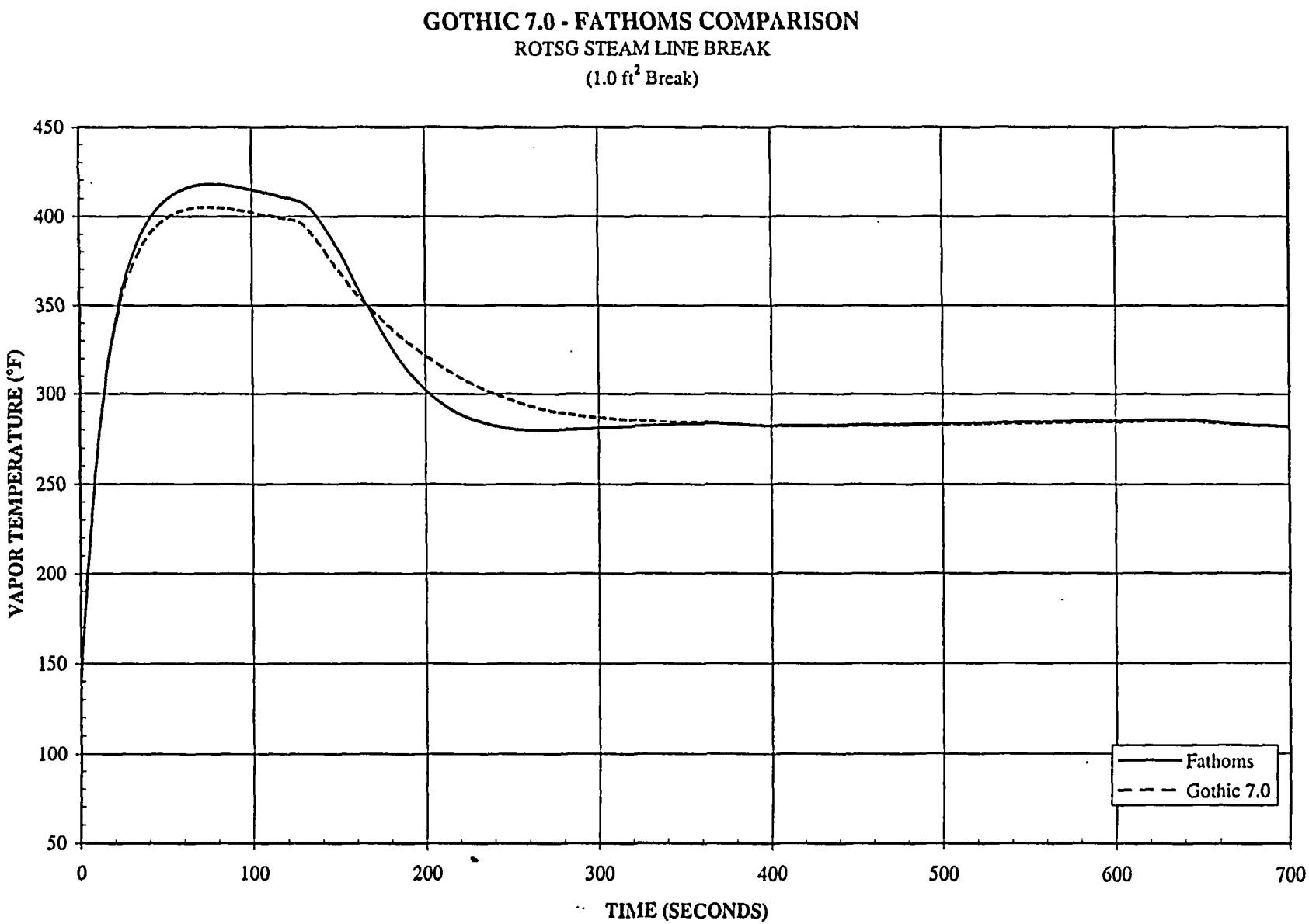


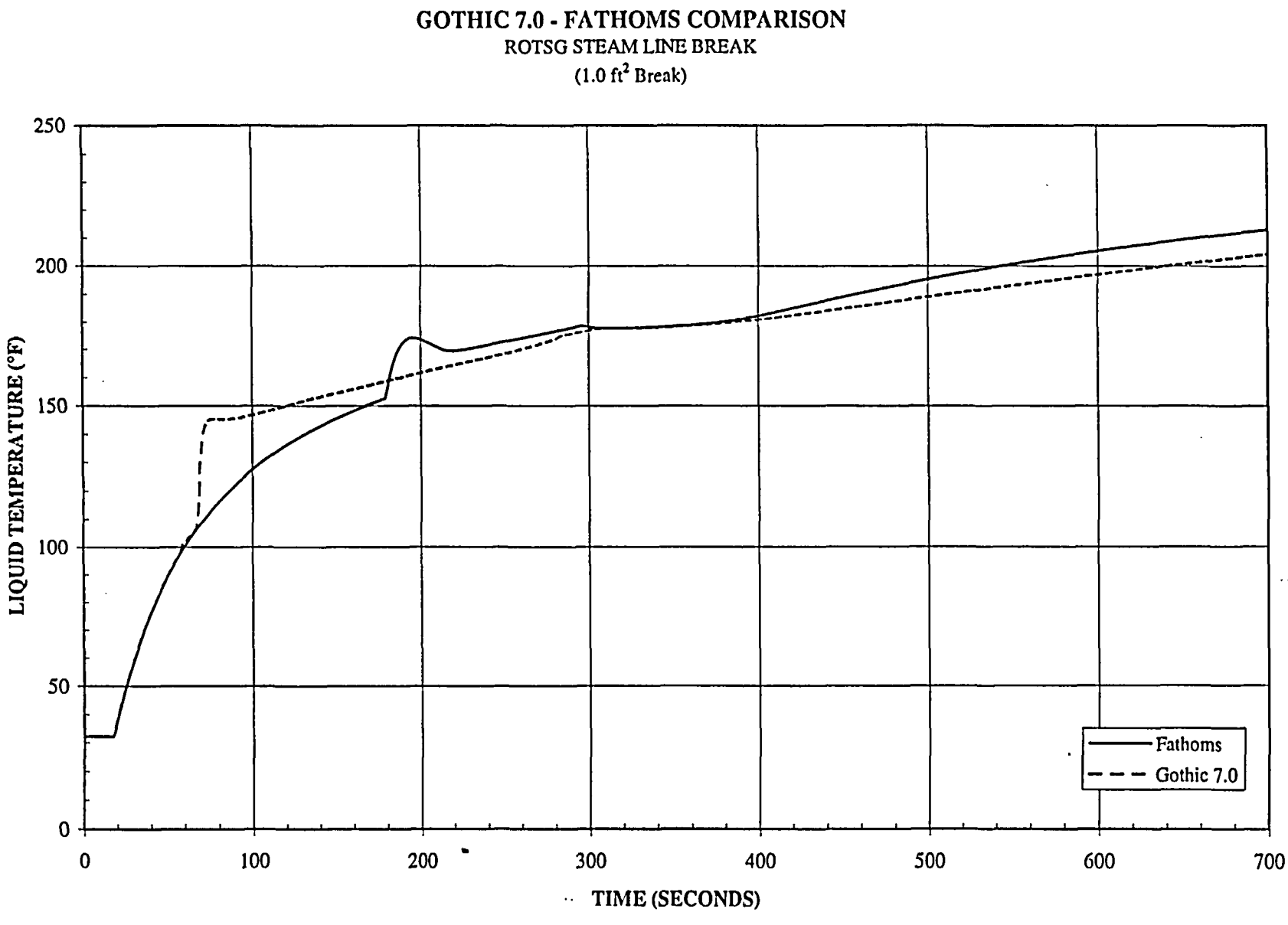
GOTHIC 7.0 - FATHOMS COMPARISON

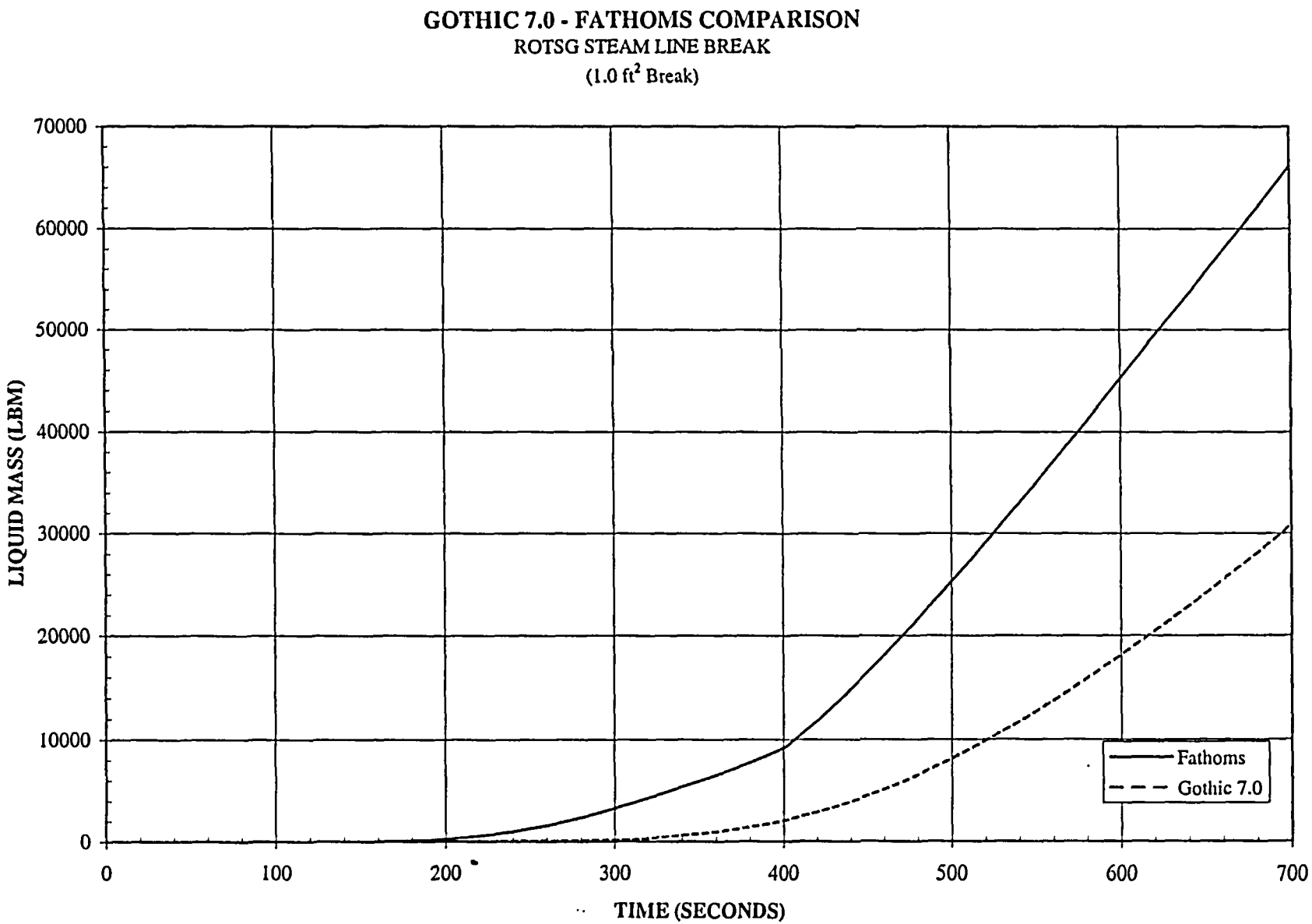
ROTSG STEAM LINE BREAK

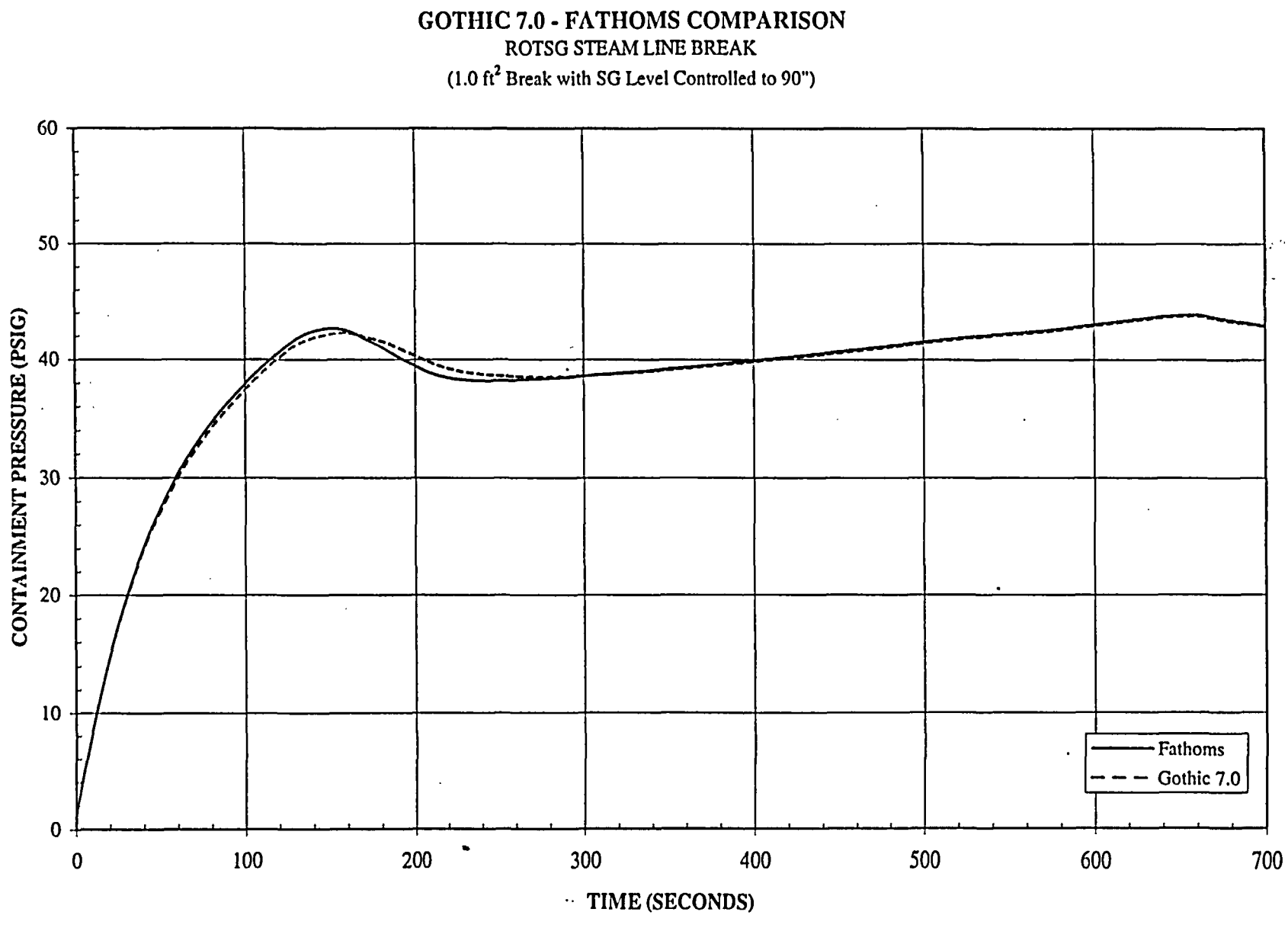
(1.0 ft^2 Break)

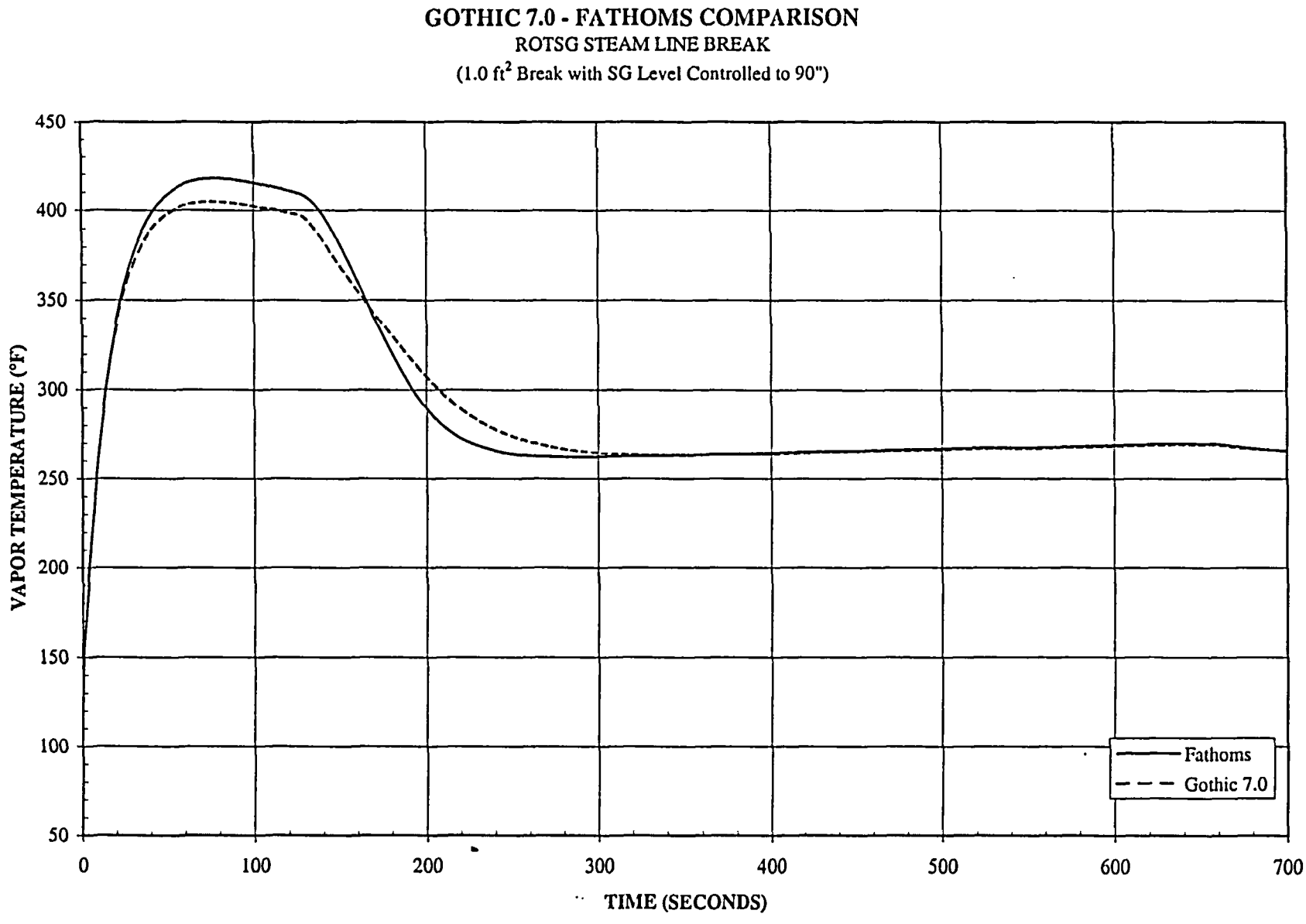


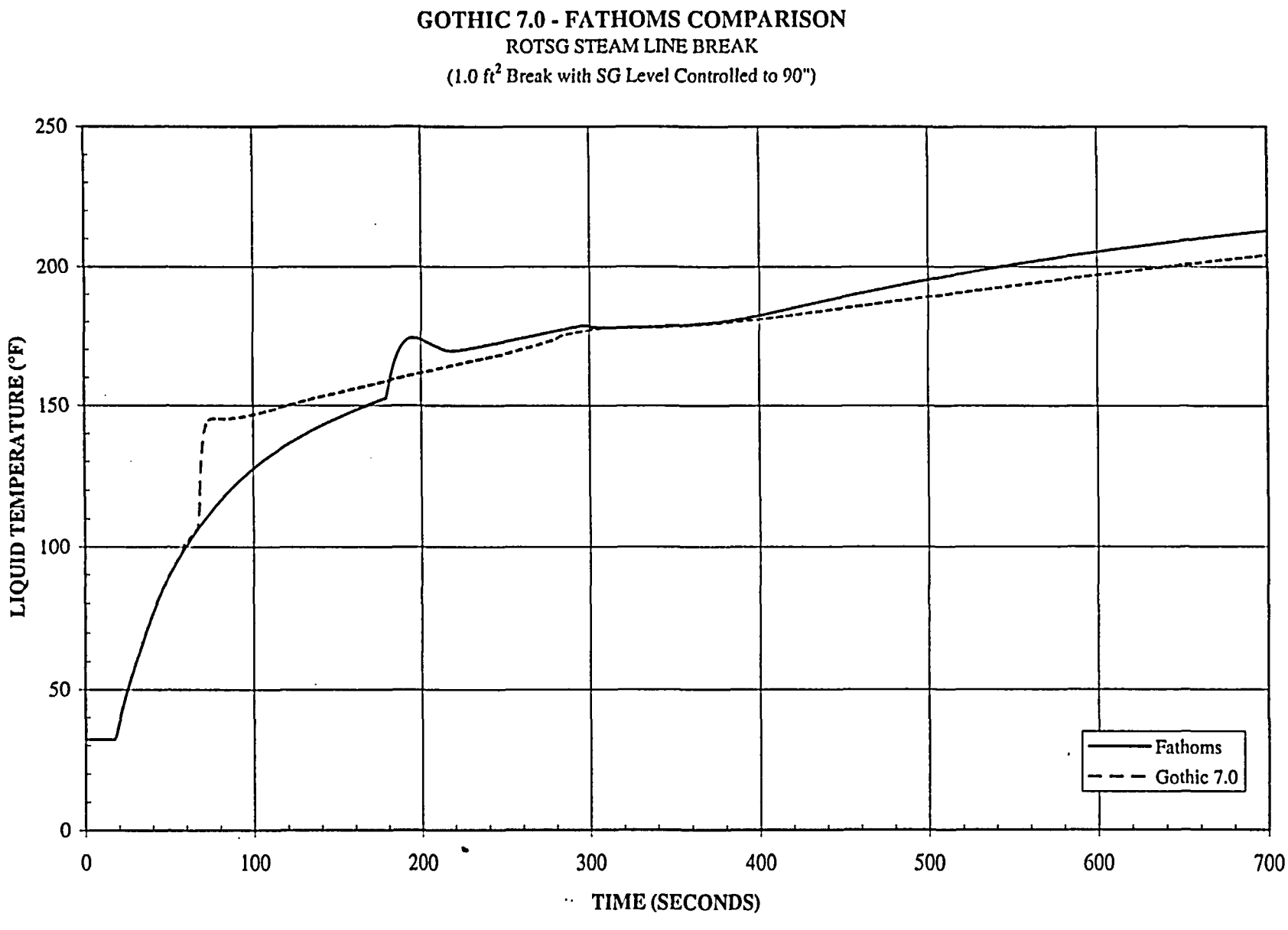

GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(1.0 ft² Break)
VAPOR TEMPERATURE (°F)
450
400
350
300
250
200
150
100
50
0
100
200
300
400
500
600
700
TIME (SECONDS)
Fathoms
Gothic 7.0

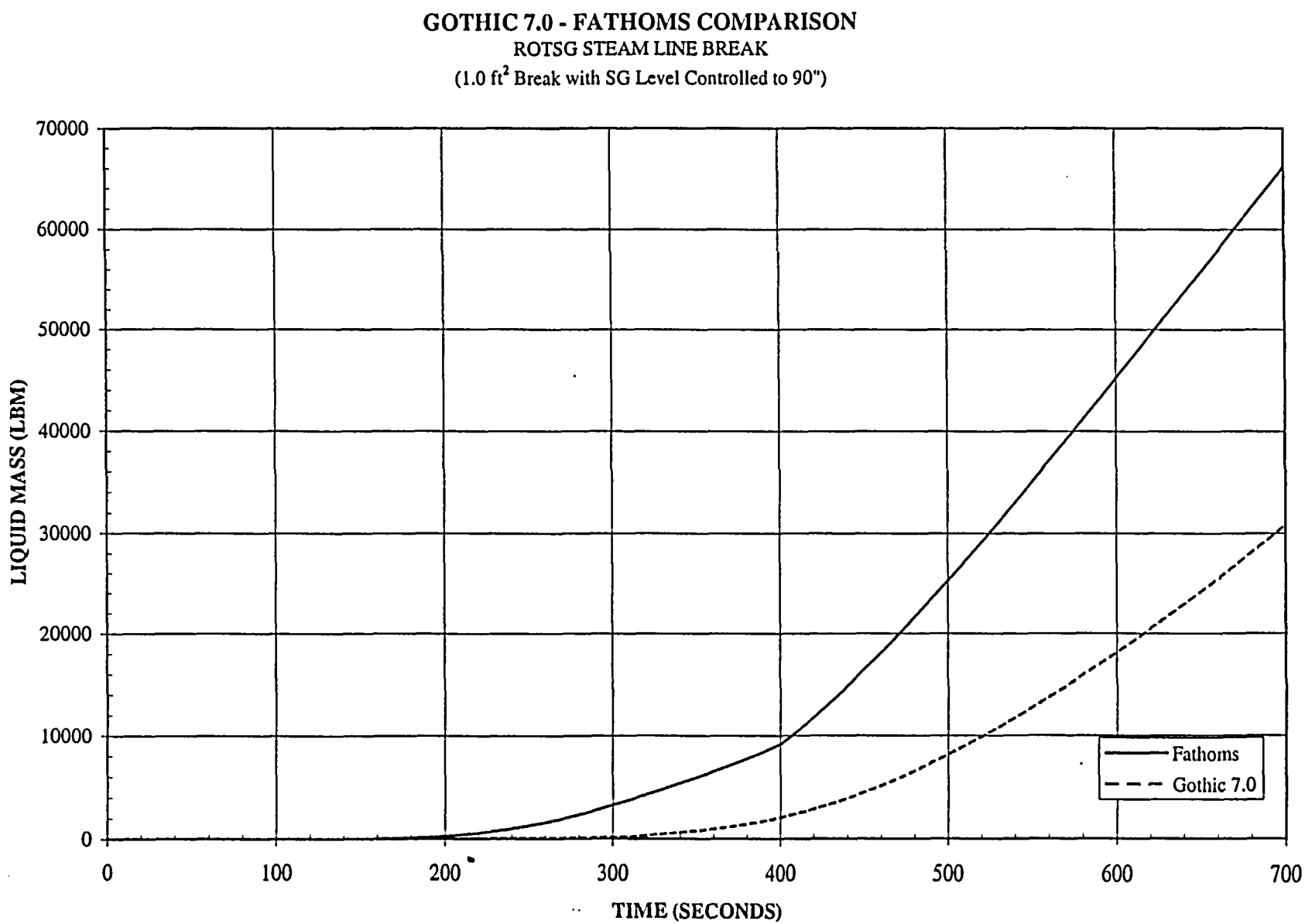

GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(1.0 ft² Break)
LIQUID TEMPERATURE (°F)
250
200
150
100
50
0
0
100
200
300
400
500
600
700
TIME (SECONDS)
Fathoms
Gothic 7.0

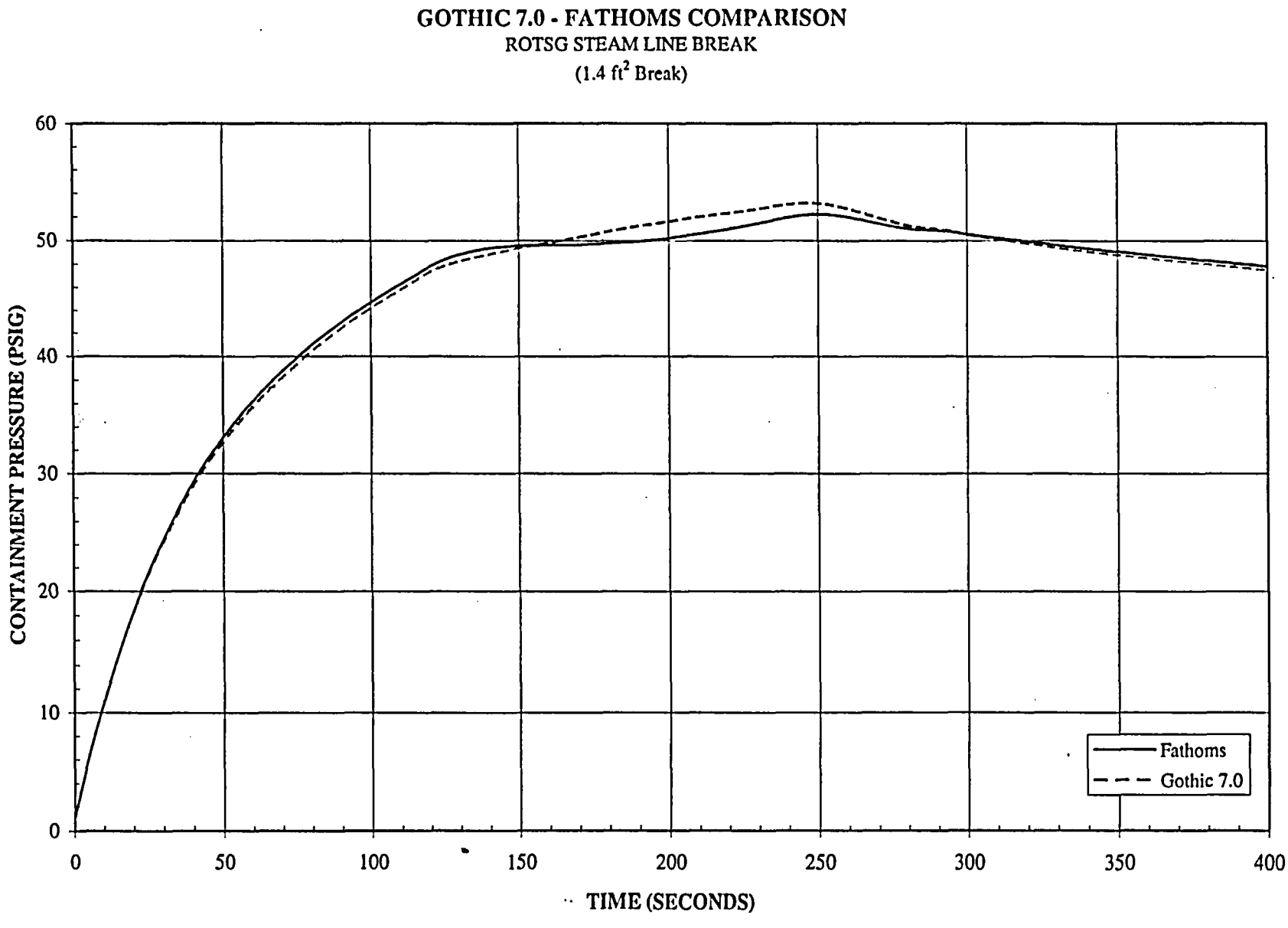

GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(1.0 ft² Break)
LIQUID MASS (LBM)
70000
60000
50000
40000
30000
20000
10000
0
0
100
200
300
400
500
600
700
TIME (SECONDS)
Fathoms
Gothic 7.0


GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(1.0 ft² Break with SG Level Controlled to 90")
CONTAINMENT PRESSURE (PSIG)
TIME (SECONDS)
0
10
20
30
40
50
60
0
100
200
300
400
500
600
700
Fathoms
Gothic 7.0


GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(1.0 ft² Break with SG Level Controlled to 90")
VAPOR TEMPERATURE (°F)
450
400
350
300
250
200
150
100
50
0
100
200
300
400
500
600
700
TIME (SECONDS)
Fathoms
Gothic 7.0


GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(1.0 ft² Break with SG Level Controlled to 90")
LIQUID TEMPERATURE (°F)
250
200
150
100
50
0
0
100
200
300
400
500
600
700
TIME (SECONDS)
Fathoms
Gothic 7.0

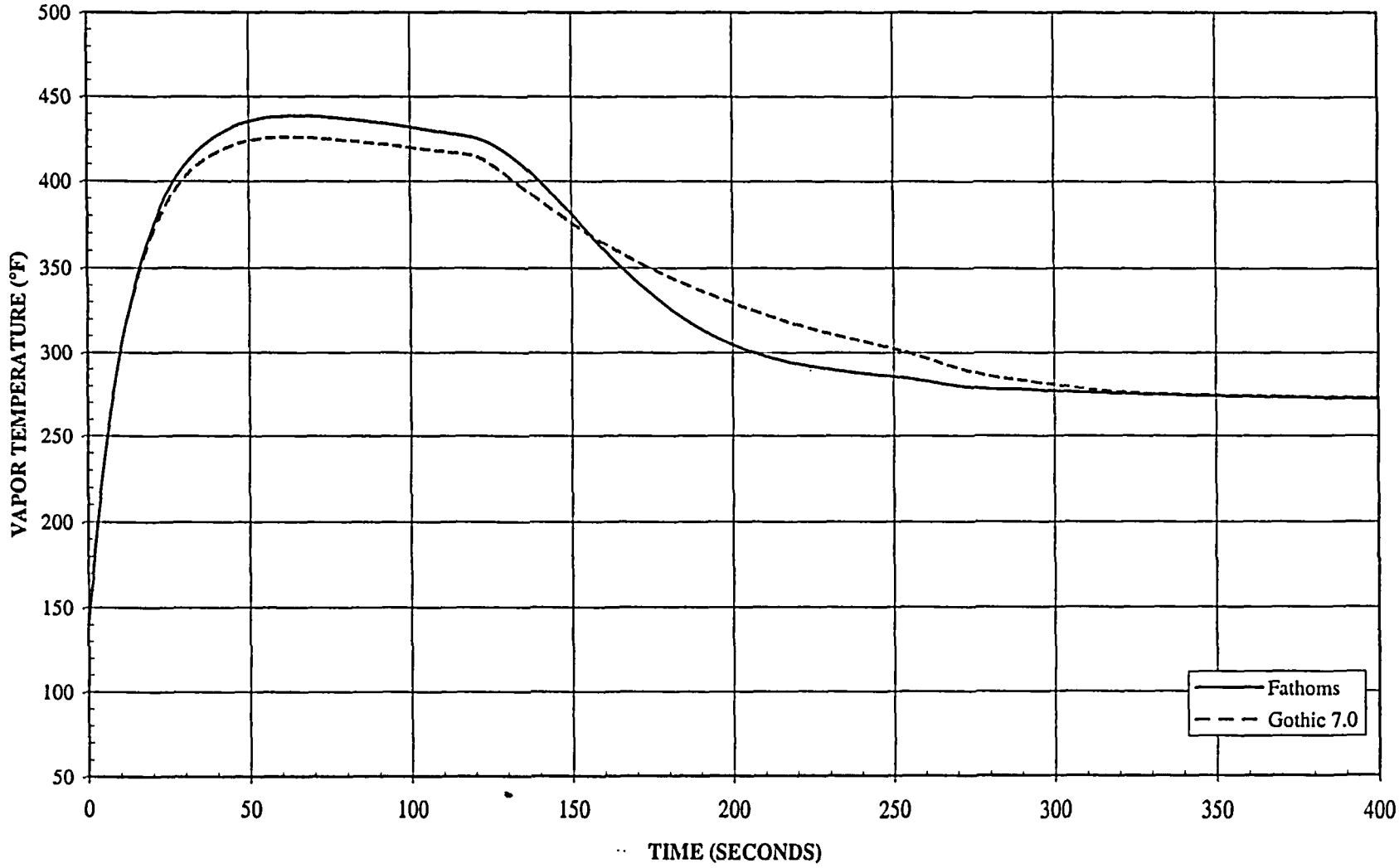

GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(1.0 ft² Break with SG Level Controlled to 90")
LIQUID MASS (LBM)
70000
60000
50000
40000
30000
20000
10000
0
0
100
200
300
400
500
600
700
TIME (SECONDS)
Fathoms
Gothic 7.0


GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(1.4 ft² Break)
CONTAINMENT PRESSURE (PSIG)
TIME (SECONDS)
0
10
20
30
40
50
60
0
50
100
150
200
250
300
350
400
Fathoms
Gothic 7.0

GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG STEAM LINE BREAK

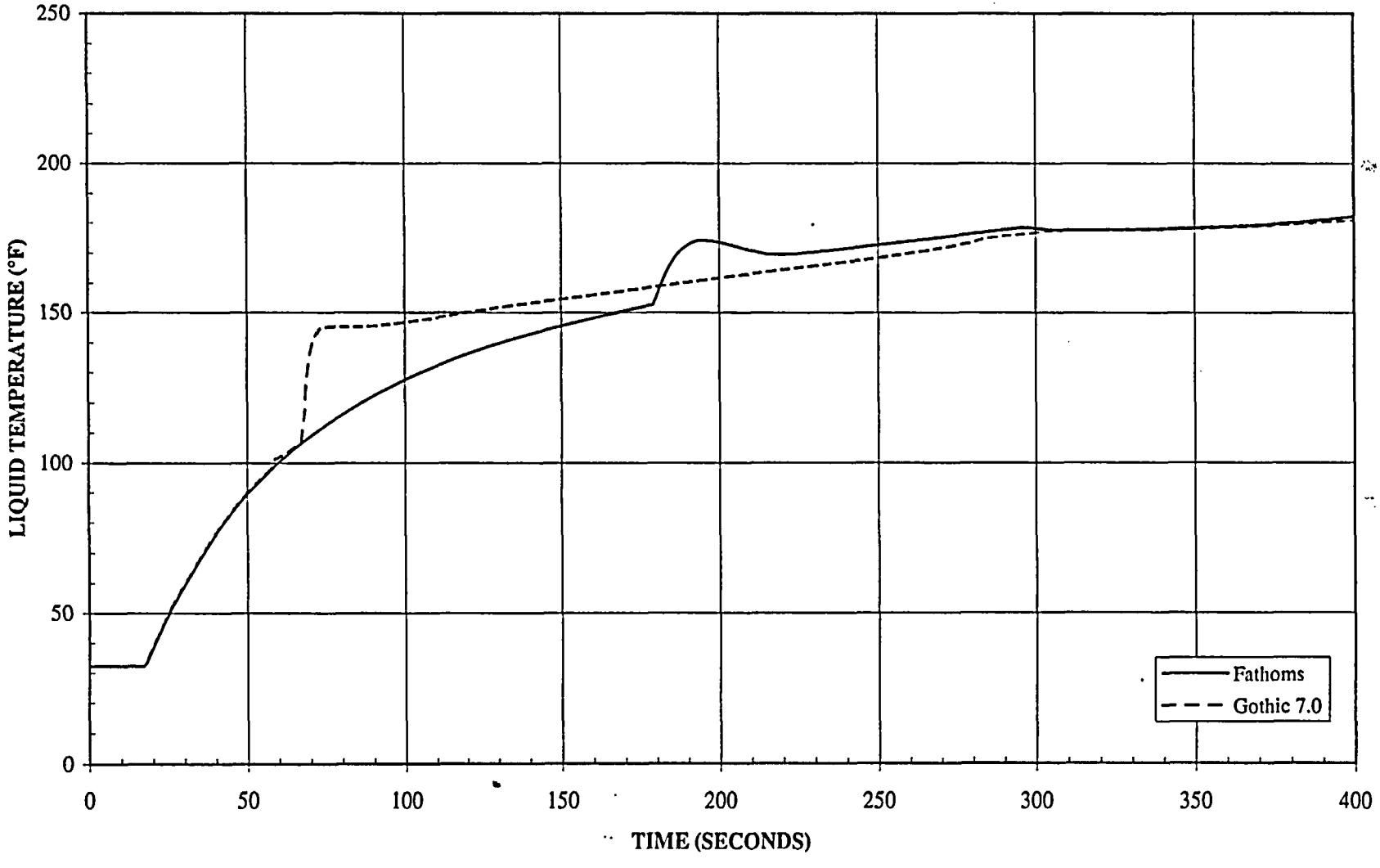
(1.4 ft^2 Break)

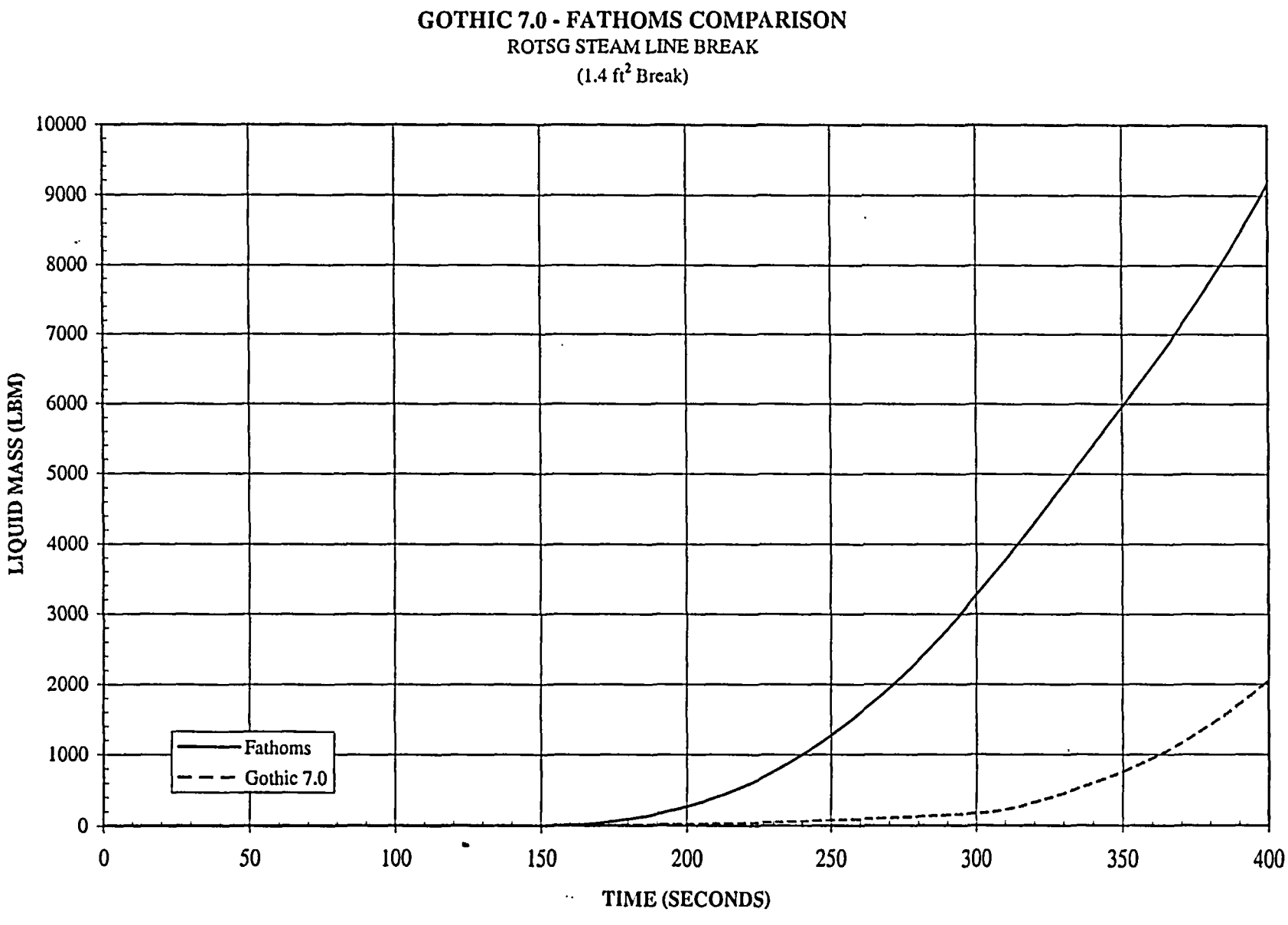


GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG STEAM LINE BREAK

(1.4 ft^2 Break)

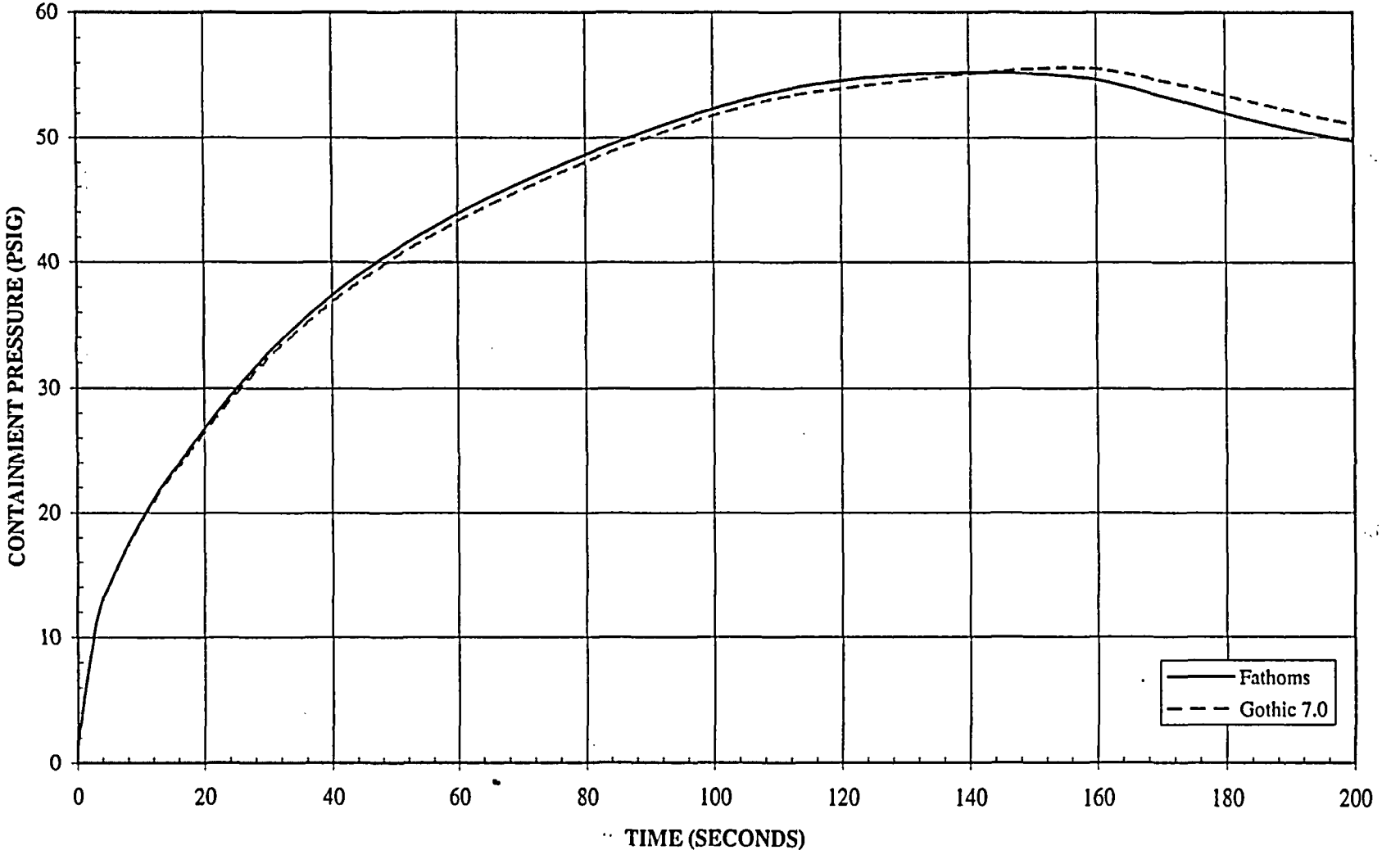



GOTHIC 7.0 - FATHOMS COMPARISON
ROTSG STEAM LINE BREAK
(1.4 ft² Break)
LIQUID MASS (LBM)
10000
9000
8000
7000
6000
5000
4000
3000
2000
1000
0
0
50
100
150
200
250
300
350
400
TIME (SECONDS)
Fathoms
Gothic 7.0

GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG STEAM LINE BREAK

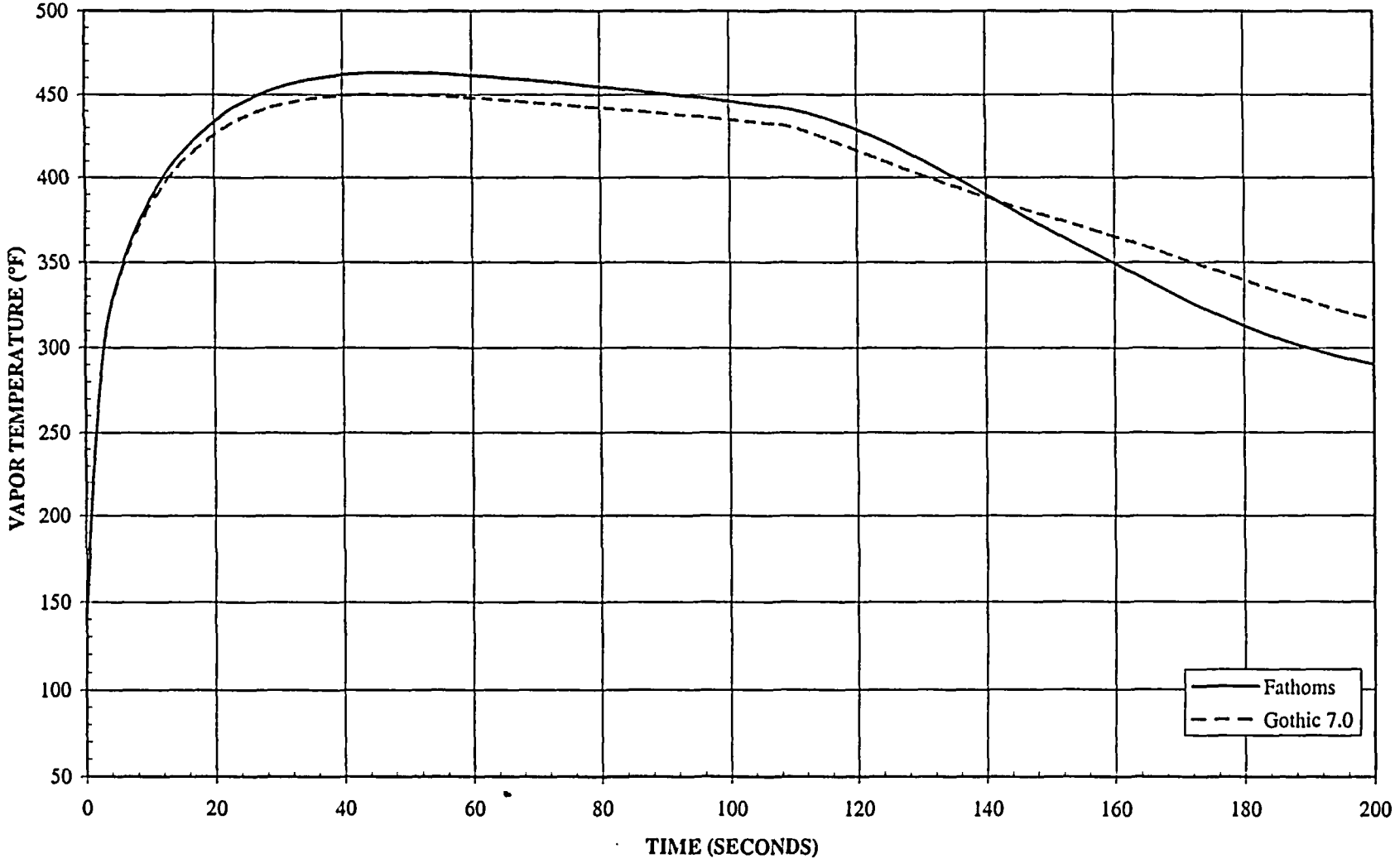
(Double-Ended Guillotine Break)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG STEAM LINE BREAK

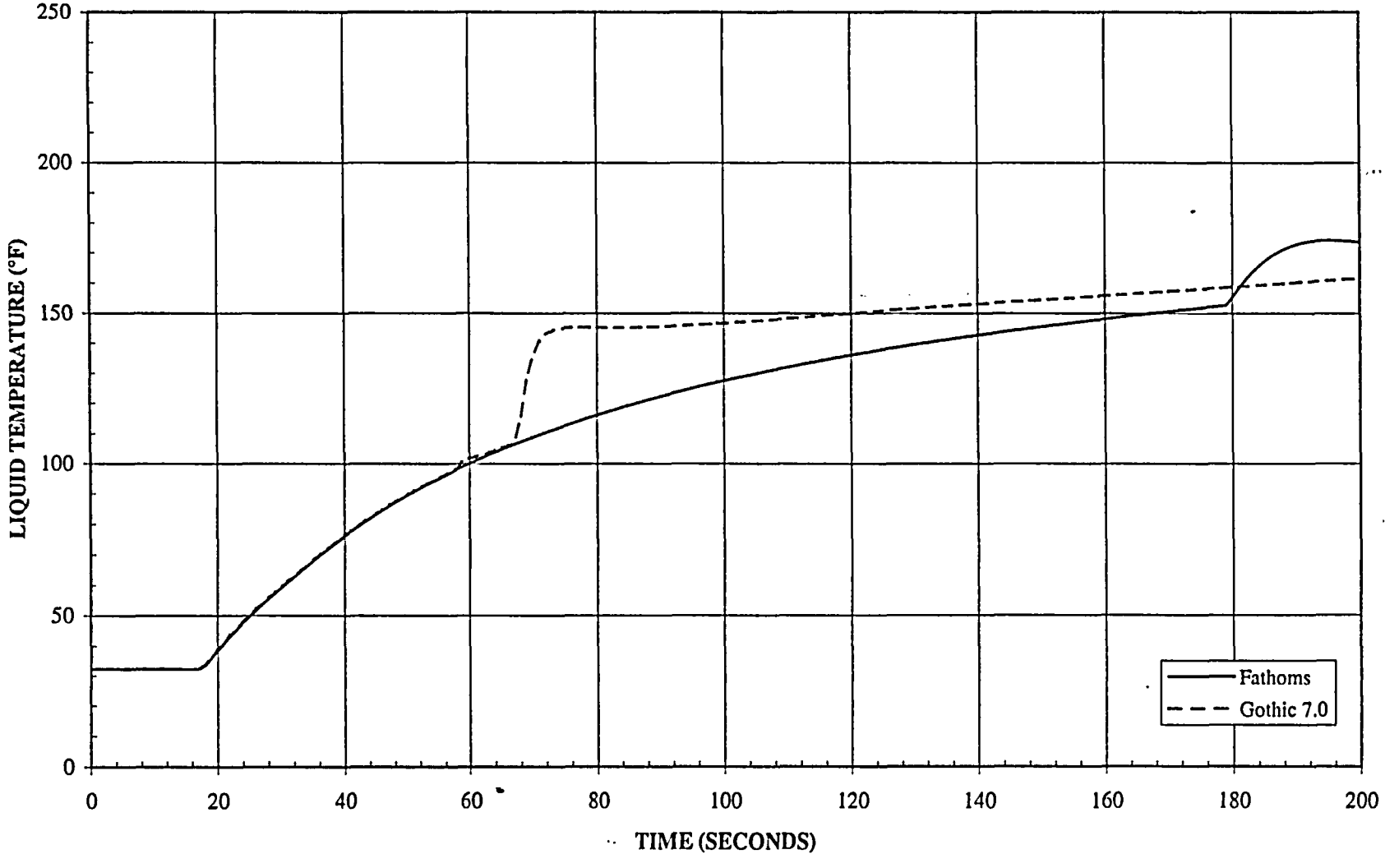
(Double-Ended Guillotine Break)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG STEAM LINE BREAK

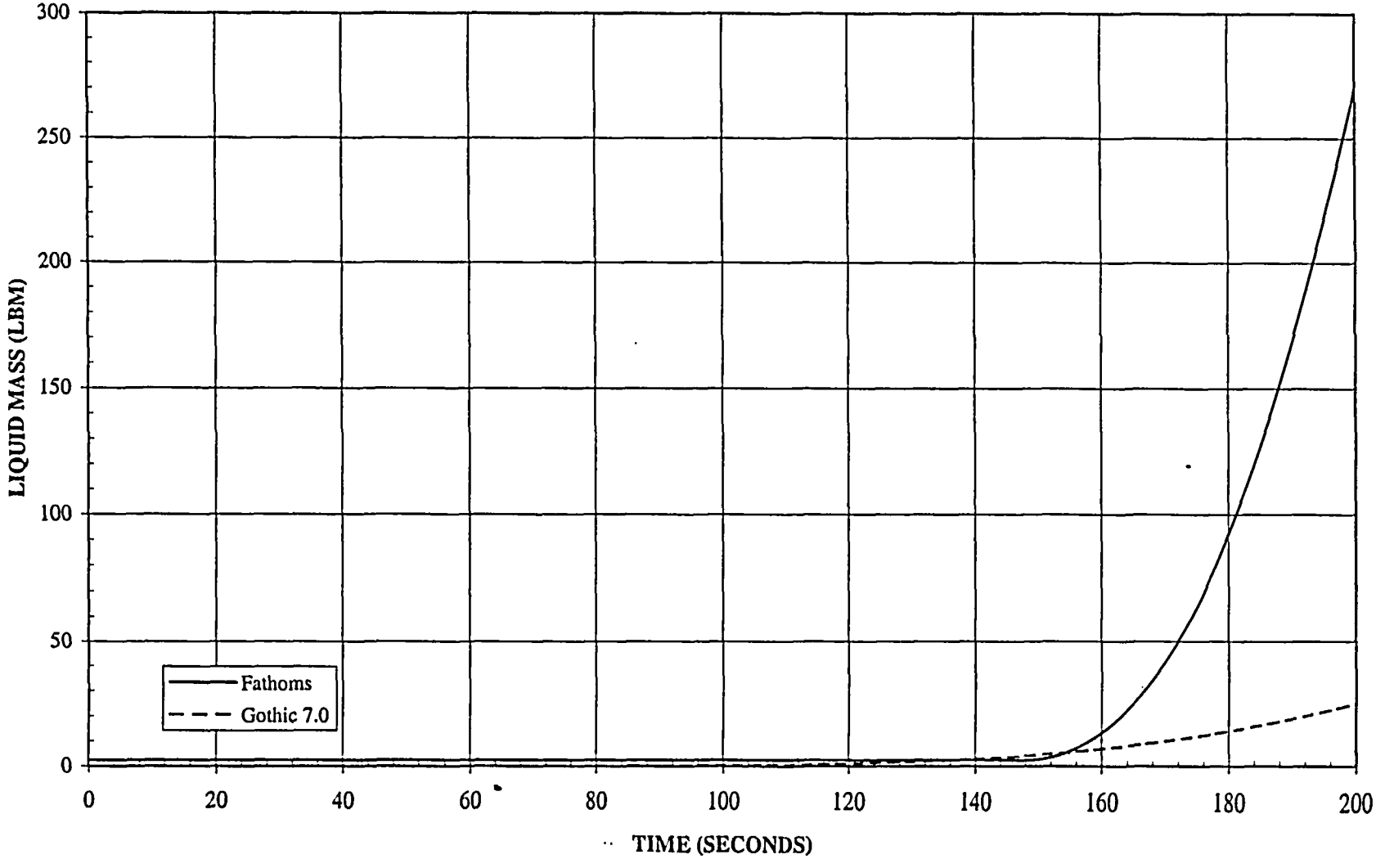
(Double-Ended Guillotine Break)



GOTHIC 7.0 - FATHOMS COMPARISON

ROTSG STEAM LINE BREAK

(Double-Ended Guillotine Break)



15. Will the Mist Diffusion Layer (MDL) model be used in GOTHIC calculations? If so, please provide curves of a main steam line break calculation with and without this model. All other input and assumptions should be as proposed for licensing basis calculations.

Response: The Uchida Condensation Option will be utilized for all heat slabs in the GOTHIC calculations. This is consistent with the guidance in ANSI/ANS 56.4-1983. This represents no change in the containment analysis methodology (consistent with FATHOMS).

16. Will GOTHIC calculations be performed with a pressure flash or temperature flash assumption concerning the distribution of vapor in the containment atmosphere?

Response: The GOTHIC boundary conditions representing LOCA mass and energy releases will partition the flow by phase. Separate flowrates and enthalpies for the steam and liquid phases of the blowdown are calculated in the RELAP5 code and passed to GOTHIC as separate boundary conditions. All liquid flow is entered as droplets when the RCS pressure exceeds the containment pressures. Subsequently the distribution of the droplets is determined by GOTHIC's heat and mass transfer equations. When the RCS pressure has decreased to the containment pressure the liquid flow is entered in the continuous liquid phase, and spills onto the containment floor with little energy transfer. This liquid phase will not flash into steam, as is permitted in the 'pressure flash' model from CONTEMPT4. The steam phase break flow from the RELAP5 blowdown analysis is added directly to the containment atmosphere.

For main steam line break mass and energy releases, the boundary condition is single phase steam, and so the question is not applicable.

17. Will the Drop Liquid Conversion Option be used in GOTHIC calculations? If so, please provide curves of a 14.1 ft^2 hot leg beak at the vessel outlet calculation with and without this model. All other input and assumptions should be as proposed for licensing basis calculations.

Response: The Drop Liquid Conversion Option, which signals the use of the GOTHIC default models for drop phase entrainment, agglomeration, and deposition, will be used in the GOTHIC calculations. This is consistent with the selection of code-calculated de-entrainment fractions, rather than a user-specified value, in previous FATHOMS analyses.

When the 14.1 ft^2 hot leg break at the vessel outlet from the set of GOTHIC runs in response to Question #3 above is re-run with the Drop Liquid Conversion Option set to IGNORE, the GOTHIC peak pressure calculation is identical to the original case with the INCLUDE option. The selection of this option has no impact on the calculated peak pressure. Since the results with this option removed are identical to the base case, no curve is provided.

Attachment 2

Additional Revisions To

DPC-NE-3003, Revision 1
"Mass and Energy Release and Containment Response Methodology"

The following two items are additional revisions that were not included in the scope of the original Revision 1 submitted for NRC review by letter dated June 13, 2002.

1. Appendix C, "GOTHIC Version 7.0 Code", of Revision 1 of DPC-NE-3003 (Attachments 3 & 4 of the June 13, 2002 submittal) describes Duke's proposed use of the GOTHIC 7.0 code for the containment response analysis. The following paragraph is from pp. 3-62 and 4-62 of Attachments 3 and 4, respectively.

> *"GOTHIC 7.0 features the use of control variables that are not available in FATHOMS. The control variables provide enhanced capability and flexibility to input boundary conditions or forcing functions into the code. For instance, the output of the BFLOW code to determine the quality of water exiting a cold leg break for long-term mass and energy releases can be input via control variables. This capability was not available for FATHOMS and specific code changes were implemented in order to input the BFLOW code results (Section 2.4.1.2)."*

Subsequent to the submittal of Revision 1, Duke has decided not to use the GOTHIC 7.0 control variables for input of the output of the BFLOW code. Rather, GOTHIC 7.0 has been modified similar to the modifications previously made to FATHOMS to input the results of the BFLOW methodology. Therefore, the GOTHIC 7.0 and FATHOMS methodology will be the same with regard to how the results of BFLOW are input. The above paragraph will be revised as follows to indicate this similarity:

> *"GOTHIC 7.0 features the use of control variables that are not available in FATHOMS. The control variables provide enhanced capability and flexibility to input boundary conditions or forcing functions into the code. These are essentially user convenience features and do not constitute elements of the methodology.*
>
> *The long-term mass and energy release boundary condition modeling in FATHOMS has been similarly installed in GOTHIC 7.0. This modeling approach, which enables interpolation of the BFLOW code results as an input boundary condition, are described in Sections 2.2.1.1, 2.4.1.2, and 6.3 of DPC-NE-3003."*

2. Due to the code modification to GOTHIC Version 7.0 described in Item #1 above, the code version must be uniquely identified. This Duke-specific version will be designated GOTHIC 7.0/DUKE in the published version of the report.

Duke Power
A Duke Energy Company

Ken S. Canady
Vice President
Nuclear Engineering

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July 28, 2003

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington D. C. 20555-0001

ATTENTION: Chief, Information Management Branch
Division of Program Management
Policy Development and Analysis Staff

Subject: Duke Energy Corporation
Oconee Nuclear Station - Units 1, 2, and 3
Docket Nos. 50-269, 50-270, and 50-287

Revisions to Topical Report DPC-NE-3003 In Support of Steam Generator Replacement
Response to NRC Staff Request for Additional Information

Reference: Duke Submittal Dated June 13, 2002

Enclosed herein, please find the Duke Energy Corporation (Duke) response to question 13 of the May 12, 2003 NRC staff's request for additional information concerning topical report DPC-NE-3003, Revision 1, "Mass and Energy Release and Containment Response Methodology."

Attachment 1 to this letter constitutes Duke's response.

This completes Duke's response to all questions asked in the staff's May 12, 2003 request for additional information. If there are any questions or if additional information is needed on this matter, please call J. A. Effinger at (704) 382-8688.

Very truly yours,

K. S. Canady

Attachment

xc: with attachment

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Attachment 1

Duke Power Response To
Request For Additional Information - Question 13

DPC-NE-3003 Revision 1
"Mass and Energy Release and Containment Response Methodology"

13. Provide a temperature envelope curve for EQ purposes calculated with GOTHIC and with FATHOMS. Discuss the differences in EQ envelopes calculated with FATHOMS and GOTHIC due to the different models in the two codes, especially those models which affect moisture in the atmosphere.

Response: A comparison of the results of analyses of the long-term cold leg break LOCA containment response with GOTHIC and FATHOMS is shown in the following figures. This event is limiting for Oconee for the long-term EQ temperature. The containment vapor temperature, liquid temperature, and pressure trends are shown for the limiting break location.

Figure 13.1 shows the predicted containment vapor temperature trends for the GOTHIC and FATHOMS analyses, overlaid with the EQ envelope. It is noted that this is an updated version of the EQ envelope shown in Figure 6.3-4 of DPC-NE-3003. There is excellent agreement between the FATHOMS and GOTHIC analyses for the Reactor Building vapor temperature trend. The GOTHIC analysis shows a slightly higher vapor temperature, exceeding the FATHOMS result by about 3°F at ten hours into the event.

Figure 13.2 shows the comparison of the containment pressure results. There is good agreement between the GOTHIC and FATHOMS analyses. The FATHOMS results are slightly higher in the first few minutes of the transient, as noted in the response to Question 3. In the long term, the GOTHIC results are about 1 to 2 psi higher, with the analyses results converging to nearly the same pressure after several days into the transient.

Figure 13.3 shows the comparison of the containment liquid temperature results. The FATHOMS result is one sump temperature curve since the liquid and droplet phases are assumed to be at the same temperature. The GOTHIC results are for the sump liquid temperature, the droplet temperature, and the film temperature. The GOTHIC droplet and film temperatures are significantly higher than the sump liquid temperature for the first hour due to significant liquid phase break flow. The curves for these parameters stop at about 6000 seconds due to an insignificant mass for these two phases. The GOTHIC sump temperature trends closely with the FATHOMS sump temperature beginning at approximately 3000 seconds. This is consistent with nearly all of the liquid phase being in the sump at this point in time and beyond.

In Figure 13.4, a mass-weighted liquid temperature from the GOTHIC analysis is compared with the results from FATHOMS. The GOTHIC results are an average for the combined liquid and droplet phases to facilitate an equivalent comparison with the FATHOMS results, in which the liquid and droplet phases are at the same temperature. Early in the GOTHIC analysis, a much greater percentage of the liquid present in the containment is in a film on the building structures, or in the form of entrained droplets, relative to the FATHOMS analyses. In the FATHOMS results, the liquid reaches the containment floor in the continuous liquid phase earlier. The

FATHOMS liquid temperature is lower than GOTHIC for about the first 10 minutes of the analysis for this reason. At about 7 to 8 minutes into the transient, the amount of droplets present in the GOTHIC analysis decreases as the total mass flow rate from the break decreases. This causes the GOTHIC liquid temperature to drop and to trend in a similar manner as the FATHOMS liquid temperature for the remainder of the analysis. In the long term, the mass of entrained droplets in the GOTHIC analysis is actually less than in the FATHOMS analysis. The liquid temperature in both analyses is in excellent agreement well before the switchover to sump recirculation mode at 5466 seconds, with a difference of 1-2°F for the remainder of the transient.

Since the GOTHIC liquid temperature is slightly less than the FATHOMS liquid temperature during sump recirculation mode, there is slightly less energy being removed by the LPI coolers in the GOTHIC analysis than in the FATHOMS analysis. Over time this results in a higher containment pressure in GOTHIC since more energy is left in the vapor phase. Also, the break quality in the GOTHIC analysis, as calculated by the BFLOW results in the long-term mass and energy release model, will be slightly higher in the GOTHIC analysis due to higher pressure. This causes the containment vapor temperature in the GOTHIC analysis to be slightly higher than the FATHOMS results after several hours in the sump recirculation mode. As the decay heat level decreases, the cooler sump temperatures in GOTHIC allows the GOTHIC vapor temperature to decrease (due to cooler ECCS water and cooler spray flow) and approach the FATHOMS analysis results in the latter stage of the transient.

ROTSG - Oconee Large Break LOCA
Cold Leg Break (Limiting EQ)
Long-Term Containment Response

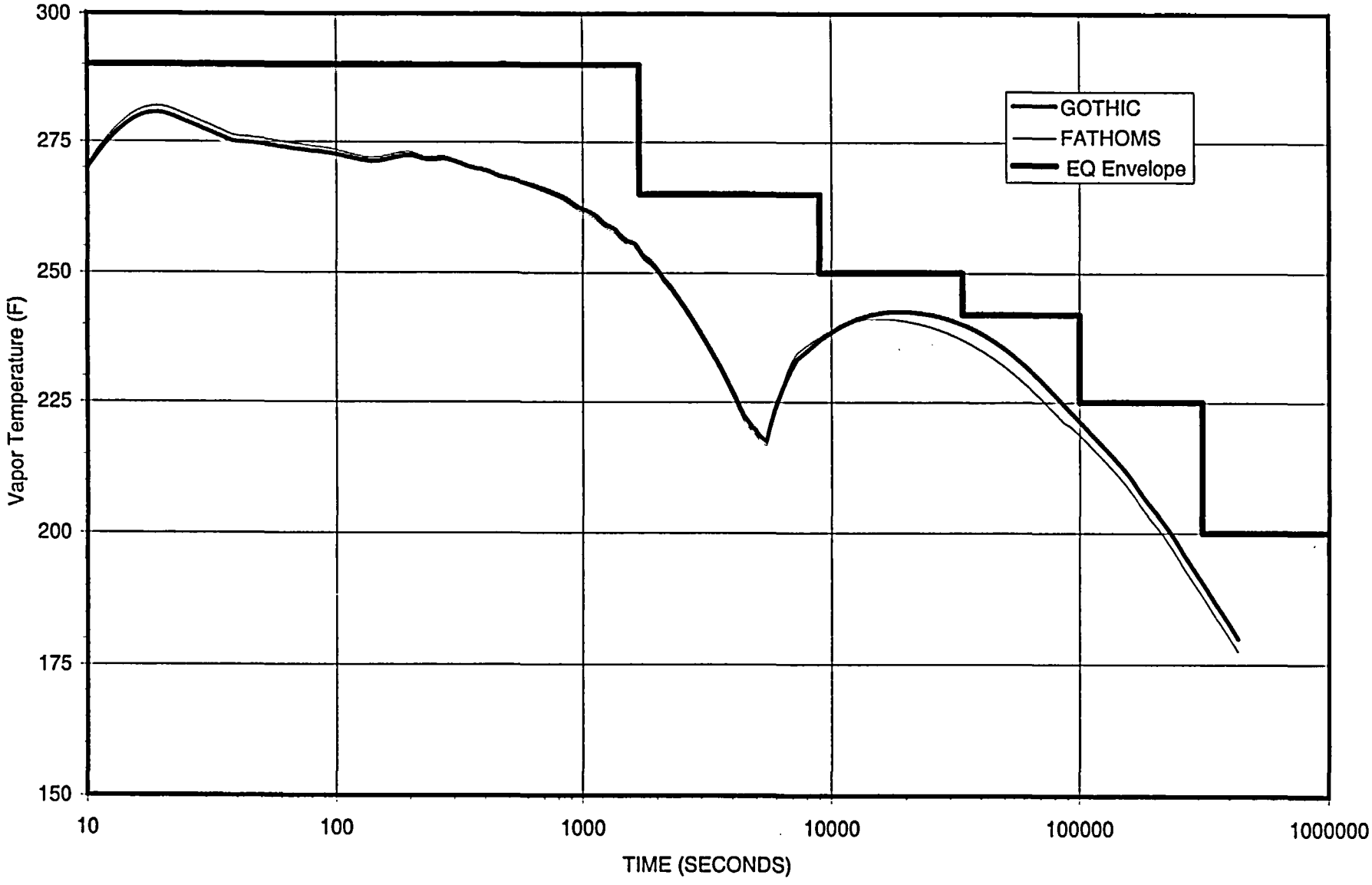

FIGURE 13.1
Reactor Building Temperature

ROTSG - Oconee Large Break LOCA
Cold Leg Break (Limiting EQ)
Long-Term Containment Response

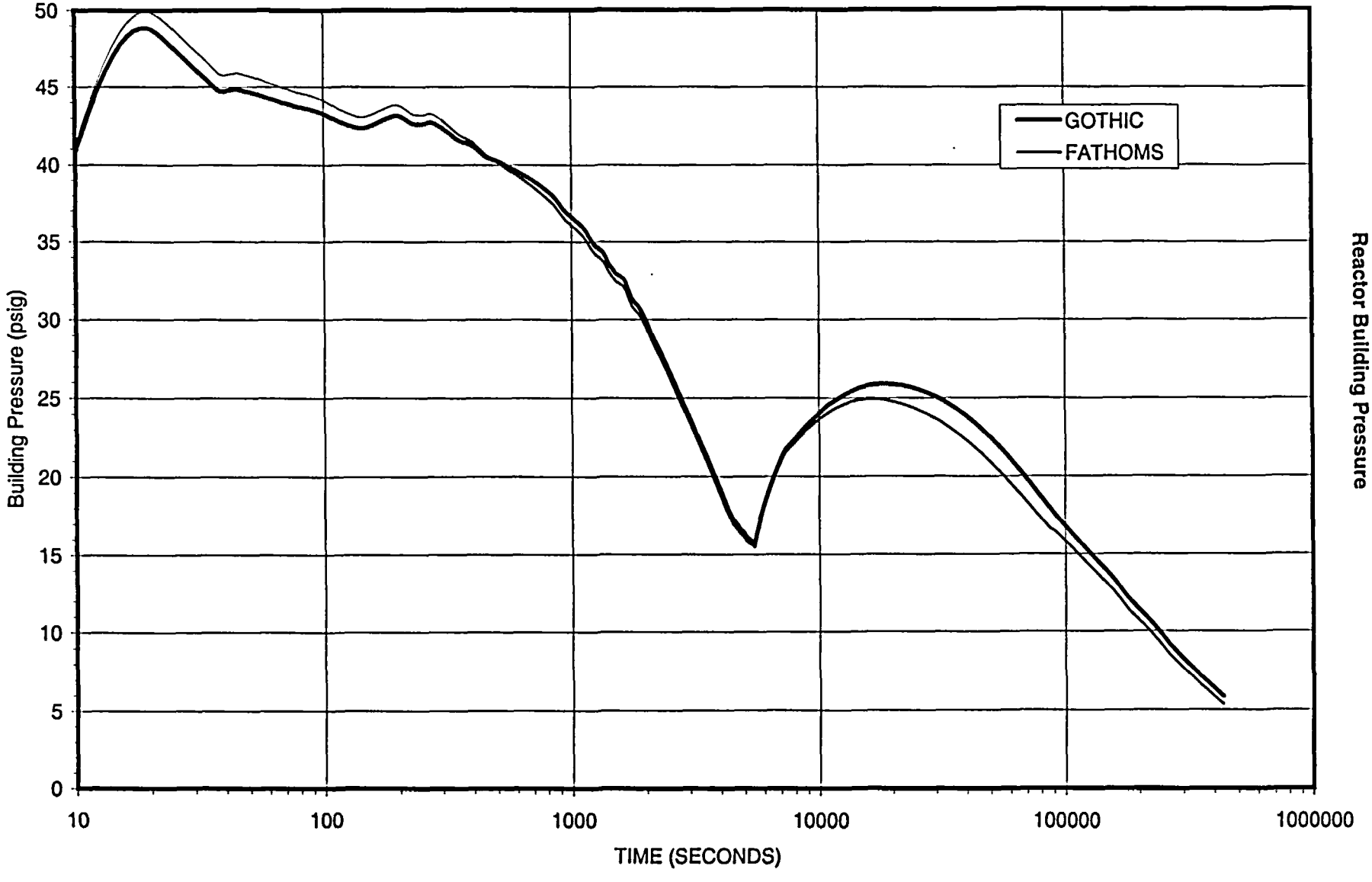

FIGURE 13.2
Reactor Building Pressure

ROTSG - Oconee Large Break LOCA
Cold Leg Break (Limiting EQ)
Long-Term Containment Response

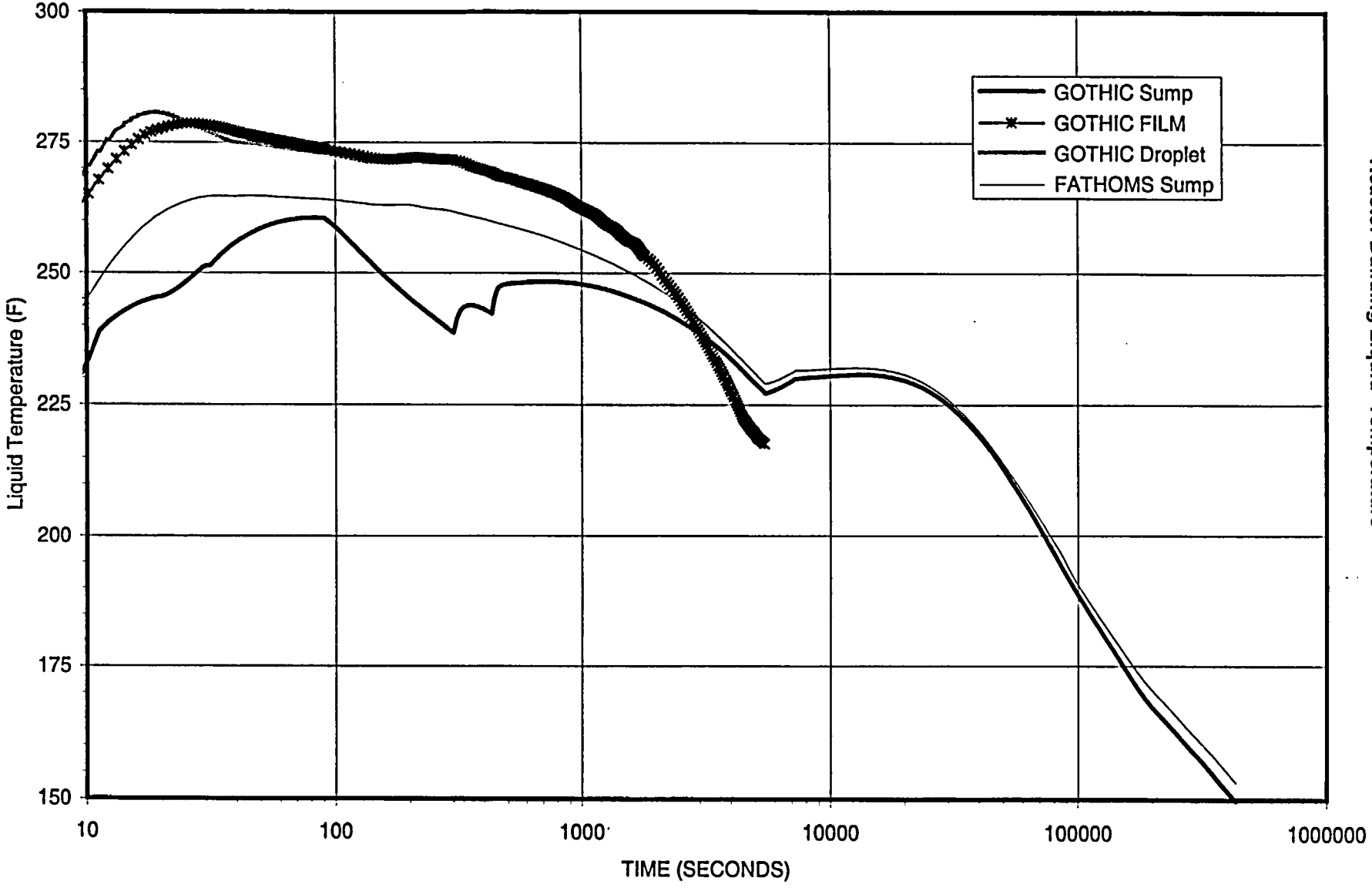

FIGURE 13.3
Reactor Building Liquid Temperature

ROTSG - Oconee Large Break LOCA
Cold Leg Break (Limiting EQ)
Long-Term Containment Response

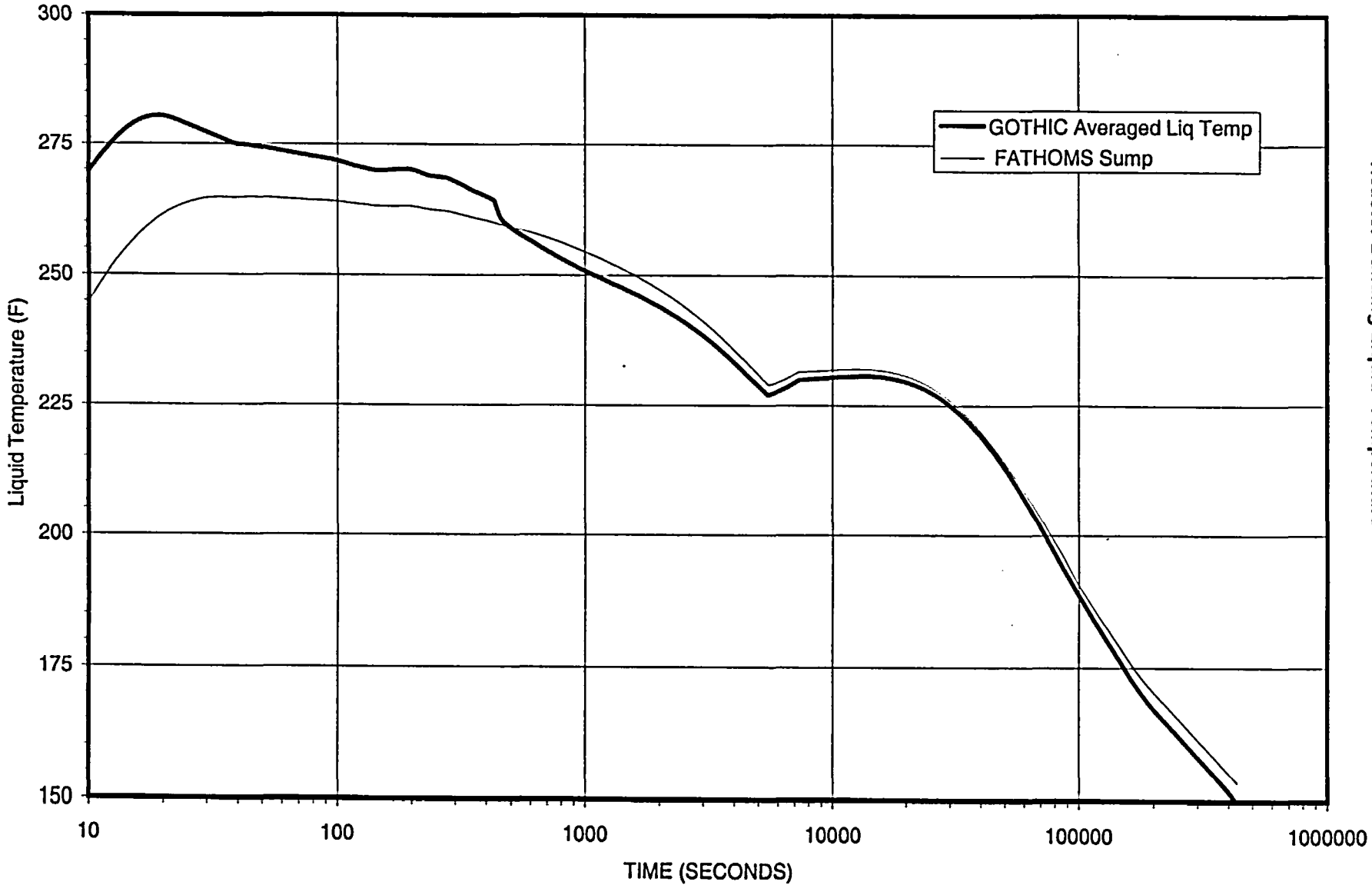

FIGURE 13.4
Reactor Building Liquid Temperature