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R. E. Ginna Boric Acid Storage Tank Boron Concentration Reduction Study

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1.0 INTRODUCTION

This report presents analyses of the R. E. Ginna plant steamline break (SLB) containment integrity and associated LOCA-related analyses, with a reduction of the boron concentration in the Boric Acid Storage Tanks (BASTs) from 20,000 ppm to 2,000 ppm. A boron reduction to this level will allow the removal of credit for the BASTs from the licensing basis accident analyses (and subsequently removal of the associated heat tracing required). The BASTs will be retained for operation requirements and redundant flow paths as discussed in Technical Specifications.

R. E. Ginna currently must maintain 20,000 ppm boron in the BASTs, which requires heat tracing to prevent boron precipitation. The BASTs and their heat tracing are part of the Safety Injection (SI) system and thus they must be maintained according to requirements which can impose operational restrictions. The only accident analyses which are significantly affected by boron concentration reduction are the secondary side steamline break transients. The core and the containment responses are affected by the steamline break transients and therefore were considered in the boron concentration reduction analysis.

2.0 CORE RESPONSE

The SLB core response analysis is documented in Reference 1 and supports a reduction in the BASTs boron concentration to 2000 ppm.

3.0 CONTAINMENT INTEGRITY ANALYSIS

3.1 Purpose

The purpose of the Containment Integrity Steamline Break analysis is to demonstrate the acceptability of the Containment Safeguards Systems to mitigate the consequences of a hypothetical rupture of a steamline pipe. The impact of steamline mass and energy releases on containment pressure is addressed to ensure the containment pressure remains below its design pressure of 60 psig at the reduced boron concentration conditions.

3.2 Relevant Acceptance Criteria

The BASTs are components of the Safety Injection System designed to mitigate the consequences of postulated steamline break accidents by providing a high concentration of boric acid to the reactor coolant. A high concentration of boric acid causes a decrease in the post trip return core power level and subsequently a decrease in heat transferred to the secondary side fluid, which results in decreased containment pressures during a SLB. The containment pressures resulting from the mass and energy releases must remain below the design pressure of the containment building. For R. E. Ginna the containment design pressure is 60 psig.

3.3 Evaluation

3.3.1 Methodology

Calculation of the steamline break containment response is a two step process. The LOFTRAN computer code (Reference 2) is first used to calculate the mass and energy released as a function of time. The releases are then used as input to the COCO code (Reference 3) to calculate containment pressures and temperatures as a function of time. Attachment 1 provides a brief description of the LOFTRAN and COCO codes.

The cases that were analyzed for peak containment pressures are listed in Table 1. The basic initial conditions, heat sink model, fan cooler data, and containment spray parameters for these cases are outlined in Tables 2 through 5. The following conservative assumptions are made for the mass and energy release analysis:

1. Maximum decay heat equivalent to the 1979 ANS decay heat $+2\sigma$ uncertainty.
2. No credit for water entrainment in the blowdown results.
3. Conservatively high values for reverse steam generator heat transfer.
4. Conservative moderator temperature coefficient for the rodded core at end-of-life.

3.3.2 Spectrum of Break Assumptions

A comprehensive set of break sizes, initial power levels, single failure assumptions, and off-site power availability must be considered so that there is reasonable assurance that the limiting cases have been covered. The complete set of steamline break cases that were addressed for the R.E. Ginna plant is listed in Table 1.

The single failures considered in this analysis have been selected based upon their potential for increasing the amount of mass and energy released into containment or for reducing the amount of heat removed from containment. The four postulated failures are as follows:

- Failure of the Main Steam Isolation Valve (MSIV) to close
- Failure of the Feedwater Control System (FCS)
- Failure of one containment spray pump (to operate)
- Failure of one diesel generator to start

The breaks considered include 4.37 ft^2 Double Ended Ruptures (DER) upstream of the flow restrictor, 1.4 ft^2 DER's downstream of the flow restrictor, and small breaks of 1.1 ft^2 or smaller. To determine the limiting break size for the small breaks, several cases were run with break sizes from 0.3 ft^2 to 1.1 ft^2 in 0.2 ft^2 increments. After it had been sufficiently demonstrated that the two largest of the small breaks consistently resulted in higher break flows and limiting peak containment pressures, the remainder of the small break cases were run with only the two largest break sizes, 0.9 ft^2 and 1.1 ft^2.

3.3.3 Consistent Off-Site Power Availability

One of the conservative assumptions that has historically been made is with respect to the availability of off-site power. Under typical SLB containment analysis methodology, the mass and energy releases are generated assuming off-site power continues to be available for the duration of the transient. This gives maximum primary-to-secondary heat transfer because of the forced reactor coolant flow from the Reactor Coolant Pumps (RCPs). The containment integrity calculation is then performed assuming that off-site power is not available, which extends the safeguards equipment startup delays due to diesel sequencing timing. These two assumptions contradict each other, but result in an analysis which bounds both with

and without off-site power with one case. To remove this unnecessary conservatism the limiting cases were analyzed with a consistent off-site power availability assumption. A small number of cases were analyzed with inconsistent assumptions and demonstrated a high margin to the pressure limit.

3.3.4 Mass and Energy Calculation Assumptions

3.3.4.1 Main and Auxiliary Feedwater Flow as a Function of Steam Generator Pressure

The cases presented in this study assumed a main feedwater flow rate as a function of both the steam generator pressure and the feedwater control valve position. The feedwater control valve (FCV) position varies with power level and postulated break location. The break location affects the FCV position in that a steamline break results in an increase in steam flow and subsequently a steam flow/feed flow mismatch. In response to the mismatch, the feedwater control system is assumed to increase feed flow to match steam flow. The typical analysis assumption is to assume that the faulted loop FCV is wide open.

However, with a break upstream of the steamline flow restrictor, no steam flow/feed flow mismatch would be present. As such, for cases with a break size larger than the flow area through the flow restrictor, it is assumed that no mismatch signal is present and that the faulted loop FCV stays in its nominal pre-break position. The intact loop FCV is conservatively assumed to remain in its nominal pre-break position until reactor trip. A turbine trip is assumed to occur at the same time as reactor trip and the intact loop FCV is assumed to close instantly in response to the decrease in steam demand.

For steamline breaks located downstream of the flow restrictors and those breaks having a break area smaller than the flow restrictor, it is assumed that the FCV on the faulted loop goes wide open in response to the increased steam flow. As with the upstream breaks, the intact loop FCV is assumed to be in its nominal position initially and closes instantly, coincident with reactor trip.

Auxiliary feedwater flow rates as a function of steam generator pressure were also assumed in the analyses. Auxiliary feedwater flow rates varied depending on the availability of offsite power and the single failure being evaluated.

At HZP the main feedwater pumps will not deliver feedwater to either steam generator. Thus, none of

the zero power cases assume any main feedwater. These cases assume auxiliary feedwater only, initiated at the time the steamline break occurs.

3.3.4.2 Core Reactivity Coefficients

LOFTRAN utilizes a point kinetics model, which uses reactivity feedback coefficients to calculate the kinetics conditions in the core. Steamline break transients initialized at hot zero power assume rodded reactivity feedback coefficients with an allowance for the most reactive Rod Cluster Control Assembly (RCCA) stuck in its fully withdrawn position. Steamline break transients initiated with the reactor at power typically assume End-Of-Life (EOL) reactivity coefficients calculated assuming that all RCCAs are fully withdrawn. However, for these analyses, since the majority of the transient is post reactor trip, rodded coefficients (again with an allowance for a stuck RCCA) were assumed. Confirmation of the conservatism of the overall reactivity model has been obtained by more detailed core neutronics calculations.

3.3.5 Containment Integrity Assumptions

The major containment integrity calculational assumptions used with COCO are as follows:

1. The mass and energy release to the containment is for a break opening time of zero.

2. The saturation temperature corresponding to the partial pressure of the containment vapor is used in calculating the condensing heat transfer to the passive heat sinks and the heat removal by containment fan coolers.

3. The Westinghouse containment model utilizes the analytical approaches described in References 3 and 4 to calculate the condensate removal from the condensate film. A convective heat flux revaporization model is used for small breaks. 100% revaporization is assumed for large breaks.

4. The small steamline break containment analyses utilized the stagnant Tagami correlation, Reference 5.

5. The diesel failure conditions (minimum safeguards), that were modeled, assumed that there were 2 fan coolers and one containment spray pump (1300 gpm) were operating. The time delays that were assumed for initiation of containment sprays and fan coolers with a diesel failure are given in Table 3.

3.4 Design-Basis Containment Integrity Analysis Results.

Figures 1 and 2 provide the pressure and temperature transient curves for the 4.37 ft^2 DER upstream of the flow restrictor case producing the highest peak containment pressure of this type of break and all other breaks analyzed. This case represents a main steam isolation valve (MSIV) failure at 30% power with offsite power available. The BASTs boron concentration of 2000 ppm was assumed in this case and all other cases identified in Table 1. The mass and energy releases for this case are shown in Figures 3 and 4.

The limiting 1.4 ft^2 downstream DER containment pressure and temperature transients are shown in Figures 5 and 6. This case represents the feedwater control system failure at 70% power without offsite power available. Note that the peak pressure is lower for the 1.4 ft^2 break than for the 4.37 ft^2 break. The smaller break area reduces the blowdown mass and energy release rate without significantly delaying actuation of protective functions and, therefore, results in a lower peak containment pressure than the 4.37 ft^2 case. The mass and energy release rates for this case are included in Figures 7 and 8.

The limiting small DER is a 1.1 ft^2 break, resulting in the pressure and temperature transients shown in Figures 9 and 10. This case was analyzed assuming a diesel failure at 102% power, without offsite power available. The mass and energy release rates for this case are included in Figures 11 and 12.

The containment pressures reached by the limiting breaks with the boric acid storage tank concentration of 2000 ppm remain below the containment design limit of 60 psig.

4.0 EVALUATION OF LOCA-RELATED ANALYSES

4.1 Large Break LOCA

The current Large Break Loss-Of-Coolant Accident (LBLOCA) analysis of record for R. E. Ginna was performed using the NRC-approved 1981 ECCS Evaluation Model, Reference 6.

The proposed reduction in the boron concentration in the BASTs will not adversely affect the Large Break LOCA because the Evaluation Model codes used in analyzing the large break do not explicitly model boron concentration in the reactor coolant system.

4.2 Small Break LOCA

The current Small Break Loss-Of-Coolant Accident (SBLOCA) analysis of record for R. E. Ginna was performed using the NRC-approved Small Break LOCA ECCS Evaluation Model with WFLASH, Reference 7.

The proposed reduction in the boron concentration in the BASTs will not adversely affect the Small Break LOCA because the Evaluation Model codes used in analyzing the small break do not explicitly model boron concentration in the reactor coolant system.

4.3 Post-LOCA Long Term Core Cooling Subcriticality Requirement

The Westinghouse licensing position for satisfying the requirements of 10CFR 50.46 Paragraph (b) Item (5) "Long Term cooling" is defined in WCAP-8339, Reference 8. The Westinghouse commitment is that the reactor will remain shutdown by borated ECCS water residing in the sump following a LOCA, Reference 9. Since credit for the control rods is not taken for large break LOCA, the borated ECCS water provided by the accumulators and the RWST must have a concentration that, when mixed with other sources of borated and non-borated water, will result in the reactor core remaining subcritical assuming all control rods are out. The large reduction in boron concentration in the BASTs will have a significant effect on the Reactor Coolant System boron concentrations assumed for this calculation.

The calculations for determining whether the reduction in the boron concentration in the BASTs will

result in the core remaining subcritical was re-done with a new concentration of 2000 ppm in the BASTs. A new RCS boron concentration curve for the 2000 ppm value was generated and used in the core design process to ensure that the core will remain subcritical with a boron concentration of 2000 ppm.

4.4 Boron Precipitation During Long Term Cooling

The post-LOCA boron precipitation long term core cooling requirement ensures no boron precipitation in the reactor vessel following boiling in the core. Since Ginna has simultaneous injection from the residual heat removal safety injection system into the upper plenum and the high head safety injection system into the cold legs, this requirement is met by requiring alternate injection within 20 hours after a LOCA. This time is dependent on power level, and the RCS, RWST, accumulator, and other water sources, volumes and boron concentrations. A reduction in the boron concentration in the BASTs will have no effect on the power level, or volumes assumed for the RCS, RWST, accumulators, and other water sources. Although the boron concentrations will be affected, it requires an increase in the concentration to adversely affect the boron precipitation. Since the boron concentration would be decreasing with the proposed change, there will be no adverse effect on the post-LOCA alternate injection requirement of 20 hours for the R. E. Ginna plant.

4.5 Post-LOCA Long Term Core Cooling Minimum Flow

Post-LOCA long term core cooling minimum flow is determined to ensure adequate flow for large break and small break at the time of recirculation switchover. A reduction of the boron concentration in the BASTs will have no effect on the inputs for this calculation. Therefore, this change will have no effect on the post-LOCA long term core cooling minimum flow for the R. E. Ginna plant.

4.6 LOCA Summary and Conclusions

The effect of reducing the boron concentration in the BASTs on the LOCA-related analyses for R. E. Ginna has been evaluated by Westinghouse. The potential effect of the change on the UFSAR analysis results for each of the LOCA-related accidents was evaluated and it was shown in all cases that the effect of the change did not result in exceeding any of the following design or regulatory limits:

1. The calculated peak fuel element cladding temperature is below the requirements of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

3. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

Therefore, it is concluded that the proposed modification to reduce the boron concentration in the BASTs is acceptable from the standpoint of the UFSAR accident analyses discussed in this section.

5.0 Conclusions

A reduction of the BASTs boron concentration to 2000 ppm at the R. E. Ginna plant will be acceptable from the standpoint of core response, steamline break containment integrity and LOCA evaluation.

References

1. RG&E to NRC Request for Amendment to Technical Specifications dated Oct.16, 1985, purpose - Revise Containment Internal Pressure Limitations.

2. Burnett, T. W. T., et. al., "LOFTRAN Code Description," WCAP-7909-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

3. Bordelon, F. M. and Murphy, E. T., "Containment Pressure Analysis Code (COCO)," WCAP - 8327, July 1974.

4. Hsieh, T. et al, "Environmental Qualification Instrument Transmitter Temperature Transient Analysis", WCAP-8936, February 1977 (Proprietary) and WCAP-8937, February 1977 (Non-Proprietary).

5. Jens, W. H., and Lottes, P. A., "Analysis of Heat Transfer, Burnout, Pressure Drop, and Density Data for High Pressure Water", USAEC Report ANL-4627, 1951.

6. WCAP-9220-P-A (Proprietary), WCAP-9221 (Non-Proprietary), Eicheldinger, C., "Westinghouse ECCS Evaluation Model - 1981 Version", Revision 1, 1981.

7. WCAP-8200 (Proprietary), "WFLASH - A FORTRAN-IV Computer Program For Simulation Of Transients In A Multi-Loop PWR", Esposito, V. J., et al., July 1973.

8. WCAP-8339 (Non-Proprietary), Bordelon, F. M., et. al., "Westinghouse ECCS Evaluation Model - Summary", June 1974.

9. Westinghouse Technical Bulletin NSID-TB-86-08, "Post-LOCA Long Term Cooling: Boron Requirements", October 31, 1986.

Table 1:
Containment Integrity Analysis - Steam Line Break Cases

CS - Failure of One Containment Spray Pump to Operate
DIESEL - Failure of one Diesel Generator to Start
MSIV - Failure of Main Steam Isolation Valve
FCS - Failure of Feedwater Control System

Case	Break Type	Break Size ft^2	Power %	Failure	Offsite Power	
					M & E	Containment
1A	UPSTREAM DER	4.37	102	CS	AVAIL	AVAIL
1B	"	"	"	DIESEL	NOT AVAIL	NOT AVAIL
2A	"	"	"	MSIV	AVAIL	AVAIL
2B	"	"	"	MSIV	NOT AVAIL	NOT AVAIL
3A	"	"	"	FCS	AVAIL	AVAIL
3B	"	"	"	FCS	NOT AVAIL	NOT AVAIL
4A	"	"	70	CS	AVAIL	AVAIL
4B	"	"	"	DIESEL	NOT AVAIL	NOT AVAIL
5A	"	"	"	MSIV	AVAIL	AVAIL
5B	"	"	"	MSIV	NOT AVAIL	NOT AVAIL
6A	"	"	"	FCS	AVAIL	AVAIL
6B	"	"	"	FCS	NOT AVAIL	NOT AVAIL
7A	"	"	30	CS	AVAIL	AVAIL
7B	"	"	"	DIESEL	NOT AVAIL	NOT AVAIL
8A	"	"	"	MSIV	AVAIL	AVAIL
8B	"	"	"	MSIV	NOT AVAIL	NOT AVAIL
9A	"	"	"	FCS	AVAIL	AVAIL
9B	"	"	"	FCS	NOT AVAIL	NOT AVAIL
10A	"	"	0	CS	AVAIL	AVAIL
10B	"	"	"	DIESEL	NOT AVAIL	NOT AVAIL
11A	"	"	"	MSIV	AVAIL	AVAIL
11B	"	"	"	MSIV	NOT AVAIL	NOT AVAIL
12A	"	"	"	FCS	AVAIL	AVAIL
12B	"	"	"	FCS	NOT AVAIL	NOT AVAIL

M&E Mass and energy released into containment

Table 1 continued

Case	Break Type	Break Size ft^2	Power %	Failure	Offsite Power	
					M & E	Containment
13A	DWNSTRM DER	1.4	102	CS	AVAIL	AVAIL
13B	"	"	"	DIESEL	NOT AVAIL	NOT AVAIL
14A	"	"	"	MSIV	AVAIL	AVAIL
14B	"	"	102	MSIV	NOT AVAIL	NOT AVAIL
15A	"	"	"	FCS	AVAIL	AVAIL
15B	"	"	"	FCS	NOT AVAIL	NOT AVAIL
16A	"	"	70	CS	AVAIL	AVAIL
16B	"	"	"	DIESEL	NOT AVAIL	NOT AVAIL
17A	"	"	"	MSIV	AVAIL	AVAIL
17B	"	"	"	MSIV	NOT AVAIL	NOT AVAIL
18A	"	"	"	FCS	AVAIL	AVAIL
18B	"	"	"	FCS	NOT AVAIL	NOT AVAIL
19A	"	"	30	CS	AVAIL	AVAIL
19B	"	"	"	DIESEL	NOT AVAIL	NOT AVAIL
20A	"	"	"	MSIV	AVAIL	AVAIL
20B	"	"	"	MSIV	NOT AVAIL	NOT AVAIL
21A	"	"	"	FCS	AVAIL	AVAIL
21B	"	"	"	FCS	NOT AVAIL	NOT AVAIL
22A	"	"	0	CS	AVAIL	AVAIL
22C	"	"	"	DIESEL	AVAIL	NOT AVAIL
23A	"	"	"	MSIV	AVAIL	AVAIL
23B	"	"	"	MSIV	NOT AVAIL	NOT AVAIL
24A	"	"	"	FCS	AVAIL	AVAIL
24C	"	"	"	FCS	AVAIL	NOT AVAIL
25A1	SMALL DER	0.3	102	CS	AVAIL	AVAIL
25A2	"	0.5	"	CS	AVAIL	AVAIL
25A3	"	0.7	"	CS	AVAIL	AVAIL
25A4	"	0.9	"	CS	AVAIL	AVAIL

Table 1 continued

Case	Break Type	Break Size ft²	Power %	Failure	Offsite Power	
					M & E	Containment
25A5	SMALL DER	1.1	102	CS	AVAIL	AVAIL
25B1		0.3	102	DIESEL	NOT AVAIL	NOT AVAIL
25B2	"	0.5	"	DIESEL	NOT AVAIL	NOT AVAIL
25B3	"	0.7	"	DIESEL	NOT AVAIL	NOT AVAIL
25B4	"	0.9	"	DIESEL	NOT AVAIL	NOT AVAIL
25B5	"	1.1	"	DIESEL	NOT AVAIL	NOT AVAIL
26A4	"	0.9	102	FCS	AVAIL	AVAIL
26A5	"	1.1	"	FCS	AVAIL	AVAIL
26B4	"	0.9	102	FCS	NOT AVAIL	NOT AVAIL
26B5	"	1.1	"	FCS	NOT AVAIL	NOT AVAIL
27A4	"	0.9	70	CS	AVAIL	AVAIL
27A5	"	1.1	"	CS	AVAIL	AVAIL
27B3	"	0.7	70	DIESEL	NOT AVAIL	NOT AVAIL
27B4	"	0.9	"	DIESEL	NOT AVAIL	NOT AVAIL
27B5	"	1.1	"	DIESEL	NOT AVAIL	NOT AVAIL
28A4	"	0.9	70	FCS	AVAIL	AVAIL
28A5	"	1.1	"	FCS	AVAIL	AVAIL
28B4	"	0.9	70	FCS	NOT AVAIL	NOT AVAIL
28B5	"	1.1	"	FCS	NOT AVAIL	NOT AVAIL
29A4	"	0.9	30	CS	AVAIL	AVAIL
29A5	"	1.1	"	CS	AVAIL	AVAIL
29B4	"	0.9	30	DIESEL	NOT AVAIL	NOT AVAIL
29B5	"	1.1	"	DIESEL	NOT AVAIL	NOT AVAIL
30A4	"	0.9	30	FCS	AVAIL	AVAIL
30A5	"	1.1	"	FCS	AVAIL	AVAIL
30B4	"	0.9	30	FCS	NOT AVAIL	NOT AVAIL
30B5	"	1.1	"	FCS	NOT AVAIL	NOT AVAIL
31A1	"	0.3	0	CS	AVAIL	AVAIL

Table 1 continued

Case	Break Type	Break Size ft^2	Power %	Failure	Offsite Power	
					M & E	Containment
31A2	SMALL DER	0.5	"	CS	AVAIL	AVAIL
31A3	"	0.7	0	CS	AVAIL	AVAIL
31A4	"	0.9	"	CS	AVAIL	AVAIL
31A5	"	1.1	"	CS	AVAIL	AVAIL
31C6		1.3	0	DIESEL	AVAIL	NOT AVAIL
31C7	"	1.5	"	DIESEL	AVAIL	NOT AVAIL
32C1	"	0.3	0	FCS	AVAIL	NOT AVAIL
32C2	"	0.5	"	FCS	AVAIL	NOT AVAIL
32C3	"	0.7	"	FCS	AVAIL	NOT AVAIL
32C4	"	0.9	"	FCS	AVAIL	NOT AVAIL
32C5	"	1.1	"	FCS	AVAIL	NOT AVAIL
33A1	"	0.3	102	MSIV	AVAIL	AVAIL
33A2	"	0.5	"	MSIV	AVAIL	AVAIL
33A3	"	0.7	"	MSIV	AVAIL	AVAIL
33A4	"	0.9	"	MSIV	AVAIL	AVAIL
33A5	"	1.1	"	MSIV	AVAIL	AVAIL
33B5	"	1.1	102	MSIV	NOT AVAIL	NOT AVAIL
33C5	"	1.1	102	MSIV	AVAIL	NOT AVAIL
34C4	"	0.9	70	MSIV	AVAIL	NOT AVAIL
34C5	"	1.1	"	MSIV	AVAIL	NOT AVAIL
35A4	"	0.9	30	MSIV	AVAIL	AVAIL
35A5	"	1.1	"	MSIV	AVAIL	AVAIL
35B5	"	1.1	"	MSIV	NOT AVAIL	NOT AVAIL
36C1	"	0.3	"	MSIV	AVAIL	NOT AVAIL
36C2	"	0.5	0	MSIV	AVAIL	NOT AVAIL
36C3	"	0.7	"	MSIV	AVAIL	NOT AVAIL
36C4	"	0.9	"	MSIV	AVAIL	NOT AVAIL
36C5	"	1.1	0	MSIV	AVAIL	NOT AVAIL

Table 1 continued

Case	Break Type	Break Size ft²	Power %	Failure	Offsite Power M & E	Offsite Power Containment
37A5	SPLIT	1.1	102	MSIV	AVAIL	AVAIL
37C4	"	0.9	102	MSIV	AVAIL	NOT AVAIL
37C5	"	1.1	"	MSIV	AVAIL	NOT AVAIL
38A4	"	0.9	70	MSIV	AVAIL	AVAIL
38A5	"	1.1	"	MSIV	AVAIL	AVAIL
38B5	"	1.1	70	MSIV	NOT AVAIL	NOT AVAIL
39A4	"	0.9	30	MSIV	AVAIL	AVAIL
39A5	"	1.1	"	MSIV	AVAIL	AVAIL
39B5	"	1.1	30	MSIV	NOT AVAIL	NOT AVAIL
40C1	"	0.3	0	MSIV	AVAIL	NOT AVAIL
40C2	"	0.5	"	MSIV	AVAIL	NOT AVAIL
40C3	"	0.7	"	MSIV	AVAIL	NOT AVAIL
40C4	"	0.9	"	MSIV	AVAIL	NOT AVAIL
40C5	"	1.1	"	MSIV	AVAIL	NOT AVAIL

Table 2 - LOFTRAN Initial Conditions/Input Assumptions

Parameter	Initial Power Level 102%	70%	30%	0%
Nominal Average RCS Temperature (°F)	573.5	565.55	554.95	547.0
RCS Flowrate (gpm)	174000	174000	174000	174000
RCS Pressure (psia)	2250	2250	2250	2250
Feedwater Temperature (°F)	425	385	322	100
Nominal Pressurizer Water Level (%NRS) **	49.0	40.2	28.4	19.5
Nominal Steam Generator Water Level (%NRS)	52.0	52.0	52.0	52.0*

* The actual steam generator level at zero power is 39% NRS ± uncertainties. 52% NRS ± uncertainties was conservatively assumed in the analyses.

** NRS -- Narrow Range Span

Initial Condition Uncertainties

Average RCS Temperature = 4°F

Pressurizer Water Level = 5% NRS

Steam Generator Water Level = 3.5% NRS (Some cases assumed 5% NRS)

Table 3: Major Containment Assumptions

Initial Pressure	15.7 psi
Initial Temperature	120°F
Initial Humidity	20%
Containment Volume	1.0 E+06 ft^3
Containment Fan Coolers	
High-1 Setpoint Used	6.0 psig
Actual Setpoint	4.0 psig
Instrument Uncertainty	2 psi
Initiate on	SI (or High-1 signal if earlier)
Heat Removal Rates	Table 5
With off-site power available	
Number of Fan Coolers	4
Delay	34.0 sec
Without off-site power available	
Number of Fan Coolers	
without Diesel Failure	4
with Diesel Failure	2
Delay	44.0 sec
Containment Sprays	
Flowrate per Spray Pump	1300 gpm
RWST Water Temperature	80 °F
Pressure Setpoint Used	32.5 psig
Actual Setpoint	28.0 psig
Pressure Instrument Uncertainties	4.5 psig

Table 3 (continued): Major Containment Assumptions

. With Off-site Power Available	
Number of Spray Pumps Operating	
without containment spray failure	2
with containment spray failure	1
Delay	
without containment spray failure	27.3 sec
with containment spray failure	28.5 sec
Without Off-site Power Available	
Number of Spray Pumps operating	
without diesel failure	2
with diesel failure	1
Delay	45.5 sec
Heat Sinks	Table 4

Table 4: PASSIVE HEAT SINKS

Wall Description	Heat Transfer Area (ft^2)	Material	Thickness (ft)
1. Insulated portion of dome and containment wall	36285.0	stainless steel	0.00158
		insulation	0.1042
		steel	0.03125
		concrete	3.364
2. Uninsulated portion of dome and containment wall	12370.0	steel	0.03125
		concrete	2.5
3. Basement floor	6576.0	concrete	2.0
		steel	0.0208
		concrete	2.0
4. Walls of sump A under sump level	8.24	steel	0.0208
		concrete	3.0
5. Wall of sump A over sump level	2052.75	steel	0.0208
		concrete	3.0
6. Floor of sumps A and B	366.0	concrete	2.0
		steel	0.0208
		concrete	1.0
7. Walls of sump B	189.0	concrete	2.0
		steel	0.0208
		concrete	1.0
8. Outer refueling cavity wall	5870.0	concrete	3.44
9. Inner refueling cavity wall	5870.0	stainless steel	0.0208
		concrete	2.0
10. Bottom of refueling cavity	1143.0	stainless steel	0.0208
		concrete	4.0
11. Loop compartments (Loops A and B)[1]	18846.0	concrete	1.4115
12. Floor of intermediate level[1]	9672.0	concrete	0.25
13. Operating floor and structure on operating floor[1]	15570.0	concrete	1.0

Table 4 (continued): PASSIVE HEAT SINKS

Wall Description	Heat Transfer Area ft^2	Material	Thickness ft
14. I-beam and beams for crane structure[1]	7120.0	steel	0.0625
15. I-beam and beams for crane structure[1]	3458.0	steel	0.03455
16 I-beam[1]	7592.0	steel	0.0217
17. I-beam, cylindrical supports for S.G. and RCPs, and containment crane rectangular support columns	5536.0	steel	0.0586
18. Containment crane rectangular support columns	342.0	steel	0.167
19. Beams for crane structure	236.0	steel	0.12
20. Grating, stairs, misc. steel[1]	14000.0	steel	0.0625

[1]- Area accounts for both sides of heat sink walls, thickness is half of actual thickness

Thermophysical Properties of Containment Heat Sinks

	Thermal Conductivity (BTU/hr ft °F)	Volumetric Heat Capacity (BTU/ft^3 °F)
Insulation	0.0208	2.0
Concrete	0.81	31.5
Steel	28.0	54.4
Stainless Steel	10.9	60.0

Table 5: Containment Fan Cooler Heat Removal Rates

Containment Temperature	Group A: with Offsite Power Available	Group B: without Offsite Power Available
deg F	BTU/hr(*10⁶)	BTU/hr(*10⁶)
200	15.90	15.22
210	17.40	16.66
220	20.70	19.82
230	25.80	24.70
240	30.60	29.30
250	34.50	33.03
260	38.10	36.48
270	41.70	39.93
280	45.00	43.09
287	47.00	45.00
290	48.30	46.24
300	50.70	48.54

Table 6 - Sequence of Events

Accident	Event	Time (sec)
1. Main Steamline Break a. 30% Power b. MSIV Failure c. 4.36 ft^2 break d. Offsite Power Available	Steamline Break Occurs	0.0
	High 1 Containment Pressure Setpoint (6.0 psig) is Reached	1.0
	Rod Motion Starts	2.4
	Steamline Isolation Occurs	7.4
	Feedwater Isolation Occurs	14.4
	Auxiliary Feedwater Starts	25.0
	Containment Sprays Start	34.5
	Fan Coolers Start	42.0
	Peak Containment Pressure is Reached	149
	Auxiliary Feedwater is Terminated	600.0
	Faulted Steam Generator Dries Out (i.e., mass releases stop)	~610.0

Table 6 - Sequence of Events (continued)

Accident	Event	Time (sec)
2. Main Steamline Break a. 70% Power b. FCS Failure c. 1.40 ft^2 break d. Offsite Power Not Available	Steamline Break Occurs	0.0
	SIS Low Steam Pressure Setpoint (372.7 psia) reached	2.7
	High 1 Containment Pressure Setpoint (6.0 psig) is Reached	3.8
	Rod Motion Starts	4.7
	Feedwater Isolation Occurs	24.7
	Auxiliary Feedwater Starts	25.0
	Containment Sprays Start	47.8
	Fan Coolers Start	126.8
	Peak Containment Pressure is Reached	569
	Auxiliary Feedwater is Terminated	600.0
	Faulted Steam Generator Dries Out (i.e., mass releases stop)	~625.0

Table 6 - Sequence of Events (continued)

Accident	Event	Time (sec)
3. Main Steamline Break a. 102% Power b. CS Failure c. 1.10 ft^2 break d. Offsite Power Not Available	Steamline Break Occurs	0.0
	High 1 Containment Pressure Setpoint (6.0 psig) is Reached	4.7
	Rod Motion Starts	7.0
	Auxiliary Feedwater Starts	25.0
	Feedwater Isolation Occurs	27.0
	Containment Sprays Start	48.8
	Fan Coolers Start	128.1
	Auxiliary Feedwater is Terminated	600.0
	Faulted Steam Generator Dries Out (i.e., mass releases stop)	~760.0
	Peak Containment Pressure is Reached	762

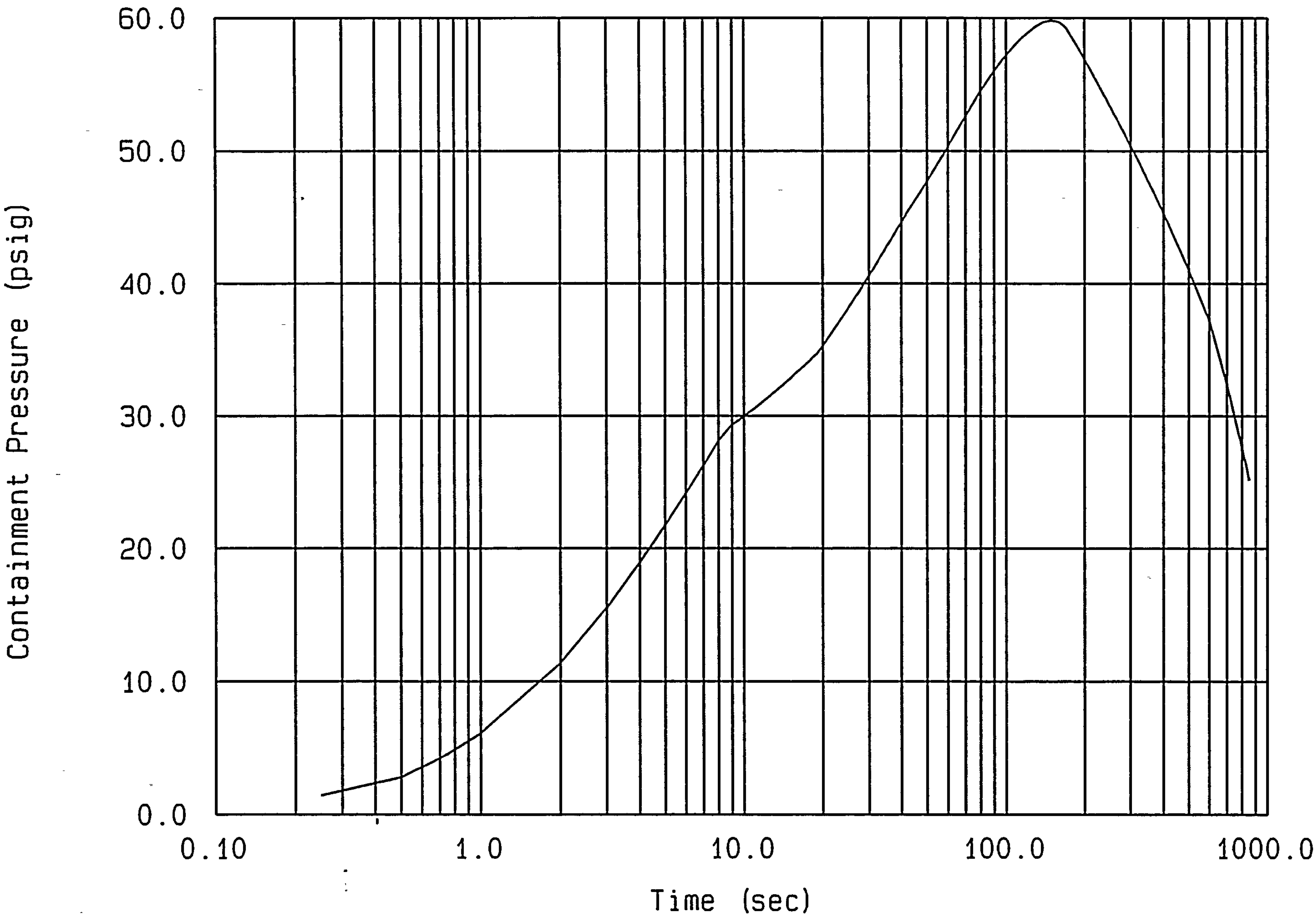

Figure 1 - Pressure Transient for Limiting Upstream DER

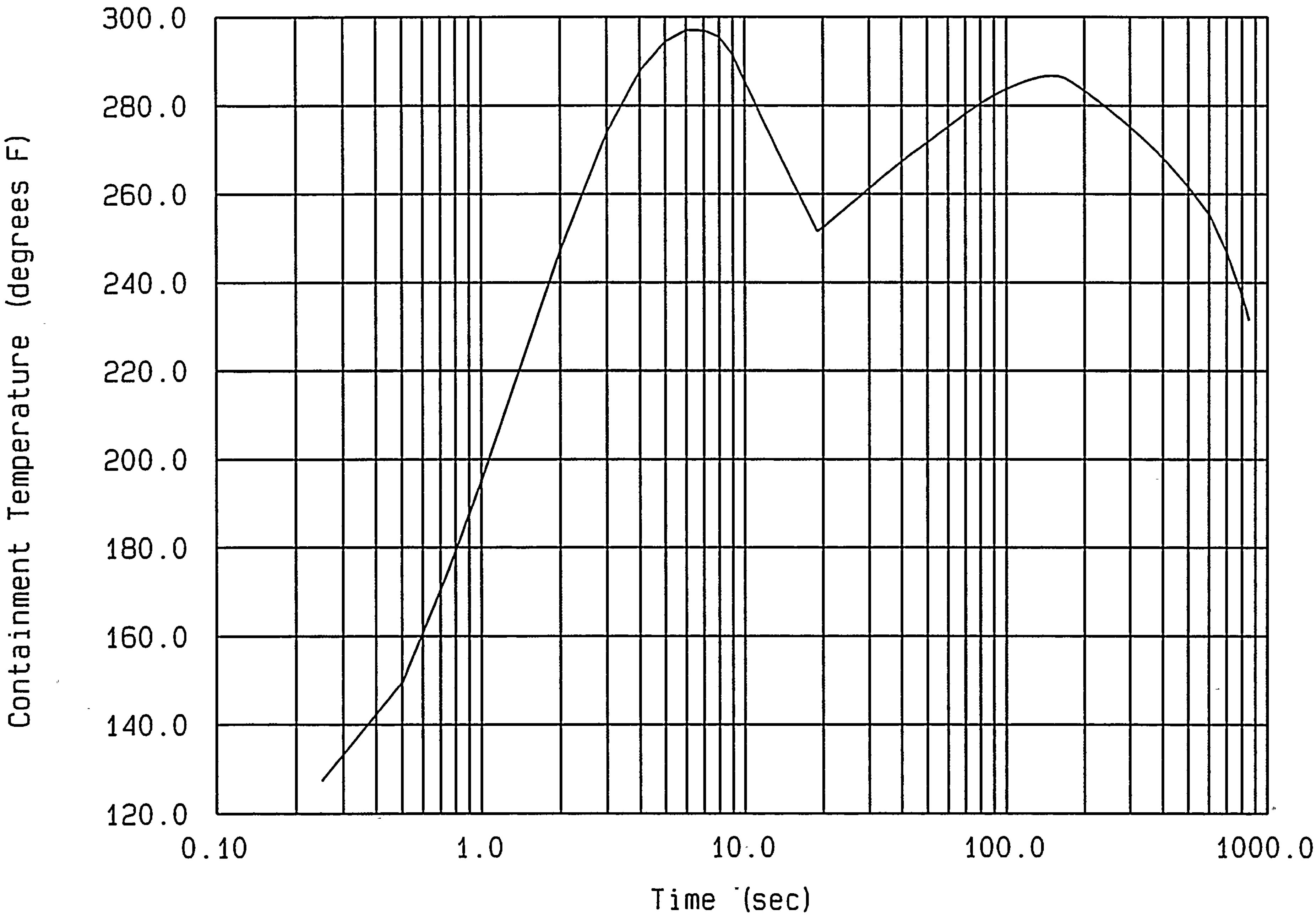

Figure 2 - Temperature Transient for Limiting Upstream DER

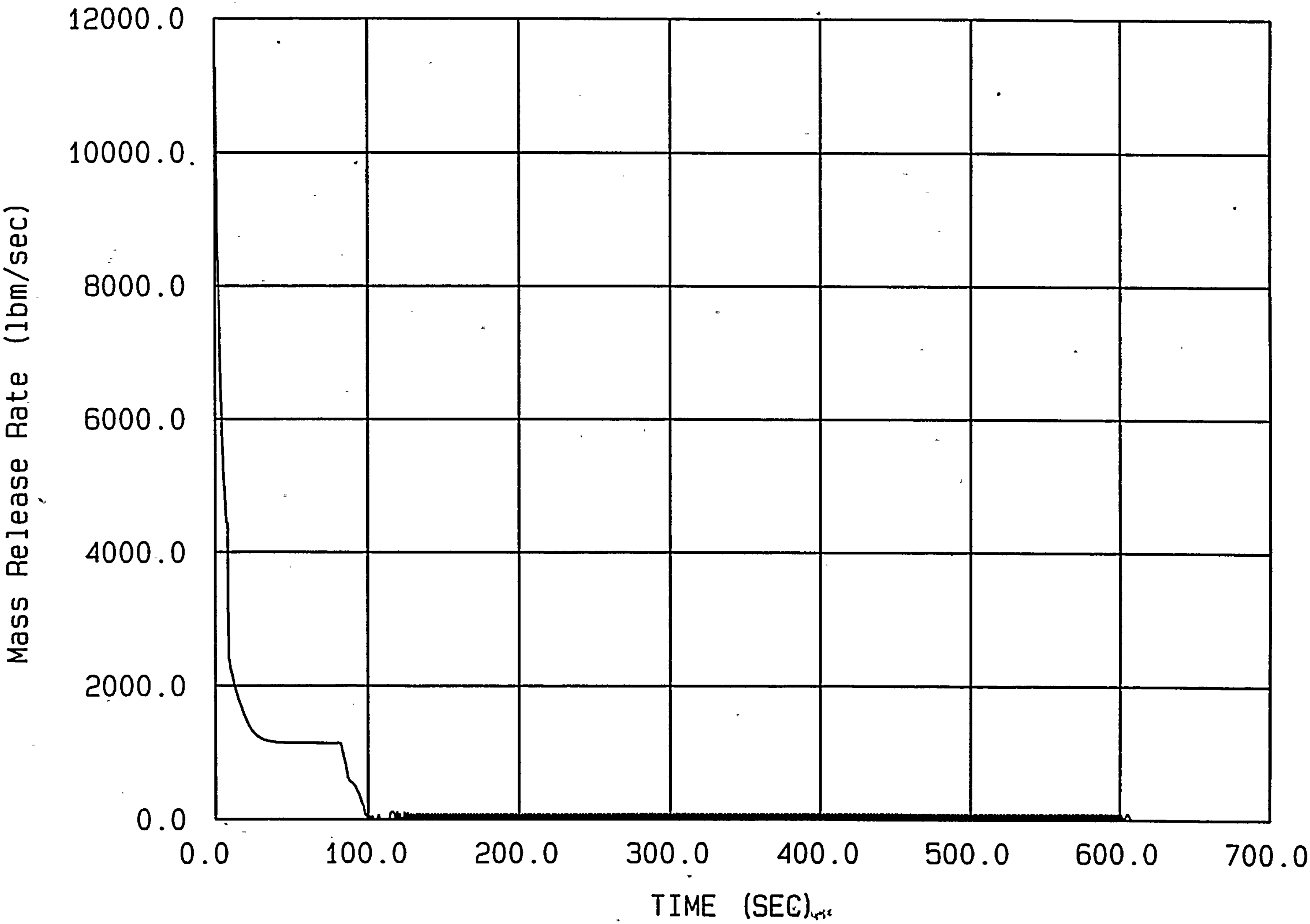

Figure 3 - Case 8A / Mass Release Rate vs. Time

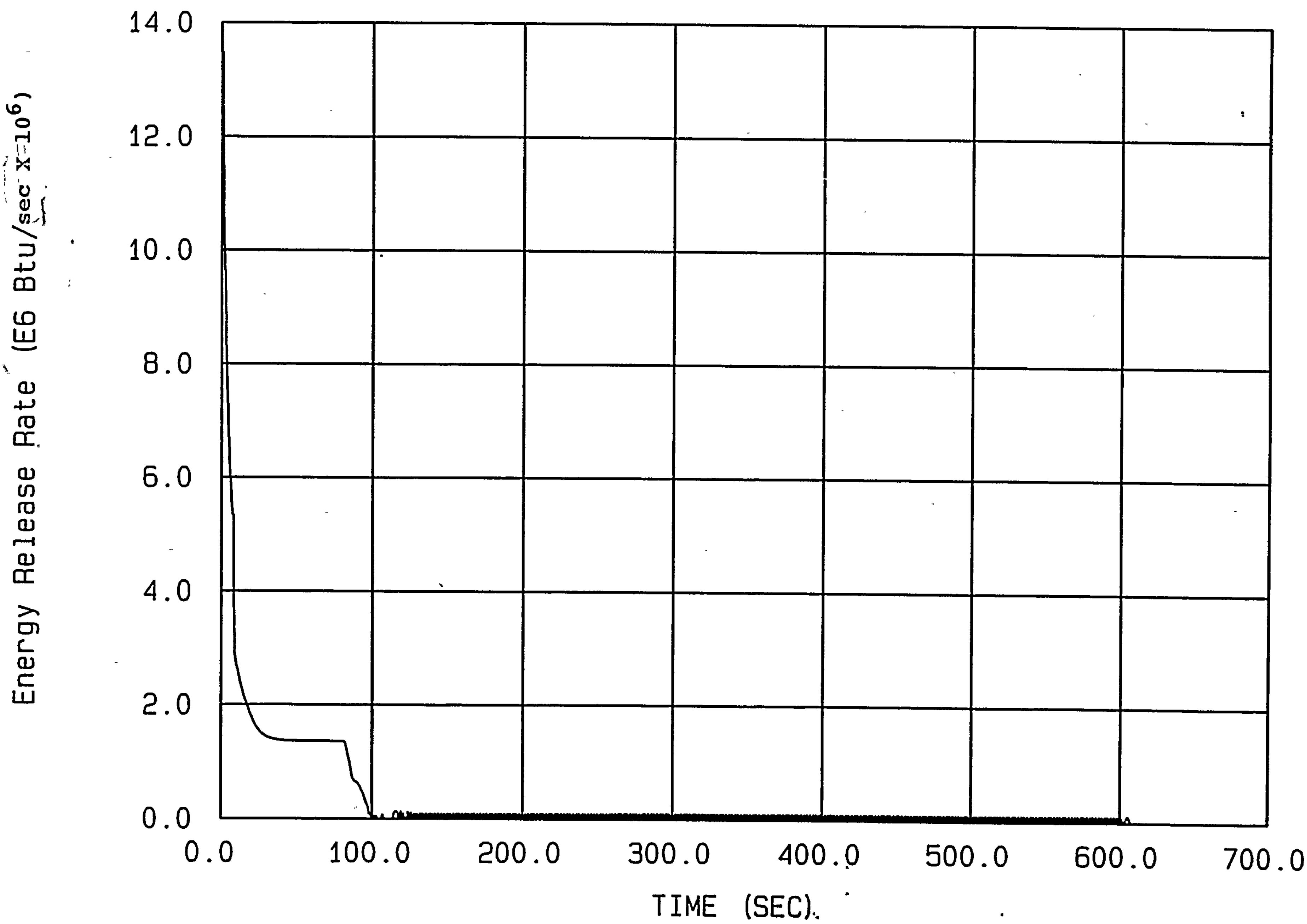

Figure 4 - Case 8A / Energy Release Rate vs. Time

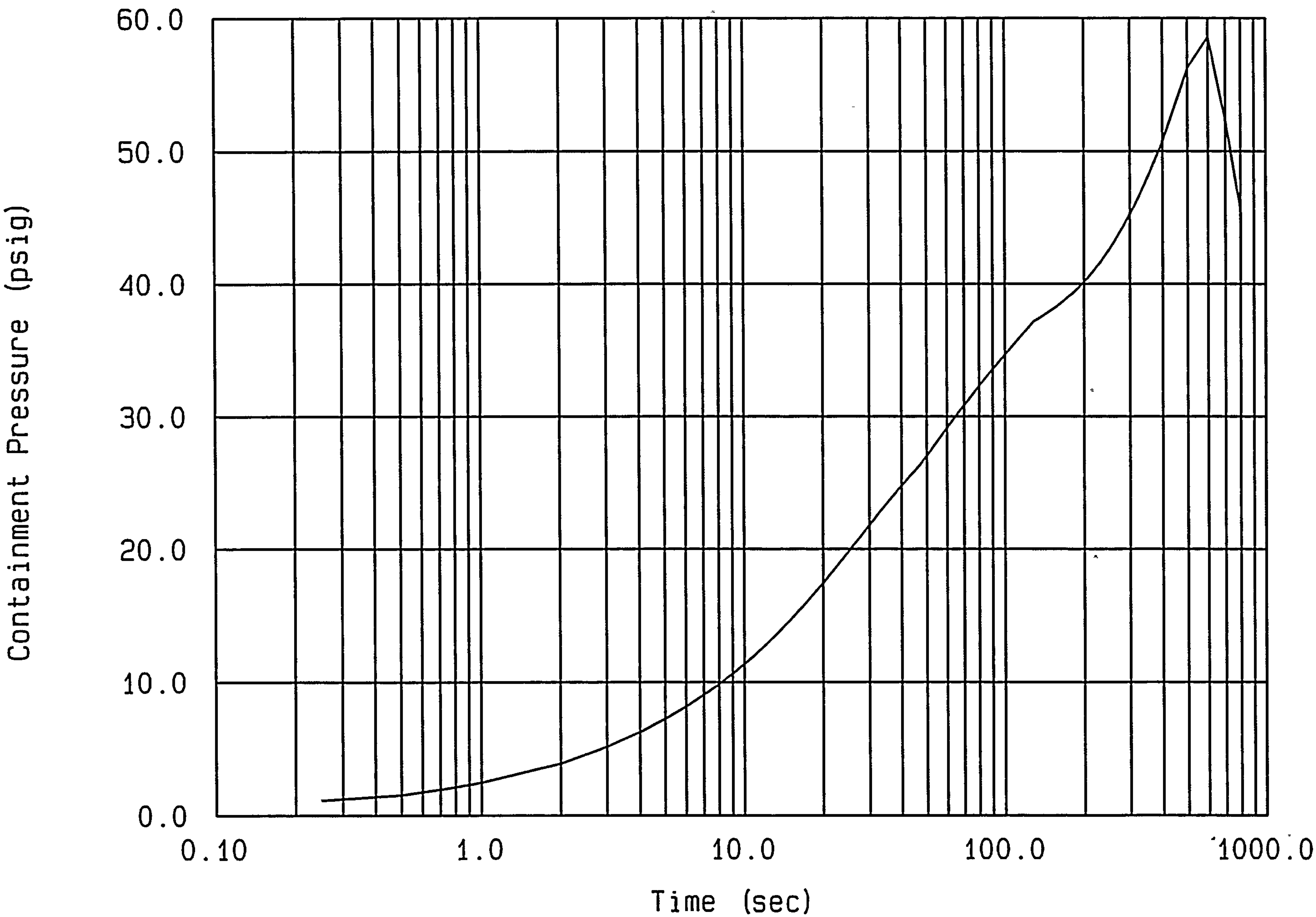

Figure 5 - Pressure Transient for Limiting Downstream DER

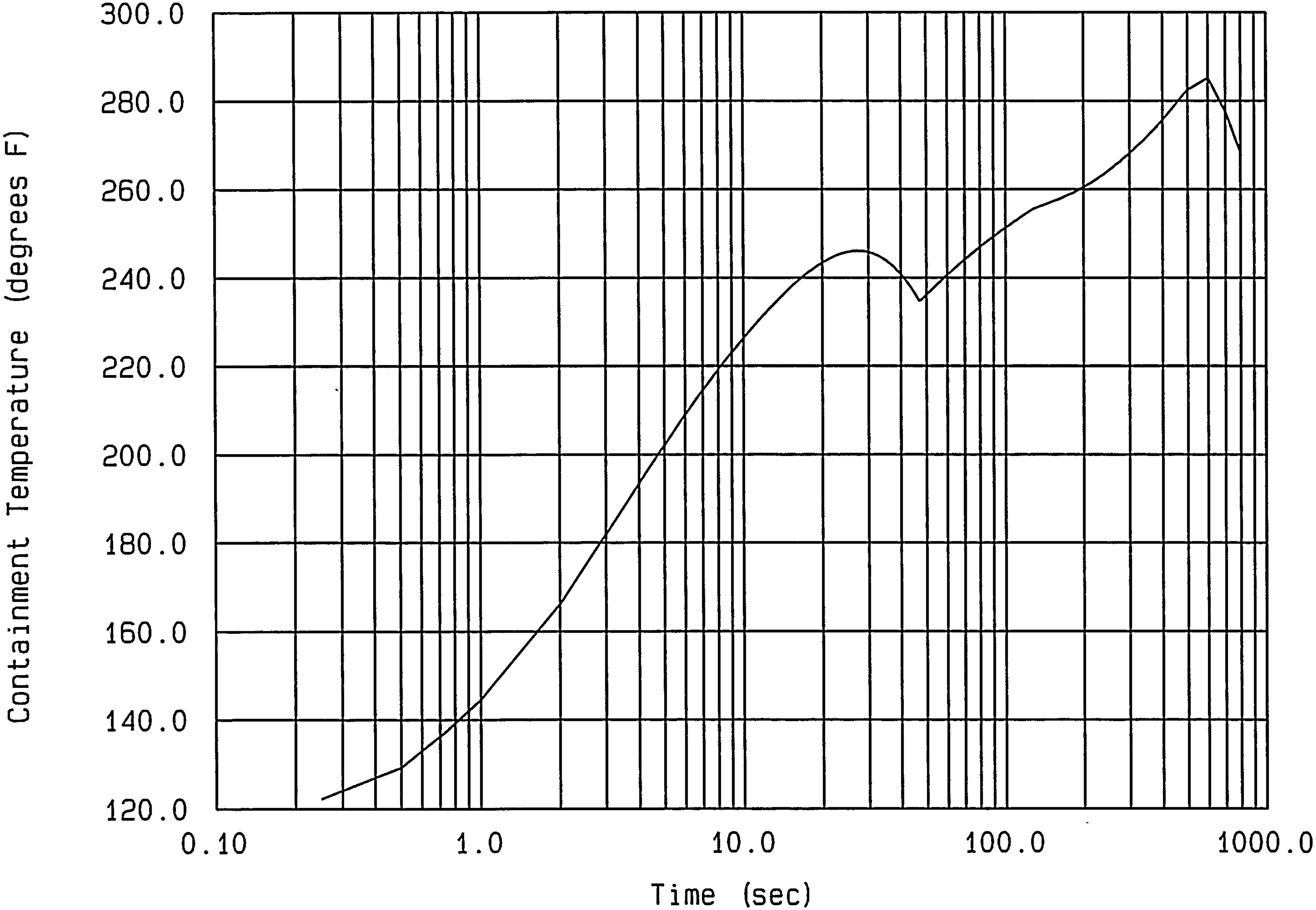

Figure 6 - Temperature Transient for Limiting Downstream DER

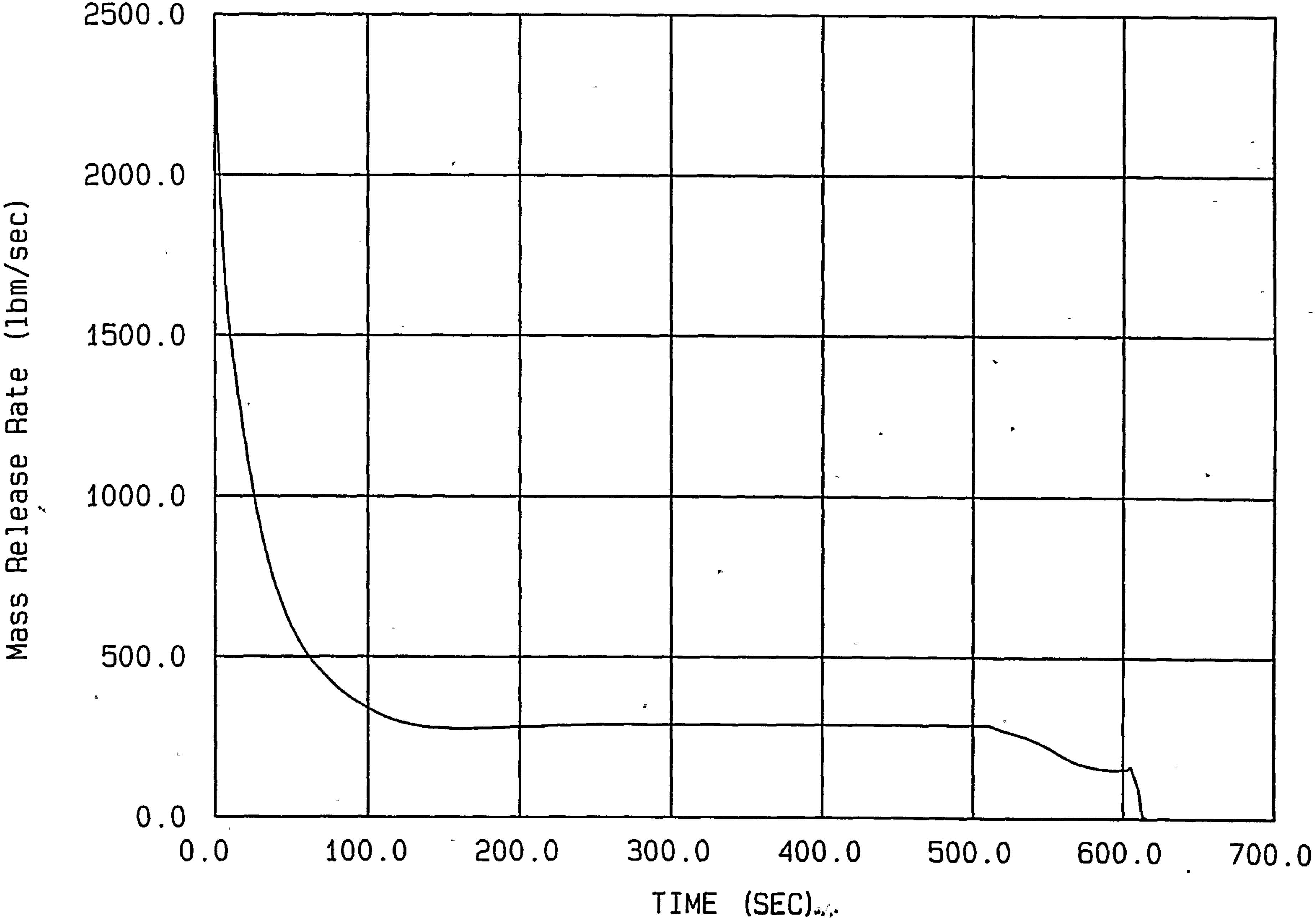

Figure 7 - Case 18B / Mass Release Rate vs. Time

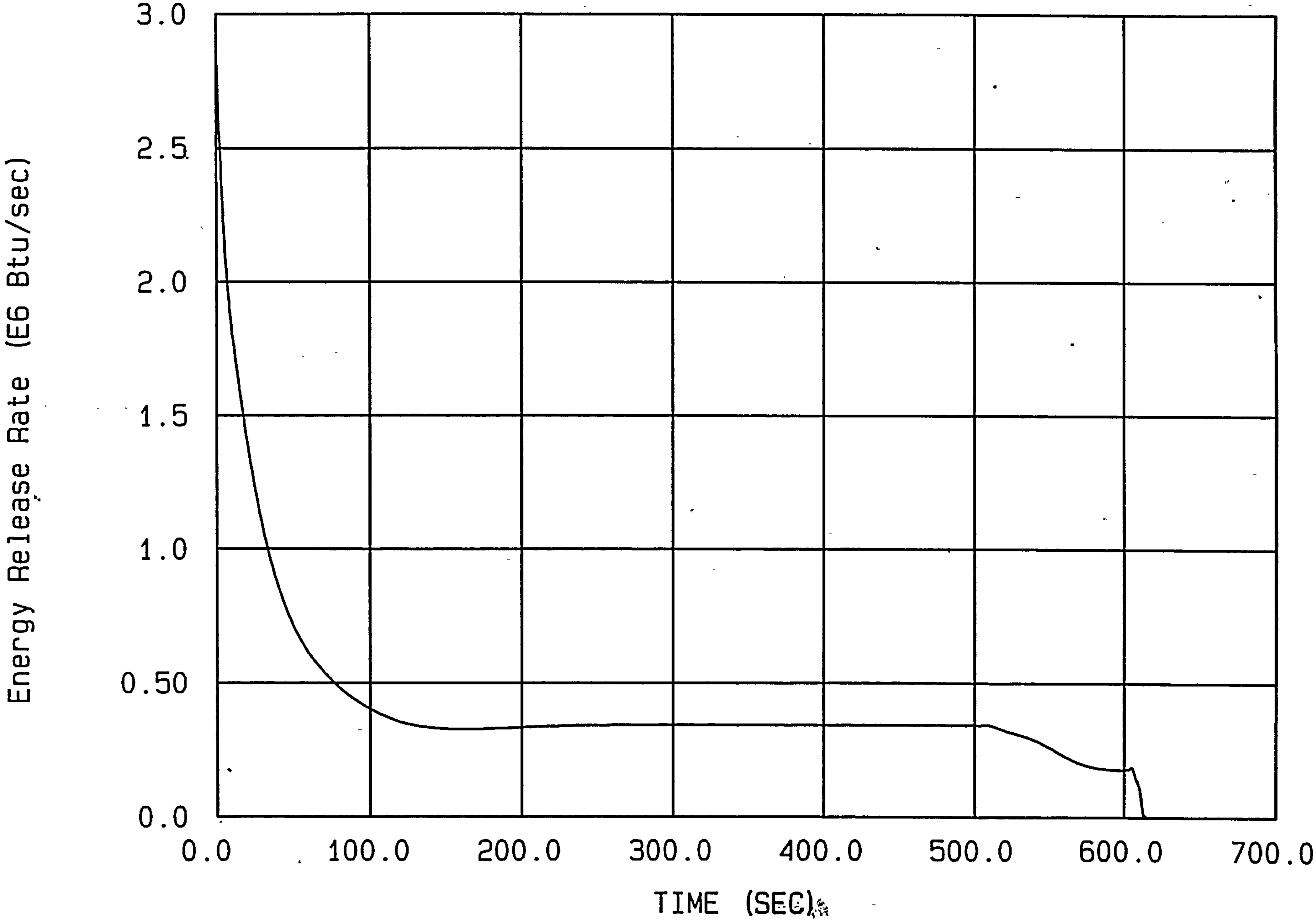

Figure 8 - Case 18B / Energy Release Rate vs. Time

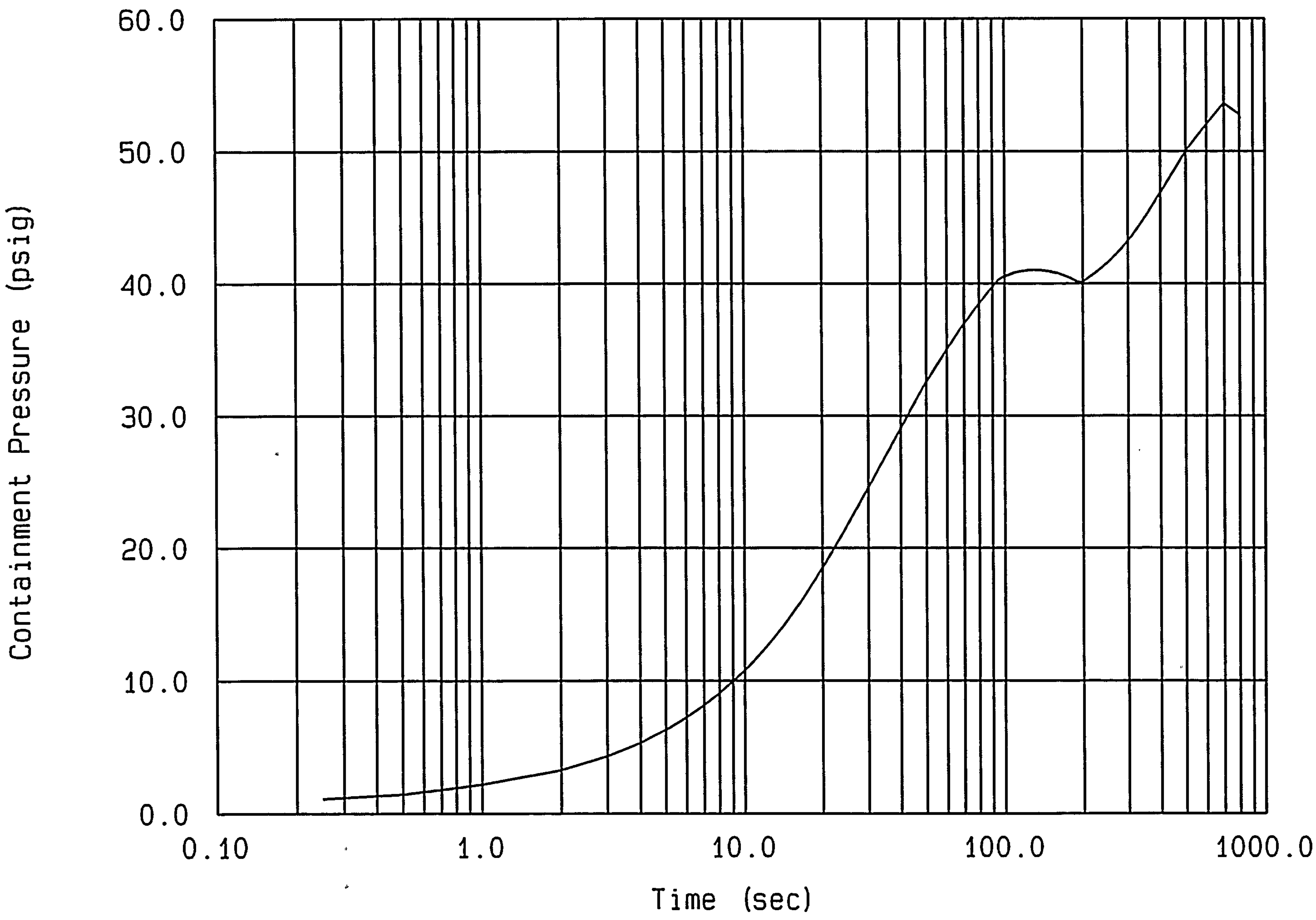

Figure 9 - Pressure Transient for Limiting Small Break

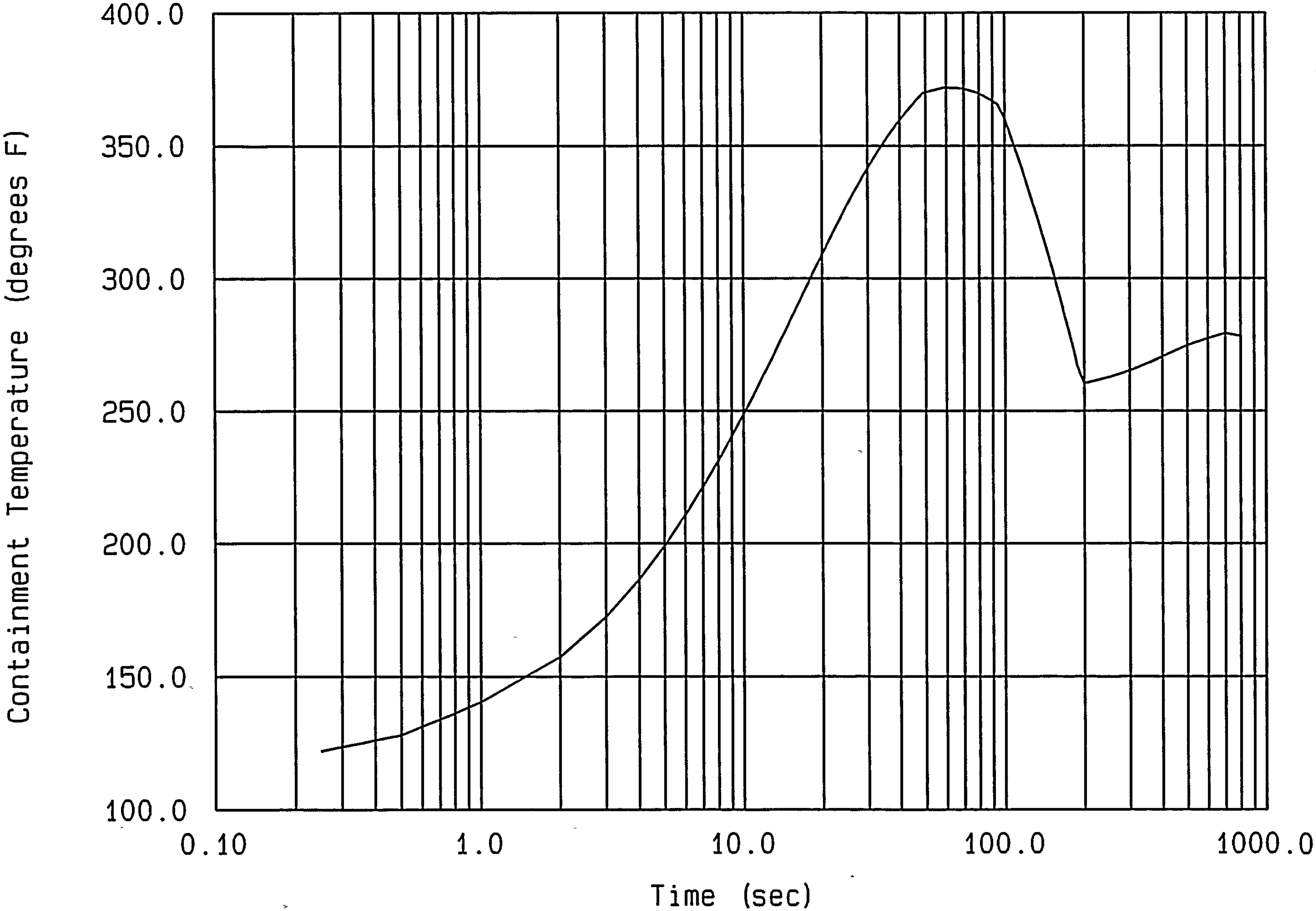

Figure 10 - Temperature Transient for Limiting Small Break

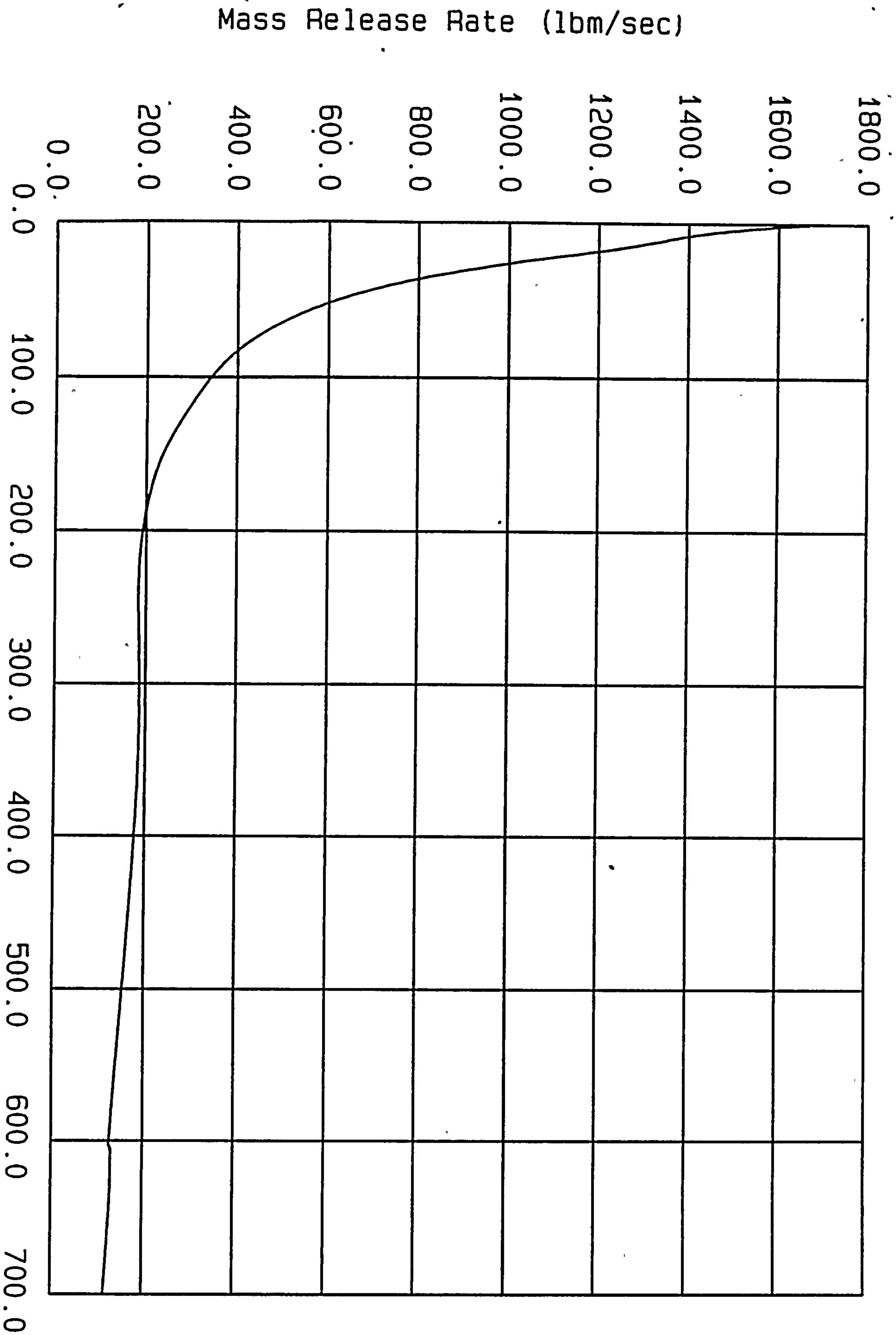

Figure 11 - Case 25B5 / Mass Release Rate vs. Time

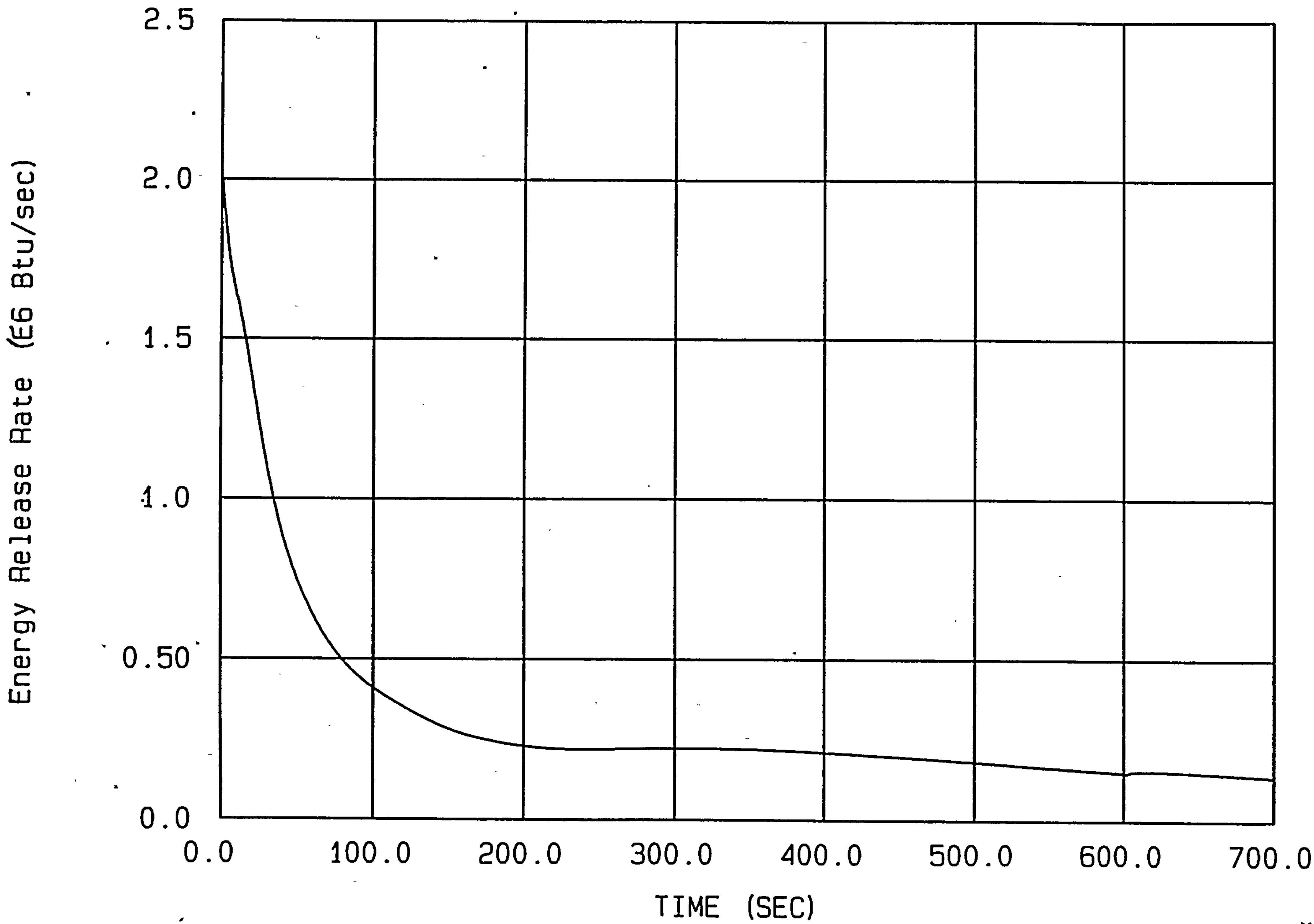

Figure 12 - Case 25B5 / Energy Release Rate vs. Time

Attachment 1: Computer Codes Used for Containment Integrity Analysis

The following is a general description of each of the computer codes used in this analysis.

LOFTRAN

The LOFTRAN program is used for studies of transient response of a PWR system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing a reactor vessel, hot and cold leg piping, steam generator (tube and shell sides) and the pressurizer. The pressurizer heaters, spray, relief, and safety valves are also considered in the program. Point neutron kinetics model, and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for a thermal transients and a water level correlation for indication and control. The Reactor Protection System is simulated to include reactor trips on high neutron flux, Overtemperature ΔT, Overpower ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure control. The Emergency Core Cooling System, including the accumulators and upper-head injection, is also modeled. LOFTRAN is discussed further in Reference A.

COCO

The COCO computer code (Reference B) is used to analyze the containment pressure transient response following a main steam line break accident. COCO is a mathematical model of a generalized containment; the proper selection of various options in the code allows the creation of a specific model for the particular containment design. The values used in the specific model for the different aspects of the containment are derived from plant-specific input data.

The COCO computer code consists of time-dependent conservation equations of mass and energy, together with steam tables, equations of state and other auxiliary relationships. Transient conditions are determined for both the containment steam-air mixture and the sump water. The energy equation is applied to the containment shell to obtain transient temperature gradients as well as heat stored in and conducted through the structure. Heat removal by means of energy storage in equipment within the containment, internal sprays, emergency containment coolers, and sump water recirculation cooling system is considered.

The containment air-steam-water mixture is separated into two distinct systems. The first system consists

of the air-steam phase, while the second system is the water phase in the containment sump. This division permits more accurate representation of the distinct physical phenomena occurring in each system.

The steam-air mixture and water phase are assumed to have uniform properties. In addition, temperature equilibrium between the air and steam is assumed. However, this does not imply continual thermal equilibrium between the steam-air mixture and water phase. Sufficient relationships to solve the problem independent of this restriction are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate equations of state and heat transfer boundary conditions.

Air inside the containment is treated as an ideal gas. Thermodynamic properties of water and steam are derived from compressed water and steam tables.

Heat transfer through, and heat storage in, interior and exterior walls of the containment structure are considered. Structural heat sinks, consisting of steel and concrete, are modeled as slabs having specific areas and layers of varying thicknesses. The thermal conductivity, density and specific heat of each layer are specified at an initial temperature.

Discharge mass and energy flow rates through the rupture are established by separate analyses of the steam generator transient. This information is supplied as time-dependent data to the code.

For the larger steam line break cases, the calculation assumes the Tagami condensation heat transfer correlation and the revaporization model. The revaporization model assumes that an equilibrium condition exists between the condensate on the containment structures and the containment steam-air atmosphere. At each time step, the conservation equations (mass, energy, and state) are solved simultaneously to determine a new containment air-steam-condensate condition. If the calculated condition is a saturated state, water mass (condensate) forms and is assumed to fall instantly into the sump. If the condition is a super-heated state, the water mass would not form at that time step. The condensate which is at a saturated state based on the interfacial temperature at a previous time step may re-evaporate under the exposure to a rapidly increasing super-heated atmosphere.

The COCO code has been benchmarked against the CVTR tests (Reference C). The CVTR tests were super-heated steam blowdown tests. The containment free volume is about one-eighth of a typical three loop PWR containment. The blowdown steam enthalpy was 1195 BTU/lbm, which is about the same as

that for a postulated steam line break with no moisture carry-over.

The COCO calculation showed good agreement with the test data when the revaporization model was used. When no revaporization was assumed, the COCO calculation predicted a much higher temperature than the test. In both cases, COCO over-predicted the containment atmosphere pressure.

For small steam line breaks, the condensation heat transfer is based on stagnant conditions and the wall condensate is assumed to fall to the sump with no revaporization. The approved mass and energy release model assumes no entrainment, i.e., dry steam blowdown. The NRC staff has approved the use of the revaporization model, on previous plant-specific applications, for break sizes which would have entrainment (Reference D). The use of the revaporization model has been approved for large steam line breaks in the LOTIC-3 code used for ice condenser plants (Reference E).

References for Attachment 1

A. Burnett, T. W. T., et. al., "LOFTRAN Code Description," WCAP-7909-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

B. Bordelon, F. M. and Murphy, E. T., "Containment Pressure Analysis Code (COCO)," WCAP - 8327, June 1974.

C. Schmitt, R. C., Bingham, G. E., and Norberg, J. A., "Simulated Design Basis Accident Tests of the Carolinas Virginia Tube Reactor Containment -- Final Report," IN-1403, Idaho Nuclear Corporation, December, 1970.

D. "Diablo Canyon Safety Evaluation Report," NUREG-0675, June 1980.

E. Hsieh, T. and Liparulo, N. J., "Westinghouse Long Term Ice Condenser Containment Code - LOTIC -3 Code," WCAP-8354-P-A, Supplement 2, February 1979.

ATTACHMENT D

Comparison of Existing to Proposed
Technical Specifications

Proposed verbage in bold print
Deleted Verbage Crossed out

1.18 Dose Equivalent I-131

The dose equivalent I-131 shall be that concentration of I-131 which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The dose conversion factors used for this calculation shall be those for the adult thyroid dose via inhalation, contained in NRC Regulatory Guide 1.109 Rev. 1 October 1977.

1.19 Reportable Event

A Reportable Event shall be any of those conditions specified in Section 50.73 to 10CFR Part 50.

1.20 Canisters Containing Consolidated Fuel Rods

Canisters containing consolidated fuel rods are stainless steel canisters containing the fuel rods of no more than two fuel assemblies which have decayed at least five years and are capable of being stored in a storage cell of the spent fuel pool.

1.21 Shutdown Margin

Shutdown margin shall be the amount of reactivity by which the reactor is subcritical, or would be subcritical from its present condition assuming all rod cluster control assemblies (shutdown and control) are fully inserted except for the single rod cluster control assembly of highest reactivity worth which is assumed to be fully withdrawn, and assuming no changes in xenon or boron concentration.

3.2 Chemical and Volume Control System

Applicability

Applies to the operational status of the chemical and volume control system.

Objective

To define those conditions of the chemical and volume control system necessary to assure safe reactor operation.

Specification

3.2.1 ~~When fuel is~~ During cold shutdown or refueling with fuel in the reactor there shall be at least one flow path to the core for boric acid injection. The minimum capability for boric acid injection shall be equivalent to that supplied from the refueling water storage tank.

3.2.1.1 With this flow path unavailable, immediately suspend all operations involving core alterations or positive reactivity changes and return a flow path to operable status as soon as possible.

3.2.2 ~~The reactor shall not be taken above cold shutdown unless the following Chemical and Volume Control System conditions are met.~~

~~a. At least two charging pumps shall be operable.~~

~~b. Both boric acid transfer pumps shall be operable.~~

~~c. The boric acid tanks together shall contain a minimum of 2000 gallons of a 12% to 13% by weight boric acid solution at a temperature of at least 145°F (See also Specification 3.3.1.1.j).~~

~~d. System piping and valves shall be operable to the extent of establishing two flow paths from the boric acid tanks to the Reactor Coolant System and a flow path from the refueling water storage tank to the Reactor Coolant System.~~

~~e. Both channels of heat tracing shall be operable for the~~

~~above flow paths.~~

When the reactor is above cold shutdown, two boron injection flow paths shall be operable with one operable charging pump for each operable flow path, and one operable boric acid transfer pump for each operable flow path from the boric acid storage tank(s).

3.2.3 ~~The requirements of 3.2.2 may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of 3.2.2 within the time period specified below, the reactor shall be placed in the hot shutdown condition within 6 hours. If the requirements of 3.2.2 are not satisfied within an additional 48 hours the reactor shall be in cold shutdown within the next 30 hours.~~

~~a. If only one charging pump is operable, then restore the second pump to operable status within 24 hours.~~

~~b. One boric acid pump may be out of service provided the pump is restored to operable status within 24 hours.~~

~~c. One boric acid tank may be out of service provided a minimum of 2,000 gallons of a 12% to 13% by weight boric acid solution at a temperature of at least 145°F is contained in the operable tank and provided that the tank is restored to operable status within 24 hours.~~

~~d. If only one flow path from the boric acid tanks is operable, then restore the second flowpath to operable status within 24 hours.~~

~~e. One channel of heat tracing may be out of service provided it is restored to operable status within 24 hours.~~

If required by specification 3.2.2 above, the Boric Acid Storage Tank(s) shall satisfy the concentration, minimum

volume and solution temperature requirements of Table 3.2-1.

With only one of the required boron injection flow paths to the RCS operable, restore at least two boron injection flow paths to the RCS to operable status within 72 hours, or within the next 6 hours be in at least hot shutdown and borated to a shutdown margin equivalent to at least 2.45% delta k/k at cold, no xenon conditions. If the requirements of 3.2.2 are not satisfied within an additional 7 days, then be in cold shutdown within the next 30 hours.

3.2.5 Whenever the RCS temperature is greater than 200°F and is being cooled by the RHR system and the over-pressure protection system is not operable, at least one charging pump shall be demonstrated inoperable at least once per 12 hours by verifying that the control switch is in the pull-stop position.

Table 3.2-1

Boric Acid Storage Tank(s)

Minimum-Volume-Temperature-Concentration[2]

Concentration ppm boron	Minimum Volume gal.	Minimum Solution Temperature °F
4700 to less than 5000	8400	40
5000 to less than 6000	7800	52
6000 to less than 7000	6400	62
7000 to less than 8000	5400	70
8000 to less than 9000	4700	78
9000 to less than 10000	4200	85
10000 to less than 11000	3800	91
11000 to less than 12000	3500	97
12000 to less than 13000	3200	103
13000 to less than 14000	3000	108
14000 to less than 15000	2700	113
15000 to less than 16000	2500	118
16000 to less than 17000	2400	123
17000 to less than 18000	2200	127
18000 to less than 19000	2100	131
19000 to less than 20000	2000	137
20000 to less than 21000	1900	140
21000 to less than 22000	1800	143
22000 to less than 23000	1800	145

Basis

The chemical and volume control system provides control of the reactor coolant system boron inventory.[1] ~~This is normally accomplished by using either of the three charging pumps in series with one of the two boric acid pumps. An alternate method of boration will be to use the charging pumps directly from the refueling water storage tank. A third method will be to depressurize and use the safety injection pumps. There are two sources of borated water available for injection through three different paths.~~

~~(1) The boric acid transfer pumps can deliver the boric acid tank contents (12% concentration of boric acid) to the charging pumps).~~

~~(2) The charging pumps can take suction from the refueling water storage tank. (2,000 ppm boron solution)~~

~~(3) The safety injection pumps can take their suctions from either the boric acid tanks or the refueling water storage tank.~~

~~The quantity of boric acid in storage from either the boric acid tanks or the refueling water storage tank is sufficient to borate the reactor coolant in order to reach cold shutdown at any time during core life. Approximately 1800 gallons of the 12% to 13% solution of boric acid are required to meet cold shutdown conditions.[2] Thus, a minimum of 2000 gallons in the boric acid tanks is specified. An upper concentration limit of 13% boric acid in the tank is specified to maintain solution solubility at the specified low temperature limit of 145°F. Two channels of heat tracing are installed on lines normally containing concentrated boric acid solution to maintain the specified low temperature limit.~~

~~References:~~

~~(1) FSAR, Sect,on 9.2~~

~~(2) FSAR, Page 9.2-37~~

[illegible]

~~(3) L.D. White, Jr. letter to A. Schwenceer, NRC, dated February 24, 1977~~

This is normally accomplished by using one or more charging pumps in series with one of the two boric acid transfer pumps.

Above cold shutdown conditions, a minimum of two of four boron injection flowpaths are required to insure single functional capability in the event that an assumed single active failure renders one of the flow paths inoperable. The boration volume available through any flow path is sufficient to provide the required shutdown margin at cold conditions from any expected operating condition and to compensate for shrinkage of the primary coolant from the cooldown process. The maximum volume requirement is associated with boration from just critical, hot zero power, peak xenon with control rods at the insertion limit, to cold shutdown with single reactor coolant loop operation. This requires 26,000[(2)] gallons of 2000 ppm borated water from the refueling water storage tank or the concentrations and volumes of borated water specified in Table 3.2-1 from the boric acid storage tanks. Two boric acid storage tanks are available. One of the two tanks may be out of service provided the required volume of boric acid is available to the operable flow paths.

Above cold shutdown, two of the following four flow paths must be operable with one operable charging pump for each operable flow path, and one operable boric acid transfer pump for each operable flow path from the boric acid storage tanks.

(1) Boric acid storage tanks via one boric acid transfer pump through the normal makeup (FCV 110A) flow path to the suction of the charging pumps.

(2) Boric acid storage tanks via one boric acid transfer pump through the emergency boration flow path (MOV 350) to the

suction of the charging pumps.

(3) Refueling water storage tank via gravity feed through AOV 112B to the suction of the charging pumps.

(4) Refueling water storage tank via gravity feed through manual bypass valve 358 to the suction of the charging pumps.

Available flow paths from the charging pumps to the reactor coolant system include the following:

(1) Charging flow path through AOV 392A to the RCS Loop B hot leg.

(2) Charging flow path through AOV 294 to the RCS Loop B cold leg.

(3) Seal injection flow path to the reactor coolant pumps.

The rate of boric acid injection must be sufficient to offset the maximum addition of positive reactivity from the decay of xenon after a trip from full power. This can be accomplished through the operation of one charging pump at minimum speed with suction from the refueling water storage tank. Also the time required for boric acid injection allows for the local alignment of manual valves to provide the necessary flow paths.

The quantity of boric acid specified in Table 3.2-1 for each concentration is sufficient at any time in core life to borate the reactor coolant to the required cold shutdown concentration and provide makeup to maintain RCS inventory during the cooldown. The temperature limits specified on Table 3.2-1 are required to maintain solution solubility at the upper concentration in each range. The temperatures listed on Table 3.2-1 are taken from Reference (4). An arbitrary 5°F is added to the Reference (4) for margin. Heat tracing may be used to maintain solution temperature at or above the Table 3.2-1 limits. If the solution temperature of either the flow path or the borated water source is not maintained at or above the minimum temperature specified, the affected flow path must be declared inoperable and the appropriate actions specified in 3.2.4 followed.

Placing a charging pump in pull-stop whenever the reactor coolant system temperature is ≥200°F and is being cooled by RHR without the overpressure protection system operable will prevent inadvertent overpressurization of the RHR system should letdown be terminated.[3]

References:

(1) UFSAR Section 9.3.4.2

(2) RG&E Design Analysis DA-NS-92-133-00 "BAST Boron Concentration Reduction Technical Specification Values" dated Dec. 14, 1992

(3) L.D. White, Jr. letter A. Schwencer, NRC, Subject: Reactor Vessel Overpressurization, dated February 24, 1977

(4) Kerr-McGee Chemical Corp. Bulletin 0151 "Boric Acid - Technical Grades" dated 5/84

3.3 Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, Containment Spray, and Charcoal/ HEPA Filters

Objective

To define those conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, (3) to remove airborne iodine from the containment atmosphere following a postulated Design Basis Accident, and (4) to minimize containment leakage to the environment subsequent to a Design Basis Accident.

Specification

3.3.1 Safety Injection and Residual Heat Removal Systems

3.3.1.1 The reactor shall not be taken above the mode indicated unless the following conditions are met:

a. Above cold shutdown, the refueling water storage tank contains not less than 300,000 gallons of water, with a boron concentration of at least 2000 ppm.

b. Above a reactor coolant system pressure of 1600 psig, **except during performance of RCS hydro test,** each accumulator is pressurized to at least 700 psig with an indicated level of at least 50% and a maximum of 82% with a boron concentration of at least 1800 ppm.

c. At or above a reactor coolant system ~~pressure and~~ temperature of ~~1600 psig and~~ 350°F, ~~except during performance of RCS hydro test~~ three safety injection pumps are operable.

d. At or above an RCS temperature of 350°F, two residual heat removal pumps are operable.

e. At or above an RCS temperature of 350°F, two residual heat removal heat exchangers are operable.

f. At the conditions required in a through e above, all valves, interlocks and piping associated with the above components which are required to function during accident conditions are operable.

g. At or above an RCS temperature of 350°F, A.C. power shall be removed from the following valves with the valves in the open position: safety injection cold leg injection valves 878B and D. A.C. power shall be removed from safety injection hot leg injection valves 878A and C with the valves closed. D.C. control power shall be removed from refueling water storage tank delivery valves 896A, 896B and 856 with the valves open.

h. At or above an RCS temperature of 350°F, check valves 853A, 853B, 867A, 867B, 878G, and 878J shall be operable with less than 5.0 gpm leakage each. The leakage requirements of Technical Specification 3.1.5.2.1 are still applicable.

i. Above a reactor coolant system pressure of 1600 psig, except during performance of RCS hydro test, A.C. power shall be removed from accumulator isolation valves 841 and 865 with the valves open.

j. ~~At or above a reactor coolant system pressure and temperature of 1600 psig and 350°F, except during performance of RCS hydro test, the boric acid tanks together shall contain a minimum of 3110 gallons of boric acid above the setpoint for switchover to the RWST. This solution shall be 12% to 13% by weight boric acid at a temperature of at least 145°F. Below 1600 psig or 350°F~~

~~the requirements of Specification 3.2.2 apply.~~
At or above an RCS temperature of 350° F, A.C. power shall be removed from Safety Injection suction valves 825A and B with the valves in the open position, and from valves 826A, B, C, D with the valves in the closed position.

At or above an RCS temperature of 350° F, A.C. power shall be removed from Safety Injection suction valves 825A and B with the valves in the open position, and from valves 826A, B, C, D with the valves in the closed position.

3.3.1.2 If the conditions of 3.3.1.1a are not met, then satisfy the condition within 1 hour or be at hot shutdown in the next 6 hours and at least cold shutdown within an additional 30 hours.

3.3.1.3 The requirements of 3.3.1.1b and 3.3.1.1i may be modified to allow one accumulator to be inoperable or isolated for up to one hour. If the accumulator is not operable or is still isolated after one hour, the reactor shall be placed in hot shutdown within the following 6 hours and below a RCS pressure of 1600 psig within an additional 6 hours.

3.3.1.4 The requirements of 3.3.1.1c may be modified to allow one safety injection pump to be inoperable for up to 72 hours. If the pump is not operable after 72 hours, the reactor shall be placed in hot shutdown within **the following** 6 hours and ~~at an RCS pressure and temperature less than 1600 psig and 350°F~~ **below a RCS temperature less than 350°F** within an additional 6 hours.

3.3.1.5 The requirements of 3.3.1.1d through h. may be modified to allow components to be inoperable at any one time. More than one component may be inoperable at any one time provided that one train of the ECCS is operable. If the requirements of 3.3.1.1d through h. are not satisfied within the time period specified below, the reactor shall be placed in hot shutdown within 6 hours and at an RCS temperature less than 350°F in an additional 6 hours.

a. One residual heat removal pump may be out of service provided the pump is restored to operable status within 72 hours.

b. One residual heat removal heat exchanger may be out of service for a period of no more than 72 hours.

c. Any valve, interlock, or piping required for the functioning of one safety injection train and/or one low head safety injection train (RHR) may be inoperable provided repairs are completed within 72 hours (except as specified in e. below).

d. Power may be restored to any valve referenced in 3.3.1.1g for the purposes of valve testing provided no more than one such valve has power restored and provided testing is completed and power removed within 12 hours.

e. Those check valves specified in 3.3.1.1h may be inoperable (greater than 5.0 gpm leakage) provided the inline MOVs are de-energized closed and repairs are completed within 12 hours.

3.3.1.6 ~~The requirements of 3.3.1.1.j may be modified to allow one boric acid tank to be out of service provided a minimum of 3110 gallons of boric acid above the setpoint for switchover to the RWST is contained in the operable tank. This solution shall be 12% to 13% by weight boric acid at a temperature of at least 145°F. If the modified requirement cannot be met within one hour, be in hot shutdown and borated to a shutdown margin equivalent to 1% delta k/k at 200°F within the next 6 hours.~~

Deleted

that the mass addition from the inadvertent operation of safety injection will not result in RHR system pressure exceeding design limits. The limitation on no safety injection pumps operable and the discharge lines isolated when overpressure protection is provided by the pressurizer PORV's removes mass injection from inadvertent safety injection as an event for which this configuration of overpressure protection must be designed to protect. Inoperability of a safety injection pump may be verified from the main control board with the pump control switch in pull stop, or the pump breaker in the test racked out position such that the pump could not start from an inadvertent safety injection signal. Isolation of a safety injection pump discharge path to the RCS may be verified from the main control board by the discharge MOV switch position indicating closed, or the discharge valve closed with A.C. power removed, or a manual discharge path isolation valve closed such that operation of the associated safety injection pump would not result in mass injection to the RCS.

~~The limitation on boric acid storage tank volume is based on the assumption that 2000 gallons of 12% to 13% solution is delivered to the RCS during a large steam line break associated with the containment integrity analysis.[10] The 3110 gallons specified is sufficient to accommodate the losses associated with the recirculation flow to the RWST and the sweep volume in the SI pump suction line and still deliver 2000 gallons to the RCS.~~

High concentration boric acid is not needed to mitigate the consequences of a design basis accident. Reference (10) demonstrates that the design basis accidents can be mitigated by safety injection flow of RWST concentration. Therefore, SI pump suction is taken from the RWST. Requiring that the safety injection suction valves (825A and B, 826A, B, C and D) are aligned with A.C. power removed ensures that the safety injection system would not be exposed to high concentration boric acid and the assumptions of the accident analysis are satisfied.

<u>References</u>

(1) Deleted

(2) UFSAR Section 6.3.3.1

(3) UFSAR Section 6.2.2.1

(4) UFSAR Section 15.6.4.3

(5) UFSAR Section 9.2.2.4

(6) UFSAR Section 9.2.2.4

(7) Deleted

(8) UFSAR Section 9.2.1.2

(9) UFSAR Section 6.2.1.1 (Containment Integrity) and UFSAR Section 6.4 (CR Emergency Air Treatment)

(10) ~~Westinghouse Analysis, "Report for the BAST Concentration for R.E. Ginna", August 1985 submitted by RG&E letter from R.W. Kober to H.R. Denton, dated October 16, 1985.~~

Westinghouse Report, "R.E. Ginna Boric Acid Storage Tank Boron Concentration Reduction Study" dated Nov. 1992 by C.J. McHugh and J.J. Spryshak

TABLE 4.1[illegible] (Continued)

Cha[illegible] Description	Check	Calibrate	Test	Remarks
10. Rod Position Bank Counters	S(1,2)	N.A.	N.A.	1) With rod position indication 2) Log rod position indications each 4 hours when rod deviation monitor is out of service
11. Steam Generator Level	S	R	M	
12. Charging Flow	N.A.	R	N.A.	
13. Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14. Boric Acid Storage Tank Level	D	R	N.A.	~~Bubbler tube Rodded weekly~~ Note 4
15. Refueling Water Storage Tank Level	N.A.	R	N.A.	
16. Volume Control Tank Level	N.A.	R	N.A.	
17. Reactor Containment Pressure	D	R	M(1)	1) Isolation Valve signal
18. Radiation Monitoring System	D	R	M	Area Monitors R1 to R9, System Monitor R17
19. Boric Acid Control	N.A.	R	N.A.	
20. Containment Drain Sump Level	N.A.	R	N.A.	
21. Valve Temperature Interlocks	N.A.	N.A.	R	
22. Pump-Valve Interlock	R	N.A.	N.A.	
23. Turbine Trip Set-Point	N.A.	R	M(1)	1) Block Trip
24. Accumulator Level and Pressure	S	R	N.A.	

Channel Description		Check	Calibrate	Test	Remarks
39.	Reactor Trip Breakers	N.A	N.A.	M	Function test - Includes independent testing of both undervoltage and shunt trip attachment of reactor trip breakers. Each of the two reactor trip breakers will be tested on alternate months.
40.	Manual Trip Reactor	N.A.	N.A.	R	Includes independent testing of both undervoltage and shunt trip circuits. The test shall also verify the operability of the bypass breaker.
41a.	Reactor Trip Bypass Breaker	N.A.	N.A.	M	Using test switches in the reactor protection rack manually trip the reactor trip bypass breaker using the shunt trip coil.
41.b	Reactor Trip Bypass Breaker	N.A.	N.A.	R	Automatically trip the undervoltage trip attachment.

NOTE 1: Logic trains will be tested on alternate months corresponding to the reactor trip breaker testing. Monthly logic testing will verify the operability of all sets of reactor trip logic actuating contacts on that train (See Note 3). Refueling shutdown testing will verify the operability of all sets of reactor trip actuating contacts on both trains. In testing, operation of one set of contacts will result in a reactor trip breaker trip; the operation of all other sets of contacts will be verified by the use of indication circuitry.

NOTE 2: Testing shall be performed monthly, unless the reactor trip breakers are open or shall be performed prior to startup if testing has not bee performed within the last 30 days.

NOTE 3: The source range trip logic may be excluded from monthly testing provided it is tested within 30 days prior to startup.

NOTE 4: When BAST is required to be operable.

TABLE 4.1-2

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

		Test	Frequency	~~FSAR Section Reference~~
1.	Reactor Coolant Chemistry Samples	Chloride and Fluoride Oxygen	3 times/week and at least every third day 5 times/week and at least every second day except when below 250°F	
2.	Reactor Coolant Boron	Boron Concentration	Weekly	
3.	Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly	
4.	Boric Acid Storage Tank	Boron Concentration	Twice/Week(4)	
5.	Control Rods	Rod drop times of all full length rods	After vessel head removal and at least once per 18 months (1)	~~7~~
6a.	Full Length Control Rod	Move any rod not fully inserted a sufficient number of steps in any one direction to cause a change of position as indicated by the rod position indication system	Monthly	~~7~~
6b.	Full Length Control Rod	Move each rod through its full length to verify that the rod position indication system transitions occur	Each Refueling Shutdown	
7.	Pressurizer Safety Valves	Set point	Each Refueling Shutdown	~~4~~
8.	Main Steam Safety Valves	Set point	Each Refueling Shutdown	~~10~~
9.	Containment Isolation Trip	Functioning	Each Refueling Shutdown	~~5~~
10.	Refueling System Interlocks	Functioning	Prior to Refueling Operations	~~9.4.5~~

	Test	Frequency	~~FSAR Section Reference~~
11. Service Water System	Functioning	Each Refueling Shutdown	~~9.5.5~~
12. Fire Protection Pump and Power Supply	Functioning	Monthly	~~9.5.5~~
13. Spray Additive Tank	NaOH Concent	Monthly	~~7~~
14. Accumulator	Boron Concentration	Bi-Monthly	~~6~~
15. Primary System Leakage	Evaluate	Daily	~~4~~
16. Diesel Fuel Supply	Fuel Inventory	Daily	~~8.2.3~~
17. Spent Fuel Pit	Boron Concentration	Monthly	~~9.5.5~~
18. Secondary Coolant Samples	Gross Activity	72 hours (2) (3)	
19. Circulating Water Flood Protection Equipment	Calibrate	Each Refueling Shutdown	

Notes:

(1) Also required for specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods.

(2) Not required during a cold or refueling shutdown.

(3) An isotopic analysis for I-131 equivalent activity is required at least monthly whenever the gross activity determination indicates iodine concentration greater than 10% of the allowable limit but only once per 6 months whenever the gross activity determination indicates iodine concentration below 10% of the allowable limit.

(4) When BAST is required to be operable.