August 27, 1982

Docket No. 50-409
LS05-82- 08-060

Mr. Frank Linder
General Manager
Dairyland Power Cooperative
2615 East Avenue South
LaCrosse, Wisconsin 54601

Dear Mr. Linder:

SUBJECT: SYSTEMATIC EVALUATION PROGRAM (SEP) FOR THE LACROSSE BOILING WATER REACTOR - EVALUATION REPORT ON TOPICS VI-2.D AND VI-3

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," which reflect the comments provided in your August 4, 1982 letter. Our evaluation compares your facility, as described in Docket No. 50-409, with the criteria currently used by the regulatory staff for licensing new facilities.

Our review has shown that the recirculation line break is the limiting event for containment temperature and pressure response and that the original analysis of this event was conservative (i.e., 48.2 psig and 272°F vs 43 psig and 265°F). Therefore, we conclude that the original LaCrosse Boiling Water Reactor design with respect to this topic is conservative and equivalent to current licensing criteria.

SE04

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.

DSU USE (38)

Sincerely,

ADD: T. Michaels

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

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*See previous yellow for additional concurrences.

Enclosure:
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DATE			8/23/82	8/23/82	8/23/82	8/25/82	8/25/82

Docket No. 50-409
LS05-82

Mr. Frank Linder
General Manager
Dairyland Power Cooperative
2615 East Avenue South
LaCrosse, Wisconsin 54601

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SUBJECT: SYSTEMATIC EVALUATION PROGRAM (SEP) FOR THE LACROSSE BOILING WATER REACTOR - EVALUATION REPORT ON TOPICS VI-2.D AND VI-3

Enclosed is a copy of our 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," which reflect the comments provided in your August 4, 1982 letter. This evaluation compares your facility, as described in Docket No. 50-409, with the criteria currently used by the regulatory staff for licensing new facilities.

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,

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

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Mr. Frank Linder

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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

LACROSSE NUCLEAR POWER PLANT

DOCKET NO. 50-409

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I. Introduction

The La Crosse Nuclear Power Plant began commercial operations in 1969. Since then the United States Nuclear Regulatory Commission staff's safety review criteria have changed. As part of the Systematic Evaluation Program (SEP), the containment pressure and heat removal capability (SEP Topic VI-3) and the mass and energy release for possible pipe break inside containment (SEP Topic VI-2.D) have been re-evaluated. The purpose of this evaluation is to document any existing deviations from current safety criteria that pertain to the containment pressure and heat removal capability and the mass/energy release for possible pipe break inside containment. Independent analysis in accordance with current criteria were performed by LLNL to determine the adequacy of the containment design and to provide input for unresolved safety issue (USI) A-24, Qualification of Class 1E Safety Related Equipment. The significance of any identified deviations, and recommended corrective measures to improve safety, will be the subject of a subsequent, integrated assessment of the LaCrosse plant.

II. Review Criteria

The review criteria used in the current evaluation of SEP Topics VI-2.D and VI-3 for the La Crosse 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 break inside containment, and/or containment pressure and heat removal capability.

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

(2) VI-7.B, ESF Switchover from Injection to Recirculation Mode (Automatic ECCS Realignment)

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

(4) X, Auxiliary Feedwater System

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

IV. General 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 pressure and temperature design conditions important to safety are not exceeded for as long as the postulated accident conditions require. GDC 38 requires that a containment heat removal system be provided to reduce the containment pressure and temperature following any loss-of-coolant accident (LOCA) and maintain them at an acceptably low level. This safety system is to function assuming a single failure. GDC 50 requires that the containment structure and the containment heat removal system shall be designed so that the structure can accommodate, with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin and the containment model are discussed in the Standard Review Plan (SRP) NUREG-0800 Section 6.2.1, Containment Functional Design; the margin is obtained from the conservative calculation of mass/energy release. The containment design basis includes the effects of stored and generated energy from the accident. Calculations of the energy available for release should be made in accordance with the requirements of 10 CFR Part 50, Section 50.46 and Appendix K, paragraph I.A, and the conservatism as specified in SRP 6.2.1.3. In general, calculations of the mass and energy release rates for a loss-of-coolant accident should be performed in a manner that conservatively establishes the containment internal design pressure and temperature (i.e. maximizes the post-accident containment pressure and temperature).

By reviewing the licensee's analysis, deviations from current licensing criteria can be identified and independent analyses performed, to evaluate the significance of these deviations. In the analysis, "the best estimate" method is used; i.e., by using actual plant design data, this best estimate analysis remains a reasonably conservative analysis of containment response. The evaluation is completed by comparing the results with the containment design basis.

V. Evaluation

In the case of BWRs, it is necessary to evaluate the effect of pipe breaks below the level of the core for maximum containment pressure and of pipe breaks above the level of the core for maximum containment temperature. Based on our review of the existing docket for LaCrosse, the break locations analyzed by Dairyland Power Cooperative are for breaks only occurring below the level of the core.

In the LaCrosse BWR Hazards Summary Report a spectrum of recirculation line breaks was analyzed to determine the peak post-accident pressure.(1) All of the resultant peak calculated pressures were determined to be below the containment design pressure of 52 psig. The maximum calculated peak pressure was determined to be 48.2 psig for a recirculation line break. The post-accident containment temperature conditions reached 272°F. The containment design temperature is 280°F. The initial conditions and assumptions presented in the report were not adequate to determine whether or not the analysis was consistent with current criteria. Therefore, in addition to reviewing the applicant's analysis, a confirmatory and independent analysis was performed by LLNL for the USNRC, which is presented in Appendix A of this report. Mass and energy release rates utilized in the analysis were calculated using RELAP-4/MOD 7 in accordance with current criteria. Calculations of the post-accident containment pressure and temperature responses were made using CONTEMPT-LT/028. One of the analyses made was the double-ended recirculation line break. The calculated transient reflects a post-accident peak drywell pressure of 43.psig and a peak temperature of 265°F. These results are plotted in Figures 1 and 2. Both the utility analysis and our analysis show that the peak pressure and temperature are below the containment's design values.

In addition to the recirculation line break case , the current criteria state that steam line breaks above the level of the core must be considered. The licensee did not perform main steam line break analyses. Therefore, independent analyses were performed. These are discussed in Appendix A. The analyses were performed for three main steam line break sizes, 0.01 ft^2, 0.10 ft^2 and 0.634 ft^2. The 0.01 ft^2 break analysis was used as a bounding case to determine the amount of time the reactor operator would have to initiate containment sprays and/or the Manual Depressurization System. The analysis indicated that the containment would reach the design pressure 56 minutes after onset of the break. The design temperature would be reached in 53 minutes. Operator action would then be needed and should become effective within 53 minutes. If accomplished on time, the containment design limits would not be exceeded.

Due to their more rapid depressurization rates, operator action would not be required for either the 0.1 ft^2 or the 0.634 ft^2 break; these blowdowns could be accommodated solely by the containment's passive heat sinks. Of the two, the 0.634 ft^2 break, corresponding to a double-ended rupture of an eight-inch main steam line, caused the more severe containment response of 33 psig for pressure and 250°F for temperature.

VI. Conclusions

The analyses submitted by the licensee have been reviewed and found to be within the design limits of 280°F and 52 psig for the LaCrosse plant. A confirmatory analysis was performed for the recirculation line break accident with resulting containment response of 43 psig maximum for pressure and 265°F maximum for temperature.

Also, independent analyses were performed for the 0.01 ft^2, 0.10 ft^2 and 0.634 ft^2 steam line break. The latter two break sizes produce containment responses less severe than the recirculation line break accident. The former the 0.01 ft^2 break, was found to require operator action which should become effective 53 minutes into the transient. This time frame is adequate for the reactor operator to activate manually the containment sprays or MDS. Having done so the upper-bound analysis indicates that the containment response would not exceed design values.

The recirculation line break analyses yield more adverse temperature and pressure responses than these of the steam line break analyses and, therefore, may be used as input for the equipment qualification of safety-related equipment effort, USI A-24.

VII. References

1. LaCrosse BWR Hazards Summary Report, Dairyland Power Cooperative, 1967.

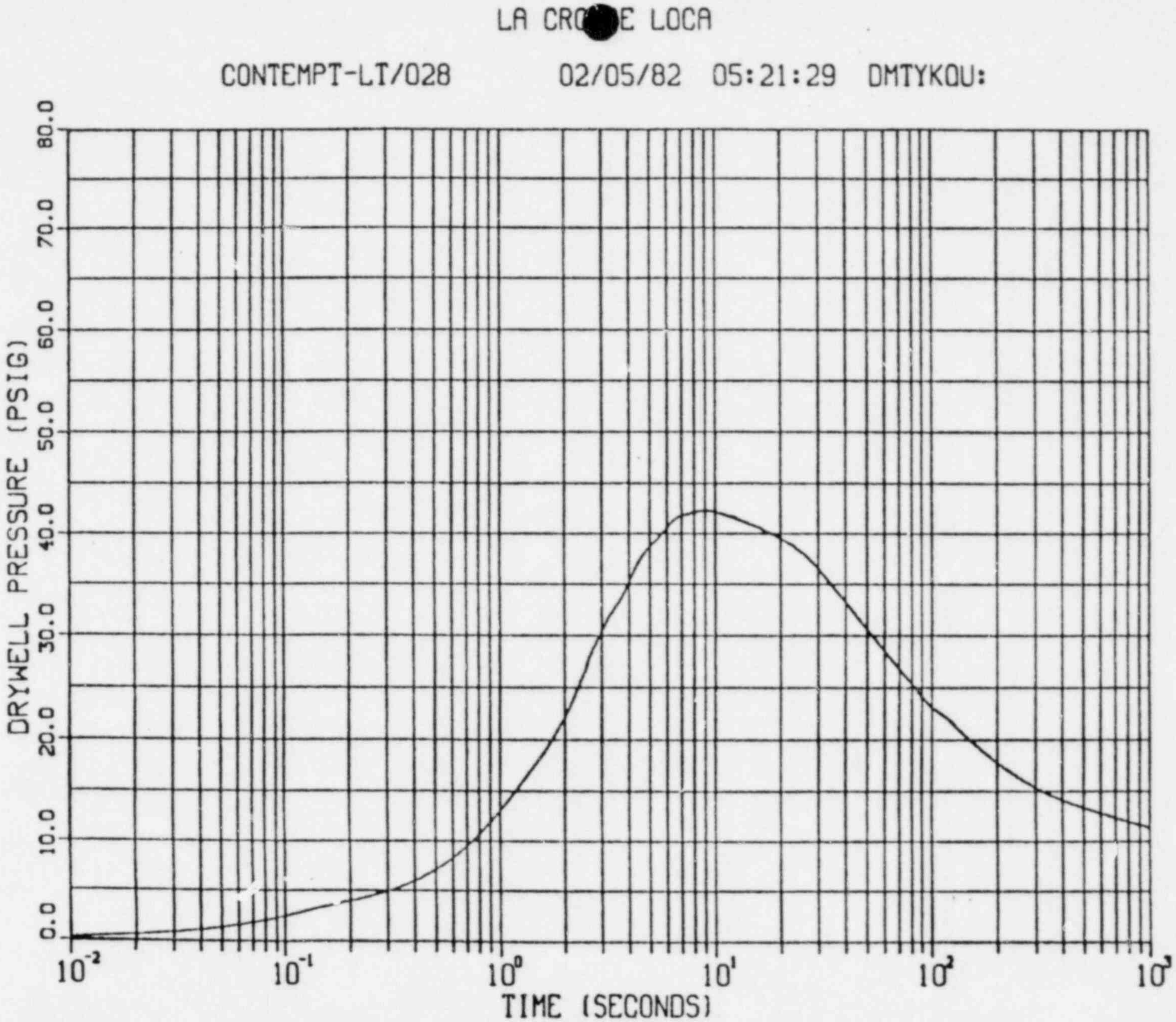

Figure 1. Containment Pressure Response to a
3.59 sq. ft. Cold Leg Discharge Break

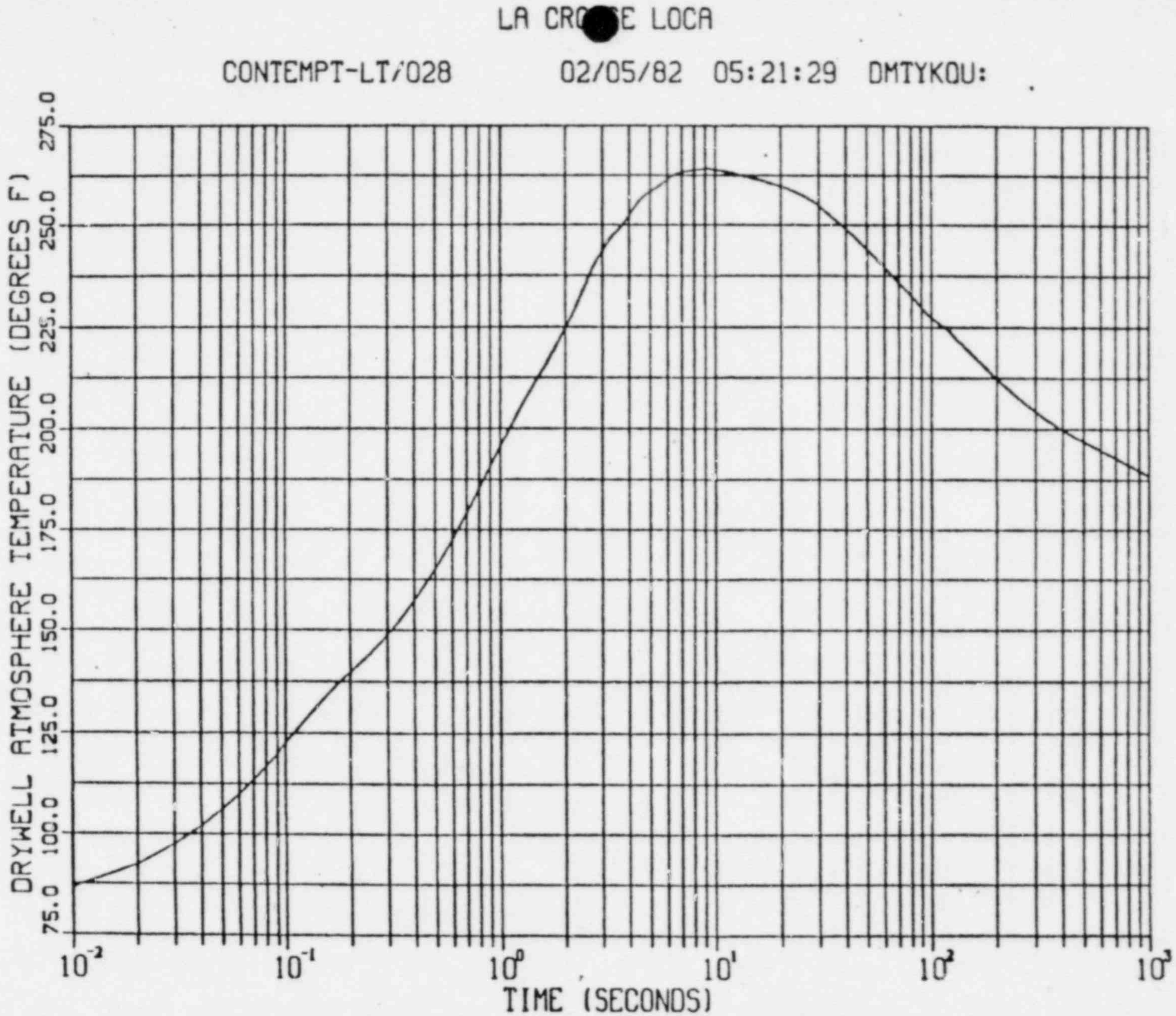

Figure 2. Containment Temperature Response to a 3.59 sq. ft. Cold Leg Discharge Break

APPENDIX A: TECHNICAL EVALUATION REPORT

SEP Containment Analysis and Evaluation for the La Crosse Nuclear Power Plant

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1.0 Introduction and Background

As part of the Systematic Evaluation Program (SEP), the containment functional design capability of the La Crosse Nuclear Power Plant has been re-evaluated. 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.[1] The significance of the identified deviations and recommended corrective measures will be the subject of a subsequent integrated assessment of the La Crosse plant.

The containment structure encloses the reactor and is the final barrier against the release of radioactive fission products to the atmosphere 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 a post-accident safety function must be capable of withstanding the resulting adverse pressure and temperature conditions.

2.0 Containment Functional Design Description

La Crosse is a 165 MWt General Electric Boiling Water Reactor (BWR). In La Crosse water enters the bottom of the reactor vessel through four 16 inch pipes and passes upward through the core passing along the fuel rods. Boiling produces steam and is separated in the steam dome. From there, the steam leaves the vessel through two 8 inch pipes. These lead to a single 10 inch line that passes from the containment building to the turbine building. The hot water which is removed from the steam in the steam dome exits the steam dome through four 16 inch pipes. These combine to make up the two 20 inch recirculation lines. The water is returned to the reactor vessel by two recirculation pumps.

The steam line can be isolated by a hydraulically operated isolation valve and can be closed in 10 seconds. This valve can be controlled remote manually from the control room and is closed automatically upon signals for low reactor water level, low steam pressure at the turbine stop valve, or low main condenser vacuum. The steam line can also be isolated by the turbine building steam isolation valve. However, this valve is not automatic and is controlled from the control room.

The main feed return line has a check valve inside the containment building and a remotely operated shutoff valve in the turbine building.

The containment structure consists of a cylindrical steel vessel, 60 feet in diameter. The vessel has an internal free volume of 264,160 cubic feet and is designed to withstand an internal pressure of 52 psig. The containment structure encloses the reactor vessel, primary recirculation pipes, and equipment needed to operate the emergency core cooling system (ECCS), and containment heat removal system.

The containment heat removal systems consist of a containment spray system and passive heat sinks. The containment spray system is manually operated. Water is supplied to the building spray system from a 42,000 gallon storage tank located at the top of the containment vessel. The piping connection to the emergency core spray system is on the bottom of the tank. The connection to the spray headers of the building spray system is a standpipe within the tank. The bottom of the standpipe is at a sufficient elevation above the bottom of the tank so that 15,000 gallons of water is available for the emergency core spray system at all times except during refueling. The minimum amount of water available for containment spray at full power is 11,300 gallons. Building spray is delivered by gravity feed at 1000 gpm to the spray headers. The containment spray system is not built to safety class 1.

In addition to the containment spray system, containment heat removal is brought about by the presence of passive heat sinks. The containment heat sink data is present in Table 6.

2.1 Review of the La Crosse Containment Design Analysis

Two separate calculations make up the containment design analysis. The first is the mass and energy release analysis for postulated LOCAs. This provides the time dependent mass and energy input from the primary system into the containment structure. The second calculation is the containment response to this mass and energy input. The containment response results in the time-dependent containment temperature and pressure profiles. The severity of the containment response depends on the magnitude and nature of the mass and energy release from the postulated LOCA. In turn, the magnitude and nature of the mass and energy release to the containment is dependent on the break location. If the break is below the core the break flow will be initially single phase liquid. This results in a fast blowdown of the mass and energy release to the containment at a relatively low enthalphy. 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 are found to produce the most severe pressure response in the containment and steam line breaks above the core produce the most severe temperature response.

The acceptance criteria used to evaluate the La Crosse Containment Design Analysis was based on the Standard Review Plan (SRP) Section 6.2.1. For the containment design analysis to be found acceptable the results from both the mass and energy release calculation and the containment response calculation must meet the acceptance criteria specified in the SRP.

2.2 Review of Pipe Breaks Inside the Reactor Coolant Pressure Boundary

The SRP specifies several acceptance criteria applied to the mass and energy release analysis for primary system pipe breaks. Among these are the break location. The only containment functional design analysis performed is described in the La Crosse BWR Hazards Summary Report.(2) In this analysis the most severe mass and energy release rate calculation for containment design was done assuming a double-ended break in the recirculation line. The break location was on the pump discharge side at a point near the bottom of the reactor vessel. The maximum calculated peak pressure was determined to 48.2 psig. The peak post accident containment temperature conditions was 272°F. A substantial amount of information needed to evaluate the analysis is not contained in this report (e.g., information pertaining to the choke flow correlation and the heat transfer assumptions used in the analysis). Without this information it is not possible to conclude whether or not the containment design analysis presented in the La Crosse BWR Hazards Summary Report is adequate.

2.3 Reanalysis of La Crosse Containment Design

As mentioned earlier in Section 2.1, Review of La Crosse Containment Analysis, there are two separate calculations which make up the containment design analysis, the mass and energy release rate and the containment response. The mass and energy release can be the result of either a recirculation line break or a steam line break. The recirculation line break results in the worst condition for calculating the peak pressure inside the containment. The steam line pipe break analysis is the worst case for temperature conditions inside the containment.

As pointed out in the previous section, the analyses submitted by Dairyland Power Cooperative lacked necessary information regarding initial conditions, assumptions or complete results to determine whether or not the current criteria were met. Both a recirculation line break and a steam line break analysis was performed again by LLNL and are discussed below.

3.0 Recirculation Line Pipe Breaks

For a recirculation line break a design basis accident (DBA) LOCA generates the highest containment temperatures and pressures for breaks which occur below the core mixture level. The LOCA analysis was performed using the RELAP4-MOD7 computer code. The RELAP4 input deck was obtained from Dairyland Power Cooperative at the request of the NRC. The deck was reviewed by LLNL to evaluate the selected code options, initial conditions and boundary conditions. The plant physical description was assumed to be as-built.

Additional information required was taken from the La Crosse BWR Hazards Summary Report.(2)

The initial conditions and boundary conditions for this analysis were selected to satisfy the requirements of the Standard Review Plan section 6.2.1.

The following is a listing of the initial conditions and a summary of the assumptions used in this analysis.

1. The reactor is operating at 102% of design power at the time the recirculation pipe breaks. This will produce the maximum core heat generation rate.
2. A complete loss of normal offsite AC power occurs simultaneously with the pipe break.
3. The recirculation pump discharge pipeline is considered to be instantly severed at both ends. Coolant being discharged from both ends of the break results in the most rapid coolant loss and depressurization. The break area is assumed to be 3.59 sq. ft. and represents a double-ended break of one of the 20 inch diameter recirculation lines.
4. The reactor is assumed to go subcritical at the time of accident initiation. Scram would normally occur in less than one second due to a high drywell pressure signal.
5. The sensible heat released in cooling the fuel rods and the core decay heat are included in the reactor vessel depressurization calculation. The rate of energy release is calculated using a conservatively high heat transfer coefficient throughout the depressurization. This maximizes the heat removal rate from the core. Calculations of heat transfer from surfaces exposed to liquid were based on nucleate boiling heat transfer. For surfaces exposed to steam, the heat transfer calculation was based on forced convection.
6. The main steam line isolation valves are assumed to be closed at the start of the accident. By assuming closure of these valves, the reactor vessel is maintained at a high pressure, which maximizes the discharge of high energy steam and water into the primary containment.

7. The feedwater flow is assumed to be closed at time zero. This conservative assumption is made because the relatively cold feedwater flow, if considered to continue, tends to depressurize the reactor vessel, and causes a reduction in the discharge of steam and water into the primary containment.
8. The vessel depressurization flow rates are calculated using a discharge coefficient of 1.0, with the Henry Fauske correlation for subcooled and Moody correlation for saturated fluid. A 14.7 psia back pressure was assumed to maximize the mass and energy release throughout the blowdown. The blowdown calculation using RELAP 4 was run until the primary system pressure dropped below the containment design pressure of 52 psig. At this time 1.2 times the ANS decay heat curve was used.
9. Emergency core cooling system was not modeled since sufficient information was not provided in the RELAP model obtained from Dairyland Power Cooperative.

The results of this analysis are the time dependent mass and energy release rates presented in Table 1.

3.1 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 and a description of the containment heat removal systems. Passive containment heat sinks were the only heat removal systems accounted for since the containment spray system is manually operated and not safety class 1. The containment heat sinks modeled are described in Table 6. The mass and energy release rate data used were taken from the blowdown calculation of the recirculation line break presented in the previous section.

The containment response calculation was made using the CONTEMPT-LT/28 computer code. The program models the containment as a one volume dry containment. The initial conditions used in the analysis are summarized in Table 2.

3.2 Containment Response Results

The containment pressure and temperature response to a recirculation line break are shown in Figures 1 and 2. The calculated transient reflects a peak post-accident containment pressure of 43 psig and a temperature of 270°F. This compares with a containment pressure of 48.2 psig calculated by Dairyland Power Cooperative presented in the FHSR for La Crosse. The containment design pressure is 52 psig. There is, therefore, an 8% margin between the peak calculated pressure and the containment design pressure.

4.0 Main Steam Line Pipe Breaks

Analyses of the containment response to a steam line break were also made. This analysis is performed to determine the most severe long term pressure and temperature condition in the containment following a pipe break. The blowdown calculation was done using RELAP4-MOD7. The input deck used was the same one as that used in the recirculation line break with the break location moved to the main steam line. Three break sizes were run, a 0.01, 0.1 and 0.634 sq. ft. The 0.634 sq. ft. break represents the area of a double-ended break of the 8 inch steam line.

The initial conditions and boundary conditions for this analysis were selected to satisfy the requirements of the Standard Review Plan section 6.2.1. The following is a listing of the initial conditions and a summary of the assumptions used in the analysis.

1. The reactor is operating at 102% of design power at the time the steam line breaks.
2. A complete loss of normal offsite AC power occur simultaneously with the pipe break.
3. The reactor is assumed to go subcritical at the time of accident initiation.
4. The sensible heat released in cooling the fuel rods and the core decay heat are included in the reactor depressurization calculation. The rate of energy release is calculated using a conservatively high heat transfer

coefficient throughout the depressurization. This maximizes the heat removal rate from the core. Calculations of heat transfer from surfaces exposed to liquid were based on nucleate boiling heat transfer. For surfaces exposed to steam, the heat transfer calculation was based on forced convection.

5. The main steam isolation valves are assumed to be closed at the initiation of the accident.
6. The feedwater flow is assumed to be closed at time zero.
7. The vessel depressurization flow rates are calculated using a discharge coefficient of 1.0, with Henry Fauske correlation for subcooled and Moody correlation for saturated fluid. A 14.7 psia back pressure was assumed to maximize the mass and energy release throughout the blowdown. The blowdown calculation using RELAP4 was run until the mass and energy release rate stabilized. At this time the blowdown rate was assumed constant for a conservative length of time to ensure the reactor vessel was depressurized. Then 1.2 times the ANS decay heat curve was used.
8. Emergency core cooling system was not modeled since sufficient information was not provided in the RELAP model obtained from Dairyland Power Cooperative.

The results of this analysis are the time dependent mass and energy release rates presented in Table 3, 4, and 5. The RELAP 4 code analyses were carried out to 200 seconds, 150 seconds and 60 seconds for the 0.01 ft^2, 0.10 ft^2 and 0.634 ft^2 breaks, respectively. From these points the blowdown was conservatively held constant until 4000 seconds for the 0.01 ft^2 break and 400 seconds for both the 0.10 ft^2 and 0.634 ft^2 breaks.

4.1 Containment Response to a Main Steam Line Break

The input data for the containment response calculation consist of the mass and energy release rates to the containment and the available containment heat sink data. The containment heat sinks modeled are described in Table 6. The mass and energy release rates were taken from the blowdown rates presented in the section 4.0.

The containment response calculation was made using the same CONTEMPT model used in the recirculation line break analysis, section 3.1. The initial conditions used in the analysis are summarized in Table 2.

4.2 Containment Response to Main Steam Line Break Results

Figures 3 and 4 show the containment pressure and temperature responses, respectively, for the 0.01 ft^2 break. These curves serve as an upper bound since the blowdown was held constant from 20ᶜ seconds to 4000 seconds. Inspection of these curves indicates that the design pressure of 52 psig and temperature of 280°F would be reached in 56 minutes and 53 minutes, respectively. This is the amount of time that exists for the reactor operator to take appropriate action such as to manually actuate the containment sprays or the Automatic Depressurization System (ADS). These actions having been taken and become effective by 53 minutes will insure that the containment will not exceed its design pressure and temperature limits. The staff believes that the LaCrosse plant reactor operators would be able to respond within this time frame to take action which would terminate the transient and limit the containment response to within allowable design values.

The response to a 0.1 ft^2 steam line break are shown in Figures 5 and 6. The calculated transient in this case reflects a peak post accident containment pressure of 21 psig and a temperature of 220°F. The response to the 0.634 ft^2 break is shown in Figures 7 and 8. This represents a double ended break of the eight inch steam line. The calculated transient reflects a peak post accident pressure of 33 psig and a temperature of 250°F. Neither the 0.10 ft^2 nor the 0.634 ft^2 break require operator action due to the fact that these larger break sizes cause the primary system to depressurize in a much faster time frame than the smaller 0.01 ft^2 break.

Assuming operator action as discussed above for the 0.01 ft^2 break, the peak containment pressures are substantially below the design value of 52 psig for all three cases. The post accident temperature for the 0.634 ft^2 steam line break results in the most severe temperature conditions for a main steam line break but is still less than 265°F resulting from a recirculation line break.

5.0 Conclusions

Based on this review of the LaCrosse docket and the subsequent analysis performed by LLNL, it is concluded that the LaCrosse containment design pressure meets current NRC criteria. The containment atmosphere conditions as a result of recirculation line break provides the most severe temperature conditions for equipment qualification of safety related equipment.

6.0 References

1. NUREG 0485, Vol. 3, No. 4, March 1, 1981, Systematic Evaluation Program.

2. La Crosse BWR Hazards Summary Report, Dairyland Power Cooperative, 1967.

Table 1. Recirculation Line Blowdown Mass and Energy Release Data (3.59 ft^2 Break)

Time (seconds)	Flow (lbm/sec)	Energy (Btu/lbm)
0.0	21180.	593
0.1	21180.	593.
0.2	11473.	555.
0.4	13203.	578.
0.6	15308.	576.
0.8	16255.	571.
1.0	15618.	566.
1.2	14320.	564.
1.4	11985.	561.
2.0	11504.	593.
2.2	13413.	593.
2.8	5440.	634.
3.0	10412.	608.
3.4	5270.	599.
4.0	8104.	615.
4.5	4178.	693.
5.0	2856.	665.
6.0	3330.	635.
7.0	1158.	687.
8.0	1374.	615.
9.0	928.	608.
10.0	428.	695.
10.5	330.	684.
30.0*	5.77	1200.
100.0	5.27	1200.
400.0	3.75	1200.
1000.0	2.95	1200.
4000.0	2.03	1200.
10000.0	1.54	1200.

* Assume steam decay heat cure at 30.0 sec.

Table 2. Initial Conditions for Containment Response Calculations (taken from Reference 2)

Containment	
Net free volume	264,160 ft^3
Pressure	14.7 psia
Temperature	80^oF
Relative humidity	100 percent
Liquid pool surface area	2827. ft^2
Outside air temperature	100^oF
Heat transfer multiplier	1.0
Mass transfer multiplier	1.0

Table 3. Main Steam Line Break Mass and Energy Release Rate Data (0.01 sq. ft. break)

Time (seconds)	Flow (lbm/sec)	Energy (Btu/lbm)
0.	28.0	1187.
1.	28.0	1187.
2.	27.8	1186.
3.	27.7	1184.
4.	26.9	1183.
5.	26.6	1181.
10.	25.9	1183.
15.	25.7	1183.
20.	25.6	1183.
30.	25.4	1184.
50.	25.0	1185.
60.	24.7	1185.
80.	24.3	1185.
90.	24.1	1186.
95.	24.0	1186.
100.	23.9	1187.
150.	22.9	1189.
200.	21.9	1190.
4000.	21.9	1190.

Table 4. Main Steamline Break Mass and Energy Release Rate Data (0.1 sq. ft. break)

Time (Seconds)	Flow (lbm/sec)	Energy (Btu/lbm)
0.0	257.9	1177.
1.0	250.4	1175.
2.0	250.4	1175.
3.0	241.5	1175.
4.0	234.7	1178.
5.0	229.4	1183.
6.0	224.4	1185.
7.0	221.7	1185.
8.0	218.8	1185.
9.0	215.8	1186.
10.0	212.7	1186.
15.0	214.8	1112.
20.0	253.9	946.
25.0	246.1	925.
30.0	218.4	959.
35.0	178.6	1068.
40.0	156.0	1131.
45.0	149.4	1149.
50.0	137.9	1131.
60.0	113.5	1201.
70.0	101.2	1202.
80.0	90.1	1202.
90.0	81.1	1202.
100.0	73.5	1202.
110.0	66.8	1201.
130.0	55.8	1200.
150.0	47.8	1199.
400.0	47.8	1199.
401.0*	3.75	1200.
1000.0	2.94	1200.
4000.0	2.80	1200.
10,000.0	1.54	1200.

* Assume constant out to 400 seconds then steam decay heat curve.

Table 5. Main Steam Line Break Mass and Energy Release Data
(.634 sq. ft. Break)

Time (Seconds)	Flow (lbm/sec)	Energy (Btu/lbm)
0.	1540.	1153.
0.1	993.8	1091.
0.2	682.2	1064.
0.3	645.6	1108.
0.4	626.6	1129.
0.5	623.6	1137.
1.0	599.6	1143.
2.0	549.6	1147.
3.0	515.2	1150.
4.0	495.2	1152.
5.0	460.2	737.
6.0	770.6	710.
7.0	724.8	728.
8.0	707.6	755.
9.0	618.8	791.
10.0	522.4	827.
15.0	508.0	753.
20.0	737.0	698.
25.0	327.4	814.
30.0	498.6	858.
35.0	281.0	907.
40.0	209.8	987.
45.0	223.8	939.
50.0	183.0	1003.
60.0	104.0	943.
400.0	104.0	943.
401.0*	3.75	1200.
1000.0	2.94	1200.
4000.0	2.80	1200.
10000.0	1.54	1200.

* Assume constant flow out to 400 seconds then steam decay heat curve.

Table 6. Containment Structural Heat Sinks
(Taken from Reference 2)

A. Heat Sink Descriptions

1. Containment dome	
Surface Area, ft^2	5670.
Composition, thickness ft	
Steel	0.05
Insulation	0.159
2. Misc. steel equipment	
Surface Area, ft^2	39300.
Composition, thickness ft	
Steel	0.0417
3. Shadow shield	
Surface Area, ft^2	14620.
Composition, thickness ft	
Concrete	0.75
4. Outside biological shield	
Surface Area, ft^2	4710.
Composition, thickness, ft	
Concrete	5.5
5. Ceilings	
Surface Area, ft^2	3600.
Composition, thickness, ft	
Concrete	1.0
6. Floors	
Surface Area, ft^2	3600.
Composition, thickness, ft	
Concrete	1.0
7. Walls	
Surface Area, ft^2	14380.
Composition, thickness, ft	
Concrete	2.4

Table 6. Containment Structural Heat Sinks (continued)

8. Pump room wall

Surface Area, ft^2	1090.0
Composition, thickness, ft	
Concrete	2.4

B. Material Properties

Material	Thermal Conductivity (Btu/hr ft^2 ^oF)	Volumetric Heat Capacity (Btu/hr ft^3 ^oF)
Steel	27.00	58.80
Concrete	0.9202	22.62
Insulation	0.020	2.0

NOTE:

1. All heat sinks modeled as rectangular slabs with one side exposure to the containment atmosphere and the other insulated.
2. Heat transfer based on Uchida correlation throughout the transient.

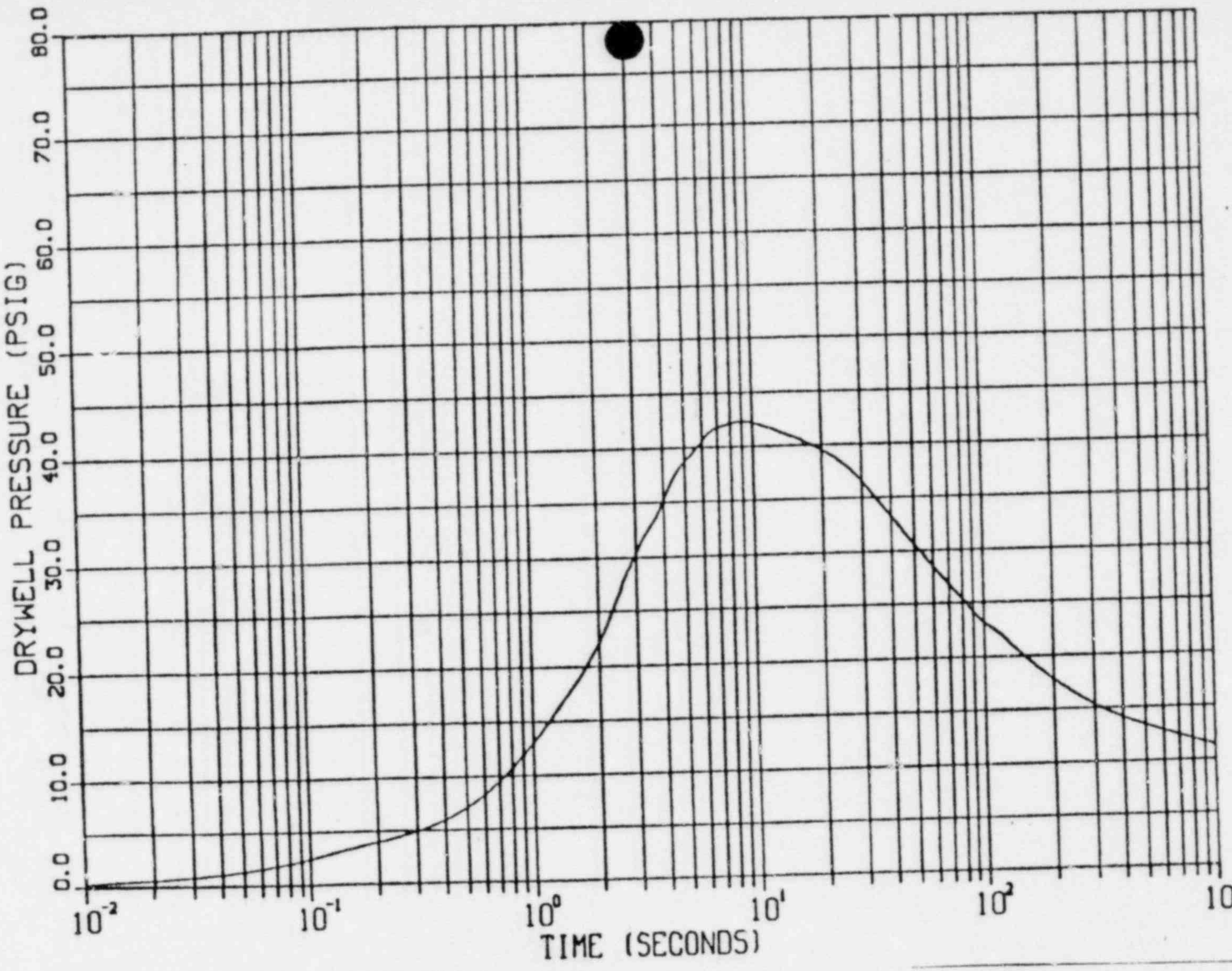

Figure 1. Containment Pressure Response to a 3.59 sq. ft. Cold Leg Discharge Break

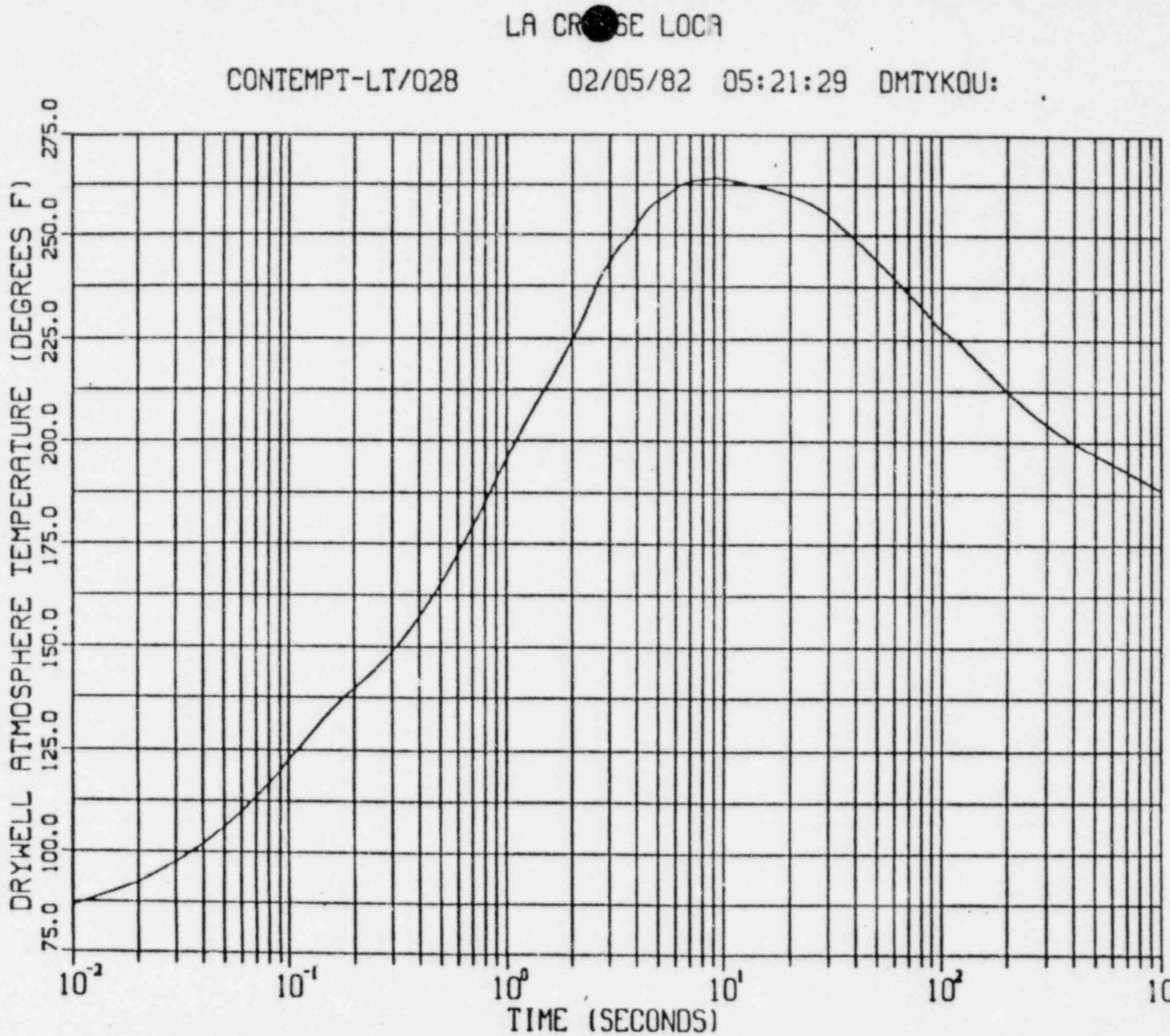

Figure 2. Containment Temperature Response to a 3.59 sq. ft. Cold Leg Discharge Break

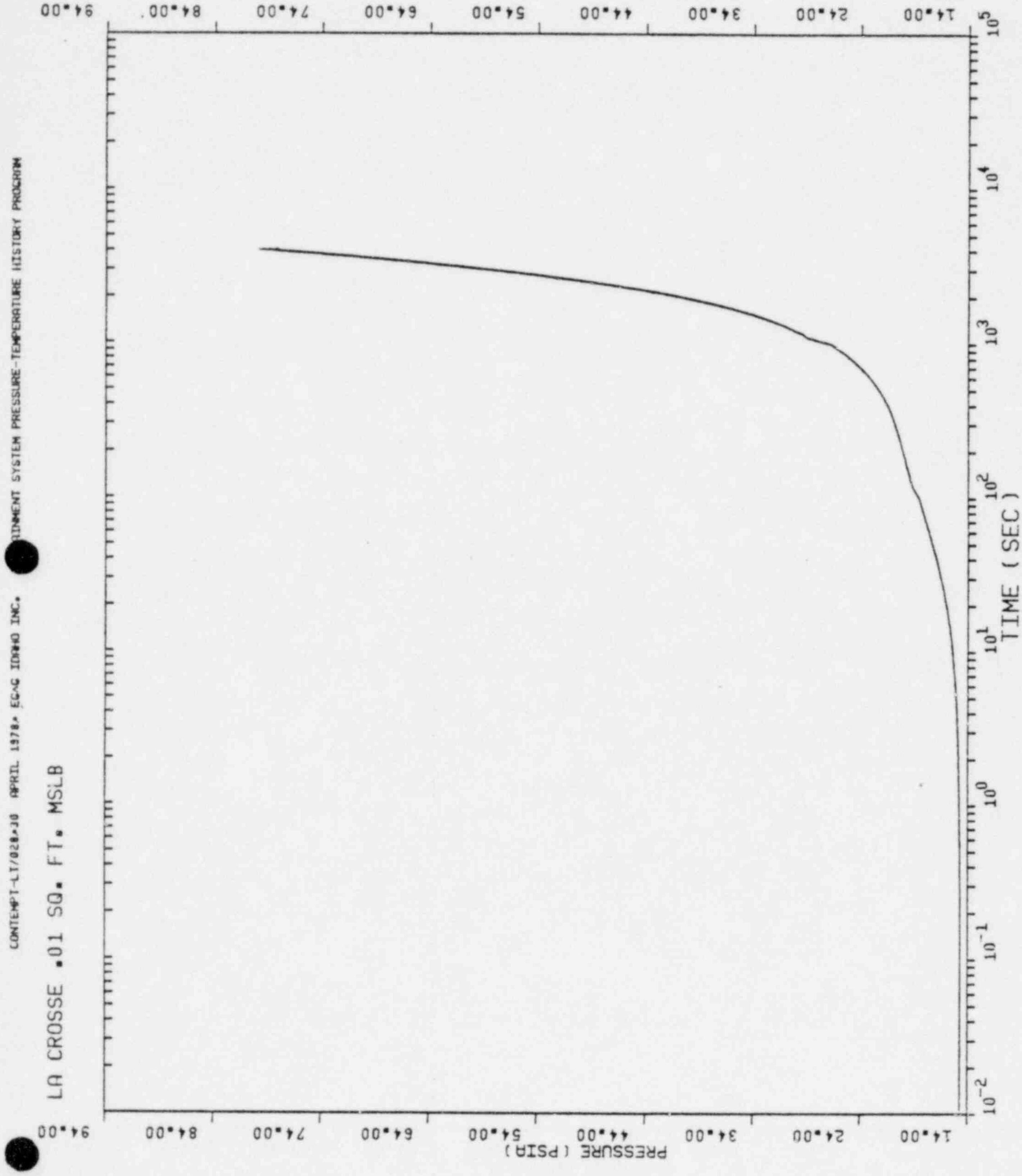

FIGURE 3

CONTAINMENT PRESSURE RESPONSE TO A 0.01 FT^2 MSLB

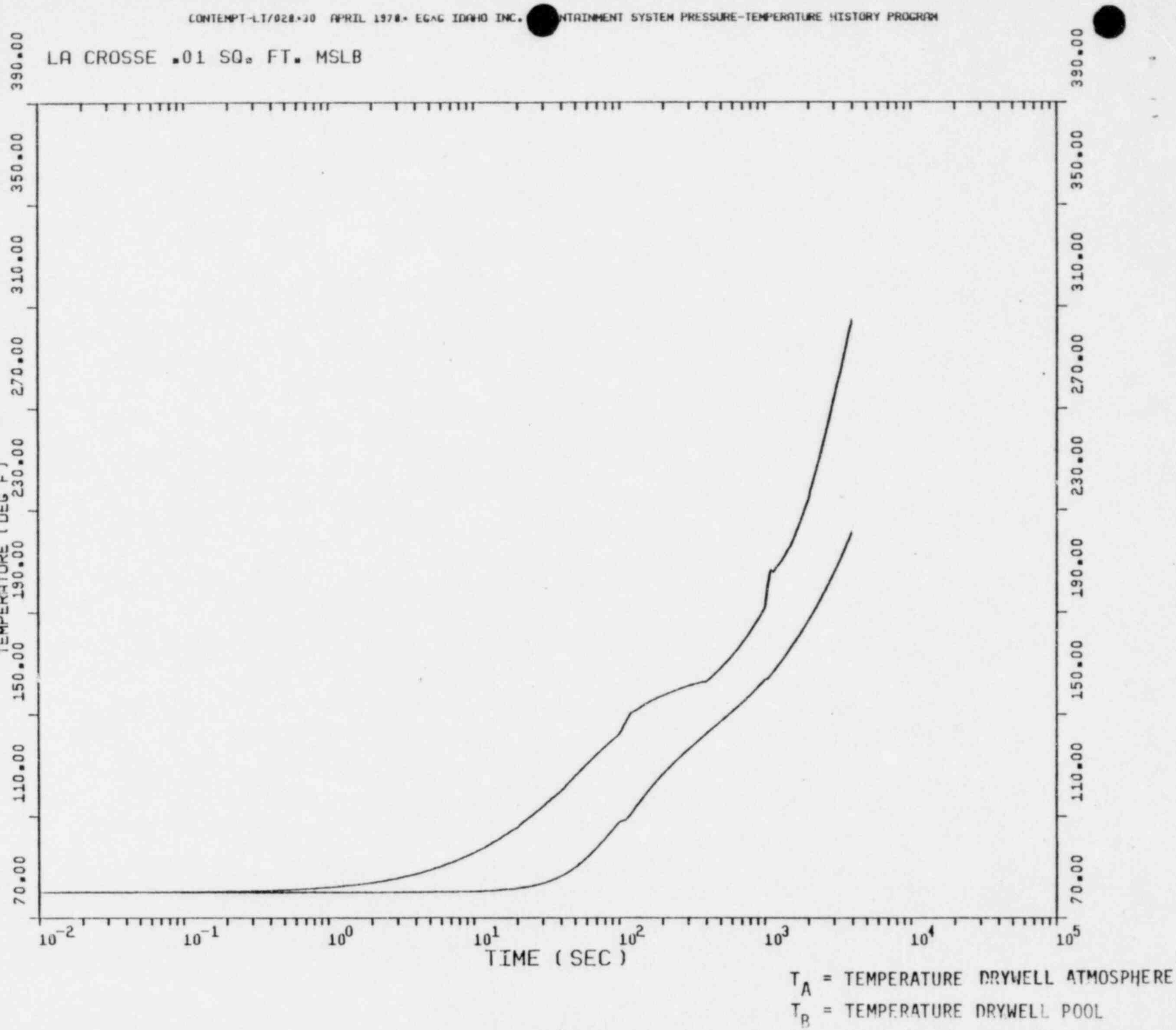

FIGURE 4

CONTAINMENT TEMPERATURE RESPONSE TO A 0.01 FT^2 MSLB

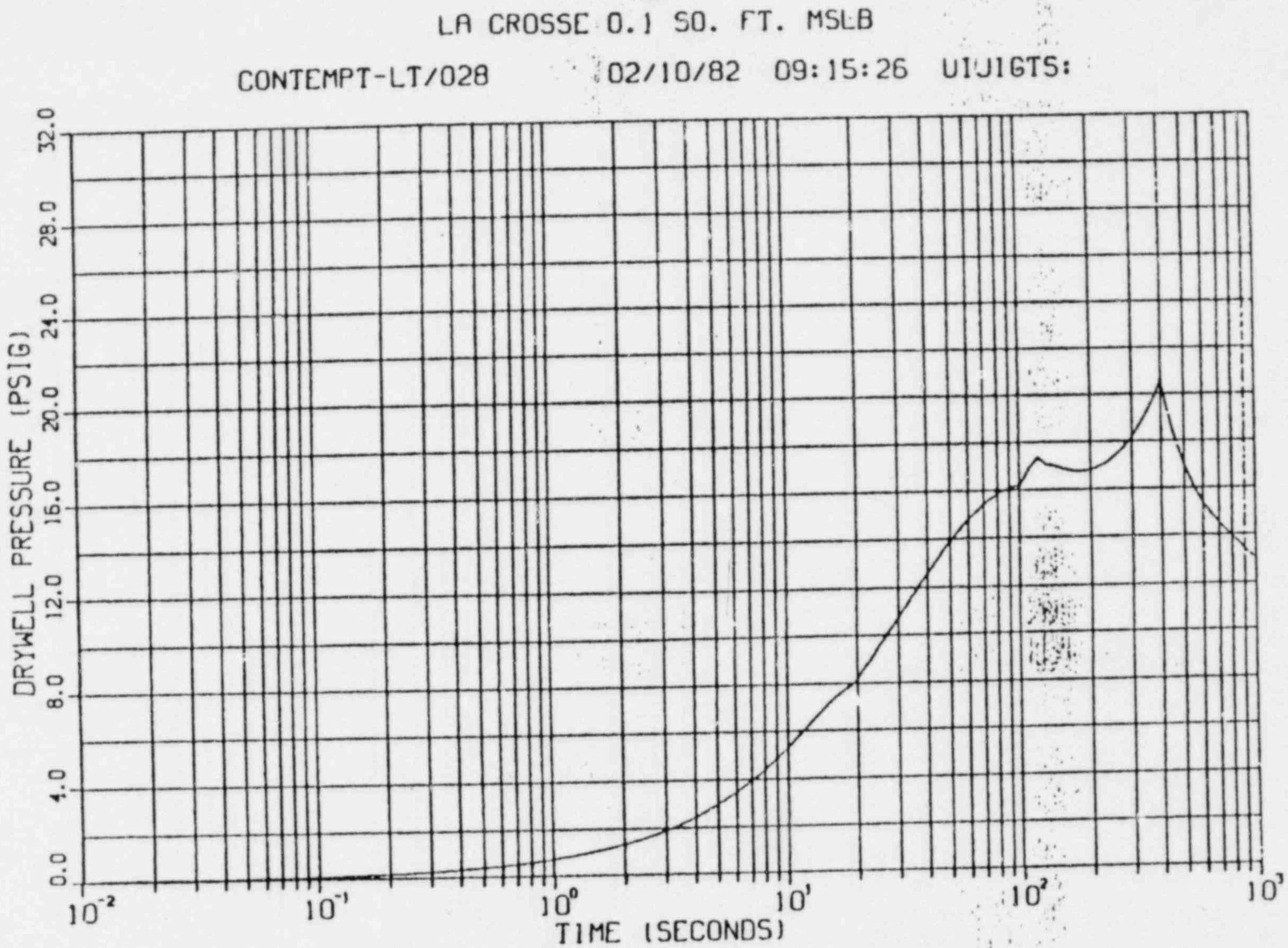

Figure 5. Containment Pressure Response to a 0.10 sq. ft. Main Steam Line Break

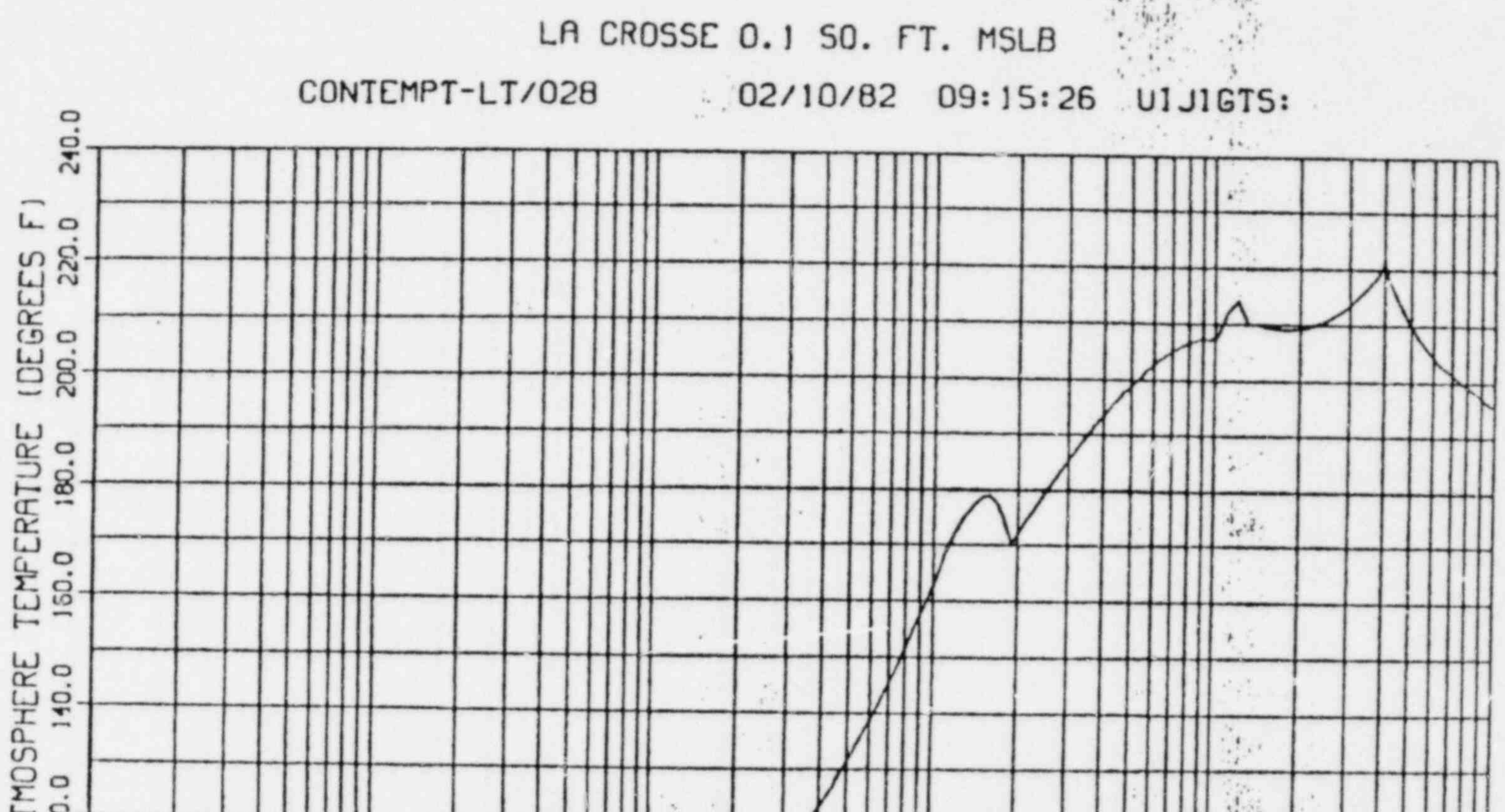

Figure 6. Containment Temperature Response to a 0.10 sq. ft. Main Steam Line Break

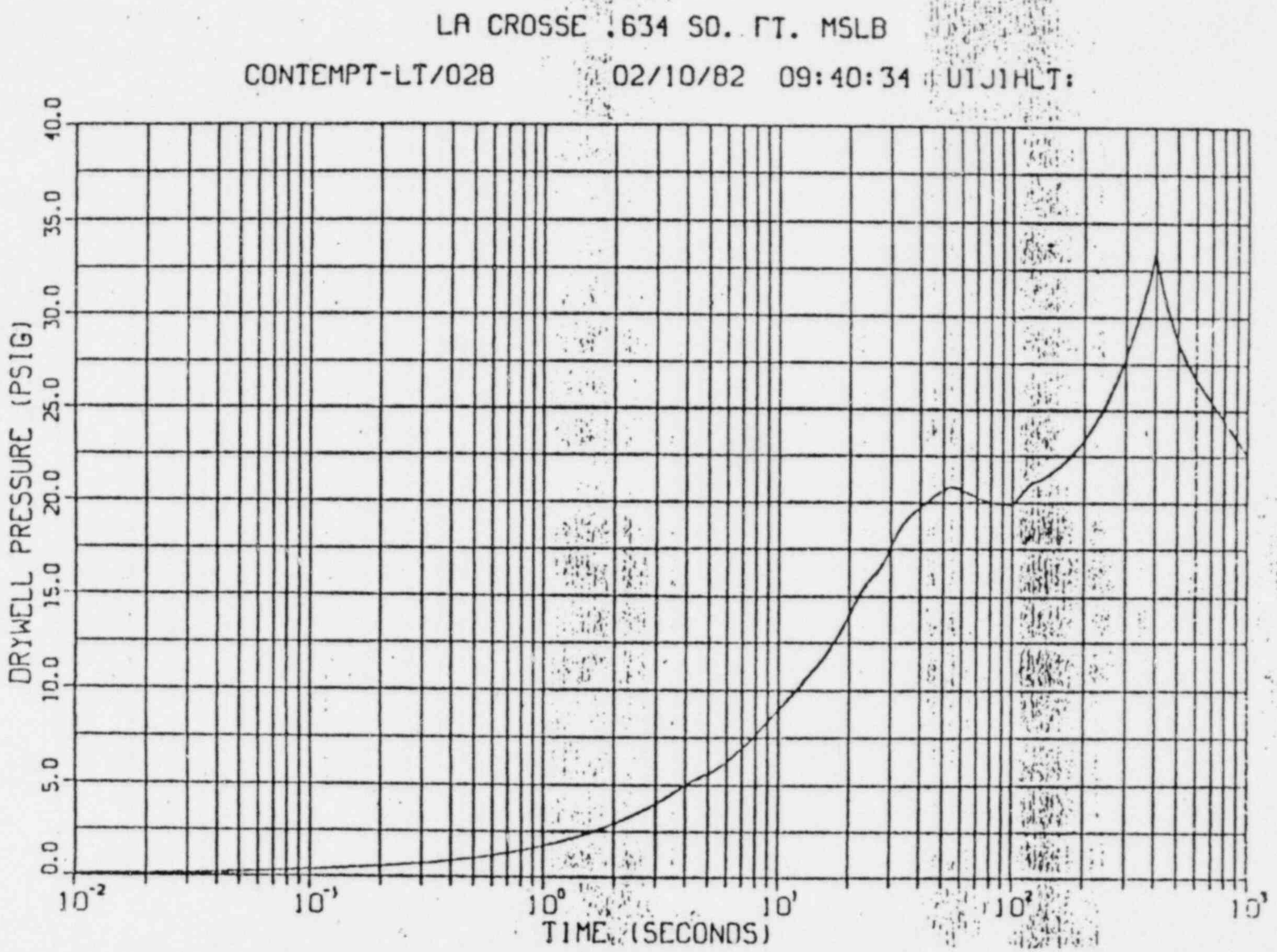

Figure 7. Containment Pressure Response to a
0.634 sq. ft. Main Steam Line Break

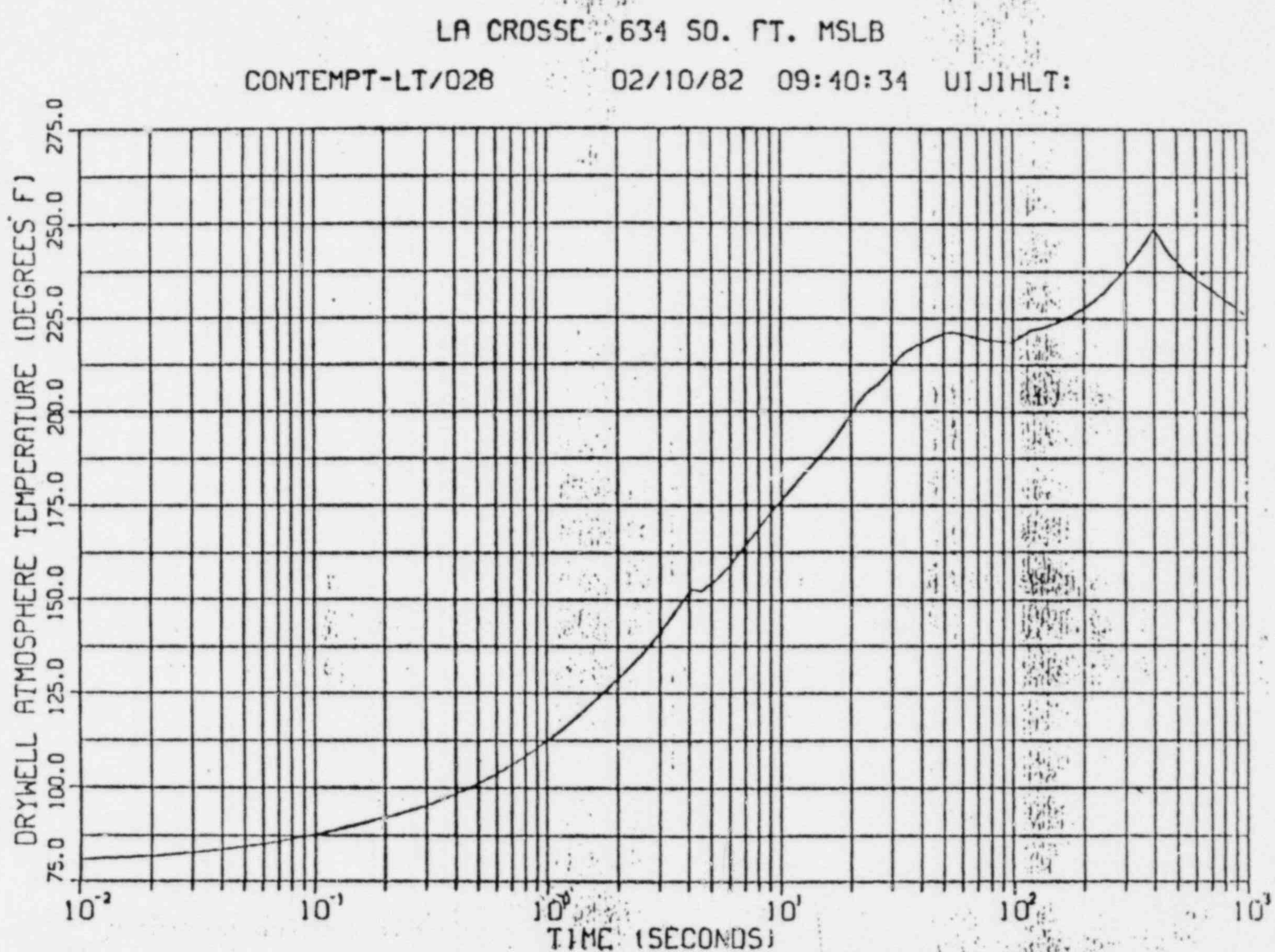

Figure 8. Containment Temperature Response to a 0.634 sq. ft. Main Steam Line Break

DIVISION OF LICENSING

CORRIGENDA FOR APPENDIX A

LAWRENCE LIVERMORE NATIONAL LABORATORY

TECHNICAL EVALUATION REPORT

FOR SEP TOPICS VI-2.D AND VI-3

Page 1 Paragraph 3 Sentence 1	LaCrosse is a 165 MWt Allis-Chalmers Boiling Water Reactor (BWR).
Page 5 Item 4 Sentence	Scram would normally occur in approximately one second due to the MISV closure on a low steam line pressure or a low reactor water.
Page 9 Paragraph 2 Sentence 4	This is the amount of time that exists for the reactor operator to take appropriate action such as to manually actuate the containment sprays or the manual depressurization system (MDS).
Page 11 Footnote	*Assume steam decay heat curve at 30.0 sec.

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