September 15, 1982

Repro 9/20

Docket No. 50-245
LS05-82-09-045

Mr. W. G. Counsil, Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06101

Dear Mr. Counsil;

SUBJECT: SEP TOPICS VI-2.D AND VI-3 EVALUATION REPORT FOR MILLSTONE NUCLEAR POWER STATION, UNIT 1

Enclosed is a copy of our final evaluation of SEP Topics VI-2.D, "Mass and Energy Release for Possible Pipe Break Inside Containment," and VI-3, "Containment Pressure and Heat Removal Capability." This evaluation compares your facility, as described in Docket No. 50-245, with the criteria currently used by the regulatory staff for licensing new facilities. Appendix A to our evaluation is a Technical Evaluation Report from our contractor, Lawrence Livermore National Laboratory.

This evaluation will be a basic input to the Integrated Safety Assessment for your facility. This assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this subject are modified before the Integrated Assessment is completed.

Sincerely,

Original signed by:

James J. Shea, Project Manager
Operating Reactors Branch No. 5
Division of Licensing

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Enclosure:
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SAFETY EVALUATION REPORT

ON

CONTAINMENT PRESSURE AND

HEAT REMOVAL CAPABILITY

SEP TOPIC VI-3

AND

MASS AND ENERGY RELEASE

FOR POSSIBLE PIPE BREAK

INSIDE CONTAINMENT,

SEP TOPIC VI-2.D

FOR THE

MILLSTONE 1 NUCLEAR POWER PLANT

DOCKET NO. 50-245

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I. INTRODUCTION

The Millstone 1 Nuclear Power Station began commercial operations in 1971. Since then the staff's safety review criteria have changed. As part of the Systematic Evaluation Program (SEP), the mass and energy release for possible pipe break inside containment (Topic VI-2.D) and the containment pressure and heat removal capability (Topic VI-3) have been re-evaluated.

The purpose of this re-evaluation is to document all deviations from current safety criteria as they relate to the containment pressure and heat removal capability and the mass and energy release for possible pipe breaks inside containment. Furthermore, independent analyses in accordance with current criteria were performed to determine the adequacy of the containment design bases (e.g., design pressure and temperature). The significance of the identified differences, and recommended corrective measures to improve safety, will be the subject of a subsequent, integrated assessment of the Millstone 1 plant.

The SEP Analysis and Evaluation and plotted results are given in Appendix A.

II. REVIEW CRITERIA

The review criteria used in the current evaluation of SEP Topics VI-2.D and VI-3 for the Millstone 1 plant are contained in the following documents:

(1) 10 CFR Part 50, Appendix A, General Design Criteria (GDC) for Nuclear Power Plants:

(a) GDC 16 - Containment design;
(b) GDC 38 - Containment heat removal; and
(c) GDC 50 - Containment design basis.

(2) 10 CFR Section 50.46, "Acceptance Criteria for Emergency Core Cooling System for Light Water Nuclear Power Reactors".

(3) 10 CFR Part 50, Appendix K, "ECCS Evaluation Models".

(4) NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP 6.2.1, Containment Functional Design).

III. RELATED SAFETY TOPICS

The review areas identified below are not addressed in this report but are related to the SEP topics of mass and energy release for possible pipe breaks inside containment and/or containment pressure and heat removal capability.

(1) III-1, Classification of Structures, Components, and Systems (Seismic and Quality).

(2) III-12, Environmental Qualification of Safety-Related Equipment.

(3) VI-7.B, ESF Switch-over from Injection to Recirculation Mode (Automatic ECCS Realignment).

(4) IX-3, Station Service and Cooling Water Systems.

(5) X, Auxiliary Feedwater System.

(6) USI-A24, Qualification of Class 1E Safety-Related Equipment.

IV. REVIEW GUIDELINES

General Design Criterion (GDC) 16 of Appendix A to 10 CFR Part 50 requires that a reactor containment and associated systems shall be provided to establish a leak-tight barrier against the uncontrolled release of radioactivity to the environment. In addition, GDC 16 requires that the containment design conditions important to safety not be exceeded for as long as the postulated accident conditions require. GDC 38 requires a containment heat removal system to be provided whose safety function shall be to reduce the containment pressure and temperature following any loss-of-coolant accident (LOCA) and to maintain them at acceptably low levels; furthermore, this safety system shall function with a single failure. GDC 50 requires that the containment structure and the containment heat removal system shall be designed so the structure can accommodate, with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin was obtained from the conservative calculation of mass and energy release, and the containment model is discussed in the Standard Review Plan (SRP) Section 6.2.1, Containment Functional Design. The containment design basis includes the effects of stored and generated energy in the accident. Calculations of the energy available for release should be performed in accordance with the requirements of 10 CFR Part 50, Section 50.46, and Appendix K, paragraph I.A, and the conservatism specified in SRP 6.2.1.3. The mass and energy release to the containment from a LOCA should be considered in terms of the mass and energy release during blowdown. Break locations should include recirculation line breaks and steam line breaks. The review also includes the analysis of postulated single active failures of components in the secondary system.

By review of the licensee's analysis, deviations from the current criteria are identified and independent analyses are performed, as required, to evaluate the significance of these deviations. The evaluation is completed by comparing the results with the containment design bases.

V. EVALUATION

In the case of BWRs with Mark I containments, it is necessary to evaluate the effect of pipe breaks below the core for maximum containment pressure and pipe breaks above the core for maximum containment temperature. In the Millstone 1 FSAR, a full double-ended-guillotine (DEG) recirculation line break was analyzed to determine the containment design pressure. The initial and boundary conditions used by the applicant were reviewed. In exception to current design criteria, the FSAR analyses were performed at 100% reactor full power condition, not the 102% required in current criteria. However, later docket material in support of reload licensing shows reanalysis of the containment pressure based on 102% reactor full power conditions. This is acceptable by current NRC criteria.

The FSAR maximum calculated containment pressure is 43 psig in response to the design basis accident (DBA) LOCA break. This is well below the containment design pressure of 62 psig. A confirmatory analysis was performed and is presented in Appendix A of this report. The confirmatory analysis response was calculated using CONTEMPT-LT/028. The calculated maximum containment pressure was 44 psig. Hence, the confirmatory analysis agrees with the FSAR calculated maximum containment pressure due to a DBA LOCA, and both values are below the containment design pressure of 62 psig.

The Millstone 1 FSAR gives a containment design temperature of 281°F. The cofirmatory containment analysis in response to a 0.1 ft^2 MSLB gives a calculated maximum drywell temperature of 328°F. The design temperature is exceeded. However, based on the results of SEP Topic III-7.B, "Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria" it was demonstrated that the containment could withstand the effect temperatures of 330°F without structural degradation.

VI. CONCLUSIONS

The deviations of the Millstone 1 plant from current criteria have been identified in Section V, above. From the independent containment analyses reported in Appendix A, it is concluded that the Millstone 1 containment design pressure meets current criteria. It is evident from the three MSLB cases analyzed that the containment design temperature is exceeded by postulated medium size main steam line break accidents.

With respect to the related matter of equipment qualification (USI A-24), the staff analyses are conservative and may be used for defining containment service conditions, if more appropriate information is lacking.

APPENDIX A

SEP CONTAINMENT ANALYSIS AND EVALUATION

OF

MILLSTONE 1 NUCLEAR POWER STATION

1.0 INTRODUCTION AND BACKGROUND

As part of the Systematic Evaluation Program (SEP), the containment functional design capability of the Millstone 1 Nuclear Power Station has been reevaluated. The purpose of this report is to document the resolution of SEP Safety Topic VI-2.D, Mass and Energy Release for Possible Pipe Break Inside Containment, and Safety Topic VI-3, Containment Pressure and Heat Removal Capability, and deviations from current safety criteria as they relate to the containment functional design. The significance of the identified deviations and recommended corrective measures will be the subject of a subsequent integrated assessment of the Millstone 1 plant.

The containment structure encloses the reactor system and is the final barrier against the release of radioactive fission products in the event of an accident. The containment structure must, therefore, be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated LOCA and steam line break accidents. Furthermore, equipment having post-accident safety functions must be environmentally qualified for the resulting adverse pressure and temperature conditions.

2.0 CONTAINMENT FUNCTIONAL DESIGN

Millstone 1 is a 2011 MWt General Electric Mark I BWR which has a primary containment consisting of a drywell, pressure suppression chamber, and interconnecting vent pipes. The pressure suppression chamber is a steel pressure vessel in the shape of a torus located below and encircling the drywell. The chamber is approximately half filled with water. The vent system from the drywell terminates below the water level in the pressure suppression chamber, so that in the event of a pipe failure in the drywell, the released steam passes directly to the water where it is condensed. This transfer of energy to the water pool rapidly reduces the post-accident pressure in the drywell and substantially reduces the potential for subsequent leakage from the primary containment.

In addition to the pressure absorption chamber, independent auxiliary cooling systems are provided for the reactor and containment cooling under various normal and abnormal conditions. These are:

(1) A low pressure coolant injection (LPCI) containment cooling system which serves three functions:

  (a) To inject water into the reactor vessel subsequent to a postulated LOCA rapidly enough to reflood the core and prevent fuel clad melting.

  (b) To remove heat from the water in the suppression chamber.

  (c) To spray water into the drywell and/or the suppression chamber as an augmented means of removing energy from the containment as required.

(2) A shutdown cooling system to remove reactor decay heat during shutdown.

(3) An isolation condenser to remove decay heat from the core when the reactor is isolated.

(4) A feedwater coolant injection (FWCI) system to remove decay heat and to provide coolant inventory control and heat dissipation from the core to the suppression chamber under postulated small break accidents. If the FWCI system should fail to operate, an automatic depressurization by blowdown will be employed through automatic opening of relief valves which vent steam to the suppression pool. This blowdown will depressurize the vessel in sufficient time to allow the core spray or the low pressure coolant injection (LPCI) function of the ECCS to adequately cool the core and prevent any clad melting.

(5) Two core spray systems designed to pump water under accident conditions from the pressure suppression chamber pool directly to the reactor core by independent spray headers or spargers mounted in the reactor vessel above the core.

(6) An auxiliary coolant supply system via a cross-tie between the service water system and the condensate storage system which makes available an inexhaustable supply of cooling water from the Long Island Sound to the reactor core and containment independent of all other cooling water sources.

In the event of loss of offsite power and failure of one diesel generator, minimum containment cooling is provided by three low pressure coolant injection pumps. After the core is flooded, these pumps are manually switched from a core injection mode to a containment cooling mode. Water from the wetwell is passed through a heat exchanger and returned to the wetwell. A containment spray system is provided to spray cooled water to either the drywell, the wetwell, or to both.

2.1 Review of the Millstone 1 Containment Design Analysis

Two separate calculations make up the containment design analysis. First is the mass and energy release rate calculation for postulated LOCAs. This provides the time-dependent mass and energy input into the containment structure. Second is the calculation of containment response to the mass and energy input to the containment structure. This results in the time-dependent containment temperature and pressure profile. The severity of the containment response depends on the magnitude and nature of the break location. If the break is below the core, the break flow will initially be single-phase liquid. This results in a rapid blowdown of the mass and energy release to the containment at a relatively low enthalpy. If the break is above the core, the break flow will be mostly single-phase steam. This results in a much longer blowdown of the mass and energy release to the containment at a much higher enthalpy. Because of these effects, breaks below the core produce the most severe pressure responses in the containment, and steam line breaks above the core produce the most severe temperature responses.

The acceptance criteria used to evaluate the Millstone 1 Containment Design Analysis were based on the Standard Review Plan (SRP), Sections 6.2.1.1.C, 6.2.1.3, and 6.2.1.4. For the containment design analysis to be found acceptable, both the mass and energy release and the containment response calculations must meet the acceptance criteria specified in the SRP.

2.2 Review of Millstone 1 Primary System Pipe Breaks

The SRP specifies several acceptance criteria to be applied to the mass and energy release analysis for primary system pipe breaks. Among these are the break location. In the Millstone 1 FSAR, the most severe mass and energy release rate calculated for containment design was done assuming a double-ended recirculation line break. The input and boundary conditions used in this analysis are in accordance with current criteria except for initial reactor power. The FSAR DBA LOCA mass and energy release calculation assumes the initial reactor power at 100% full power, not 102% as required by current criteria. However, later docket material documents an analysis of 102% of full power initial conditions, which is acceptable.

The FSAR calculated peak post-accident containment pressure resulting from a double-ended recirculation line break is 43 psig. The peak wetwell pressure was 25 psig. The peak drywell temperature was 291°F.

In addition to recirculation line breaks, the current criteria state that steam line breaks above the core must be considered. The licensee has not performed a steam line break analysis specifically for containment design evaluation.

3.0 ANALYSIS OF MILLSTONE 1 CONTAINMENT DESIGN

The recirculation line break results in the limiting condition for calculating the peak pressure inside the containment. The steam line pipe break analysis is the most limiting case for temperature conditions inside the containment. Both of these analyses were performed.

3.1 Recirculation Line Break

There are no reactor coolant system (RCS) blowdown decks available for the Millstone 1 plant. The specific input decks for reactor blowdown analysis are proprietary to the General Electric Company and are not available in the open literature. A search of the Millstone 1 docket material did not provide any documented LOCA mass and energy release data. In this situation, it was decided to apply the LOCA blowdown data from the most similar BWR plant as the best engineering estimate. A literature search concluded that the best available LOCA blowdown data would be from the Dresden 2 plant, which was also under review for SEP. Refer to Appendix B for more details pertaining to the selection of the Dresden 2 LOCA blowdown.

The Dresden 2 RCS design is nearly identical to that of Millstone 1. The full DEG recirculation line break area is 5.62 ft^2 for Dresden 2 versus 5.82 ft^2 for Millstone 1, assuming the equalizer line valve is open for both plants. The initial reactor pressure is 1000 psig at Dresden 2 versus 1035 psig at Millstone 1. Hence, the saturated liquid enthalpy during the initial recirculation line blowdown will differ by 5 Btu/lbm, which is less than a 1% difference. Due to its larger physical size and higher power rating, the Dresden 2 reactor coolant system mass and energy inventory at full power is approximately 25% larger than the Millstone 1 RCS mass and energy inventory. Therefore, the application of the Dresden 2 LOCA blowdown to the Millstone 1 containment model will provide a conservatively high calculated containment response. Table 1 lists the recirculation line break assumptions that were used in the Dresden 2 SEP analysis (Re: Memorandum, T. Speis to G. Lainas, December 2, 1981, Dresden 2 TER, SEP Topics VI.2-D and VI-3). These assumptions remain valid for Millstone 1.

The resulting recirculation line break mass and energy release rates are shown in Table 2.

3.1.1 Containment Response Calculation to a Recirculation Line Break

The input data for the containment response calculation consists of the mass and energy release to the containment, a description of the containment geometry, heat removal systems, and containment heat sink data. The mass and energy release rate data used were taken from the blowdown of the recirculation line described in the previous section.

The containment heat removal system consists of a pressure suppression pool, LPCI containment cooling subsystem, containment sprays, and containment fan coolers. For this analysis, the containment fan coolers will not operate due to the assumed loss of auxiliary ac power. The containment sprays must be manually activated by the operator and therefore, conservatively will not be accounted for in this analysis. The LPCI containment cooling subsystem will be assumed to be switched from core flooding mode to wetwell cooling mode at 600 seconds. This switchover requires operator action. The pressure suppression pool and vent and downcomer model information was taken from the Millstone 1 FSAR and subsequent docket material.

No Millstone 1 containment heat sink data were available in the open literature or in docket material. Therefore, Oyster Creek containment heat sink data were used as the best available substitutes.

The containment response calculation was performed with the CONTEMPT-LT/028 computer code. The node model is composed of three regions: reactor vessel, drywell, and wetwell. The geometric descriptions and initial boundary conditions were obtained from the Millstone 1 FSAR. A summary of the contaiment input model characteristics are given in Table 3.

3.1.2 Containment Response Results

The containment pressure and temperature responses to a recirculation line break are shown in Figures 1 and 2. The calculated transient reflects a peak post-accident containment drywell pressure of 44 psig and a temperature of 291^{o}F. The peak containment wetwell pressure and temperature are 14 psig and 144^{o}F. The containment design pressure for the drywell and the wetwell is 62 psig. There is, therefore, a substantial margin between the peak calculated pressure and the containment design pressure.

3.2 Main Steam Line Pipe Breaks

Analyses of the containment response to various steam line breaks were performed to reveal the most severe temperature condition. Break sizes of 0.01 ft^2, 0.10 ft^2 and 0.75 ft^2 were examined to identify the most limiting steam line break. Blowdown data consisting of mass and energy release rates were provided by Northeast Utilities for these breaks. These data were developed by General Electric based on their licensing code as part of the equipment qualification effort. (Ref.: Letter, W. B. Counsil, Northeast Utilities to H. R. Denton, NRR, June 9, 1981). The blowdown calculation was performed using the assumptions given in Table 4. The staff has reviewed the assumptions listed in Table 4 and agrees with them with the exception of the manual actuation of the ADS valves at 600 seconds. Also, the utility has indicated to the NRC that the Low Pressure Core Spray (LPCS) injection was assumed to be constant at full flow once it was initiated. The result of these two actions was that the resulting blowdown exhibited a switchover from the steam to the liquid phase. The time that this switchover occurred ranged from about 2007 seconds for the 0.01 ft^2 break to 900 seconds for the 0.75 ft^2 break. The switchover is accompanied by a corresponding drop in enthalpy of water as it goes from steam to liquid.

Although this switchover is possible in the accident sequence it appeared more plausible to the NRC that operator action would be taken beyond the 10 minute time limit into the transient to avoid the condition of having solid water above the level of the main steam line. It seemed appropriate to take into account the operator action which would likely be to cycle the LPCS pump to control the water level in the vessel. The resulting blowdown, then, would consist of pure steam or a two phase mixture. Rather than try to determine the relative percentages of steam and liquid in the blowdown it was decided to conservatively specify all the blowdown to be steam with its accompanying enthalpy in the vicinity of 1200 Btu/lbm.

Based on the considerations mentioned and after examining the utility transmittals it was felt that enough information had been submitted to perform the containment response calculations without deriving new blowdown output from the RELAP code. The blowdown data was modified to eliminate the crossover from steam to liquid. The initial values of flow rate and enthalpy were taken, as is, up until ADS actuation. For the 0.75 ft^2 and 0.01 ft^2 break the blowdown values were held constant from this point on through the remainder of the transient. For the 0.10 ft^2 break, the blowdown was correlated to the 0.75 ft^2 break after that break flow was adjusted.

0.75 ft^2 Main Steam Line Break

The blowdown data submitted by Northeast Utilities was revised to eliminate the switchover from pure steam to liquid at 900 seconds. The revision consisted of simply maintaining the blowdown values of vapor mass flow rate, 60.19 lb/sec, and enthalpy, 1169 Btu/lb constant from 816.3 seconds to 3000 seconds. Table 5 contains the revised data. The effect of this modification compared with the original utility data is marginal since the peak temperature is the same for both sets of data.

0.10 ft^2 Main Steam Line Break

The blowdown curve for the 0.10 ft^2 break exhibits a sharp change in slope in the time frame immediately following 600 seconds, the point at which the automatic depressurization system valves actuate. The ADS actuation causes an increase in the rate of reactor vessel depressurization which manifests itself by a decrease in vapor flow rates.

To arrive at a modified version of this blowdown it was decided to correlate with the 0.75 ft^2 break the relationship between the integral of the product of steam flow rate and enthalpy over time versus the mass flux, i.e.,

$$\int \dot{m}\, H\, dt \text{ vs. } \dot{m}/A_{break}$$

i.e.,

TOTAL DEPLETED ENERGY vs. MASS FLUX

where $\dot{m}$ = steam flow rate, lbm/second

H = enthalpy, Btu/lbm

$\frac{\dot{m}}{A_{break}}$ = mass flux, lbm/second - ft^2

(A_{break} = cross sectional area of break, ft^2)

First the curve was plotted, in Figure 3, using the revised data for the 0.75 ft^2 MSLB given in Table 5. Then the data was plotted for the 0.10 ft^2 MSLB through 611.4 seconds and extended graphically so that it asymptotically approached the curve generated from the 0.75 ft^2 break. This was done as an assumption that both breaks would converge. From this curve, representative points were selected and from them the values of mass flow rate and time were determined. The revised blowdown is shown in Figure 4 and compared with the original. For example, point 1 on Figure 3 indicates that the mass flux equals 1031 lb/ft^2 seconds. Thus $\dot{m}$ = 103 lb/sec.

ΔT = (change in total energy depleted)/($\dot{m}$)(h) = 185.1 sec.

where H = enthalpy, 1205 Btu/lbm

T total = 611.4 + 185.1 = 796.5 sec.

Table 6 gives the revised blowdown data and Figure 4 shows the revised blowdown curve.

As with the 0.75 ft^2 break, the effects on containment temperature of this modifications compared to the utility deck is marginal. The peak temperature is the same for both sets of data.

0.01 ft^2 Main Steam Line Break

The data submitted by Northeast Utilities at time t = 0 consists of pure steam with a flow rate of 21.16 lb/sec and an energy content (enthalpy) of 1191 Btu/lbm. The revised data contained in Table 7 simply consists of maintaining the blowdown at these levels for the duration of the transient. The effect of this change was to raise the peak temperature up to about 300^{0}F and shift the peak to 3000 seconds. The utility data yielded a peak temperature of 261 F at 2000 seconds. By comparison with the Oyster Creek nuclear power plant, a 0.01 ft^2 MSLB break was examined by LLNL and the peak temperature was determined to be 310^{0}F. This break was also examined by LLNL for Dresden 2 and the peak temperature was determined to be 328^{0}F.

3.2.1 Containment Response to a Main Steam Line Break

The containment response to a Main Steam Line Break was calculated using the CONTEMPT-LT/028 computer code. The input data comprised the same geometric containment model as that used to analyze the previous LOCA break described in Section 3.1.1. The three MSLB cases of 0.75, 0.1, 0.01 ft^2 were analyzed using the blowdown data shown in Tables 5, 6, and 7. In each analysis, when blowdown input ends, the mass and energy release rate to the drywell is calculated by CONTEMPT based on ECCS injection and 120 percent of ANS standard for core decay heat. The containment response analysis assumes a loss of auxiliary ac power and both FWCI and LPCI failure. One core spray is assumed to operate. No credit is taken for the manually initiated containment sprays.

3.2.2 Containment Response to Main Steam Line Break Results

The calculated pressure and temperature responses to the postulated main steam line breaks are shown in Figures 5 through 19. The calculated containment response to the .75 ft^2 MSLB shows a maximum drywell pressure of 19.2 psig and a maximum temperature of 325°F. The wetwell maximum pressure and temperature are 17.1 psig and 157°F.

The calculated containment response to the 0.1 ft^2 MSLB shows a maximum drywell pressure of 17.6 psig and maximum temperature of 328°F. The wetwell maximum pressure and temperature are 15.5 psig and 125°F.

The calculated containment response to the 0.01 ft^2 MSLB shows a maximum drywell pressure of 16.7 psig and maximum temperature of 300°F. The wetwell maximum pressure and temperature are 14.7 psig and 111°F.

As the drywell and wetwell design pressures for Millstone 1 are 62 psig, the results above confirm that the containment pressures due to steam line breaks are substantially below the design pressures. The maximum post-accident drywell temperature was approximately the same for the 0.1 ft^2 and the 0.75 ft^2 MSLBs. The design temperature of the Millstone 1 containment is 281°F. Both the 0.1 ft^2 and 0.75 ft^2 MSLB cases exceed the design temperature.

4.0 CONCLUSIONS

Based on the review of the Millstone 1 docket material and the above discussed reanalysis, it is concluded that the Millstone 1 containment design pressure meets all current NRC criteria. The containment atmosphere temperature profile as the result of a 0.1-ft^2 MSLB exceeds the containment design temperature. The drywell atmosphere reaches a maximum temperature of 328°F which exceeds the containment structure design temperature of 281°F.

TABLE 1

RECIRCULATION LINE BREAK ASSUMPTIONS

BASED ON DRESDEN 2 DEG BREAK

---

1. Reactor initial condition is at 102% of full power.

2. Recirculation pump suction line instantly separates to a full double-ended guillotine break. Break area = 5.62 ft^2.

3. Equalizer line is open.

4. Loss of offsite ac power and diesel generator.

5. Main steam isolation valves start closing at 0.5 second and are fully closed within 3 seconds.

6. Feedwater flow stopped at time of accident.

7. Mass discharge through the broken pipe calculated using Moody critical flow with a multiplier of 1.0.

8. All reactor vessel mass is discharged through the break.

9. Reactor scrams at time zero.

TABLE 2

DOUBLE-ENDED-GUILLOTINE RECIRCULATION LINE BREAK

RELEASE DATA (5.62 ft^2 BREAK)

Time (seconds)	Flow (lbm/sec)	Energy (Btu/lbm)
0.0	27211.	552.
1.0	27211.	552.
2.0	27211.	553.
3.0	27211.	553.
4.0	27211.	554.
5.0	27211.	556.
10.0	27211.	562.
15.0	27218.	572.
20.0	19412.	591.
25.0	3012.	570.
30.0	1659.	711.
35.0	1037.	1198.
45.0	516.	1189.
55.0	291.	1180.

DECAY HEAT at 55 seconds (1.2 ANS)

TABLE 3

CONTAINMENT MODEL INPUT DATA

(TAKEN FROM MILLSTONE 1 FSAR)

Drywell/Wetwell Data

	Drywell	Wetwell
Free Air Volume (ft^3)	146,900.0	114,600.0
Initial Pool Water Volume (ft^3)	0.C	94,000.0
Initial Temperature of Atmosphere (^{o}F)	150.0	95.0
Initial Temperature of Pool (^{o}F)	150.0	95.0
Initial Pressure (psia)	15.7	14.7
Relative Humidity	1.0	1.0
Pool Surface Area (ft^2)	1271.8	9151.4

Vent System

Vent Pipes	
Number	8
Internal Diameter	6 ft. 9 in.
Vent Tubes Flow Area, Total	286.3 ft^2
Downcomer Pipes	
Number	96
Internal Diameter	2 ft.
Submergence Below Absorption Pool Water Level	4 ft. 9 in.

TABLE 4

MAIN STEAM LINE BREAK ASSUMPTIONS

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1. Scram at time zero.
2. FWCI failure.
3. No credit for isolation condenser.
4. No LPCI operation.
5. One of two core sprays operate.
6. Feedwater controller maintains normal reactor vessel water level until manually shut off at 600 seconds.
7. Reactor initially at 102% of full power.
8. Steam line isolation valves close within 3.5 seconds of break.

TABLE 5

0.75 ft^2 Main Steam Line Break Blowdown

TIME (s)	MASS FLOW RATE (lbm/s)	ENTHALPY (BTU/lbm)
0.0	1587.0	1191.0
32.7	1247.0	1199.0
83.3	662.4	1205.0
163.1	385.3	1201.0
248.4	238.3	1195.0
328.0	172.3	1189.0
405.1	135.1	1185.0
488.4	109.6	1181.0
576.7	91.9	1177.0
660.6	77.67	1174.0
738.5	67.53	1171.0
816.3	60.19	1169.0
3000.0	60.19	1169.0

TABLE 6

0.10 ft^2 MAIN STEAM LINE BREAK BLOWDOWN

TIME (s)	MASS FLOW RATE (lbm/s)	ENTHALPY (BTU/lbm)
0.0	211.6	1191
73.6	200.5	1193
144.4	177.2	1197
216.2	169.0	1198
291.7	148.3	1201
376.2	141.5	1202
454.7	124.7	1203
535.7	120.3	1204
611.4	103.1	1205
796.5	90.0	1200
981.7	77.0	1200
1198.1	70.0	1200
1317.1	50.0	1200
1900.0	40.0	1200
2316.7	30.0	1200
2816.7	24.0	1200
3233.0	19.0	1200

TABLE 7

0.01 - ft^2 MAIN STEAM LINE BREAK BLOWDOWN

TIME (sec)	MASS FLOW RATE (lbm/s)	ENTHALPY (BTU/lbm)
0.0	21.16	1191
5000.0	21.16	1191

TABLE 8

MILLSTONE 1 CONTAINMENT DESIGN CONDITIONS

VERSUS CALCULATED ACCIDENT CONDITIONS

Event	Containment Design	Calculated
DBA LOCA	62 psig and 281°F	44 psig and 291°F
0.01 ft^2 MSLB	62 psig and 281°F	16.7 psig and 300°F
0.1 ft^2 MSLB	62 psig and 281°F	17.6 psig and 328°F
0.75 ft^2 MSLB	62 psig and 281°F	19.2 psig and 325°F

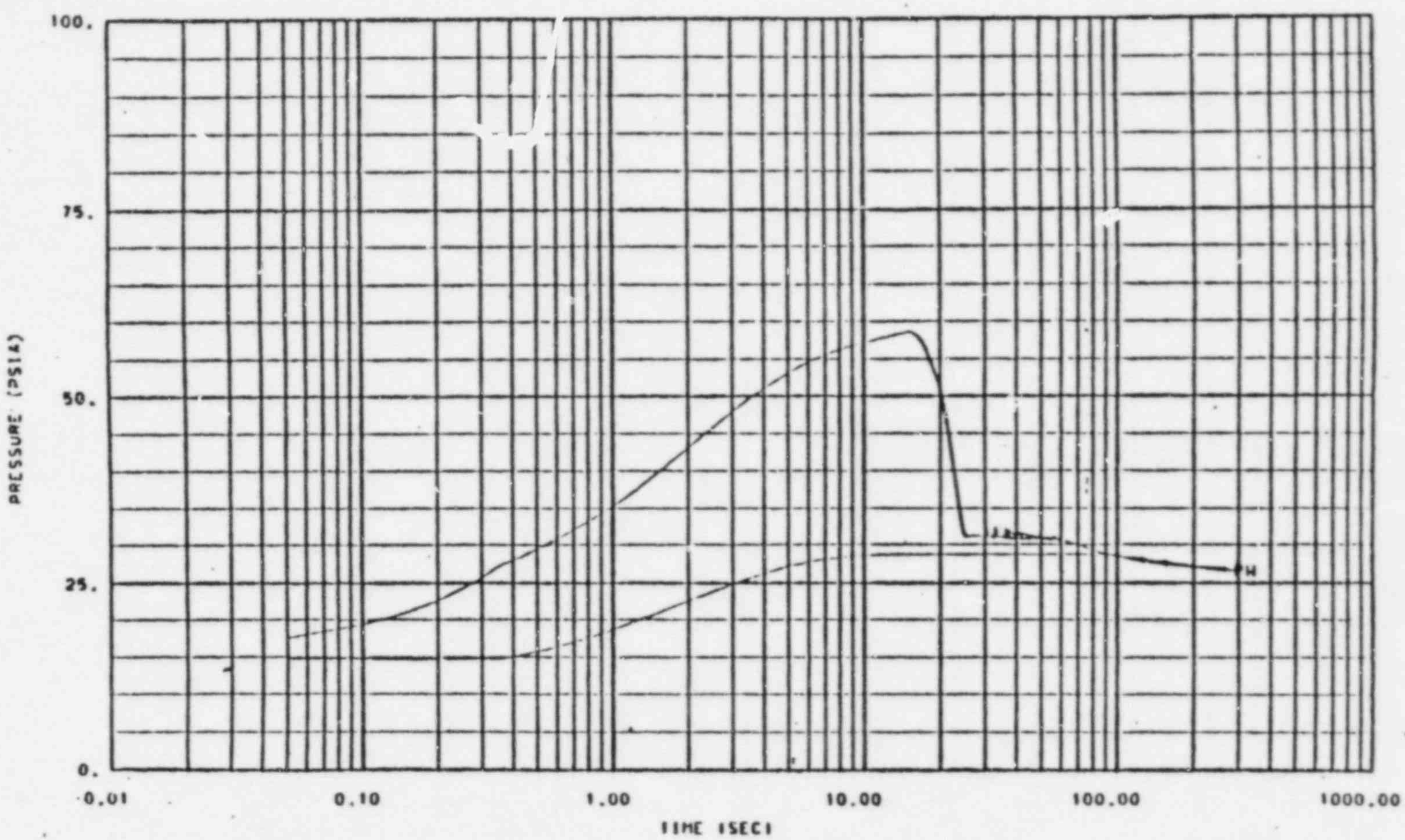

Figure 1. Millstone 1 Containment Pressure Response to a Double-Ended-Guillotine Recirculation Line Break

Top Curve = Drywell

Bottom Curve = Wetwell

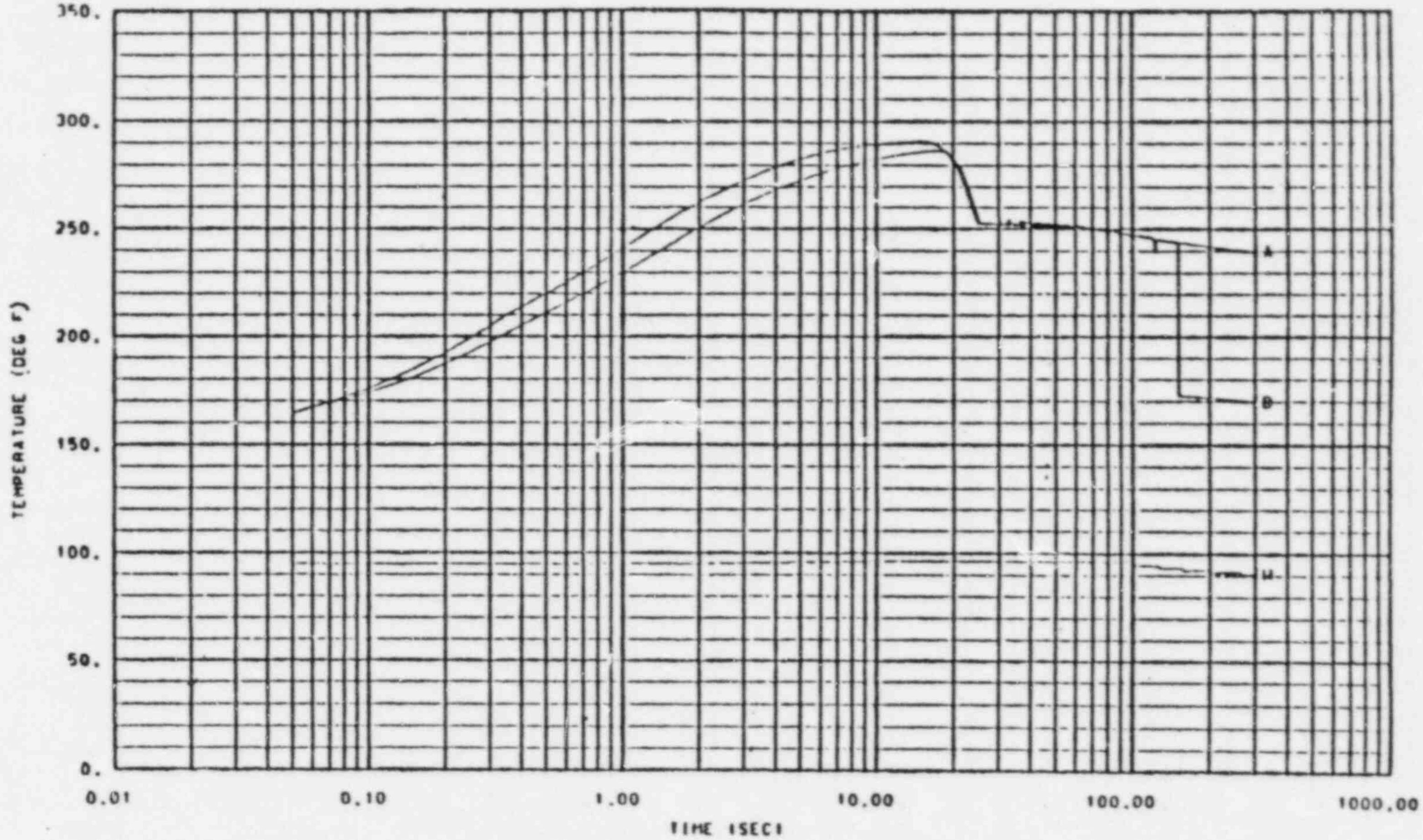

Figure 2. Millstone 1 Containment Temperature Response to a Double-Ended-Guillotine Recirculation Line Break

A = Drywell Atmosphere Temperature
B = Drywell Pool Temperature
W = Wetwell Atmosphere Temperature

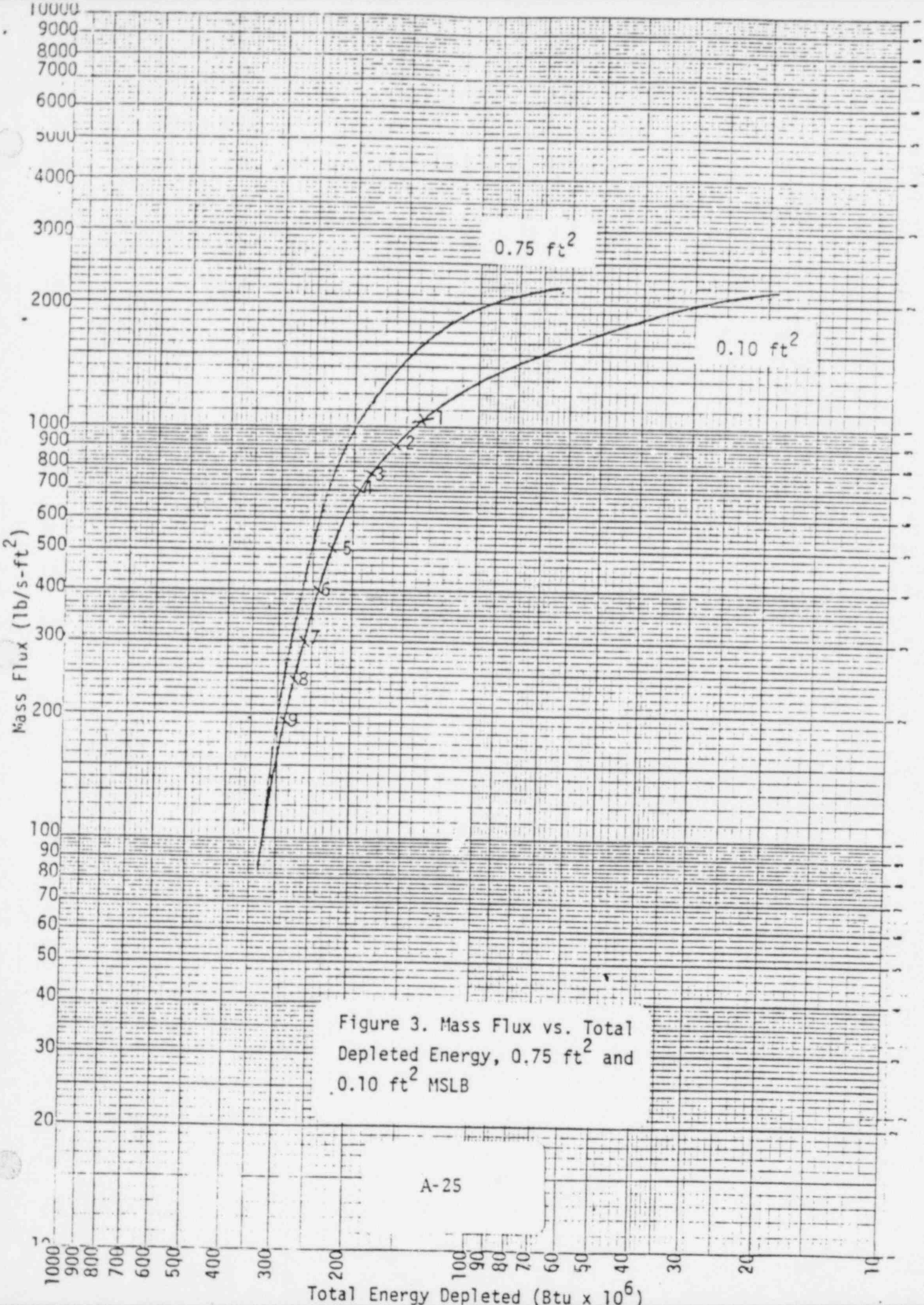

Figure 3. Mass Flux vs. Total Depleted Energy, 0.75 ft^2 and 0.10 ft^2 MSLB

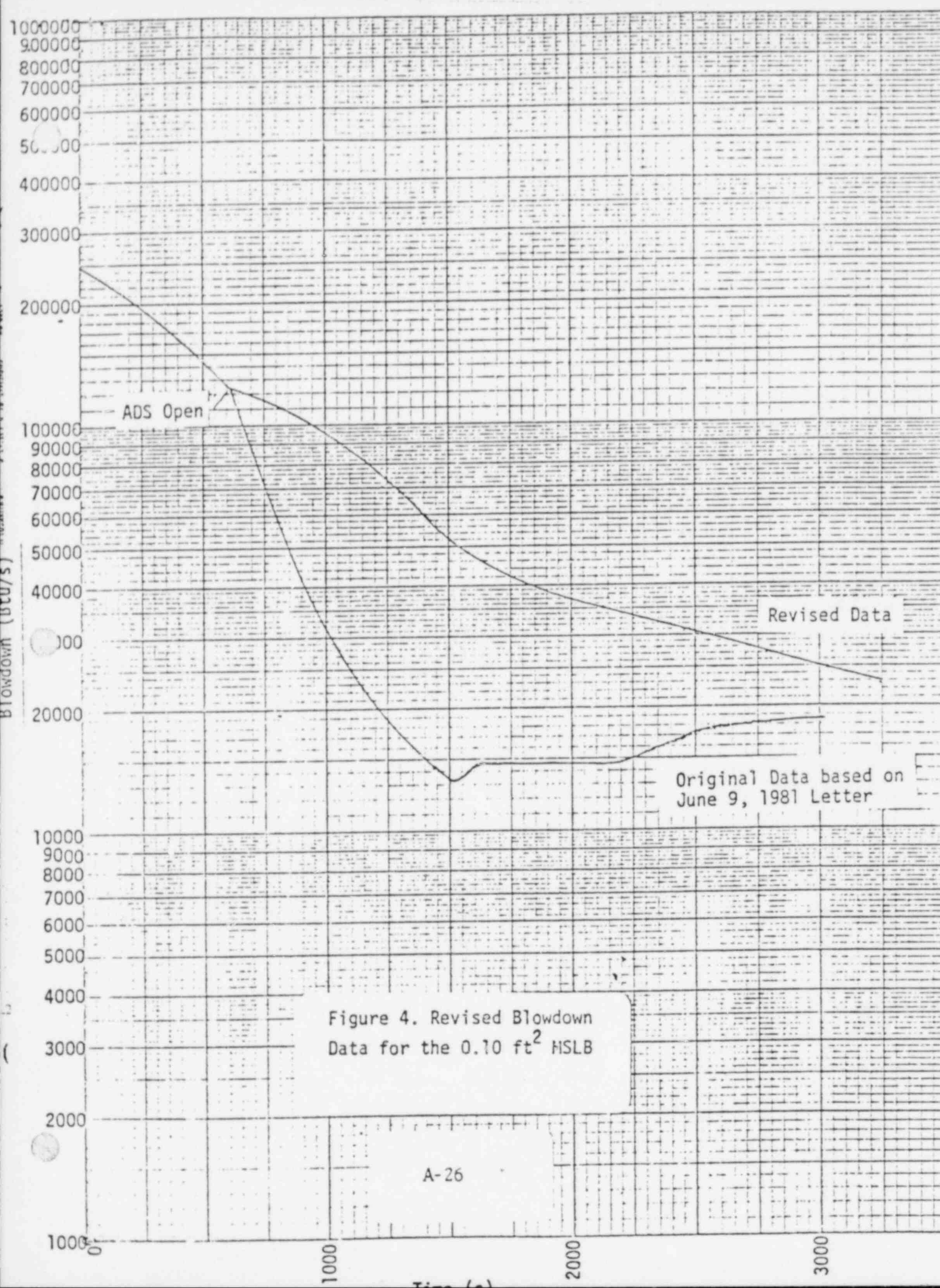

Figure 4. Revised Blowdown Data for the 0.10 ft^2 MSLB

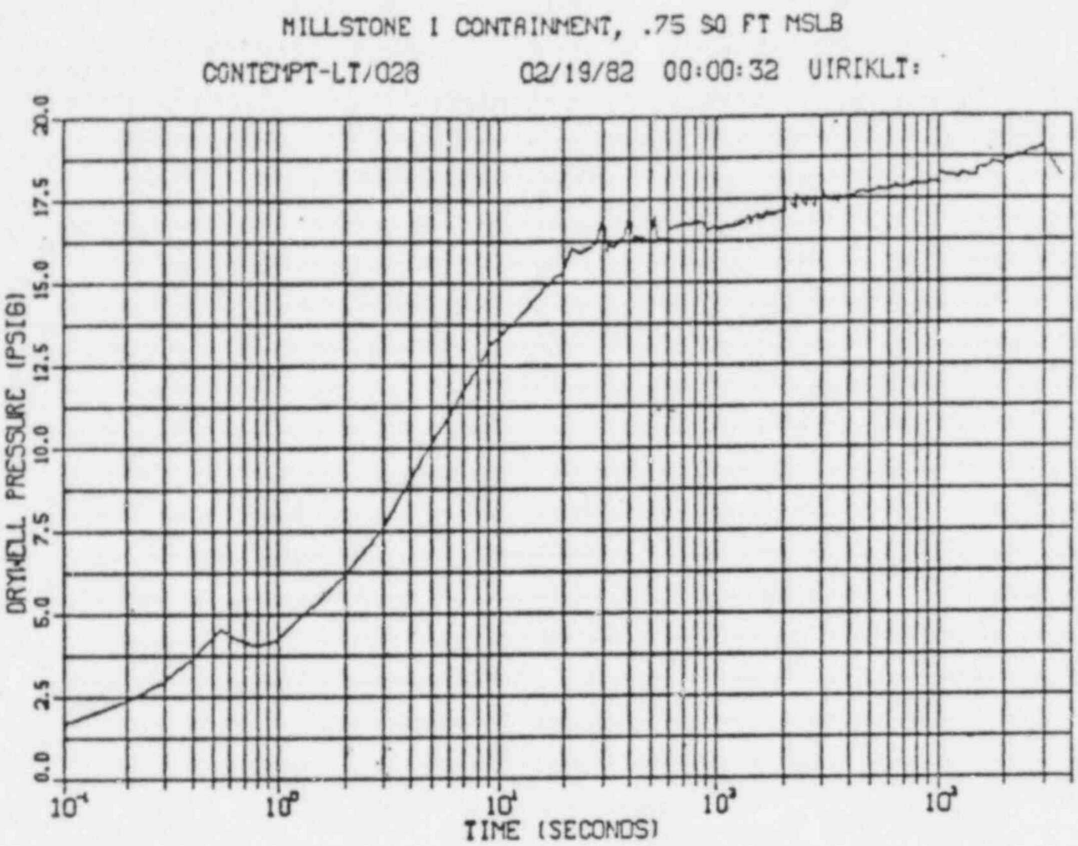

Figure 5. Drywell Pressure
Response to a 0.75 ft^2 MSLB

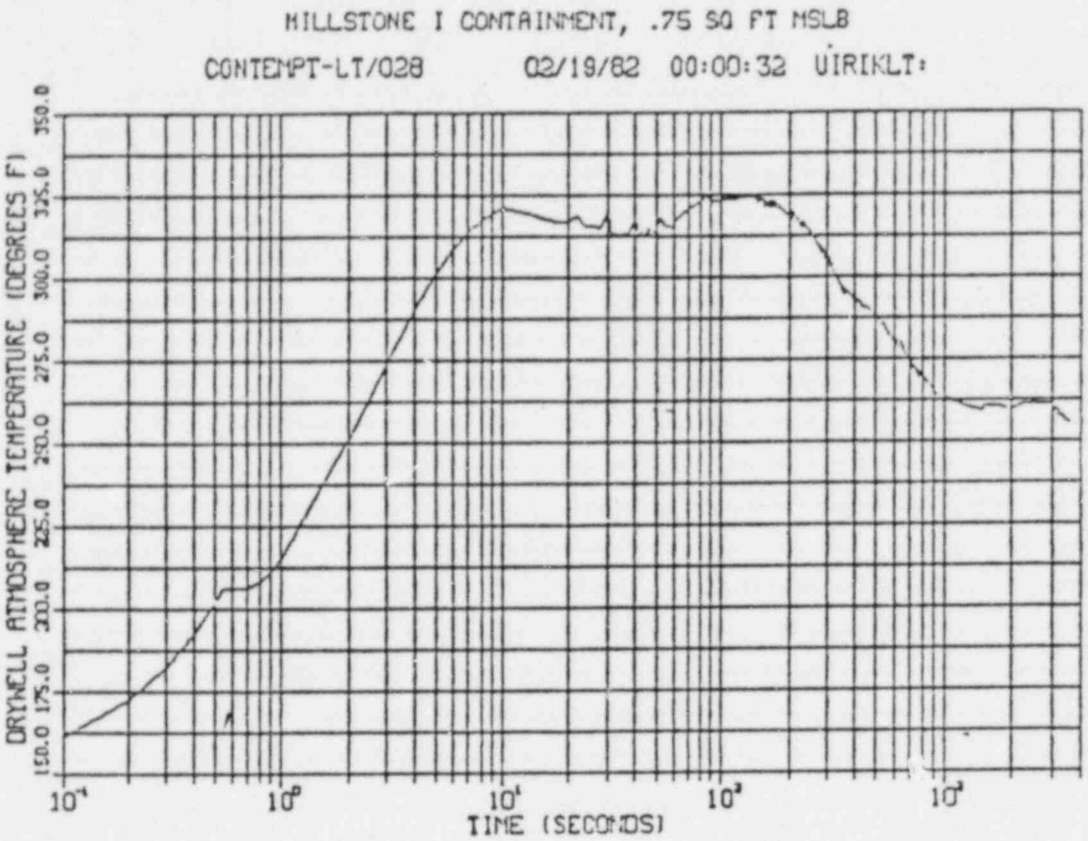

Figure 6. Drywell Atmosphere Temperature Response to a 0.75 ft^2 MSLB

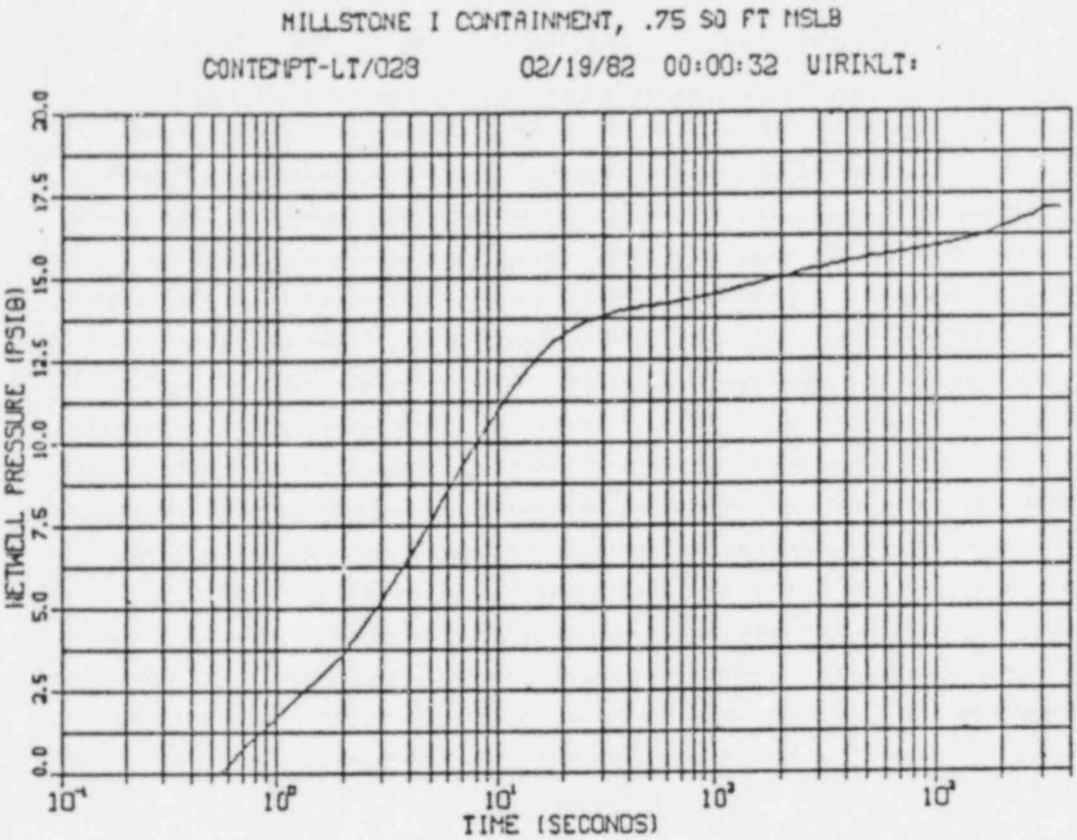

Figure 7. Wetwell Pressure
Response to a 0.75 ft^2 MSLB

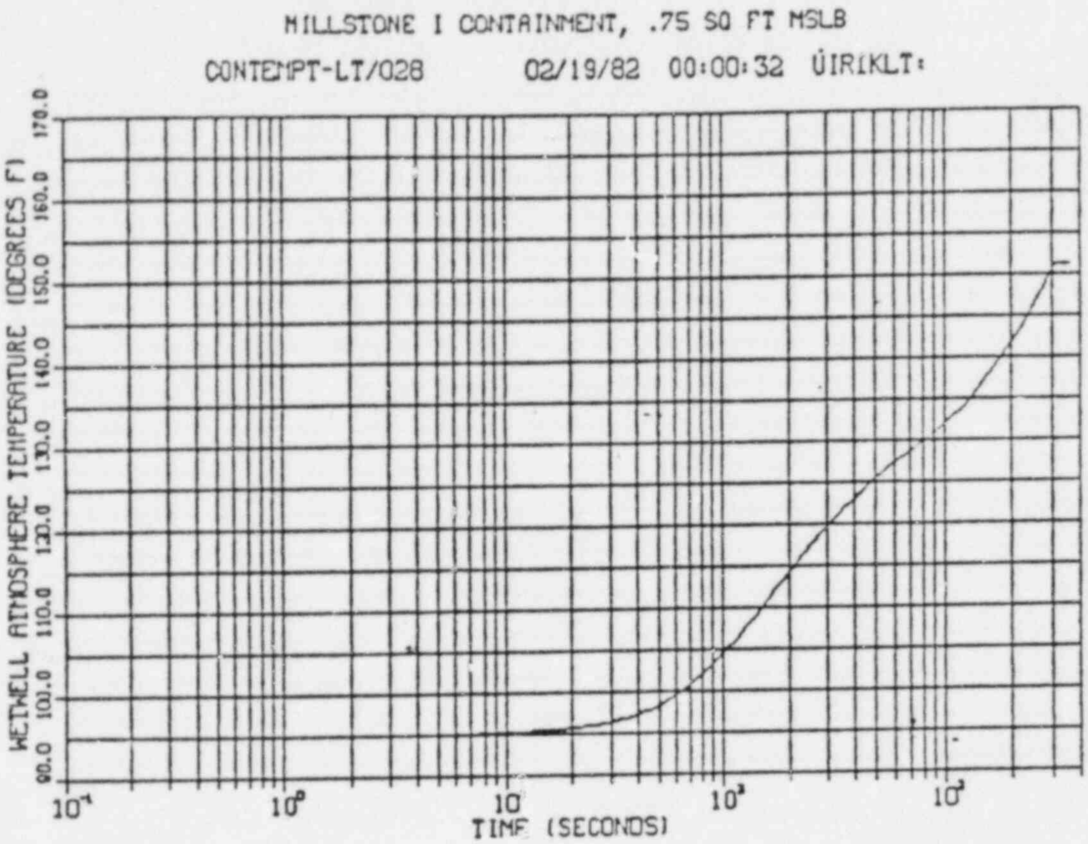

Figure 8. Wetwell Atmosphere Temperature Response to a 0.75 ft^2 MSLB

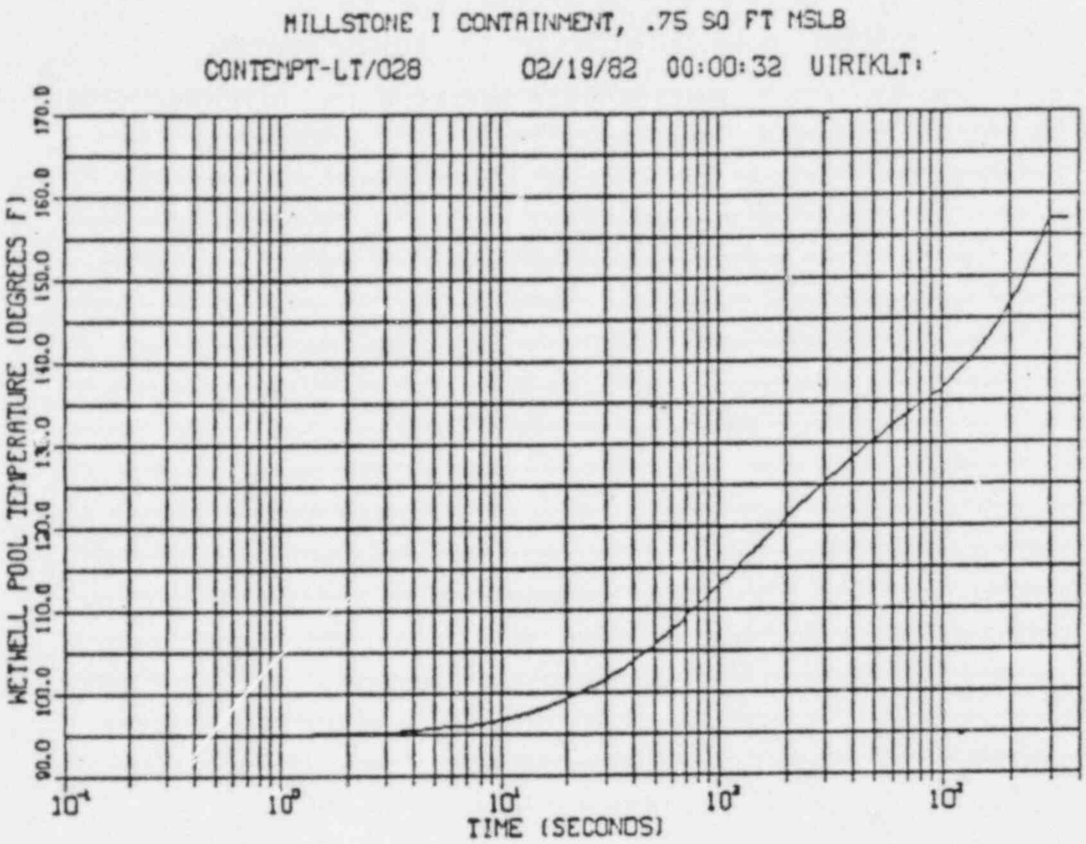

Figure 9. Wetwell Pool Temperature Response to a 0.75 ft^2 MSLB

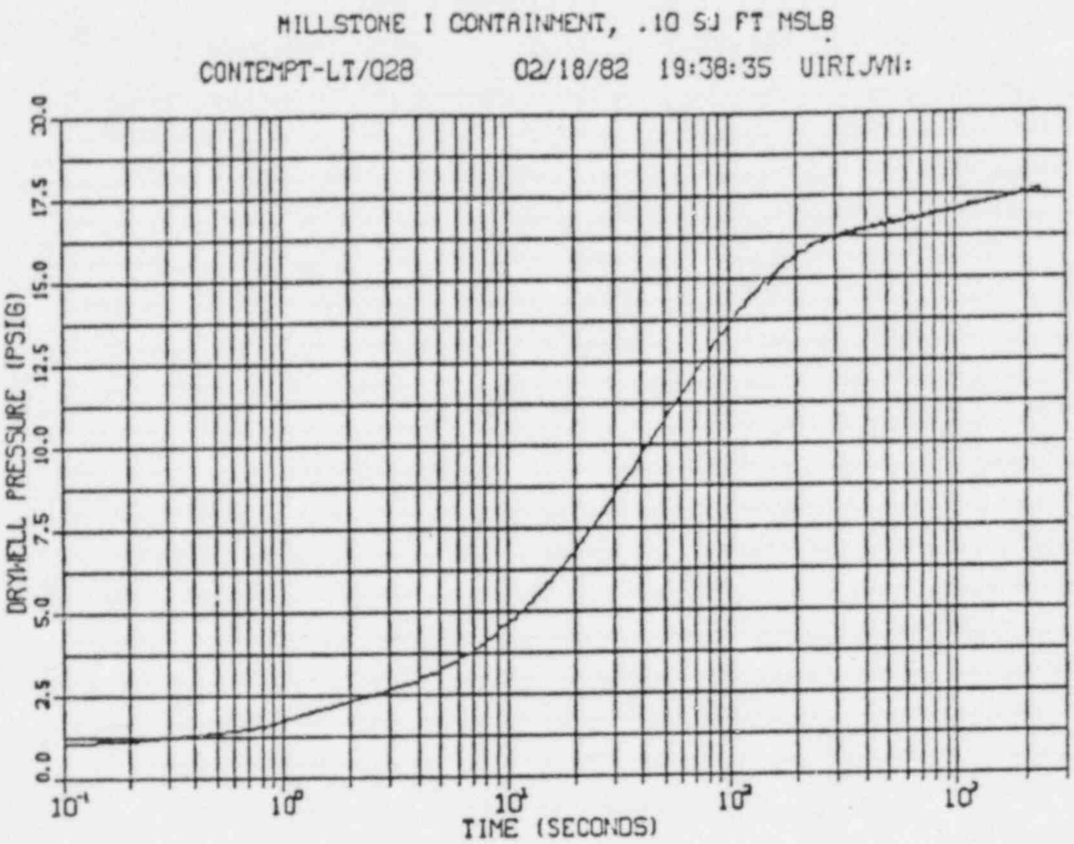

Figure 10. Drywell Pressure
Response to a 0.10 ft^2 MSLB

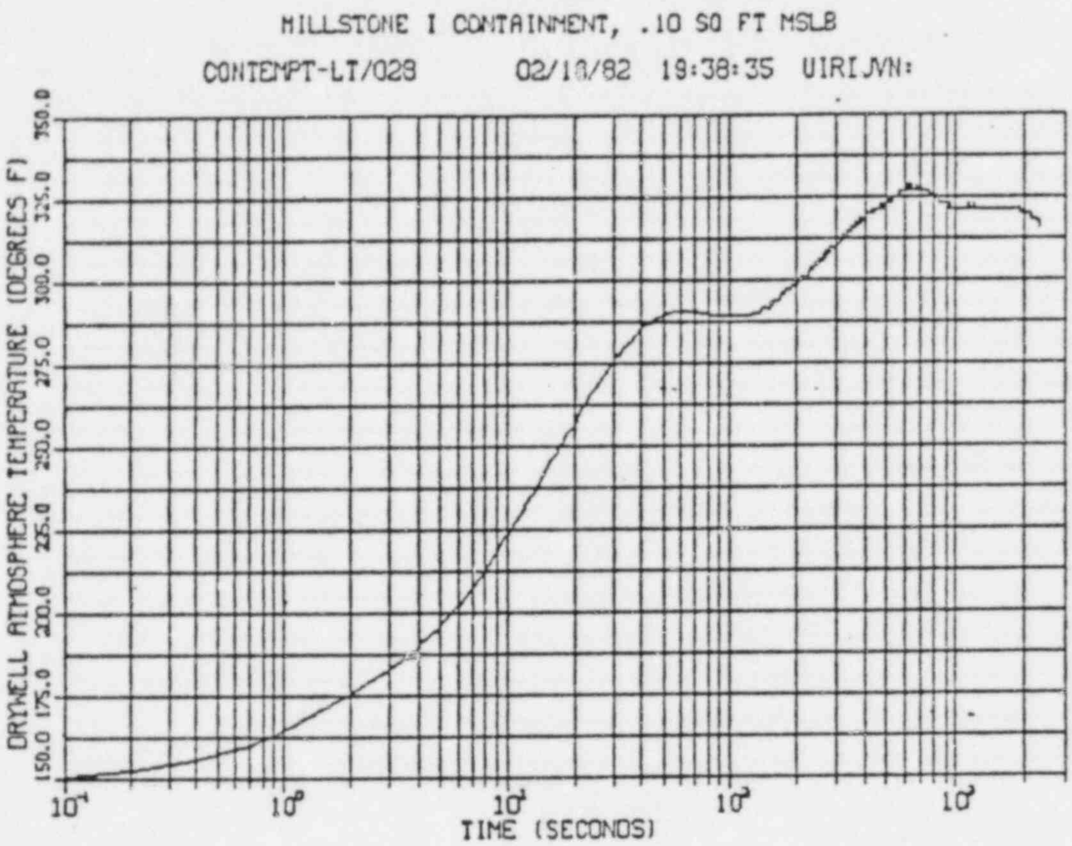

Figure 11. Drywell Atmosphere Temperature Response to a 0.10 ft^2 MSLB

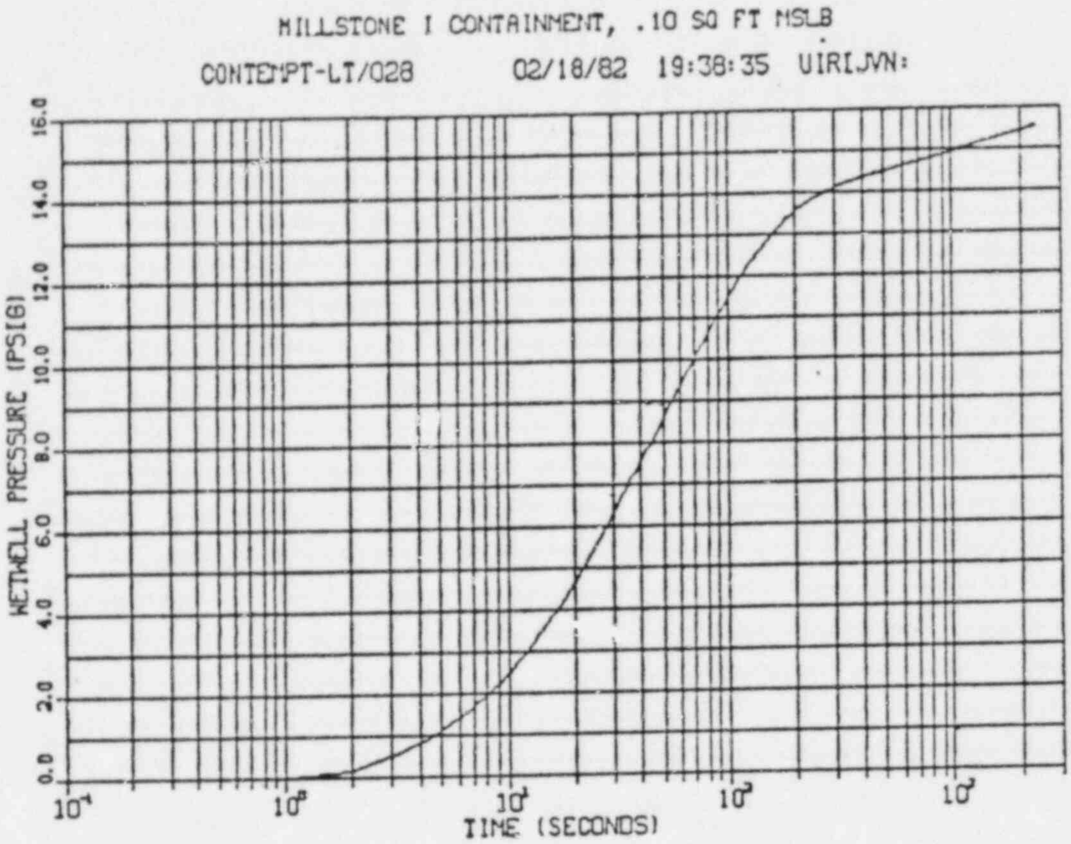

Figure 12. Wetwell Pressure Response to a 0.10 ft^2 MSLB

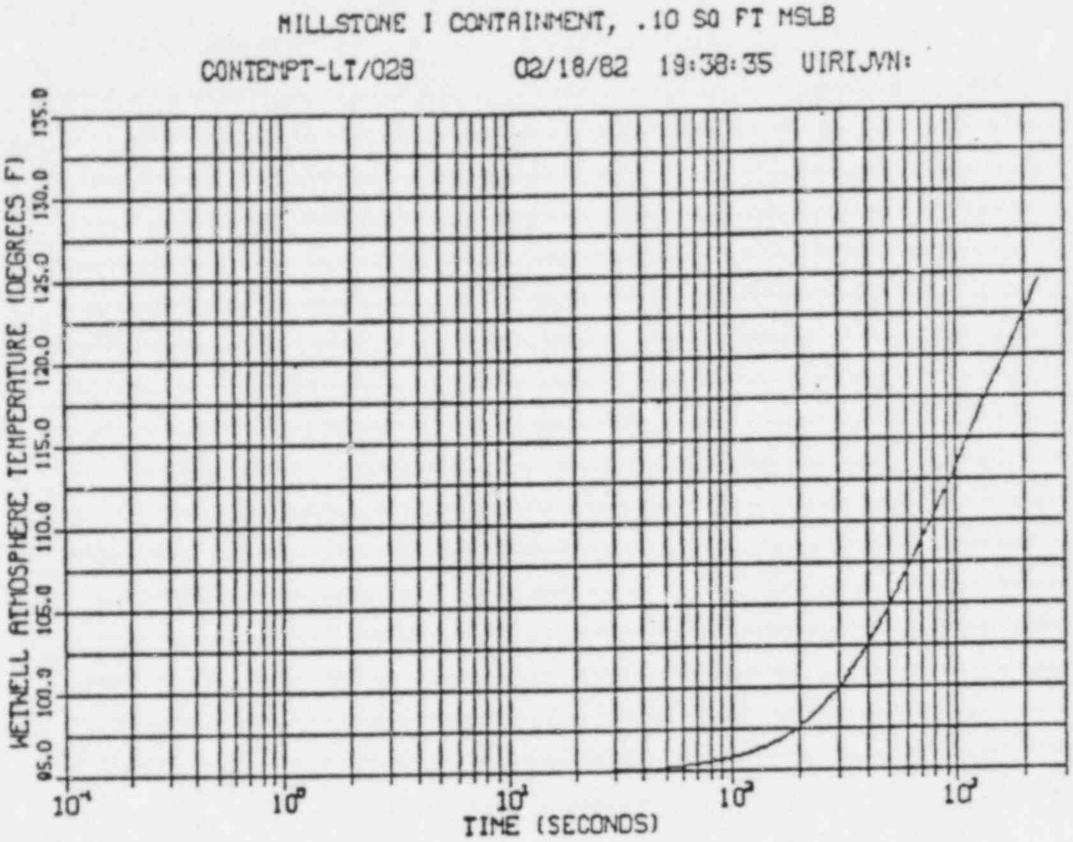

Figure 13. Wetwell Atmosphere Temperature Response to a 0.10 ft^2 MSLB

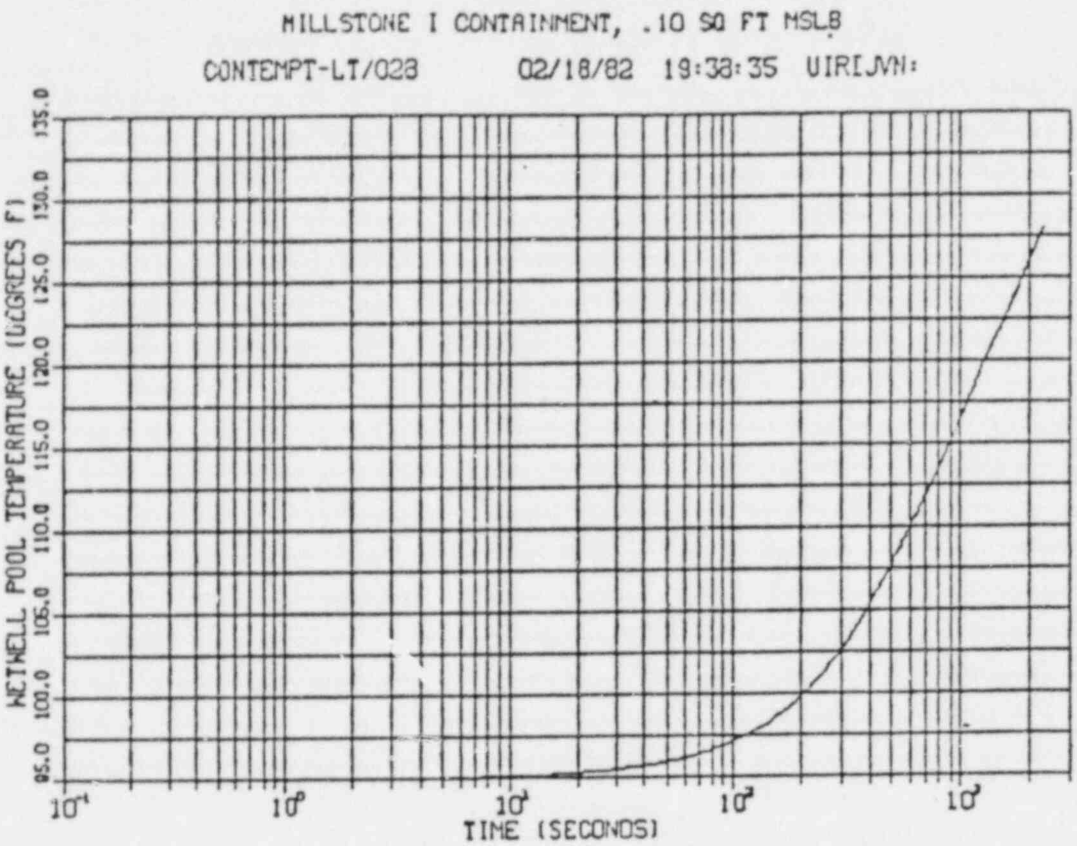

Figure 14. Wetwell Pool Temperature Response to a $0.10\ ft^2$ MSLB

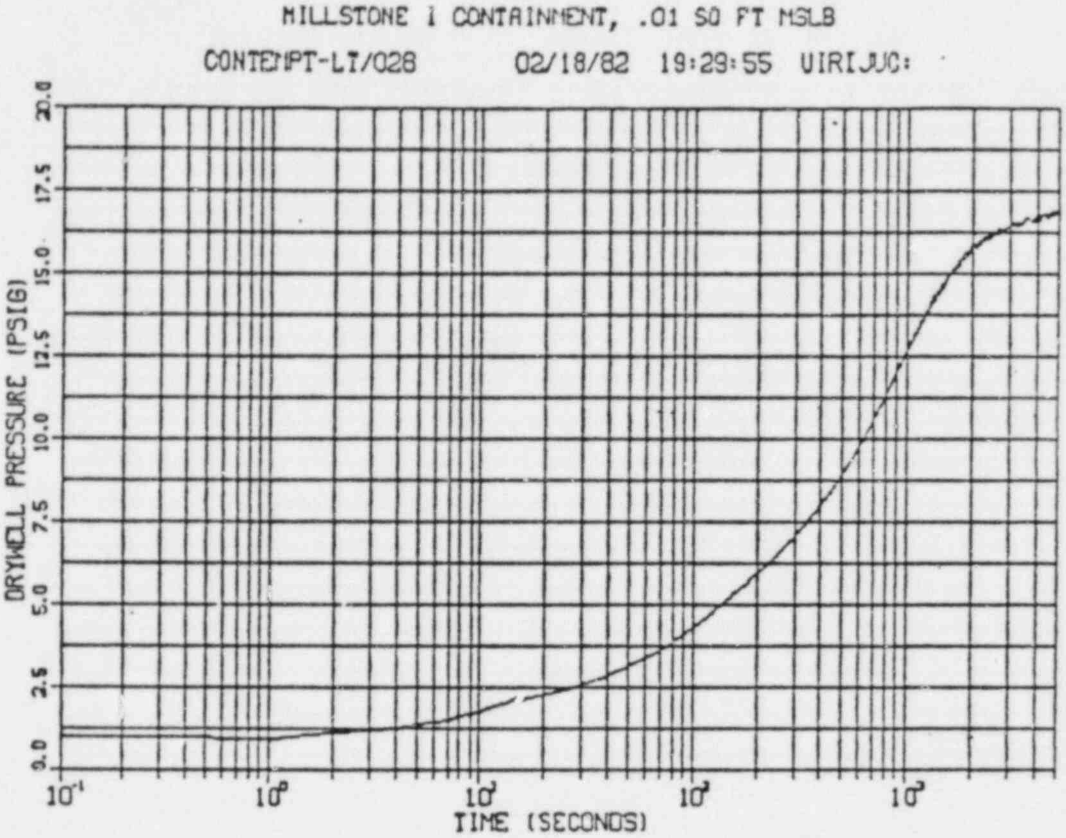

Figure 15. Drywell Pressure
Response to a 0.01 ft^2 MSLB

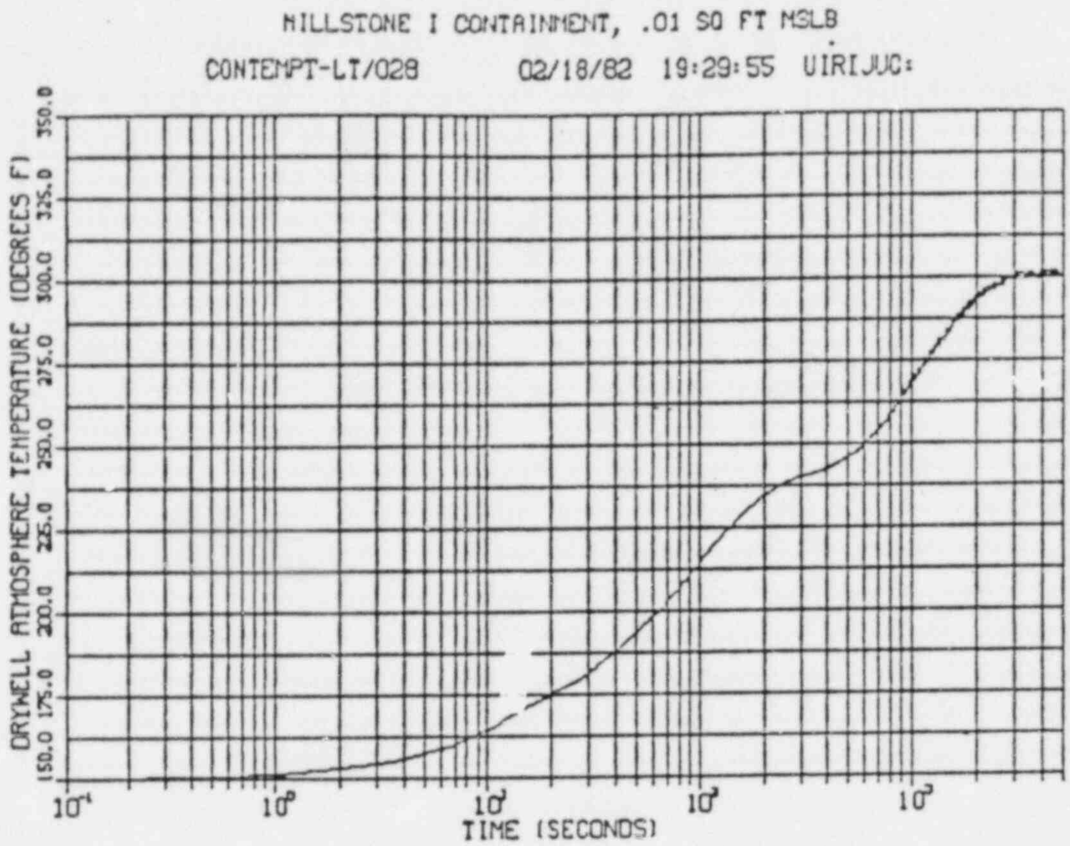

Figure 16. Drywell Atmosphere
Temperature Response to a
0.01 ft^2 MSLB

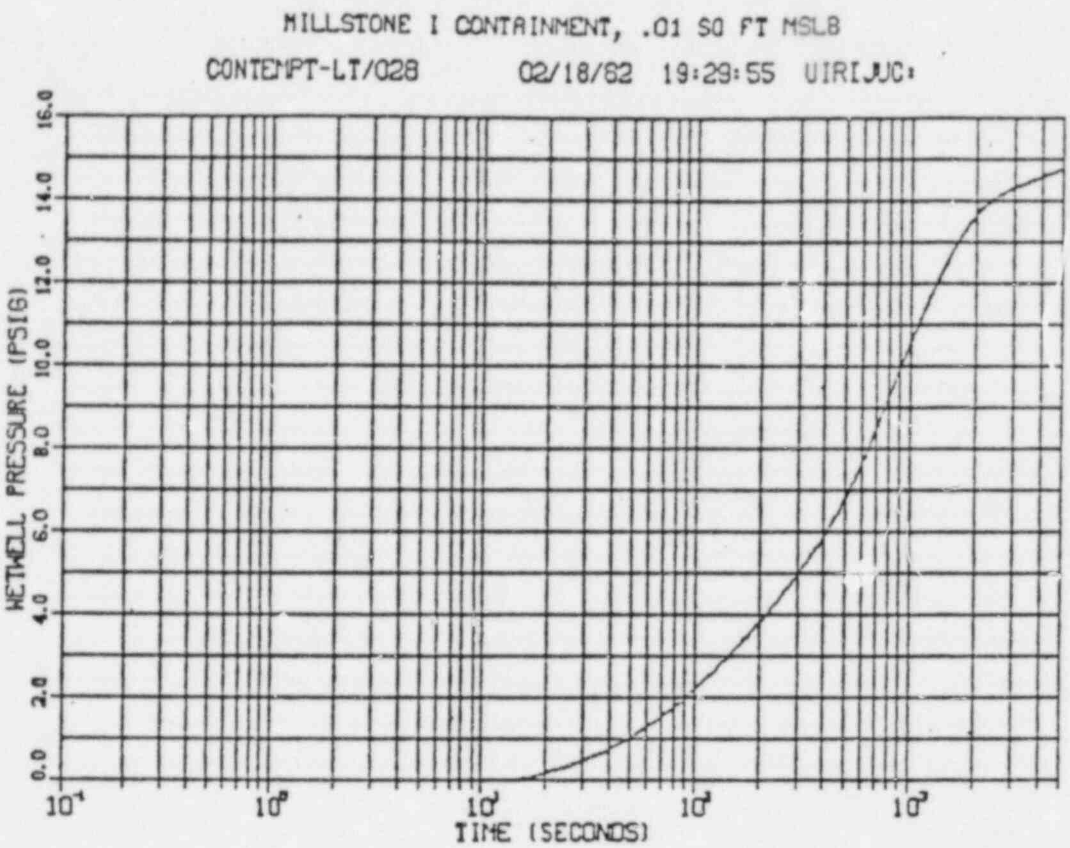

Figure 17. Wetwell Pressure
Response to a 0.01 ft^2 MSLB

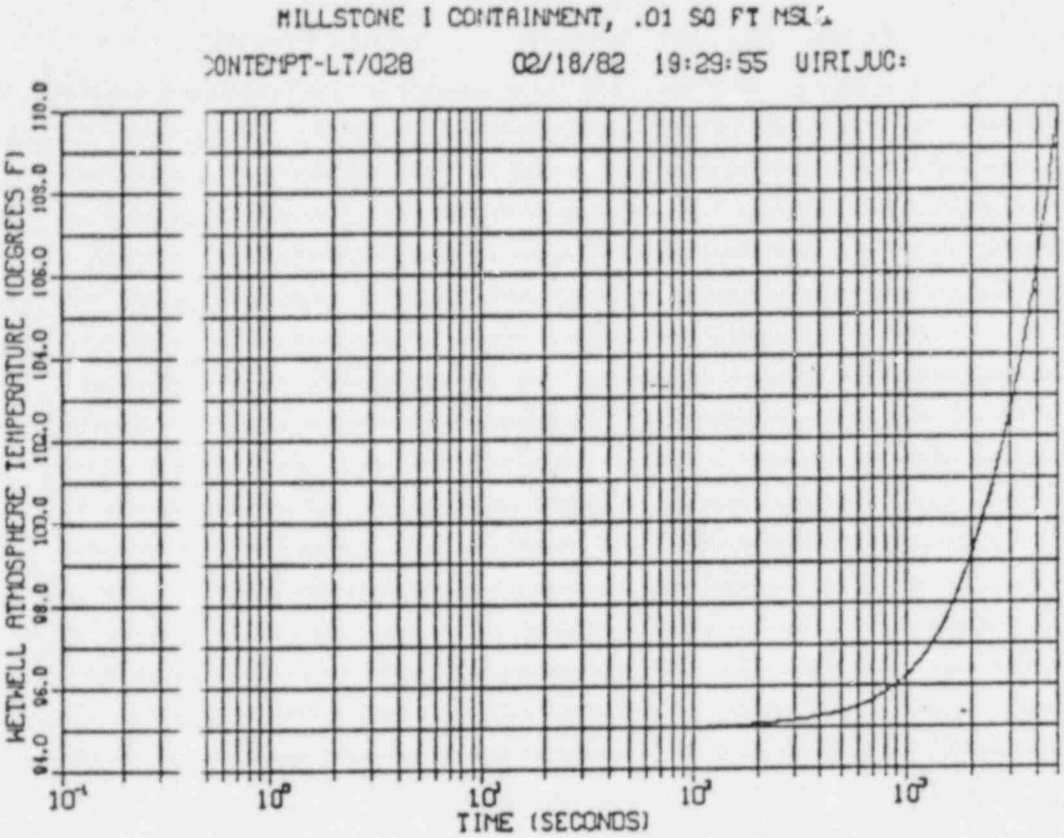

Figure 18. Wetwell Atmosphere Temperature Response to a $0.01\ ft^2$ MSLB

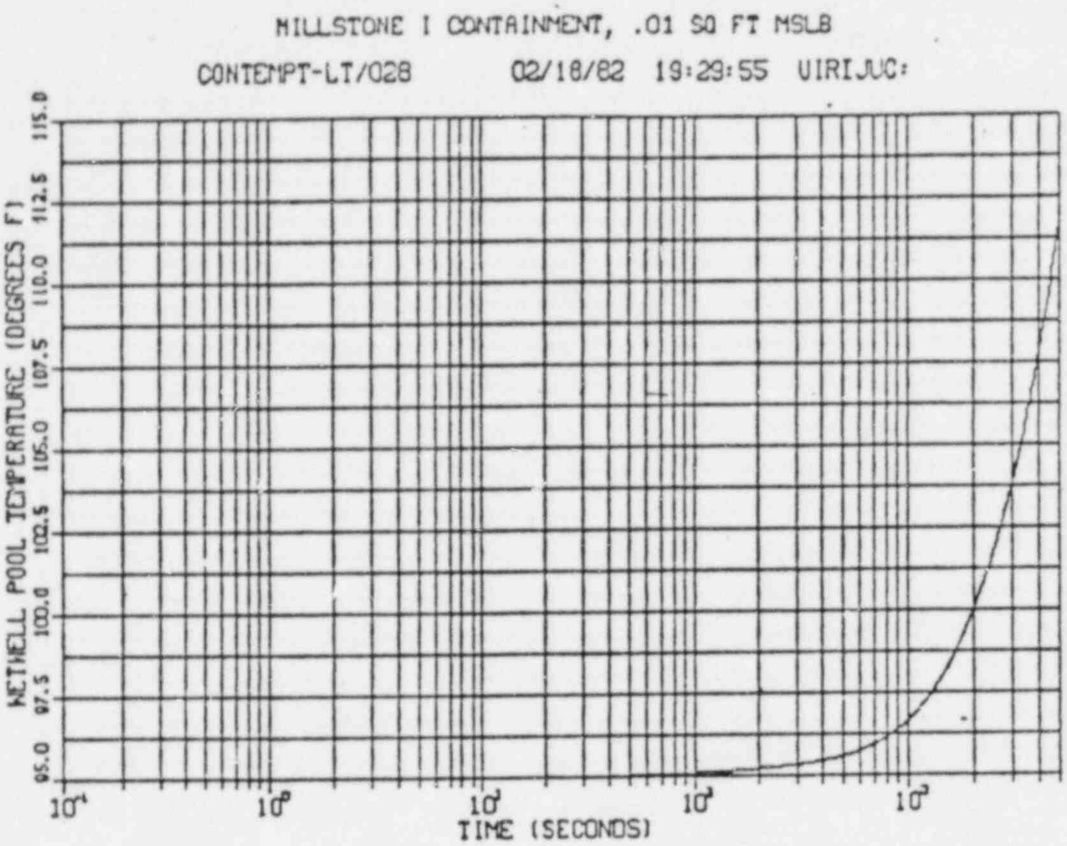

Figure 19. Wetwell Pool Temperature Response to a 0.01 ft^2 MSLB

APPENDIX B

CS&A-125-81, J. D. Atchison Letter to D. G. Vreeland, 2/17/81

ENERGY INCORPORATED

REF: CS&A-125-81

February 17, 1981

Mr. David Vreeland, L-90
University of California
Lawrence Livermore Laboratory
P.O. Box 808
Livermore, CA 94550

SUBJECT: SEP Containment Analysis, Recommended Action on Millstone I

Dear Dave:

The purpose of this letter is to discuss the possible methods of obtaining the necessary pipe rupture blowdown data for Millstone I to be used for containment analysis. The following recommendations are different than those proposed in the telephone discussion of 2/10/81 involving G. R. Sawtelle, J. D. Atchison, and D. G. Vreeland. Since that time, further reference research and discussion concerning the Millstone I plant have led to a new conclusion. The associated background and reasons for this decision are given below.

Of all the plants to be analyzed in the SEP Containment Analysis, only Millstone I does not have an available RELAP deck for blowdown calculations. The two alternative solutions to this problem are to either create a Millstone I RELAP deck from scratch or use a RELAP deck from a nearly identical BWR system plant that would be available to us. The first alternative is not desirable given the scope of this project and budget constraints. Therefore, an attempt was made to find a nearly identical BWR sister plant to Millstone I that would have an available RELAP deck.

The search narrowed down the choices to two plants, Oyster Creek and Dresden II. Both plants are included in the SEP Containment Analysis work, so the RELAP blowdown decks are immediately available. Plant parameters comparing Millstone I, Oyster Creek, and Dresden II are shown in Table I.

Based on available data for BWRs and available RELAP decks for blowdown calculations, Oyster Creek is the best match for rated thermal power and reactor coolant inventory. The main problem is that Oyster Creek has a slightly smaller reactor vessel than Millstone I and it is a nonjet pump plant. The major change that would have to be implemented to the Oyster Creek deck is the renoding of the recirculation loops and inclusion of the jet pumps. This would be necessary in order to obtain the correct recirculation line break flow areas and blowdown rates. The same would have to be done to the main steam lines for the steam line break case as the two plants have different main steam configurations. However, at the present time we have on hand none of the information required to compare the Oyster Creek and Millstone water inventories at normal operating conditions. This information along with the downcomer flow areas is critical to the correct blowdown response. The RELAP deck would then have to be debugged to ensure proper results. There is extremely limited information available in the Millstone I FSAR as is needed to ensure the validity of the Oyster Creek deck modifications. After contacting NUSCO, it is not known at this time if the updated information required can be obtained in a timely manner. Therefore, this alternative will not be pursued any further.

The other choice is to use the blowdown data already being generated for the Dresden II plant as part of this SEP analysis. Millstone I and Dresden II are the same generation jet pump BWR plants and have nearly identical configurations and safety features. Dresden II is a larger plant at 800 MWe versus Millstone I at 560 MWe. It is proposed that the best course of action is to apply the Dresden II blowdown to the Millstone I containment analysis. The following reasons substantiate this action along with the comparison data of Table I:

(1) The resulting Millstone I containment analysis will definitely be conservative. Whereas with the other option of using the Oyster Creek deck, the uncertainties involved were not on the conservative side.

(2) Dresden's drywell free air volume is only 8% larger than Millstone. Both plants have the same vent pipe flow area and configuration.

(3) From FSAR information, the design basis accidents (double-ended recirculation line rupture) for both plants have the same sequence of events following the rupture. Both plants use the same ECCS equipment to control the accident, namely LPCI and core spray. The two important setpoints in this analysis are high drywell pressure and low-low water level. These setpoints are compared in Table I. The ECCS systems for both plants are shown in Table II. There are slight differences in the two plants' systems.

(4) For the double-ended steam line break inside the drywell, both plants have the same sequence of events following the pipe rupture. The important setpoints in this analysis are high steam flow, MSIV closure scram setpoint, and high drywell pressure. These are shown in Table I. Both plants use the

assumption of 10.5 seconds for complete MSIV closure. Both plants use core spray and LPCI to control the accident.

(5) Dresden's FSAR peak pressure response to the design basis accident is approximately 10% higher than Millstone. Millstone's FSAR peak pressure response is approximately 30% under its design value. Taking the above factors into consideration, using the Dresden II SEP blowdown applied to the Millstone I containment should give a Millstone peak pressure response that is within the design values and is also very conservatively calculated.

(6) Using the Dresden II blowdown will save several weeks of time and expense that would be needed for the other alternative to modify the Oyster Creek RELAP deck, debug it, and run the blowdown cases.

The only drawback at this time concerning the recommended method is the lack of FSAR design basis accident mass and energy release data for both Millstone I and Dresden II. If this were available, it would confirm the validity of the recommended approach. However, based on engineering judgment and the deficiencies of the other alternatives, the recommended course of action will give the most valid and conservative results within the framework and objectives of this project.

Based on this letter and our recent telephone conversations on this subject, work will proceed by applying the Dresden II blowdown data to the Millstone I containment. The contents of this letter show this to be an acceptable methodology.

Very truly yours,

John D. Atchison

John D. Atchison
Engineer

JDA:db

Enclosures

TABLE I

BWR PLANT PARAMETERS*

Parameter	Millstone I	Oyster Creek	Dresden II
Thermal Power (MW)	2011	1600 (now uprated to 1930 MW)	2527
Operating Pressure (psig)	1000	1000	1000
Recirculation Flow (lb/hr)	69×10^6	61×10^6	98×10^6
Steam Flow (lb/hr)	7.94×10^6	5.85×10^5	9.94 [illegible] 0^6
Circumscribed Core Dia.	177.1 in.	170.55 in.	189.7 [illegible]n.
Heat Transfer Surface Area	50,796 ft^2	49,137 ft^2	62,640 ft^2
Average Heat Flux ($Btu/hr\text{-}ft^2$)	129,640	107,470	131,860
Max. Heat Flux ($Btu/hr\text{-}ft^2$)	310,000	295,600	312,800
Core Subcooling (Btu/lb)	23.5	25.7	22.4
Core Ave. Void Fraction	38.9%	32%	29.9%
Core Ave. Exit Quality	13.1%	9.8%	10.1%
Water/UO_2 Volume Ratio (cold)	2.41	2.38	2.41
Number of Fuel Assemblies	580	560	724
Fuel Rod Array	7 x 7	7 x 7	7 x 7
Fuel Rod Pitch (inch)	.738	.738	.738
Fuel Pellet O.D. (inch)	.488	.488	.488
Clad Thickness (inch)	.0355	.0355	.032
Clad O.D. (inch)	.570	.570	.563
Active Fuel Length (inch)	144	144	144
Number of Control Rods	145	137	177
Core Equivalent Dia. (inch)	163.1	160.21	182.2

TABLE I (continued)

Parameter	Millstone I	Oyster Creek	Dresden II
Reactor Vessel ID	18'8"	17'9"	20'11"
Reactor Vessel Height	64'8"	63'10"	68'7"
Number of Recir. Loops	2	5	2
Recir. Pipe Size	28"	26"	28"
Number of Jet Pumps	20	none	20
Type of Primary Containment	pressure suppression	pressure suppression	pressure suppression
Drywell Cylindrical Dia.	34'2"	33'	37'
Drywell Spherical Dia.	64'	70'	66'
Drywell Free Air Volume	146,900 ft^3	180,000 ft^3	158,236 ft^3
Number Vent Pipes	8	10	8
Vent Pipe ID	6'9"	6'6"	6'9"
Vent Tubes Total Flow Area	286.3 ft^2	331.9 ft^2	285 ft^2
Vent Header ID	4'9"	4'7"	4'10"
Number of Downcomer Pipes	96	120	96
Downcomer Pipe ID	2'	1'11.5"	2'
Wetwell Water Volume	98,700 ft^3 max. 83,500 ft^3 min.	83,400	112,203
Wetwell Free Air Volume	125,100 ft^3 max. 109,900 ft^3 min.	127,000	117,245

TABLE I (continued)

Parameter	Millstone I	Oyster Creek	Dresden II
Torus ID	29'6"	30'	30'
Torus Major Diameter	102'	101'	109'
Number Main Steam Lines	4	2	4
Main Steam Line Dia.	20"	24"	20"

Protection System Setpoints	Millstone I	Oyster Creek	Dresden II
Reactor High Pressure	1085 psig	1050 psig	1070 psia
Reactor Low Level	2" above bottom of separator	1' below normal	1" above bottom of separator
Reactor Low-Low Level	49" below bottom of separator	5' below normal	59" below bottom of separator
High Neutron Flux	120% rated power	120% rated power	120% rated power
Drywell High Pressure	2 psig	2 psig	2 psig
Scram on MSIV % closure	10%	10%	10%
Main Steam High Flow	120% rated flow	--	120% rated flow

*All values are taken from the respective plant's FSAR and Tech. Spec. The information may not be up to date, especially Oyster Creek which has been uprated in power.

TABLE II

EMERGENCY COOL COOLING SYSTEMS

Function - Plant	Number of Pumps	Design Coolant Flow*	Pressure Range*
Core Spray			
Millstone I	2 (100% each)	3600 gpm @ 90 psi	245 to 0 psi
Dresden II	2 (100% each)	4500 gpm @ 90 psi	260 to 0 psi
LPCI			
Millstone I	4 (33% each)	7500 gpm @ 165 psi 15,000 gpm @ 0 psi	235 to 0 psi
Dresden II	4 (33% each)	8000 gpm @ 200 psi 14,500 gpm @ 20 psi	275 to 0 psi
FWCI			
Millstone I	3 (100% each)	8000 gpm	1125 to 100 psig
HPCI			
Dresden II	1 (100%)	5600 gpm	1125 to 150 psig

*RV internal pressure to drywell pressure differential.