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 AUTH.NAME   AUTHOR AFFILIATION
CRUTCHFIELD,D.   Operating Reactors Branch 5
 RECIP.NAME   RECIPIENT AFFILIATION
MAIER,J.E.   Rochester Gas & Electric Corp.

SUBJECT: Forwards draft safety evaluation of SEP Topics VI-2.D, "Mass
         & Energy Release for Possible Pipe Break Inside
         Containment," & VI-3, "Containment Pressure & Heat Removal
         Capability" & LLL draft technical evaluation.
                  SEE Repts.
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Docket No. 50-244
LS05-81- 11-004

Mr. John E. Maier, Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: SYSTEMATIC EVALUATION PROGRAM (SEP) FOR THE R. G. GINNA
           NUCLEAR POWER PLANT - EVALUATION REPORT ON TOPICS VI-2.D
           AND VI-3

Enclosed is a copy of our draft 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. 50-244, with the
criteria currently used by the regulatory staff for licensing new faci-
lities. Appendix A to our draft evaluation is a draft Technical Evaluation
Report from our contractor, Lawrence Livermore National Laboratory.
Please inform us if your as-built facility differs from the licensing
basis assumed in our assessment. Comments are requested within 30 days
of the receipt of this letter so that they may be considered in our final
evaluation.

This evaluation will be a basic input to the integrated safety assessment
for your facility unless you identify changes needed to reflect the as-
built conditions at 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
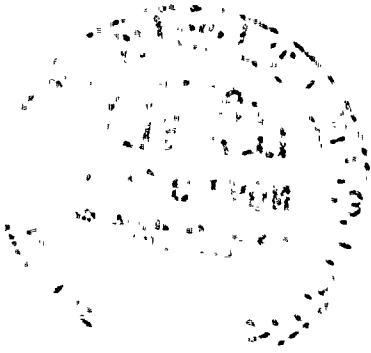Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
Draft SEP Topics VI-2.D and VI-3

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Docket No. 50-244
LS05-81-11-004

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Electric and Steam Production
Rochester Gas & Electric Corporation
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Rochester, New York 14649

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

Enclosure:
Draft SEP Topics VI-2.D and VI-3

cc w/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-2.0

FOR THE

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

TABLE OF CONTENTS

I. Introduction

The R. E. Ginna Nuclear Power Plant began commercial operations in 1970. Since then the staff's safety review criteria have changed. As part of the Systematic Evaluation Program (SEP), the containment pressure and heat removal capability (Topic VI-3) and the mass and energy release for possible pipe break inside containment (Topic VI-2.D) have been re-evaluated.

The purpose of this evaluation is to document the deviations from current safety criteria as they relate to the containment pressure and heat removal capability and the mass/energy release for possible pipe break inside containment. Furthermore, independent analyses in accordance with current criteria were performed to determine the adequacy of the containment design basis (e.g., design pressure and temperature) and to provide input for Unresolved Safety Issue (USI) A-24, Qualification of Class 1E Safety Related Equipment. The significance of the identified deviations, and recommended corrective measures to improve safety, will be the subject of a subsequent, integrated assessment of the Ginna Plant.

II. Review Criteria

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

(1) 10 CFR Part 50, Appendixc A, General Design Criteria for Nuclear Powr 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 systems for Light Water Nuclear Power Reactors."

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

(4) NUREG 75/087, 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) III-7B, Design Codes, Design Criteria, Load Combinations, and Reactor
     Cavity Design Criteria

(3) VI-7.B, ESF Switchover 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 an essentially leak-tight barrier against the uncontrolled release
of radioactivity to the environment and to assure that the containment design
conditions important to safety are not exceeded for as long as the postulated
accident conditions require. GDC 38 requires a containment heat removal

system be provided whose system safety function shall be to reduce the containment pressure and temperature following any loss-of-coolant accident (LOCA) and maintain them at acceptably low levels; furthermore, the system safety function shall be achievable 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 as obtained from the conservative calculation of mass/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 done 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. The mass and energy release to the containment from a LOCA should be considered in terms of blowdown, reflood, and post-reflood. The mass and energy release as a result of postulated secondary system pipe ruptures should be calculated in accordance with SRP 6.2.1.4. Our review also included the analysis of postulated single active failures of components in the secondary system.

In reviewing the licensee's analysis, deviations from current criteria have been identified. Independent analyses, as required, were performed, to evaluate the significance of these deviations. The evaluation was completed by comparing the results with the licensee's containment design basis.

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V. Evaluation

A review of the existing containment analysis for Ginna, as described in the Final Safety Analysis Report (FSAR), indicated two basic deviations from the current safety criteria. First, for the LOCA analysis, the licensee had not considered a cold leg pump suction break location, the core reflood phase of mass and energy release, or the release of secondary system energy to the containment. These aspects of the LOCA analysis are addressed in Standard Review Plan (SRP) 6.2.1.3. Second, the main steam line break (MSLB) analysis had not been performed for the Ginna Plant. SRP Sections 6.2.1.1.A and 6.2.1.4 address the MSLB analysis.

To assess the significance of these two deviations, our consultant, the Lawrence Livermore National Laboratory (LLNL) performed independent LOCA and MSLB analyses which are presented in Appendix A to this report. Mass and energy release rates utilized in the analysis were calculated using RELAP-4 MOD 7, and the calculation of the containment pressure and temperature response was done using CONTEMPT-LT/-28.

For the primary system (LOCA analysis), a double-ended break at the pump suction of the cold leg was analyzed in Appendix A since it typically is the design basis LOCA for a PWR plant. Blowdown, reflood and post-reflood phases were considered in the calculation of mass and energy release data; the release of the secondary system energy was also factored into the calculation. The calculated transient results show a peak containment pressure of 74 psia, and a peak containment atmosphere temperature of 282°F. The containment design pressure and temperature for Ginna are 75 psia and 286°F. There is, therefore, a slight margin between the calculated containment pressure and temperature and the design values.

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Based on our review of the LLNL LOCA analysis, we concur with their findings and conclude that the Ginna containment design basis is adequate for postulated LOCAs.

For the secondary system, the worst case MSLB accident identified in Appendix A is that postulated to occur during hot standby concurrent with the failure of one spray line. The peak calculated containment pressure is 85.8 psia which exceeds the containment design pressure by 11 psi. The worst peak calculated containment temperature is 421°F occurring at 102% of full power with the single failure of one spray line.

It is acknowledged that the MSLB accident analysis was done in a conservative manner. This was prompted by the unavailability of a water entrainment model to more explicitly define the mass and energy release during the blowdown of a steam generator. An appropriate water entrainment model is necessary to determine the maximum break size that results in a pure steam blowdown, which is smaller than a double-ended break. Without the water entrainment model, it was necessary to assume a double-ended steam line break with pure steam blowdown. Also, heat transfer from the primary system to the steam generator secondary side was treated conservatively. Steam addition from the unaffected steam generator and feedwater addition to the affected steam generator were treated in a manner that was consistent with the performance of system isolation valves or single active failure assumptions.

The hot standby condition was assumed for the analysis since the steam generator would have the largest inventory of water. Calculations were also performed

assuming the reactor was at 102% of full power, a condition of low steam generator inventory, to bracket the results of the analysis.

VI. Conclusions

Our review of the containment functional design of the Ginna plant, as reported in Appendix A, identified deviations from current safety criteria. To assess the significance of these deviations independent containment analyses were performed. The results of the analyses show that the containment design conditions are not exceeded for postulated LOCAs, but are exceeded for the MSLB accident. For the MSLB analysis, the peak calculated pressure exceeded the containment design pressure by 11 psi; i.e., 85.8 psia. It should be noted, however, that a more refined MSLB analysis may show that the containment design pressure would not be exceeded. Moreover, the Structural Integrity Test for the Ginna containment was performed at a pressure 1.15 times the containment design pressure, or 69 psig (84 psia). Therefore, as conservative as the MSLB analysis is, the peak calculated containment pressure is very close to the test pressure for the Structural Integrity Test. The implications of exceeding the containment design pressure are, therefore, not of great concern. Therefore, the need for the licensee to upgrade the Ginna containment analysis (for both loss of coolant and MSLB accidents) to reflect the application of current NSSS vendor analytical capabilities to the Ginna plant for future reference in licensing actions, will be deferred to the outcome of the integrated assessment of the Ginna plant.

The results of LLNL analyses are extremely conservative, especially for equipment qualification purposes. However, if the licensee chooses to use the results of this report then the MSLB temperature profile in Figure 10 of Appendix A and the LOCA temperature profile in Figure 2 of Appendix A may be used to assess the environmental qualification of Class IE safety-related electrical equipment (USI A-24). Alternatively, the licensee may choose to perform a more realistic analysis to establish environmental conditions for postulated LOCA and steam line breaks.
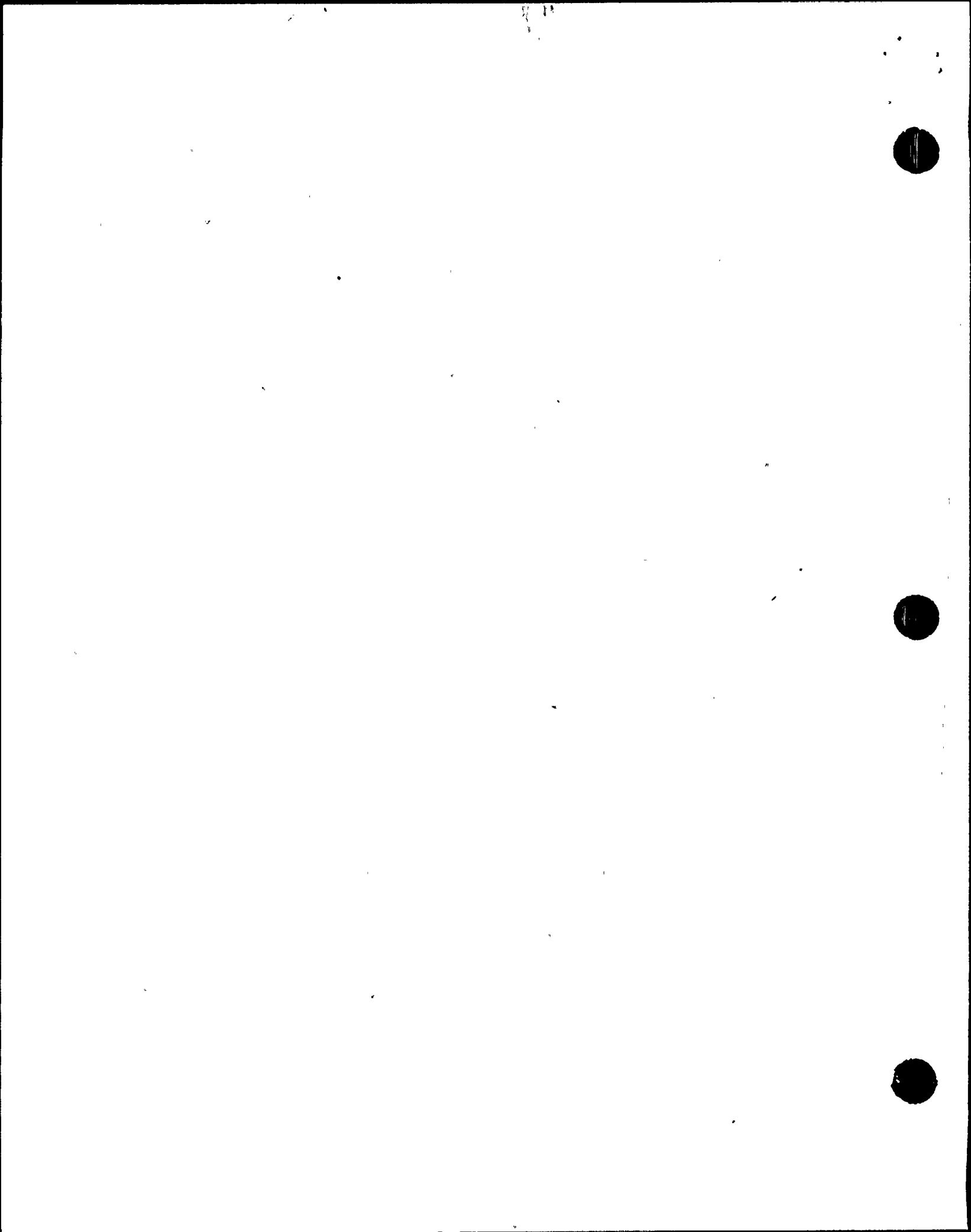
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Appendix A

SEP Containment Analysis and Evaluation

for the R. E. Ginna 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 R. E. Ginna Nuclear Power Plant 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 SEP 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 R. E. Ginna 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 loss-of-coolant (LOCA) and steam-line break accidents. Furthermore, equipment with a post-accident safety function must be environmentally qualified for the resulting adverse pressure and temperature conditions.

2.0 CONTAINMENT FUNCTIONAL DESIGN

Ginna is a 1520-MWt Westinghouse pressurized water reactor (PWR) which uses a dry cylindrical reinforced concrete type containment. It is very similar to the San Onofre Unit 1 power plant also designed by Westinghouse. The reactor coolant system of Ginna consists of 2 loops, compared with 3 loops' for San Onofre 1.

The engineered safety sytems provided include the containment air recirculation system, containment spray system, and safety injection system. The safety injection system consists of two passive accumulators, three

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high-pressure pumps, and two low-pressure pumps. In the event of loss of off-site power and failure of one diesel generator, minimum safety injection is provided by two high-pressure pumps and one low-pressure pump, and minimum containment heat removal is provided by one containment spray pump and two fan coolers.

2.1 Review of Analysis of Ginna Containment Functional Design

For PWR plants the high-energy line break types that must be analyzed include primary system pipe breaks and secondary system pipe breaks. A break on the primary side generally results in the most severe pressure response in the containment, while a break on the secondary side results in the most severe temperature conditions in the containment.

There are two separate calculations which comprise the containment analysis for a postulated pipe break. The first calculation includes the mass and energy release analysis which, for primary system pipe breaks (LOCAs), includes blowdown, reflood and post-reflood phases. The results are mass and energy release rates into the containment. The second calculation is the containment response analysis, which results in the containment temperature and pressure response to the mass and energy release from the postulated break.

The acceptance criteria used to evaluate the Ginna containment functional design analysis are based on the Standard Review Plan (SRP), NUREG-75/087. In order for the containment 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 Primary System Pipe Break

In the Ginna FSAR,[1] the most severe primary system pipe break was identified as a double-ended cold-leg discharge break. For the postulated

break, the reflood phase, and hence the energy in the secondary system, was not included in the analysis. This analysis, therefore, does not meet the acceptance criteria specified in the SRP. Since the analysis of mass and energy release rates is unacceptable, so too is the containment response calculation based on the mass and energy release rate data.

2.3 Secondary System Pipe Break

In the Ginna FSAR,[1] the licensee's secondary system pipe-break analysis consisted of analyzing the reactor response to a steam-line break occurring at various locations inside and outside the containment. The analysis was performed to demonstrate that:

(a) with a stuck rod and minimum engineered safety features, the core remains in place and essentially intact so as not to impair effective cooling of the core; and

(b) with no stuck rod and all equipment operating at design capacity, insignificant cladding rupture occurs.

This analysis was not intended to be used to evaluate the containment functional design calculation, and the results would not be appropriate for that purpose. Therefore, an acceptable secondary system pipe-break analysis has not been performed.

2.4 Reanalysis of Ginna Containment Functional Design

As mentioned above in Section 2.2, Review of Analysis of Ginna Containment Functional Design, the containment response analysis for primary system pipe-breaks (LOCA analysis) does not satisfy current criteria, and a MSLB analysis suitable for evaluating the containment functional design has not been performed. The secondary system pipe-break (MSLB) analysis generally is

the most limiting case for temperature conditions inside the containment. The primary system pipe-break (LOCA) analysis generally results in the limiting peak pressure condition inside the containment. Both of these analyses were performed and are discussed below.

3.0 PRIMARY SYSTEM PIPE BREAK ANALYSIS

For a primary system pipe break, three phases are involved in the calculation of mass and energy release rates, namely the blowdown, reflood, and post-reflood phases. The mass and energy release rate calculations were based on the guidelines of Standard Review Plan Section 6.2.1.3; in the calculations the carryout rate fraction during reflood was set equal to 0.80 at the bottom of the core. In general, the analysis was done in a manner that conservatively establishes the containment design pressure; i.e., maximizes the post-accident containment pressure. The worst break location was determined to be at the pump suction side of the cold leg, because of the consideration of energy input from the steam generator in the affected loop during the reflood phase.

3.1 Initial and Boundary Conditions

The initial mass of water in the reactor coolant system was based on the system volume calculated for the temperature and pressure conditions existing at 102% of full power (safeguards design rating) or 1550.4 MWt. The initial conditions within the containment and the reactor coolant system prior to accident initiation are given in Table 1.

For the containment peak pressure analysis, a double-ended guillotine break at the pump suction with loss of off-site power, was postulated. In addition, the loss of one diesel generator was assumed as the worst single active failure. This assumption of postulated break and single active failure

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typically results in the maximum calculated containment internal pressure.

The components of the available safety injection and containment heat removal systems, if off-site power and one diesel generator are lost, are shown in Table 2. The containment heat sink data used in the analysis are described in Table 3.

3.2 Blowdown Phase

Following a postulated rupture of the Reactor Coolant System (RCS), steam and water are released into the containment. Initially, the water in the RCS is sub-cooled at a high pressure. When the break occurs, the water passes through the break where a portion flashes to steam at the low pressure in the containment. Break flow rates are calculated with the Moody critical flow model for saturated flow and the Henry-Fauske model for sub-cooled flow. A discharge coefficient of 1.0 was used.

Reactor scram was assumed to occur with loss of off-site power, at the initiation of the break. The recirculation pumps were tripped off and the steam generators were isolated at the time of the break. The containment back-pressure was conservatively assumed to be constant throughout the accident at 14.7 psia. The end of blowdown was defined as the time when the primary system pressure dropped below the containment design pressure of 74.7 psia. Natural convection heat transfer was used for the secondary coolant in the steam generator for tube surfaces immersed in water.

The mass and energy release rate was calculated with the code RELAP4-MOD7. The RELAP4 input deck was obtained from the NRC and carefully reviewed for code options and for initial and boundary conditions. The plant physical description was assumed to be correct. Additional information required to perform the analysis was obtained from information on the Ginna docket and conversations with personnel of Rochester Gas and Electric Corp.

The results of the blowdown analysis are summarized in Tables 4 and 5. Table 4 provides a detailed energy balance prior to the accident and at the end of the blowdown phase, which occurred 14.7 seconds after break initiation. The total energy released to the containment during blowdown was approximately 211.9 million Btu.

Table 5 provides mass and energy release rates from the blowdown phase for use in the containment response analysis.

3.3 Reflood Phase

Following blowdown, the lower plenum below the reactor core is refilled by water from the safety injection system. This phase, known as refill, was conservatively omitted and reflood was assumed to begin immediately after blowdown. Initial conditions for the start of the reflood phase were based on the end-of-blowdown (EOB) results. At the start of reflood, 14.7 seconds after break initiation, the water remaining in the reactor vessel was assumed to be saturated at the design pressure of 74.7 psia and at the level of the bottom of the active core.

At 14.7 seconds, the core power level dropped to 100.11 MWt or approximately 6% of the initial power. The accumulator flows had been initiated on low cold-leg pressure of 700 psia, which occurred at about 7 seconds into blowdown. At the start of reflood the accumulator flows totaled 4550 lbm/s. For numerical stability of the RELAP4 computer code, the Emergency Core Cooling System (ECCS) flow was set at the saturation temperature of 272.9°F. The reactor coolant pumps had coasted down and the rotors were locked.

In the reflood phase, Safety Injection (SI) water enters into the downcomer. As the downcomer is filled, a driving head across the vessel

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forces water into the core. SI water entering the core is converted to steam, which entrains water into the hot legs at a high velocity. Water continues to enter the core and releases the stored energy of the fuel and cladding as the mixture level in the core increases. The carryout rate fraction (CRF), which is the mass ratio of liquid exiting the core to liquid entering the core, is assumed to be at a constant value of 0.80 throughout the reflood phase. The core is assumed to be quenched when the liquid level is 2 feet from the top of the core.

The flow split between the broken and unbroken loop and any steam quenching was calculated by RELAP4-MOD7 using the homogeneous equilibrium model. The heat transfer from the secondary coolant to the steam generator tubes was based on natural convection heat transfer for tube surfaces immersed in water. For tubes not immersed in water, condensing heat transfer is assumed. Steam leaving the steam generator was conservatively assumed to be superheated to the temperature of the secondary coolant.

The results of the reflood analysis are summarized in Tables 6 and 7. Table 6 provides a detailed energy balance at the end of blowdown just prior to reflood and at the end of the reflood phase, which ocurred 20.1 seconds after the start of reflood. Table 7 provides mass and energy release rates from the reflood phase needed for input into the containment response analysis.

3.4 Post-Reflood Phase

The post-reflood phase consists of removing all remaining stored energy in the primary and secondary systems and accounting for decay heat. This is done by conservatively assuming that all the energy in the secondary system and primary heat structures is released in one hour after the end of reflood. The amount of energy in the secondary system and primary heat structures was

calculated by assuming these structures would return to 212°F in one hour following reflood. This is conservative since the containment pressure will not return to 14.7 psia within one hour, and therefore the saturation temperature will be hotter than 212°F. The decay heat released over the one hour duration was based on the ANS standard decay heat curve plus 20%.

3.5 Containment Response Calculation

The containment spray systems and containment structures available for energy removal were mentioned earlier in Section 3.1, Initial and Boundary Conditions; they are given in Tables 1 and 2. The Tagami and Uchida heat transfer correlations were used for all structural heat sinks. The Tagami correlation was used until the end of blowdown or 14.7 seconds; thereafter the Uchida correlation was used.

The containment response calculation was done using the CONTEMPT-LT/028 computer code. The program uses a three-region containment model consisting of the containment atmosphere (vapor region), the sump (liquid region), and the water in the reactor vessel. Mass and energy are transferred between the liquid and vapor regions by boiling, condensation, or liquid dropout. Each region is homogeneous, but a temperature difference can exist between regions. The physical model was obtained from references 1, 2 and 3.

3.6 Containment Response Results

The containment pressure and temperature response was calculated by assuming that the blowdown, reflood, and post-reflood energy is released directly to the containment. This method is conservative since it does not take into account the energy that may be required to heat the water in the primary system to saturation. In addition, it was also assumed that the

reflood and post-reflood energy were released as superheated steam at the temperature of the secondary side (approximately 500°F). The results are shown in Figures 1 and 2. The calculated transient reflects a post-accident containment pressure of 74 psia and a temperature of 282°F. The containment design pressure and temperature are 74 psia and 286°F, respectively. There is, therefore, a slight margin between the peak calculated pressure and temperature and the design values.

4.0 SECONDARY SYSTEM PIPE BREAK ANALYSIS

The containment response to a secondary system pipe break was also analyzed. For PWRs, the most limiting break is a main steam-line break with pure steam blowdown. The steam-line break accident was analyzed for various plant conditions from hot standby to 102% of full power. A detailed parametric study is required to determine the most limiting combination of consistent initial conditions and system operation modes. To circumvent an extensive parametric study, the most limiting set of conditions was considered.

The postulated accidents analyzed were a double-ended guillotine break in a main steam line at 102% of full power and the same break at hot standby.[1] In both of these cases, the mass and energy release rates were calculated assuming that off-site power was available. Since no liquid entrainment was assumed during steam generator blowdown, a spectrum of break sizes was not analyzed. In addition, three different single active failures were considered for the 102% of full power case, and two different single active failures were considered for the hot standby case. These were a main steam isolation valve (MSIV) and main feed isolation valve (MFIV) failure and loss

of one train of containment heat removal systems for the 102% of full power case and MSIV failure and loss of one train of containment heat-removal system for the hot standby case. Thus, five different containment response calculations were performed to determine the most limiting pressure and temperature conditions resulting from a steam-line break, (see Table 8). The model and assumptions that were used in analyzing the main steam-line break are given in the following discussion.

4.1 Analytical Model

The blowdown mass and energy release rates were calculated using a four-volume RELAP4 model. One volume models the primary side of the affected steam generator and the other three volumes model the feedwater line, secondary side of the steam generator, and the steam line. A schematic of the four-volume model is shown in Figures 3 and 4. A description of the four-volume model follows.

Steam Generator - On the primary side of the steam generator, steady state flow conditions are conservatively assumed throughout the blowdown for both the 102% of full power and hot standby cases. On the secondary side of the steam generator, the actual plant conditions representing 102% of full power and hot standby are used in each case. An infinite bubble rise velocity was assumed on the secondary side, which precludes moisture carryover and ensures a pure steam blowdown. One heat slab was used in the steam generator model to model the heat transfer between the primary and secondary sides. The heat transfer coefficient on the primary side was calculated by RELAP4; forced convection was assumed. On the secondary side, nucleate boiling heat transfer was assumed. The height of the heat slab used to model the steam-generator-tube surface area was set to a small value to ensure that the tubes remained covered during the entire transient.

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Steam Line and Feedwater Line - The blowdown of the steam line and feedwater line was accounted for by a one-volume RELAP4 model for each line. The size of each volume was adjusted to account for the mass of steam or water in the line up to their respective isolation valves. Both lines have redundant isolation valves. The specific isolation valve considered depends on the single-failure assumption being used in the analysis. The blowdown of the unaffected steam generator through the connecting steam header before isolation valves close was conservatively modeled by assuming a constant back-pressure fill for this line. Feedwater flow before isolation valve closure was modeled as a constant mass flow rate fill. The main feedwater isolation valves are assumed to start closing 10.54 seconds after a steam-line break; these valves require 5 seconds to fully close. The main steam isolation valves are assumed to start closing at zero seconds after a steam-line break and require 5 seconds to fully close.

Auxiliary Feedwater Injection - The auxiliary feedwater injection was assumed to be 200 gpm at 80°F for each case calculated.[3] For breaks at 102% of full power, injection is assumed to start 30 seconds after the line-break occurs. At hot standby conditions, injection is assumed to start at the time of the break.

4.2 Initial Conditions and Other Assumptions

The initial conditions for the three cases analyzed at 102% of full power and hot standby are summarized in Figures 2 and 3. In all cases the sources of energy include the following:

The stored energy in the affected steam-generator vessel tubing.

The stored energy in the water contained within the affected steam generator.

The stored energy in the feedwater transferred to the affected steam generator before the isolation valves in the feedwater line close.

The stored energy in the steam from the unaffected steam generator before the isolation valves in the unaffected steam generator close.

The energy transferred from the primary coolant to the water in the affected steam generator during blowdown.

The stored energy in the auxiliary feedwater transferred to the affected steam generator after auxiliary feedwater system initiation.

In addition, the mass release rate was calculated with the Moody model..

4.3 Containment Response Calculation

The containment for secondary system line breaks was modeled in a similar manner as for primary system blowdown as described in Section 3.5 with initial conditions as in Table 1. One exception is that the Tagami heat transfer correlation was used with a peak time of 100 seconds for all cases analyzed. The containment engineered safety systems are described in Table 2. For cases 1, 2, and 4, (see Table 8) full capacity of the systems is assumed. For cases 3 and 5, (see Table 8) loss of one containment spray line is assumed and all the four fan coolers are assumed to remain in full capacity. In each case, the containment sprays are initiated at a containment pressure of 30 psig and require 35 seconds to come on line.

4.4 Steam Generator Blowdown and Containment Response Results

Three different cases at 102% of full power and two cases at hot standby conditions were analyzed, as described in Table 8. The blowdown mass and energy release rates for the five cases are tabulated in Tables 9 - 13. The

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resultant containment pressure and temperature response is given in
Figures 5 - 14. As shown by Figures 13 and 14, case 5 results in the highest
containment pressure, 85.8 psia at 91 seconds after steam-line break, with a
containment temperature of $413^{\circ}F$ at 34 seconds after steam-line break. Case
5 represents a hot standby plant configuration failure of one containment
spray pump. As shown by Figure 10, case 3 results in the highest containment
temperature, $421^{\circ}F$ at 34 seconds after steam line break, with a correspond-
ing containment peak pressure of 75 psia at 60 seconds after steam line break.
Case 3 represents a 102% of full power plant configuration with failure of one
containment spray pump.

The containment design conditions are 74.7 psia and $286^{\circ}F$; thus, both
values are exceeded as a result of a main steam-line break.

5.0 REFERENCES

1. Ginna Nuclear Power Plant Unit 1, "Updated Final Facility Description and
   Safety Analysis Report," Docket No's. 50-244-A1 to 50-244-A4.

2. Exxon Nuclear Company, Inc., ECCS Analysis for the R. E. Ginna Reactor
   with ENC WREM-2 PWR Evaluation Model," XN-NF-77-58 dated December 1977.

3. Memo from S. Brown to C. Tinkler, "Ginna Containment Analysis Data," dated
   20 September 1979.

Table 1. Initial conditions for containment analysis[2,3]

Parameter	Value
Reactor coolant system	
Reactor power level (a)	1550.4 MWt
Mass of RCS	392×10^3 lbm
Total Liquid Energy(b)	386 MBtu
Containment	
Net free volume	972000 ft^3
Pressure	14.7 psia
Temperature	$100^\circ F$
Relative humidity	50%
Refueling water temperature	$80^\circ F$
Outside air temperature	$100^\circ F$
Refueling water storage tank	230,000 gal.

a. 102% of full power
b. all energies are relative to $32^\circ F$

Table 2. Engineered safety systems
Operating Assumptions for Containment Peak Pressure Analysis[1,2]

System/Item	Full Capacity	Value Used for Peak Pressure Analyses
Safety injection system		
Number of trains	2	
Number of Injection lines	3	
Number of pumps		
High-pressure pumps	3	2
Low-pressure pumps	2	1
Flowrate, gal/min/train	4020	2160
Containment spray system		
Number of lines	1	1
Number of refueling water pumps	2	1
Flowrate, gal/min	2400	1200
Recirculation system		
Number of lines	1	1
Number of refueling water pumps	2	1
Number of heat exchangers	2	2
Type	Shell & U-Tube	Shell & U-Tube
Design UA Btu/hr$^\circ$F	750,000	750,000
Flowrates		
Recirculation side, gal/min	3120	1560
Exterior side, gal/min	5560	2780
Source of cooling water	Component cooling water	Component cooling water

Table 3. Containment Structural Heat Sinks.[2]

A. Material Properties

Material	Thermal Conductivity (Btu/hr ft °F)	Volumetric Heat Capacity (Btu/ft^3 °F)
Steel	30.0	54.0
Concrete	0.8	30.0
Insulation	0.02	1.0

B. Heat Sink Descriptions

1. Insulated dome and wall

Surface Area, ft^2	36,181
Composition, ft	
Steel	0.03125
Concrete	2.5
Insulation	0.10417

2. Uninsulated dome and wall

Surface Area, ft^2	12,474
Composition, ft	
Steel	0.0315
Concrete	2.5

3. Sump walls

Surface Area, ft^2	2,342
Composition, ft	
Steel	0.03125
Concrete	5.0

4. Refueling cavity inside wall and floor

Surface Area, ft^2	6,900
Composition, ft	
Steel	0.02083
Concrete	2.5

Table 3.   Containment Structural Heat Sinks (cont'd)

5.   Outside refueling cavity wall and steam generator compartment

Surface Area, ft^2	21,800
Composition, ft	
Concrete	1.25

6.   Intermediate level floor

Surface Area, ft^2	6,170
Composition, ft	
Concrete	0.25

7.   Operating floor

Surface Area, ft^2	9,162
Composition, ft	
Concrete	1.0

8.   Heavy steel beam and crane structure

Surface Area, ft^2	9,174
Composition, ft	
Steel	0.0625

9.   Steel beam

Surface Area, ft^2	5,016
Composition, ft	
Steel	0.04167

10.   Cylindrical supports and beam

Surface Area, ft^2	8,586
Composition, ft	
Steel	0.02088

11.   Crane support columns

Surface Area, ft^2	5,756
Composition, ft	
Steel	0.03125

12.   Grating and stairs

Surface Area, ft^2	7,000
Composition, ft	
Steel	0.0052

Note: Boundary conditions on all heat slabs are adiabatic on the inside and Tagami/Uchida on the outside..

Table 4. Blowdown Energy Balance,

Ginna Double-Ended Guillotine

Suction Leg Break

	Inventory @ 0.0 s		Inventory @ 14.7 s		Decrease	
	10^3 lbm	10^6 Btu	10^3 lbm	10^6 Btu	10^3 lbm	10^6 Btu
Reactor coolant system	312.4	211.9	136.9	5.87	255.1	206.00
Accumulator system[a]	13.67	7.93	10.84	6.28	2.83	0.63
Core stored energy[b]	--	10.76	--	5.55	--	2.38
Primary sensible energy	--	85.64	--	83.1	--	2.53
Decay heat	--	--	--	--	--	4.768

[a] The SI water temperature was 272.9°F to prevent numerical instabilities. Actual value should be 90 F.

[b] Based on ANS + 20% decay heat curve.

Table 5.   Blowdown Mass and Energy Release Rates

Ginna Double-Ended Guillotine
Suction Leg Break

Time (s)	Mass (lbm/s)	Energy (Btu/lbm)
0.0	8.05×10^4	535.8
1.0	4.57×10^4	540.8
2.0	3.77×10^4	557.7
3.0	2.81×10^4	576.3
4.0	1.93×10^4	637.5
5.0	1.87×10^4	601.6
6.0	1.62×10^4	610.9
7.0	1.42×10^4	601.3
8.0	1.26×10^4	606.9
9.0	9.82×10^3	603.1
10.0	7.88×10^3	566.4
11.0	6.25×10^3	558.5
12.0	4.07×10^4	530.7
13.0	2.81×10^3	512.4
14.0	2.21×10^3	476.6
14.7	1.99×10^3	458.0

Table 6. Reflood Energy Balance. Ginna Double-Ended Guillotine Suction Leg Break.

	Inventory @ 14.7 s		Inventory @ 40 s		Decrease	
	10^3 lbm	10^6 Btu	10^3 lbm	10^6 Btu	10^3 lbm	10^6 Btu
Reactor coolant system	43.9	30.3	39.7	28.6	4.2	1.7
Accumulator system	10.84	6.28	0.0	0.0	10.84	6.28
Core stored energy	--	18.8	--	6.17	--	12.63
Decay heat	--	--	--	--	--	1.96
Steam generator (secondary side)	17.8	97.7	17.8	24.6	--	73.1

Table 7. Reflood mass and energy release rate. Ginna Double-Ended
Guillotine Suction Leg Break.

Time (s)	Mass (lbm/s)	Energy (Btu/lbm)
14.7	2.11×10^3	1282
15.0	2.98×10^3	1282
16.0	3.15×10^3	1282
17.0	3.60×10^3	1282
18.0	1.87×10^3	1282
19.0	9.39×10^2	1282
20.0	5.69×10^2	1282
21.0	1.21×10^2	1282
22.0	1.06×10^2	1282
23.0	9.82×10^1	1282
24.0	8.99×10^1	1282
25.0	7.58×10^1	1282
26.0	7.21×10^1	1282
27.0	6.89×10^1	1282
28.0	5.63×10^1	1282
29.0	5.40×10^1	1282
30.0	4.86×10^1	1282
31.0	1.08×10^1	1282
7200.	1.08×10^1	1282
7201.	0.0	1282
1.0×10^5	0.0	1282

Table 8 Description of steam-line break cases[3]

Power level		102% of pull power			Hot standby
RCS primary flow		982 LBM/s			962 LBM/s
Steam generator					
Feed flow		869 LBM/s			0.0 LBM/s
Mixture level		24.8 ft			41.8 ft
Water volume		1681 ft^3			2821 ft^3
Pressure		779 psia			1020 psia

	Case 1	Case 2	Case 3	Case 4	Case 5
Failure Of	MSIV	MFIV	1 spray line	MSIV	1 spray line
Steam-line					
Mass (LBM)	2,462	2,412	2,412	2,462	2,412
Feedwater line					
Mass (LBM)	18,300	76,895	18,300	18,300	18,300
Temp. (^{O}F)	432	432	432	432	432
Auxiliary feedwater					
Flow (GPM)	200 @ 30 s	200 @ 30 s	200 @ 30 s	200 @ 0 s	200 @ 0 s
Temp. (^{O}F)	80	80	80	80	80
Containment heat removal system	Full capacity	Full capacity	One spray line & 4 fan coolers	Full capacity	One spray line & 4 fan coolers

Table 9. Main steam-line break mass and energy release rates - case 1.

Time (s)	Mass (lbm/s)	Energy (Btu/lbm)
0.0	8894	1199
0.1	8894	1199
0.2	8551	1197
0.3	8250	1198
0.4	8004	1197
0.5	7812	1197
1.0	7104	1199
2.0	5595	1194
3.0	4637	1189
5.0	3388	1182
7.0	2262	1201
9.0	2019	1200
10.0	1940	1199
15.0	1747	1198
20.0	1699	1197
25.0	1625	1197
30.0	1588	1196
35.0	1578	1196
50.0	991	1188
70.0	8	1174
100.0	8	1174
100.1*	0	0

*At this time, the steam generator has reached a dryout condition, and the steam generator and containment are in pressure equilibrium. The continued injection of auxiliary feedwater will result in oscillation in the blowdown flow. However, the mass release rate will be less than 28 lbs/sec (200 gpm water) and will not significantly influence the course of the accident since the containment pressure and temperature have already passed their peak and are rapidly decreasing. The analysis was, therefore terminated.

Table 10. Main steam-line break mass and energy release rates - case 2.

Time (s)	Mass (lbm/s)	Energy (Btu/lbm)
0.0	8891	1199
0.1	8891	1199
0.2	8543	1197
0.3	8245	1197
0.4	8003	1197
0.5	7809	1197
1.0	7103	1199
2.0	5592	1194
3.0	4632	1189
5.0	3384	1182
7.0	2261	1201
10.0	1932	1199
15.0	1735	1198
20.0	1680	1197
25.0	1661	1197
65.0	1604	1196
75.0	246	1175
100.0	57	1174
110.0*	0	1174

* See footnote at the bottom of Table 9.

Table 11. Main steam-line break mass and energy release rates - case 3.

Time (s)	Mass (lbm/s)	Energy (Btu/lbm)
0.0	8891	1199
0.1	8891	1199
0.2	8542	1197
0.3	8244	1197
0.4	8003	1197
0.5	7810	1197
1.0	7104	1199
2.0	5592	1194
3.0	4632	1189
5.0	3384	1182
7.0	2262	1201
19.0	2019	1200
10.0	1940	1199
15.0	1747	1198
20.0	1699	1197
25.0	1625	1197
30.0	1588	1196
35.0	1578	1196
50.0	911	1188
70.0	8	1174
100.0	8	1174
100.1*	0	0

*See footnote at the bottom of Table 9

Table 12. Main steam-line break mass and energy release rates – case 4.

Time (s)	Mass (lbm/s)	Energy (Btu/lbm)
0.0	11935	1192
0.1	11825	1191
0.2	11362	1190
0.3	10973	1190
0.4	10655	1191
0.5	10400	1191
1.0	9454	1194
2.0	7373	1190
3.0	6012	1187
4.0	4963	1183
5.0	4189	1180
6.0	3245	1182
8.0	2314	1200
10.0	1956	1199
15.0	1536	1196
20.0	1370	1194
25.0	1307	1193
30.0	1269	1193
40.0	1232	1192
50.0	1214	1192
70.0	1211	1191
80.0	935	1187
90.0	367	1176
100.0	353	1176
101.0	97	1174
102.0	28	1174
120.0	14	1174
131.0	6	1174
132.0*	0	1174

*See footnote at the bottom of Table 9.

Table 13. Main steam-Line break mass and energy release rates - case 5.

Time (s)	Mass (lbm/s)	Energy (Btu/lbm)
0.0	11935	1192
0.1	11820	1191
0.2	11355	1190
0.3	10965	1190
0.4	10649	1191
0.5	10396	1191
1.0	9454	1194
2.0	7368	1190
3.0	6005	1187
4.0	4958	1183
5.0	4183	1180
6.0	3236	1182
8.0	2297	1201
10.0	1956	1199
15.0	1536	1196
20.0	1370	1194
25.0	1307	1193
30.0	1269	1193
40.0	1232	1192
50.0	1214	1192
70.0	1211	1191
80.0	935	1187
90.0	367	1176
100.0	353	1176
101.0	97	1174
102.0	28	1174
120.0	14	1174
131.0	6	1174
132.0*	0	1174

*See footnote at the bottom of Table 9

Figure 1. Containment Atmosphere Pressure, Ginna Double-Ended Suction Leg Break
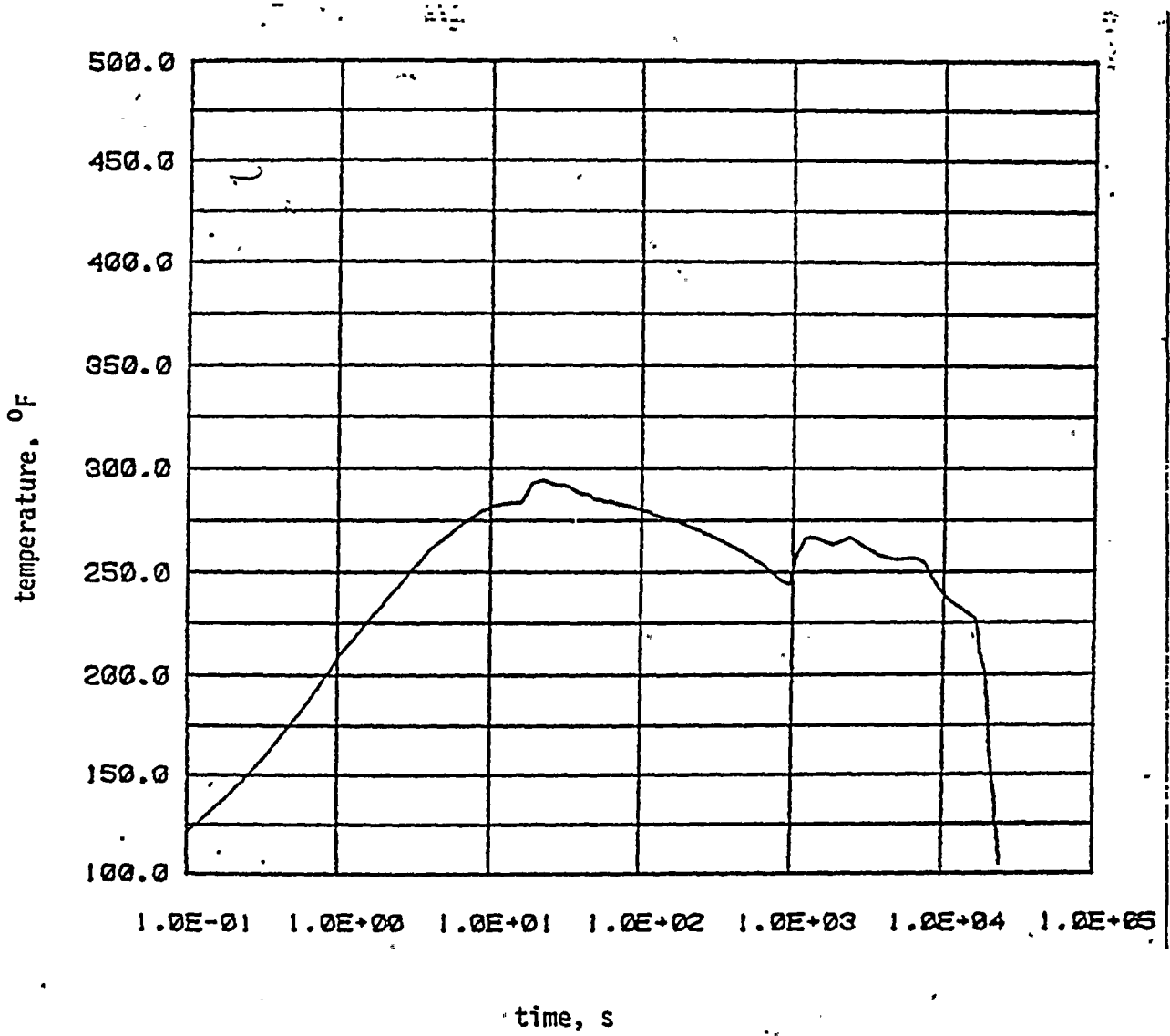
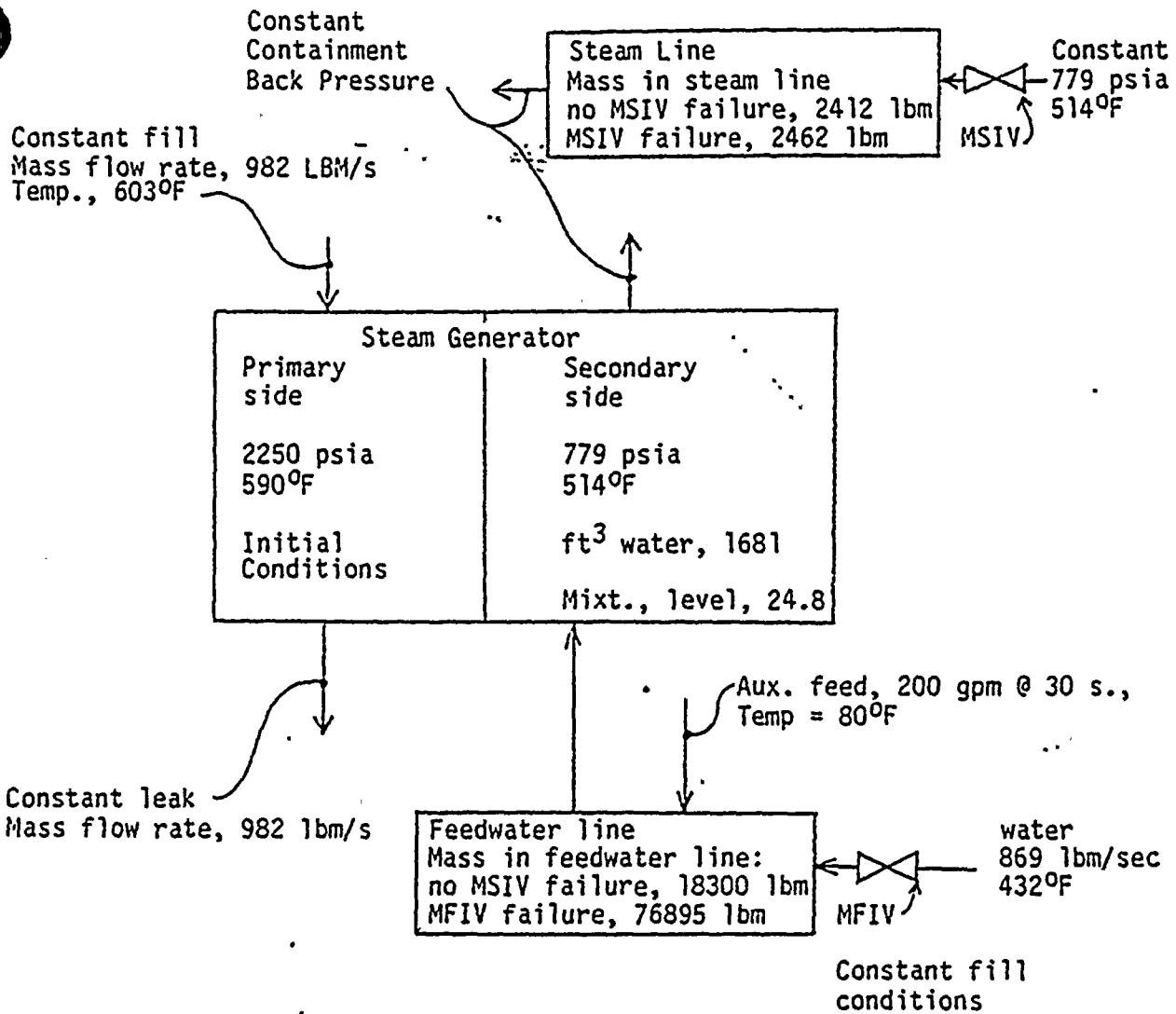Figure 2. Containment Atmosphere Temperature, Ginna Double-Ended Suction Leg Break

Constant
Containment
Back Pressure

Steam Line
Mass in steam line
no MSIV failure, 2412 lbm
MSIV failure, 2462 lbm

Constant
779 psia
514°F

MSIV

Constant fill
Mass flow rate, 982 LBM/s
Temp., 603°F

Steam Generator

Primary
side

Secondary
side

2250 psia
590°F

779 psia
514°F

Initial
Conditions

ft^3 water, 1681

Mixt., level, 24.8

Aux. feed, 200 gpm @ 30 s.,
Temp = 80°F

Constant leak
Mass flow rate, 982 lbm/s

Feedwater line
Mass in feedwater line:
no MSIV failure, 18300 lbm
MFIV failure, 76895 lbm

MFIV

water
869 lbm/sec
432°F

Constant fill
conditions

Figure 3.   Schematic of Analytical Model for Steam Line Break at 102% of Full
Power.

Constant
Containment
Back Pressure

Steam Line
Mass in steam line
no MSIV failure, 2412 lbm
MSIV failure, 2462 LBM

MSIV

Constant
1020 psia
547°F

Constant fill
Mass flow rate, 962 lbm/s
Temp., 547°F

Steam Generator
Primary side	Secondary side
2250 psia 547°F	1020 psia 547°F
Initial conditions	ft³ water, 2821
	Mixt., level, 41.8 ft

Aux. feed, 200 GPM @
0 s., Temp. = 80°F

Constant leak
Mass flow rate, 962 lbm/s

Valve closed
at hot standby

Feedwater line
Mass in feedwater line:
18300 lbm
432°F

MFIV

Constant fill
conditions

Figure 4. Schematic of Analytical Model for Steam Line Break at Hot
Standby Conditions.

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GINNA MSLB, CASE 1 - 102% POWER WITH MSIV FAILURE
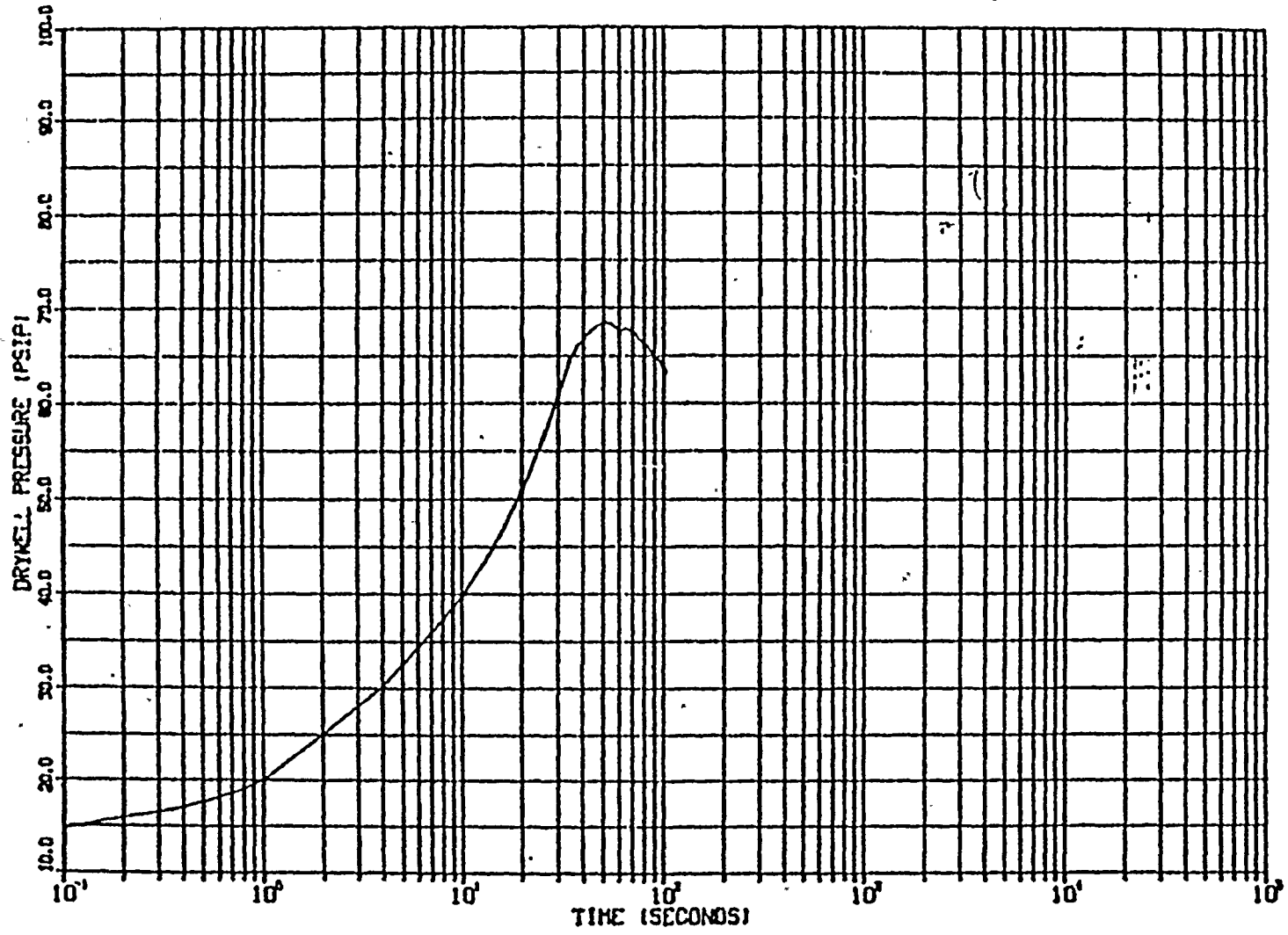
Figure 5. Ginna MSLB, Case 1 - 102% Power with MSIV Failure
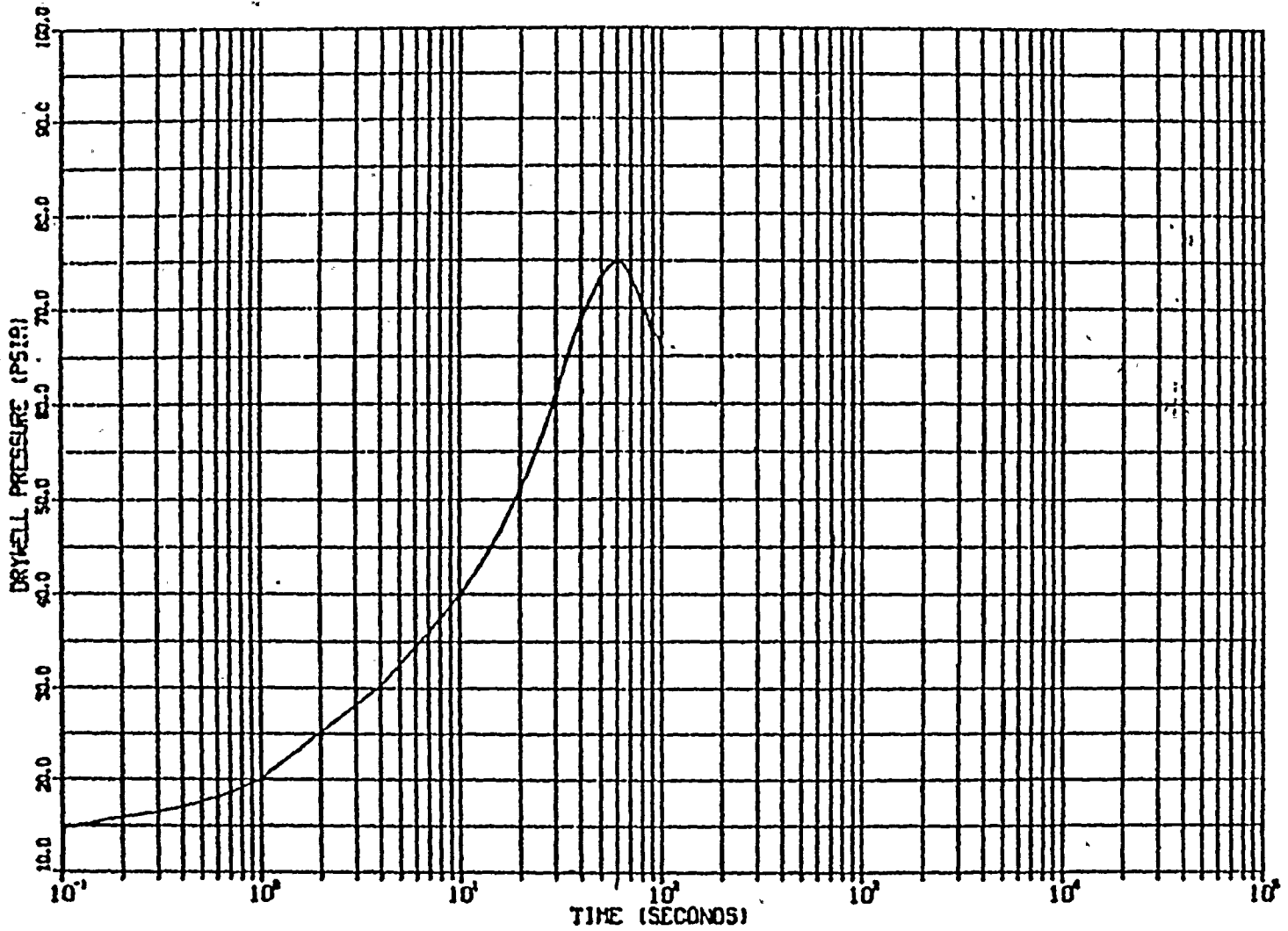
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GINNA MSLB, CASE 1 - 102% POWER WITH MSIV FAILURE
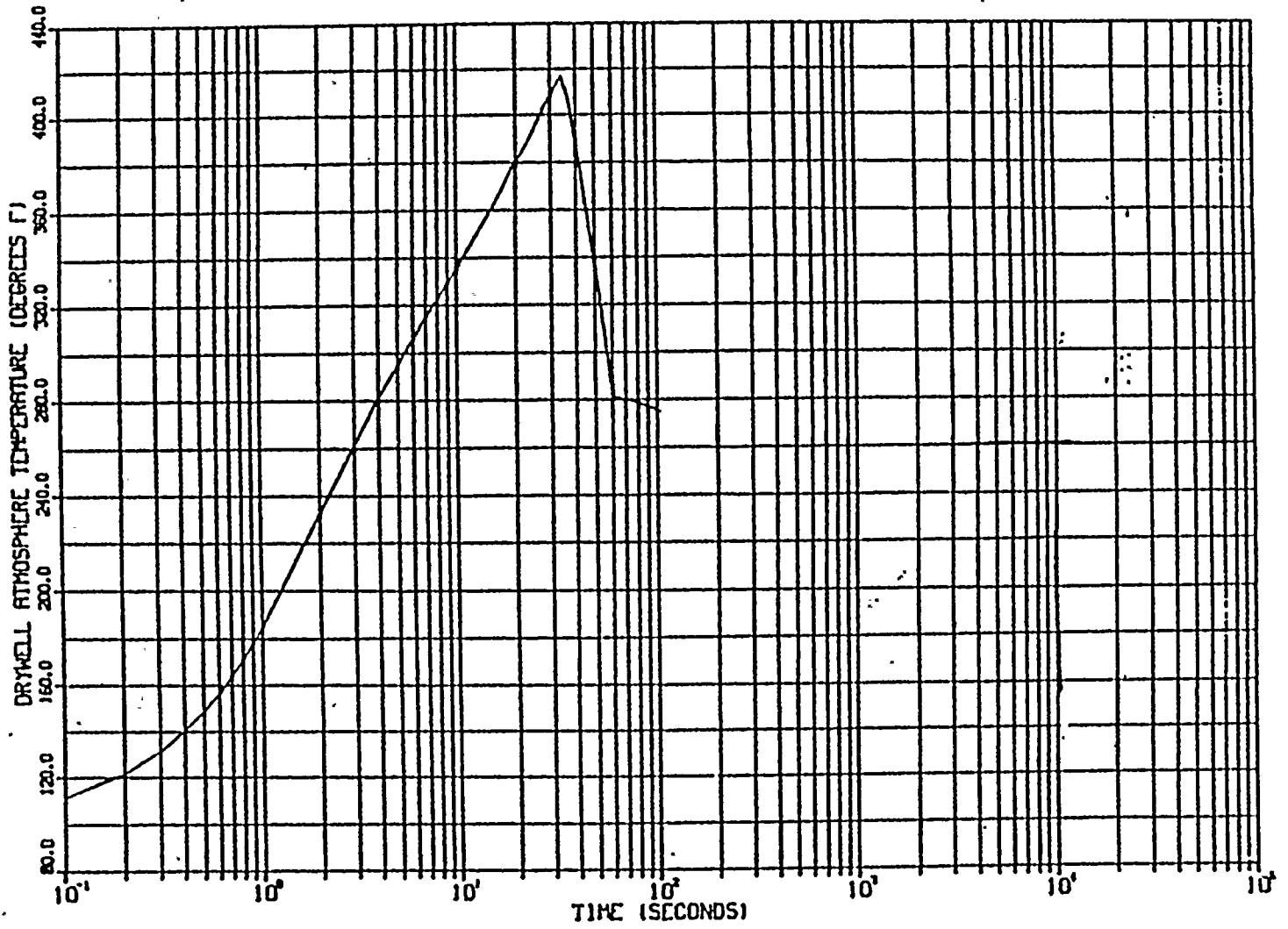
Figure 6. Ginna MSLB, Case 1 - 102% Power with MSIV Failure

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GINNA MSLB, CASE 2 - 102% POWER WITH MFIV FAILURE
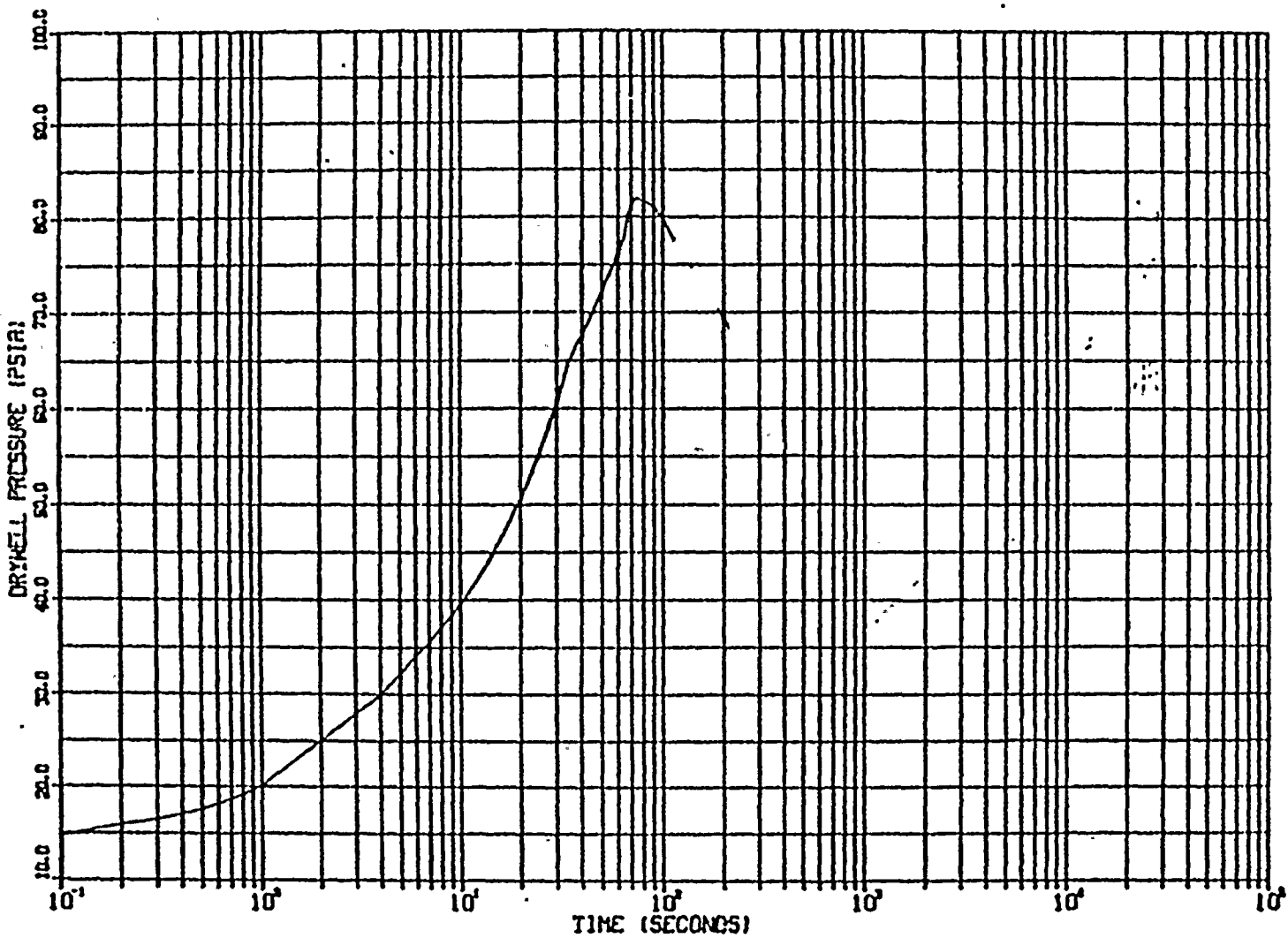
Figure 7. Ginna MSLB, Case 2 - 102% Power with MFIV Failure

GINNA MSLB, CASE 2 - 102% POWER WITH MFIV FAILURE
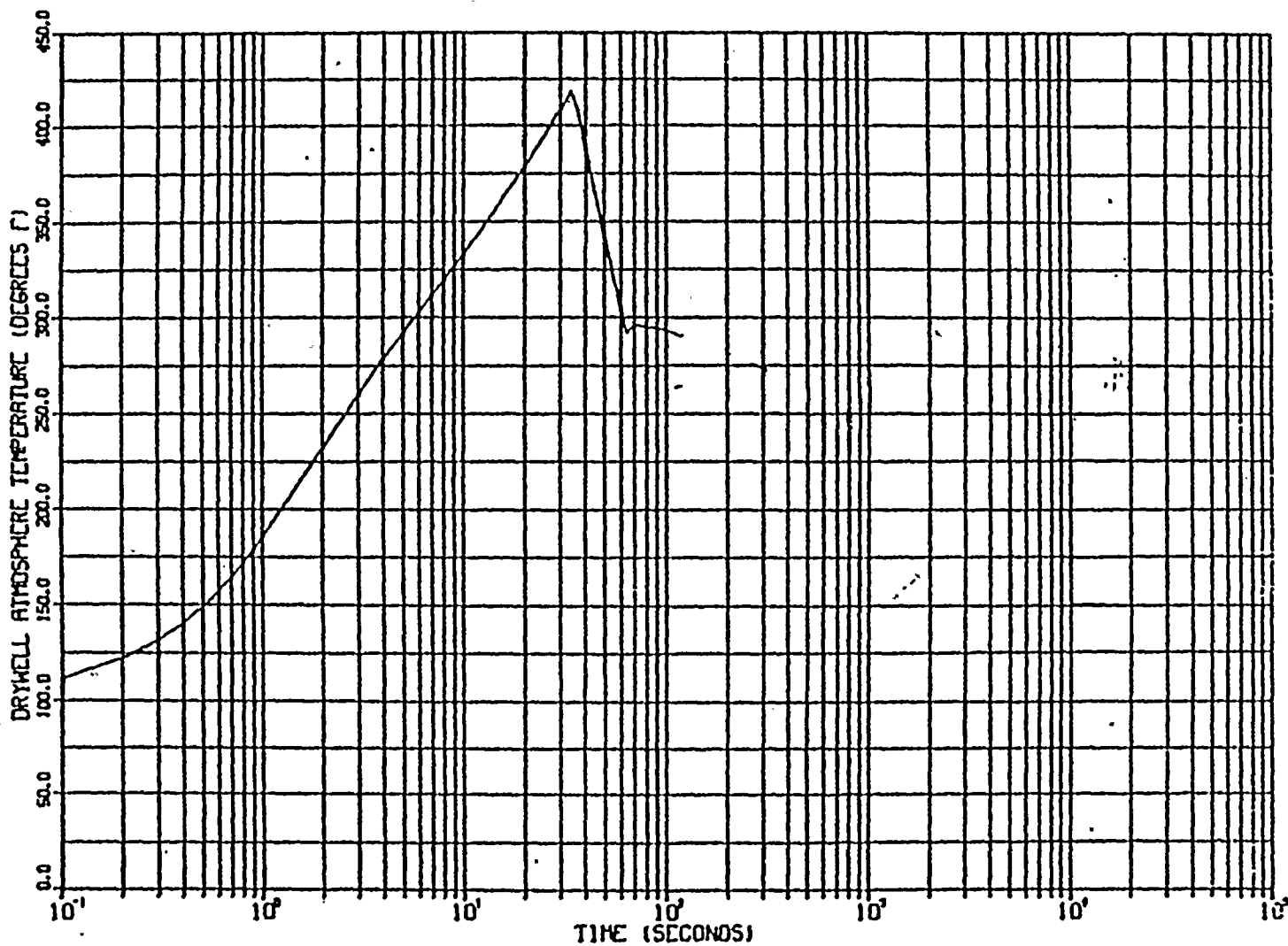
Figure 8.   Ginna MSLB, Case 2 - 102% Power with MFIV Failure

GINNA MSLB, CASE 3 — 102% POWER WITH CONT. SPRAY FAILURE

Figure 9. Ginna MSLB, Case 3 — 102% Power with Containment Spray Failure.
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GINNA MSLB, CASE 3 - 102% POWER WITH CONT. SPRAY FAILURE
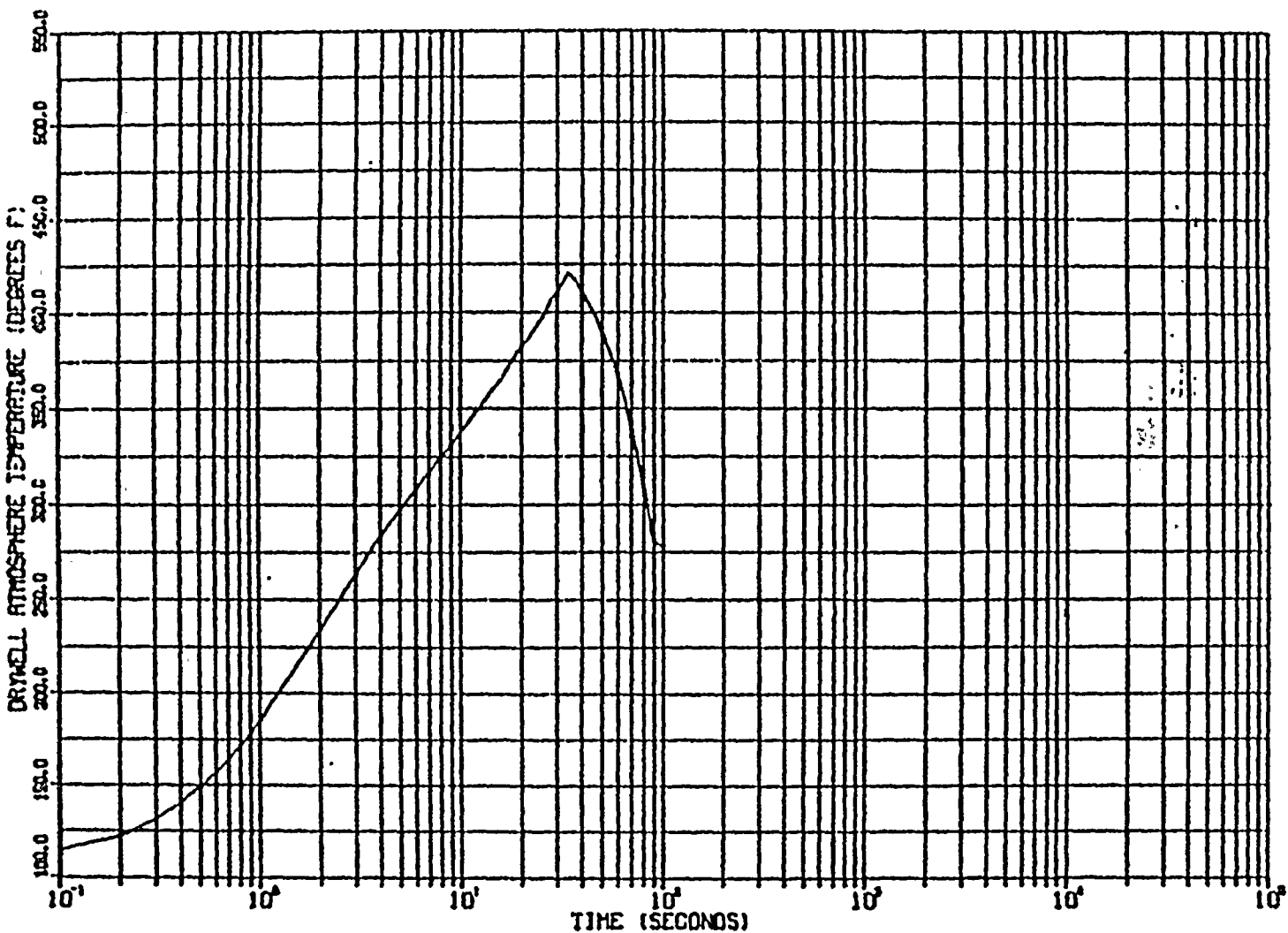
Figure 10. Ginna MSLB, Case 3 - 102% Power with Containment Spray Failure.

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GINNA MSLB, CASE 4 – HOT STANDBY WITH MSIV FAILURE
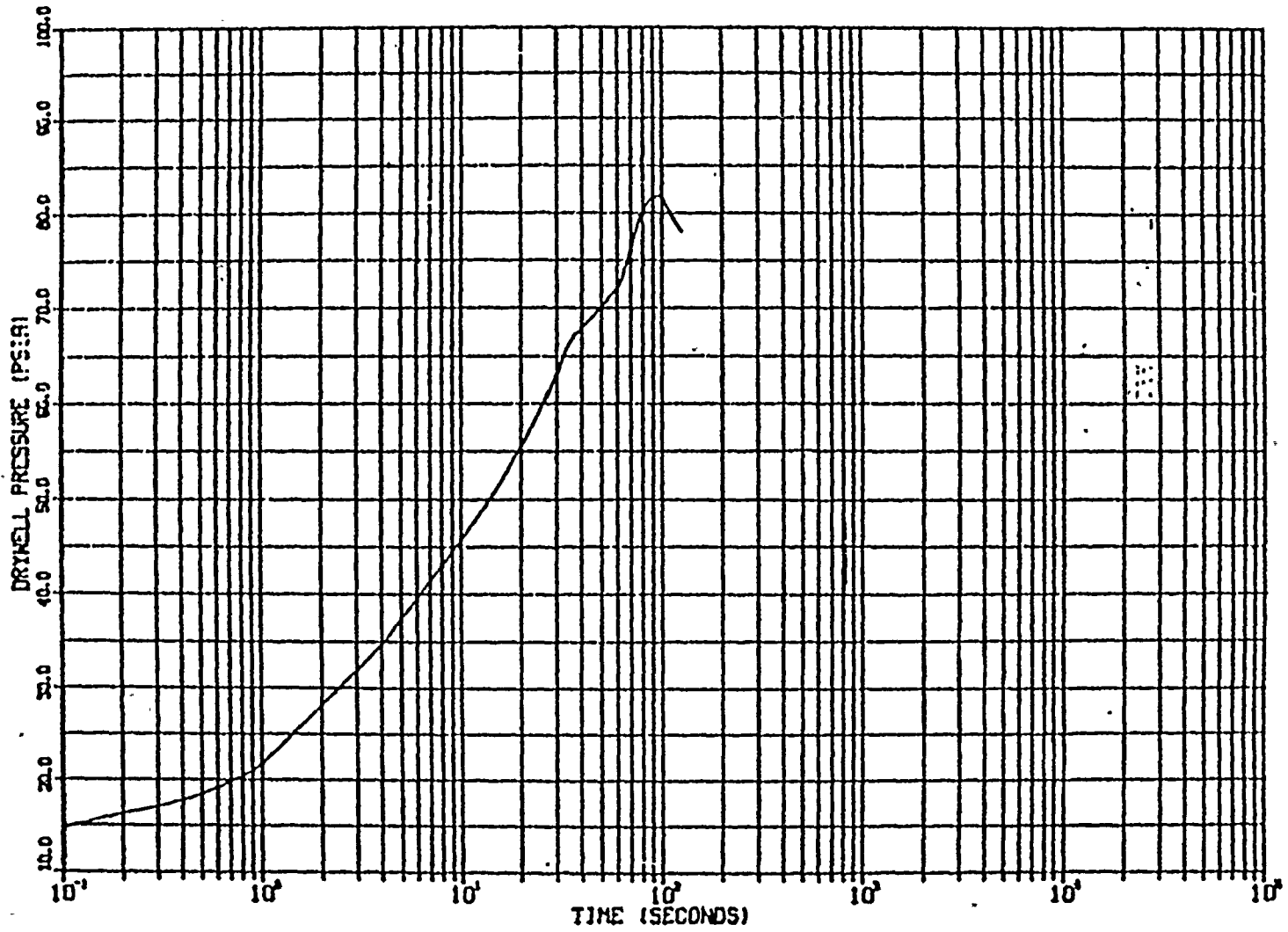
Figure 11. Ginna MSLB, Case 4 – Hot Standby with MSIV Failure
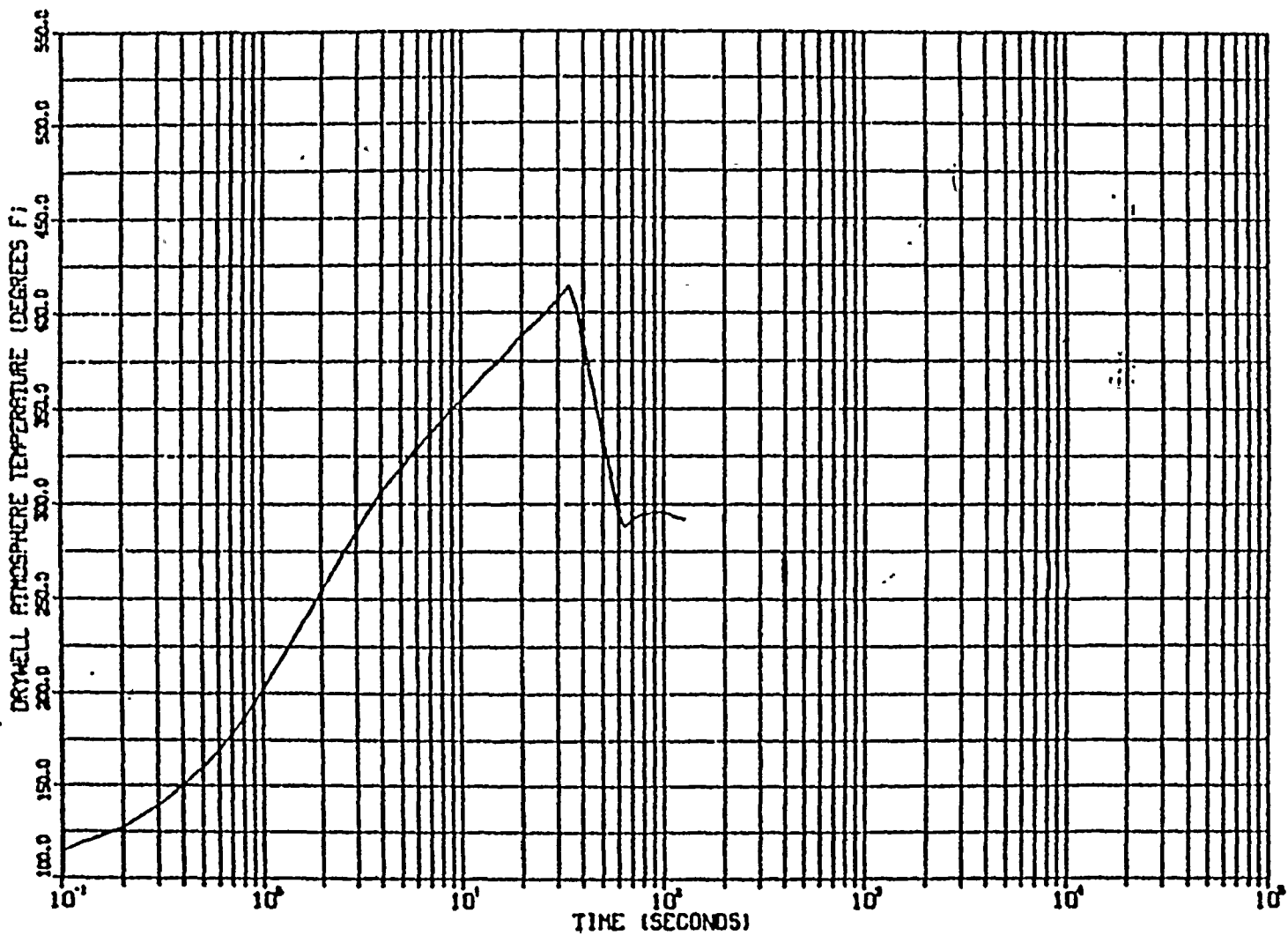
GINNA MSLB, CASE 4 - HOT STANDBY WITH MSIV FAILURE

Figure 12. Ginna MSLB, Case 4 - Hot Standby with MSIV Failure

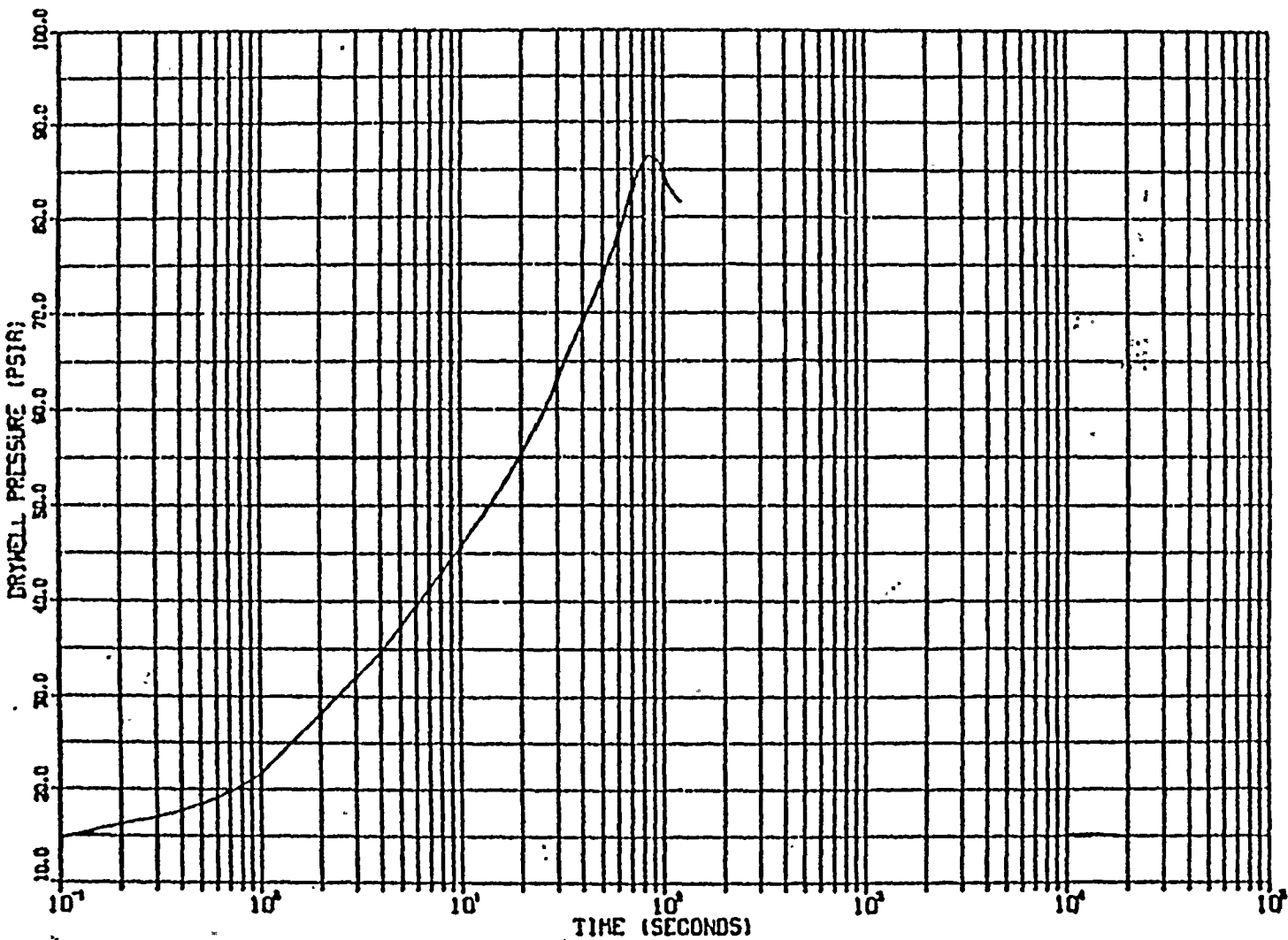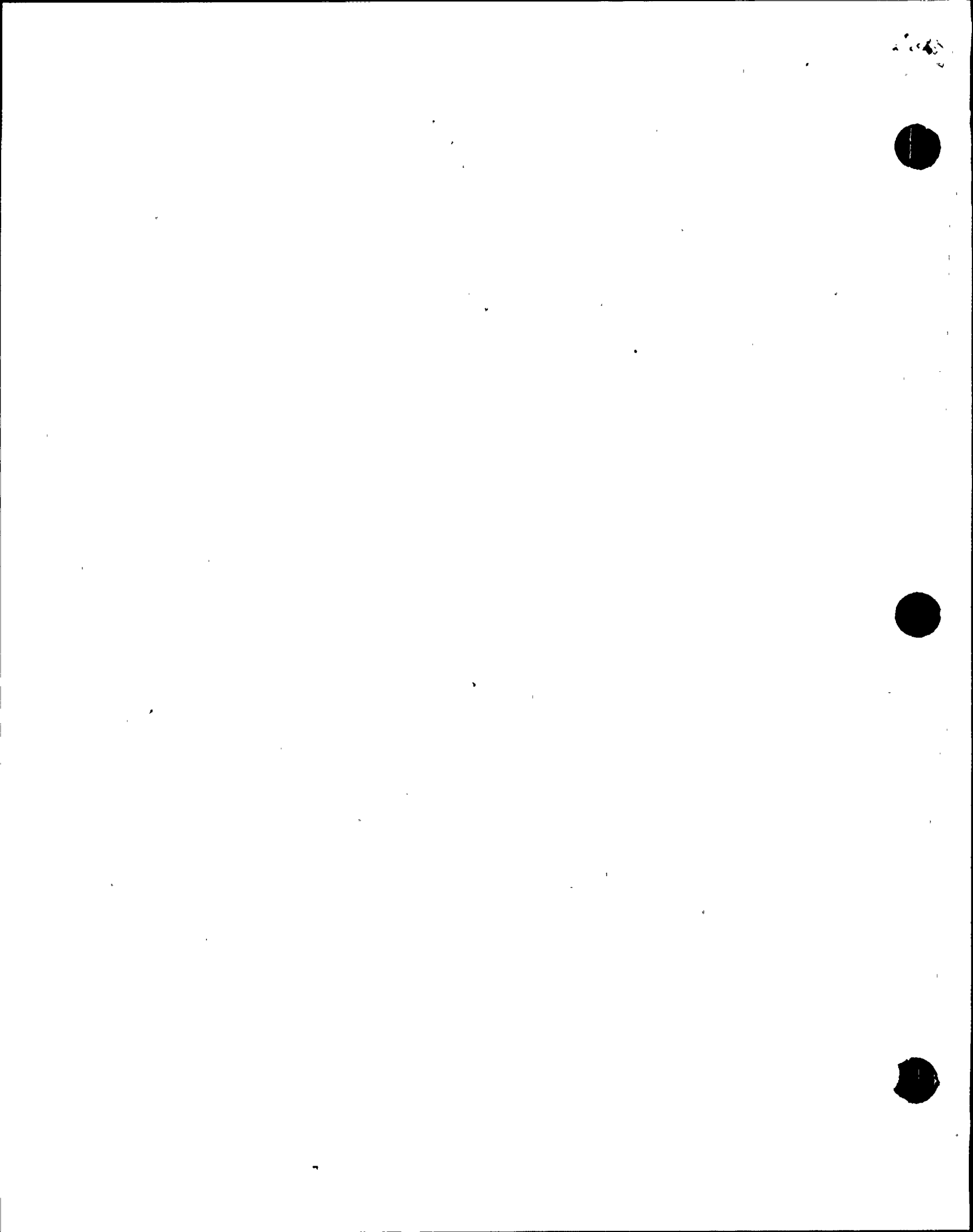# GINNA MSLB, CASE 5 - HOT STANDBY WITH CONT. SPRAY FAILURE

Figure 13. Ginna MSLB, Case 5 - Hot Standby with Containment Spray Failure

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GINNA MSLB, CASE 5 - HOT STANDBY WITH CONT. SPRAY FAILURE

Figure 14. Ginna MSLB, Case 5 - Hot Standby with Containment Spray Failure

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