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January 13, 1993

Docket No. 50-336
B14330

Re: NRC IEB 80-04

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Inspection and Enforcement Bulletin 80-04
Revised Response

In letters dated January 25, 1980[1] and July 6, 1982,[2] Northeast Nuclear Energy Company (NNECO) responded to NRC Inspection and Enforcement Bulletin (IEB) 80-04. Subsequent to these letters, analysis associated with the Millstone Unit No. 2 steam generator replacement project (SGRP) determined that portions of these previous IEB 80-04 responses were in error. The purpose of this letter is to submit the revised response to IEB 80-04, as originally committed in Licensee Event Report (LER) 91-10-00, dated November 15, 1991,[3] and subsequently revised based on telephone conversation with NRC Region I Staff.

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(1) W. G. Counsil letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2 Automatic Initiation of Auxiliary Feedwater," dated January 25, 1980.

(2) W. G. Counsil letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2 I&E Bulletin 80-04 on Main Steam Line Break With Continued Feedwater Addition—Response to Request for Additional Information," dated July 6, 1982.

(3) S. E. Scace letter to U.S. Nuclear Regulatory Commission, "Facility Operating Licensing No. DPR-65, Docket No. 50-336, Licensee Event Report 91-010-00," dated November 15, 1991.

Background

In letters dated September 13, 1979,[4] October 22, 1979,[5] and October 30, 1979,[6] the NRC Staff informed NNECO of the original Staff requirements for automatic initiation of auxiliary feedwater. NNECO's then ongoing evaluations of the requirement were discussed in detail in a letter dated November 30, 1979.[7] In a letter dated December 21, 1979,[8] the NRC Staff acknowledged correspondence transmitted by NNECO regarding automatic initiation of auxiliary feedwater systems upon the loss of main feedwater flow. The Staff acknowledged that NNECO had submitted the correspondence in response to Short-Term Recommendation 2.1.7.a, "Auto Initiation of the Auxiliary Feedwater System." During preparation of that submittal, NNECO raised the issue of the applicability of the then current main steam line break (MSLB) or main feedwater line break analysis, assuming early initiation of auxiliary feedwater flow with a failure to limit flow to the affected steam generator. The basic question concerned whether the changes in assumptions would increase the calculated containment pressure or the likelihood of return to power. In the December 21, 1979,[9] letter, the Staff requested, among other things, that NNECO resolve the concern by submitting analysis for Staff review.

In response to the NRC Staff request for information on automatic initiation of the auxiliary feedwater system, the design basis steam line break analysis was reevaluated. In the analysis, the additional mass releases to the containment due to auxiliary feedwater addition were added to the Final Safety Analysis Report (FSAR) case and shown to have no impact on the peak

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(4) D. G. Eisenhut letter to All Operating Nuclear Power Plants, "Followup Actions Resulting From the NRC Staff Revisions Regarding the Three Mile Island Unit 2 Accident," dated September 13, 1979.

(5) D. G. Eisenhut letter to W. G. Council, "NRC Requirements for Auxiliary Feedwater Systems at Millstone Nuclear Power Station Unit 2," dated October 22, 1979.

(6) H. R. Denton letter to All Operating Nuclear Power Plants, "Discussion of Lessons Learned Short-Term Requirements," dated October 30, 1979.

(7) W. G. Counsil letter U.S. Nuclear Regulatory Commission (J. M. Hendrie), "Haddam Neck Plant, Millstone Nuclear Power Station, Unit No. 2," dated November 30, 1979.

(8) R. W. Reid letter to W. G. Counsil, "Automatic Initiation of Auxiliary Feedwater Systems at Haddam Neck and Millstone Unit No. 2," dated December 21, 1979.

(9) Ibid.

containment pressure and temperature. Since this study was aimed at only assessing the impact of the new automatic feedwater initiation system, the original FSAR assumptions were not reevaluated. This was supported by evaluations performed by the nuclear steam supply system vendor, Combustion Engineering. This comprehensive analysis was submitted to the NRC Staff in a letter dated January 25, 1980.[10] Since the information requested in IEB 80-04 issued in February 1980 was very similar to the request made in December 1979, NNECO presumed that this analysis was also sufficient to respond to the Bulletin. Therefore, no new analysis was performed for the Bulletin. A Safety Evaluation Report was received from the NRC Staff on October 7, 1982.[11]

On October 18, 1991, at 1305 hours, with the plant in Mode 1 at 100 percent power, a reportability determination was made concerning a reanalysis of the MSLB event inside the containment. The reanalysis indicated that the assumptions made for the existing (1979) MSLB analysis were nonconservative with respect to power level, break size, and single active failure. Using more restrictive assumptions, design limits for containment pressure and temperature could have been exceeded. NNECO determined that this condition was reportable as a condition outside the design basis of the plant. An Immediate Report was made to the NRC, and the unit immediately commenced an orderly downpower to approximately 3 percent power (Mode 2). The existing MSLB analysis remained valid for Mode 2 operation.

A Justification for Continued Operation (JCO) was developed to allow the unit to return to power operation by stationing a dedicated reactor operator to close the main feedwater block valves following any reactor trip. That JCO documented the basis for reasonable assurance that, with the actions of a dedicated operator, containment pressure would remain below the design basis value for all postulated MSLB events. The unit was returned to power operation on October 22, 1991. The details of this discovery were discussed in LER 91-010-00, dated November 15, 1991.[12]

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(10) W. G. Counsil letter to U.S. Nuclear Regulatory Commission (R. Reid), "Millstone Nuclear Power Station, Unit No. 2 Automatic Initiation of Auxiliary Feedwater," dated January 25, 1980.

(11) R. A. Clark letter to W. G. Counsil, "Resolution of Main Steam Line Break With Continued Feedwater Addition Even for Millstone Nuclear Power Station, Unit No. 2," dated October 7, 1982.

(12) S. E. Scace letter to U.S. Nuclear Regulatory Commission, "Facility Operating License No. DPR-65, Docket No. 50-336, Licensee Event Report 91-010-00," dated November 15, 1991.

Later, in supplemental LER 91-010-01, dated January 17, 1992,[13] NNECO informed the NRC Staff that the short-term corrective actions, to close the main feedwater block valves given a containment isolation actuation signal (CIAS), had been installed and tested in December 1991. These changes eliminated the need for the dedicated operator. That LER also informed the Staff that changes for which hardware was available would be installed during the 1992 refueling outage. The remaining long-term hardware changes (i.e., qualified replacement components) would be completed during the 1994 refueling outage.

During the refueling outage, on August 4, 1992, at 1600 hours, with the plant in Mode 6 at 0 percent power and all fuel stored in the spent fuel pool, a new reportability determination was made which identified two new postulated single failures for the MSLB event inside the containment which resulted in the calculated containment pressure exceeding the design pressure limit.

The first was a failure of the feedwater regulating bypass valve to terminate flow to the affected steam generator. As part of the October 1991 MSLB evaluations, a failure of feedwater regulating bypass valve to close was considered, but was assumed to only provide 10 percent of full power feedwater flow to the affected steam generator. This assumption was consistent with the original analysis and resulted in acceptable containment pressures. Therefore, no provision was made in the JCO, or plant modifications proposed, to isolate the bypass feedwater flowpath. In response to the October 1991 determination, a reanalysis of containment pressure response to a MSLB had been initiated. In June 1992, the reanalysis of the failure of the bypass valve to close was performed using actual condensate pump curves and feedwater regulating bypass valve flow characteristics. This reanalysis resulted in the calculated containment pressure exceeding the 54 psig design limit.

The second was a failure of the vital buses to fast transfer to the reserve station services transformer (RSST). In that case, power to the condensate pumps would remain available, while power to close the feedwater regulating valves and start the containment pressure control systems would be delayed, due to diesel start and sequencing times. These delay times were not previously considered in the MSLB analysis. NNECO committed, in LER 91-010-02, dated September 3, 1992,[14] to perform plant modifications, during the outage then in progress, to ensure an acceptable containment

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(13) S. E. Scace letter to U.S. Nuclear Regulatory Commission, "Facility Operating License No. DPR-65, Docket No. 50-336, Licensee Event Report 91-010-01," dated January 17, 1992.

(14) S. E. Scace letter to U.S. Nuclear Regulatory Commission, "Facility Operating License No. DPR-65, Docket No. 50-336, Licensee Event Report 91-010-02," dated September 3, 1992.

pressure response for a MSLB inside containment given these newly identified failures.

The October 1991 MSLB evaluations also considered the possibility of various loss of power cases. Proposed plant modifications would have eliminated all of the cases identified at that time. However, the newly identified single failure of the vital buses to fast transfer to the RSST introduced delay times that were not previously considered in the MSLB analyses.

In response to the August 4, 1992, reportability determination, a multi-disciplinary task force was established to investigate the issue, ensure that all required single failures were considered, and to propose modifications which would ensure that the containment response is acceptable. Various design modifications were proposed and evaluated. Based upon the evaluations concerning all identified single failures, plant modifications have been completed which included adding redundant main steam isolation (MSI) signals to MSI actuated components; adding MSI signals to components which did not receive an MSI signal; modifying the MSI logic to actuate on high containment pressure, as well as low steam generator pressure; upgrading power supplies to vital power for selected valves; lowering the containment spray actuation setpoint; and reinstalling the emergency diesel generator start on a safety injection actuation signal. NNECO believes that implementation of these modifications eliminated the need to upgrade 2-FW-42A and 2-FW-42B (feedwater block valves) to full safety grade status as stated in LER 92-010-01. Following the modifications made during the 1992 refueling outage, the predicted MSLB peak containment pressure and temperature will be equal to or less than 54 psig and 426°F. In a letter dated December 4, 1992,[15] NNECO provided additional information, as requested by NRC Staff during their review of the license amendment associated with the MSLB modifications. Attached to that letter was NNECO's evaluation of equipment qualification (EQ) for postulated MSLB. Based on this evaluation, the required safe shutdown electrical equipment will remain qualified.

Attachment 1 provides the revised response to IEB 80-04, based on a comprehensive containment reanalysis which was preliminarily reviewed by the NRC Staff on December 17, 1992. Attachment 2 provides an augmented legend to the EQ figure, as requested by NRC Staff, for containment response which was previously submitted in our December 4, 1992, letter.[16]

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(15) J. F. Opeka letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2 Proposed Revision to Technical Specifications Main Steam Line Break Design Limits Response to Request for Additional Information," dated December 4, 1992.

(16) Ibid.

Conclusion

The results in Attachment 1, Table 1, show that for all cases, the peak containment pressure is less than the containment design pressure of 54 psig. Figure 6 gives the bounding EQ containment temperature profile. The peak temperature for this profile is 426°F. NNECO notes that Technical Specification 5.2.2 specifies the containment building temperature limit as 289°F. Because the containment atmosphere exceeds 289°F for only a short period of time, the containment building remains well below 289°F. Further, NNECO has identified the need to revise the bases of various technical specifications to reflect the results of recent analyses and to improve consistency among the bases. The previously docketed EQ evaluation indicates that the safety-related equipment is qualified for this temperature peak due to the very brief duration during which the temperature spike exists.

If you have any questions on this issue, please contact my staff.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

FOR: J. F. Opeka
Executive Vice President

BY: W D R

W. D. Romberg
Vice President

cc: T. T. Martin, Region I Administrator
G. S. Vissing, NRC Project Manager, Millstone Unit No. 2
P. D. Swetland, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3

Subscribed and sworn to before me

this 13th day of January, 1993

Ruth J. Dubois
Notary Public

Date Commission Expires: 3/31/95

Attachment 1

Revised Response to IEB 80-04

January 1993

Revised Response to Bulletin 80-04

1.0 CONTAINMENT PRESSURE RESPONSE

Item #1 of IE Bulletin No. 80-04 pertains to containment response for a main steam line break (MSLB) event. Specifically, Item #1 states:

> Review the containment pressure response analysis to determine if the potential for containment overpressure for a main steam line break inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources, such as continuation of feedwater or condensate flow. In your review, consider your ability to detect and isolate the damaged steam generator from these sources and the ability of the pumps to remain operable after extended operation at runout flow.

The analysis of record was reviewed during the reanalysis performed in support of the planned steam generator replacement. It was determined that nonconservative assumptions had been made with respect to power level, break size, and single active failure. Interim analyses, using more appropriate assumptions with respect to these parameters, predicted that design limits for containment pressure and temperature would be exceeded before the damaged steam generator could be isolated and containment spray would be effective.

A review of our previous response to IE Bulletin 80-04 determined that, due to these nonconservative assumptions, continuation of feedwater could occur, which had not been analyzed. As described in detail in Section 3.0, this finding required corrective actions to ensure isolation of feed sources under any single failure scenario. The corrective actions were put in place in three phases. The first phase involved stationing a dedicated operator to close the main feedwater block valves on any reactor trip. The second phase involved replacing the dedicated operator by wiring Containment Isolation Actuation Signals to the main feedwater block valves. The third phase involved completing an in-depth review to determine if other single failures existed, analyses to evaluate the consequences of those failures, and implementing hardware modifications as necessary.

2.0 CORE REACTIVITY RESPONSE

Item #2 of IE Bulletin No. 80-04 pertains to the core reactivity analysis for a MSLB event. Specifically, Item #2 states:

> Review your analysis of the reactivity increase which results from a main steam line break inside or outside containment. This

review should consider the reactor cooldown rate and the potential for the reactor to return to power with the most reactive control rod in a fully withdrawn position. If your previous analysis did not consider all potential water sources (i.e., runout from the auxiliary feedwater system, continuation of feedwater or condensate flow) and if the reactivity increase is greater than previous analysis indicated, the report of this review should include:

a. The Boundary conditions for the analysis, e.g., the end of life shutdown margin, the moderator temperature coefficient, power level and the net effect of the associated generator water inventory on the reactor system cooling, etc.

b. The most restrictive single active failure in the safety injection system and the effect of that failure on delaying the delivery of high concentration boric acid solution to the reactor coolant system,

c. The effect of extended water supply to the affected steam generator on the core criticality and return to power,

d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn position at the end of life, and Minimum Departure from Nucleate Boiling Ratio (MDNBR) values for the analyzed events.

The analysis of record for the core was reviewed relative to the above items and the following responses are provided:

a. The end-of-cycle shutdown margin supported in the analysis or record is the Technical Specification value of 3.6 percent$\Delta\rho$. The end-of-cycle moderator temperature coefficient is conservatively modeled as -28 pcm/°F. The analysis of record considers four cases, i.e., HFP and HZP both with and without offsite power. Analysis of this event for these four cases is judged to bound the consequences at intermediate initial power levels. The net effect of the steam generator water inventory determines the potential overcooling that the primary coolant system will sustain which adversely impacts the reactivity transient.

b. The single failure assumed in the analysis of record is the loss of one of two HPSI pumps. The time delay assumed in the analysis of record is 45 seconds for the HPSI pumps, for loss of offsite power cases. The time delay assumed for cases with offsite power is 30 seconds for the HPSI pumps. The analysis accounts for the delay that occurs from flushing the nonborated water from the safety injection lines prior to boron injection. These assumptions delay the time that highly concentrated boron is injected into the core and, thus, exacerbate the return-to-power.

An additional single failure that could impact the core response to a MSLB was identified. Following a reactor trip, plant electrical loads will fast transfer from the normal station services transformer to the reserve station services transformer. Failure of bus 24G would result in power remaining to buses 25A and B which power the reactor coolant pumps. However, power would be lost to the vital buses 24C and D until the diesels are started. The HPSI pumps would be sequenced on after the diesels have energized the buses.

This single failure affects cases when offsite power is available by adding a diesel start time delay to the actuation of the safety injection system (SIS). This single failure assumption could be more limiting than the limiting analysis of record (i.e., loss of one HPSI). For cases with offsite power available, the analysis of record supports a HPSI delay time of 30 seconds. With the plant changes implemented in the 1992 refueling outage, the Technical Specification time delay that includes the start of a diesel generator is 25 seconds for the HPSI pumps. This revised time delay is clearly bounded by the analysis of record. Additionally, the reactivity transient will be less limiting relative to the analysis of record since two HPSI pumps would be available to deliver borated water to the core sooner and with more capacity. Thus, failure of nonvital bus 24G is considered less limiting than the loss of a HPSI pump in the analysis of record.

c. The primary effect of an extended water supply to the affected steam generator is an increase in the duration of the event with potentially a higher return-to-power. Extended water supply can also slightly reduce the primary system pressure which can lead to an earlier SIAS. The analysis of record specifically includes the effect of runout flow from the auxiliary feedwater system after 180 seconds.

A failure of the feedline isolation system can occur only at initial power conditions above HZP when the main feedwater system is operating (e.g., HFP). A single failure in the feedwater isolation system that results in extended feedwater supply will contribute to the cooldown of the primary system by prolonging the event which can lead to a high return-to-power. The changes to the feedline isolation system in the 1992 refueling outage ensure a redundant valve alignment and a change to the power supply to the feedwater regulating valves (vital instrument AC powered from vital DC buses). Also, the feedwater regulating valve bypass valves fail in a closed position if power is lost. Thus, a single failure that leads to a continuous supply of feedwater has a low probability of occurrence. However, if a feedline failure that results in an extended water supply were to occur, the additional overcooling would be counteracted by the slower energy release through the integral flow restriction in the replacement steam generators relative to the analysis of record. Also, diesel generator start on SIAS, faster diesel generator start time, and the availability of the HPSI pumps would lead to a less severe reactivity transient compared to the bounding case for the analysis of record.

d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn position and Minimum Departure from Nucleate Boiling Ratio (MDNBR) values for the analyzed events are given below. MDNBRs were calculated with the modified Barnett correlation with a 95/95 limit of 1.135. The limiting MDNBR case for the analysis of record was initiated from HZP with a coincident loss of offsite power. The limiting peak linear heat rate (LHR) was calculated for the HZP case with offsite power and is less than the conservative centerline melt limit of 21 kW/ft.

Initial Power	Offsite Power	Hot Channel Factor	Max. LHR kW/ft	MDNBR
HZP	Y	10.7	20.9	2.40
HZP	N	8.7	16.5	1.18
HFP	Y	12.8	17.1	3.00
HFP	N	13.5	5.7	4.60

In summary, the existing MSLB analysis for core response predicts a slight return to power, but with no failure of fuel. Assumptions used in the analysis were sufficiently bounding that return to power is already maximized.

Those plant design changes which were determined necessary for containment analysis have not been incorporated into the core response MSLB, although doing so would produce less impact to the core.

3.0 CORRECTIVE ACTIONS

As discussed in Section 1.0, corrective actions were necessary to prevent exceeding the containment design pressure and temperature. Since this condition was discovered while the plant was in Mode 1 at 100 percent power, interim action was necessary until permanent modifications could be implemented. As described in LER 91-010-00, dated November 15, 1991,[1] the interim action included testing the main feedwater block valves to demonstrate adequate closure time and stationing a dedicated operator to close the main feedwater block valves following any reactor trip. The need for the dedicated operator was subsequently eliminated by the implementation of a short-term modification which caused automatic closure of the main feedwater block valves given a containment isolation actuation signal (CIAS). This action was described in LER 91-010-01, dated January 17, 1991.[2] That LER also informed the Staff that changes, for which hardware was available, would be installed during the 1992 refueling outage.

A description of the reanalysis to determine the effect of MSLB on the containment peak pressure, which incorporates the modifications made during the 1992 refueling outage, is presented in Section 3.1. As noted in Section 2.0, the existing MSLB analysis for core response does not require reanalysis and the plant modifications, which were necessary to obtain an acceptable containment response, would have a beneficial effect on the core response.

3.1 MSLB CONTAINMENT ANALYSIS

The analysis to determine the effect of a MSLB on the containment peak pressure and temperature was completed by ABB-CE in October 1992. This analysis reflects the design changes that were implemented in the 1992 refueling outage. Of these changes, the following were a direct result of the MSLB analysis:

1. Actuation of the main steam isolation signal (MSIS) upon a containment high pressure signal (CHPS) of 4.75 psig;

2. Automatic closure of the feed regulating valve (FRV) block valves (FW-42A and 42B) and the feed pump discharge valves

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(1) S. E. Scace letter to U.S. Nuclear Regulatory Commission, "Facility Operating License No. DPR-65, Docket No. 50-336, Licensee Event Report 91-010-00," dated November 15, 1991.

(2) S. E. Scace letter to U.S. Nuclear Regulatory Commission, "Facility Operating License No. DPR-65, Docket No. 50-336, Licensee Event Report 91-010-01," dated January 17, 1992.

(FW-38A and 38B) to include automatic actuation (shutting) on an MSIS;

3. Automatic actuation of MSIS components from either MSI-A or MSI-B;

4. Powering the FRVs from vital AC power, backed up with a DC alternate supply, and

5. Actuation of emergency diesel generator (EDG) start on SIAS.

In addition, the following Technical Specification changes have been made to ensure that the MSLB accident analysis is bounding:

1. Reduction of the minimum containment air recirculation (CAR) Fans starting delay time with normal AC power available to 15 seconds;

2. Reduction of the minimum containment spray starting delay time with normal AC power available to 16 seconds;

3. Reduction of the minimum EDG starting time to 15 seconds;

4. Reduction of the feedwater block valves, FRVs, feed pump discharge valves and FRV bypass valves closure times, including stroke time and signal delay time, to 14 seconds, and;

5. Reduction of the High-High containment pressure signals to 9.48 psig.

Method of Analysis

A complete MSLB spectrum study has been performed to determine the limiting cases for peak containment pressure and for environmental profiles for Electrical Equipment Qualification (EEQ). The NRC-approved methodology associated with the ABB-CE SGN-III computer program was used. This methodology includes consideration for the following: (a) inclusion of the steam line and feed line volumes into the overall determination of blowdown volume available; (b) determination of temperature/pressure expansion factors and manufacturing tolerances for the steam generators (SGs) and reactor coolant system (RCS); (c) feed spiking due to the increasing pressure imbalance between the ruptured and intact SG; (d) inclusion of current core physics and thermal-hydraulic data; (e) inclusion of SG shell metal heat transfer as part of the energy release; and lastly, (f) a complete determination of the effects of different component single failures during the accident. This methodology is consistent with the Standard Review Plan (SRP) guidance.

Major Assumptions

The major assumptions are as follows:

1. Offsite power is conservatively assumed to be available for most of the cases. This increases the primary to secondary heat transfer since the reactor coolant pumps (RCPs) are operating. To verify this assumption, loss of offsite power cases were included as part of the single failure analysis.

2. For determination of peak containment pressure, the initial containment pressure/temperature is conservatively assumed to be at the Technical Specification maximum of 16.8 psia and 120°F. For determination of peak containment temperature, the initial containment pressure was conservatively assumed to be 14.7 psia. With a lower initial pressure, containment spray will be actuated later, resulting in a high containment temperature. Relative humidity is assumed at 30 percent except for the EEQ cases where it is conservatively set at 100 percent.

3. Consistent with the NRC approved methodology, moisture carryover was determined by the SGN-III computer code.

4. Feedwater spiking is accounted for by conservatively doubling the initial feedwater flow rate for each case.

5. Credit is taken for the main steam nonreturn valves to prevent blowdown of the unaffected SG into the containment.

6. The maximum RCS flow rate was conservatively assumed to maximize the heat transfer from the primary to secondary side.

7. Auxiliary feedwater was conservatively assumed to initiate at 180 seconds based upon the time delay of 3 minutes given in Technical Specification Table 3.3-5.

8. RCP heat was included.

9. All actuation signals are redundant and safety grade. In some cases credit is taken for actuation of nonsafety grade components initiated by the safety grade signals. This is fully consistent with original license design bases where

nonsafety components have always been credited for feed isolation.

Results

In order to determine the limiting conditions, four different spectrum studies were performed. These are as follows:

1. Power level and break size

2. Feed system single failures

3. Containment heat removal systems single failures

4. EEQ spectrum study

A summary of the results of these studies is given in Table 1 and Figures 1 through 6. The results of the spectrum studies are summarized below:

Power level and Break Size

A comprehensive sensitivity study was performed to determine the limiting break size for each power level. A sensitivity study was needed because of the interaction of power level with SG inventory and moisture carryover. The limiting break size at a given power level is the largest break size that would result in a pure steam blowdown, since a pure steam blowdown results in the greatest amount of energy being transferred to the containment atmosphere in a short period of time. The limiting break sizes were shown to be 3.51 ft^2 at 50 percent power and above, 3.35 ft^2 at 25 percent, and 1.75 ft^2 at 0 percent. Results are shown as cases A1 through A5 in Table 1. No single failures are assumed in this set of cases.

Feed System Single Failures

A comprehensive single failure study was performed for feedwater system isolation. For each single failure, a range of power levels was analyzed, using the insights from the Power Level/Break Size sensitivity study. The results of this sensitivity study are shown as Cases B1 through B4 in Table 1 and are briefly described below.

B1. Feed Pump Failure to Trip - The failure of a feed pump to trip on MSIS results in additional feedwater being pumped preferentially into the affected SG until the FRV or the isolation valves shut. For power levels 50 percent and below, only one feedwater pump was assumed to be running when the accident commences.

B2. Auxiliary Feedwater Regulating Valve Fails Open - A failed open auxiliary feedwater (AFW) regulating valve was assumed to result in the addition of the maximum flow from the two electrically driven AFW pumps to the affected SG. AFW was assumed to commence at 180 seconds. The AFW flow addition results in a slow but steady increase in containment pressure and temperature. Credit is taken for operator action at ten minutes to isolate AFW to the affected steam generator. The results show that the operator has at least 15 minutes to isolate the affected generator.

B3. Feedwater Bypass Valve Fails Open - The failed open feedwater bypass valve results in additional feedwater being pumped preferentially into the affected SG until the feedwater pump discharge valves shut. In addition, even with the feed pump discharge valves shut, flashing in the feedwater lines continues to add energy into the affected steam generator. This effect has been taken into account. This case is the limiting MSLB for peak containment pressure. The plant response for this case is shown in Figures 1 through 5 and the sequence of events is given in Table 2.

B4. Failure of Vital Instrument AC Bus VA-10 or VA-20 - This failure could prevent closure of the FRVs and loss of one train of the containment heat removal systems. Feedwater addition to the affected SG will continue until closure of the feedwater isolation valves.

Containment Heat Removal Systems Single Failures

A comprehensive containment heat removal systems single failure study was performed. For each single failure, a range of power levels was analyzed, using the insights from the Power Level Break Size sensitivity study. The results of this sensitivity study are shown as Cases C1 through C5 in Table 1 and are briefly described below.

C1. Failure of Two CAR Fans to Start - This failure is bounded by Case B4 described above.

C2. Failure of One Spray Train to Start - This failure is bounded by Case B4 described above.

C3. Failure of the Vital Bus Transfer Mechanism - This failure results in a loss of the normal power supply for the vital buses. Thus, initiation of the containment sprays and CAR fans is delayed until the EDGs are powering the vital buses and auto sequencing has occurred. Since the FRV's power supplies are powered from a vital DC power source, they are unaffected by this failure and will isolate the affected SG. The RCPs and certain other nonvital loads are also unaffected by this failure, which contributes to the severity of this accident by providing more rapid heat transfer from the primary to the affected SG.

C4. Loss of Offsite Power with a Loss of One EDG - A loss of offsite power will result in loss of power to the RCPs, the condensate pumps, and feedwater heater drains pumps. While only one train of containment heat removal systems is available, the loss of power to these pumps results in a greatly degraded heat transfer in the affected SG and less limiting results. Feedwater isolation will be unaffected since the FRV's power supplies are powered by vital DC supplies.

C5. Loss of Offsite Power with a Loss of VA-10/20 - This case is similar to C4, with the exception of the effect on feedwater isolation to the affected SG. With this failure, there is failure of the FRV and the other isolation valves to close. However, with the loss of the condensate and feedwater heater drains pumps, feedwater addition to the affected SG is terminated. The effect of continued energy addition to the affected SG from flashing in the feedwater lines has been taken into account.

<u>EEQ Spectrum Study</u>

In order to develop a bounding profile for EEQ, the sensitivity studies (with the exception of Case B2) were repeated with initial conditions selected to maximize peak containment temperature. The major changes in assumptions for determining the bounding profile are as follows:

1. The initial containment pressure was reduced to 14.7 psia. This results in the maximum delay in containment spray actuation.

2. The relative humidity was increased to 100 percent.

3. The guidelines of NUREG-0588 and IE-IN #84-90 were used to set the containment wall reevaporization at 8 percent and

the modeling of SG superheating as it passes the uncovered portion of the SG tubes before exiting the break.

These assumptions result in some changes in the timing of the sequence of events. The results of these cases were used to generate the EEQ containment temperature profile. Case D1 shows the limiting peak containment temperature.

4.0 Conclusion

The results in Table 1 show that for all cases, the peak containment pressure is less than the containment design pressure of 54 psig. Figure 6 gives the bounding EEQ containment temperature profile. The peak temperature for this profile is 426°F. It is noted that Technical Specification 5.2.2 specifies the containment building temperature limit as 289°F. Because the containment atmosphere exceeds 289°F for only a short period of time, the containment building remains well below 289°F. Further, NNECO has identified the need to revise the bases of various Technical Specifications to reflect the results of recent analyses and to improve consistency among the bases.

TABLE 1

Case	Description	Power Level	Peak Pressure (psig)	Peak Temp (°F)	
A1	Base Case	102%	48.6	394	
A2	Base Case	75%	49.0	389	
A3	Base Case	50%	49.8	386	
A4	Base Case	25%	47.2	379	
A5	Base Case	0%	50.0	345	
B1	Feed Pump Fails to Trip	50%	52.5	385	
B2	AFW Regulating Valve Fails Open	0%	50.9 at 600 sec	345	
B3	Feed Regulating Bypass Valve Fails Open	50%	53.7	385	
B4	Vital Bus Cabinet (VA-10/20) Fails - Preventing FRV Closure and Eliminating 1/2 of the sprays and Fans	102%	53.3	395	
C1	Two Car Fans Fail to Start	0%	51.8	346	
C2	One Spray Train Fails to Start	102%	51.8	394	
C3	Vital Bus Fast Transfer Fails	102%	52.2	416	
C4	Loss of Offsite Power and Loss of one Diesel	25%	48.0	382	
C5	Loss of Offsite Power and Loss of VA-10/20	50%	51.20	386	
D1	EEQ Calculation - Vital Bus Fast Transfer Fails	102%	NA	426	

TABLE 2

SEQUENCE O F EVENTS
MP2-MSLB: FEED BYPASS FAILURE CASE @ 50% POWER

TIME (seconds)	EVENT	SETPOINT/VALUE
0.00	MSLB occurs from 50% power, break size is 3.51 ft.	
0.01	Feed spiking occurs which causes feed to ruptured SG to double from its initial flow.	
1.90	Containment High Pressure Signal (CHPS) is generated. This will cause a Reactor Trip and MSIS after a 1.15 second signal delay.	5.83 psig with uncert.
3.05	Reactor trip and turbine trip occur. An MSIS signal causes the feed pumps to trip off and FRV's, feed isolation valves begin closing. Feed flow begins ramping down as the feed pumps coast down. feed bypass valve fail open.	
5.28	Containment High-High pressure signal (CHHPS) occurs.	11.08 psig with uncert.
8.05	Feed pump coastdown to "low flow" condition complete.	
15.90	Feed isolation valves shut. Feed to the ruptured SG ceases.	
16.90	Containment cooling fans energize. Time based on CHPS + 15 second delay.	
35.58	Containment spray flow commences. Time based on CHHPS + 30.3 seconds for pump start, valve stroke time, and header fill time.	2600 gpm
35.58	Peak containment temperature reached.	385 °F
133.98	Peak containment pressure reached.	53.7 psig
180.00	Auxiliary feedwater flow commences to the ruptured SG.	600 gpm, 120 °F
600.00	Problem time ends.	

FIGURE 1

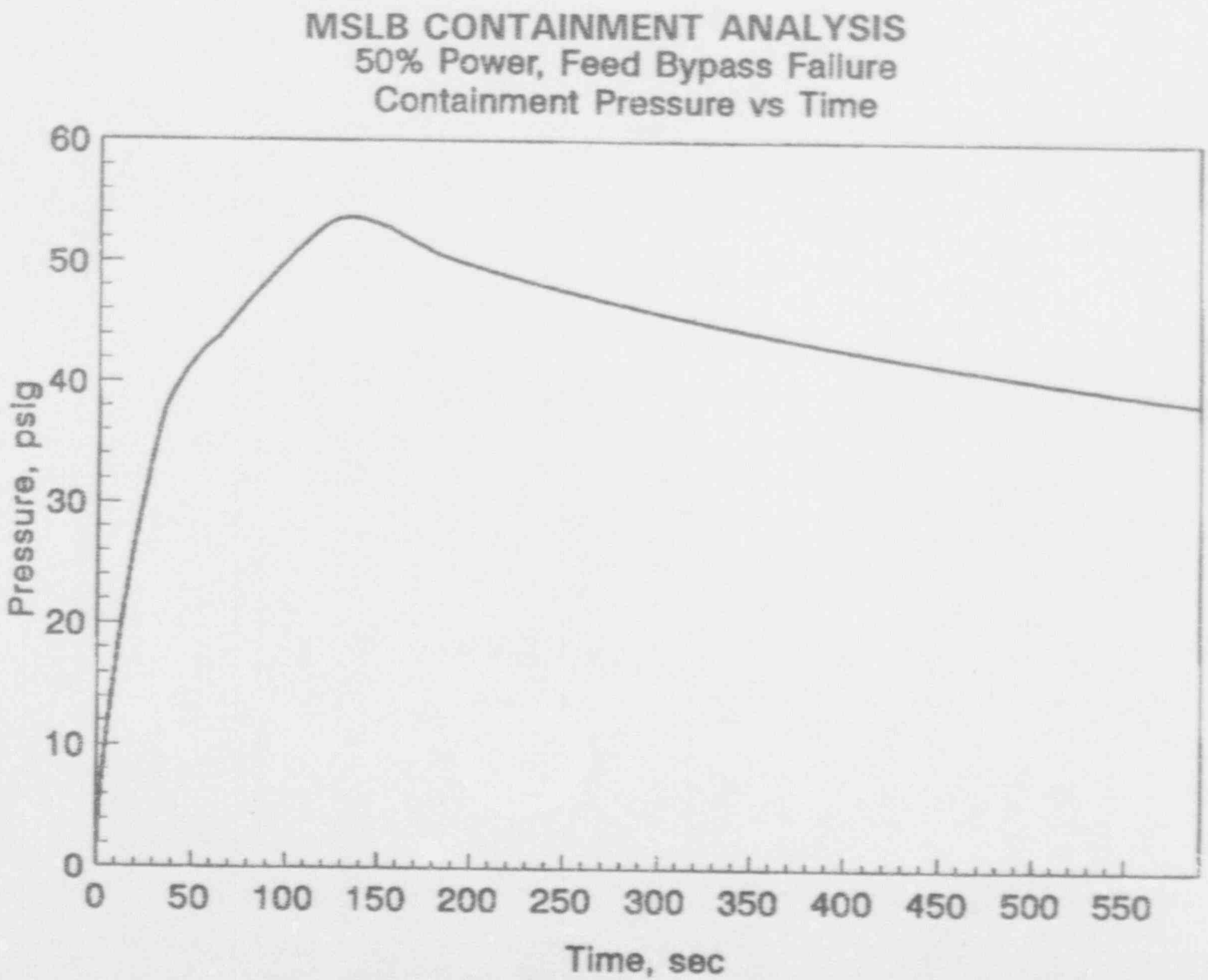

MSLB CONTAINMENT ANALYSIS
50% Power, Feed Bypass Failure
Containment Pressure vs Time
Pressure, psig
Time, sec
0
10
20
30
40
50
60
0
50
100
150
200
250
300
350
400
450
500
550

FIGURE 2

MSLB CONTAINMENT ANALYSIS
50% Power, Feed Bypass Failure
Containment Temperature vs Time

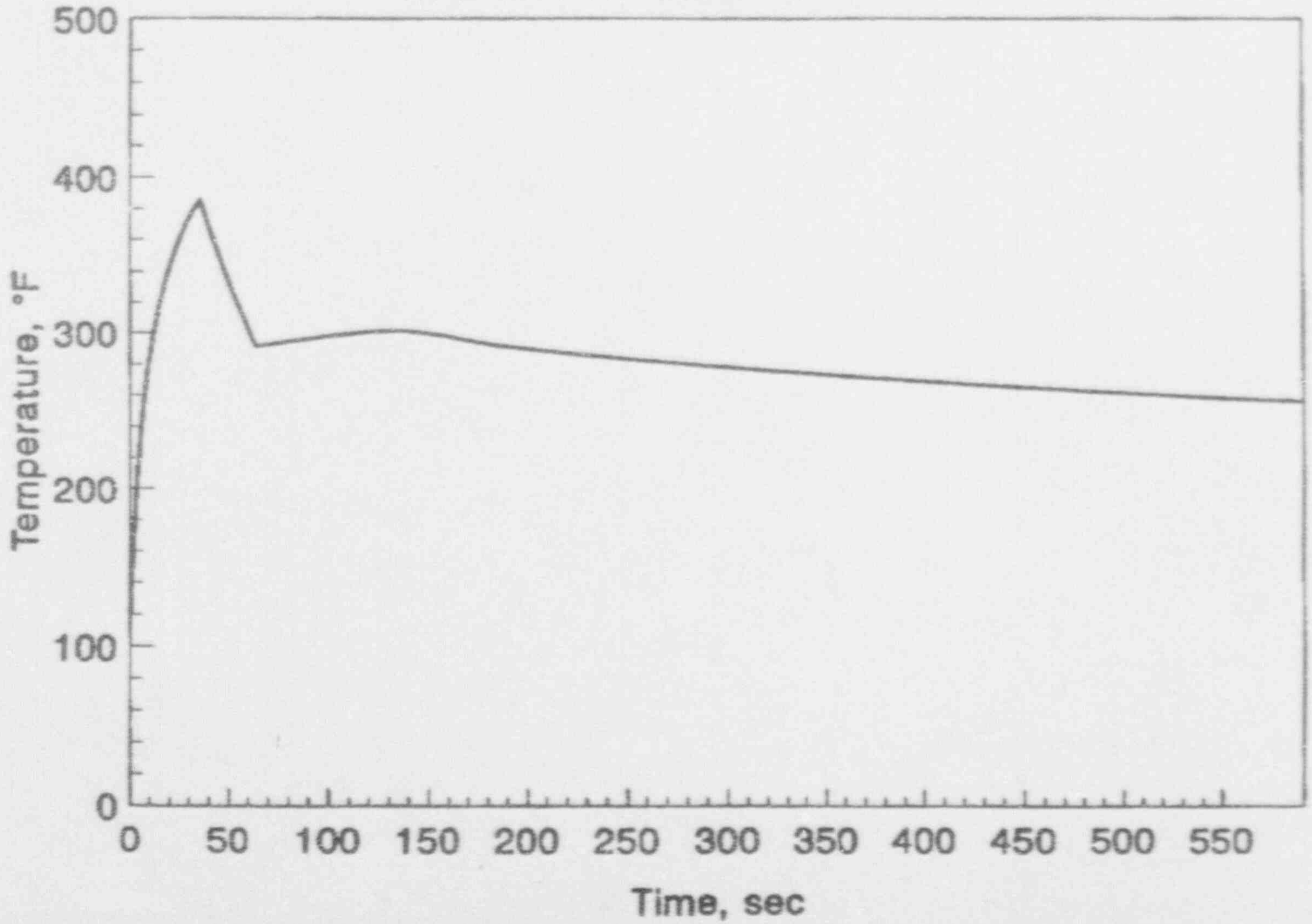


FIGURE 3

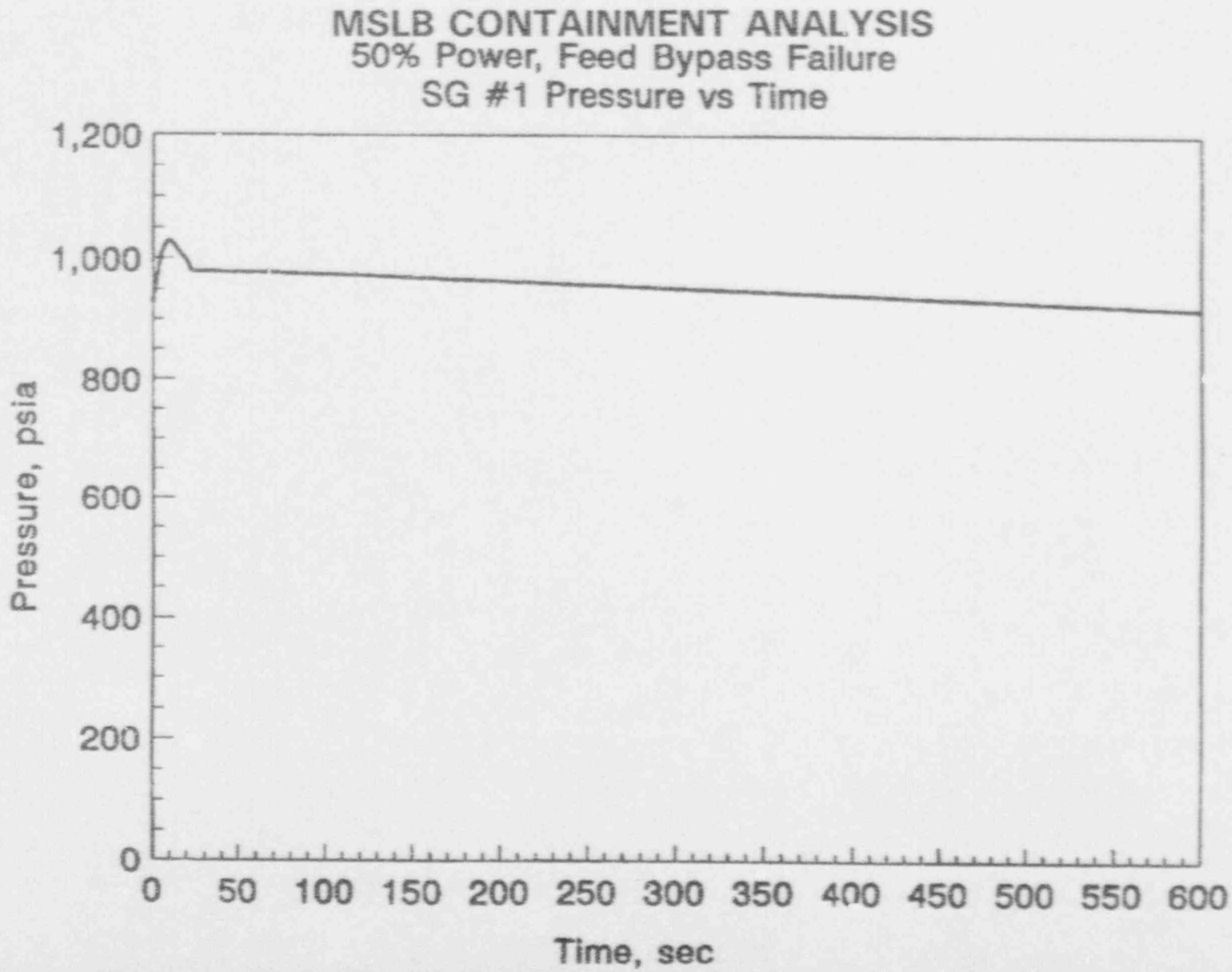

MSLB CONTAINMENT ANALYSIS
50% Power, Feed Bypass Failure
SG #1 Pressure vs Time
1,200
1,000
800
600
400
200
0
Pressure, psia
0
50
100
150
200
250
300
350
400
450
500
550
600
Time, sec

FIGURE 4

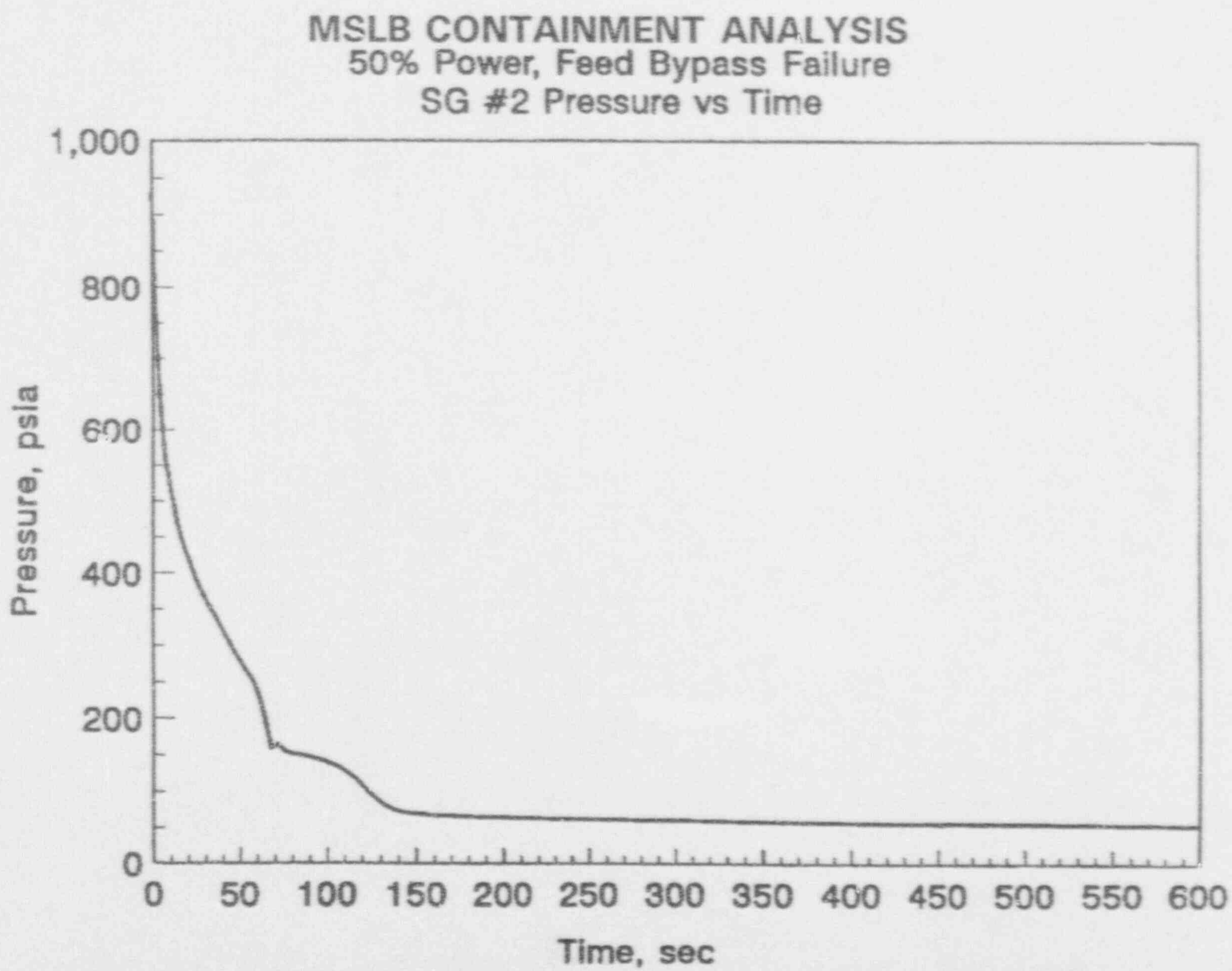

MSLB CONTAINMENT ANALYSIS
50% Power, Feed Bypass Failure
SG #2 Pressure vs Time
1,000
800
600
400
200
0
Pressure, psia
0
50
100
150
200
250
300
350
400
450
500
550
600
Time, sec

FIGURE 5

MSLB CONTAINMENT ANALYSIS
50% Power, Feed Bypass Failure
Two Phase Ruptured SG Water Volume vs Time

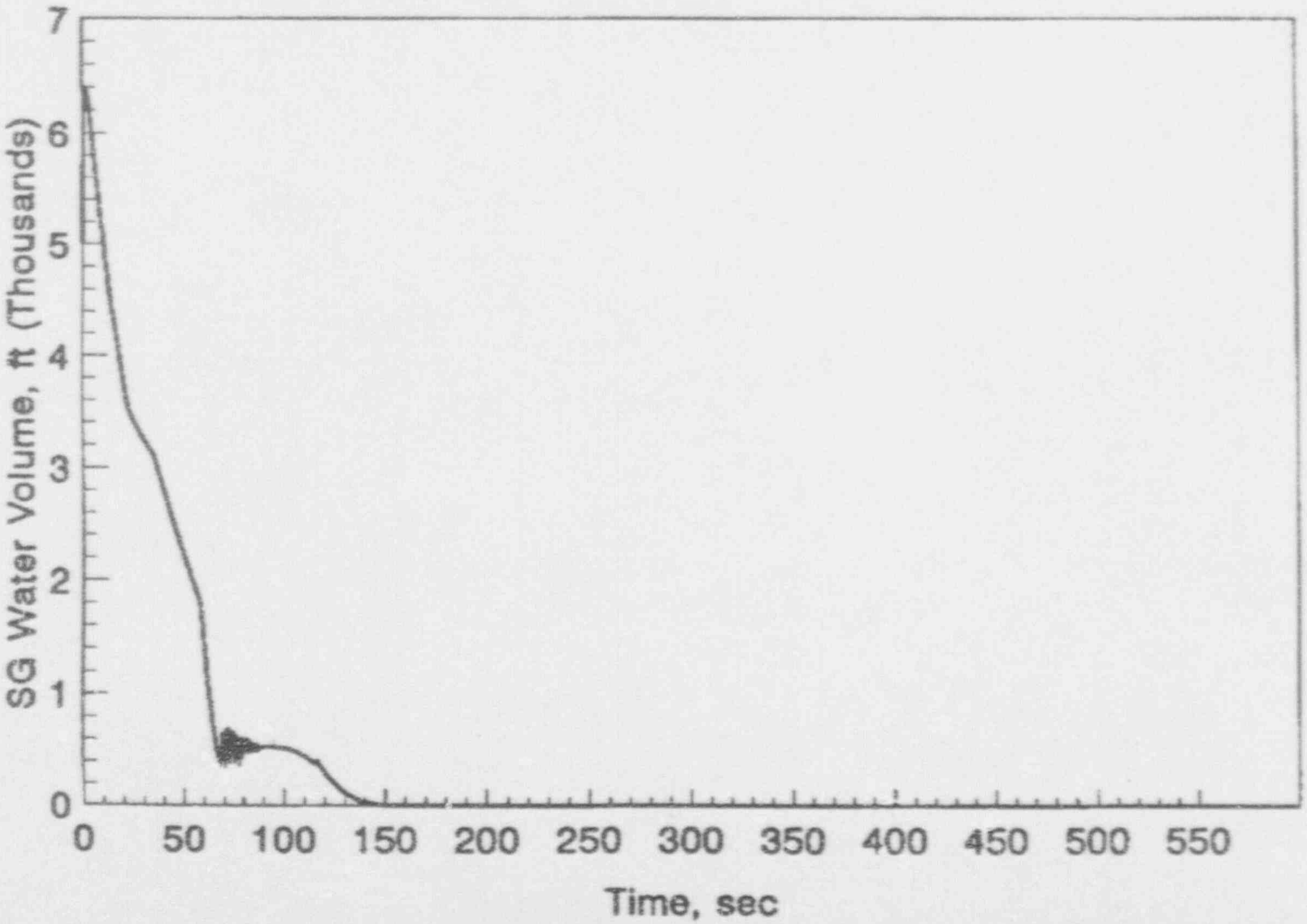
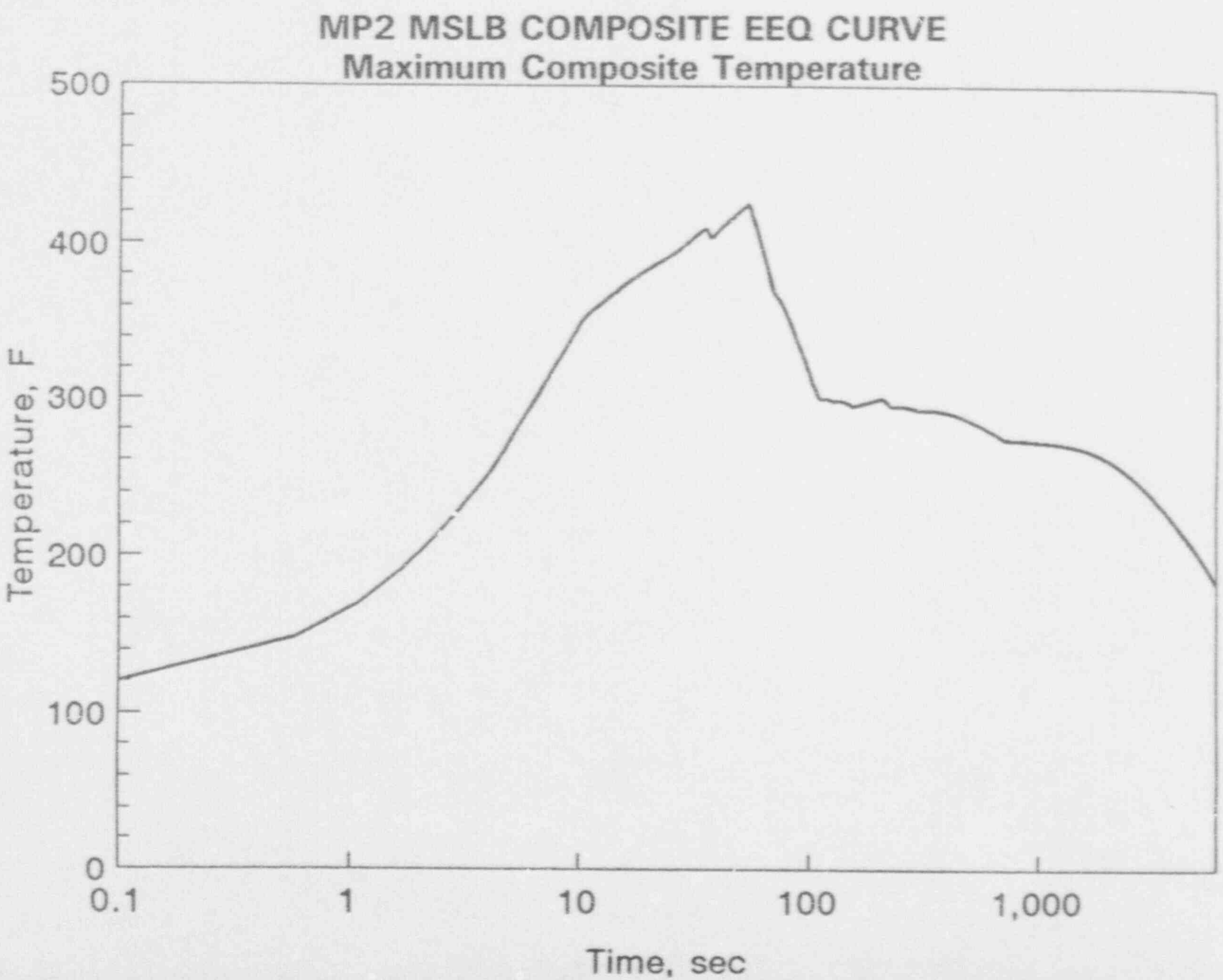


FIGURE 6


MP2 MSLB COMPOSITE EEQ CURVE
Maximum Composite Temperature
Temperature, F
500
400
300
200
100
0
0.1
1
10
100
1,000
Time, sec

Attachment 2

Augmented Legend for EQ Profile Figure

January 1993

Augmented Legend of EQ Profile Previously Submitted

103	General Atomic Rad Monitors
121	Rockbestos Coaxial
107FR	Kerite FR cable
116	Ideal Setscrews
107HTK	Kerite HTK Cable
111	ASCO Soleniods
125	Namco Limit Switch
122	Limitorque Motor Operators
115	Westinghouse Motors
102	Conax Penetrations
135	Anaconda Cable
134	Rosemount Transmitters
128	Litton Connectors
101	Gems-Delaval Transmitters
120	Litton Connectors
119	Weidmuller Terminal Blocks