WESTINGHOUSE PROPRIETARY CLASS 3

WCAP-12338

CONTAINMENT MARGIN IMPROVEMENT ANALYSIS
FOR
INDIAN POINT UNIT 3

By

J. A. Kolano
L. C. Smith
L. A. Wooten
G. G. Ament

MAY 1989

WESTINGHOUSE ELECTRIC CORPORATION
Energy Systems Business Unit
P.O. Box 355
Pittsburgh, Pennsylvania 15230

EXECUTIVE SUMMARY

Containment Integrity Analyses have been performed as part of the Indian Point Unit 3 Containment Margin Improvement Program. The objective of the progr a was to provide containment integrity analysis results using the current Indi Point Unit 3 specific information and new more realistic models. In this way the licensing basis for Unit 3 is clarified and updated, and pressure margin for operation of Unit 3 has been determined and thus made available for possible future use.

The results of the analysis ensure that the pressure inside containment will remain below the containment building design pressure if a Loss-of-Coolant-Accident (LOCA) or a rupture of main steamline pipe (MSLB) inside containment should occur during plant operation. The peak calculated pressure for the limiting LOCA and MSLB events respectively are 40.3 psig and 42.42 psig. The design pressure is 47 psig.

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF FIGURES

TABLE OF CONTENTS

LIST OF FIGURES (Continued)

TABLE OF CONTENTS

LIST OF FIGURES (Continued)

TABLE OF CONTENTS

LIST OF FIGURES (Continued)

TABLE OF CONTENTS

LIST OF TABLES

TABLE OF CONTENTS

LIST OF TABLES (Continued)

1.0 INTRODUCTION

Containment Integrity Analyses are performed during nuclear plant design to ensure that the pressure inside containment wi l remain below the containment building design pressure if a Loss-of-Coolant-Accident (LOCA) or a rupture of main steamline pipe (MSLB) inside containment should occur during plant operation. The analysis ensures that the containment heat removal capability is sufficient to remove the maximum possible discharge of mass and energy to containment from the Nuclear Steam Supply System without exceeding design pressure. The analysis can also be used as a basis for the containment leak rate test pressure to ensure that dose limits will be met in the event of release of radioactive material to containment.

1.1 LOCA BACKGROUND

The most limiting analysis case for Containment Integrity for LOCA, the double ended pump suction guillotine break case, currently described in the Indian Point 3 FSAR was originally performed by Westinghouse in 1974. The methods used in determining the LOCA mass and energy releases are described in Reference 1. The Containment pressure calculation was done with the COCO computer code (Reference 2). The peak pressure from this calculation was 40.6 psig, which is well below the containment design pressure of 47 psig.

Subsequent analysis was performed by Westinghouse in 1986 (Reference 3) for justification of an increase in maximum normal operation of containment temperature to 130 degrees Fahrenheit and in 1987 (References 4&5) to investigate margins in the containment fan cooler heat removal system. For these analyses the same mass and energy releases were used as the 1974 FSAR analysis. The COCO computer code was also used, but with input modified to reflect the conditions under study.

In 1988, Westinghouse provided results of analysis performed using recently computed bounding Indian Point Unit 2 specific mass and energy information and new more realistic computer models to be used as a justification of continued operation for Indian Point Unit 3 (Reference 6). The mass and energy models utilized are discussed in Reference 7.

1.2 MSLB BACKGROUND

The containment response to a steam line break has been found to be dependent upon several factors including break size, power level single failure assumption and containment wall heat transfer modelling. he original FSAR steamline break limiting case was a full double-ended break (3.687 ft^2 for each end) at 0 % power and included failures of the main steamline non-return valve and one train of safety injection and containment heat removal. The 0% power assumption was justified based on the initial inventory of the steam generators being greater at 0% power than at higher power. The original and existing FSAR analysis for Main Steamline break was performed in 1970 and resulted in a peak pressure of 41 psig.

In 1986, Westinghouse performed a study to support removal of the Boron Injection Tank. The assumptions used for the mass/energy releases in this study included 0 % power and multiple equipment failures comparable to those assumed in the original FSAR analysis. Westinghouse provided a revised Boron Injection Tank analysis result of 42.93 psig (Reference 6) in 1988. This analysis did not assume failure of the Main Steam Check Valve (MSCV), but did assume failure of a containment heat removal train and an SI pump and 30 minute operator action time. This was the most limiting case for the 0% power level analyzed. The initial containment temperature assumed was 130 degrees Fahrenheit.

In 1988, Westinghouse provided (Reference 8) revised zero power double-ended analyses for Main Steamline break comparable to the BIT Removal Analysis with the BIT in service for various failure assumption cases. The limiting cases from these revised analyses were a diesel failure (peak pressure of 37.79 psig) and MSCV failure (peak pressure of 37.49 psig).

Also in 1988, Westinghouse performed a feasibility study to determine if removing multiple failure assumptions used in previous analyses would result in lower calculated peak pressures without the BIT in service. The intent of this study was to provide confidence that a more comprehensive study would be worthwhile in providing additional margin. The study indicated that significant pressure benefit could be gained by a single failure study and also if the operator action time assumed was reduced to 10 minutes from 30 minutes. The analysis result was 34.98 psig assuming the failure of a single train of safeguards (e.g., 1 train of SI, and 1 train of containment heat removal) and 10 minute operator action time. This result could be compared to a multiple failure case with 30 minute operator action time that exc eded design pressure and a single failure case with 30 minute operator action time which resulted in a calculated peak pressure of 42.93 psig.

1.3 PURPOSE OF ANALYSES

Numerous design and operational parameters affect the results of Containment Integrity Analysis and the more significant parameters are limited by Technical Specifications and other administrative controls. The purpose of the analyses described herein is to present the results of a program of analyses that optimizes the operation at Indian Point 3 with respect to containment pressure design margin as well as ensures that adequate safety margin is maintained. This margin can be utilized for future applications, such as: (a) to offset any future degradation of Emergency Safeguards Systems, such as the Containment Heat Removal Systems; or (b) to improve maintenance efficiency, such as performing work on more than one Emergency Fan Cooler at a time. This margin can also be utilized to develop a flexible Technical Specification that could support a more flexible operating state.

The objective of the program discussed herein is to provide containment integrity analysis results using the latest Indian Point Unit 3 specific information and the new more realistic models (Reference 7) which were invoked for the May 1988 JCO (Reference 6) for Indian Point Unit 3. In this way the licensing basis for Unit 3 is clarified and updated, and the maximum margin for operation of Unit 3 can be determined and thus made available for possible future use.

The analyses presented in this report address the consequences of the mass and energy that is released to containment as a result of a design basis Loss-of-Coolant Accident (LOCA) and a design basis main steamline break (MSLB). The mass and energy release data is subsequently used to verify, via calculations, that the containment design pressure is not exceeded in the event of a LOCA or a MSLB. In this manner, the analysis results demonstrate that the containment integrity has not been compromised. Section 2.0 and 3.0 present the LOCA and MSLB mass and energy release analysis respectively, and Section 4.0 and 5.0 discuss the results of the containment integrity response calculations.

Bounding initial temperatures and pressures for the containment integrity analysis were selected to envelop the limiting conditions for operation. In this manner, the most limiting conditions for operation at full power (3025 MWt) were conservatively chosen.

1.4 MAJOR ANALYTICAL ASSUMPTIONS FOR LOCA

The evaluation model for the long term LOCA mass and energy release calculations used was the March 1979 model described in reference 7. This evaluation model has been reviewed and approved by the NRC, and has been used in the analysis of other dry containment plants.

For the long term mass and energy release calculations, operating temperatures for the highest average coolant temperature case were selected as the bounding analysis conditions. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures which are at the maximum levels attained in steady state operation. Additionally, an allowance of +5.0 °F is reflected in the temperatures in order to account for instrument error and deadband. The initial RCS pressure in this analysis is based on a nominal value of 2250 psia. Also included is an allowance of +40 psi, which accounts for the uncertainty on pressurizer pressure. The selection of 2250 psia as the limiting pressure is considered to affect the blowdown phase results only since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure (2250 psia). Additionally the RCS has a higher fluid density at 2250 psia (assuming a constant temperature) and subsequently has a higher RCS mass available for release. Thus, 2250 psia initial pressure was selected as the limiting case for the long term mass and energy release calculations. These assumptions conservatively maximize the mass and energy in the Reactor Coolant System.

The selection of fuel allowance for the long term mass and energy calculation and subsequent LOCA containment integrity calculation is based on the need to conservatively maximize the core stored energy. The margin in core stored energy was chosen to be +15 percent. Thus, the analysis very conservatively accounts for the stored energy in the core.

The break locations and sizes analyzed are the Double-ended Pump Suction and the Double-ended Hot Leg pipe breaks. The Double-ended Pump Suction case has been the limiting containment integrity LOCA break for Indian Point 3 and also has been the limiting break for the post-blowdown period for all analyses completed with the 1979 Model (Reference 7) to date. The Double-ended Hot Leg case is typically the most limiting with respect to blowdown peak pressure and temperature and can be the most limiting overall case if the containment heat removal capability can sufficiently mitigate the post-blowdown mass and energy releases for the Pump Suction break. For the Hot Leg break case only the blowdown portion of the transient is analyzed, since the containment pressure quickly decreases following the end of blowdown.

The limiting single failure scenario is the failure of a diesel following a loss of offsite power. This scenario has also been demonstated to be limiting for all 1979 Model analyses because it results in failure of one train of containment heat removal and reduced safety injection capability.

The analysis models employ design-basis input assumptions relative to plant operation and hardware. Specific input limited by Technical Specifications will be based on the limiting condition allowed.

Thus, based on the previously mentioned conditions and assumptions, a conservative bounding analysis of Indian Point Unit 3 is made for the discharge of mass and energy from the RCS in the event of a LOCA at 3025 MWt.

In the case of the containment integrity peak pressure calculations, the analysis utilizes the COCO [Reference 2] evaluation model. This model has been successfully used for the other dry containment plants in their FSAR analyses.

As input to the COCO computer code, LOCA mass and energy release rates as described in Section 2 of this report are used.

1.5 MAJOR ANALYTICAL ASSUMPTIONS FOR MSLB

A series of cases to determine the peak pressures for main steamline breaks inside containment are investigated. The analysis is performed in steps such that determination of limiting conditions with respect to a single equipment failure and power are determined. By eliminating all break size dependent items, such as revaporization, non-stagnant wall heat transfer coefficients and break size dependent protection signals, the double-ended ruptures are pressure limiting. Consideration is also given to additional benefits such as reduced valve closure times and reduced feed flashing volume that may provide additional margin but also which may affect the limiting break size, the limiting failure and the power assumptions.

The LOFTRAN code, Reference 9, is used to generate mass and energy releases specific to Indian Point 3. The COCO containment analysis code, as described in Section 2 of this report is used to calculate the containment pressure and temperature time histories.

The following assumptions are utilized for the MSLB analyses:

- 44F Steam Generators with 1.4 Ft^2 integral flow restrictors
- BIT Removal - 0 ppm boron concentration in BIT Tank
- Fan Cooler heat removal based on 95 degrees Fahrenheit Service Water Temperature

- No entrainment of water in steam blowdown (This is a break size dependent assumption with entrainment above a certain break size and no entrainment below. Therefore, no entrainment is conservatively assumed.)

- 30 minute operator action time for isolation of auxiliary feedwater to faulted steam generator

- Minimum SI (with a 6 second pure time delay) and Containment Spray performance characteristics consistent with the number of operating trains

- Vantage 5 Fuel, which is being loaded in Cycle 7 and up, is modeled.

- Reduced Shutdown Margin of 1.3% - for future core design flexibility

- SG Tube Plugging levels less than or equal to 24% uniform and asymmetric

The following assumptions are used to determine the limiting power and single failure conditions for the determination of the peak containment pressure:

- Full Double-ended rupture between the flow restrictor and the containment wall with effective break area limited by flow restrictor as appropriate

- The operation of SI, RCPs, Feedwater pumps and Containment heat removal equipment is consistent with the failure assumption (The most limiting values are used.)

- Elimination of all break size step change dependencies including:

    No revaporization of containment wall condensate assumed

    Limiting wall heat transfer coefficients

- Credit for Containment Signals only (High 1 and High 2 pressure) for reactor trip, SI Steamline Isolation and feedwater isolation (By eliminating all break size dependent items the double-ended ruptures are pressure limiting. Although taking credit for other signals would result in improved results, credit for other signals may result in a break size other than the largest double-ended rupture being more limiting in pressure).

The Feedwater Control Valve (FCV) failure case addresses I.E. Bulletin 80-04 concerns regarding additional feedwater flow due to FCV failure and failure of AFW runout protection. Feedwater flow as a function of pressure in the steam generator is maximized to the faulted steam generator and minimized to the intact steam generators. Maximizing flow to the faulted steam generator provides more inventory for release and tends to keep pressures up thereby maximizing blowdown. Initial assumptions include conservative conditions to bound I.E. Bulletin 80-04 concerns, including AFW runout. The maximum auxiliary feed runout flow is 450 gpm and is conservatively modeled as a constant flow.

The following base scope cases are analyzed:

1. 0%, 70%, and 100% power, Main Steam Check Valve (MSCV) failure cases with offsite power available.

2. 0% and 100% power, diesel failure cases with loss of offsite power and loss of one train of engineered safeguards.

3. 0%, 30%, 70% and 100% power, Feedwater Control Valve (FCV) failure cases and offsite power available.

4. 100% power case with Auxiliary Feed Runout (AFW) failure.

Optional Cases

An investigation and assessment of other areas of margin is made. The most limiting case of the base scope is reviewed to determine if credit for Primary or Secondary System trip signals can provide earlier automatic actuation times and improved results. Credit for Primary and Secondary trips/actuations from signals other than High 1 and High 2 is investigated for the limiting pressure case. The limiting full double-ended rupture analysis results are reviewed for consideration of enhanced containment wall heat transfer based on break size considerations. This includes turbulent Tagami wall heat transfer and revaporization of wall condensate. Although taking credit for break size dependencies would result in improved results, the models may also result in a smaller break size becoming limiting. Additionally, the impact of reduced operator action time (i.e., 10 minutes) is assessed.

MSLB OBJECTIVE

The objective of the MSLB analysis described herein is to increase available margin in peak pressure compared to a design pressure of 47 psig by limiting failures of equipment coincident with the event to a single failure. A program is outlined which will enable limiting conditions to be determined and then be used to calculate limiting results for a broad range of postulated operating and failure conditions.

The goal is to minimize the number of cases by performing the base scope work utilizing a matrix approach and considering only full double-ended breaks, conservatively calculating peak containment pressure while eliminating all break size step change dependencies and assuming only one single failure at a time.

Following the base scope work, optional work is performed to demonstrate potential additional pressure margin for the limiting peak pressure case. Conservatisms in the analysis are reduced and more appropriate values are used which show additional margin in pressure. In each case, only the value examined is modified in the sensitivity analyses. All other assumptions and values remain unchanged from the base scope case in order to provide a true sensitivity.

2.0 LONG TERM LOCA MASS AND ENERGY RELEASE ANALYSIS

2.1 INTRODUCTION

This report section presents the long term LOCA mass and energy releases that were generated in support of the Containment Margin Improvement program effort for Indian Point Unit 3. These mass and energy releases are then subsequently used in the COCO containment integrity analysis peak pressure calculation.

2.2 LOCA MASS AND ENERGY RELEASE PHASES

The LOCA transient is typically divided into four phases:

1. Blowdown - which includes the period from accident initiation (when the reactor is at steady state operation) to the time that the RCS reaches initial equilibrium with containment.

2. Refill - the period of time when the lower plenum is being filled by accumulator and safety injection water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.

3. Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-Reflood (Froth) - describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators. After the broken loop steam generator cools, the break flow becomes two phase.

2.3 BREAK SIZE AND LOCATION

Generic studies have been performed with respect to the effect on the LOCA mass and energy releases relative to postulated break size. The double ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture:

1. Hot leg (between vessel and steam generator)
2. Cold leg (between pump and vessel)
3. Pump suction (between steam generator and pump)

The break location analyzed and described herein is the double-ended pump suction guillotine break (10.48 ft^2). Pump Suction break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA. The following information provides a discussion on each break location.

The double ended hot leg guillotine has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be highest for this break location, the amount of energy released from the steam generator secondary is minimal because the majority of the fluid which exits the core bypasses the steam generators in venting to containment. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break.

For the hot leg break, there is no reflood peak as determined by generic studies (i.e., from the end of the blowdown period the releases would continually decrease). Therefore the reflood (and subsequent post-reflood) releases are not calculated for a hot leg break. The mass and energy releases for the hot leg break blowdown phase have been included in the scope of this containment integrity analysis.

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment peak pressure. The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient is, in general, less limiting than the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold leg break is not included in the scope of this report.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the Reactor Coolant System in calculating the releases to containment. This break location has been determined to be the limiting break for typical dry containment plants. The analysis of this break location for Indian Point Unit 3 as the limiting break is consistent with other dry containment plants for the post blowdown phase of the event.

In summary then, the analysis of the limiting break location for a dry containment has been performed and is shown in this report. The double-ended pump suction guillotine break has historically been considered to be the limiting break location for the post blowdown phase of the event, by virtue of its consideration of all energy sources present in the RCS. The analyses presented in this document support the conclusions of the double ended pump suction (DEPS) as the limiting break case for the post blowdown period and the double ended hot leg (DEHL) Break Case for the blowdown phase of the transient. These break locations provides a mechanism for the release of the available energy in the Reactor Coolant System, including both the broken and intact loop steam generators.

2.4 APPLICATION OF SINGLE FAILURE CRITERIA

An analysis of the effects of the single failure criteria has been performed on the mass and energy release rates for the (DEPS) break. For the DEPS results presented in this report, an inherent assumption in the generation of the mass and energy releases is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the safety injection system. This is not an issue for the blowdown period which is limited by the DEHL break.

The limiting case for Indian Point Unit 3 is the minimum safeguards case. This was determined by prior generic and specific Indian Point 3 analyses. In the case of minimum safeguards, the single failure postulated to occur is the loss of an emergency diesel generator. This results in the loss of one pumped safety injection train and the containment safeguards components on that diesel, thereby minimizing the safety injection flow. The analysis further considers the safety injection pump head curves to be degraded by 5%. This results in the greatest reduction possible for the Emergency Core Cooling System (ECCS) components.

2.5 MASS AND ENERGY RELEASE DATA

2.5.1 Blowdown Mass and Energy Release Data

The SATAN-VI code is used for computing the blowdown transient, and is the same as that used for the ECCS calculation in Reference 10. The methodology for the use of this model is described in Reference 7.

Tables 2-1 and 2-2 present the calculated mass and energy releases for the blowdown phase of the break analyzed for the DEPS and DEHL breaks, respectively.

The mass and energy releases for the double-ended pump suction break and the double-ended hot leg break, given in Table 2-1 and 2-2 terminate 24.8 and 30.3 seconds respectively after the initiation of the postulated accident.

2.5.2 Reflood Mass and Energy Release Data

The WREFLOOD code is used for computing the reflood transient, and is a modified version of that used in the ECCS calculation in Reference 10. The methodology for the use of this model is described in Reference 7.

An exception to the mass and energy evaluation model described in Reference 7 is taken, in that steam/water mixing in the broken loop has been included in this analysis. This assumption is justified and is supported by test data, and is summarized as follows:

The model assumes a complete mixing condition (i.e.,thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two phase interaction with condensation of steam by cold injection water. The second is a single phase mixing of condensate and injection water. Since the mass and energy of the steam released is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data is that generated in 1/3 scale tests (Reference 11), which are the largest scale data available and thus most closely simulates the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

From the entire series of 1/3 scale tests, a group corresponds almost directly to containment integrity reflood conditions. The injection flow rates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 7. For all of these tests, the data clearly indicates the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is therefore wholly supported by the 1/3 scale steam/water mixing data.

Additionally, the following justification is also noted. The limiting break for the containment integrity peak pressure analysis during the post-blowdown phase is the double ended pump suction break. For this break, there are two flow paths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam which is not condensed by ECC injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECC injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam which is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECC injection nozzle. A description of the test and test results is contained in References 7 and 11.

The methodology previously discussed and described in Reference 7 has been utilized and approved on the Dockets for Catawba Units 1 & 2, McGuire Units 1 & 2, Sequoyah Units 1 & 2, Watts Bar Units 1 & 2, Millstone Unit 3, Beaver Valley Unit 2 and Surry Units 1 & 2.

Table 2-3 presents the calculated mass and energy releases for the reflood phase of the Double-ended Pump Suction break, with minimum safety injection. A significantly higher discharge occurs during the period the accumulators are injecting (from 33.3 to 64.9 seconds as illustrated in Table 2-3).

The transient of the principal parameters during reflood are given in Table 2-4 for this break case.

2.5.3 Post-Reflood Mass and Energy Release Data

The FROTH code [Reference 1] is used for computing the post-reflood transient. The methodology for the use of this model is described in Reference 7. The mass and energy release rates calculated by FROTH are used in the containment analysis until the time of containment depressurization.

After depressurization, the mass and energy release from decay heat is based on the 1979 ANSI/ANS Standard, shown in Reference 12, and the following input:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

2. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Table 10 of ANS (1979).

5. Operation time before shutdown is 3 years.

6. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

7. Two sigma uncertainty (2 times the standard deviation) has been applied to the fission product decay.

Table 2-5 presents the two phase (froth) mass and energy release data.

2.6 SOURCES OF MASS AND ENERGY

The sources of mass considered in the LOCA mass and energy release analysis are given in Tables 2-6 and 2-7. These sources are the reactor coolant system, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analysis are given in Tables 2-8 and 2-9. The energy sources include:

1. Reactor Coolant System Water

2. Accumulator Water

3. Pumped Injection Water

4. Decay Heat

5. Core Stored Energy

6. Reactor Coolant System Metal

7. Steam Generator Metal

8. Steam Generator Secondary Energy

9. Secondary Transfer of Energy (feedwater into and steam out of the steam generator secondary)

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the clad temperature did not rise high enough for the rate of the Zirc-water reaction heat to be of any significance.

System parameters needed to perform confirmatory analyses are provided in Table 2-10.

The consideration of the various energy sources in the mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Thus the review guidelines presented in Standard Review Plan section 6.2.1.3 have been satisfied.

The mass and energy inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)

2. End of blowdown time

3. End of refill time

4. End of reflood time

5. Time of full depressurizations

6. End of analysis

The methods and assumptions used to release the various energy sources are given in Reference 7, except as noted in section 2.5.2, which has been approved as a valid evaluation model by the Nuclear Regulatory Commission.

2.7 SIGNIFICANT MODELING ASSUMPTIONS

The following items ensure that the mass and energy releases are conservatively calculated for maximum containment pressure:

1. Maximum expected operating temperature of the reactor coolant system

2. Allowance in temperature for instrument error and dead band (+5.0°F)

3. Margin in volume of +3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty)

4. Power level of 3025 MWt

5. Allowance for calorimetric error (+2 percent of power)

6. Conservative coefficients of heat transfer (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer)

7. Allowance in core stored energy for effect of fuel densification

8. Margin in core stored energy (+15 percent)

9. Allowance for RCS pressure uncertainty (+40 psi)

TABLE 2-1

BLOWDOWN MASS AND ENERGY RELEASES

DEPS

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
0.0000	0.0	0.0	0.0	0.0
0.0501	40202.8	21895.1	22225.7	11923.9
0.100	40241.3	21784.4	18828.3	10138.4
0.200	44039.8	24009.8	21515.7	11597.0
0.351	45176.9	25037.4	22865.9	12223.8
0.800	41983.2	24458.9	22175.1	11878.2
1.10	38567.5	23212.7	21014.1	11356.9
1.40	30992.0	19925.5	20395.1	11027.7
2.20	23233.1	15394.4	19659.2	10639.0
2.70	21081.9	14113.6	18997.7	10288.9
3.70	19347.1	12931.6	17292.5	9390.7
4.80	18153.1	11870.2	15119.6	8237.4
4.90	20672.1	13494.6	14868.1	8102.8
5.25	14520.6	10342.1	14198.7	7744.2
5.75	15631.7	10800.1	13128.6	7168.5
6.00	16134.8	10982.4	13275.3	7255.6
6.75	18621.5	12263.1	12647.6	6922.3
7.00	23621.8	15254.2	12342.8	6758.8
7.25	25185.9	15981.8	12133.7	6647.8
8.50	26929.0	16152.6	10612.5	5812.0
9.25	26111.2	15721.3	9716.8	5317.5
9.75	25002.7	15086.4	9158.2	5009.0
10.0	10511.5	6393.1	9279.9	5079.4
10.3	8045.3	5126.4	9214.6	5038.1
10.5	7385.7	4805.4	9445.7	5161.5
11.8	7774.9	4870.6	9301.4	5091.9
12.3	8922.6	5682.1	9201.1	5053.5
12.8	5540.2	4655.3	8905.2	4907.9
14.0	6095.3	4235.7	8035.5	4469.9
17.8	3547.0	3310.2	5272.9	3047.7
18.3	2892.9	3074.4	4250.0	2380.6
19.3	1963.7	2409.3	3739.2	1611.4
22.3	378.3	476.5	1540.4	828.3
22.8	197.8	249.6	832.2	284.8
24.8	0.0	0.0	0.0	0.0

TABLE 2-2

BLOWDOWN MASS AND ENERGY RELEASES

DEHL

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
0.0000	0.0	0.0	0.0	0.0
0.0503	43245.9	27061.5	26844.5	16539.9
0.100	49948.5	31020.7	25982.8	15812.3
0.250	30729.0	19594.3	20713.7	12648.7
0.500	30304.0	19266.6	18278.0	10784.3
1.10	29177.6	18764.7	16479.8	9205.5
1.40	28594.2	18585.1	16502.8	9118.6
2.40	25620.1	17080.2	17563.5	9405.1
3.30	22861.0	15360.2	17530.5	9355.6
4.20	20936.6	13941.0	16969.3	9063.1
5.25	20055.8	13001.3	15838.5	8512.7
6.00	20732.5	13095.6	14627.6	7926.4
6.25	12101.9	8638.5	14062.3	7642.4
6.50	14179.9	10298.8	13582.8	7404.9
7.75	16369.5	11177.9	11474.8	6329.9
8.00	18271.8	12265.9	11104.8	6137.0
8.25	27970.3	17685.6	10755.8	5954.0
8.50	25635.3	16005.2	10381.7	5755.7
9.00	24417.9	16003.3	9495.8	5283.3
10.0	26227.9	15663.1	7802.3	4398.7
11.0	24613.4	14584.1	6341.6	3854.3
11.3	15515.6	9005.1	6030.2	3499.2
11.5	15221.5	8819.2	5738.0	3354.9
11.8	9968.5	6719.2	6499.1	3240.9
12.0	9003.4	6545.2	5318.7	3154.9
13.8	10376.9	6886.2	4500.9	2760.1
14.3	9667.9	6586.1	4184.4	2575.7
14.8	7344.7	5390.6	3685.0	2340.6
16.0	5950.6	5094.2	2815.5	1868.6
16.5	4236.7	4036.4	2620.6	1745.9
19.5	1412.6	1669.7	1422.1	1236.1
21.0	805.5	1025.2	1092.3	1028.3
23.8	247.9	323.6	192.7	236.6
29.3	440.7	553.3	197.2	246.8
30.3	0.0	0.0	0.0	0.0

TABLE 2-3

REFLOOD MASS AND ENERGY RELEASES

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
24.8	0.0	0.0	0.0	0.0
25.2	0.0	0.0	205.4	18.0
25.7	0.0	0.0	205.4	18.0
25.9	39.0	46.0	205.4	18.0
26.1	12.1	14.3	205.4	18.0
27.0	53.6	63.2	205.4	18.0
30.8	130.0	153.3	205.4	18.0
32.8	155.7	183.6	205.4	18.0
33.8	246.2	290.6	2100.6	321.8
34.9	375.3	444.[illegible]	3727.4	819.6
35.9	378.6	448.0	3759.1	833.6
39.9	356.0	421.1	3522.7	803.8
41.9	345.4	408.5	3409.3	589.0
45.9	326.6	388.1	3201.2	561.5
49.9	310.2	366.6	3015.4	536.9
53.9	295.8	349.6	2848.4	514.7
57.9	283.1	334.5	2697.0	494.5
58.9	280.2	330.9	2661.2	489.7
60.9	274.5	324.2	2592.0	480.4
64.9	263.9	311.6	2461.7	462.7
65.9	362.8	429.0	341.3	188.7
66.9	424.3	502.5	384.9	224.4
69.9	411.2	486.6	359.2	218.5
87.9	340.4	402.5	329.9	175.6
95.9	319.0	377.1	321.5	163.9
97.9	314.5	371.7	319.7	161.5
107.9	296.3	350.1	312.6	151.8
115.9	287.6	339.8	309.8	147.3
121.9	285.0	336.6	312.9	146.2
129.9	285.0	336.7	324.8	147.1
137.9	285.5	337.3	341.6	148.9
145.9	284.3	335.9	360.6	150.4
153.9	280.6	331.5	381.2	151.5
169.9	264.7	312.6	426.0	152.4
171.5	262.5	310.0	430.8	152.5

TABLE 2-4

REFLOOD PRINCIPAL PARAMETERS

TIME	FLOODING TEMP	FLOODING RATE	CARRYOVER FRACTION	CORE HEIGHT	DOWNCOMER HEIGHT	FLOW FRACTION	INJECTION TOTAL	INJECTION ACCUMULATOR	INJECTION SPILL	ENTHALPY
SECONDS	DEGREE F	IN/SEC		FT	FT		(POUNDS MASS PER SECOND)			BTU/LBM
24.8	227.4	0.000	0.000	0.00	0.00	0.250	0.0	0.0	0.0	0.00
25.4	224.9	20.016	0.000	0.50	0.93	0.000	6482.5	5640.7	0.0	98.76
25.6	223.4	22.616	0.000	0.87	0.97	0.000	6419.5	5597.7	0.0	98.75
25.7	222.6	22.482	0.000	1.05	0.98	0.000	6398.3	5576.5	0.0	98.74
26.1	221.9	2.185	0.101	1.31	1.52	0.215	6300.4	5478.6	0.0	98.69
26.3	221.8	2.857	0.134	1.35	1.93	0.251	6260.4	5438.6	0.0	98.68
27.3	222.2	2.387	0.303	1.50	3.88	0.333	6071.0	5249.2	0.0	98.59
27.8	222.4	2.317	0.369	1.57	4.92	0.342	5973.4	5151.7	0.0	98.54
32.3	224.1	2.608	0.622	2.00	13.40	0.360	5306.5	4484.7	0.0	96.17
34.9	224.9	3.878	0.682	2.23	16.06	0.538	4579.6	3810.8	0.0	95.89
36.9	225.5	3.817	0.705	2.43	16.07	0.537	4367.8	3599.3	0.0	95.72
37.7	225.8	3.749	0.711	2.50	16.07	0.535	4302.2	3532.3	0.0	95.65
43.8	228.4	3.400	0.734	3.00	16.07	0.522	3874.6	3094.6	0.0	95.17
50.9	232.2	3.149	0.744	3.50	16.07	0.507	3482.8	2694.8	0.0	94.83
58.6	236.7	2.952	0.748	4.00	16.07	0.493	3138.8	2344.3	0.0	94.06
64.9	240.5	2.822	0.751	4.38	16.07	0.483	2899.8	2101.2	0.0	93.58
65.9	241.2	3.445	0.755	4.44	16.07	0.566	777.0	0.0	0.0	78.00
66.9	241.9	3.759	0.755	4.52	15.98	0.580	748.1	0.0	0.0	78.00
73.5	246.9	3.513	0.758	5.00	15.47	0.575	758.1	0.0	0.0	78.00
80.9	253.0	3.267	0.760	5.51	15.10	0.568	768.1	0.0	0.0	78.00
88.8	259.2	3.055	0.763	6.00	14.89	0.562	776.1	0.0	0.0	78.00
97.9	265.1	2.869	0.767	6.53	14.85	0.555	782.6	0.0	0.0	78.00
106.6	269.8	2.738	0.770	7.00	14.96	0.549	786.7	0.0	0.0	78.00
117.9	274.9	2.634	0.774	7.58	15.23	0.545	789.5	0.0	0.0	78.00
126.7	278.3	2.599	0.778	8.00	15.48	0.546	789.8	0.0	0.0	78.00
137.9	282.0	2.569	0.782	8.54	15.75	0.552	789.4	0.0	0.0	78.00
148.1	284.6	2.521	0.785	9.00	15.91	0.556	789.5	0.0	0.0	78.00
159.9	287.7	2.431	0.789	9.52	16.01	0.558	791.0	0.0	0.0	78.00
171.5	290.0	2.308	0.793	10.00	16.06	0.557	793.8	0.0	0.0	78.00

TABLE 2-5

POST-REFLOOD MASS AND ENERGY RELEASES

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
171.6	186.1	228.3	635.6	174.1
176.6	184.4	226.2	637.3	174.2
186.6	181.0	221.9	640.8	174.4
201.6	175.5	215.2	646.3	174.9
221.6	168.0	206.0	653.8	175.5
231.6	163.9	201.0	657.9	176.0
236.6	162.1	198.8	659.6	176.1
256.6	153.4	188.2	668.3	177.1
261.6	150.9	185.1	670.8	177.4
266.6	149.3	183.1	672.5	177.5
281.6	148.8	182.5	673.0	178.6
286.6	162.6	199.5	659.1	178.3
291.6	160.3	196.5	661.5	178.6
311.6	150.7	184.8	671.0	177.7
316.6	148.5	182.1	673.3	177.9
341.6	146.9	180.2	674.8	176.5
346.6	147.0	180.3	674.7	176.1
371.6	146.0	179.0	675.8	174.5
376.6	146.0	179.1	675.7	174.1
401.6	145.0	177.8	676.7	172.8
411.6	145.0	177.8	676.8	171.7
426.6	144.5	177.2	677.2	174.1
771.6	144.5	177.2	677.2	174.1
771.7	81.4	99.0	740.4	180.8
801.6	80.7	98.2	741.1	181.1
876.6	79.3	96.5	742.5	180.0
881.6	79.2	96.3	742.6	178.7
986.6	77.6	94.4	744.2	177.4
1081.9	77.6	94.4	744.2	177.4
1082.0	74.8	85.8	746.9	86.9
3600.0	55.9	64.2	105.1	32.3
3600.1	46.0	52.8	115.0	18.8
10000.0	33.5	38.4	127.6	20.6
100000.0	17.9	20.5	143.1	23.3
1000000.0	7.7	8.8	153.4	25.0

TABLE 2-6

MASS BALANCE - DEPS

	TIME (SECONDS)	0.00	24.75	24.75	171.53	776.80	1061.84	3600.00
				MASS (THOUSAND LBM)				
INITIAL	IN RCS AND ACC	729.77	729.77	729.77	729.77	729.77	729.77	729.77
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00	115.27	612.43	790.44	1195.96
	TOTAL ADDED	0.00	0.00	0.00	115.27	612.43	790.44	1195.96
*** TOTAL AVAILABLE ***		729.77	729.77	729.77	845.04	1342.20	1520.21	1925.73
DISTRIBUTION	REACTOR COOLANT	528.72	44.01	66.24	138.54	138.54	138.54	138.54
	ACCUMULATOR	201.05	156.46	134.23	0.00	0.00	0.00	0.00
	TOTAL CONTENTS	729.77	200.47	200.47	138.54	138.54	138.54	138.54
EFFLUENT	BREAK FLOW	0.00	529.30	529.30	706.50	1203.66	1381.67	1787.20
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	529.30	529.30	706.50	1203.66	1381.67	1787.20
*** TOTAL ACCOUNTABLE ***		729.77	729.77	729.77	845.04	1342.20	1520.21	1925.73

TABLE 2-7

MASS BALANCE - DEHL

	TIME (SECONDS)	0.00	30.25	30.25
	MASS (THOUSAND LBM)			
INITIAL	IN RCS AND ACC	729.77	729.77	729.77
ADDED MASS	PUMPED INJECTION	0.00	0.00	0.00
	TOTAL ADDED	0.00	0.00	0.00
*** TOTAL	AVAILABLE ***	729.77	729.77	729.77
DISTRIBUTION	REACTOR COOLANT	528.72	94.64	116.88
	ACCUMULATOR	201.05	120.27	98.04
	TOTAL CONTENTS	729.77	214.92	214.92
EFFLUENT	BREAK FLOW	0.00	514.85	514.85
	ECCS SPILL	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	514.85	514.85
*** TOTAL	ACCOUNTABLE ***	729.77	729.77	729.77

TABLE 2-8

ENERGY BALANCE - DEPS

	TIME (SECONDS)	0.00	24.75	24.75	171.53	778.80	1081.94	3300.00
				ENERGY (MILLION BTU)				
INITIAL ENERGY	IN RCS,ACC,S GEN	817.10	817.10	817.10	817.10	817.10	817.10	817.10
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00	8.99	47.77	63.17	129.31
	DECAY HEAT	0.00	7.77	7.77	23.53	70.96	91.19	222.33
	HEAT FROM SECONDAR	0.00	-1.78	-1.78	-1.78	2.63	4.05	4.05
	TOTAL ADDED	0.00	5.99	5.99	30.74	121.37	158.41	355.69
*** TOTAL	AVAILABLE ***	817.10	823.09	823.09	847.84	938.48	975.51	1172.79
DISTRIBUTION	REACTOR COOLANT	306.70	12.45	14.66	37.84	37.84	37.86	37.84
	ACCUMULATOR	20.01	15.57	13.36	0.00	0.00	0.00	0.00
	CORE STORED	23.53	13.17	13.17	3.95	3.78	3.64	2.56
	PRIMARY METAL	169.45	160.98	160.98	129.45	86.03	75.21	54.25
	SECONDARY METAL	78.06	77.76	77.76	68.99	48.65	40.49	29.58
	STEAM GENERATOR	219.34	223.54	223.54	194.36	136.10	113.93	84.12
	TOTAL CONTENTS	817.10	503.47	503.47	434.59	312.80	271.12	208.36
EFFLUENT	BREAK FLOW	0.00	319.62	319.62	405.66	618.28	696.79	956.84
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	319.62	319.62	405.66	618.28	696.79	956.84
*** TOTAL	ACCOUNTABLE ***	817.10	823.10	823.10	840.25	930.87	967.92	1165.20

TABLE 2-9

ENERGY BALANCE - DEHL

	TIME (SECONDS)	0.00	30.25	30.25
	ENERGY (MILLION BTU)			
INITIAL ENERGY	IN RCS,ACC,S GEN	817.10	817.10	817.10
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00
	DECAY HEAT	0.00	8.98	8.98
	HEAT FROM SECONDAR	0.00	-3.67	-3.67
	TOTAL ADDED	0.00	5.31	5.31
*** TOTAL	AVAILABLE ***	817.10	822.41	822.41
DISTRIBUTION	REACTOR COOLANT	308.70	22.58	24.80
	ACCUMULATOR	20.01	11.97	9.75
	CORE STORED	23.53	9.76	9.76
	PRIMARY METAL	169.45	157.83	157.83
	SECONDARY METAL	76.06	75.83	75.83
	STEAM GENERATOR	219.34	216.80	216.80
	TOTAL CONTENTS	817.10	494.78	494.78
EFFLUENT	BREAK FLOW	0.00	327.63	327.63
	ECCS SPILL	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	327.63	327.63
*** TOTAL	ACCOUNTABLE ***	817.10	822.40	822 40

TABLE 2-10

SYSTEM PARAMETERS
FOR 3025 MWt

PARAMETER	VALUE
Core Inlet Temperature (includes +5.0°F Allowance for Instrument Error and Deadband)	547.6 °F
Initial Steam Generator Steam Pressure	817.0 psia
Assumed Maximum Containment Back Pressure	61.7 psia

3.0 MSLB MASS AND ENERGY RELEASE ANALYSIS

3.1 INTRODUCTION

As a part of the Indian Point 3 Containment Margin Improvement Program, the impact of the Mass and Energy Releases due to the rupture of a main steam line on containment Pressure Response is analyzed. The analyses determine if containment pressure will remain below the design limit of 47 psig. Conservative assumptions are made to limit the number of cases needed to identify peak containment pressures. Optional cases are run to determine the sensitivity of the limiting case to some of the conservative assumptions made.

3.2 DISCUSSION

The LOFTRAN computer code, (Reference 9), is used to generate the mass and energy release to the containment for a large double-ended guillotine break. Analysis is done for four plant power levels: 102%, 70%, 30% and 0% power. In addition, a spectrum of single failures is imposed on each postulated break scenario. The single failures that are assumed are: 1) failure of the MSIV and check valve in the faulted loop; 2) failure of the feedwater regulating valve in the faulted loop; and 3) failure of auxiliary feedwater pump runout protection to the faulted loop. The methodology employed is consistent with Reference 16.

The mass and energy releases to containment are input to the COCO computer code, (Reference 2), which determines the containment pressure and temperature response. Important containment design and operating conditions used in the analysis are discussed in Section 4.0.

3.3 SIGNIFICANT MODELING ASSUMPTIONS

Section 1.5 gives many of the major analytical assumptions used in this analysis. Additional details are provided below on the trip signals available and on other important considerations in the analyses.

For the steamline break analysis, the following safety features were taken credit for: reactor trip, steamline isolation, feedwater isolation and safety injection. These events reduce or terminate the mass and energy releases to containment. The signals that normally provide protection are:

1. Reactor Trip

   High 1 Containment Pressure (SI)
   Overpower Delta T
   High Differential Pressure (SI)
   High Steam Flow coincident with Low Steam Pressure (SI)

2. Feedwater Isolation

   Due to Safety Injection Signal

3. Safety Injection

   High 1 Containment Pressure
   Low Pressurizer Pressure
   High Differential Pressure
   High Steam Flow coincident with Low Steam Pressure

4. Steamline Isolation

   High 2 Containment Pressure
   High Steam Flow coincident with Low Steam Pressure

In this analysis to avoid any break dependency, only High-1 and High-2 were credited. The setpoints which were assumed in the analyses are as follows:

1. High-1 Containment Pressure - 5.12 psig

2. High-2 Containment Pressure - 24.6 psig

The BIT water volume used is 120 ft^3, and no boron is in the BIT. The delay time for delivering the 2300 ppm borated RWST water to the primary system includes an 18 second delay. This is the delay between when the process parameters being measured reach the setpoint to the time when the charging pumps are at full flow. The 18 second delay is composed of a 6 second pure delay and a 12 second linear ramp to full flow. In addition to this delay, there is an additional delay in delivering borated water to the RCS since there is water which must be purged between the RWST and the primary system.

The main steamline isolation valve closure time used is 12 seconds. The 12 second time allows for a typical 2 second electronic delay and a 10 second valve stroke time. The overall time of 12 seconds and not the breakdown is all that is important to the analysis.

The analyses assume a 10000 ft^3 dead volume to blow down between the intact steam generator isolation valves and the break. This assumption applies when the isolation valves shut and the check valve on the faulted steam generator has been assumed to fail.

A dead volume of 800 ft^3 is assumed to be available for feedwater flashing between the feed valves and faulted steam generator feedwater line inlet.

3.4 MASS AND ENERGY RELEASE DATA

The MSLB mass and energy releases are shown in Tables 3-1 thru 3-10 for the following double-ended breaks:

1. 0% power MSCV failure with offsite power available

2. 70% power MSCV failure with offsite power available

3. 100% power MSCV failure with offsite power available

4. 0% power diesel failure (one engineering safeguards train assumed lost, RCP's conservatively assumed to continue to operate)

5 100% power diesel failure (one engineering safeguards train assumed lost, RCP's conservatively assumed to continue to operate)

6. 0% power Feedwater Control Valve failure (FCV) with offsite power available

7. 30% power Feedwater Control Valve failure (FCV) with offsite power available

8. 70% power Feedwater Control Valve failure (FCV) with offsite power available

9. 100 % power Feedwater Control Valve failure (FCV) with offsite power available

10. 100 % power Auxiliary Feed Runout failure (AFW) with offsite power available

As seen from some of the tables (i.e., Table 3-4), minor flow perturbations may occur during certain time periods of the transient. These variations have been determined to have an insignificant affect on the results. For example, between 234.0 and 2098.0 seconds in Table 3-4 there exists a minor flow oscillation. This is attributed to a minor flow fluctuation during steam generator dryout.

As discussed in Section 1.5, an assessment of other areas of pressure margin was made. The most limiting case of the base scope containment response work, as discussed in Section 5.1, was reviewed to determine if such things as enhanced wall heat transfer, an increased FCV response time, credit for break size dependent primary/secondary trips, etc., would demonstrate additional avenues of pressure margin. The containment response illustrating the sensitivity to these optional items is discussed in Section 5.1. The MSLB mass and energy releases for these additional cases are shown in Tables 3-11 thru 3-15 for the following cases:

1. 70% power Feedwater Control Valve Failure (FCV) with offsite power available assuming a 30 second FIV closure time as opposed to 122 seconds.

2. 70% power Feedwater Control Valve Failure (FCV) with offsite power available assuming a reduced feed flashing volume of 260 cubic feet (between the SG inlet nozzle and the FIV) as opposed to 800 cubic feet.

3. 70% power Feedwater Control Valve Failure (FCV) with offsite power available assuming the BIT has 20000 ppm boron as opposed to 0 ppm.

4. 70% power Feedwater Control Valve Failure (FCV) with offsite power available assuming a minimum, more realistic SI time delay as opposed to a conservatively long delay.

5. 70% power Feedwater Control Valve Failure (FCV) with offsite power available assuming a break size of 0.6 square feet as opposed to a full double ended rupture. The 0.6 square foot break is chosen to approximate the largest break size for which entrainment, revaporization, and Tagami could not be credited at the 70% power level.

In addition, an assessment was made to determine if crediting primary or secondary protection setpoints other than containment High-1 and High-2 would provide some margin. The limiting pressure 70% FCV failure case and the 100% power MSCV failure case were examined. It was determined that none of the other protection signals would have provided reactor trip, SI, or steamline isolation soon enough to alter the results.

Finally, an assessment was made to determine if a 10 minute operator action time versus 30 minutes would affect the pressure margin. In all of the base scope cases the peak pressure is reached prior to 10 minutes. Therefore, 10 minute operatc action makes no difference.

TABLE 3-1

MSLB MASS AND ENERGY RELEASES

0% POWER MSCV FAILURE

WITH OFFSIT POWER

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-1 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-1 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-2
MSLB MASS AND ENERGY RELEASES
70% POWER MSCV FAILURE
WITH OFFSITE POWER

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-2 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-2 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-3

MSLB MASS AND ENERGY RELEASES

100% POWER MSCV FAILURE

WITH OFFSITE POWER

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (LBM/SEC)

a,c

TABLE 3-3 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (LBM/SEC)

a,c

TABLE 3-3 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (LBM/SEC)

a,c

TABLE 3-3 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (LBM/SEC)

a,c

TABLE 3-4

MSLB MASS AND ENERGY RELEASES

0% POWER DIESEL FAILURE

WITHOUT OFFSITE POWER

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-4 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-4 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-5

MSLB MASS AND ENERGY RELEASES

100% POWER DIESEL FAILURE

WITHOUT OFFSITE POWER

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-5 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-5 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-6

MSLB MASS AND ENERGY RELEASES

0% POWER FCV FAILURE

WITH OFFSITE POWER

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-6 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a

TABLE 3-6 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-6 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-7

MSLB MASS AND ENERGY RELEASES

30% POWER FCV FAILURE

WITH OFFSITE POWER

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-7 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-7 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-7 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-7 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-7 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-8

MSLB MASS AND ENERGY RELEASES

70% POWER FCV FAILURE

WITH OFFSITE POWER

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-8 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-8 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-8 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-9

MSLB MASS AND ENERGY RELEASES

100% POWER FCV FAILURE

WITH OFFSITE POWER

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-9 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-9 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a, c

TABLE 3-9 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-10

MSLB MASS AND ENERGY RELEASES

100% POWER AFW FAILURE

WITH OFFSITE POWER

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-10 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-10 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-11

MSLB MASS & ENERGY RELEASES

70% POWER FCV FAILURE

WITH OFFSITE POWER

30 SECOND FIV CLOSURE

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-11 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-11 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-11 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-11 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-11 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-11 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-12

MSLB MASS & ENERGY RELEASES

70% POWER FCV FAILURE

WITH OFFSITE POWER

REDUCED FEED FLASHING VOLUME

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-12 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-12 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-12 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-13

MSLB MASS & ENERGY RELEASES

70% POWER FCV FAILURE

WITH OFFSITE POWER

BIT IN

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-13 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-13 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-13 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-13 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-14

MSLB MASS & ENERGY RELEASES

70% POWER FCV FAILURE

WITH OFFSITE POWER

MINIMUM SI TIME DELAY

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-14 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-14 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-14 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-14 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-15

MSLB MASS & ENERGY RELEASES

70% POWER FCV FAILURE

WITH OFFSITE POWER

0.60 SQUARE FOOT BREAK

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-15 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-15 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-15 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-15 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

TABLE 3-15 (Cont.)

TIME (SEC)	BREAK FLOW (LBM/SEC)	BREAK ENERGY (BTU/SEC)

a,c

4.0 LOCA CONTAINMENT INTEGRITY (PEAK PRESSURE) ANALYSIS

4.1 DESCRIPTION OF COCO MODEL

Calculation of containment pressure and temperature transients is accomplished by use of the digital computer code, COCO (Reference [2]). The COCO code has been used and found acceptable to calculate containment pressure transients for many dry containment plants. Transient phenomena within the reactor coolant system affect containment conditions by means of convective mass and energy transport through the pipe break.

For analytical rigor and convenience, the containment air-steam-water mixture is separated into systems. The first system consists of the air-steam phase; the second consists of the water phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate boundary conditions. As thermodynamic equations of state and conditions may vary during the transient, the equations have been derived for all possible cases of superheated or saturated steam and subcooled or saturated water. Switching between states is handled automatically by the code. The following are the major assumptions made in the analysis:

(a) Discharge mass and energy flow rates through the reactor coolant system break are established from the analysis in Section 2.

(b) For the blowdown portion of the LOCA analysis, the discharge flow separates into steam and water phases at the break point. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment. For the post-blowdown portion of the LOCA analysis, steam and water releases are input separately.

(c) Homogeneous mixing is assumed. The steam-air mixture and the water phase each have uniform properties. More specifically, thermal equilibrium between the air and steam is assumed. This does not imply thermal equilibrium between the steam-air mixture and water phase.

(d) Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.

(e) The saturation temperature at the partial pressure of the steam is used for heat transfer to the heat sinks and the fan coolers.

4.2 CONTAINMENT PRESSURE CALCULATION

The following are the major input assumptions used in the COCO analysis for the pump suction pipe rupture case with the steam generators considered as an active heat source for the Indian Point Unit 3 Nuclear Plant Containment:

1. Minimum safeguards are employed in all calculations, e.g., one of two spray pumps and three of five fan coolers. Additionally used are two of two RHR pumps, one of two recirculation pumps and one of two RHR heat exchangers providing flow to the core; and two high head safety injection pumps.

2. The blowdown, reflood, and post reflood mass and energy releases described in Section 2 are used.

3. Essential service water temperature of 95°F is used for the component cooling heat exchanger.

4. The initial conditions in the containment are a temperature of 130°F and a pressure of 2.5 psig.

5. Containment structural heat sinks are assumed with conservatively low heat transfer rates. (See Tables 4-1,4-2)

6. The operation of one RHR heat exchanger (UA = 6.1×10^5 Btu/hr-°F) was assumed for core cooling during recirculation. The component cooling heat exchanger was modeled at $1.08 * 10^6$ Btu/hr-°F per heat exchanger.

7. The essential service water flow to the component cooling heat exchanger was modeled as 7221 gpm, which includes a degradation of 5% for conservatism. The service water flow basis is discussed in Reference 18.

4.3 HEAT REMOVAL SYSTEMS

The significant heat removal source during the early portion of the transient is the structural heat sinks. Provision is made in the containment pressure transient analysis for heat transfer through, and heat storage in, both interior and exterior walls. Every wall is divided into a large number of nodes. For each node, a conservation of energy equation expressed in finite-difference form accounts for transient conduction into and out of the node and temperature rise of the node. Tables 4-1 and 4-2 are summaries of the containment structural heat sinks used in the analysis.

The heat transfer coefficient to the containment structure is calculated by the code based primarily on the work of Tagami (Reference [13]). From this work, it was determined that the value of the heat transfer coefficient increases parabolically to a peak value at the end of blowdown for LOCA. The value then decreases exponentially to a stagnant heat transfer coefficient which is a function of steam-to-air-weight ratio.

Tagami presents a plot of the maximum value of h as a function of "coolant energy transfer speed," defined as follows:

$$\frac{\text{total coolant energy transferred into containment}}{\text{(containment volume) (time interval to peak pressure)}}$$

From this, the maximum h of steel is calculated:

$$h_{max} = 75 \left(\frac{E}{t_p V}\right)^{0.60} \qquad (4.3\text{-}1)$$

where:

h_{max} = maximum value of h (Btu/hr ft^2 °F).

t_p = time from start of accident to end of blowdown for LOCA and steam line isolation for secondary breaks (sec).

V = containment volume (ft^3).

E = coolant energy discharge (Btu).

The parabolic increase to the peak value is given by:

$$h_s = h_{max} \left(\frac{t}{t_p}\right)^{0.5}, \; 0 \leq t \leq t_p \qquad (4.3\text{-}2)$$

where:

h_s = heat transfer coefficient for steel (Btu/hr ft^2 °F).

t = time from start of accident (sec).

For concrete, the heat transfer coefficient is taken as 40 percent of the value calculated for steel (Reference 19).

The exponential decrease of the heat transfer coefficient is given by:

$$h_s = h_{stag} + (h_{max} - h_{stag})\, e^{-0.05\,(t-t_p)} \qquad t > t_p \qquad (4.3\text{-}3)$$

where:

h_{stag} = $2 + 50X,\ 0 \leq X \leq 1.4.$

h_{stag} = h for stagnant conditions (Btu/hr ft^2 °F).

X = steam-to-air weight ratio in containment.

For a large break, the engineered safety features are quickly brought into operation. Because of the brief period of time required to depressurize the reactor coolant system, the containment safeguards are not a major influence on the blowdown peak pressure; however, they reduce the containment pressure after the blowdown and maintain a low long-term pressure. Also, although the containment structure is not a very effective heat sink during the initial reactor coolant system blowdown, it still contributes significantly as a form of heat removal.

During the injection phase of post-accident operation, the emergency core cooling system pumps water from the refueling water storage tank into the reactor vessel. Since this water enters the vessel at refueling water storage tank temperature, which is less than the temperature of the water in the vessel, it can absorb heat from the core until saturation temperature is reached. During the recirculation phase of operation, water is taken from the containment sump and cooled in the residual heat removal heat exchanger. The cooled water is then pumped back to the reactor vessel to absorb more decay heat. The heat is removed from the residual heat exchanger by component cooling water.

Another containment heat removal system is the containment spray. Containment spray is used for rapid pressure reduction. During the injection phase of operation, the containment spray pumps draw water from the RWST and spray it into the containment through nozzles mounted high above the operating deck. As the spray droplets fall, they absorb heat from the containment atmosphere. Since the water comes from the RWST, the entire heat capacity of the spray from the RWST temperature to the temperature of the containment atmosphere is available for energy absorption. During the recirculation phase of post-accident operation, water can be drawn from the residual heat removal heat exchanger outlet and sprayed into the containment atmosphere via the recirculation spray system.

When a spray drop enters the hot, saturated, steam-air containment environment following a loss-of-coolant accident, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the drop. This mass flow will carry energy to the drop. Simultaneously, the temperature difference between the atmosphere and the drop will cause the drop temperature and vapor pressure to rise. The vapor pressure of the drop will eventually become equal to the partial pressure of the steam, and the condensation will cease. The temperature of the drop will essentially equal the temperature of the steam-air mixture.

The equations describing the temperature rise of a falling drop are as follows:

$$\frac{d}{dt}(Mu) = mh_g + q \tag{4.3-4}$$

$$\frac{d}{dt}(M) = m \tag{4.3-5}$$

where:

$q = h_c A (T_s - T).$

$m = K_g A (P_s - P_v).$

The coefficients of heat transfer (h_c) and mass transfer (k_g) are calculated from the Nusselt number for heat transfer, Nu, and the Nusselt number for mass transfer, Nu'.

Both $\underline{Nu}$ and $\underline{Nu'}$ may be calculated from the equations of Ranz and Marshall (Reference [14]).

$$Nu = 2 + 0.6 \, \underline{(Re)}^{1/2} \, \underline{(Pr)}^{1/3} \qquad (4.3\text{-}6)$$

$$Nu' = 2 + 0.6 \, \underline{(Re)}^{1/2} \, \underline{(Sc)}^{1/3} \qquad (4.3\text{-}7)$$

Thus, Equations 4.3-4 and 4.3-5 can be integrated numerically to find the internal energy and mass of the drop as a function of time as it falls through the atmosphere. Analysis shows that the temperature of the (mass) mean drop produced by the spray nozzles rises to a value within 99 percent of the bulk containment temperature in less than 2 seconds.

Drops of this size will reach temperature equilibrium with the steam-air containment atmosphere after falling through less than half the available spray fall height.

Detailed calculations of the heatup of spray drops in post-accident containment atmospheres by Parsly (Reference [15]) show that drops of all sizes encountered in the containment spray reach equilibrium in a fraction of their residence time in a typical pressurized water reactor containment.

These results confirm the assumption that the containment spray will be 100 percent effective in removing heat from the atmosphere. Nomenclature used in this section is as follows:

<u>Nomenclature</u>

A = area.

h_c = coefficient of heat transfer.

k_g = coefficient of mass transfer.

h_g = steam enthalpy.

M = droplet mass.

m = diffusion rate.

$\underline{Nu}$ = Nusselt number for heat transfer.

$\underline{Nu'}$ = Nusselt number for mass transfer,

P_s = steam partial pressure.

P_v = droplet vapor pressure.

$\underline{Pr}$ = Prandtl number.

q = heat flow rate.

$\underline{Re}$ = Reynolds number.

$\underline{Sc}$ = Schmidt number.

T_s = droplet temperature.

T = steam temperature.

t = time.

u = internal energy.

The reactor containment fan coolers are a final means of heat removal. The main aspect of a fan cooler from the heat removal standpoint are the fan and the banks of cooling coils. The fans draw the dense atmosphere through banks of finned cooling coils and mix the cooled steam/air mixture with the rest of the containment atmosphere. The coils are kept at a low temperature by a constant flow of cooling water. Since this system does not use water from the RWST, the mode of operation remains the same both before and after the spray system and emergency core cooling system change to the recirculation mode.

With these assumptions, the heat removal capability of the containment is sufficient to absorb the energy releases and still keep the maximum calculated pressure below the design pressure.

4.4 ANALYSIS RESULTS

The results of the analysis show that the maximum calculated containment pressure for the double-ended pump suction minimum safeguards break case is 39.8 psig and is 40.3 psig for the double-ended hot leg break case. The pressure peaks occur at approximately 799.0 seconds and 22.4 seconds, respectively.

The following plots show the containment integrity transient, as calculated by the COCO code.

Figure 4-1, Containment Pressure Transient - DEPS
Figure 4-2, Containment Temperature Transient - DEPS
Figure 4-3, Containment Pressure Transient - DEHL
Figure 4-4, Containment Temperature Transient - DEHL

Tables 4-1 and 4-2 show the containment structural heat sink and material properties data used in the analysis, respectively.

The accident Chronology for the double-ended pump suction loss-of-coolant accident is shown in Table 4-3.

4.5 RELEVANT ACCEPTANCE CRITERIA

The LOCA mass and energy analysis has been performed in accordance with the criteria shown in the Standard Review Plan (SRP) section 6.2.1.3. In this analysis, the relevant requirements of General Design Criteria (GDC) 50 and 10 CFR Part 50 Appendix K have been met since the calculated pressure is less than the design pressure, and because all available sources of energy have been included. The sources considered include: reactor power, decay heat, core stored energy, energy stored in the reactor vessel and internals, metal water reaction energy, and stored energy in the secondary system.

The containment integrity peak pressure analysis has been performed in accordance with the criteria shown in the SRP section 6.2.1.1.A, for dry PWR containments. Conformance to GDC's 16, 38, and 50 is demonstrated by showing that the containment design pressure is not exceeded at any time in the transient. This analysis also demonstrates that the containment heat removal systems function to rapidly reduce the containment pressure and temperature in the event of a LOCA.

FIGURE 4-1

CONTAINMENT PRESSURE vs. TIME

DEPS

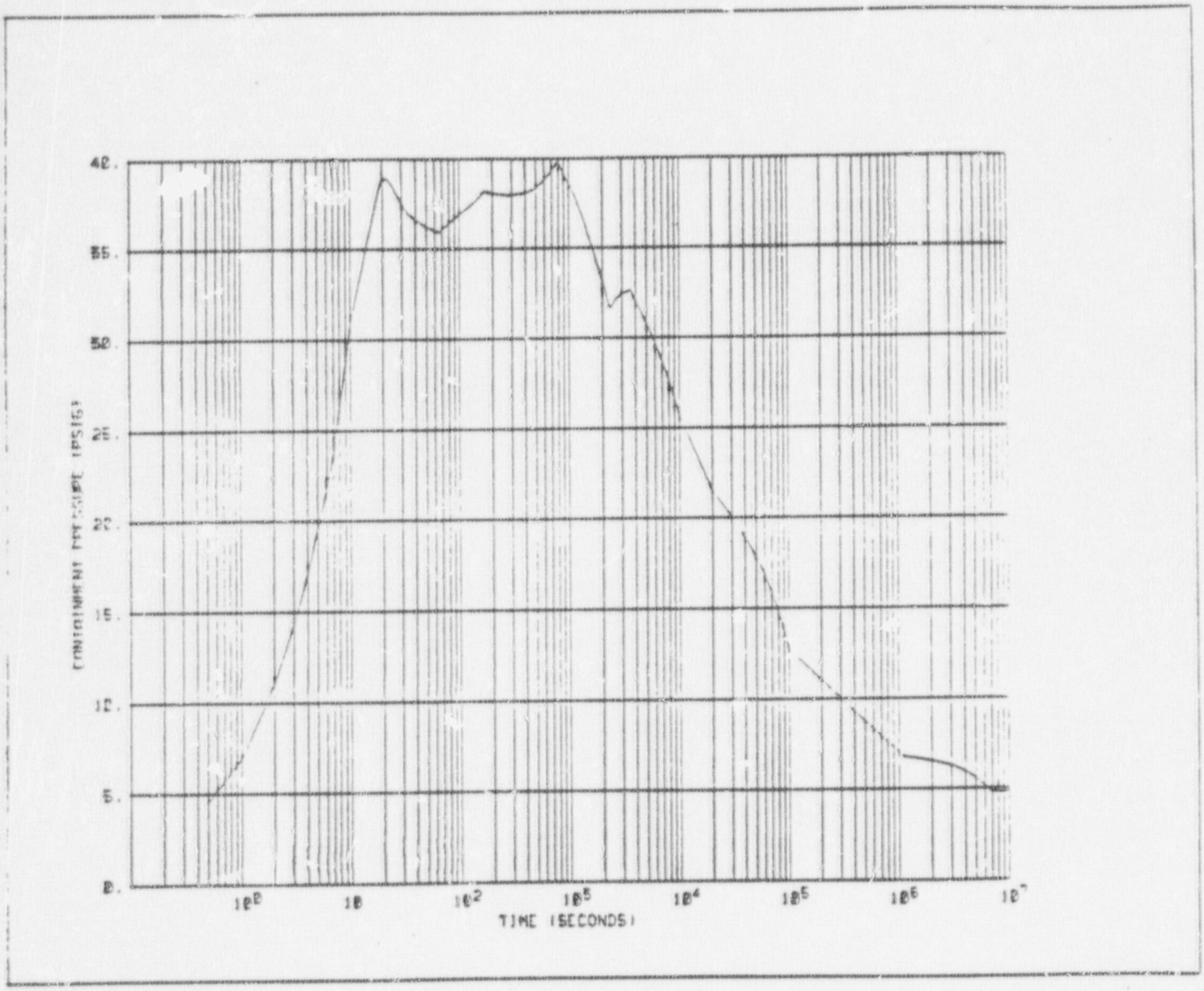

FIGURE 4-2

CONTAINMENT TEMPERATURE vs. TIME

DEPS

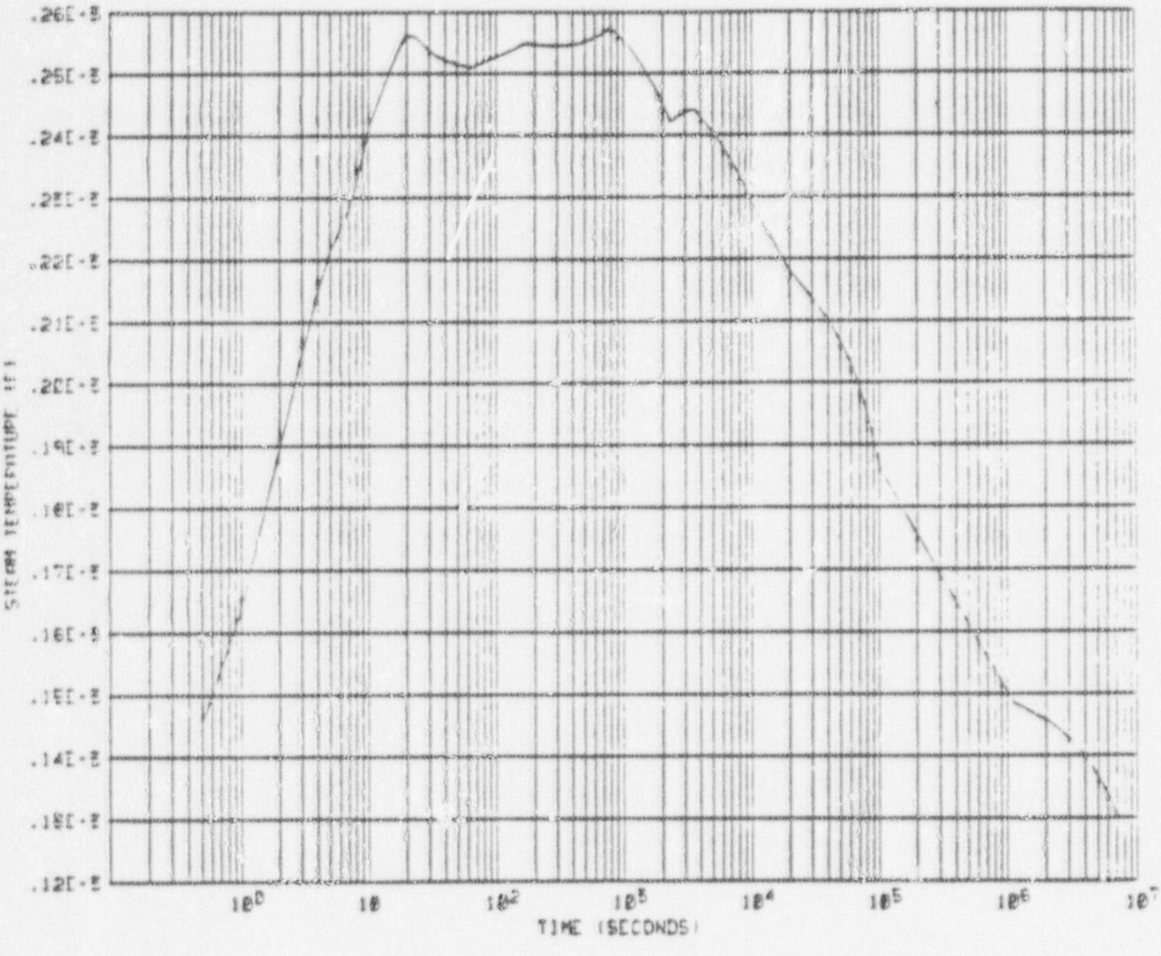

FIGURE 4-3
CONTAINMENT PRESSURE vs. TIME
DEHL

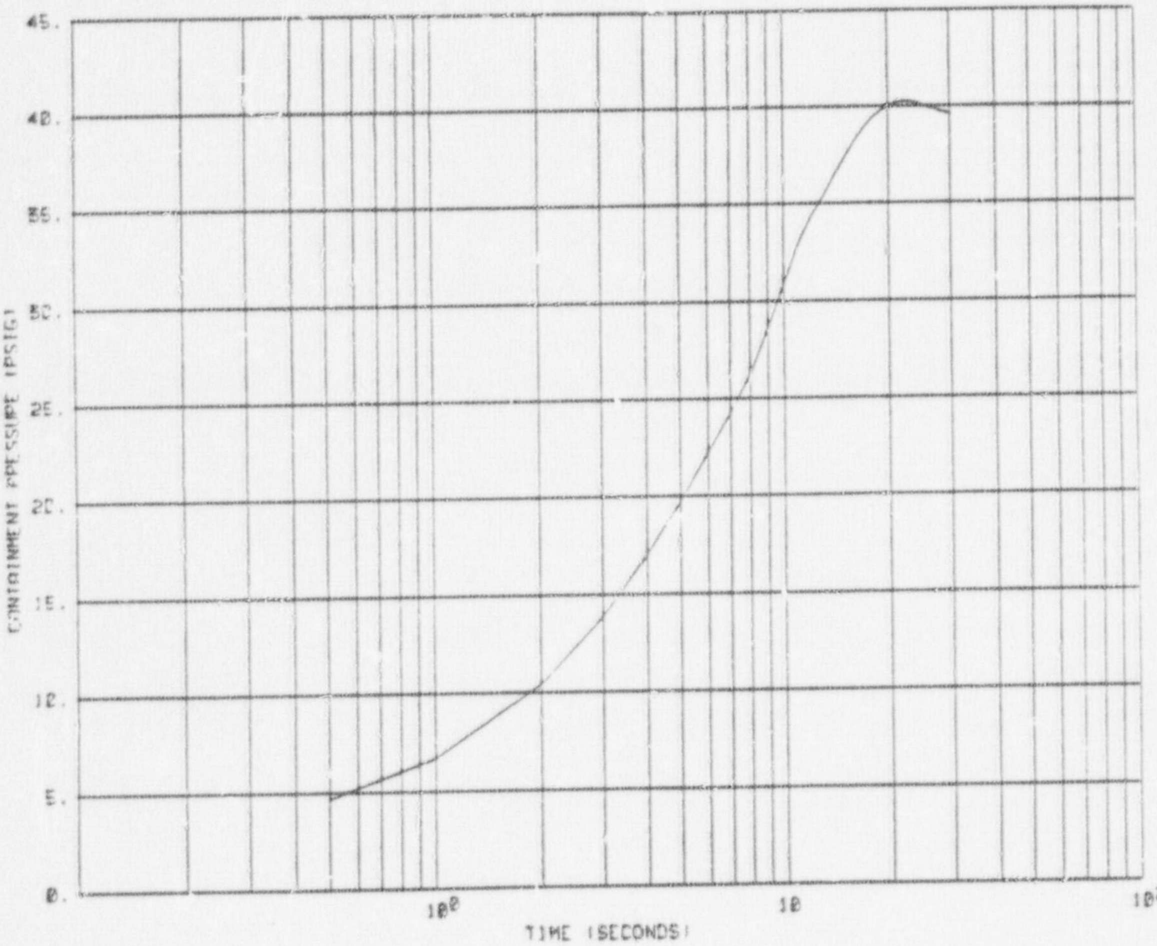

FIGURE 4-4
CONTAINMENT TEMPERATURE vs. TIME
DEHL

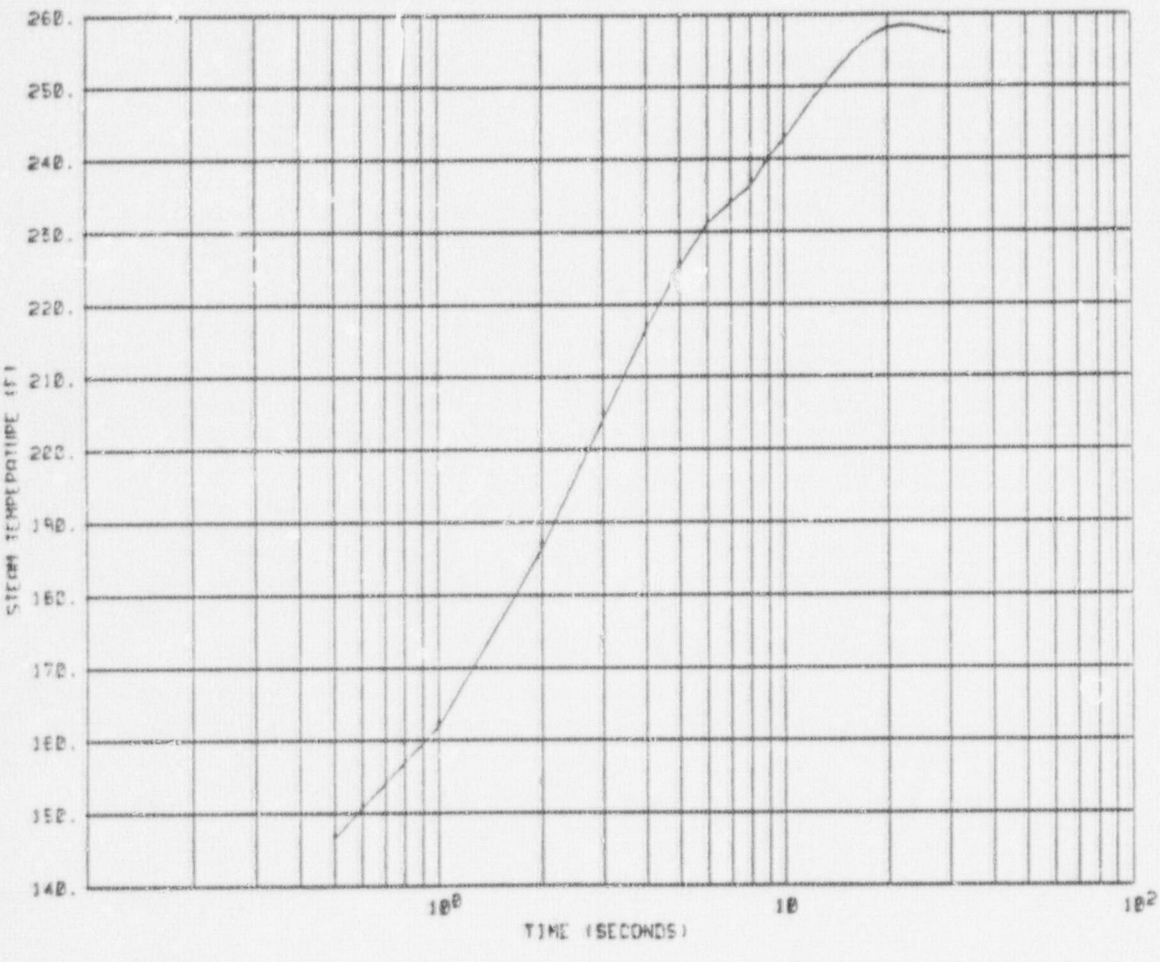

TABLE 4-1

CONTAINMENT HEAT SINKS

No.	Material	Heat Transfer Area ft^2	Thickness ft
1	Paint	41302.	0.000625
	Steel		0.03125
	Concrete		1.0
2	Paint	28613.	0.000625
	Steel		0.04167
	Conrete		1.0
3	Paint	15000.	0.000625
	Concrete		1.0
4	Stainless Steel	10000.	0.03125
	Concrete		1.0
5	Paint	61000.	0.000625
	Concrete		1.0
6	Paint	68792.	0.000625
	Steel		0.0417
7	Paint	81704.	0.000625
	Steel		0.03125
8	Paint	27948.	0.000625
	Steel		0.02083
9	Paint	69800.	0.000625
	Steel		0.015625
10	Paint	3000.	0.000625
	Steel		0.01042
11	Paint	22000.	0.000625
	Steel		0.01152
12	Paint	10000.	0.000625
	Steel		0.0052

TABLE 4-2

THERMOPHYSICAL PROPERTIES OF CONTAINMENT HEAT SINKS

Material	Thermal Conductivity (Btu/hr-ft - °F)	Volumetr Heat Capacity (Btu/ t^3 - °F)
Paint	0.2083	36.86
Steel	26.0	56.35
Stainless Steel	8.6	56.35
Concrete	0.8	28.8

TABLE 4-3

CHRONOLOGY OF EVENTS FOR LOCA - DEPS

Time (Seconds)	Event
0.0	Start of accident
24.8	End of blowdown phase
39.0	Containment fan coolers start
68.0	Containment sprays start
171.5	End of reflood phase
799.0	Peak Pressure Reached
2351.3	Sump recirculation starts

5.0 MSLB CONTAINMENT INTEGRITY (PEAK PRESSURE) ANALYSIS

5.1 BASE SCOPE MSLB ANALYSIS

The purpose of the Containment Integrity Main Steam Line Break (MSLB) analysis is to demonstrate the acceptability of the Containment Safeguards Systems to mitigate the consequences of a hypothetical rupture of a main steam line pipe. The impact of steam line break mass and energy releases on containment pressure is addressed to assure the containment pressure remains below its design pressure of 47 psig at the rated 3025 MWt power conditions.

The LOFTRAN computer program, Reference 9, was used to generate the mass and energy released to the containment. The following plots show the containment responses as calculated by the COCO computer program:

Figure 5-1 Containment Pressure vs. Time at 0 % Power, MSCV Failure, With Offsite Power

Figure 5-2 Containment Temperature vs. Time at 0 % Power, MSCV Failure, With Offsite Power

Figure 5-3 Containment Pressure vs. Time at 70 % Power, MSCV Failure, With Offsite Power

Figure 5-4 Containment Temperature vs. Time at 70 % Power, MSCV Failure, With Offsite Power

Figure 5-5 Containment Pressure vs. Time at 100 % Power, MSCV Failure, With Offsite Power

Figure 5-6 Containment Temperature vs. Time at 100 % Power, MSCV Failure, With Offsite Power

Figure 5-7 Containment Pressure vs. Time at 0 % Power, Diesel Failure, Without Offsite Power

Figure 5-20 Containment Temperature vs. Time at 100 % Power, AFW Failure, With Offsite Power

The COCO computer code, Reference 2, was used to generate the containment response. For the MSLB diesel failure case, the containment model was similar to that used for the Long Term LOCA Containment Integrity Analysis. For the other cases full containment safeguards were used. The limiting single failure for pressure consideration was the FCV failure case which assumed an initial ambient containment temperature of 130°F, 95°F service water temperature and full safeguards of 5 fan coolers and two spray pumps. The containment peak pressure was calculated to be 42.28 psig compared to the containment design value of 47 psig. Note that the MSLB peak pressure is higher than the peak containment pressure for LOCA. The calculated pressure and temperature time histories for the MSLB cases are shown in Figures 5-1 thru 5-20. Table 5-1 summarizes the peak pressure and temperature for the MSLB cases.

Based on the analysis performed, the Containment Safeguards Systems are capable of mitigating the pressure consequences of a hypothetical rupture of a main steam line break.

5.2 ADDITIONAL INVESTIGATION

An additional investigation of other areas of margin was made. The most limiting case was reviewed to determine if credit for Primary or Secondary System trip signals can provide earlier automatic actuation times and improved results. The limiting full double-ended rupture analysis results were reviewed for consideration of enhanced containment wall heat transfer based on break size considerations. This included turbulent Tagami wall heat transfer and revaporization of wall condensate. Prior to initiation of fan coolers and containment sprays enhanced heat transfer would cause the containment pressure and temperature to build up slower than without enhanced heat transfer. For cases where the peak pressure occurs late in time, i.e., later than 500 seconds, although not typical, a delay in actuation of containment spray and fan cooler initiation could result in a slightly higher containment pressure. These models typically result in reduced pressures but may also result in a smaller break size becoming limiting. As a sensitivity case, a 0.6 ft^2 break size was modeled for the limiting base scope 70% power FCV failure transient to provide insight as to what the peak pressure change would be for the largest break size which could not credit revaporization and enhanced heat transfer. Additionally, the impact of reduced operator action times (i.e., 10 minutes) was assessed.

The limiting peak pressure case as illustrated in Table 5-1 is the Feedwater Control Valve (FCV) Failure Case at 70% Power with offsite power available. The peak pressure for this case is calculated to be 42.28 psig. Following are results of additional sensitivities for this limiting case:

- 30 SECOND FCV CLOSURE TIME

  A 122 second total FCV closure time was utilized in the base run. Assuming the valve would close in 30 seconds results in a pressure reduction of 12.1. Because of this, the MSCV case, with a calculated peak pressure of 38.68 psig, would become the limiting MSLB case. Therefore, the overall reduction is 3.6 psi.

- REDUCED FEEDWATER FLASHING VOLUME

  The feedwater flashing volume is defined as the volume between the SG inlet nozzle and the last Feedwater Isolation Valve (FIV). Reducing the volume from 800 cubic feet to a more appropriate volume of 260 cubic feet reduces the peak pressure by 2.3 psi.

- BORON INJECTION TANK WITH BORON

  The base analysis assumed 0% Boron concentration. The effect of including the tank with 20,000 ppm Boron is a reduction in pressure of 4.9 psi.

- REDUCED SI TIME DELAY

  The peak pressure case assumed a 6 second SI delay followed by a 12 second linear pump ramp to full flow. Reducing this delay to 5 seconds followed by a 6 second linear pump ramp results in no peak pressure benefit.

- TURBULENT TAGAMI WALL HEAT TRANSFER & 100% REVAPORIZATION OF WALL CONDENSATE

  Turbulent Tagami heat transfer is discussed in Section 4.3. A major thermodynamic assumption that can be made is that complete revaporization of the saturated condensate occurs under superheated containment conditions. For smaller breaks, revaporization is not typically assumed. Including these effects produces variations from essentally no change to 2.8 psi depending upon the failure mode.

  Because of a delay in the actuation of the containment sprays due to the setpoint being reached later and due to the corresponding reduction in integrated spray heat removal at the time of the peak pressure, the peak pressure increased slightly for the FCV failure at 70 percent power case when Turbulent Tagami heat transfer and 100% revaporization were included. Therefore the peak pressure (without including additional benefits) for the MSLB event is 42.42 psig. Figures 5-37 and 5-38 illustrate the pressure and temperature time histories for this case.

- 10 MIN. OPERATOR ACTION TIME

  Since all of the pressure peaks for the base case calculations occur before 10 minutes the effect of this is only to increase the rate of pressure decay after 600 seconds. A separate computer calculation was not made to assess this effect.

- BREAK SIZE CONSIDERATION

  For double-ended breaks or any breaks where sufficient turbulence in containment atmosphere occurs, enhanced Tagami heat transfer is applicable. Some of the previously mentioned effects are break size dependent, including the Turbulent Tagami heat transfer. If these break size dependent effects are considered then other factors associated with break size considerations should also be considered. Based upon available data and engineering judgement it was determined that a break size of 0.60 square feet should be evaluated for potential pressure impacts. The peak pressure for this case was calculated to be 33.55 psig, therefore this demonstrates that the double-ended break size is limiting for pressure.

- OTHER SAFETY ACTUATIONS

  In evaluating the effect of primary and secondary protection signals it was concluded that there would be no benefit for the 70% power FCV failure case or the 100% power MSCV case because the High-1 signal is reached before any primary or secondary trip signals would be reached.

In summary, there is potential pressure margin that can be removed from the worst MSLB base case if more detailed work is performed and the FIV is assumed to respond faster.

The purpose of the MSLB analysis and evaluation described in this report is to show that sufficient margin exists in containment pressure. In order to obtain the peak pressure case and to limit the number of cases, all break size dependencies, including Turbulent Tagami heat transfer and 100% revaporization, were factored out of the base analyses. Because of the exclusion of this enhanced heat transfer phenomena higher than expected transient temperatures were observed.

The current MSLB design basis break size is a full double-ended rupture of the steam line. Turbulent Tagami and 100% revaporization are considered. For the double-ended break results described herein, the peak temperature for the MSCV failure, 0% power case is reduced from 378.6 degrees Fahrenheit to 269.9 degrees Fahrenheit when turbulent Tagami heat transfer and 100% revaporization are included. (See Table 5-2 for the effects of other cases that included Turbulent Tagami and 100% revaporization.) This is below the present EQ peak of 287 degrees Fahrenheit (Reference 20).

The times that HI-1 and HI-2 are reached are utilized in the generation of the mass and energy releases. These times are determined from the COCO computer program results. For the base case runs iterations were made to ensure that a conservative set of values were utilized in the mass and energy release calculations. No iterations quantifying the effect of changes in HI-1 and HI-2 were explicitly made for the runs that included turbulent Tagami and 100% revaporization. For the FCV failure and Diesel failure cases only HI-1 is relevant. The time HI-1 was reached for all failure cases did not change when turbulent Tagami and 100% revaporization were considered. Therefore there is no impact on the results presented in this document for these two failure cases whenever turbulent Tagami and 100% revaporization were considered. For the MSCV failure case, the time to reach HI-2 is delayed. The impact of this is approximately a 1.3 psi increase in peak pressure and a 5°F increase in peak temperature (above those illustrated in Table 5-2 and Figures 5-33 and 5-34).

In order to quantify the effect of MSLB on the containment temperature transient different break sizes and types could be considered. For example, for a split break where enhanced containment wall heat transfer may not be applicable, the peak temperature could approach the double-ended break case result that did not include the enhanced heat transfer. However, consistent with Reference 17, the duration of the MSLB temperature spike is so short that the thermal capacity of the equipment damps the higher environmental temperatures, therefore LOCA produces the most severe equipment conditions. Westinghouse has performed detailed MSLB analyses, considering break type, break size, power, single failure, etc., and equipment thermal lag analyses for a similar application. Although a detailed comparison of limiting equipment for the similar application and Indian Point Unit 3 was not made, that evaluation demonstrated that LOCA produces the most severe equipment qualification conditions when thermal lag of equipment is considered.

Table 5-2 illustrates the pressures and temperatures for the MSLB sensitivity. The following plots show the containment response:

FIGURE 5-1

CONTAINMENT PRESSURE vs. TIME

0% POWER MSCV FAILURE WITH OFFSITE POWER

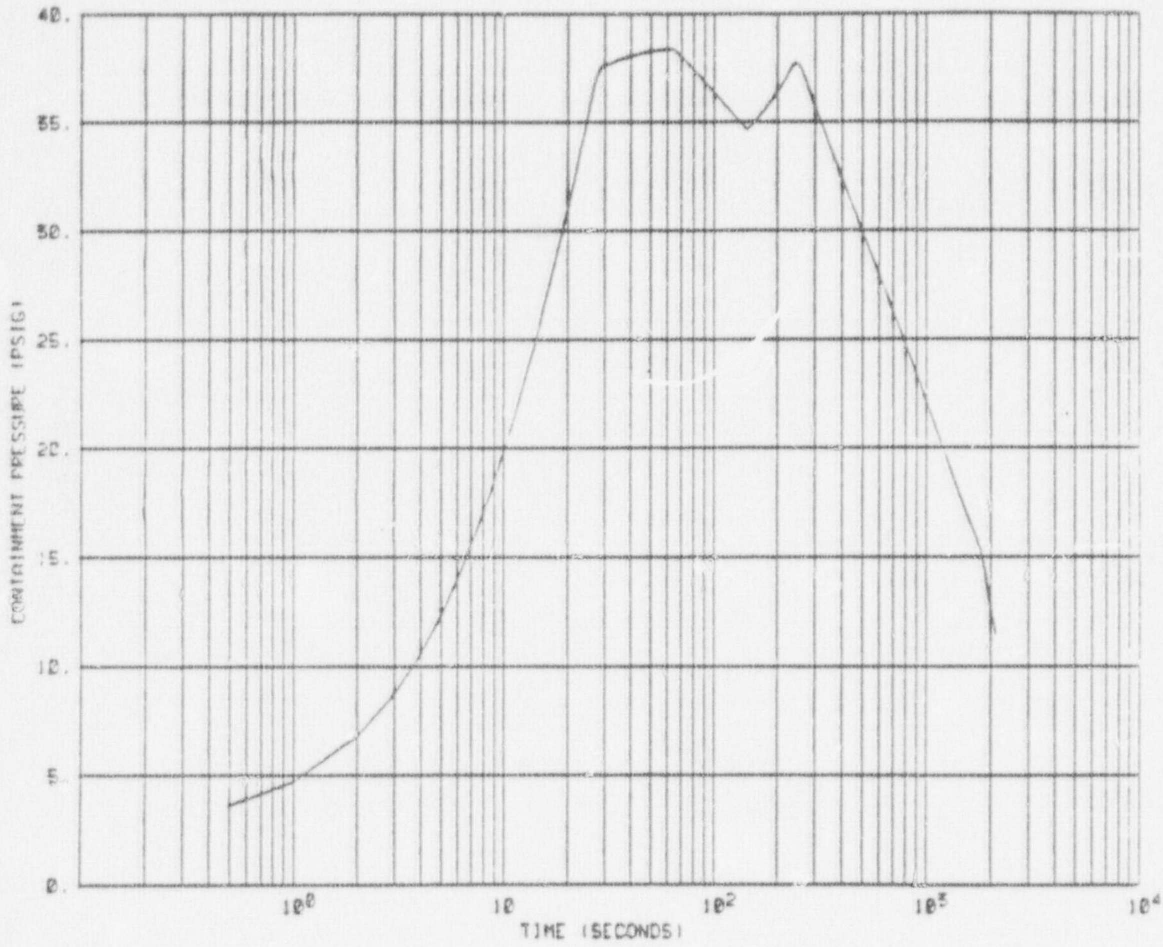

FIGURE 5-2

CONTAINMENT TEMPERATURE vs. TIME

0% POWER MSCV FAILURE WITH OFFSITE POWER

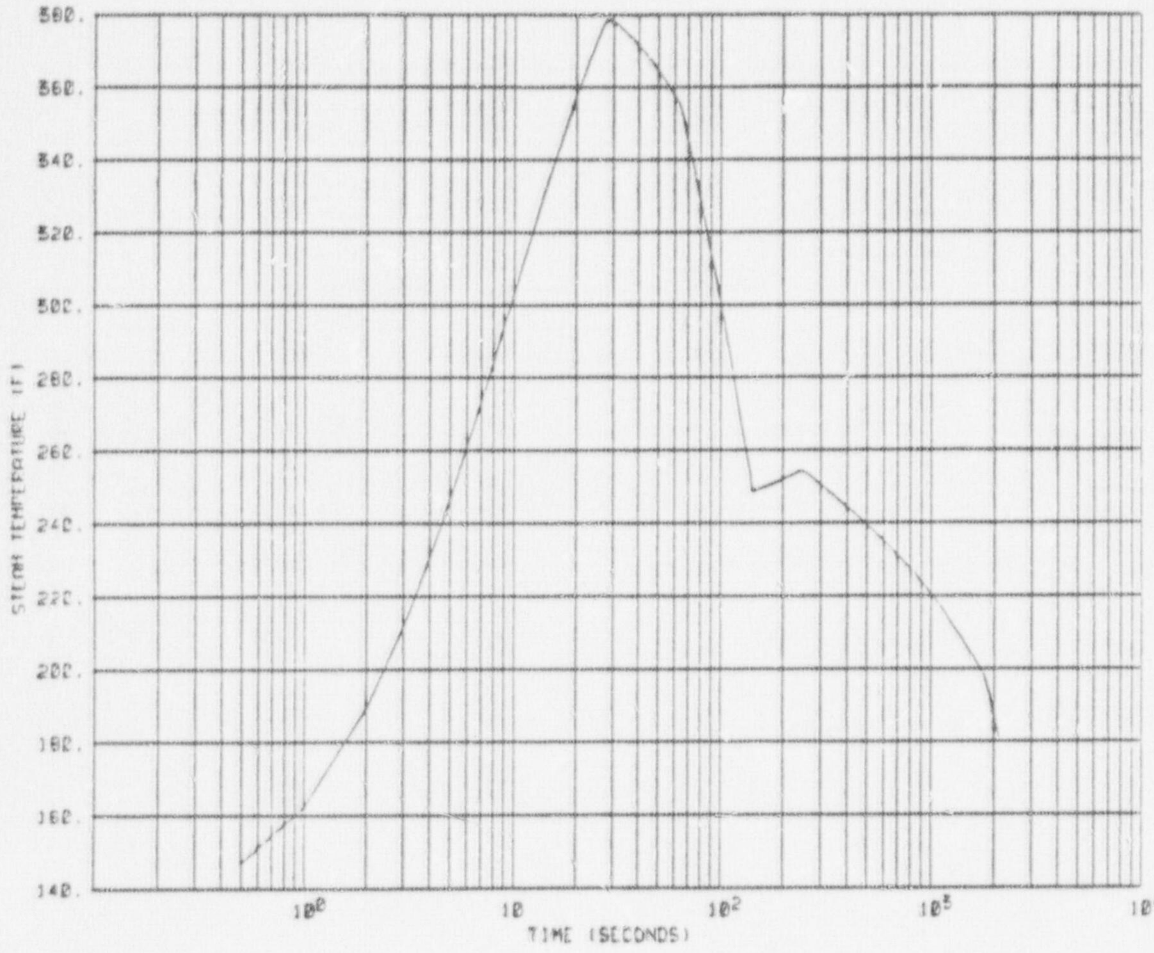

FIGURE 5-3

CONTAINMENT PRESSURE vs. TIME

70% POWER MSCV FAILURE WITH OFFSITE POWER

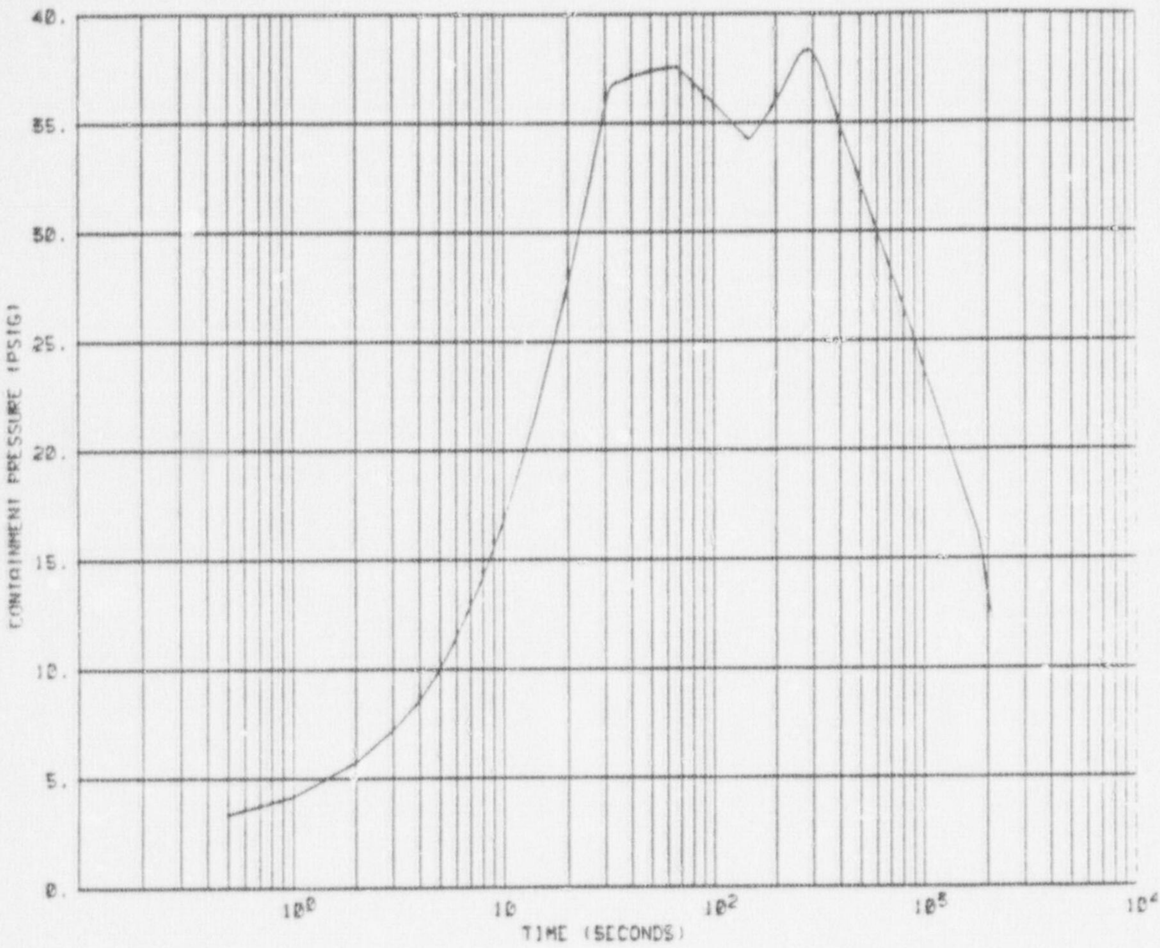

FIGURE 5-4

CONTAINMENT TEMPERATURE vs. TIME

70% POWER MSCV FAILURE WITH OFFSITE POWER

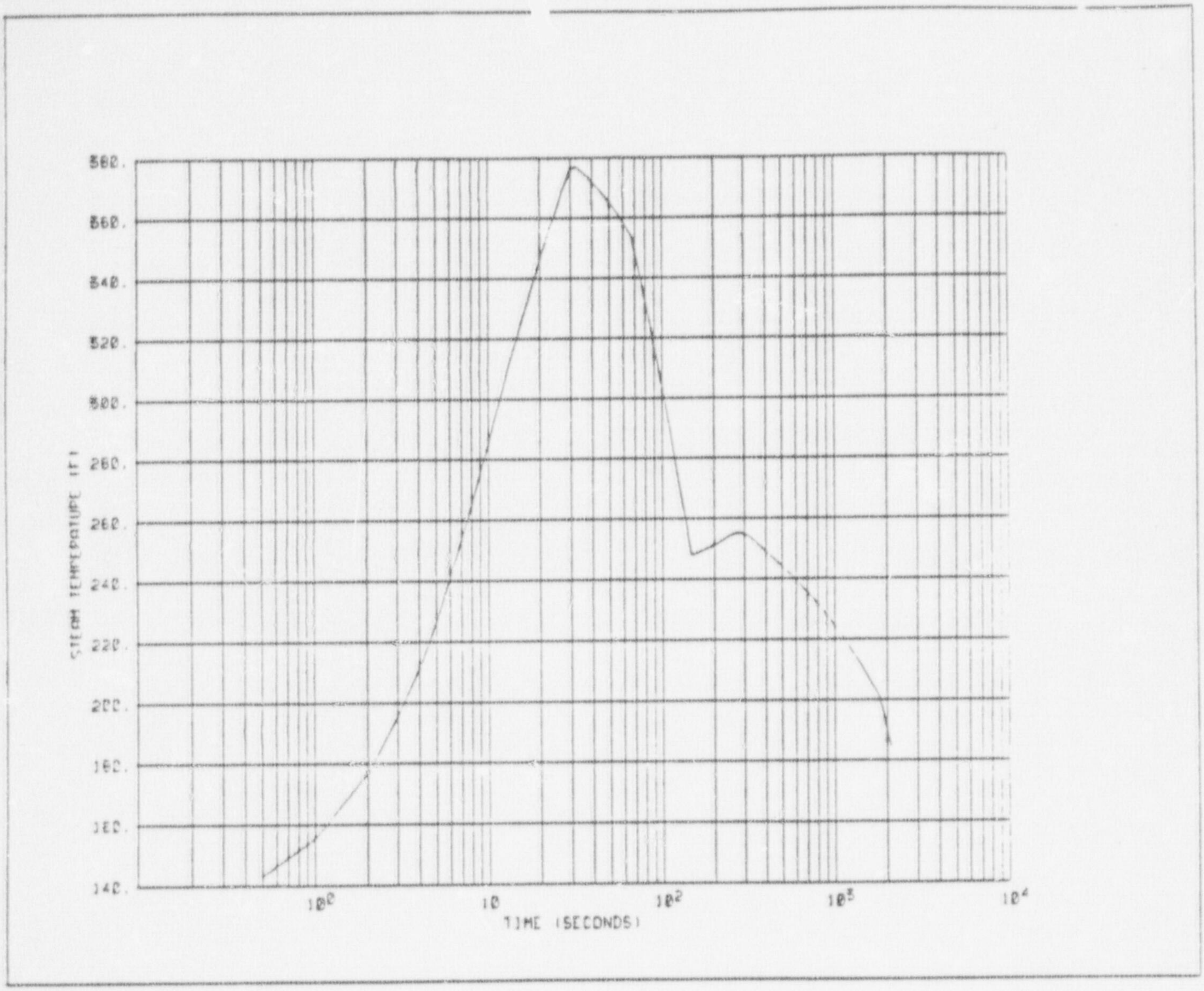

FIGURE 5-5

CONTAINMENT PRESSURE vs. TIME

100% POWER MSCV FAILURE WITH OFFSITE POWER

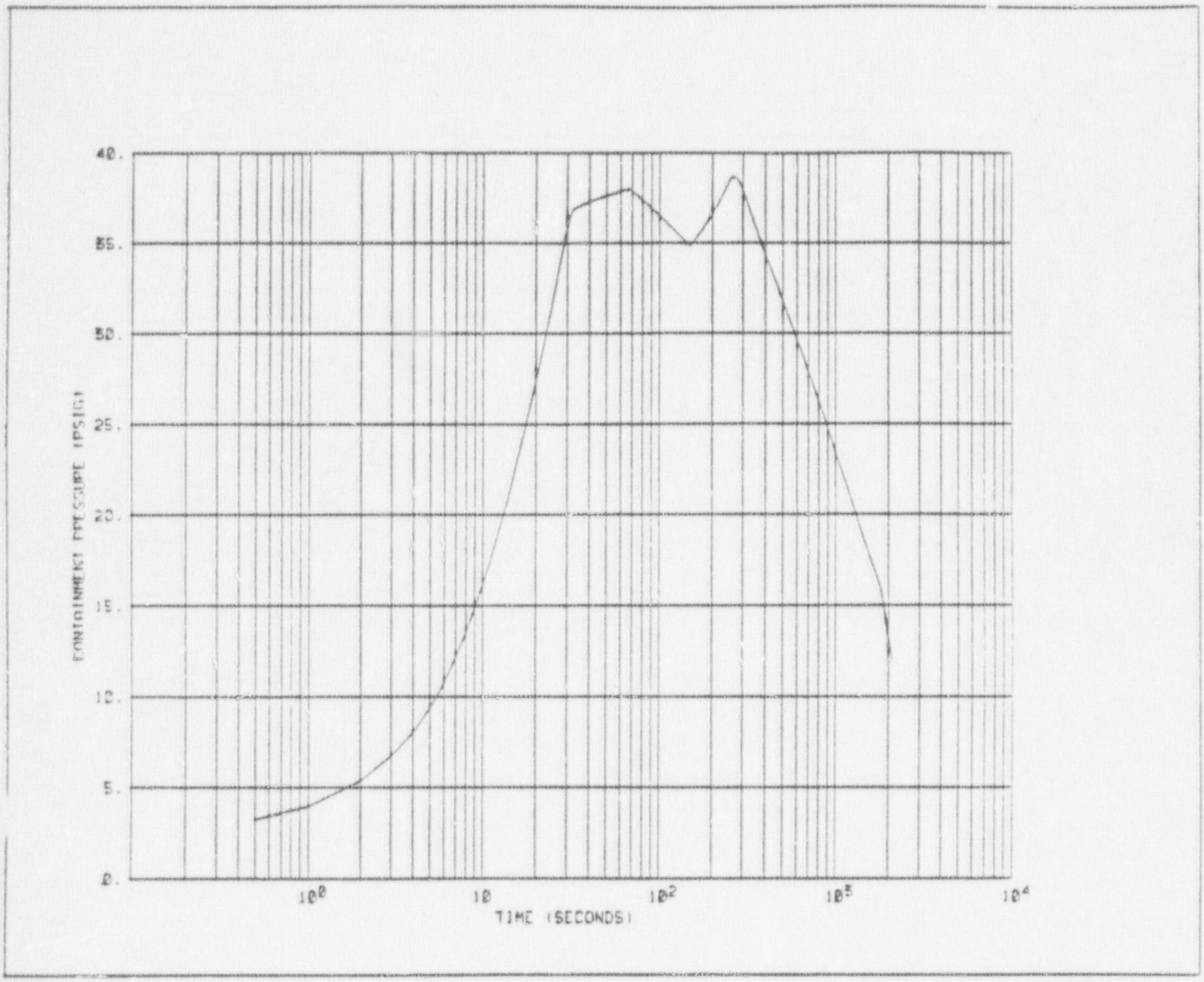

FIGURE 5-6

CONTAINMENT TEMPERATURE vs. TIME

100% POWER MSCV FAILURE WITH OFFSITE POWER

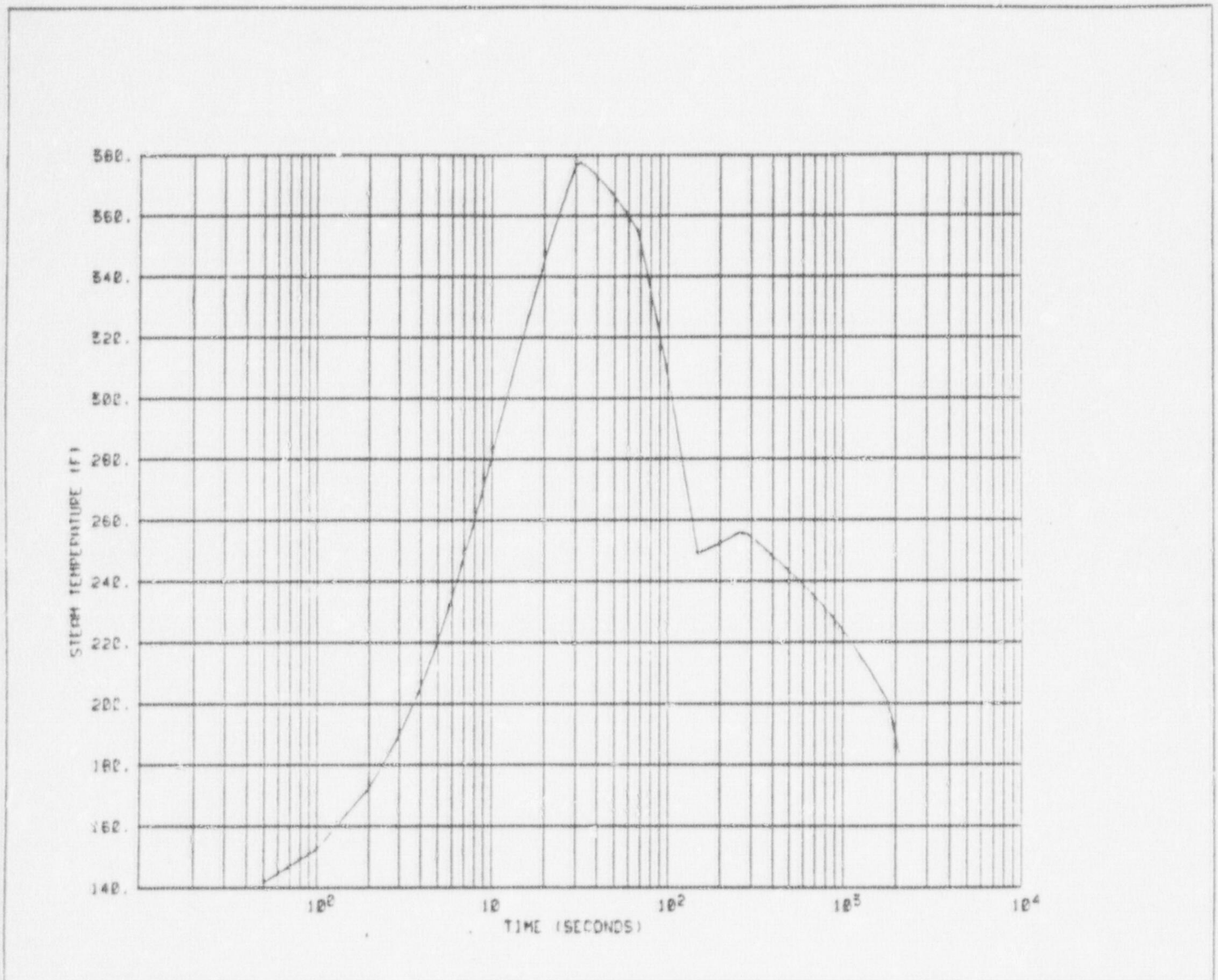

FIGURE 5-7

CONTAINMENT PRESSURE vs. TIME 0% POWER

DIESEL FAILURE WITHOUT OFFSITE POWER

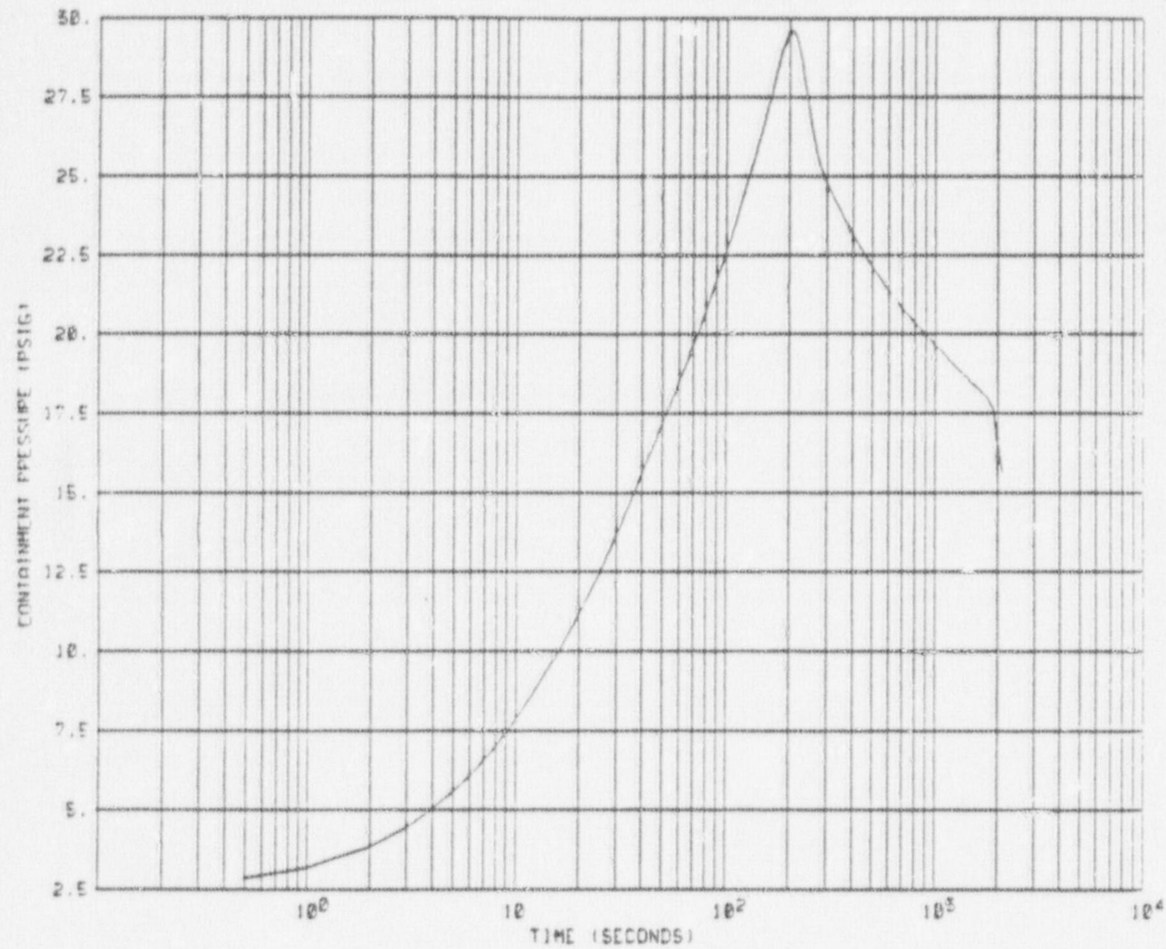

FIGURE 5-8

CONTAINMENT TEMPERATURE vs. TIME

0% POWER DIESEL FAILURE WITHOUT OFFSITE POWER

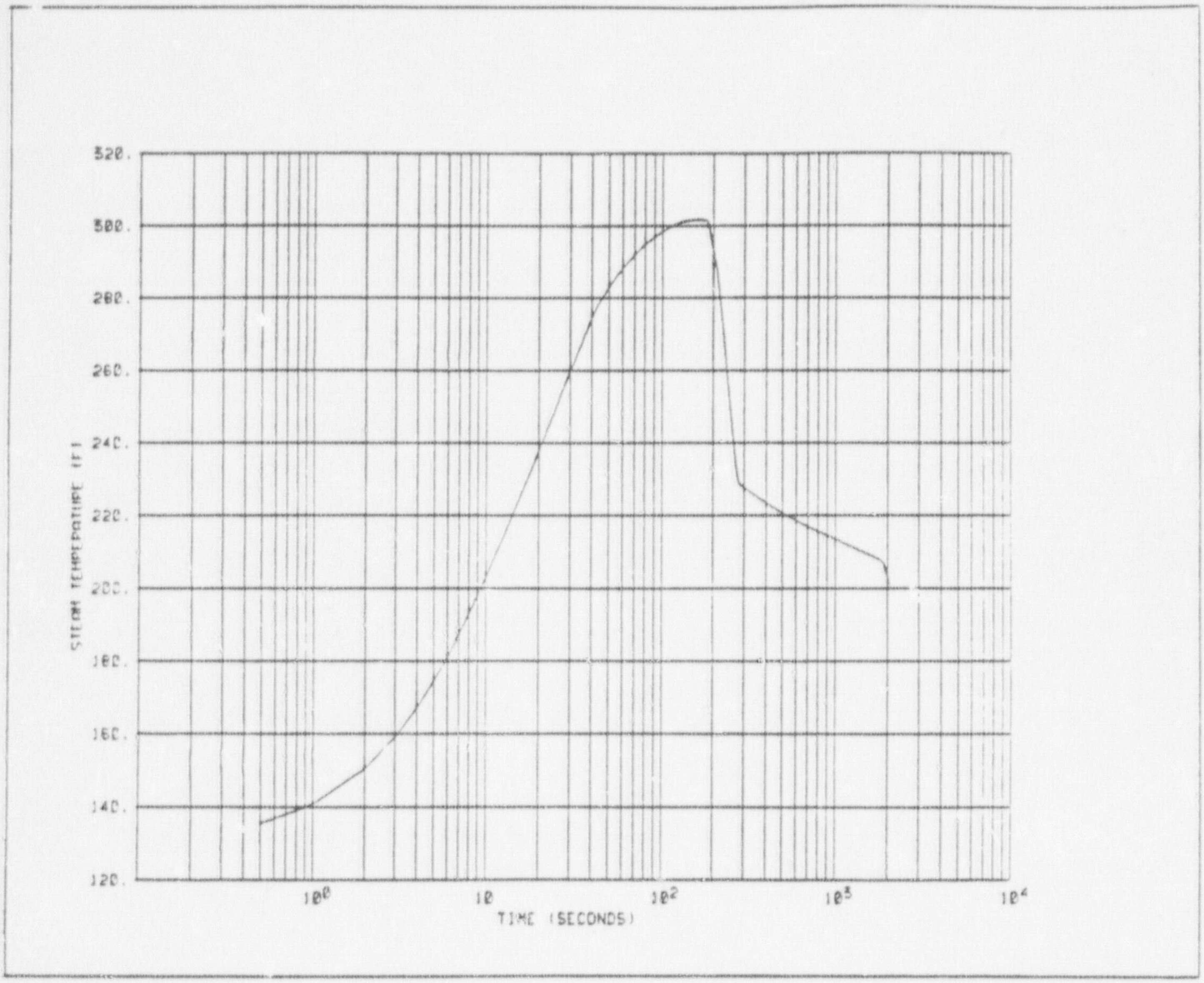

FIGURE 5-9

CONTAINMENT PRESSURE vs. TIME

100% POWER DIESEL FAILURE WITHOUT OFFSITE POWER

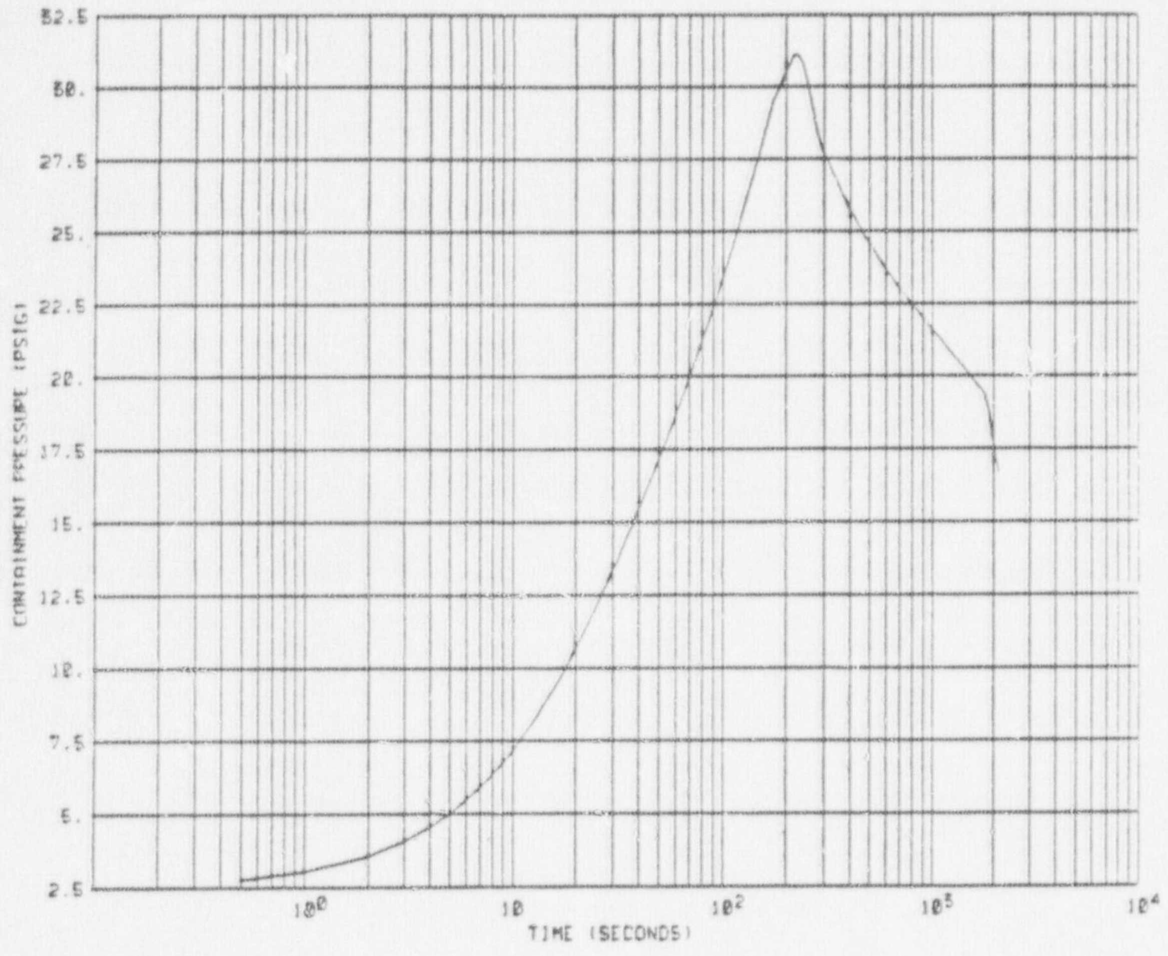

FIGURE 5-10

CONTAINMENT TEMPERATURE vs. TIME 100% POWER

DIESEL FAILURE WITHOUT OFFSITE POWER

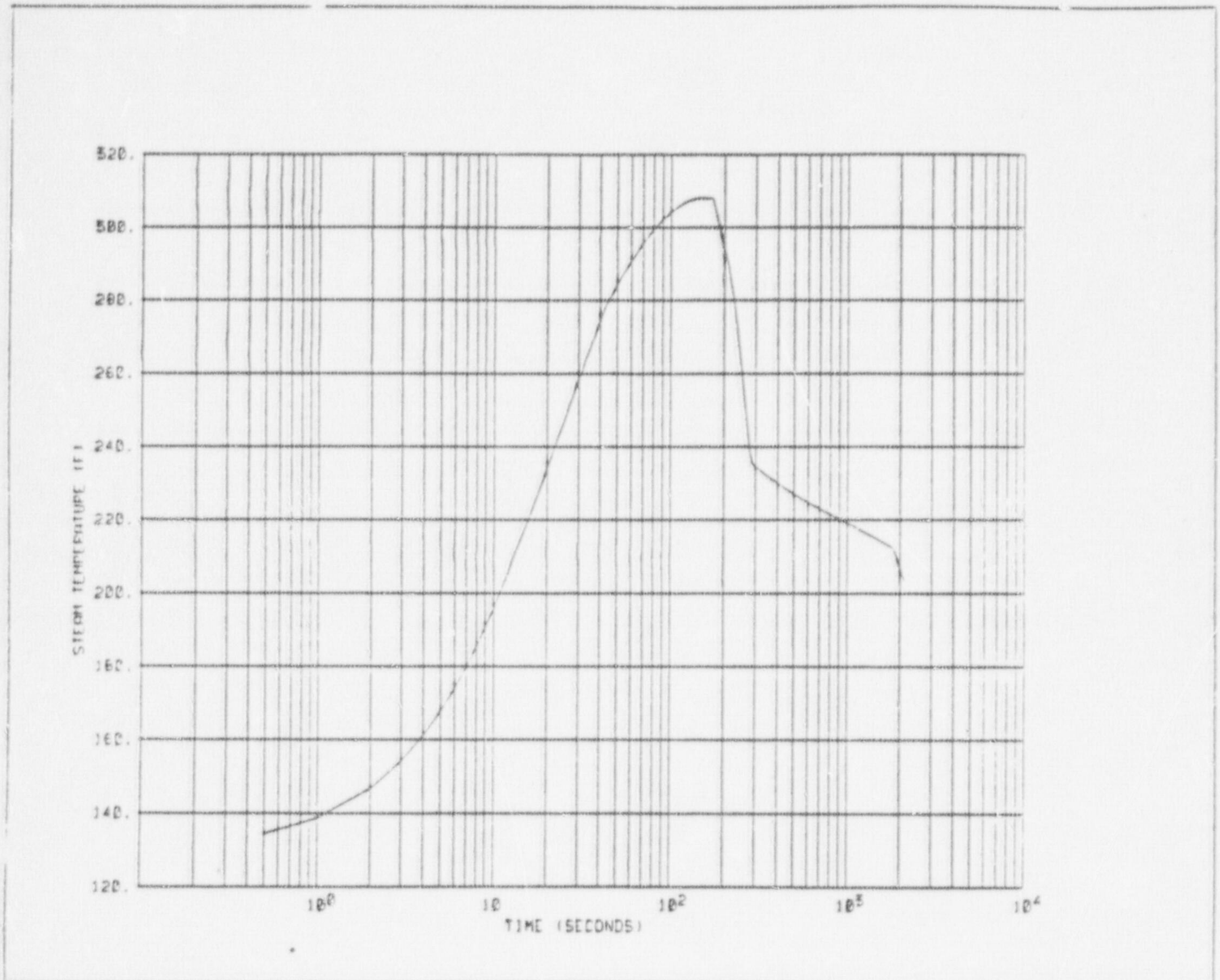

FIGURE 5-11

CONTAINMENT PRESSURE vs. TIME

0% POWER FCV FAILURE WITH OFFSITE POWER

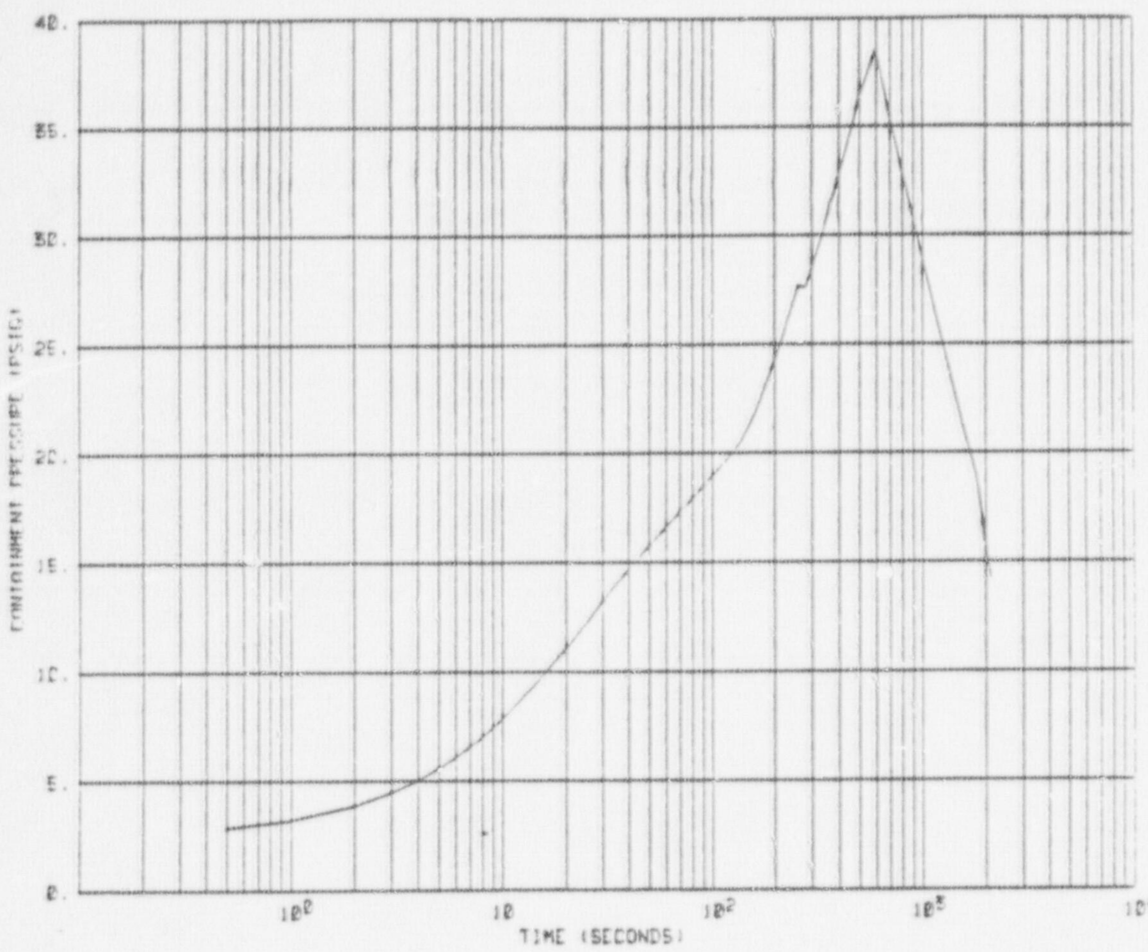

FIGURE 5-12

CONTAINMENT TEMPERATURE vs. TIME

0% POWER FCV FAILURE WITH OFFSITE POWER

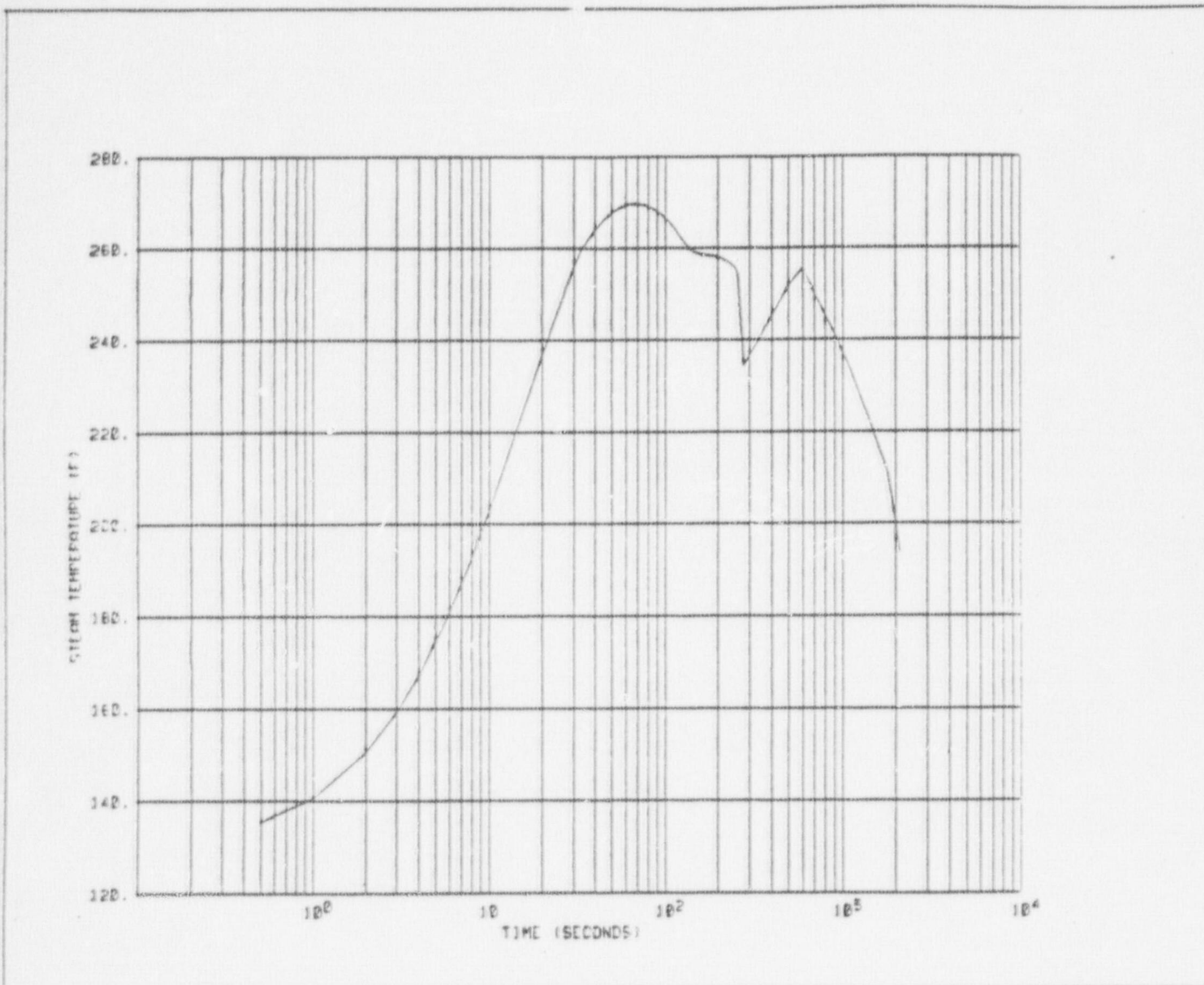

FIGURE 5-13
CONTAINMENT PRESSURE vs. TIME
30% POWER FCV FAILURE WITH OFFSITE POWER

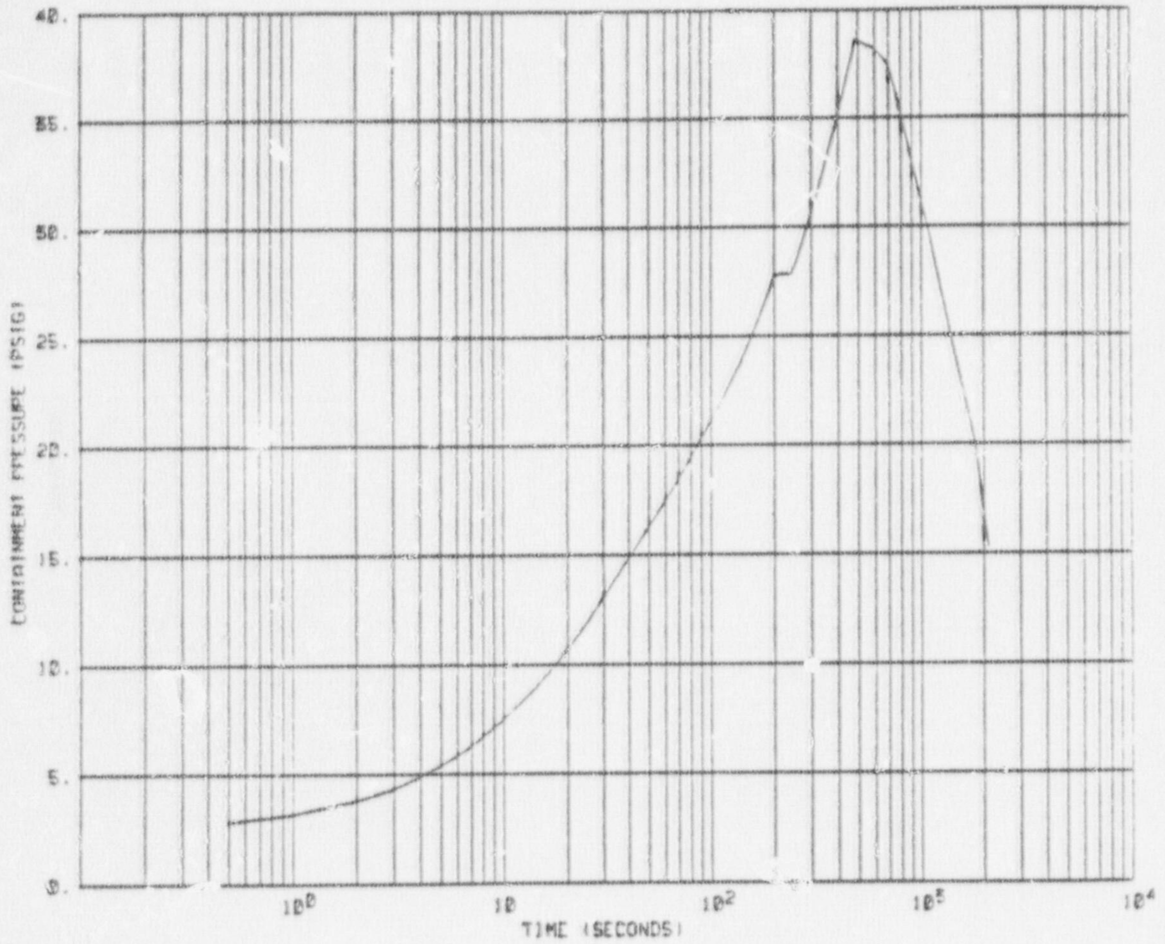

FIGURE 5-14
CONTAINMENT TEMPERATURE vs. TIME
30% POWER FCV FAILURE WITH OFFSITE POWER

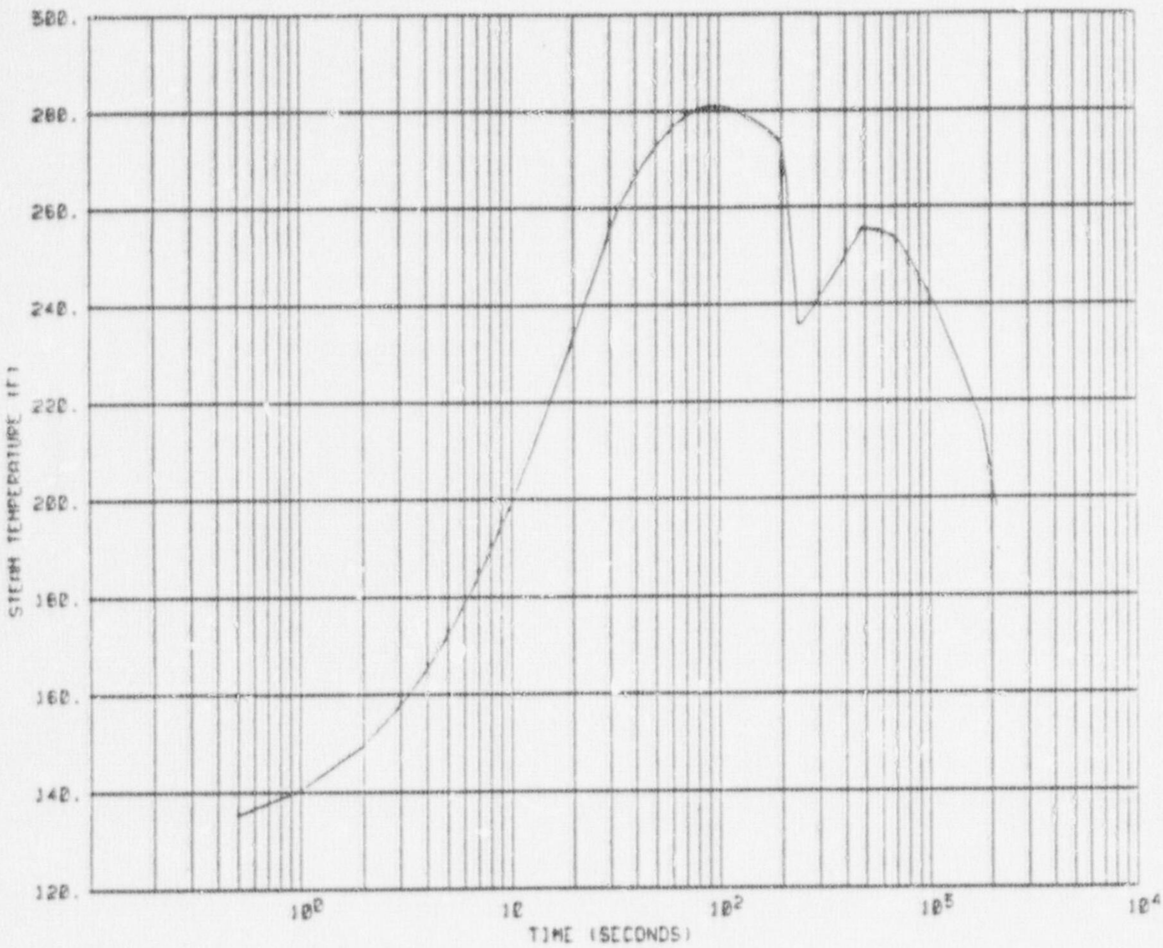

FIGURE 5-15

CONTAINMENT PRESSURE vs. TIME

70% POWER FCV FAILURE WITH OFFSITE POWER

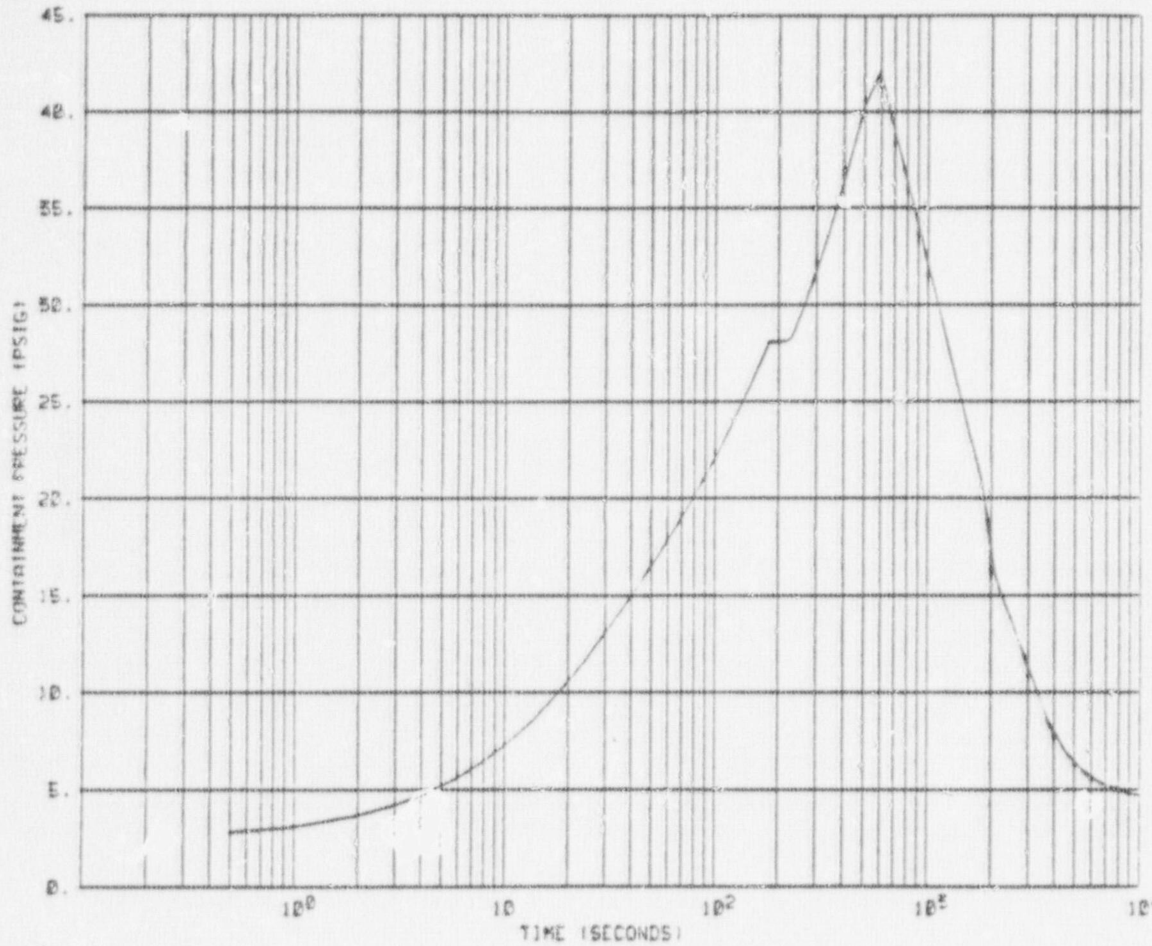

FIGURE 5-16
CONTAINMENT TEMPERATURE vs. TIME
70% POWER FCV FAILURE WITH OFFSITE POWER

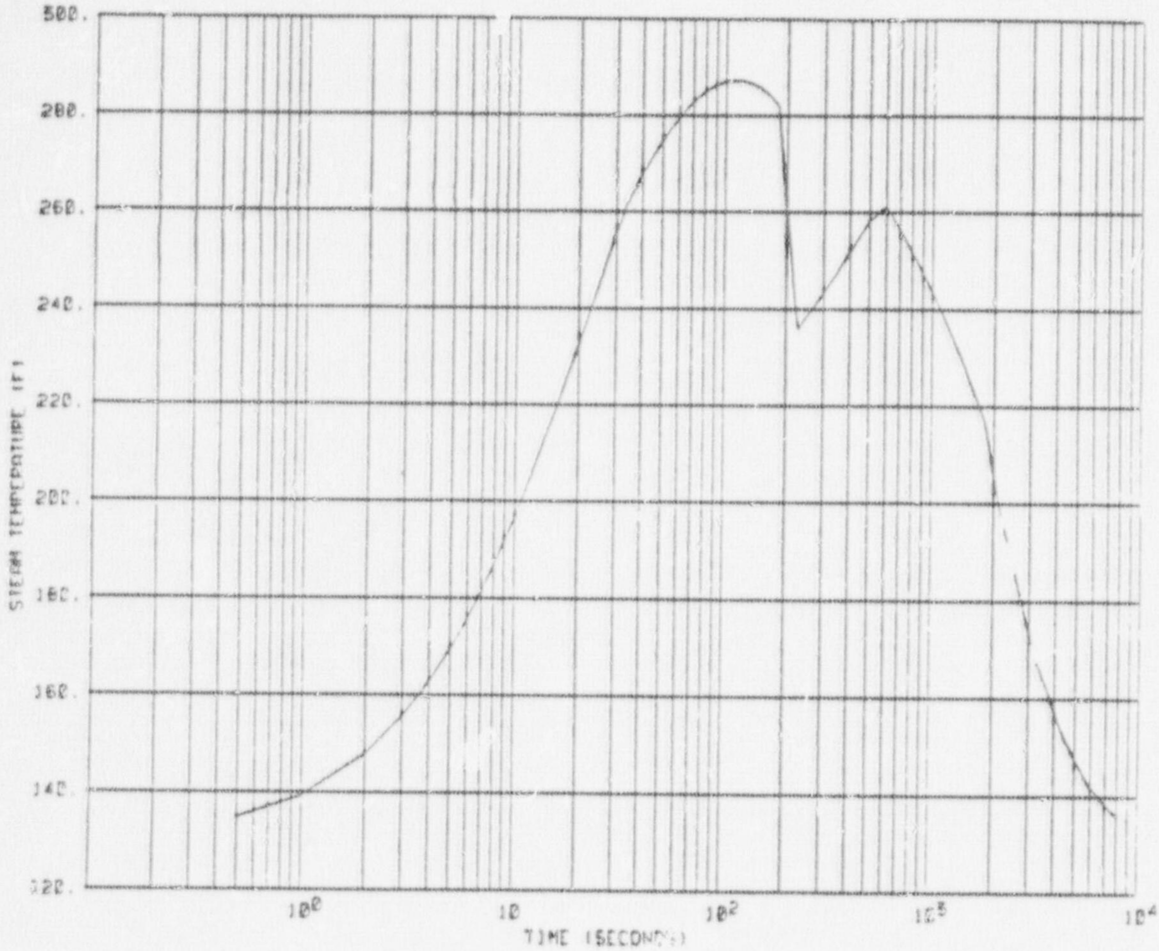

FIGURE 5-17
CONTAINMENT PRESSURE vs. TIME
100% POWER FCV FAILURE WITH OFFSITE POWER

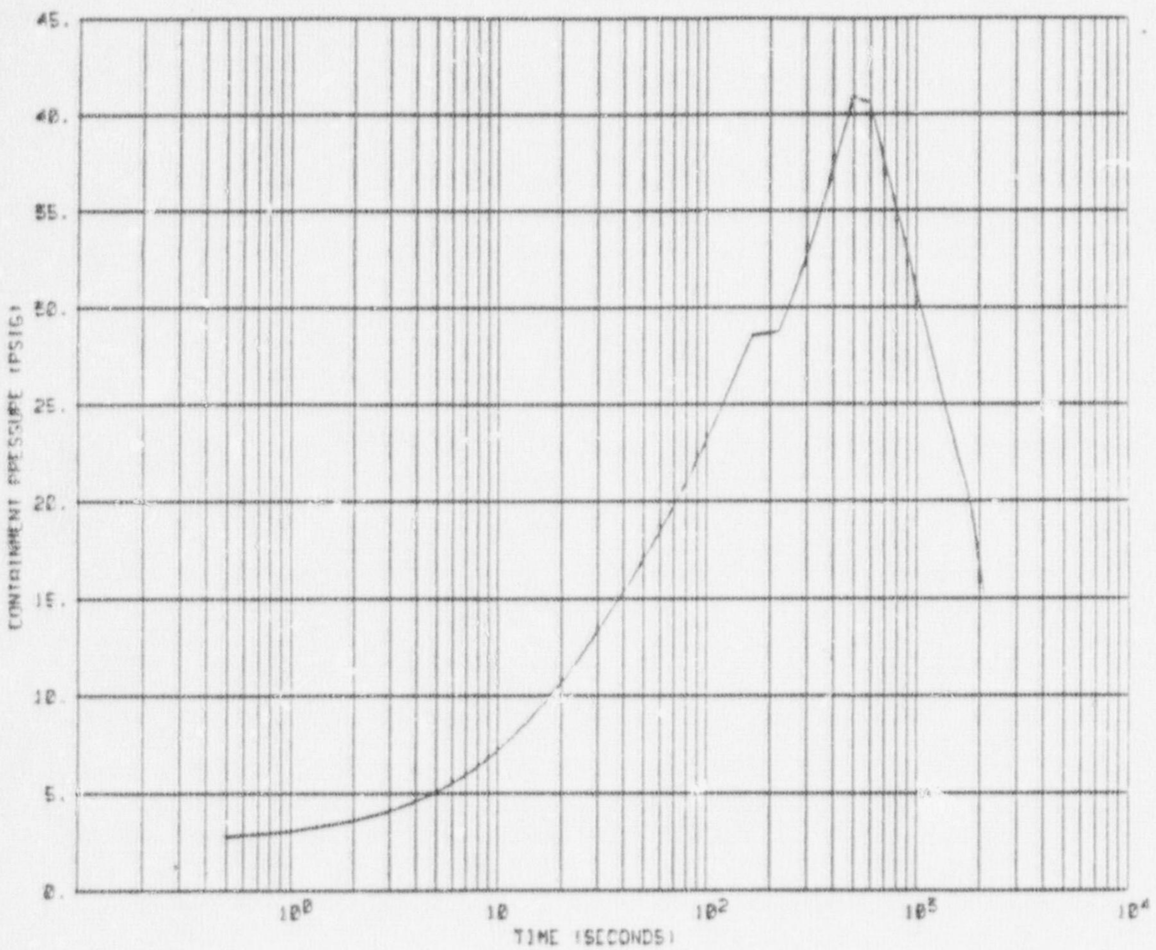

FIGURE 5-18
CONTAINMENT TEMPERATURE vs. TIME
100% POWER FCV FAILURE WITH OFFSITE POWER

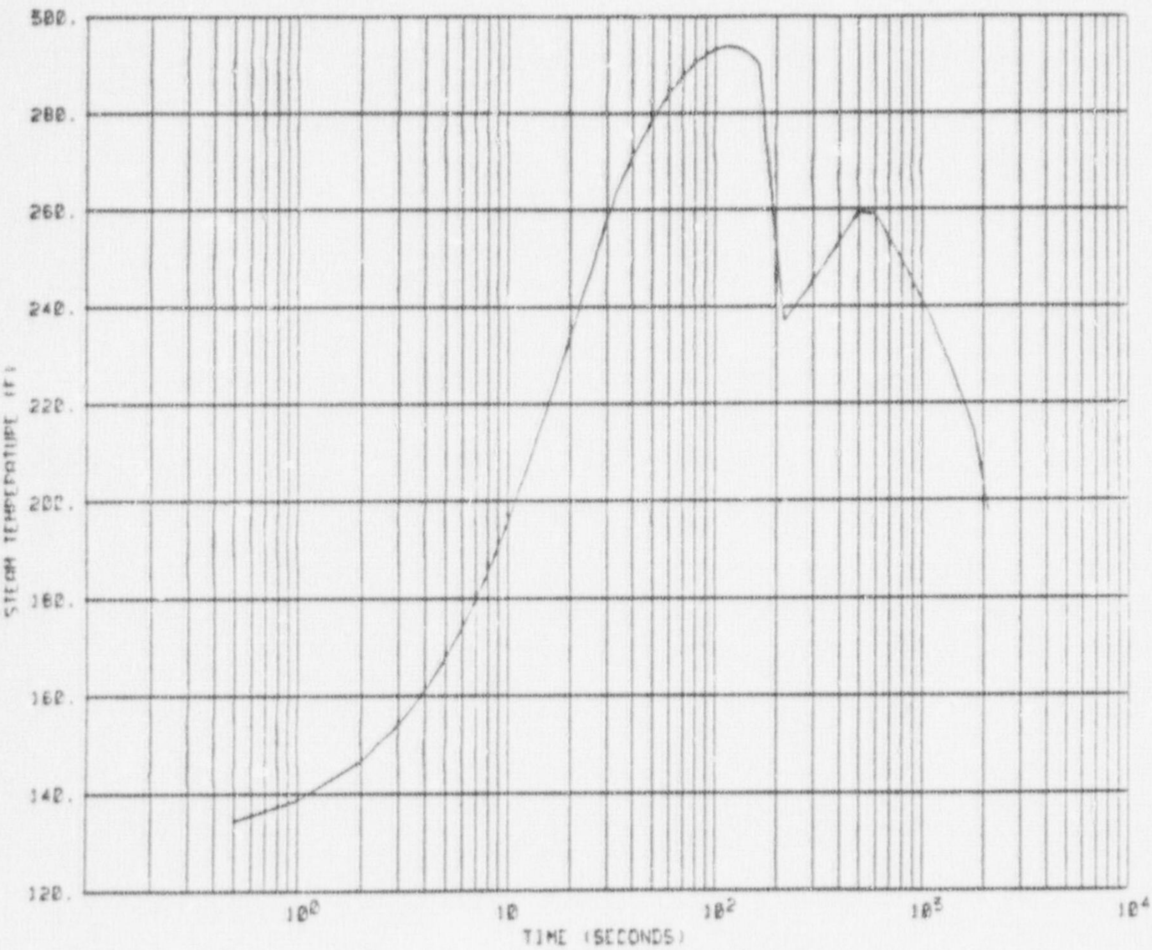

FIGURE 5-19

CONTAINMENT PRESSURE vs. TIME

100% POWER AFW FAILURE WITH OFFSITE POWER

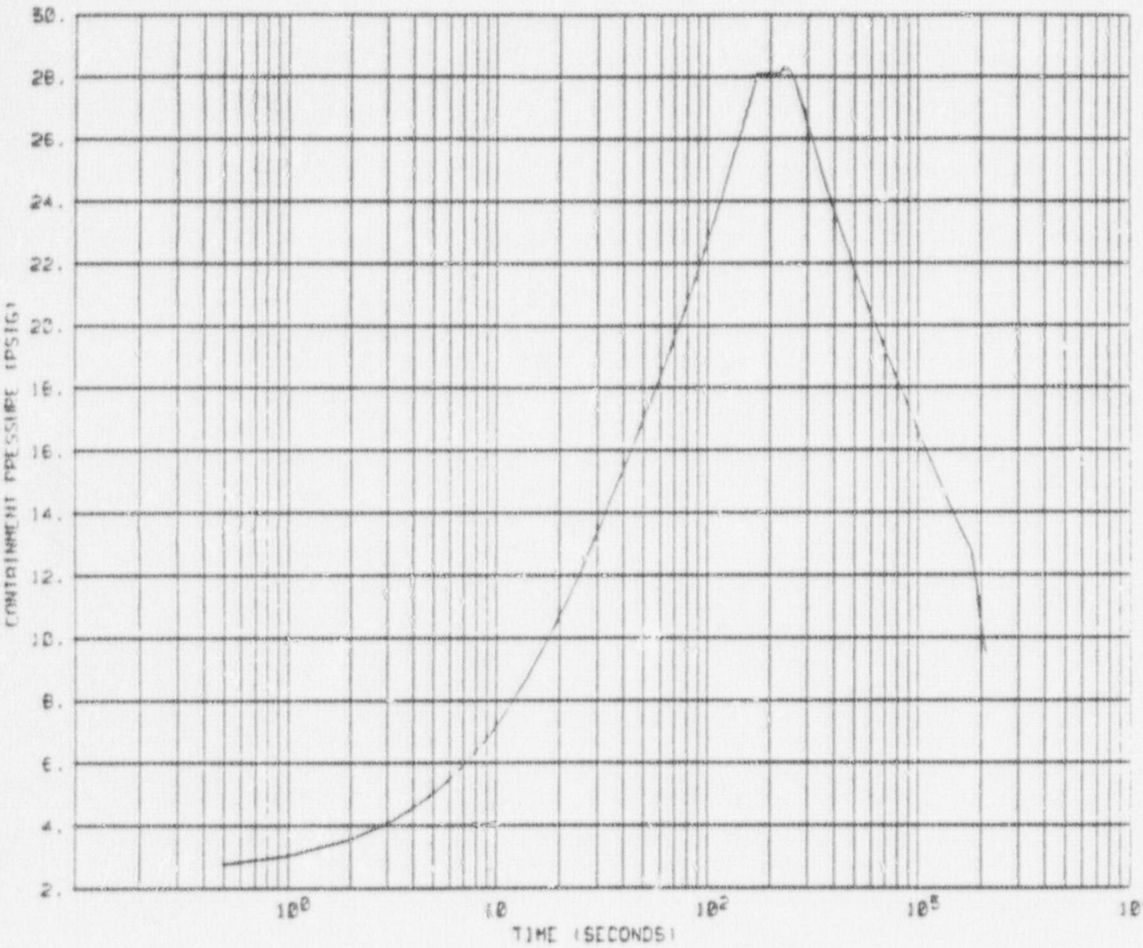

FIGURE 5-20

CONTAINMENT TEMPERATURE vs. TIME

100% POWER AFW FAILURE WITH OFFSITE POWER

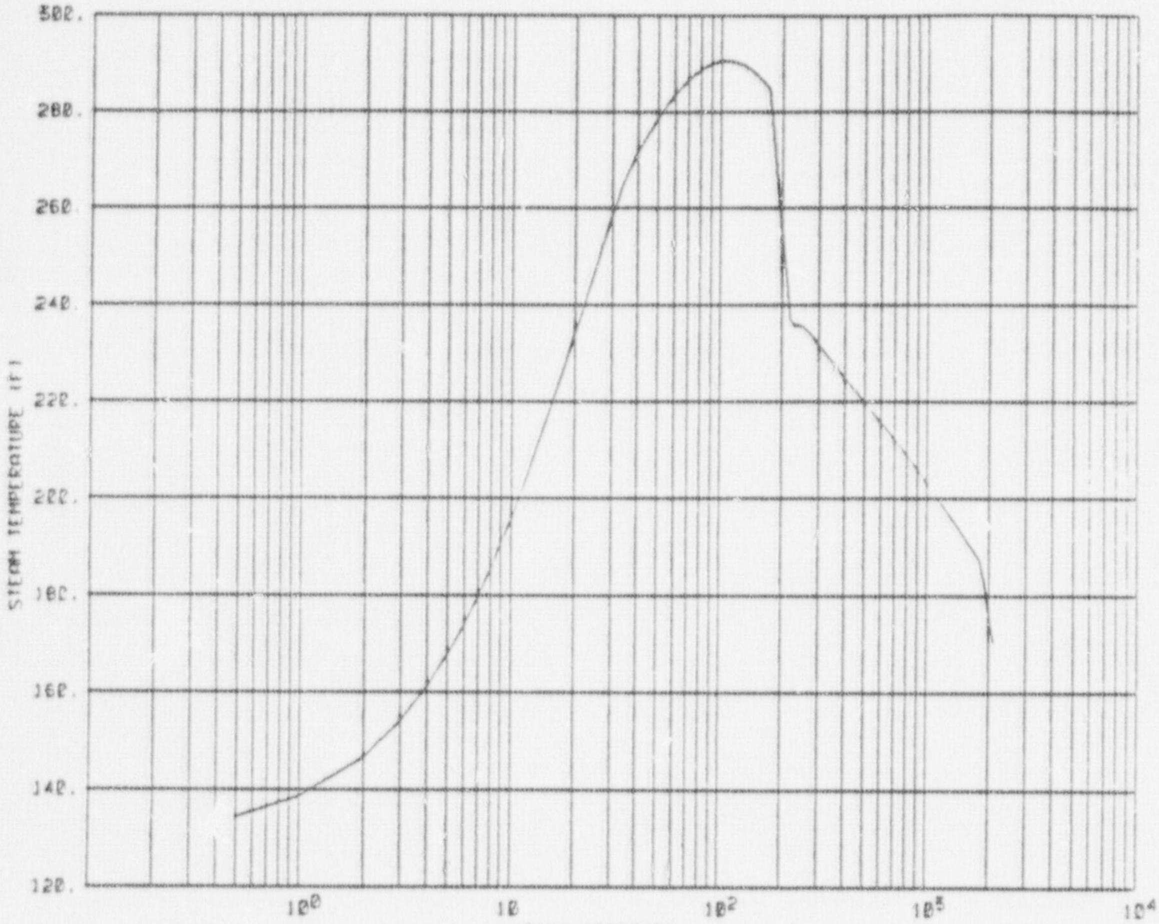

FIGURE 5-21
CONTAINMENT PRESSURE vs. TIME
70 % POWER FCV FAILURE WITH OFFSITE POWER
30 SECOND FIV CLOSURE

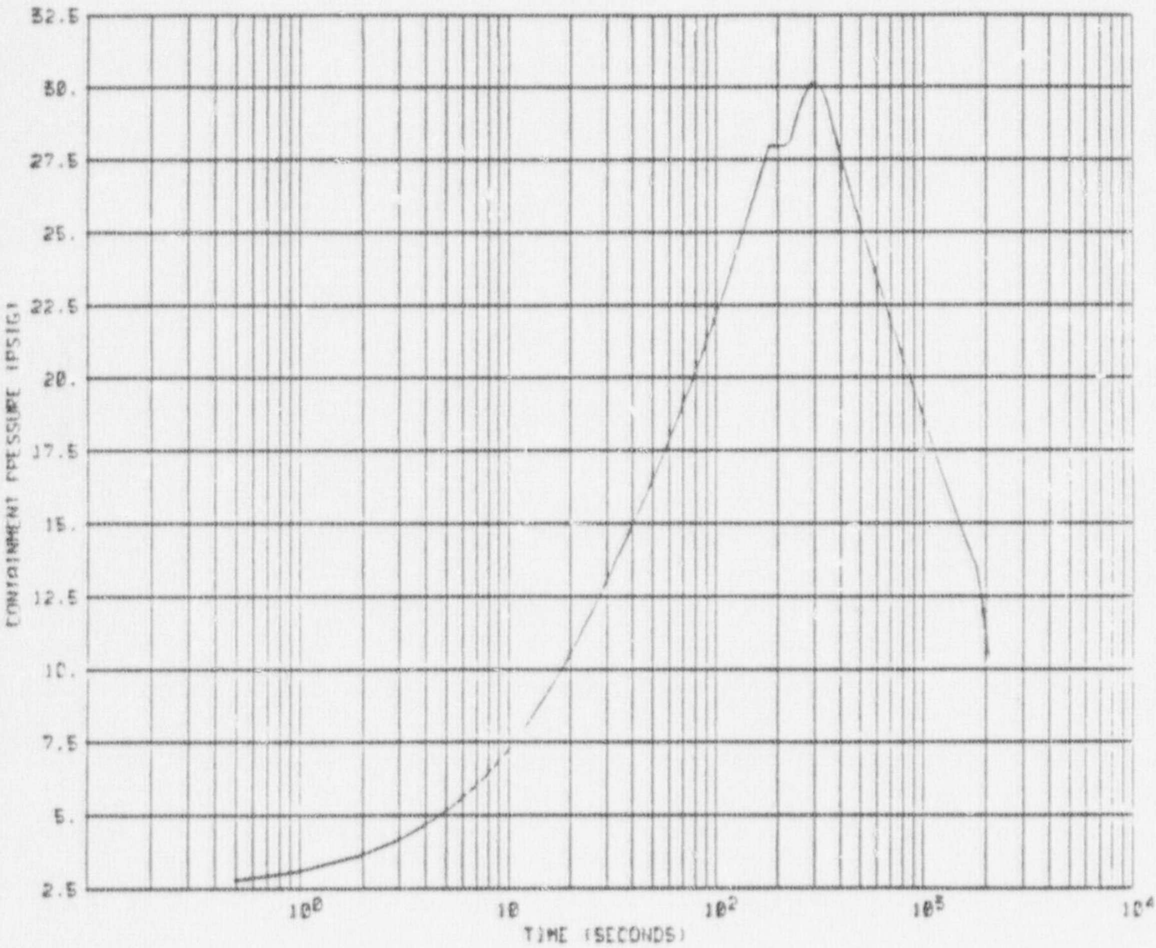

FIGURE 5-22

CONTAINMENT TEMPERATURE vs. TIME

70 % POWER FCV FAILURE WITH OFFSITE POWER

30 SECOND FIV CLOSURE

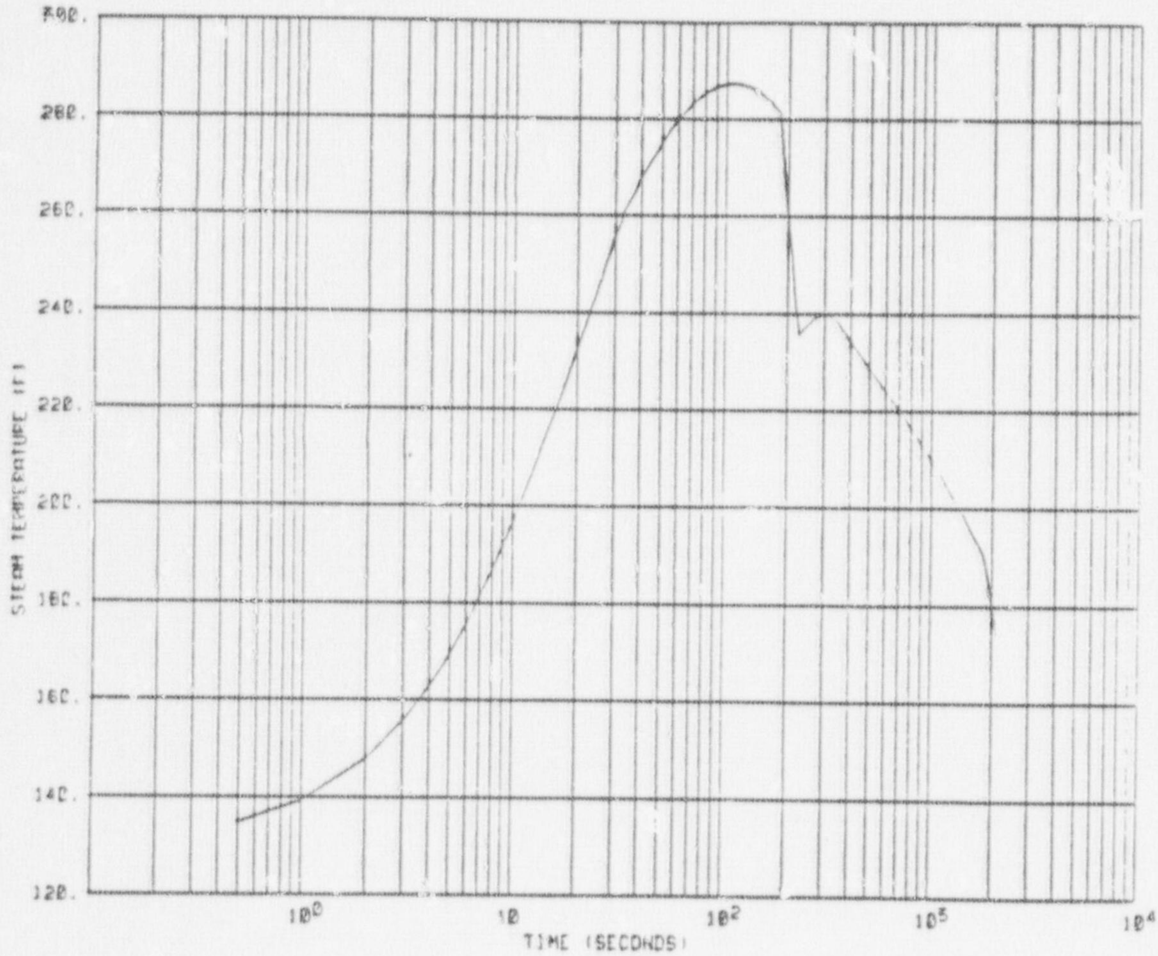

FIGURE 5-23

CONTAINMENT PRESSURE vs. TIME

70 % POWER FCV FAILURE WITH OFFSITE POWER

REDUCED FEED FLASHING VOLUME

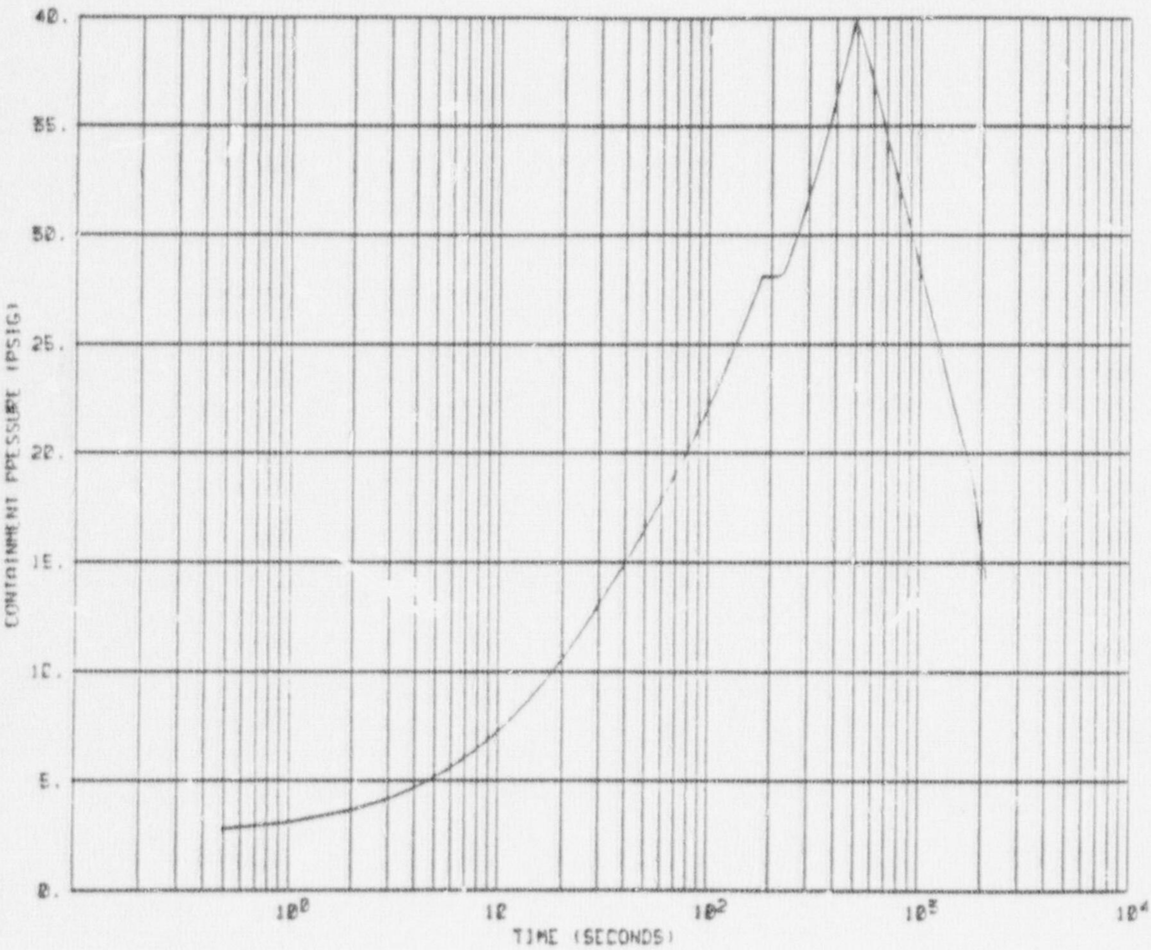

FIGURE 5-24

CONTAINMENT TEMPERATURE vs. TIME

70 % POWER FCV FAILURE WITH OFFSITE POWER

REDUCED FEED FLASHING VOLUME

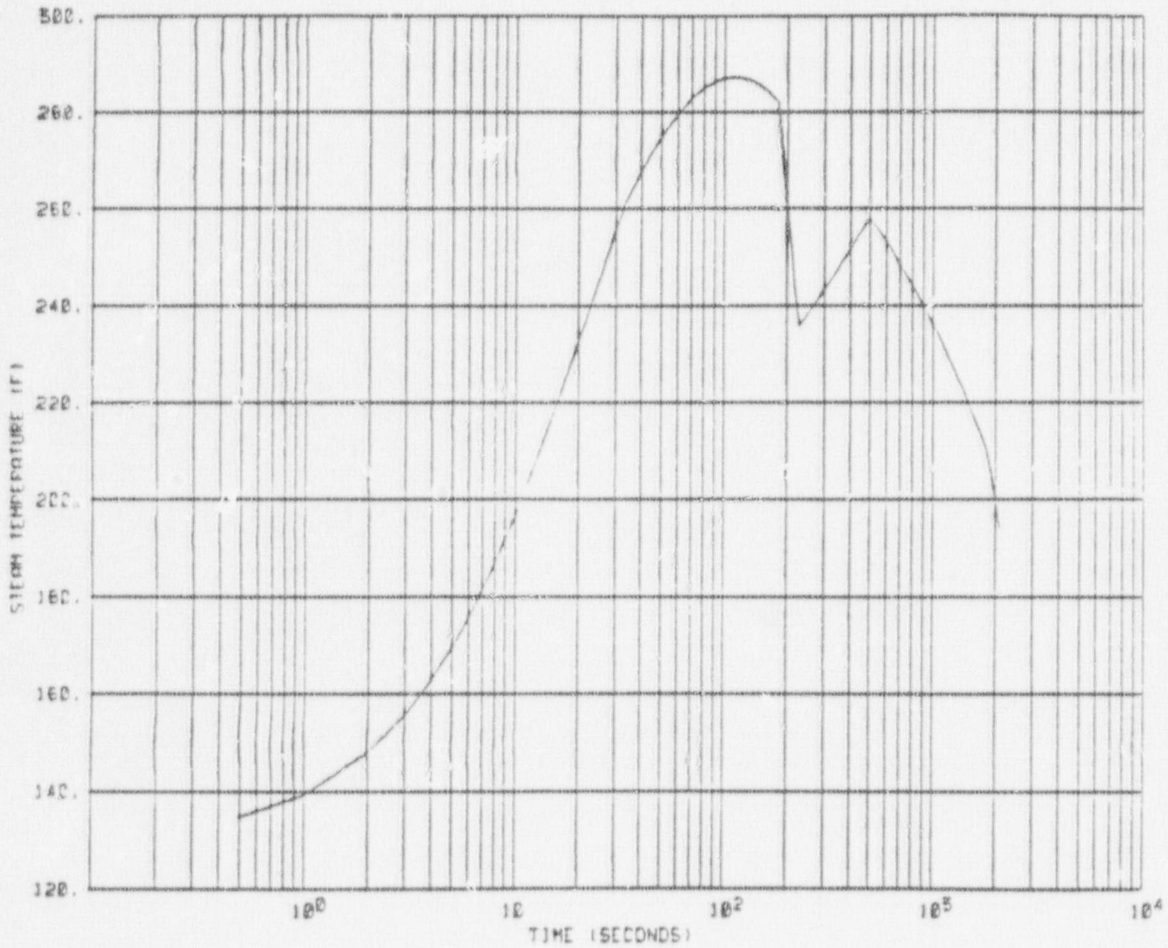

FIGURE 5-25

CONTAINMENT PRESSURE vs. TIME

70 % POWER FCV FAILURE WITH OFFSITE POWER

BIT IN

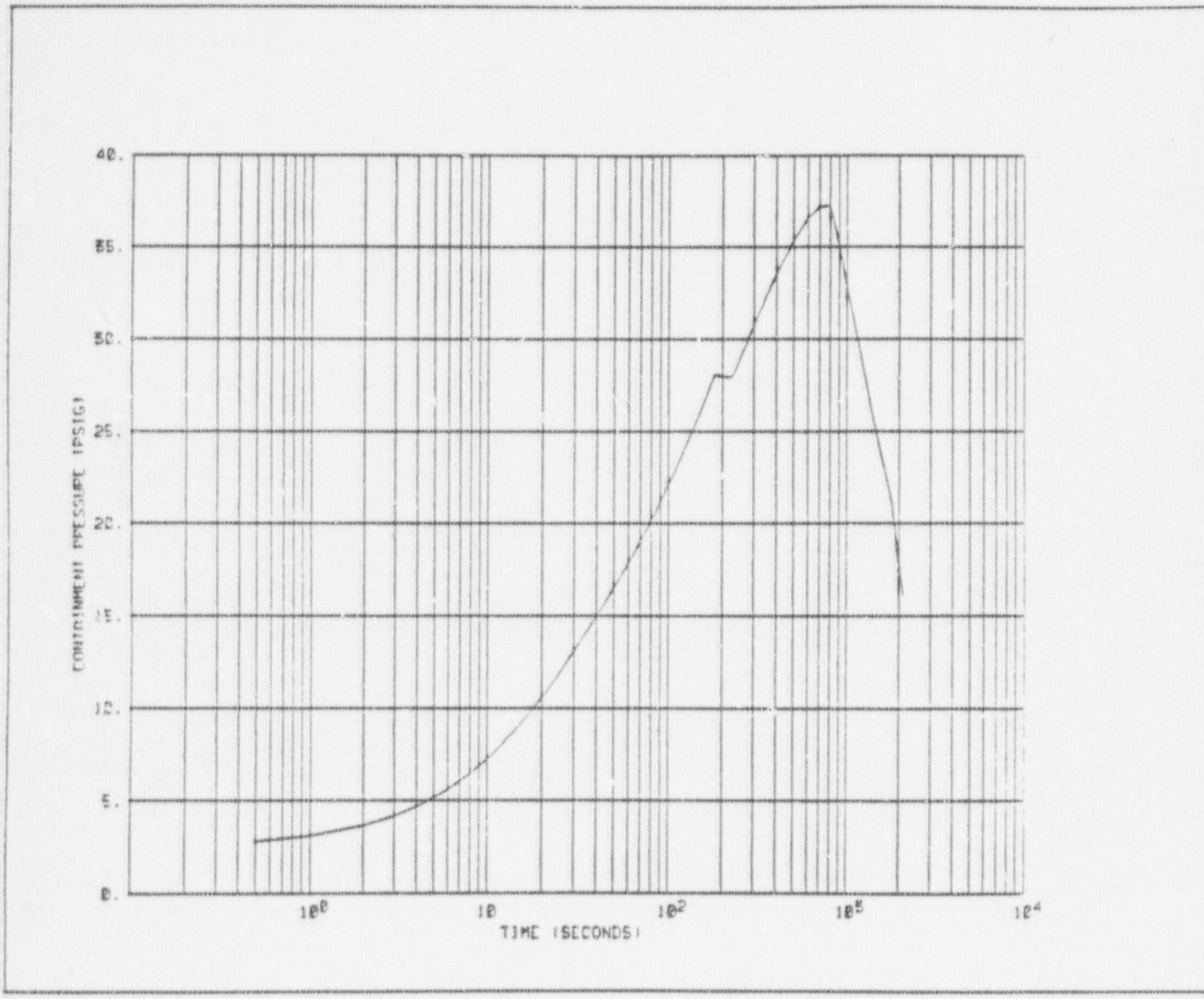

FIGURE 5-26

CONTAINMENT TEMPERATURE vs. TIME

70 % POWER FCV FAILURE WITH OFFSITE POWER

BIT IN

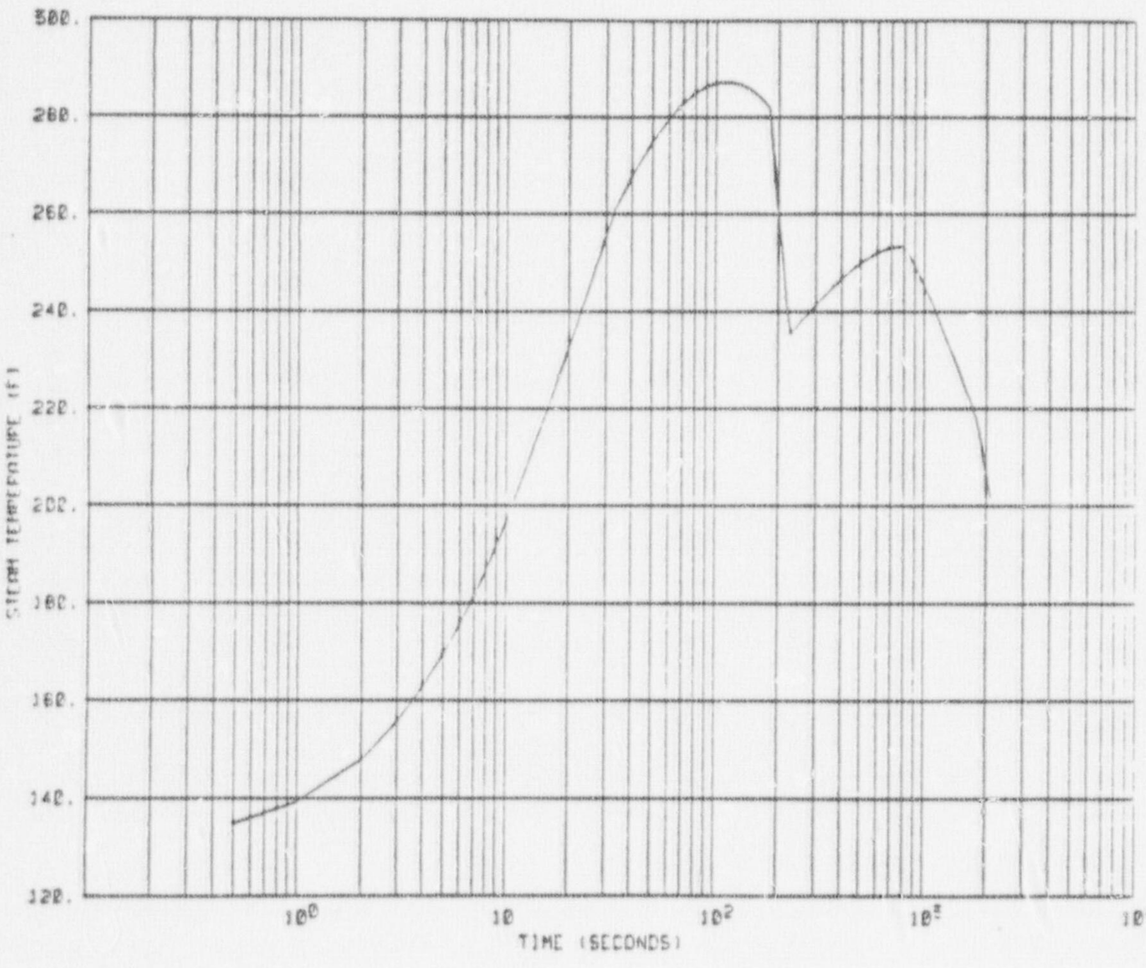

FIGURE 5-27

CONTAINMENT PRESSURE vs. TIME

70 % POWER FCV FAILURE WITH OFFSITE POWER

MINIMUM SI DELAY

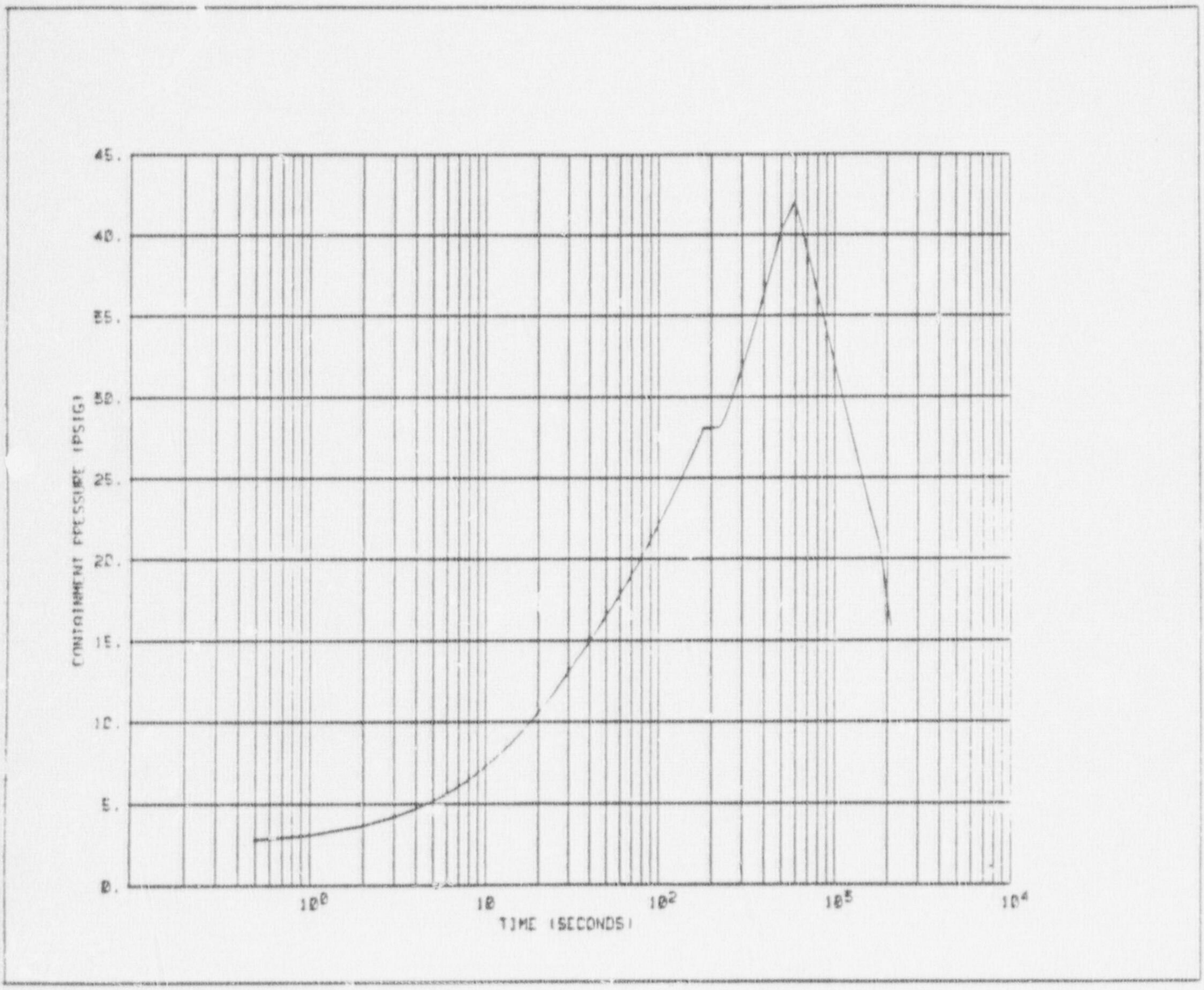

FIGURE 5-28

CONTAINMENT TEMPERATURE vs. TIME

70 % POWER FCV FAILURE WITH OFFSITE POWER

MINIMUM SI DELAY

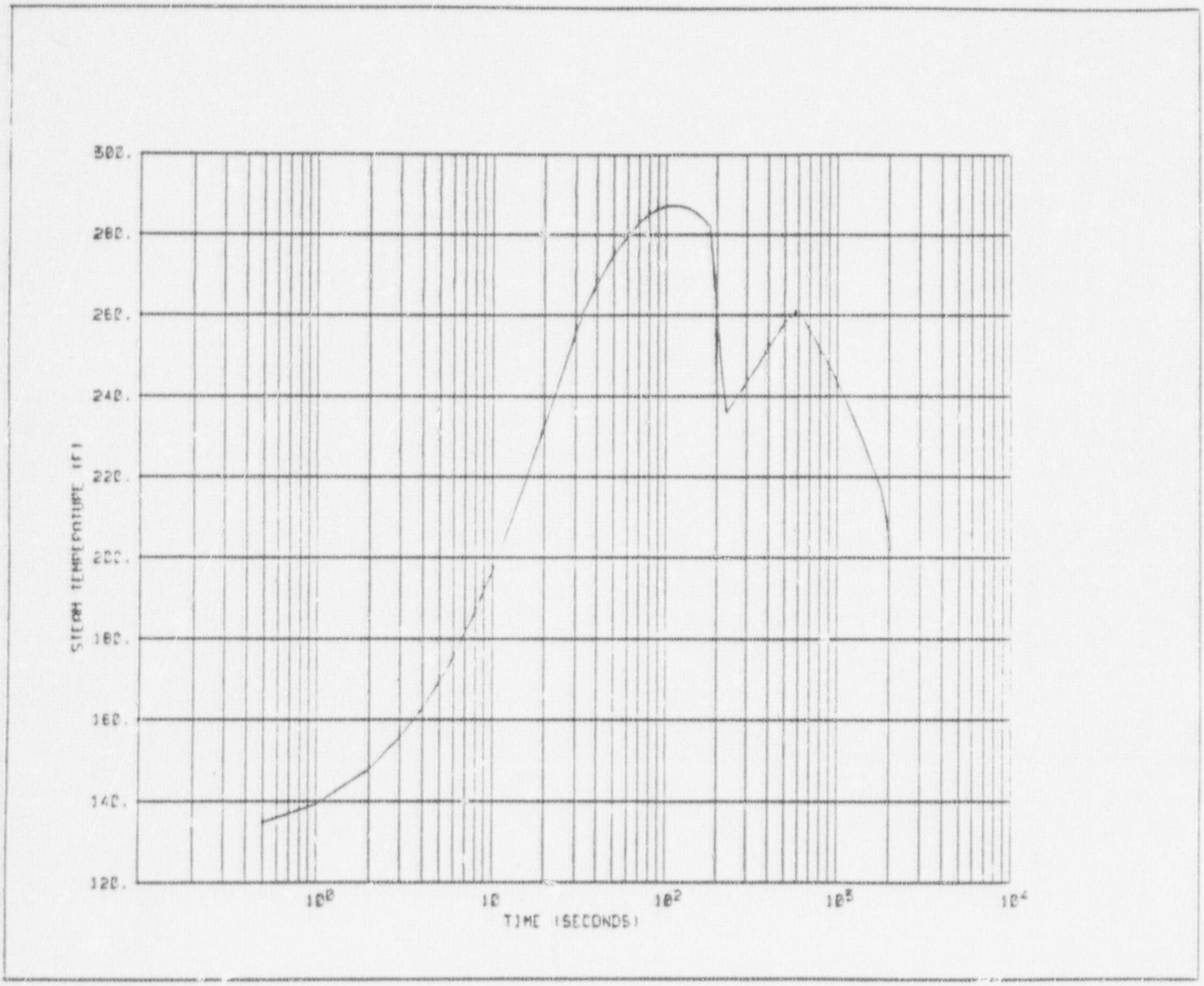

FIGURE 5-29
CONTAINMENT PRESSURE vs. TIME
70 % POWER FCV FAILURE WITH OFFSITE POWER
0.60 SQUARE FOOT BREAK

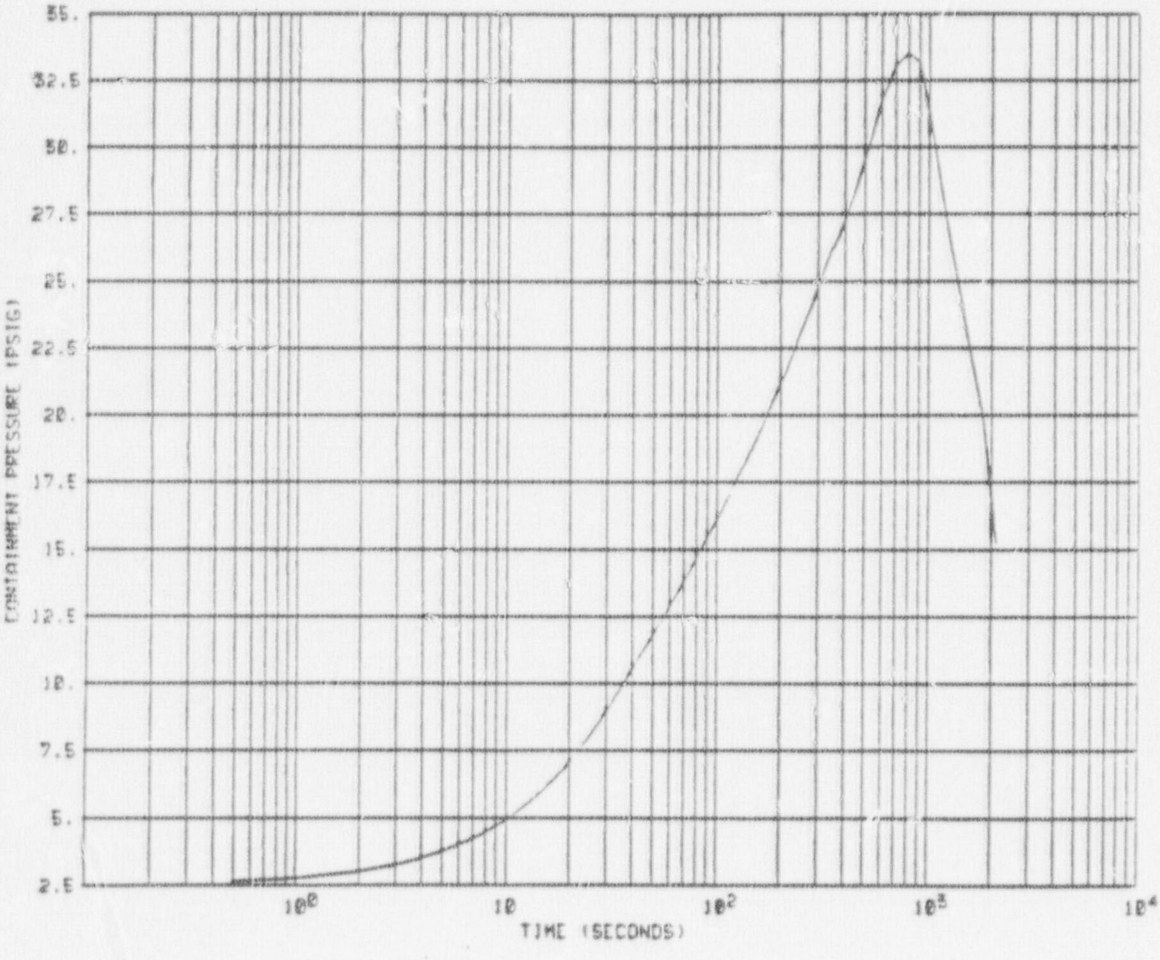

FIGURE 5-30
CONTAINMENT TEMPERATURE vs. TIME
70 % POWER FCV FAILURE WITH OFFSITE POWER
0.60 SQUARE FOOT BREAK

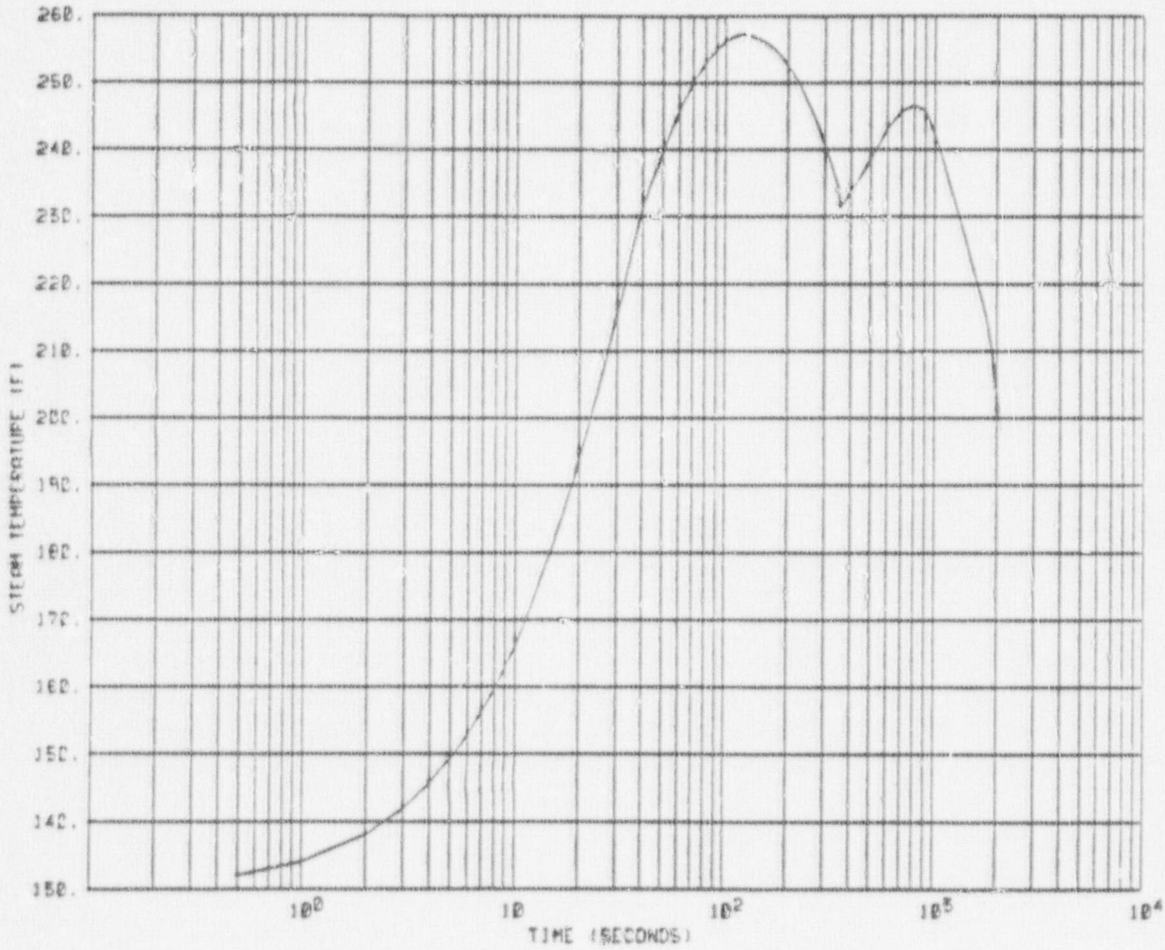

FIGURE 5-31

CONTAINMENT PRESSURE vs. TIME

0 % POWER MSCV FAILURE WITH OFFSITE POWER

CREDIT FOR 100% REVAPORIZATION

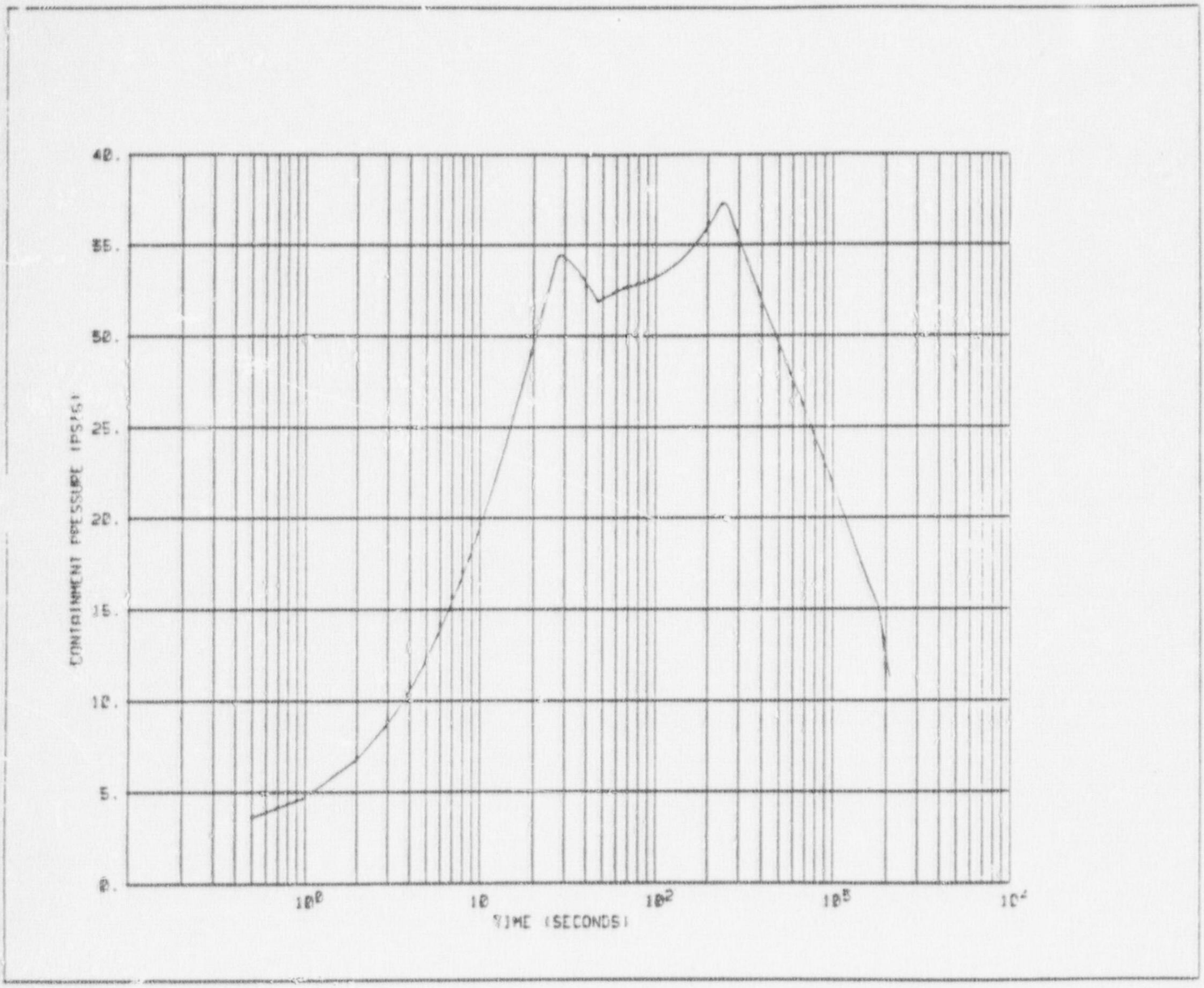

FIGURE 5-32

CONTAINMENT TEMPERATURE vs. TIME

0 % POWER MSIV FAILURE WITH OFFSITE POWER

CREDIT FOR 100% REVAPORIZATION

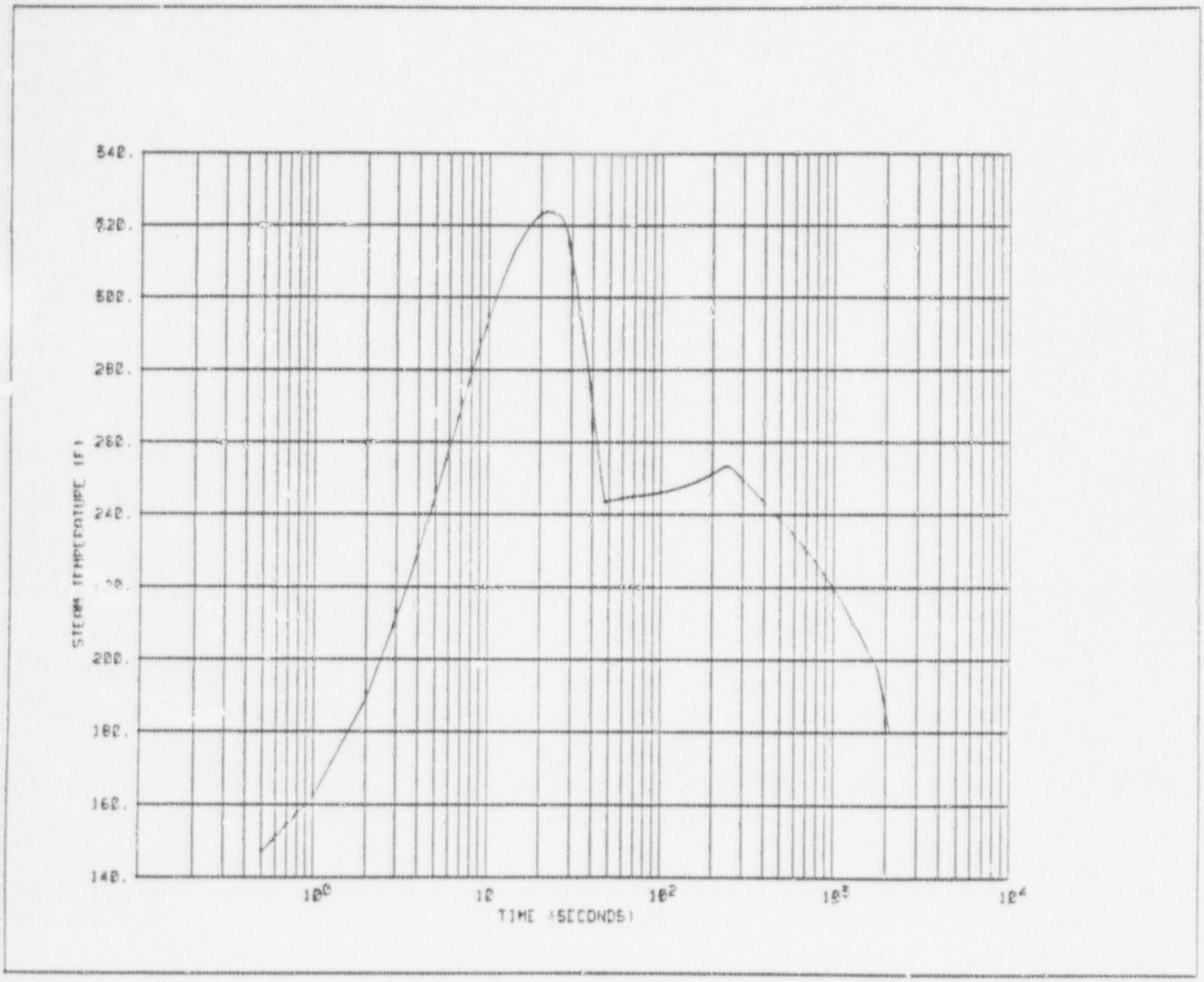

FIGURE 5-33

CONTAINMENT PRESSURE vs. TIME

0 % POWER MSCV FAILURE WITH OFFSITE POWER

CREDIT FOR TURBULENT TAGAMI & 100% REVAPORIZATION

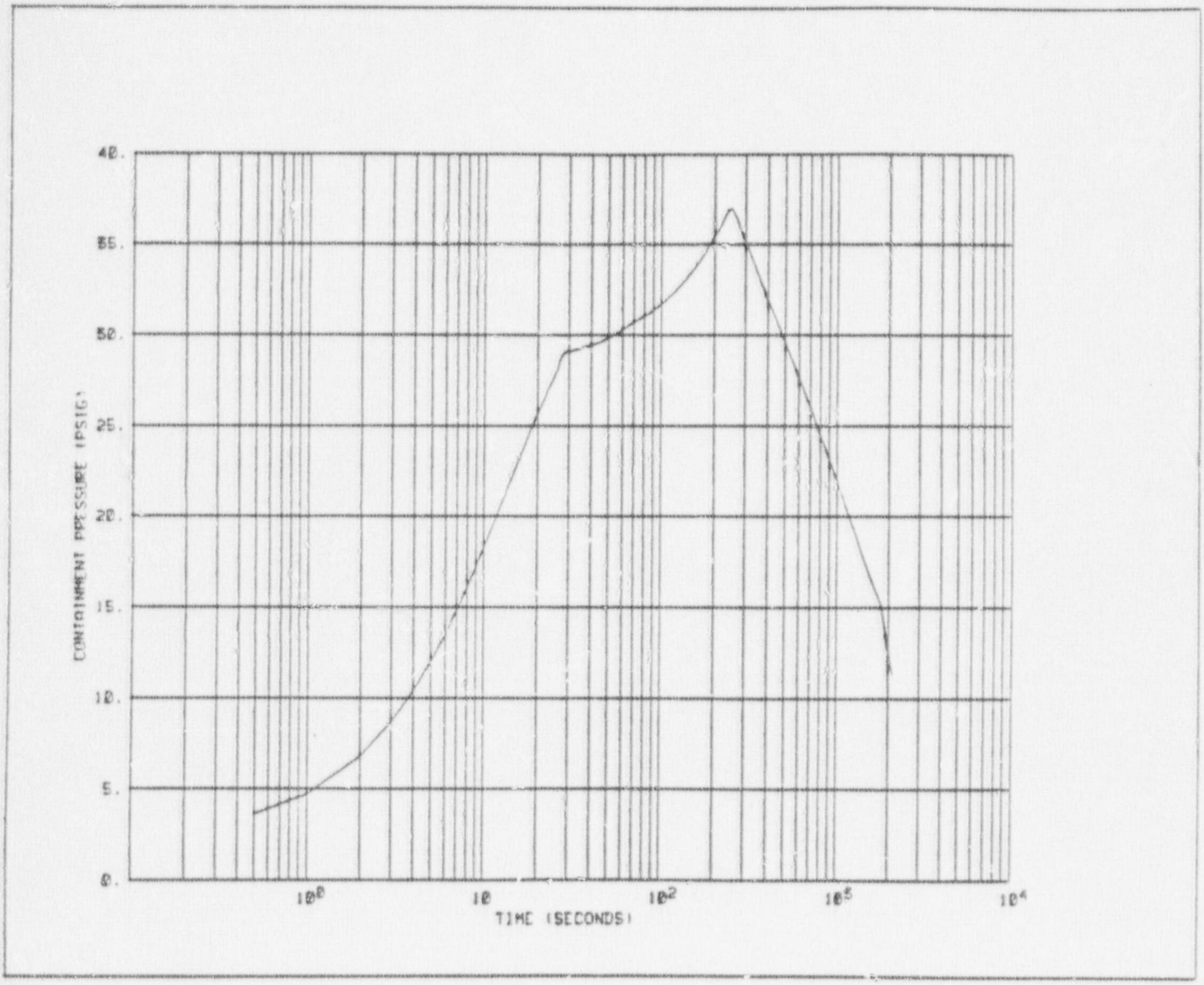

FIGURE 5-34

CONTAINMENT TEMPERATURE vs. TIME

0 % POWER MSCV FAILURE WITH OFFSITE POWER

CRED'T FOR TURBULENT TAGAMI & 100% REVAPORIZATION

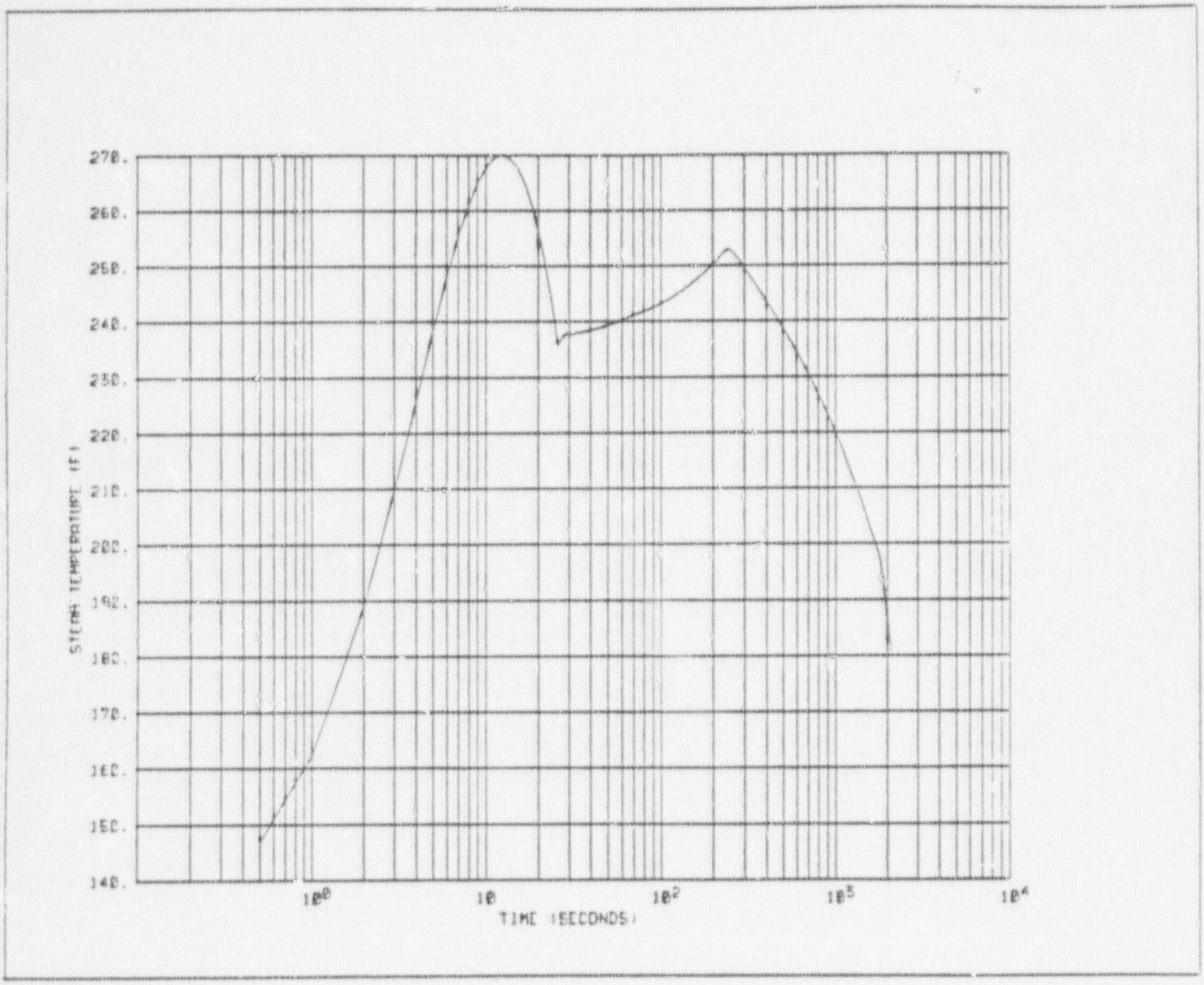

FIGURE 5-35
CONTAINMENT PRESSURE vs. TIME
100 % POWER DIESEL FAILURE WITHOUT OFFSITE POWER
CREDIT FOR TURBULENT TAGAMI & 100% REVAPORIZATION

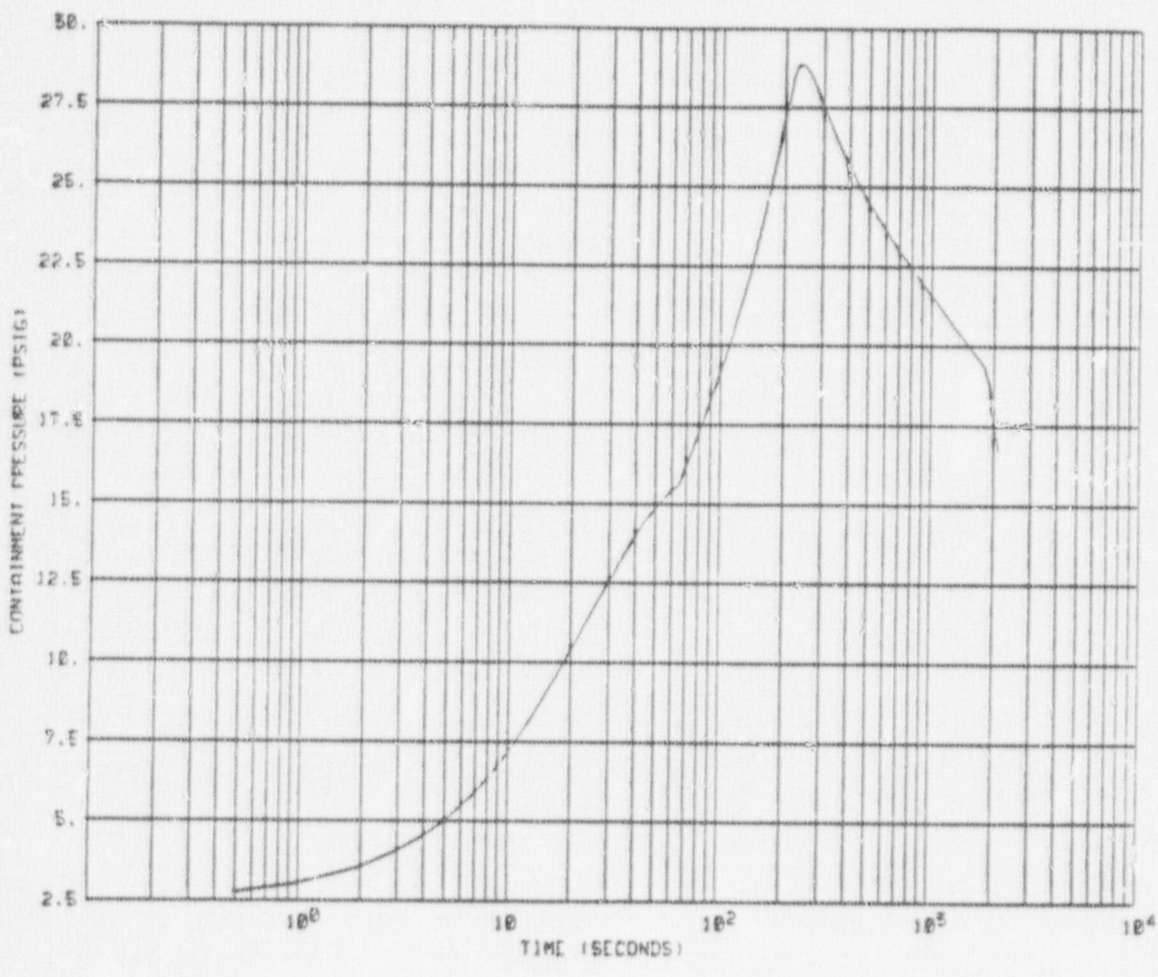

FIGURE 5-36

CONTAINMENT TEMPERATURE vs. TIME

100 % POWER DIESEL FAILURE WITHOUT OFFSITE POWER

CREDIT FOR TURBULENT TAGAMI & 100% REVAPORIZATION

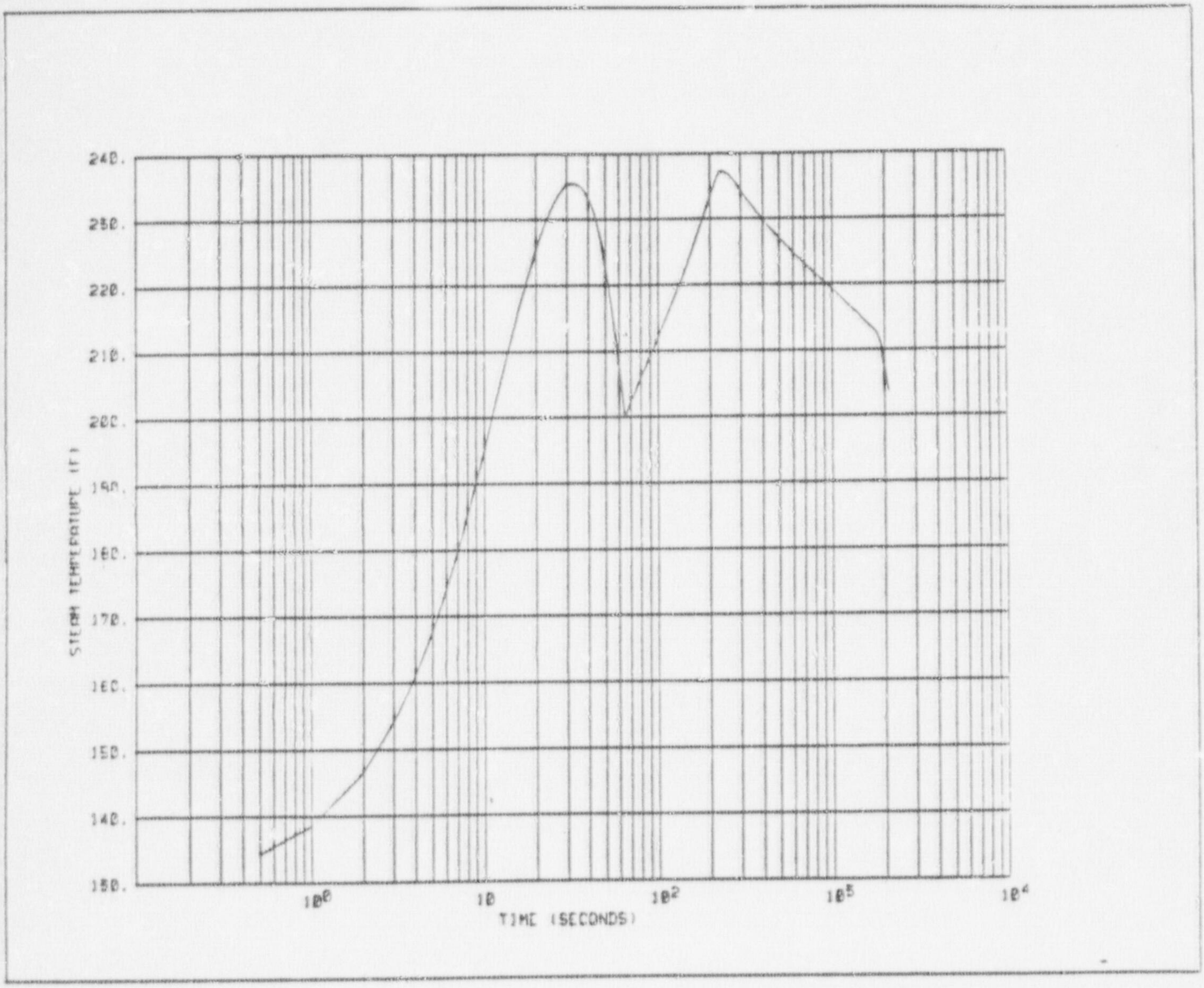

FIGURE 5-37
CONTAINMENT PRESSURE vs. TIME
70 % POWER FCV FAILURE WITH OFFSITE POWER
CREDIT FOR TURBULENT TAGAMI & 100% REVAPORIZATION

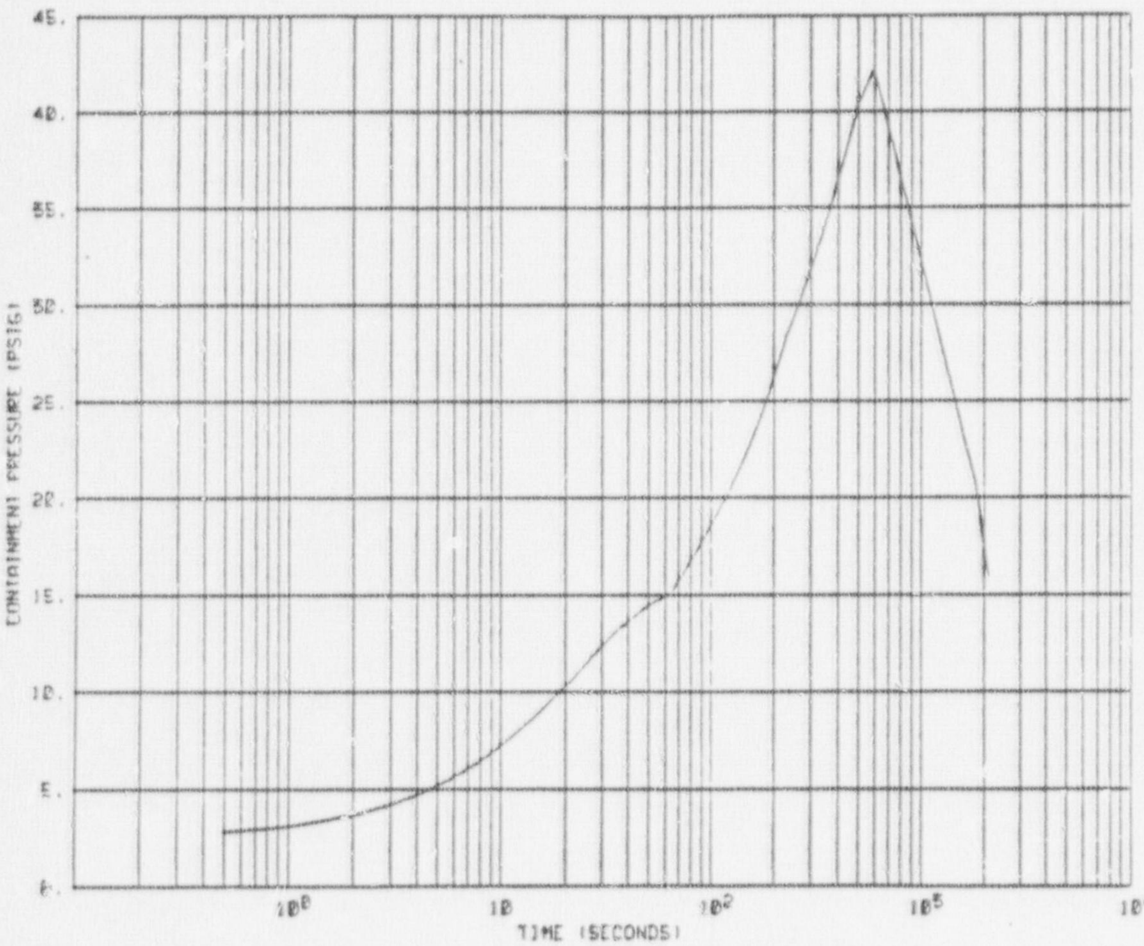

FIGURE 5-38

CONTAINMENT TEMPERATURE vs. TIME

70 % POWER FCV FAILURE WITH OFFSITE POWER

CREDIT FOR TURBULENT TAGAMI & 100% REVAPORIZATION

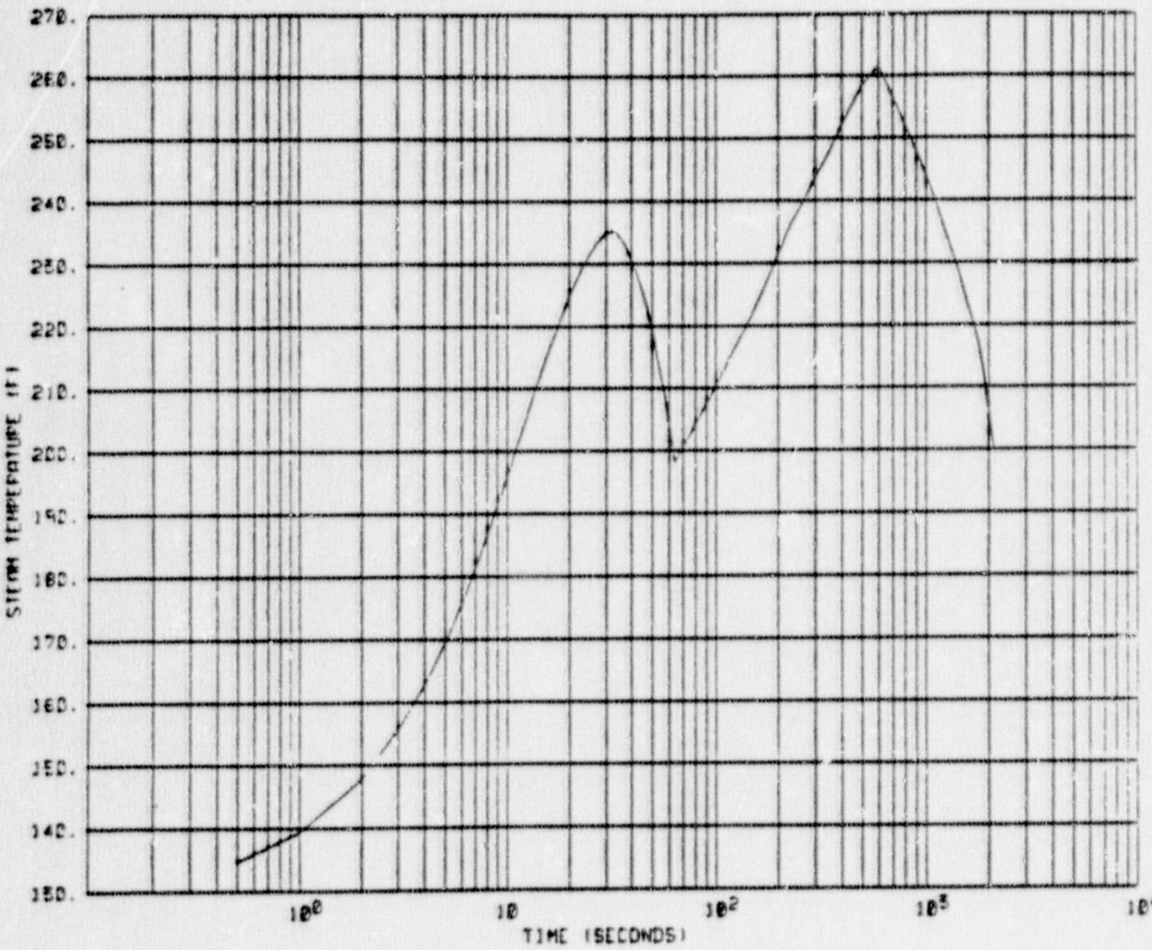

TABLE 5-1

SUMMARY OF MSLB RESULTS

CASE DESCRIPTION	PEAK PRESSURE (PSIG)	PEAK TEMPERATURE (°F)
MSCV FAILURE, 0% POWER	38.39	378.6
MSCV FAILURE, 70% POWER	38.32	377.0
MSCV FAILURE, 100% POWER	38.68	377.7
DIESEL FAILURE, 0% POWER	29.60	301.6
DIESEL FAILURE, 100% POWER	31.12	308.1
FCV FAILURE, 0% POWER	38.75	269.7
FCV FAILURE, 30% POWER	38.67	281.0
FCV FAILURE, 70% POWER	42.28	287.3
FCV FAILURE, 100% POWER	41.98	293.8
AFW FAILURE, 100% POWER	28.29	290.6

TABLE 5-2

SUMMARY OF CONTAINMENT PRESSURE SENSITIVITIES

CASE DESCRIPTION	PEAK PRESSURE (PSIG)	PEAK TEMPERATURE (F)
FCV FAIL., 70% POWER, 30 sec FIV closure	30.17	287.3
FCV FAIL., 70% POWER, reduced flash volume	39.94	287.2
FCV FAIL., 70% POWER, BIT in	37.37	287.3
FCV FAIL., 70% POWER, minimum SI delay	42.28	287.3
FCV FAIL., 70% POWER, 0.60 ft2 break	33.55	257.2
MSCV FAIL., 0% POWER, Revap.	37.31	323.8
MSCV FAIL., 0% POWER, Turb.Tagami & Revap.	36.99	269.9
Diesel FAIL., 100% POWER, Turb.Tagami & Revap.	28.87	237.3
FCV FAIL., 70% POWER, Turb.Tagami & Revap.	42.42	261.5

6.0 CONCLUSIONS

Containment Integrity Analyses have been performed as part of the Indian Point Unit 3 Containment Margin Improvement Program. The objective of the program was to provide containment integrity analysis results using the current Indian Point Unit 3 specific information and new more realistic LOCA models. In this way the licensing basis for Unit 3 is clarified and updated, and the maximum pressure margin for operation of Unit 3 can be determined and thus made available for possible future use.

The results of the analysis ensured that the pressure inside containment will remain below the containment building design pressure if a Loss-of-Coolant- Accident (LOCA) or a rupture of main steamline pipe (MSLB) inside containment should occur during plant operation. The peak pressure calculated for the limiting LOCA and MSLB events respectively are 40.3 psig and 42.42 psig. The design pressure is 47 psig.

The purpose of the analysis and evaluation described herein is to show that sufficient margin exists in containment pressure for the LOCA and MSLB events. Because of the exclusion of enhanced heat transfer phenomena higher than anticipated transient temperatures were observed for the MSLB cases. Whenever assumptions consistent with the current MSLB design basis were made it was determined that the peak calculated temperature is below the present EQ peak of 287 degrees Fahrenheit. Furthermore, based upon NUREG-0458 and detailed analyses performed on a similar design it is judged that LOCA produces the most severe environmental conditions for equipment, and the LOCA results are below the equipment qualification limits.

7.0 REFERENCES

1. WCAP-8264-P-A, Rev.1, "Topical Report Westinghouse Mass And Energy Release Data For Containment Design", August 1975.

2. WCAP-8327, "Containment Pressure Analysis Code (COCO)", July 1974.

3. Westinghouse Letter # NS-OPLS-OPLI-86138, 4/30/86.

4. Westinghouse Letter # INT-87-760, 7/9/87.

5. Westinghouse Letter # INT-87-761, 7/9/87.

6. Westinghouse letter # INT-88-650, Rev. 1, 6/20/88.

7. WCAP-10325-A, "Westinghouse LOCA Mass And Energy Release Model For Containment Design - March 1979 Version", May 1983.

8. Westinghouse Letter # INT-88-641, 5/27/88.

9. WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary) "LOFTRAN Code Description", April 1984.

10. "Westinghouse ECCS Evaluation Model - 1981 Version," WCAP-9220-P-A, Rev. 1, February 1982 (Proprietary), WCAP-9221-A, Rev. 1 (Non-Proprietary).

11. EPRI 294-2, Mixing of Emergency Core Cooling Water with Steam: 1/3 Scale Test and Summary, (WCAP-8423), Final Report June 1975.

12. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors", August 1979.

13. Takashi Tagami, "Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965," No. 1.

14. E. W. Ranz and W. R. Marshall, Jr., "Evaporation for Drops," Chemical Engineering Progress, 48, pp. 141-146, March 1952.

15. L. F. Parsly, "Spray Tests at the Nuclear Safety Pilot Plant," Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1970, ORNL-4647, 1971, p.82.

16. WCAP-8822-S1-P-A, "Mass and Energy Releases following A Steamline Rupture," R. E. Land, September, 1980.

17. NUREG-0458, "Short Term Safety Assessment On The Environmental Qualification Of Safety-Related Electrical Equipment Of SEP Operating Plants", May 1978.

18. United Engineers & Constructors Inc. letter # IUP-8265, 12/1/88.

19. Final Safety Analysis Report Indian Point 3 Nuclear Power Plant, Section 14.3.6.3, Containment Pressure Calculation.

20. IPN-84-63, Corbin A. McNeill to Steven A. Varga, "Indian Point 3 Nuclear Power Plant Docket No. 50-286, Indian Point 3 Environmental Qualification Program," December 27, 1984.