Attachment 1
to Appendix B.

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649

February 1, 1982

RECEIVED FEB 3 1982

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: SEP Topics VI-2.D and VI-3
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Crutchfield:

This letter is in response to the draft evaluation of SEP Topic VI-2.D, "Mass and Energy Release Inside Containment" and SEP Topic VI-3, "Containment Pressure and Heat Removal Capability," which was transmitted by your letter dated November 3, 1981. We have reviewed the draft evaluation and have identified several conservatisms in the analysis for the loss of coolant accident (LOCA). These conservatisms and a qualitative discussion of their impact on the LOCA results, as well as a number of general comments on the evaluation, are dicussed in Attachment A. We also identified a number of conservatisms in the analysis for the main steam line break (MSLB). Because of the degree of conservatisms in the NRC evaluation, we performed a sensitivity study with the code CONTEMPT-EI/28B. This code is very similar to CONTEMPT-LT028, as discussed in Attachment B. Therefore, while this code has not been completely qualified for use as a licensing code, we believe that it is accurate and adequately represents Ginna. The sensitivity study, presented in Attachment B to this letter, confirmed the conservatism of the NRC results for MSLB.

Regarding the LOCA analysis, since the Ginna design basis pressure envelopes the NRC results, we conclude that the Ginna design basis pressure profile remains acceptable. The Ginna design basis temperature profile exceeds the NRC results except between 10,000 seconds and approximately 20,000 seconds after the design basis event. We propose that the Ginna design basis temperature profile remain as shown in the FSAR for times less than 10,000 seconds and be revised beyond 10,000 seconds as follows: from 10,000 seconds to 20,000 seconds, temperature = 250°F, beyond 20,000 seconds, temperature < 100°F. Since the containment temperature is already decreasing at this time, it is not considered

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that this revised profile would invalidate any conclusions drawn as part of our review of environmental qualification of electrical equipment, since the affected equipment temperatures would also be decreasing at this time from their peak temperatures.

Regarding the NRC steam break analyses, the NRC results for pressure and temperature exceed the Ginna design basis as revised above. However, as shown in Attachment B, acceptable results for containment pressure are obtained when more reasonable assumptions are used. It is our conclusion, based on Attachment B, that the Ginna design basis pressure profile exceeds the pressure profile calculated for main steam line break and, therefore, remains acceptable. We conclude that the temperature resulting from a steam line break in containment may exceed the Ginna design basis temperature profile, but that this is of no consequence due to the short duration of this exceedance and may therefore be ignored. This conclusion is based on NRC guidance provided in the Division of Operating Reactors (DOR) Guidelines which in turn is based, for example, on the short duration of the temperature spike, lower heat transfer coefficient, and the elevation of the steam lines relative to equipment. Thus, we conclude that the Ginna design basis temperature profile, as revised based on LOCA results discussed above, remains valid.

Very truly yours,

John E. Maier

John E. Maier

ATTACHMENT A

COMMENTS ON NRC SAFETY EVALUATION

1.	p. 5	Containment conditions resulting from a main steam break were assessed in a gross fashion in FSAR page 14.2.5-10. That analysis does not, however, comply with current criteria.
2.	p. 5	The LOCA analysis underestimated the effectiveness of the containment fan coolers by a heat removal rate of approximately 25 to 30 million BTU per hour. This resulted from an incorrect set of data being provided by RG&E to the NRC consultant (see also LER 81-022 transmitted by latter dated January 4, 1982 from John E. Maier, RG&E, to Ronald C. Haynes, NRC). We have estimated the impact of correcting the fan cooler heat removal rate to be on the order of a 1 to 2 psi reduction in peak containment pressure. This, therefore, is an additional conservation in the analysis.
3.	p. 6	The SER discussion of the result of a main steam line break should be revised to reflect the sensitivity study presented in Attachment B.
4.	p. 7	The last paragraph of the SER should be revised to reflect the conclusions presented in our letter transmitting this attachment.
5.	p.15	See our comment above regarding FSAR Section 14.2.5.
6.	p.16	The assumption of a constant containment pressure of 14.7 psia throughout the transient will result in an overestimate of the mass and energy release and, therefore, an overestimate of containment pressure and temperature.
7.	p.16	All information provided in conversations was also available on the docket.
8.	p.18	It is our understanding that only accumulator water was (conservatively) set at 272.9°F, not all ECCS flow. This should be clarified. If our understanding is not correct, then Table 1, which indicates RWST temperature, should be revised.
9.	p.21	The containment design pressure is incorrectly stated at the top of the page as being 74 psia; it is 75 psia.
10.	General	A number of other conservatisms are discussed in the LOCA evaluation. A more reasonable assessment would not require the level of conservatism employed here.
11.	p.21-25	See comments provided in Attachment B.

Attachment B

Containment Temperature/Pressure Following a Main Steam Line Break

Introduction

The purpose of this study is to provide a reevaluation of the containment conditions following a main steam line break. The first step will be to reconstruct the worst case containment temperature/pressure transients presented by the NRC in Reference 1 for a large steam break. Once the Reference 1 results have been reproduced, the assumptions necessary to reproduce those results can be evaluated. It may then be possible to remove some of the conservatism and calculate a more reasonable result.

Discussion

The containment temperatures and pressures presented in this study were calculated using the CONTEMPT-EI/28B computer code. The results presented in Reference 1 were calculated using CONTEMPT-LT/028. The CONTEMPT-EI/28B code is quite similar to the CONTEMPT-LT/028 code with changes which allow more user flexibility.

Hot Zero Power Case

The highest containment pressure was calculated in Reference 1 to occur for a large steam line break at HZP with failure of one spray train. The input for this case was run using CONTEMPT-EI/28B. Figure 1 and 2 illustrates the results of this run and points taken from Reference 1. The following peak temperature and pressure was obtained:

Reference 1 Case #5	85.8 psia @ 91 sec.
	413^{o} @ 34 sec.
CONTEMPT-EI/28B	83.4 psia @ 99.8 sec.
	403.9^{o} @ 35 sec.

While reproducing this case from the Reference 1 input one inconsistency was noted. Reference 1 states that spray was initiated 35 seconds after the setpoint at 30 psig was reached. In general, this pressure setpoint is reached at approximately 10 sec. Therefore, spray would start at approximately 45 sec. Since the temperature rise is terminated by spray, the peak temperature would occur when spray starts. All curves in Reference 1 illustrate peak temperature at approximately 35 sec. Therefore, it appears that the Reference 1 analysis neglects the time to reach the spray setpoint when actuating spray.

Using the reconstruction of Reference 1 case #5 as the base case, several cases were run to determine the sensitivity of containment temperature and pressure to various parameters. The results of these sensitivities are listed on Table 1 and are discussed below.

- Q/V, where Q is the total energy released to the time of peak containment pressure and is the containment volume, is a parameter associated with the Tagami film heat transfer correlation. This should represent total energy release to the time of peak pressure. A Q/V of ~ 165 results from the energy released to containment up to the time of peak pressure. However, a better approximation of the Reference 1 results can be obtained by reducing this parameter. The effect of reducing this parameter can be seen by comparing #1 and #3 on Table 1. Increasing Q/V results in increasing the film heat transfer coefficient. Changing Q/V from 87 to 165 results in approximately -1.0 psi pressure change and approximately -3.1° change in temperature (#3 temperature - 400.8° @ 35.0 sec.). Therefore, the Q/V term in Tagami may be doubled and still have only a small effect on containment temperature and pressure.

- The Uchida film heat transfer correlation has traditionally been used for steam breaks. When Uchida is used in the Ref. 1 Model a 3.7 psi pressure reduction and a 15.8° temperature reduction results (#1 versus #4 on Table 1).

- Exxon Nuclear Company (ENC) mass and energy release for the most limiting large steam line break (Ref. 3) was used in the evaluation. The ENC mass and energy was normalized to the total mass in the broken steam generator at HZP plus the mass released from the unaffected steam generator until main steam isolation occurs. The normalized ENC mass and energy is illustrated on Figure 3 with the mass and energy release used in the Reference 1 analysis. The mass associated with auxilary feed was not included. The effect of auxiliary feed on peak containment pressure and temperature would be negligible since the mass added during the time frame of interest is a very small fraction of the secondary side inventory (<1%).

  The effect of using the normalized Exxon mass and energy release is a pressure reduction of 3.3 psi and a temperature reduction of 17.4° (#4 and #5 on Table 1).

- A comparison between the RG&E containment model and the containment model used in Ref. 1 is illustrated on Table 2. The major difference between the models is the inclusion of the accumulators and ducting in the RG&E model. The area of the ducting is an assumed value based on values used by other plants, i.e.,

  Palisades = 20,072 @ 0.10 in.
  Indian Point = 22,000 @ 0.1382 in.
  Prairie Island = 22,000 @ 0.1875 in.

  Therefore, an assumption was made that Ginna had 20,000 sq. ft. @ 0.10 in.

  The CONTEMPT codes are sensitive to node spacing. A large spacing will result in lower surface temperatures which will result in removing too much energy from containment. The effect of node spacing can be seen by comparing #6 and #7. The effect of inclusion of accumulators and ducting is also illustrated on Table 1.

- In the process of doing this study it was determined that the heat removal capacity of the fan coolers used in the Reference 1 analysis was the capacity of one fan cooler at a service water temperature of 35°F. This corresponds to maximum cooling capability for one cooler. The minimum capability should be used in this analysis. The minimum capability is associated with the maximum service water temperature (80°F). Reference 4 presents a curve of heat removal versus containment pressure and equipment specifications presents the capacity at 120° and 286°.

  The following illustrates the heat removal capacity used in case #11 of Table 1: The capacities represent the minimum values of Reference 4 and equipment specifications; therefore, the values are conservative.

containment temperature °F	heat removal per fan MBTU/hr	total heat removal (4 fans) MBTU/hr
120	1.575	6.30
286	50.0	200.
308	54.72	218.9
320	56.52	226.0

  The effect of using the appropriate fan cooler capacity can be seen by comparing #10 and #11 on Table 1. This represents a 2.2 psi reduction in containment pressure and a 4.7° reduction in containment temperature.

- The effect of containment volume is illustrated on Table 1. Increasing the volume by 28,000 cu. ft. results in a 1.4 psi reduction in pressure and a 1.9° reduction in temperature. Since the gross volume of containment is approximately 1.13E6; 28,000 cu. ft. represents approximately 2.5% of the gross volume. Calculations show a net volume of approximately 1.037E6 cu. ft. Based on the FSAR the net volume of 9.72E5 cu. ft. represents a conservative small volume containing at least 3% margin. Therefore, a best estimate volume would be between 9.72E5 and 1.037E6 cu. ft. This represents available margin that was not used in this study.

Figures 4 and 5 illustrate the effect of Uchida, ENC mass and energy, and the RG&E containment heat sink model on the worst case containment response presented in Reference 1 (#1, #4, #5, and #11 of Table 1).

Using the RG&E containment heat sink model (volume = 9.72E5), ENC mass and energy release, Uchida correlation and fan cooler capacity of four fans results in:

72.4 psia @ 128.6 sec.
356.1° @ 20.3 sec.

<u>Full Power Case</u>

The highest containment temperature was calculated in Reference 1 to occur for a large steam break occurring at full power with failure of one spray train. The mass and energy release presented in Reference 1 for this case was coupled with the Reference 1 model discussed previously and containment temperature and pressure was calculated using the CONTEMPT-EI/28B code. The following results were obtained:

Reference 1 Case #3	75 psia @ 60 sec. 421° @ 34 sec.
CONTEMPT-EI/28B	73.3 psia @ 59.0 sec. 412.1° @ 35.0 sec.

The mass and energy release presented in Reference 1 was used with the RG&E containment heat sink model (volume = 9.72E5) previously described, Uchida correlation, and fan cooler capacity of four fans. This resulted in the following peak temperature and pressure:

63.2 psia @ 51.8 sec.
374.0° @ 32.0 sec.

The temperature and pressure versus time is illustrated on Figures 6 and 7 together with the reproduction of Reference 1 Case #3 using the CONTEMPT-EI/28B code.

Summary

In summary, the following compares the Reference 1 worst case with the comparable worst case calculated by RG&E as previously described:

case	Reference 1 Results	RG&E Results
Steam Break - HZP	85.8 psia @ 91 sec 413° @ 34 sec	72.4 psia @ 128.6 sec 356.1° @ 20.3 sec
Steam Break - HFP	75 psia @ 60 sec 421° @ 34 sec	63.2 psia @ 51.8 sec 374.0° @ 32.0 sec

For the worst case transients the calculated peak pressure is less than the design pressure for the Ginna containment (60 psig). The temperature is above the design temperature for Ginna containment (286°F). However, the temperature is exceeded only for a short period of time.

The RG&E calculations do not account for revaporization, entrainment, or best estimate containment volume. Inclusion of these effects would result in additional margin to design limits.

1. Revaporization - Reference 2 presents a discussion on revaporization. A temperature response is presented for a large steam line break using the Uchida heat transfer coefficient. When revaporization is used the temperature profile is reduced by approximately 40°. This would also result in a reduction in containment pressure.

2. Entrainment - In reality the steam flowing out of the break would not be dry steam but would contain some moisture. As the moisture content of the steam increases, the energy associated with the steam decreases; therefore, the energy added to containment decreases. This would result in a decrease in containment pressure and temperature. It has been estimated that the decrease in containment pressure and temperature resulting from accounting for entrainment would be similar to the decrease associated with revaporization.

3. Containment Volume - As previously described, a best estimate containment volume would be between 1.037E6 and 9.72E5 cu. ft. Increasing the containment volume used in the analysis would result in a slight pressure decrease.

References

1. NRC letter from D. M. Crutchfield to J. E. Maier, "Systematic Evaluation Program (SEP) for the R. E. Ginna Nuclear Power Plant - Evaluation Report on Topics VI-2.D and VI-3," November 3, 1981.

2. NRC letter from R. Tedesco to R. Mattson, V. Stello, and R. Boyd, "Best Estimate Evaluation for Environmental Qualification of Equipment Inside Containment Following A Main Steam Line Break," February 21, 1978.

3. Exxon Report XN-NF-77-40 Supplement 1, "Plant Transient Analysis for the R. E. Ginna Unit 1 Nuclear Power Plant," March 1980.

4. Rochester Gas and Electric Corporation, R.E. Ginna, FSAR.

TABLE 1
SENSITIVITY STUDY
Hot Zero Power Case

Case	Peak Pressure (psia)	Peak Temperature (psia)	P	T
1. Ref. 1 Model, Q/V = 87	83.4 @ 99.8 sec.	403.9 @ 35.0 sec.	-	-
2. Ref. 1 Model, Q/V = 87, no fan coolers	83.7 @ 99.4 sec.	404.8 @ 35.0 sec.	+0.3	+ 0.9
3. Ref. 1 Model, Q/V = 165, spray @ 45 sec.	82.4 @ 84.4 sec.	408.0 @ 44.6 sec.	-1.0	+ 4.1
4. Ref. 1 Model, Uchida	79.7 @ 100.4 sec.	388.1 @ 34.8 sec.	-3.7	-15.8
5. Ref. 1 Model, Uchida, ENC mass/energy	76.4 @ 129.4 sec.	370.7 @ 32.6 sec.	-3.3	-17.4
6. RGE Model (large spacing), Uchida, ENC mass/energy	73.5 @ 129 sec.	359.7 @ 32.8 sec.	-2.9	-11.0
7. RGE Model (small spacing), Uchida, ENC mass/energy	75.1 @ 129.2 sec.	365.4 @ 32.6 sec.	+1.6	+ 5.7
8. same as 7 with containment volume = 1.0E6	73.7 @ 129.2 sec.	363.5 @ 32.8 sec.	-1.4	- 1.9
9. same as 7 with accumulators	74.9 @ 128.8 sec.	365.4 @ 34.6 sec.	-0.2	- 0.0
10. same as 9 with ducting	74.6 @ 129.0 sec.	360.8 @ 30.0 sec.	-0.3	- 4.6
11. same as 10 with 4 fan coolers	72.4 @ 128.6 sec.	356.1 @ 20.29 sec.	-2.2	-4.7

TABLE 2
CONTAINMENT MODEL

Reference 1		RG&E	
Containment Volume	9.72E5 cu. ft.	Containment Volume	9.72E5 cu. ft.
Insulated Dome and Walls	36,181 sq. ft.	same	36,181 sq. ft.
Uninsulated Dome and Walls	12,474 sq. ft.	same	12,474 sq. ft.
Sump Walls	2,342 sq. ft.	same	2,342 sq. ft.
-		Sump Floor	297
			*2,639 sq. ft.
-		Basement Floor	*7,955 sq. ft.
Refueling Cavity Wall and Floor	6,400 sq. ft.	same	6,400 sq. ft.
Outside Refueling Cavity and S.G. Comp.	21,800 sq. ft.	same	21,800
Operating Floor	9,162 sq. ft.	same	9,162
			30,962
Intermediate Floor	6,170 sq. ft.	same but 2 X Area =	12,340 sq. ft.
1.5 in. Beams	9,174 sq. ft.	same	9,174 sq. ft.
1.0 in. Beams	5,016 sq. ft.	same	5,016 sq. ft.
0.5 in. Beams	8,586 sq. ft.	same	8,586 sq. ft.
Crane Supports	5,756 sq. ft.	same	5,756 sq. ft.
Grating etc.	7,000 sq. ft.	same	7,000 sq. ft.
-		Accumulators	1,756 sq. ft.
-		Ducting @ 0.10 in.	20,000 sq. ft.

*Surface assumed to be in contact with pool water.

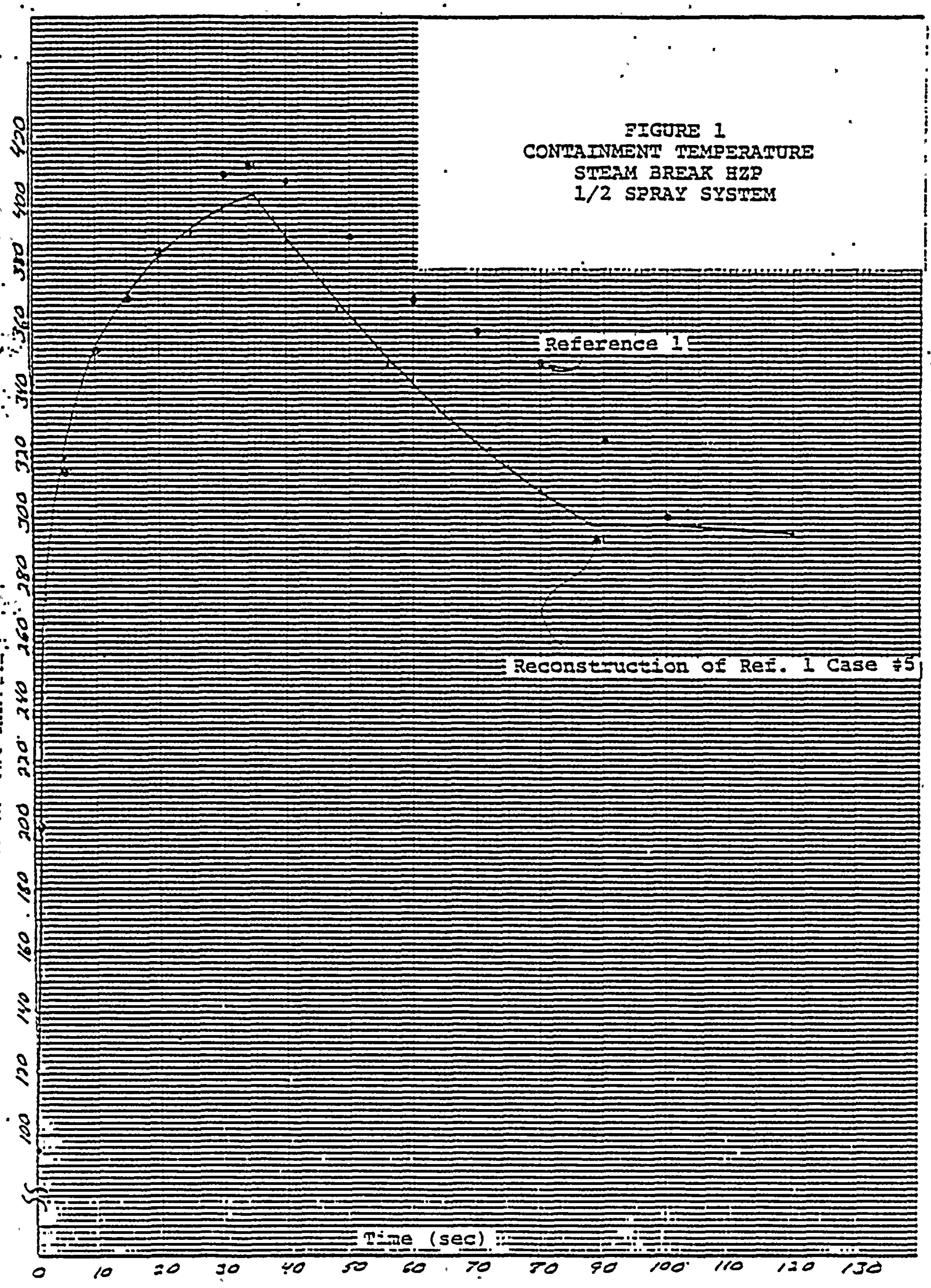

FIGURE 1
CONTAINMENT TEMPERATURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM
Reference 1
Reconstruction of Ref. 1 Case #5
Temperature (°F)
Time (sec)
0 10 20 30 40 50 60 70 80 90 100 110 120 130
100 120 140 160 180 200 220 240 260 280 300 320 340 360 380 400 420

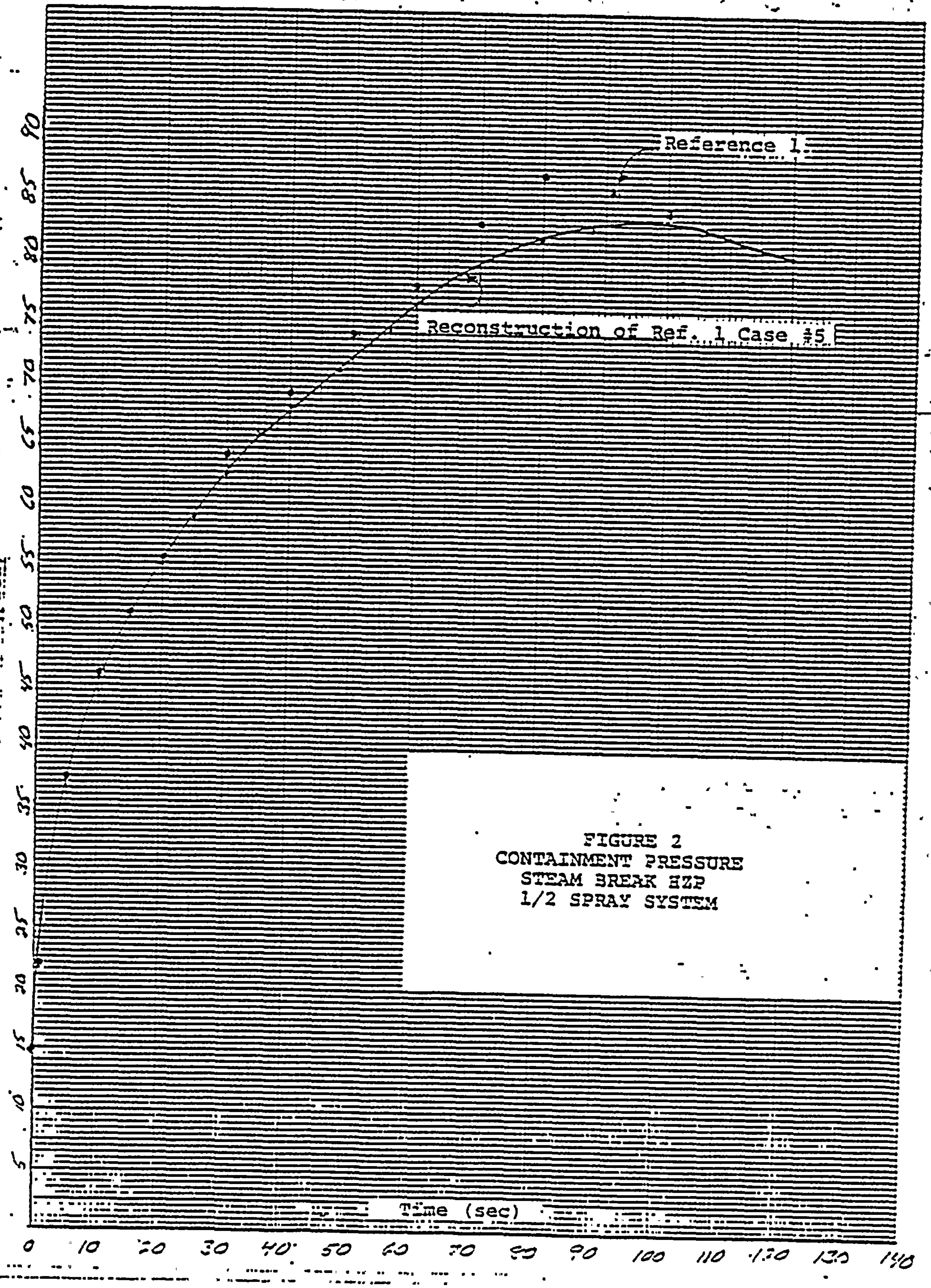

FIGURE 2
CONTAINMENT PRESSURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM

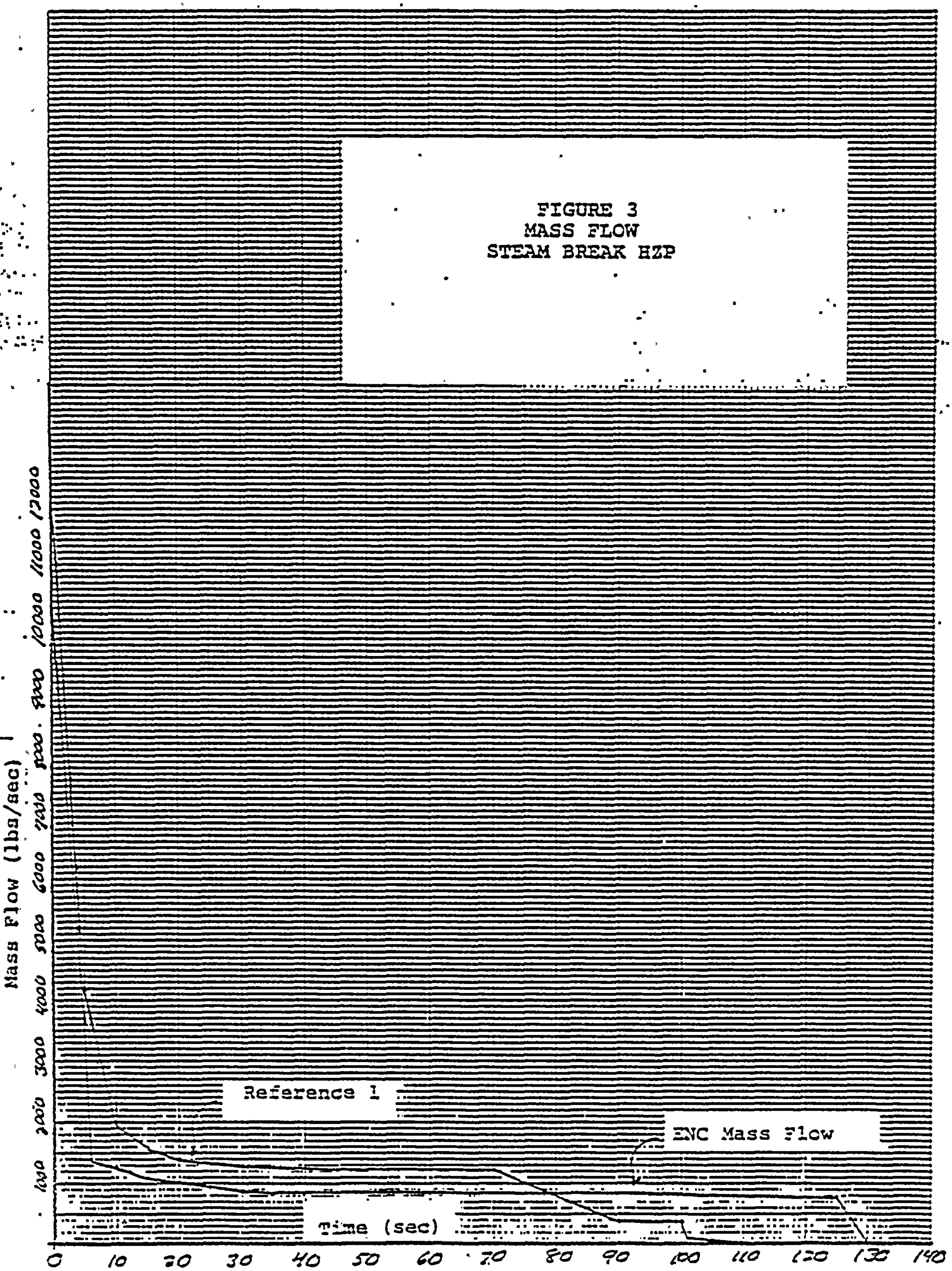

FIGURE 3
MASS FLOW
STEAM BREAK HZP
Mass Flow (lbs/sec)
Reference 1
ENC Mass Flow
Time (sec)
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
1000
2000
3000
4000
5000
6000
7000
8000
9000
10000
11000
12000

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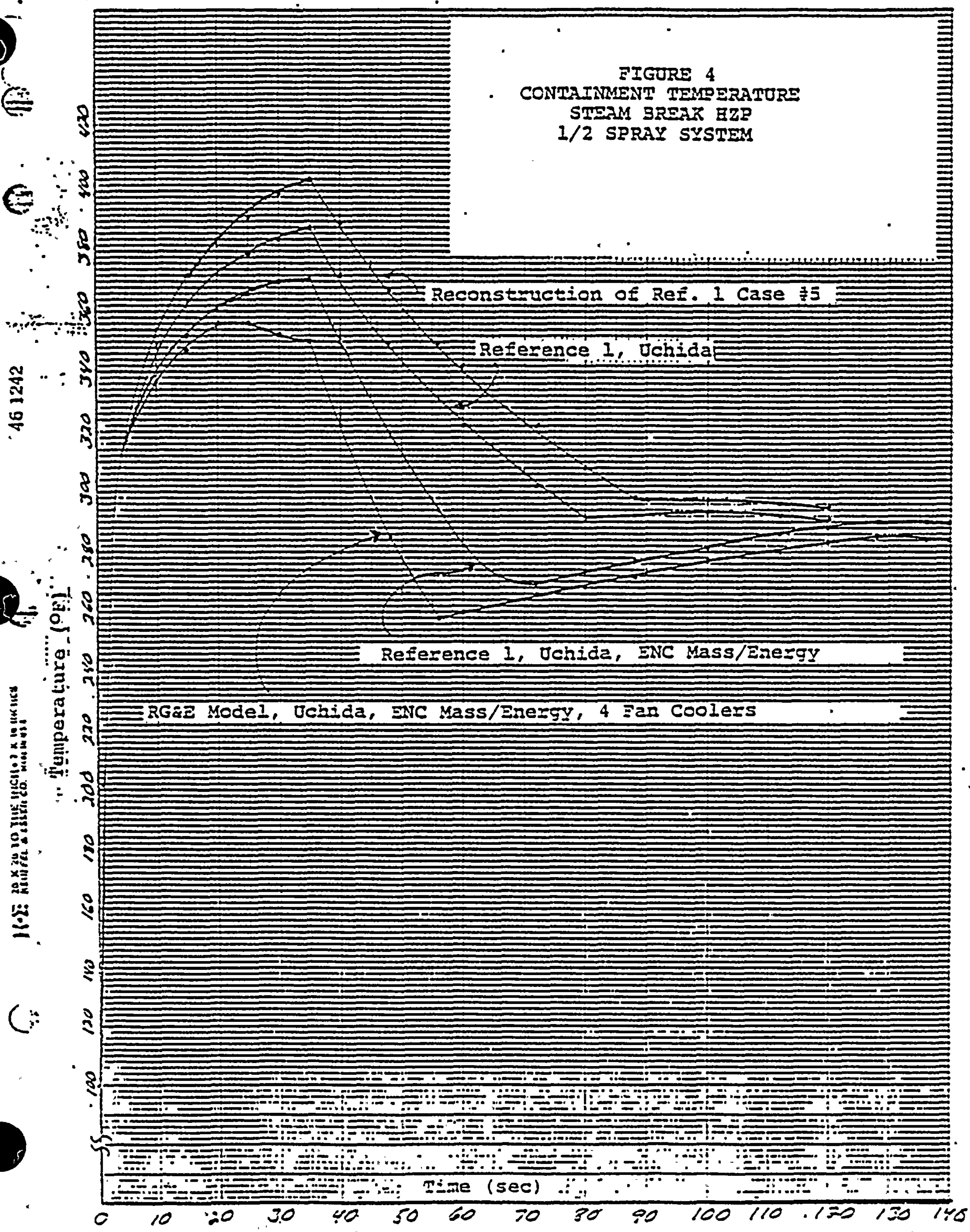

FIGURE 4
CONTAINMENT TEMPERATURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM
Reconstruction of Ref. 1 Case #5
Reference 1, Uchida
Reference 1, Uchida, ENC Mass/Energy
RG&E Model, Uchida, ENC Mass/Energy, 4 Fan Coolers
Temperature (°F)
Time (sec)
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
100
120
140
160
180
200
220
240
260
280
300
320
340
360
380
400
420
46 1242

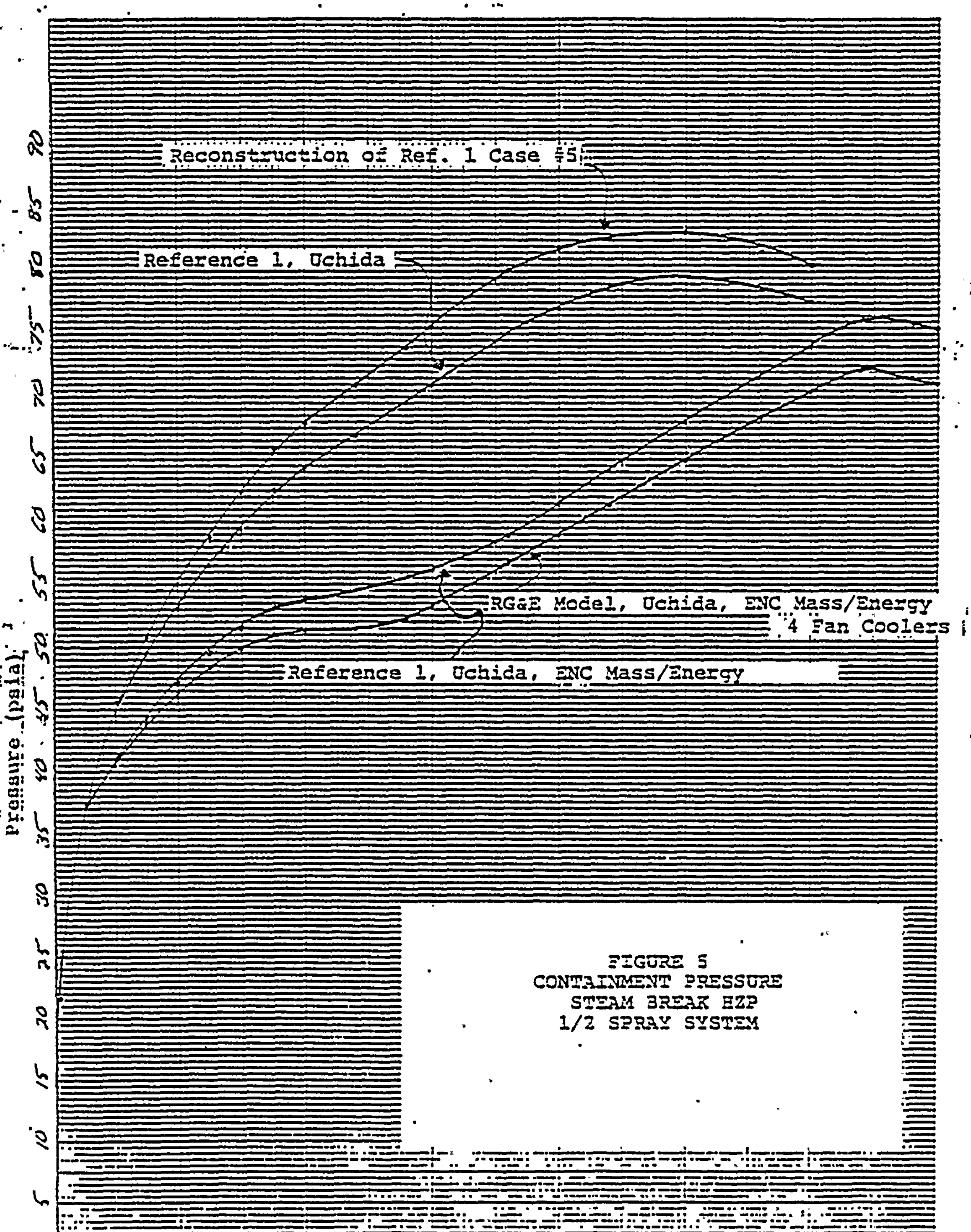

FIGURE 5
CONTAINMENT PRESSURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM

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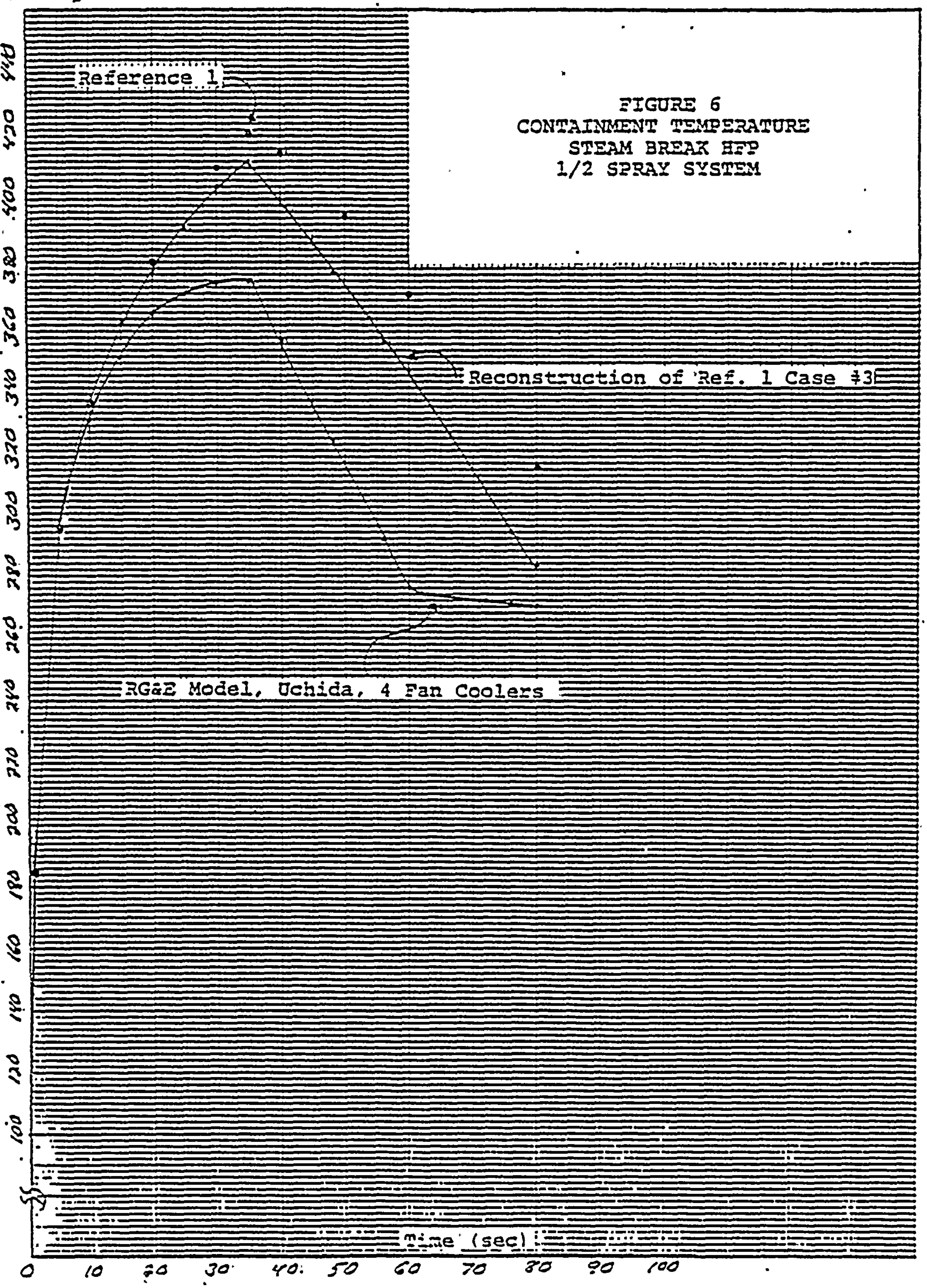

FIGURE 6
CONTAINMENT TEMPERATURE
STEAM BREAK HFP
1/2 SPRAY SYSTEM

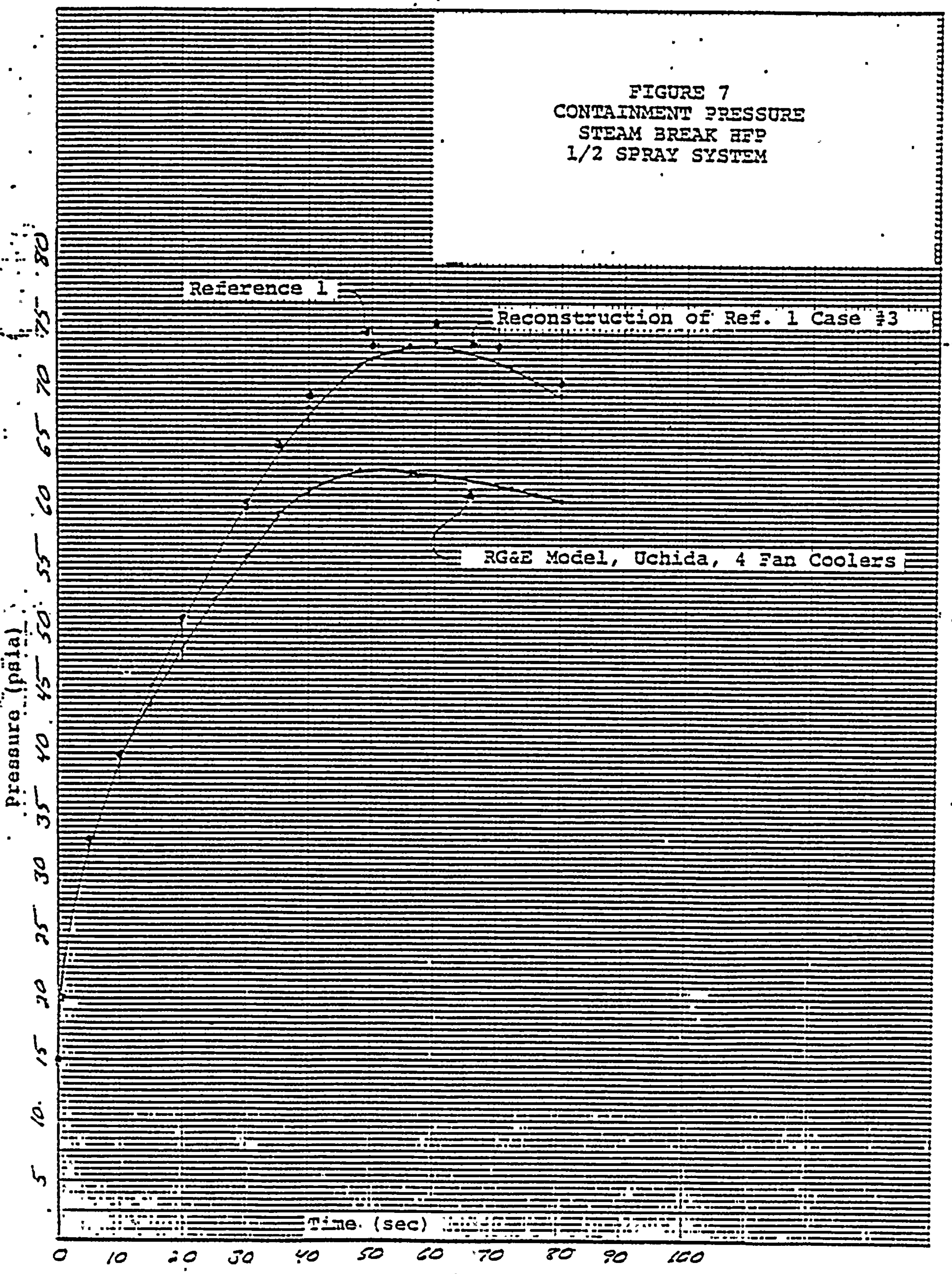
FIGURE 7
CONTAINMENT PRESSURE
STEAM BREAK HFP
1/2 SPRAY SYSTEM
Reference 1
Reconstruction of Ref. 1 Case #3
RG&E Model, Uchida, 4 Fan Coolers
Pressure (psia)
Time (sec)
0
10
20
30
40
50
60
70
80
90
100
5
15
25
35
45
55
65
75

TOPIC VI-3

SEE TOPIC VI-2.D

Ray Scholl

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
March 19, 1982

Docket No. 50-244
LS05-82-03-088

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: ELECTRICAL, INSTRUMENTATION AND CONTROL ASPECTS OF THE OVERRIDE OF CONTAINMENT PURGE VALVE ISOLATION (INCLUDING RESOLUTION OF SEP TOPIC VI-4, CONTAINMENT ISOLATION) R. E. GINNA

The staff has determined that the scope of review and evaluation performed for multi-plant generic activity B-24 addresses the electrical aspects of SEP Topic VI-4. Additional electrical review and evaluation is, therefore, not required.

Enclosed is a copy of our revised evaluation of the electrical override portion of generic activity B-24 for Ginna. This assessment compares your facility as described in Docket No. 50-244, with the criteria currently used by the regulatory staff for presently operating facilities. Our report replaces that issued by my letter of January 12, 1982 and reflects the information provided in your letter of March 2, 1981 and your responses to I&E Bulletin 80-06.

We require that by September 30, 1982, you provide physical features to augment existing administrative controls for each manual override. With regard to radiation monitoring, should further reviews of operating plants and/or additional requirements be deemed necessary, the Ginna plant will be included with that operating plant action.

Sincerely,

Dennis M. Crutchfield

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

Enclosures:
As stated

cc w/enclosures:
See next page

Mr. John E. Maier

cc

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REVISED
SAFETY EVALUATION REPORT

GINNA PLANT

OVERRIDE OF CONTAINMENT PURGE ISOLATION AND OTHER ENGINEERED SAFETY FEATURE ACTUATION SIGNALS

INTRODUCTION

As a result of Abnormal Occurrence #78-5, the NRC issued a generic letter requesting each licensee to take certain actions.

EVALUATION

The enclosed revised report (EGG-EA-5723, Rev. 2) was prepared for us by EG&G, Idaho, as part of our technical assistance program for SEP. Also, enclosed is EG&G Report 1183-4204, "Technical Evaluation of the Licensee's Response to I&E Bulletin 80-06." These reports provide a technical evaluation of the electrical, instrumentation and control design aspects of the override of containment purge valves isolation and other engineered safety feature actuation signals and is based upon review of these design aspects against the six NRC criteria provided for the review. The technical evaluation concludes that the modifications made by the licensee at the plant have not brought the designs of the engineered safety features into conformance with our review criteria.

The reports identify several areas in which the present plant does not satisfy the review criteria. The most important design problems are that the radiation monitors have not been demonstrated to satisfy Class 1E requirements and some ESF reset pushbuttons are unguarded.

We have reviewed the licensee's justification for not modifying the Containment Spray Additive Tank Discharge valves and find it acceptable on the basis that, following reset actuation, these valves close and that this would likely be the desired position. Further, the containment spray pumps remain in operation and chemical additives can be reinstated manually if required.

We have also reviewed the licensee's justification for not modifying the Main Feedwater Isolation and Bypass valves and find it acceptable on the basis that once the Feedwater Isolation reset has been actuated, the Feedwater Isolation and Bypass valves will not assume the position called for by their controllers unless the valves are in manual control. Since the plant is not operated at power levels above 15% with the valves in manual control, there is little consequence in the feedwater valves reopening. (Reopening of the Feedwater Isolation and Bypass valves may result in the addition of feedwater to a failed steam generator. This condition would occur if the pump discharge valves fail to close or fail to remain closed and the condensate booster pumps remain in operation.)

CONCLUSION

Based upon our review of the consultant's technical evaluations, we conclude that the electrical, instrumentation and control design aspects of the override of engineered safety feature actuation signals are acceptable, except for a lack of adequate physical protection for some of the ESF reset pushbuttons. The licensee must modify such pushbuttons to provide protection against inadvertent actuation.

With regard to radiation monitoring, should further reviews of operating plants and/or additional requirements be deemed necessary, the Ginna plant will be included with that operating plant action.

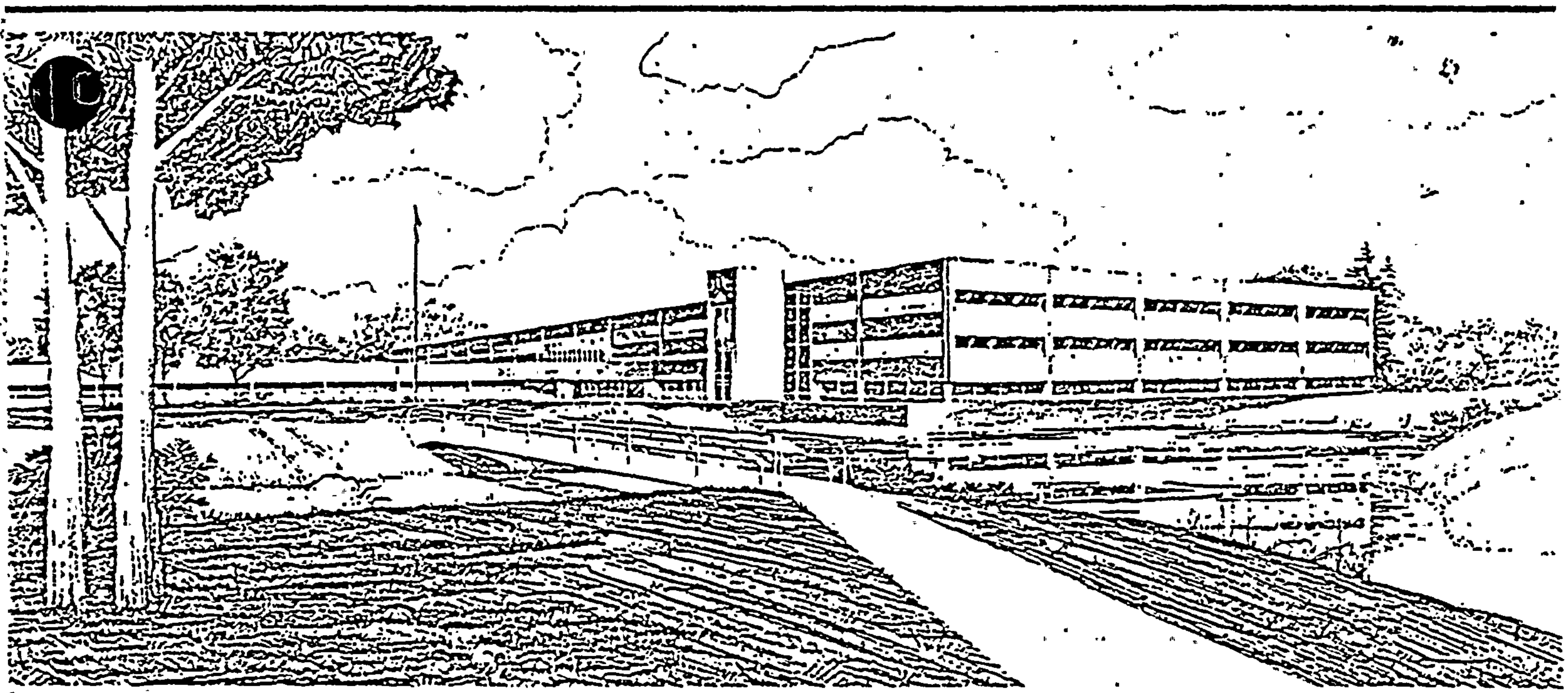
EGG-EA-5723, Rev. 2

JANUARY 1982

SYSTEMATIC EVALUATION PROGRAM TOPIC VI-4, ELECTRICAL, INSTRUMENTATION, AND CONTROL ASPECTS OF THE OVERRIDE OF CONTAINMENT PURGE VALVE ISOLATION, R. E. GINNA NUCLEAR POWER PLANT

A. C. Udy

U.S. Department of Energy
Idaho Operations Office • Idaho National Engineering Laboratory



This is an informal report intended for use as a preliminary or working document

Prepared for the
U. S. Nuclear Regulatory Commission
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6425

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INTERIM REPORT

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SYSTEMATIC EVALUATION PROGRAM

TOPIC VI-4

ELECTRICAL, INSTRUMENTATION, AND CONTROL ASPECTS OF THE OVERRIDE OF CONTAINMENT PURGE VALVE ISOLATION

R. E. GINNA NUCLEAR POWER PLANT

Revision 2

Docket No. 50-244

January 1982

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1-7-82

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SYSTEMATIC EVALUATION PROGRAM
TOPIC VI-4

ELECTRICAL, INSTRUMENTATION, AND CONTROL ASPECTS OF THE OVERRRIDE OF CONTAINMENT PURGE VALVE ISOLATION

R. E. GINNA NUCLEAR POWER PLANT

1.0 INTRODUCTION

Based on the information supplied by the Rochester Gas and Electric Company (RGE) this report addresses the electrical, instrumentation, and control system design aspects of the Containment Ventilation Isolation (CVI) system and other related Engineered Safety Feature (ESF) functions for the Ginna plant.

Several instances have been reported where the automatic closure of the containment ventilation or purge isolation valves would not have occurred because the safety actuation signals were manually overridden or blocked during normal plant operations. Lack of proper management controls, procedural inadequacies, and circuit design deficiencies contributed to these instances. These events also brought into question the mechanical operability of the valves themselves. These events were determined by the Nuclear Regulatory Commission (NRC) to be an Abnormal Occurrence (#78-05) and accordingly, were reported to Congress.

The NRC is now reviewing the electrical override aspects of containment purging and venting for all operating reactors. On November 28, 1978, the NRC issued a letter, "Containment Purging During Normal Plant Operation"[1] to all Boiling Water Reactor and Pressurized Water Reactor licensees, which required a review of these systems by the licensee. RGE responded on February 16, 1979[3], March 30, 1979[4], and March 17, 1980[5]. The Final Safety Analysis Report (FSAR) and Westinghouse Drawing No. 882D612, Sheet 6,[6] also contain design information reviewed for this report. RGE letters of March 2, 1981,[7] November 19, 1979[8] and December 1, 1981[9] also contain information on the control systems that was reviewed for this report.

2.0 EVALUATION OF THE R. E. GINNA NUCLEAR POWER PLANT

2.1 Review Guidelines. The intent of this evaluation is to determine if the actuating signals for the ESF equipment meet the following NRC criteria:

1. Guideline No. 1--In keeping with the requirements of General Design Criteria 55 and 56, the override[a] of one

---

a. The following definitions are given for clarity of use in this evaluation:

Override: the signal is still present, and it is blocked in order to perform a function contrary to the signal.

Reset: the signal has come and gone, and the circuit is being cleared in order to return it to the normal condition.

1. High containment radiation

2. Safety injection signal (high containment pressure can initiate a safety injection signal).

RGE has indicated that these signals are derived from equipment "designed and constructed as a Class 1E system."[5] However, the radiation channels have not been shown to be Class 1E.

These eight valves (except for the radiation monitor valves) are air-operated butterfly valves and are used so that one is redundant for another on the same air line. Valve position lights show the actual valve position. The solenoid valves fail closed on loss of air or on loss of power. The radiation monitor valves are air-operated diaphragm valves which have either a check valve or a manual valve for redundancy.

The logic of the containment isolation and the CVI valves is shown in reference 6. In both systems, the manual actuation is overridden along with the automatic actuation signals by operation of a reset switch (one per safeguards train). This logic has since been modified as outlined below.[7]

As a result of the short-term lessons learned, the CVI valve control circuits have been modified to provide individual resetting of each isolation valve. Resetting a valve after automatic closure now requires operation of a key-locked reset switch and a valve reset (guarded) pushbutton switch. The valve then goes to the position the valve control circuit requires. Administrative procedure requires the valve controller to be in the closed position before resetting the valve logic.

2.3 <u>Containment Ventilation Isolation System Design Evaluation.</u> Guideline 1 requires that no signal override can prevent another safety actuation signal from functioning. Ginna has override provision in the reset switches. The circuits involved have been modified to comply with this guideline.[7]

Guideline 2 requires that reset and override switches have physical provisions to aid in the administrative control of these switches. The reset switches are keylocked. The individual valve reset switches are guarded. This guideline is satisfied.

Guideline 3 requires system level annunciation whenever an override affects the performance of a safety system. The literal intent of this guideline is not satisfied by the Ginna design; however, individual status lights monitor the status of each individual valve override. Thus, operators will be aware of the status of any overrides.

Guideline 4 requires that isolation of the CVI valves be actuated by several diverse signals. This criterion is met in that:

1. Safety injection will initiate isolation.

2. High pressure in the reactor building will initiate safety injection.

4.0 REFERENCES

1. NRC/DOR letter, A. Schwencer, to RGE and all BWR and PWR licensees, "Containment Purging During Normal Plant Operation," dated November 28, 1978.

2. RGE letter, L. D. White, Jr., to Director of Nuclear Reactor Regulation, U.S. NRC, "Containment Purging During Normal Plant Operations," January 2, 1979.

3. RGE letter, L. D. White, Jr., to Director of Nuclear Reactor Regulation, U.S. NRC, "Review of Safety Actuation Circuits with Overrides," February 16, 1979.

4. RGE letter, L. D. White, Jr., to Director of Nuclear Reactor Regulation, U.S. NRC, "Review of Safety Actuation Circuits with Overrides," March 30, 1979.

5. RGE letter, L. D. White, Jr., to Director of Nuclear Reactor Regulation, U.S. NRC, "SEP Topic VI-4, Containment Isolation System," March 17, 1980.

6. Drawing, Westinghouse Logic Diagram No. 882D612, Sheet 6, Revision 7, "Safeguards Actuation Signals."

7. RGE letter, J. E. Maier to Director of Nuclear Reactor Regulation, NRC, "SEP Topic VI-4, Containment Isolation (Purge Valve Reset)," March 2, 1981.

8. RGE letter, L. D. White to Director of Nuclear Reactor Regulation, NRC, "Discussion of Lessons Learned Short Term Requirements," November 19, 1979.

9. RGE letter, L. D. White to Director of Nuclear Reactor Regulation, NRC, "SEP Topic VI-4, Containment Isolation (Electrical)," December 1, 1981.

TECHNICAL EVALUATION OF THE LICENSEE'S RESPONSE TO I&E BULLETIN 80-06 CONCERNING ESF RESET CONTROLS AT THE R. E. GINNA NUCLEAR POWER STATION UNIT 1

INTERIM REPORT

EG&G
Energy Measurements Group
San Ramon Operations

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INTERIM REPORT

EG&G
Energy Measurements Group
San Ramon Operations

EGG 1183-4
May 1

TECHNICAL EVALUATION OF THE LICENSEE'S RESPONSE TO I&E BULLETIN 80-06 CONCERNING ESF RESET CONTROLS FOR THE R. E. GINNA NUCLEAR POWER PLANT, UNIT 1

(DOCKET NO. 50-244)

by

D. H. Laudenbach

Approved for Publication

J. R. Radosevic
Department Manager

This document is UNCLASSIFIED

Derivative Classifier:
Nicholas E. Broderick
Department Manager

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INTRODUCTION

On March 13, 1980, the USNRC Office of Inspection and Enforcement (I&E), issued I&E Bulletin 80-06, entitled "Engineered Safety Feature (ESF) Reset Controls," to all PWR and BWR facilities with operating licenses. I&E Bulletin 80-06 requested that the following actions be taken by the licensees:

(1) Review the drawings for all systems serving safety-related functions at the schematic/elementary diagram level to determine whether or not upon the reset of an ESF actuation signal all associated safety-related equipment remains in its emergency mode.

(2) Verify that the actual installed instrumentation and controls at the facility are consistent with the schematics reviewed in Item 1 above by conducting a test to demonstrate that all equipment remains in its emergency mode upon removal of the actuating signal and/or manual resetting of the various isolating or actuation signals. Provide a schedule for the performance of the testing in your response to this bulletin.

(3) If any safety-related equipment does not remain in its emergency mode upon reset of an ESF signal at your facility, describe proposed system modification, design change, or other corrective action planned to resolve the problem.

(4) Report in writing within 90 days the results of your review, include a list of all devices which respond as discussed in Item 3 above, actions taken or planned to assure adequate equipment control, and a schedule for implementation of corrective action.

This technical evaluation addresses the licensee's response to I&E Bulletin 80-06 and the licensee's proposed system modification, design change, and/or other corrective action planned to resolve the problem. In evaluating the licensee's response to the four Action Item requirements of the bulletin, the following NRC staff guidance is also used:

> Upon the reset of ESF signals, all safety-related equipment shall remain in its emergency mode. Multiple reset sequencing shall not cause the affected equipment to deviate from its emergency mode. Justification should be provided for any exceptions.

EVALUATION AND CONCLUSIONS

In a letter dated June 3, 1980 [Ref. 1], Rochester Gas and Electric Corporation, the licensee for R. E. Ginna Nuclear Power Plant, Unit 1, replied to I&E Bulletin 80-06. In a telephone conference call conducted on March 3, 1981 [Ref. 2], the licensee provided additional information and clarification to their written response.

The licensee reported [Ref. 1] that a drawing review has been completed at Ginna station for all systems serving safety-related functions. This review was conducted at the schematic level to determine whether all associated safety-related equipment would remain in its emergency mode upon the reset of an engineered safety feature actuation signal. The licensee identified [Ref. 1] the following equipment as not remaining in the emergency mode upon ESF reset:

1. Containment Spray additive tank discharge valves.
2. Main Feedwater isolation and bypass valves.

We conclude that the licensee has complied with the requirements of Action Items 1 and 4 of I&E Bulletin 80-06 by completing the drawing review of all systems serving safety-related functions and by identifying the devices that do not remain in their emergency mode upon ESF reset.

The licensee reported [Ref. 1] that testing to verify that actual installed instrumentation and controls were consistent with the schematics reviewed was completed during the May 1980 refueling outage. We conclude that the licensee has complied with the requirements of Action Item 2 of I&E Bulletin 80-06 by providing a schedule and completion date for the performance of testing.

The licensee indicated [Ref. 1] that no modifications or design changes were planned for the Containment Spray additive tank discharge valves nor for the Main Feeedwater isolation and bypass valves. The licensee offered justification [Ref. 1] for not modifying these devices and also provided [Ref. 2] a verbal explanation to enhance the justification offered in reference 1.

The licensee offered [Ref. 1] the following justification for not modifying the Containment Spray additive tank discharge valves:

> The Containment Spray circuit has a reset switch which gives the operator the means of resetting containment spray. Once the reset switch has been actuated, the spray additive tank discharge valves will return automatically to the position called for by their controllers. The containment spray pumps and their discharge valves would require operator action to change state. This capability is necessary so the operator has the flexibility in dealing with post-accident conditions within containment (i.e., LOCA or steam line break).

The licensee offered [Ref. 2] the following additional justification for not modifying the Containment Spray additive tank discharge valves:

> The valves associated with the spray additive tank will be opened automatically two minutes after the containment spray signal is actuated. The sodium hydroxide will flow due to the suction of the spray pumps and mix with refueling water prior to being discharged through the spray nozzle into the containment. After the containment spray signal is actuated, the operator has the capability to stop the timer if it has been detemined that actuation of the sodium hydroxide addition is not warranted. The operator also has the capability to reinstate the sodium hydroxide addition, if required. Emergency procedures set forth guidelines for this action based on one or more of the following:
>
> (1). High containment pressure in combination with a total loss of RCS pressure.
>
> (2) High radiation levels in combination with elevated containment pressure.
>
> (3) Pressure signals indicative of accumulator discharge into the RCS.

The licensee offered [Ref. 1] the following justification for not modifying the Main Feedwater isolation and bypass valves:

> The Feedwater Isolation circuit has a reset switch which gives the operator the means of resetting the isolation signal to the feedwater bypass valves. Once the reset switch is actuated, the feedwater bypass valves will assume the position called for by their controllers. The main feedwater valves will remain closed until the isolation logic clears, and then they will automatically assume the position called for by their controllers. It should be noted that a safety injection signal also causes the main feedwater pumps to be tripped and their discharge valves to automatically close; therefore, closing the main feedwater valves on a safety injection signal is redundant.

The licensee offered [Ref. 2] the following additional justification for not modifying the Main Feedwater isolation and bypass valves:

> While reset will result in the feedwater isolation valves returning to their demand position, reset does not affect the status of the feedwater pumps or the pump discharge valves. Thus, re-opening of the feedwater isolation (and bypass) valves would not result in the addition of feedwater to the steam generator via the feedwater lines.

The above justifications were offered by the licensee in lieu of any system modification, design change, or other corrective action. We have reviewed the justifications submitted by the licensee to insure that sufficient information is provided as a basis for the NRC staff to prepare a Safety Evaluation Report.

FINDINGS

Based on our review of the information and documents provided by the licensee, we find that the ESF reset controls for R. E. Ginna Nuclear Power Plant, Unit 1, satisfy the requirements of Action Items 1, 2, and 4 of I&E Bulletin 80-06.

In response to Action Item 3 of I&E Bulletin 80-06, the licensee identified several valves as not remaining in their emergency mode upon ESF reset and offered justification in lieu of any system modification, design change, or other corrective action.

REFERENCES

1. Rochester Gas and Electric Corporation letter (L.D. White, Jr.) to NRC I&E (B.H. Grier), "Response to I&E Bulletin 80-06," dated June 3, 1980.

2. Telephone conference call, NRC (P. Bender); Rochester Gas and Electric Corporation (R. McCready, G. Daniels); EG&G San Ramon (D. Hackett, D. Laudenbach), March 3, 1981.

Attachment I to Appendix B.

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649

February 1, 1982

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: SEP Topics VI-2.D and VI-3
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Crutchfield:

This letter is in response to the draft evaluation of SEP Topic VI-2.D, "Mass and Energy Release Inside Containment" and SEP Topic VI-3, "Containment Pressure and Heat Removal Capability," which was transmitted by your letter dated November 3, 1981. We have reviewed the draft evaluation and have identified several conservatisms in the analysis for the loss of coolant accident (LOCA). These conservatisms and a qualitative discussion of their impact on the LOCA results, as well as a number of general comments on the evaluation, are dicussed in Attachment A. We also identified a number of conservatisms in the analysis for the main steam line break (MSLB). Because of the degree of conservatisms in the NRC evaluation, we performed a sensitivity study with the code CONTEMPT-EI/28B. This code is very similar to CONTEMPT-LT028, as discussed in Attachment B. Therefore, while this code has not been completely qualified for use as a licensing code, we believe that it is accurate and adequately represents Ginna. The sensitivity study, presented in Attachment B to this letter, confirmed the conservatism of the NRC results for MSLB.

Regarding the LOCA analysis, since the Ginna design basis pressure envelopes the NRC results, we conclude that the Ginna design basis pressure profile remains acceptable. The Ginna design basis temperature profile exceeds the NRC results except between 10,000 seconds and approximately 20,000 seconds after the design basis event. We propose that the Ginna design basis temperature profile remain as shown in the FSAR for times less than 10,000 seconds and be revised beyond 10,000 seconds as follows: from 10,000 seconds to 20,000 seconds, temperature = 250°F, beyond 20,000 seconds, temperature < 100°F. Since the containment temperature is already decreasing at this time, it is not considered

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that this revised profile would invalidate any conclusions drawn as part of our review of environmental qualification of electrical equipment, since the affected equipment temperatures would also be decreasing at this time from their peak temperatures.

Regarding the NRC steam break analyses, the NRC results for pressure and temperature exceed the Ginna design basis as revised above. However, as shown in Attachment B, acceptable results for containment pressure are obtained when more reasonable assumptions are used. It is our conclusion, based on Attachment B, that the Ginna design basis pressure profile exceeds the pressure profile calculated for main steam line break and, therefore, remains acceptable. We conclude that the temperature resulting from a steam line break in containment may exceed the Ginna design basis temperature profile, but that this is of no consequence due to the short duration of this exceedance and may therefore be ignored. This conclusion is based on NRC guidance provided in the Division of Operating Reactors (DOR) Guidelines which in turn is based, for example, on the short duration of the temperature spike, lower heat transfer coefficient, and the elevation of the steam lines relative to equipment. Thus, we conclude that the Ginna design basis temperature profile, as revised based on LOCA results discussed above, remains valid.

Very truly yours,

John E. Maier

John E. Maier

ATTACHMENT A

COMMENTS ON NRC SAFETY EVALUATION

1. p. 5 Containment conditions resulting from a main steam break were assessed in a gross fashion in FSAR page 14.2.5-10. That analysis does not, however, comply with current criteria.

2. p. 5 The LOCA analysis underestimated the effectiveness of the containment fan coolers by a heat removal rate of approximately 25 to 30 million BTU per hour. This resulted from an incorrect set of data being provided by RG&E to the NRC consultant (see also LER 81-022 transmitted by letter dated January 4, 1982 from John E. Maier, RG&E, to Ronald C. Haynes, NRC). We have estimated the impact of correcting the fan cooler heat removal rate to be on the order of a 1 to 2 psi reduction in peak containment pressure. This, therefore, is an additional conservation in the analysis.

3. p. 6 The SER discussion of the result of a main steam line break should be revised to reflect the sensitivity study presented in Attachment B.

4. p. 7 The last paragraph of the SER should be revised to reflect the conclusions presented in our letter transmitting this attachment.

5. p.15 See our comment above regarding FSAR Section 14.2.5.

6. p.16 The assumption of a constant containment pressure of 14.7 psia throughout the transient will result in an overestimate of the mass and energy release and, therefore, an overestimate of containment pressure and temperature.

7. p.16 All information provided in conversations was also available on the docket.

8. p.18 It is our understanding that only accumulator water was (conservatively) set at 272.9°F, not all ECCS flow. This should be clarified. If our understanding is not correct, then Table 1, which indicates RWST temperature, should be revised.

9. p.21 The containment design pressure is incorrectly stated at the top of the page as being 74 psia; it is 75 psia.

10. General A number of other conservatisms are discussed in the LOCA evaluation. A more reasonable assessment would not require the level of conservatism employed here.

11. p.21-25 See comments provided in Attachment B.

Attachment B

Containment Temperature/Pressure Following a Main Steam Line Break

Introduction

The purpose of this study is to provide a reevaluation of the containment conditions following a main steam line break. The first step will be to reconstruct the worst case containment temperature/pressure transients presented by the NRC in Reference 1 for a large steam break. Once the Reference 1 results have been reproduced, the assumptions necessary to reproduce those results can be evaluated. It may then be possible to remove some of the conservatism and calculate a more reasonable result.

Discussion

The containment temperatures and pressures presented in this study were calculated using the CONTEMPT-EI/28B computer code. The results presented in Reference 1 were calculated using CONTEMPT-LT/028. The CONTEMPT-EI/28B code is quite similar to the CONTEMPT-LT/028 code with changes which allow more user flexibility.

Hot Zero Power Case

The highest containment pressure was calculated in Reference 1 to occur for a large steam line break at HZP with failure of one spray train. The input for this case was run using CONTEMPT-EI/28B. Figure 1 and 2 illustrates the results of this run and points taken from Reference 1. The following peak temperature and pressure was obtained:

Reference 1 Case #5	85.8 psia @ 91 sec. 413^{o} @ 34 sec.
CONTEMPT-EI/28B	83.4 psia @ 99.8 sec. 403.9^{o} @ 35 sec.

While reproducing this case from the Reference 1 input one inconsistency was noted. Reference 1 states that spray was initiated 35 seconds after the setpoint at 30 psig was reached. In general, this pressure setpoint is reached at approximately 10 sec. Therefore, spray would start at approximately 45 sec. Since the temperature rise is terminated by spray, the peak temperature would occur when spray starts. All curves in Reference 1 illustrate peak temperature at approximately 35 sec. Therefore, it appears that the Reference 1 analysis neglects the time to reach the spray setpoint when actuating spray.

Using the reconstruction of Reference 1 case #5 as the base case, several cases were run to determine the sensitivity of containment temperature and pressure to various parameters. The results of these sensitivities are listed on Table 1 and are discussed below.

- Q/V, where Q is the total energy released to the time of peak containment pressure and is the containment volume, is a parameter associated with the Tagami film heat transfer correlation. This should represent total energy release to the time of peak pressure. A Q/V of ~ 165 results from the energy released to containment up to the time of peak pressure. However, a better approximation of the Reference 1 results can be obtained by reducing this parameter. The effect of reducing this parameter can be seen by comparing #1 and #3 on Table 1. Increasing Q/V results in increasing the film heat transfer coefficient. Changing Q/V from 87 to 165 results in approximately -1.0 psi pressure change and approximately -3.1° change in temperature (#3 temperature - 400.8° @ 35.0 sec.). Therefore, the Q/V term in Tagami may be doubled and still have only a small effect on containment temperature and pressure.

- The Uchida film heat transfer correlation has traditionally been used for steam breaks. When Uchida is used in the Ref. 1 Model a 3.7 psi pressure reduction and a 15.8° temperature reduction results (#1 versus #4 on Table 1).

- Exxon Nuclear Company (ENC) mass and energy release for the most limiting large steam line break (Ref. 3) was used in the evaluation. The ENC mass and energy was normalized to the total mass in the broken steam generator at HZP plus the mass released from the unaffected steam generator until main steam isolation occurs. The normalized ENC mass and energy is illustrated on Figure 3 with the mass and energy release used in the Reference 1 analysis. The mass associated with auxilary feed was not included. The effect of auxiliary feed on peak containment pressure and temperature would be negligible since the mass added during the time frame of interest is a very small fraction of the secondary side inventory (<1%).

  The effect of using the normalized Exxon mass and energy release is a pressure reduction of 3.3 psi and a temperature reduction of 17.4° (#4 and #5 on Table 1).

- A comparison between the RG&E containment model and the containment model used in Ref. 1 is illustrated on Table 2. The major difference between the models is the inclusion of the accumulators and ducting in the RG&E model. The area of the ducting is an assumed value based on values used by other plants, i.e.,

  Palisades = 20,072 @ 0.10 in.
  Indian Point = 22,000 @ 0.1382 in.
  Prairie Island = 22,000 @ 0.1875 in.

  Therefore, an assumption was made that Ginna had 20,000 sq. ft. @ 0.10 in.

  The CONTEMPT codes are sensitive to node spacing. A large spacing will result in lower surface temperatures which will result in removing too much energy from containment. The effect of node spacing can be seen by comparing #6 and #7. The effect of inclusion of accumulators and ducting is also illustrated on Table 1.

- In the process of doing this study it was determined that the heat removal capacity of the fan coolers used in the Reference 1 analysis was the capacity of one fan cooler at a service water temperature of 35°F. This corresponds to maximum cooling capability for one cooler. The minimum capability should be used in this analysis. The minimum capability is associated with the maximum service water temperature (80°F). Reference 4 presents a curve of heat removal versus containment pressure and equipment specifications presents the capacity at 120° and 286°.

  The following illustrates the heat removal capacity used in case #11 of Table 1: The capacities represent the minimum values of Reference 4 and equipment specifications; therefore, the values are conservative.

containment temperature °F	heat removal per fan MBTU/hr	total heat removal (4 fans) MBTU/hr
120	1.575	6.30
286	50.0	200.
308	54.72	218.9
320	56.52	226.0

  The effect of using the appropriate fan cooler capacity can be seen by comparing #10 and #11 on Table 1. This represents a 2.2 psi reduction in containment pressure and a 4.7° reduction in containment temperature.

- The effect of containment volume is illustrated on Table 1. Increasing the volume by 28,000 cu. ft. results in a 1.4 psi reduction in pressure and a 1.9° reduction in temperature. Since the gross volume of containment is approximately 1.13E6; 28,000 cu. ft. represents approximately 2.5% of the gross volume. Calculations show a net volume of approximately 1.037E6 cu. ft. Based on the FSAR the net volume of 9.72E5 cu. ft. represents a conservative small volume containing at least 3% margin. Therefore, a best estimate volume would be between 9.72E5 and 1.037E6 cu. ft. This represents available margin that was not used in this study.

Figures 4 and 5 illustrate the effect of Uchida, ENC mass and energy, and the RG&E containment heat sink model on the worst case containment response presented in Reference 1 (#1, #4, #5, and #11 of Table 1).

Using the RG&E containment heat sink model (volume = 9.72E5), ENC mass and energy release, Uchida correlation and fan cooler capacity of four fans results in:

72.4 psia @ 128.6 sec.
356.1° @ 20.3 sec.

<u>Full Power Case</u>

The highest containment temperature was calculated in Reference 1 to occur for a large steam break occurring at full power with failure of one spray train. The mass and energy release presented in Reference 1 for this case was coupled with the Reference 1 model discussed previously and containment temperature and pressure was calculated using the CONTEMPT-EI/28B code. The following results were obtained:

Reference 1 Case #3	75 psia @ 60 sec. 421° @ 34 sec.
CONTEMPT-EI/28B	73.3 psia @ 59.0 sec. 412.1° @ 35.0 sec.

The mass and energy release presented in Reference 1 was used with the RG&E containment heat sink model (volume = 9.72E5) previously described, Uchida correlation, and fan cooler capacity of four fans. This resulted in the following peak temperature and pressure:

63.2 psia @ 51.8 sec.
374.0° @ 32.0 sec.

The temperature and pressure versus time is illustrated on Figures 6 and 7 together with the reproduction of Reference 1 Case #3 using the CONTEMPT-EI/28B code.

Summary

In summary, the following compares the Reference 1 worst case with the comparable worst case calculated by RG&E as previously described:

case	Reference 1 Results	RG&E Results
Steam Break - HZP	85.8 psia @ 91 sec 413° @ 34 sec	72.4 psia @ 128.6 sec 356.1° @ 20.3 sec
Steam Break - HFP	75 psia @ 60 sec 421° @ 34 sec	63.2 psia @ 51.8 sec 374.0° @ 32.0 sec

For the worst case transients the calculated peak pressure is less than the design pressure for the Ginna containment (60 psig). The temperature is above the design temperature for Ginna containment (286°F). However, the temperature is exceeded only for a short period of time.

The RG&E calculations do not account for revaporization, entrainment, or best estimate containment volume. Inclusion of these effects would result in additional margin to design limits.

1. Revaporization - Reference 2 presents a discussion on revaporization. A temperature response is presented for a large steam line break using the Uchida heat transfer coefficient. When revaporization is used the temperature profile is reduced by approximately 40°. This would also result in a reduction in containment pressure.

2. Entrainment - In reality the steam flowing out of the break would not be dry steam but would contain some moisture. As the moisture content of the steam increases, the energy associated with the steam decreases; therefore, the energy added to containment decreases. This would result in a decrease in containment pressure and temperature. It has been estimated that the decrease in containment pressure and temperature resulting from accounting for entrainment would be similar to the decrease associated with revaporization.

3. Containment Volume - As previously described, a best estimate containment volume would be between 1.037E6 and 9.72E5 cu. ft. Increasing the containment volume used in the analysis would result in a slight pressure decrease.

References

1. NRC letter from D. M. Crutchfield to J. E. Maier, "Systematic Evaluation Program (SEP) for the R. E. Ginna Nuclear Power Plant - Evaluation Report on Topics VI-2.D and VI-3," November 3, 1981.

2. NRC letter from R. Tedesco to R. Mattson, V. Stello, and R. Boyd, "Best Estimate Evaluation for Environmental Qualification of Equipment Inside Containment Following A Main Steam Line Break," February 21, 1978.

3. Exxon Report XN-NF-77-40 Supplement 1, "Plant Transient Analysis for the R. E. Ginna Unit 1 Nuclear Power Plant," March 1980.

4. Rochester Gas and Electric Corporation, R.E. Ginna, FSAR.

TABLE 1
SENSITIVITY STUDY
Hot Zero Power Case

Case	Peak Pressure (psia)	Peak Temperature (psia)	P	T
1. Ref. 1 Model, Q/V = 87	83.4 @ 99.8 sec.	403.9 @ 35.0 sec.	-	-
2. Ref. 1 Model, Q/V = 87, no fan coolers	83.7 @ 99.4 sec.	404.8 @ 35.0 sec.	+0.3	+ 0.9
3. Ref. 1 Model, Q/V = 165, spray @ 45 sec.	82.4 @ 84.4 sec.	408.0 @ 44.6 sec.	-1.0	+ 4.1
4. Ref. 1 Model, Uchida	79.7 @ 100.4 sec.	388.1 @ 34.8 sec.	-3.7	-15.8
5. Ref. 1 Model, Uchida, ENC mass/energy	76.4 @ 129.4 sec.	370.7 @ 32.6 sec.	-3.3	-17.4
6. RGE Model (large spacing), Uchida, ENC mass/energy	73.5 @ 129 sec.	359.7 @ 32.8 sec.	-2.9	-11.0
7. RGE Model (small spacing), Uchida, ENC mass/energy	75.1 @ 129.2 sec.	365.4 @ 32.6 sec.	+1.6	+ 5.7
8. same as 7 with containment volume = 1.0E6	73.7 @ 129.2 sec.	363.5 @ 32.8 sec.	-1.4	- 1.9
9. same as 7 with accumulators	74.9 @ 128.8 sec.	365.4 @ 34.6 sec.	-0.2	- 0.0
10. same as 9 with ducting	74.6 @ 129.0 sec.	360.8 @ 30.0 sec.	-0.3	- 4.6
11. same as 10 with 4 fan coolers	72.4 @ 128.6 sec.	356.1 @ 20.29 sec.	-2.2	-4.7

TABLE 2
CONTAINMENT MODEL

Reference 1		RG&E	
Containment Volume	9.72E5 cu. ft.	Containment Volume	9.72E5 cu. ft.
Insulated Dome and Walls	36,181 sq. ft.	same	36,181 sq. ft.
Uninsulated Dome and Walls	12,474 sq. ft.	same	12,474 sq. ft.
Sump Walls	2,342 sq. ft.	same	2,342 sq. ft.
-		Sump Floor	297
			*2,639 sq. ft.
-		Basement Floor	*7,955 sq. ft.
Refueling Cavity Wall and Floor	6,400 sq. ft.	same	6,400 sq. ft.
Outside Refueling Cavity and S.G. Comp.	21,800 sq. ft.	same	21,800
Operating Floor	9,162 sq. ft.	same	9,162
			30,962
Intermediate Floor	6,170 sq. ft.	same but 2 X Area =	12,340 sq. ft.
1.5 in. Beams	9,174 sq. ft.	same	9,174 sq. ft.
1.0 in. Beams	5,016 sq. ft.	same	5,016 sq. ft.
0.5 in. Beams	8,586 sq. ft.	same	8,586 sq. ft.
Crane Supports	5,756 sq. ft.	same	5,756 sq. ft.
Grating etc.	7,000 sq. ft.	same	7,000 sq. ft.
-		Accumulators	1,756 sq. ft.
-		Ducting @ 0.10 in.	20,000 sq. ft.

*Surface assumed to be in contact with pool water.

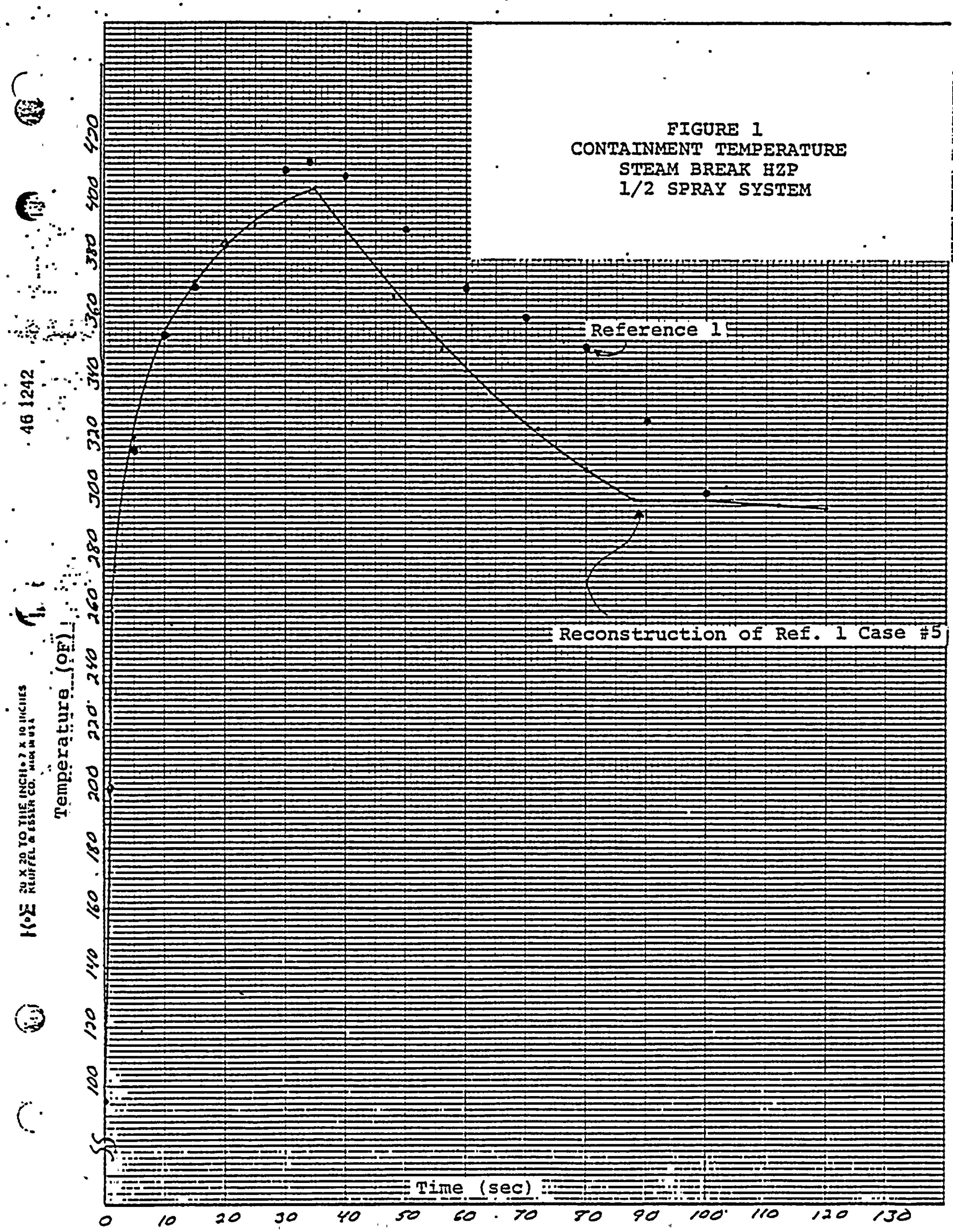

FIGURE 1
CONTAINMENT TEMPERATURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM
Reference 1
Reconstruction of Ref. 1 Case #5
Temperature (OF)
Time (sec)
0
10
20
30
40
50
60
70
80
90
100
110
120
130
100
120
140
160
180
200
220
240
260
280
300
320
340
360
380
400
420
46 1242
20 X 20 TO THE INCH • 7 X 10 INCHES
KEUFFEL & ESSER CO.

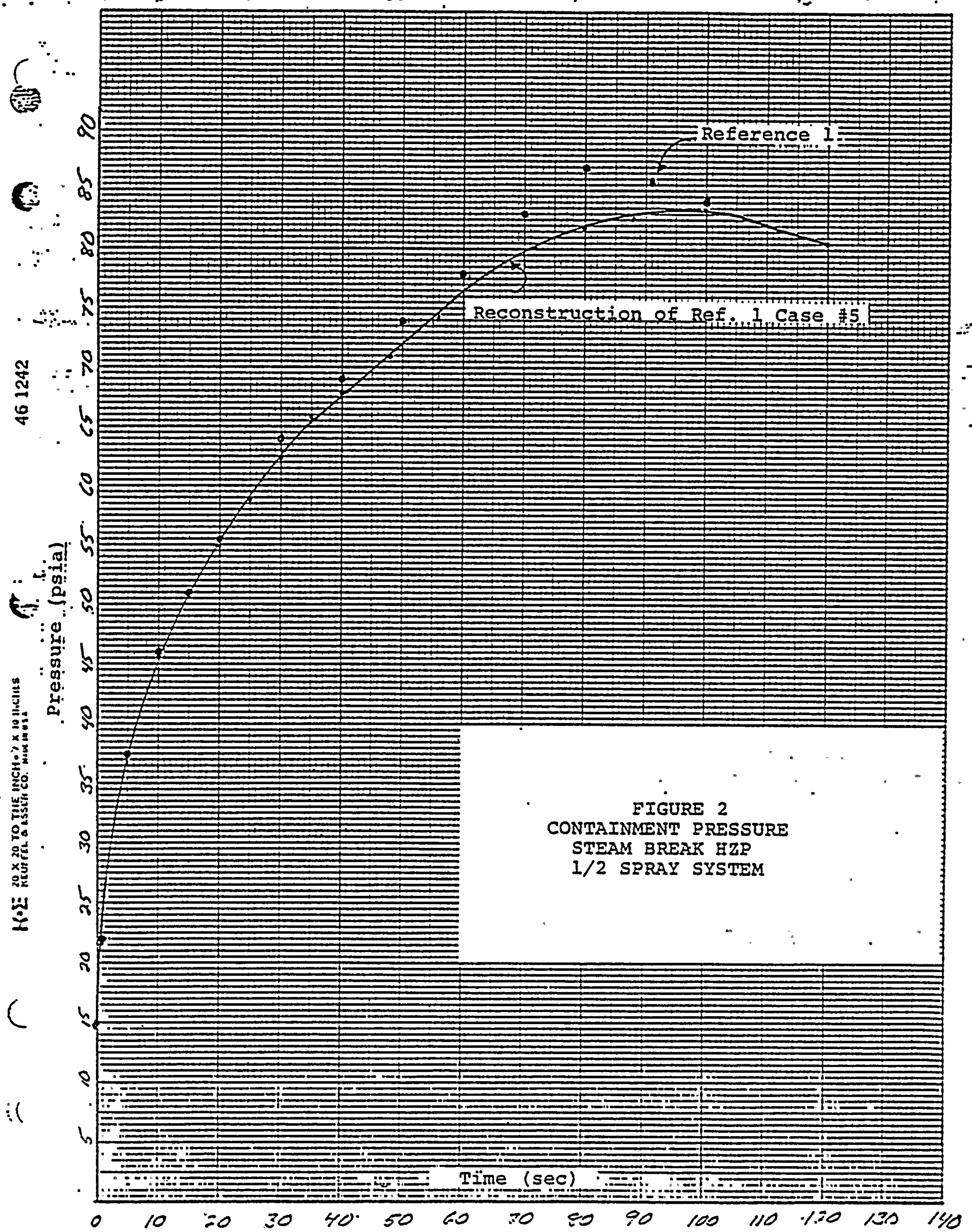

FIGURE 2
CONTAINMENT PRESSURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM

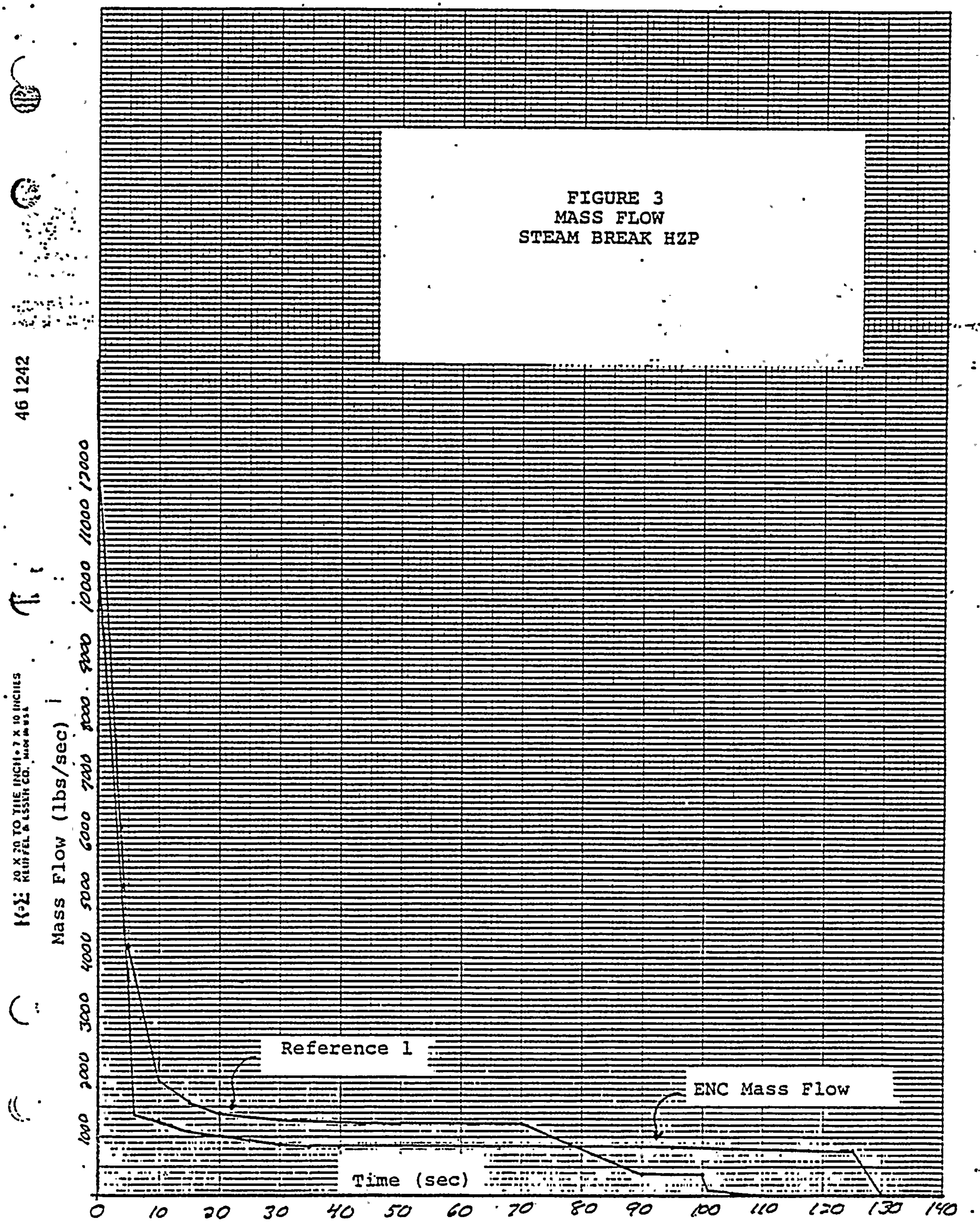

FIGURE 3
MASS FLOW
STEAM BREAK HZP

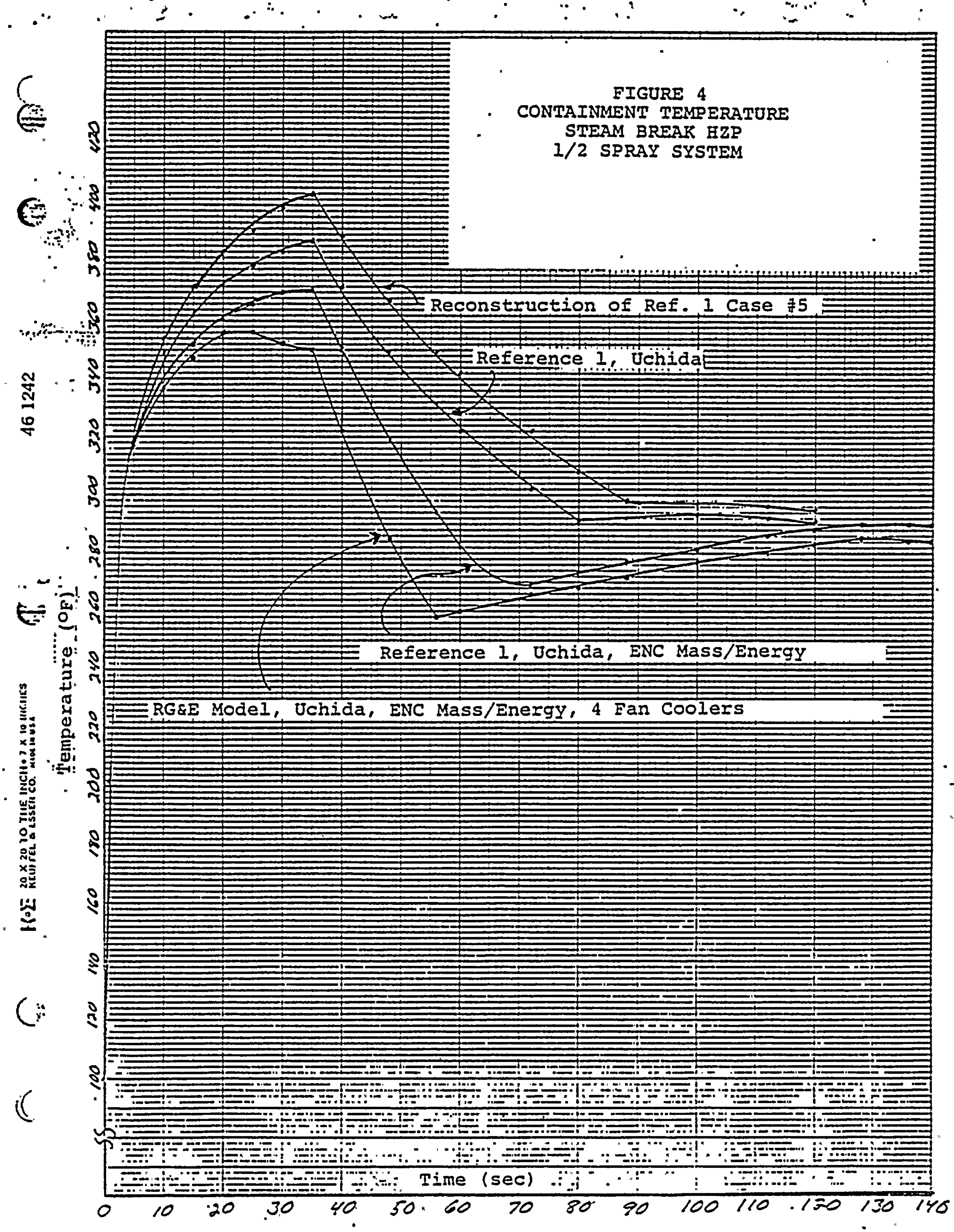
FIGURE 4
CONTAINMENT TEMPERATURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM
Reconstruction of Ref. 1 Case #5
Reference 1, Uchida
Reference 1, Uchida, ENC Mass/Energy
RG&E Model, Uchida, ENC Mass/Energy, 4 Fan Coolers
Temperature (°F)
Time (sec)
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
100
120
140
160
180
200
220
240
260
280
300
320
340
360
380
400
420
46 1242
K&E 20 X 20 TO THE INCH • 7 X 10 INCHES
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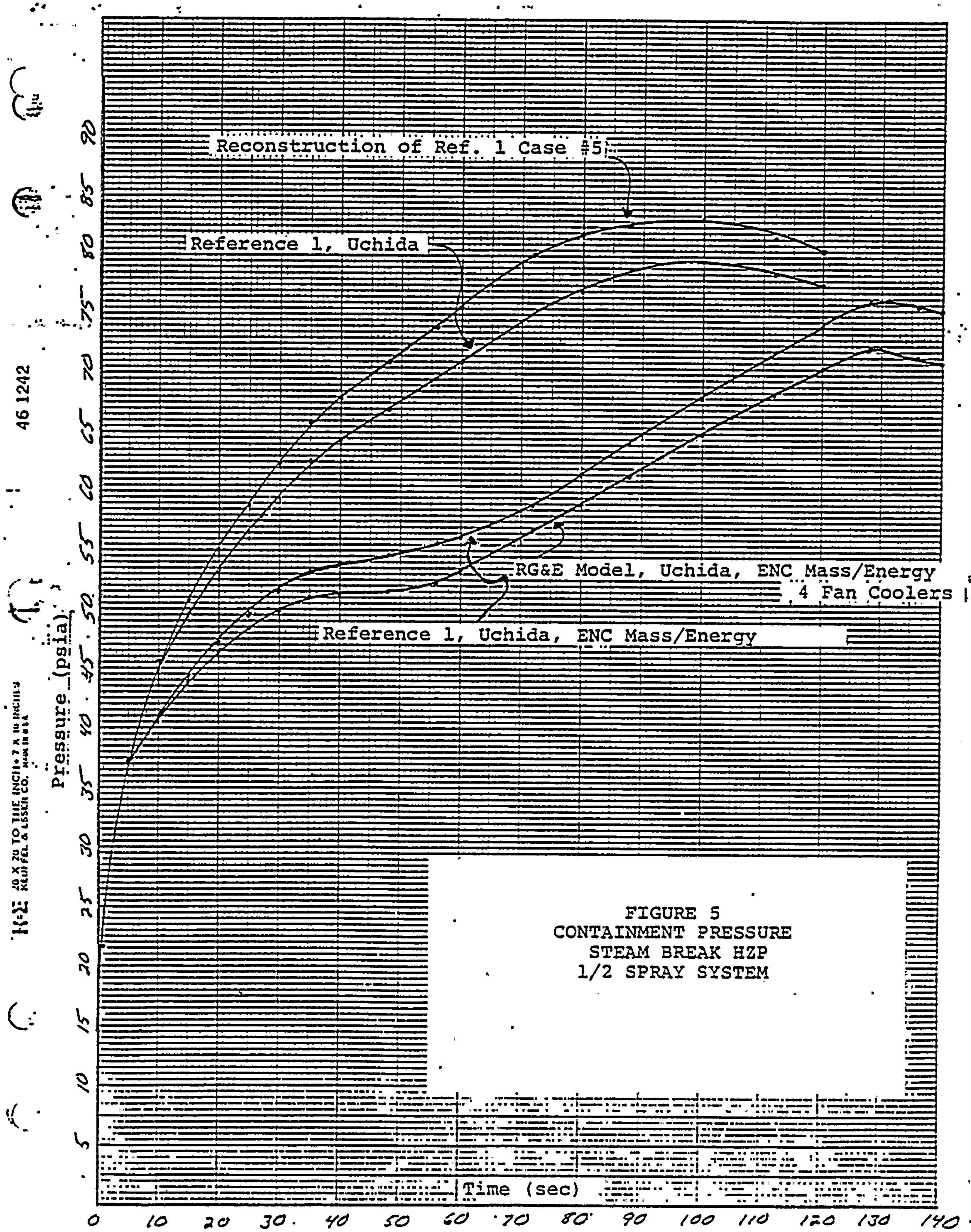

FIGURE 5
CONTAINMENT PRESSURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM

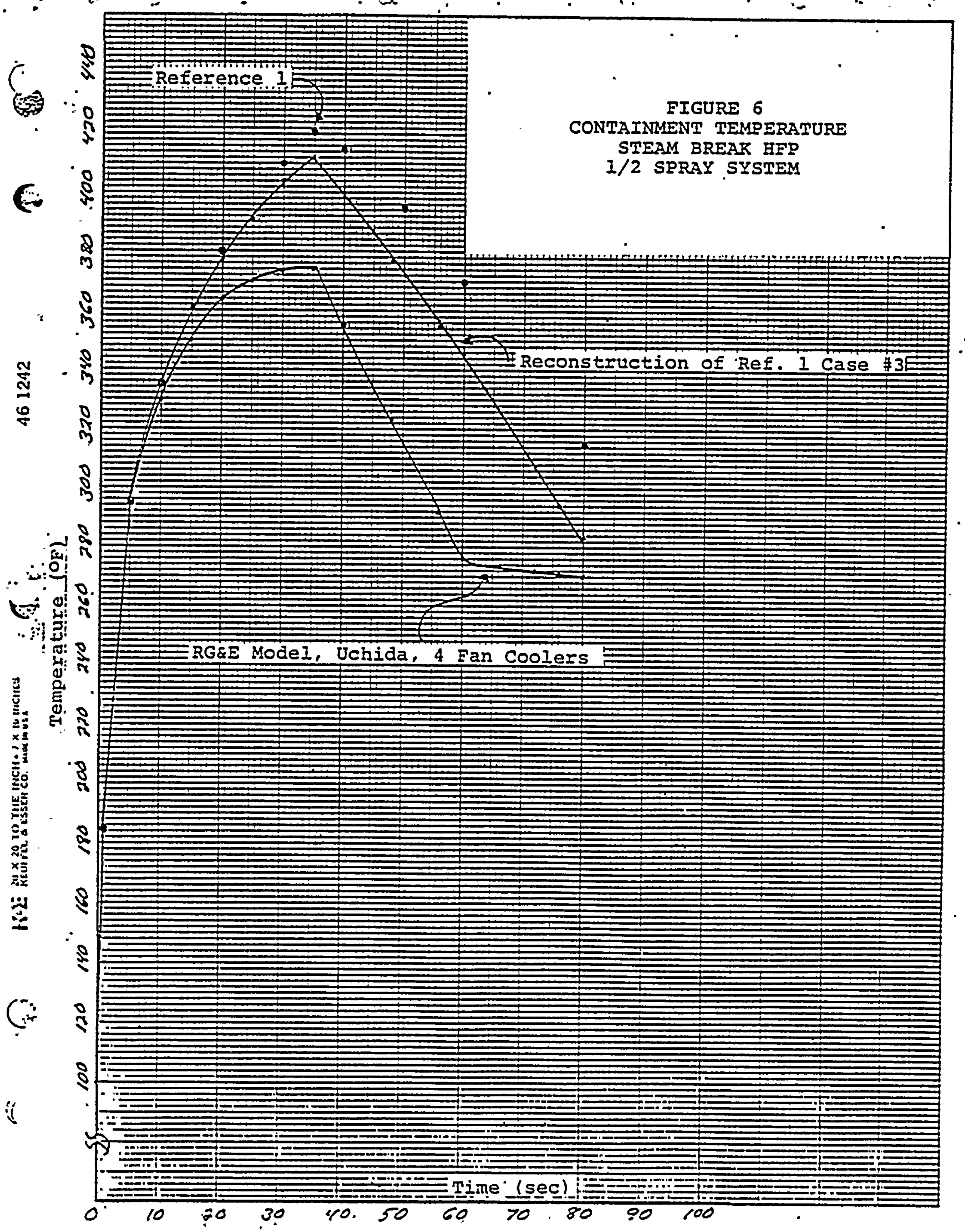

FIGURE 6
CONTAINMENT TEMPERATURE
STEAM BREAK HFP
1/2 SPRAY SYSTEM
Reference 1
Reconstruction of Ref. 1 Case #3
RG&E Model, Uchida, 4 Fan Coolers
Temperature (°F)
Time (sec)
0
10
20
30
40
50
60
70
80
90
100
440
420
400
380
360
340
320
300
280
260
240
220
200
180
160
140
120
100
46 1242
K+E 20 X 20 TO THE INCH • 7 X 10 INCHES
KEUFFEL & ESSER CO. MADE IN U.S.A.

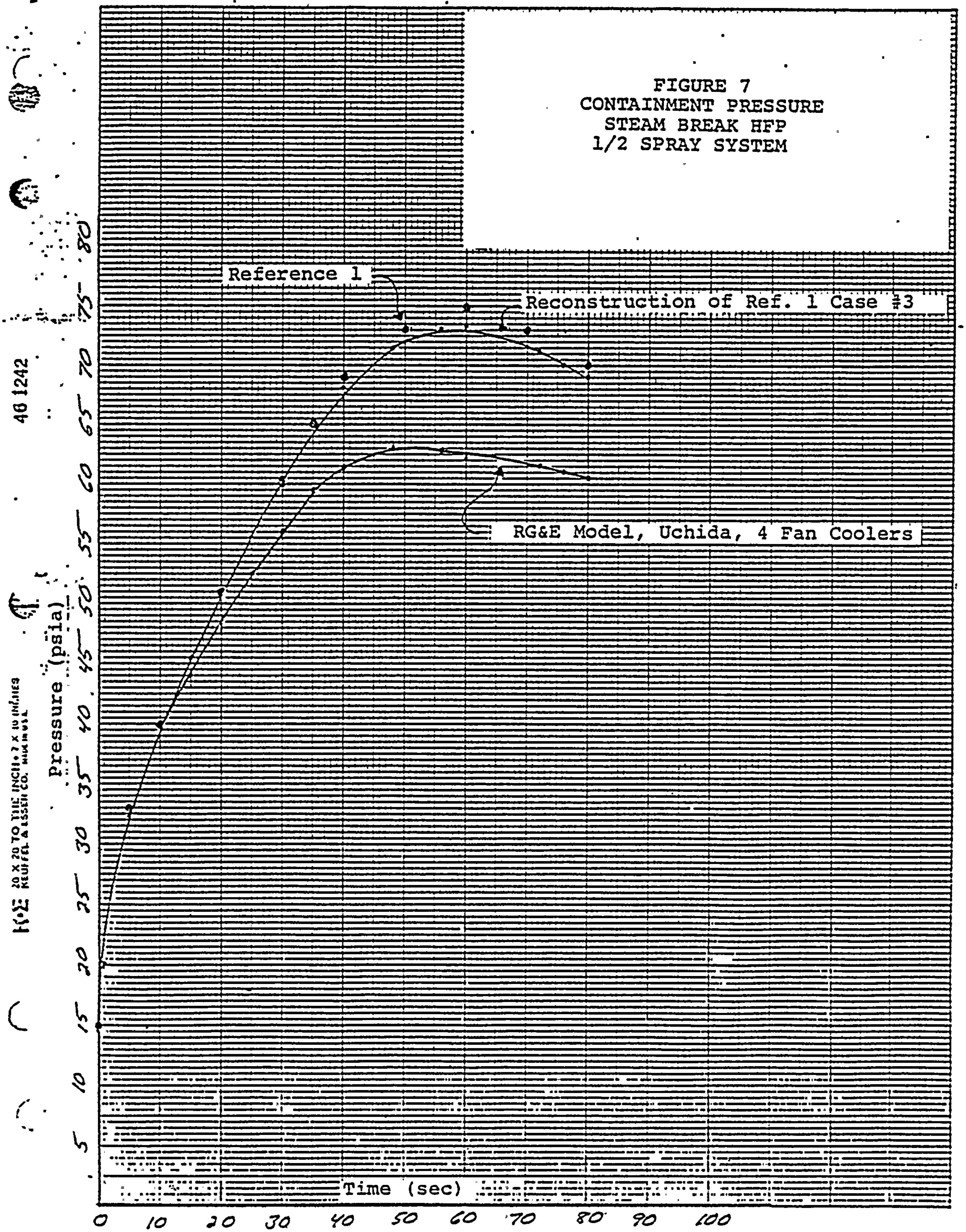
FIGURE 7
CONTAINMENT PRESSURE
STEAM BREAK HFP
1/2 SPRAY SYSTEM
Reference 1
Reconstruction of Ref. 1 Case #3
RG&E Model, Uchida, 4 Fan Coolers
Pressure (psia)
Time (sec)
0
10
20
30
40
50
60
70
80
90
100
5
10
15
20
25
30
35
40
45
50
55
60
65
70
75
80
46 1242
K&E 20 X 20 TO THE INCH • 7 X 10 INCHES
KEUFFEL & ESSER CO.

Attachment 1
to Appendix B.

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649

February 1, 1982

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: SEP Topics VI-2.D and VI-3
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Crutchfield:

This letter is in response to the draft evaluation of SEP Topic VI-2.D, "Mass and Energy Release Inside Containment" and SEP Topic VI-3, "Containment Pressure and Heat Removal Capability," which was transmitted by your letter dated November 3, 1981. We have reviewed the draft evaluation and have identified several conservatisms in the analysis for the loss of coolant accident (LOCA). These conservatisms and a qualitative discussion of their impact on the LOCA results, as well as a number of general comments on the evaluation, are dicussed in Attachment A. We also identified a number of conservatisms in the analysis for the main steam line break (MSLB). Because of the degree of conservatisms in the NRC evaluation, we performed a sensitivity study with the code CONTEMPT-EI/28B. This code is very similar to CONTEMPT-LT028, as discussed in Attachment B. Therefore, while this code has not been completely qualified for use as a licensing code, we believe that it is accurate and adequately represents Ginna. The sensitivity study, presented in Attachment B to this letter, confirmed the conservatism of the NRC results for MSLB.

Regarding the LOCA analysis, since the Ginna design basis pressure envelopes the NRC results, we conclude that the Ginna design basis pressure profile remains acceptable. The Ginna design basis temperature profile exceeds the NRC results except between 10,000 seconds and approximately 20,000 seconds after the design basis event. We propose that the Ginna design basis temperature profile remain as shown in the FSAR for times less than 10,000 seconds and be revised beyond 10,000 seconds as follows: from 10,000 seconds to 20,000 seconds, temperature = 250°F, beyond 20,000 seconds, temperature < 100°F. Since the containment temperature is already decreasing at this time, it is not considered

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that this revised profile would invalidate any conclusions drawn as part of our review of environmental qualification of electrical equipment, since the affected equipment temperatures would also be decreasing at this time from their peak temperatures.

Regarding the NRC steam break analyses, the NRC results for pressure and temperature exceed the Ginna design basis as revised above. However, as shown in Attachment B, acceptable results for containment pressure are obtained when more reasonable assumptions are used. It is our conclusion, based on Attachment B, that the Ginna design basis pressure profile exceeds the pressure profile calculated for main steam line break and, therefore, remains acceptable. We conclude that the temperature resulting from a steam line break in containment may exceed the Ginna design basis temperature profile, but that this is of no consequence due to the short duration of this exceedance and may therefore be ignored. This conclusion is based on NRC guidance provided in the Division of Operating Reactors (DOR) Guidelines which in turn is based, for example, on the short duration of the temperature spike, lower heat transfer coefficient, and the elevation of the steam lines relative to equipment. Thus, we conclude that the Ginna design basis temperature profile, as revised based on LOCA results discussed above, remains valid.

Very truly yours,

John E. Maier

John E. Maier

ATTACHMENT A

COMMENTS ON NRC SAFETY EVALUATION

1.	p. 5	Containment conditions resulting from a main steam break were assessed in a gross fashion in FSAR page 14.2.5-10. That analysis does not, however, comply with current criteria.
2.	p. 5	The LOCA analysis underestimated the effectiveness of the containment fan coolers by a heat removal rate of approximately 25 to 30 million BTU per hour. This resulted from an incorrect set of data being provided by RG&E to the NRC consultant (see also LER 81-022 transmitted by letter dated January 4, 1982 from John E. Maier, RG&E, to Ronald C. Haynes, NRC). We have estimated the impact of correcting the fan cooler heat removal rate to be on the order of a 1 to 2 psi reduction in peak containment pressure. This, therefore, is an additional conservation in the analysis.
3.	p. 6	The SER discussion of the result of a main steam line break should be revised to reflect the sensitivity study presented in Attachment B.
4.	p. 7	The last paragraph of the SER should be revised to reflect the conclusions presented in our letter transmitting this attachment.
5.	p.15	See our comment above regarding FSAR Section 14.2.5.
6.	p.16	The assumption of a constant containment pressure of 14.7 psia throughout the transient will result in an overestimate of the mass and energy release and, therefore, an overestimate of containment pressure and temperature.
7.	p.16	All information provided in conversations was also available on the docket.
8.	p.18	It is our understanding that only accumulator water was (conservatively) set at 272.9°F, not all ECCS flow. This should be clarified. If our understanding is not correct, then Table 1, which indicates RWST temperature, should be revised.
9.	p.21	The containment design pressure is incorrectly stated at the top of the page as being 74 psia; it is 75 psia.
10.	General	A number of other conservatisms are discussed in the LOCA evaluation. A more reasonable assessment would not require the level of conservatism employed here.
11.	p.21-25	See comments provided in Attachment B.

Attachment B

Containment Temperature/Pressure Following a Main Steam Line Break

Introduction

The purpose of this study is to provide a reevaluation of the containment conditions following a main steam line break. The first step will be to reconstruct the worst case containment temperature/pressure transients presented by the NRC in Reference 1 for a large steam break. Once the Reference 1 results have been reproduced, the assumptions necessary to reproduce those results can be evaluated. It may then be possible to remove some of the conservatism and calculate a more reasonable result.

Discussion

The containment temperatures and pressures presented in this study were calculated using the CONTEMPT-EI/28B computer code. The results presented in Reference 1 were calculated using CONTEMPT-LT/028. The CONTEMPT-EI/28B code is quite similar to the CONTEMPT-LT/028 code with changes which allow more user flexibility.

Hot Zero Power Case

The highest containment pressure was calculated in Reference 1 to occur for a large steam line break at HZP with failure of one spray train. The input for this case was run using CONTEMPT-EI/28B. Figure 1 and 2 illustrates the results of this run and points taken from Reference 1. The following peak temperature and pressure was obtained:

Reference 1 Case #5	85.8 psia @ 91 sec.
	413° @ 34 sec.
CONTEMPT-EI/28B	83.4 psia @ 99.8 sec.
	403.9° @ 35 sec.

While reproducing this case from the Reference 1 input one inconsistency was noted. Reference 1 states that spray was initiated 35 seconds after the setpoint at 30 psig was reached. In general, this pressure setpoint is reached at approximately 10 sec. Therefore, spray would start at approximately 45 sec. Since the temperature rise is terminated by spray, the peak temperature would occur when spray starts. All curves in Reference 1 illustrate peak temperature at approximately 35 sec. Therefore, it appears that the Reference 1 analysis neglects the time to reach the spray setpoint when actuating spray.

Using the reconstruction of Reference 1 case #5 as the base case, several cases were run to determine the sensitivity of containment temperature and pressure to various parameters. The results of these sensitivities are listed on Table 1 and are discussed below.

- Q/V, where Q is the total energy released to the time of peak containment pressure and is the containment volume, is a parameter associated with the Tagami film heat transfer correlation. This should represent total energy release to the time of peak pressure. A Q/V of ~ 165 results from the energy released to containment up to the time of peak pressure. However, a better approximation of the Reference 1 results can be obtained by reducing this parameter. The effect of reducing this parameter can be seen by comparing #1 and #3 on Table 1. Increasing Q/V results in increasing the film heat transfer coefficient. Changing Q/V from 87 to 165 results in approximately -1.0 psi pressure change and approximately -3.1^{o} change in temperature (#3 temperature - 400.8^{o} @ 35.0 sec.). Therefore, the Q/V term in Tagami may be doubled and still have only a small effect on containment temperature and pressure.

- The Uchida film heat transfer correlation has traditionally been used for steam breaks. When Uchida is used in the Ref. 1 Model a 3.7 psi pressure reduction and a 15.8^{o} temperature reduction results (#1 versus #4 on Table 1).

- Exxon Nuclear Company (ENC) mass and energy release for the most limiting large steam line break (Ref. 3) was used in the evaluation. The ENC mass and energy was normalized to the total mass in the broken steam generator at HZP plus the mass released from the unaffected steam generator until main steam isolation occurs. The normalized ENC mass and energy is illustrated on Figure 3 with the mass and energy release used in the Reference 1 analysis. The mass associated with auxilary feed was not included. The effect of auxiliary feed on peak containment pressure and temperature would be negligible since the mass added during the time frame of interest is a very small fraction of the secondary side inventory (<1%).

  The effect of using the normalized Exxon mass and energy release is a pressure reduction of 3.3 psi and a temperature reduction of 17.4^{o} (#4 and #5 on Table 1).

- A comparison between the RG&E containment model and the containment model used in Ref. 1 is illustrated on Table 2. The major difference between the models is the inclusion of the accumulators and ducting in the RG&E model. The area of the ducting is an assumed value based on values used by other plants, i.e.,

  Palisades = 20,072 @ 0.10 in.
  Indian Point = 22,000 @ 0.1382 in.
  Prairie Island = 22,000 @ 0.1875 in.

  Therefore, an assumption was made that Ginna had 20,000 sq. ft. @ 0.10 in.

  The CONTEMPT codes are sensitive to node spacing. A large spacing will result in lower surface temperatures which will result in removing too much energy from containment. The effect of node spacing can be seen by comparing #6 and #7. The effect of inclusion of accumulators and ducting is also illustrated on Table 1.

- In the process of doing this study it was determined that the heat removal capacity of the fan coolers used in the Reference 1 analysis was the capacity of one fan cooler at a service water temperature of 35°F. This corresponds to maximum cooling capability for one cooler. The minimum capability should be used in this analysis. The minimum capability is associated with the maximum service water temperature (80°F). Reference 4 presents a curve of heat removal versus containment pressure and equipment specifications presents the capacity at 120° and 286°.

  The following illustrates the heat removal capacity used in case #11 of Table 1: The capacities represent the minimum values of Reference 4 and equipment specifications; therefore, the values are conservative.

containment temperature °F	heat removal per fan MBTU/hr	total heat removal (4 fans) MBTU/hr
120	1.575	6.30
286	50.0	200.
308	54.72	218.9
320	56.52	226.0

  The effect of using the appropriate fan cooler capacity can be seen by comparing #10 and #11 on Table 1. This represents a 2.2 psi reduction in containment pressure and a 4.7° reduction in containment temperature.

- The effect of containment volume is illustrated on Table 1. Increasing the volume by 28,000 cu. ft. results in a 1.4 psi reduction in pressure and a 1.9° reduction in temperature. Since the gross volume of containment is approximately 1.13E6; 28,000 cu. ft. represents approximately 2.5% of the gross volume. Calculations show a net volume of approximately 1.037E6 cu. ft. Based on the FSAR the net volume of 9.72E5 cu. ft. represents a conservative small volume containing at least 3% margin. Therefore, a best estimate volume would be between 9.72E5 and 1.037E6 cu. ft. This represents available margin that was not used in this study.

Figures 4 and 5 illustrate the effect of Uchida, ENC mass and energy, and the RG&E containment heat sink model on the worst case containment response presented in Reference 1 (#1, #4, #5, and #11 of Table 1).

Using the RG&E containment heat sink model (volume = 9.72E5), ENC mass and energy release, Uchida correlation and fan cooler capacity of four fans results in:

72.4 psia @ 128.6 sec.
356.1° @ 20.3 sec.

<u>Full Power Case</u>

The highest containment temperature was calculated in Reference 1 to occur for a large steam break occurring at full power with failure of one spray train. The mass and energy release presented in Reference 1 for this case was coupled with the Reference 1 model discussed previously and containment temperature and pressure was calculated using the CONTEMPT-EI/28B code. The following results were obtained:

Reference 1 Case #3	75 psia @ 60 sec.
	421° @ 34 sec.
CONTEMPT-EI/28B	73.3 psia @ 59.0 sec.
	412.1° @ 35.0 sec.

The mass and energy release presented in Reference 1 was used with the RG&E containment heat sink model (volume = 9.72E5) previously described, Uchida correlation, and fan cooler capacity of four fans. This resulted in the following peak temperature and pressure:

63.2 psia @ 51.8 sec.
374.0° @ 32.0 sec.

The temperature and pressure versus time is illustrated on Figures 6 and 7 together with the reproduction of Reference 1 Case #3 using the CONTEMPT-EI/28B code.

Summary

In summary, the following compares the Reference 1 worst case with the comparable worst case calculated by RG&E as previously described:

case	Reference 1 Results	RG&E Results
Steam Break - HZP	85.8 psia @ 91 sec 413° @ 34 sec	72.4 psia @ 128.6 sec 356.1° @ 20.3 sec
Steam Break - HFP	75 psia @ 60 sec 421° @ 34 sec	63.2 psia @ 51.8 sec 374.0° @ 32.0 sec

For the worst case transients the calculated peak pressure is less than the design pressure for the Ginna containment (60 psig). The temperature is above the design temperature for Ginna containment (286°F). However, the temperature is exceeded only for a short period of time.

The RG&E calculations do not account for revaporization, entrainment, or best estimate containment volume. Inclusion of these effects would result in additional margin to design limits.

1. Revaporization - Reference 2 presents a discussion on revaporization. A temperature response is presented for a large steam line break using the Uchida heat transfer coefficient. When revaporization is used the temperature profile is reduced by approximately 40°. This would also result in a reduction in containment pressure.

2. Entrainment - In reality the steam flowing out of the break would not be dry steam but would contain some moisture. As the moisture content of the steam increases, the energy associated with the steam decreases; therefore, the energy added to containment decreases. This would result in a decrease in containment pressure and temperature. It has been estimated that the decrease in containment pressure and temperature resulting from accounting for entrainment would be similar to the decrease associated with revaporization.

3. Containment Volume - As previously described, a best estimate containment volume would be between 1.037E6 and 9.72E5 cu. ft. Increasing the containment volume used in the analysis would result in a slight pressure decrease.

References

1. NRC letter from D. M. Crutchfield to J. E. Maier, "Systematic Evaluation Program (SEP) for the R. E. Ginna Nuclear Power Plant - Evaluation Report on Topics VI-2.D and VI-3," November 3, 1981.

2. NRC letter from R. Tedesco to R. Mattson, V. Stello, and R. Boyd, "Best Estimate Evaluation for Environmental Qualification of Equipment Inside Containment Following A Main Steam Line Break," February 21, 1978.

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4. Ref. 1 Model, Uchida	79.7 @ 100.4 sec.	388.1 @ 34.8 sec.	-3.7	-15.8
5. Ref. 1 Model, Uchida, ENC mass/energy	76.4 @ 129.4 sec.	370.7 @ 32.6 sec.	-3.3	-17.4
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8. same as 7 with containment volume = 1.0E6	73.7 @ 129.2 sec.	363.5 @ 32.8 sec.	-1.4	- 1.9
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10. same as 9 with ducting	74.6 @ 129.0 sec.	360.8 @ 30.0 sec.	-0.3	- 4.6
11. same as 10 with 4 fan coolers	72.4 @ 128.6 sec.	356.1 @ 20.29 sec.	-2.2	-4.7

TABLE 2
CONTAINMENT MODEL

Reference 1		RG&E	
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Sump Walls	2,342 sq. ft.	same	2,342 sq. ft.
-		Sump Floor	297
			*2,639 sq. ft.
-		Basement Floor	*7,955 sq. ft.
Refueling Cavity Wall and Floor	6,400 sq. ft.	same	6,400 sq. ft.
Outside Refueling Cavity and S.G. Comp.	21,800 sq. ft.	same	21,800
Operating Floor	9,162 sq. ft.	same	9,162
			30,962
Intermediate Floor	6,170 sq. ft.	same but 2 X Area =	12,340 sq. ft.
1.5 in. Beams	9,174 sq. ft.	same	9,174 sq. ft.
1.0 in. Beams	5,016 sq. ft.	same	5,016 sq. ft.
0.5 in. Beams	8,586 sq. ft.	same	8,586 sq. ft.
Crane Supports	5,756 sq. ft.	same	5,756 sq. ft.
Grating etc.	7,000 sq. ft.	same	7,000 sq. ft.
-		Accumulators	1,756 sq. ft.
-		Ducting @ 0.10 in.	20,000 sq. ft.

*Surface assumed to be in contact with pool water.

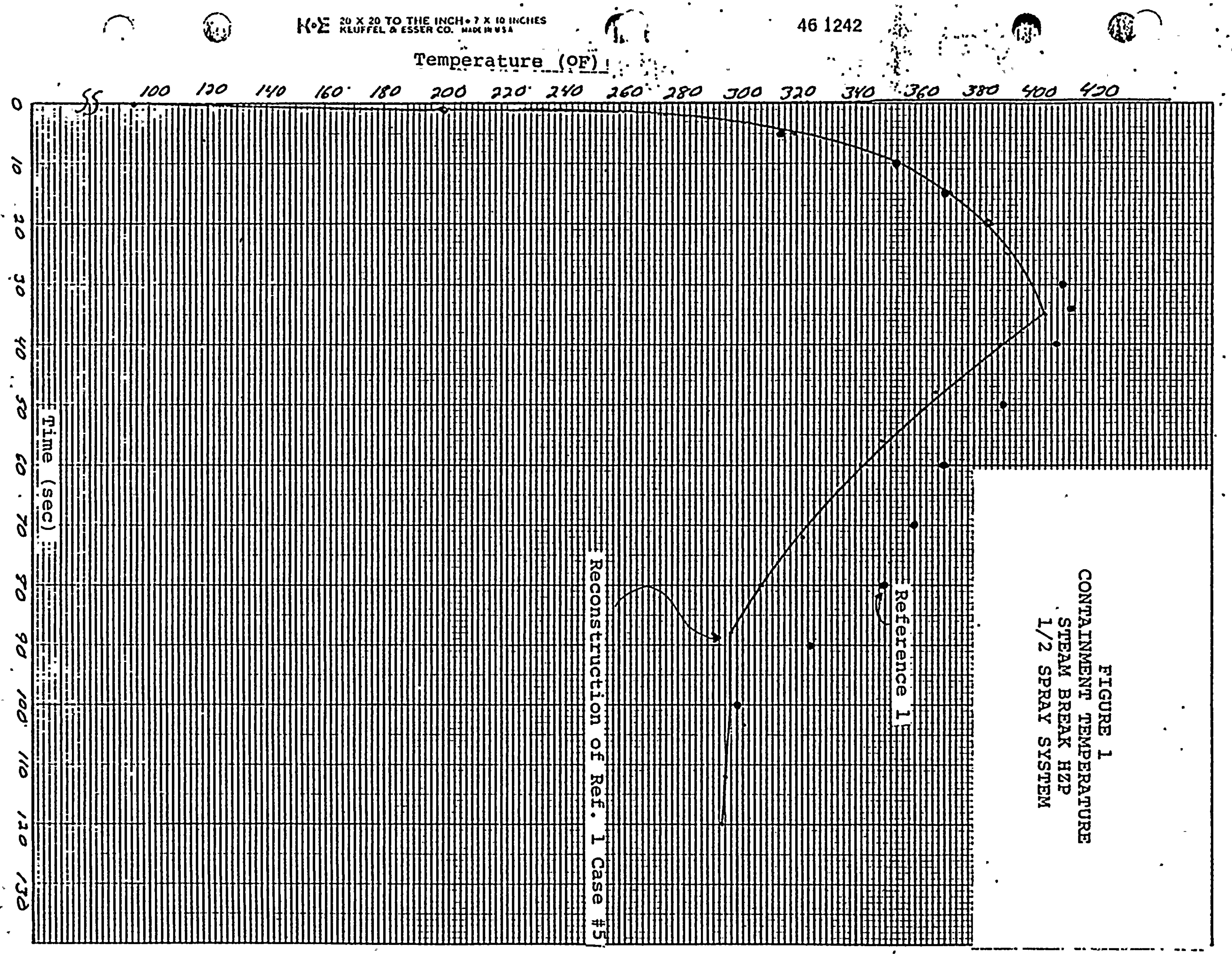


FIGURE 1
CONTAINMENT TEMPERATURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM

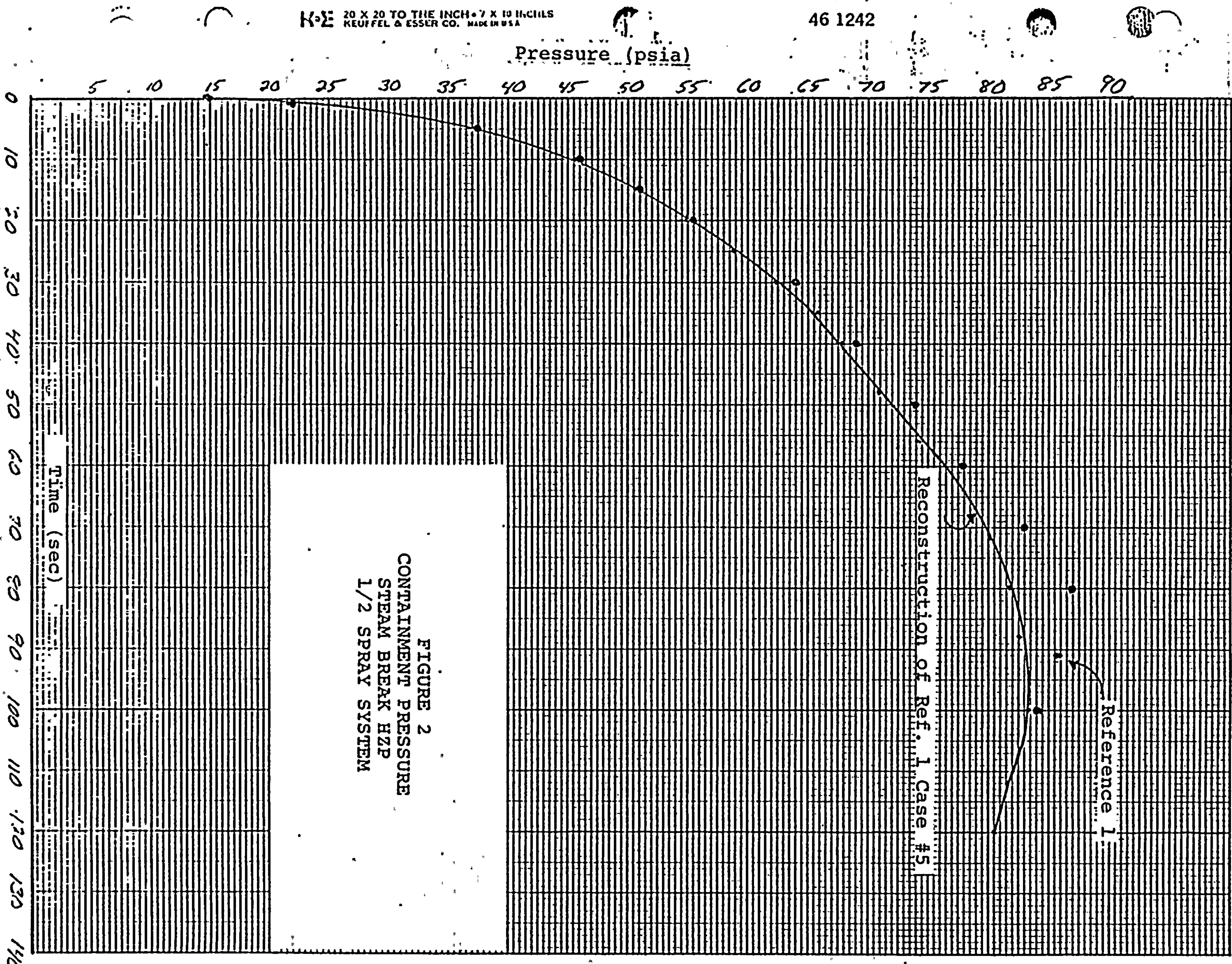
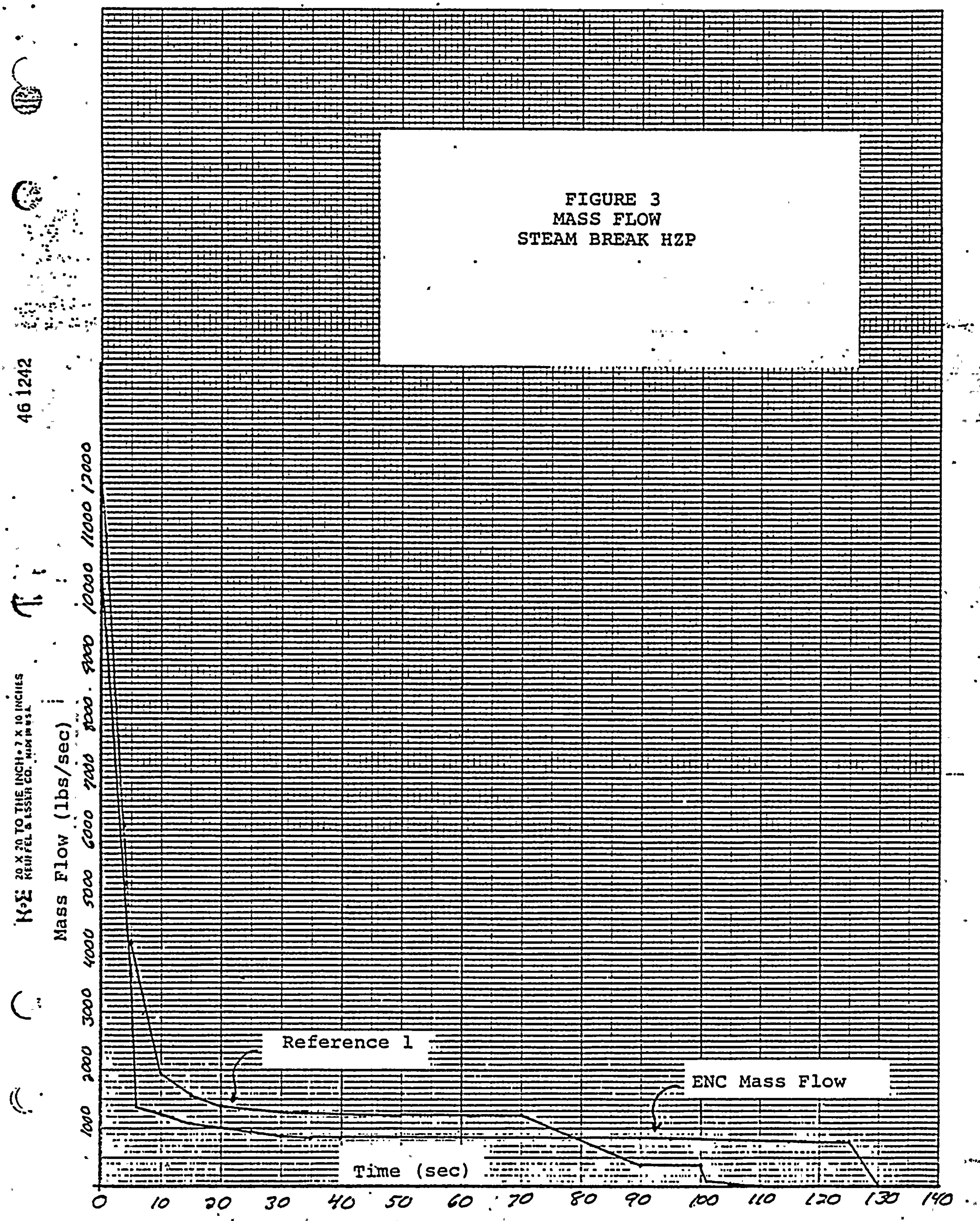


FIGURE 2
CONTAINMENT PRESSURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM

FIGURE 3
MASS FLOW
STEAM BREAK HZP



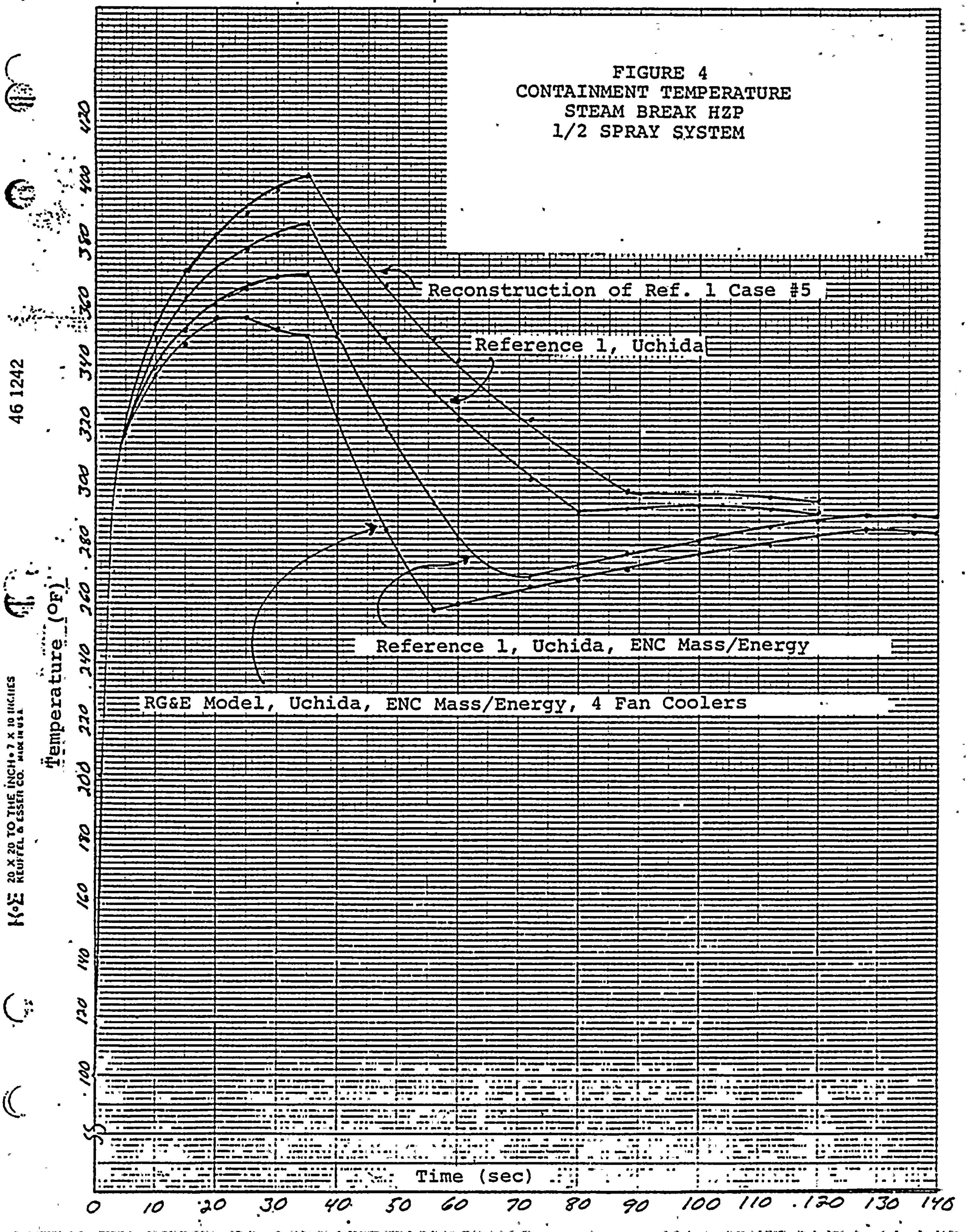


FIGURE 4
CONTAINMENT TEMPERATURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM
Reconstruction of Ref. 1 Case #5
Reference 1, Uchida
Reference 1, Uchida, ENC Mass/Energy
RG&E Model, Uchida, ENC Mass/Energy, 4 Fan Coolers
Temperature (°F)
Time (sec)
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
100
120
140
160
180
200
220
240
260
280
300
320
340
360
380
400
420
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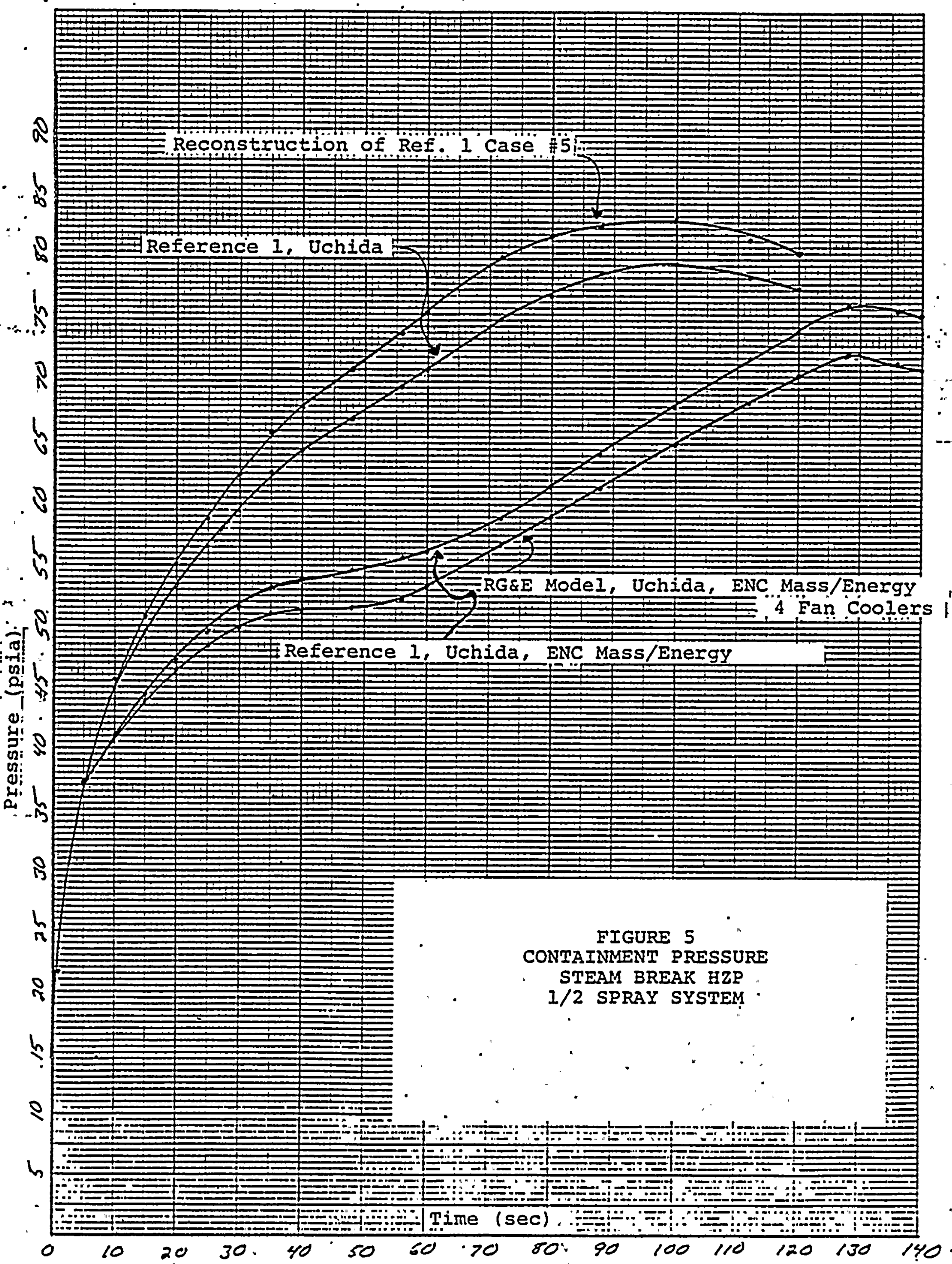

FIGURE 5
CONTAINMENT PRESSURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM

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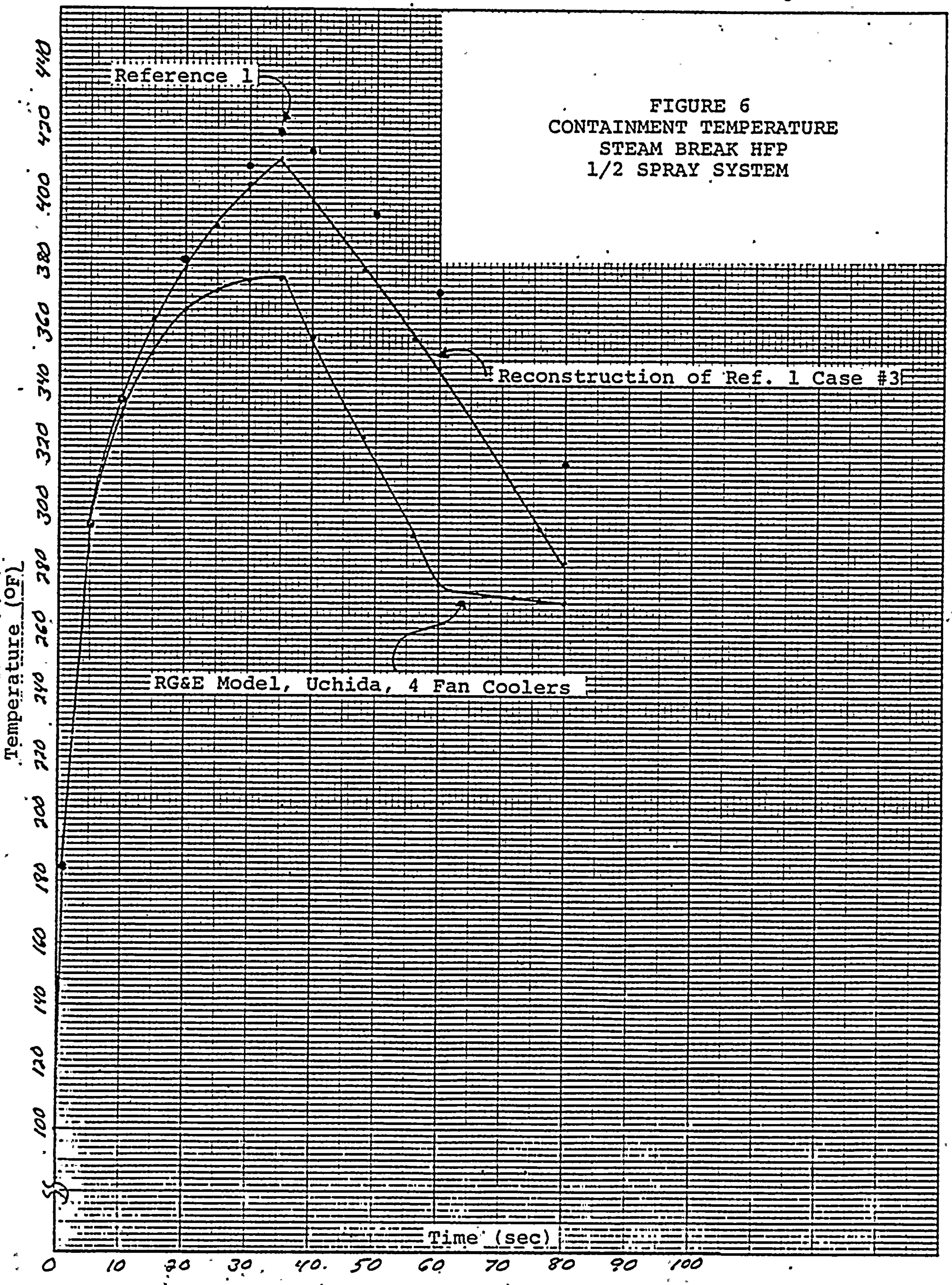

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K&E 20 X 20 TO THE INCH • 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

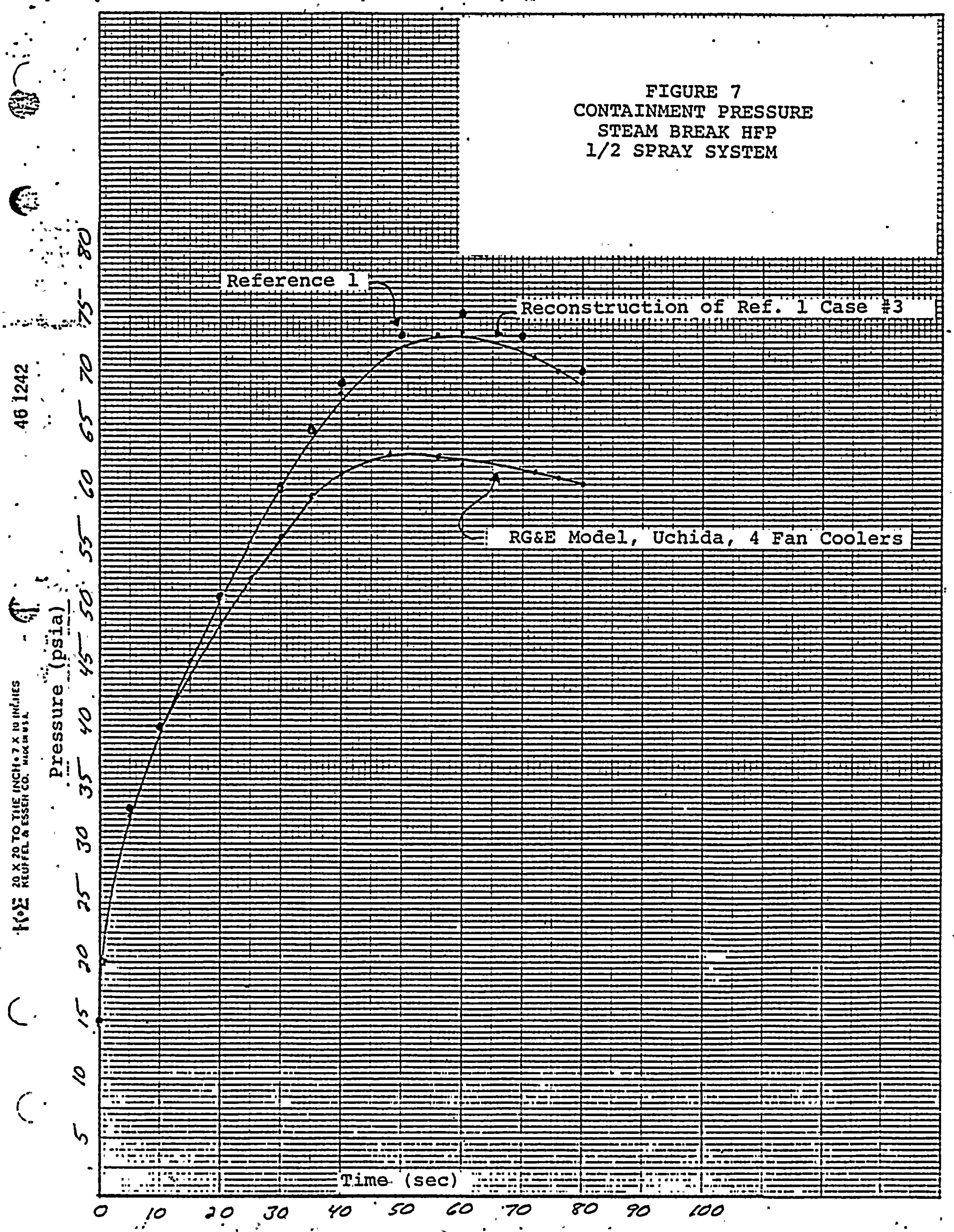
FIGURE 7
CONTAINMENT PRESSURE
STEAM BREAK HFP
1/2 SPRAY SYSTEM
Reference 1
Reconstruction of Ref. 1 Case #3
RG&E Model, Uchida, 4 Fan Coolers
Pressure (psia)
5
10
15
20
25
30
35
40
45
50
55
60
65
70
75
80
Time (sec)
0
10
20
30
40
50
60
70
80
90
100
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RECIP.NAME RECIPIENT AFFILIATION
CRUTCHFIELD,D. Operating Reactors Branch 5

SUBJECT: Forwards comments on NRC 811103 safety evaluation & draft evaluation of SEP Topics VI-2D, "Mass & Energy Release Inside Containment," & VI-3, "Containment Pressure & Heat Removal Capability."

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February 1, 1982

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Director of Nuclear Reactor Regulation
Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: SEP Topics VI-2.D and VI-3
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Crutchfield:

This letter is in response to the draft evaluation of SEP Topic VI-2.D, "Mass and Energy Release Inside Containment" and SEP Topic VI-3, "Containment Pressure and Heat Removal Capability," which was transmitted by your letter dated November 3, 1981. We have reviewed the draft evaluation and have identified several conservatisms in the analysis for the loss of coolant accident (LOCA). These conservatisms and a qualitative discussion of their impact on the LOCA results, as well as a number of general comments on the evaluation, are dicussed in Attachment A. We also identified a number of conservatisms in the analysis for the main steam line break (MSLB). Because of the degree of conservatisms in the NRC evaluation, we performed a sensitivity study with the code CONTEMPT-EI/28B. This code is very similar to CONTEMPT-LT028, as discussed in Attachment B. Therefore, while this code has not been completely qualified for use as a licensing code, we believe that it is accurate and adequately represents Ginna. The sensitivity study, presented in Attachment B to this letter, confirmed the conservatism of the NRC results for MSLB.

Regarding the LOCA analysis, since the Ginna design basis pressure envelopes the NRC results, we conclude that the Ginna design basis pressure profile remains acceptable. The Ginna design basis temperature profile exceeds the NRC results except between 10,000 seconds and approximately 20,000 seconds after the design basis event. We propose that the Ginna design basis temperature profile remain as shown in the FSAR for times less than 10,000 seconds and be revised beyond 10,000 seconds as follows: from 10,000 seconds to 20,000 seconds, temperature = 250°F, beyond 20,000 seconds, temperature < 100°F. Since the containment temperature is already decreasing at this time, it is not considered

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that this revised profile would invalidate any conclusions drawn as part of our review of environmental qualification of electrical equipment, since the affected equipment temperatures would also be decreasing at this time from their peak temperatures.

Regarding the NRC steam break analyses, the NRC results for pressure and temperature exceed the Ginna design basis as revised above. However, as shown in Attachment B, acceptable results for containment pressure are obtained when more reasonable assumptions are used. It is our conclusion, based on Attachment B, that the Ginna design basis pressure profile exceeds the pressure profile calculated for main steam line break and, therefore, remains acceptable. We conclude that the temperature resulting from a steam line break in containment may exceed the Ginna design basis temperature profile, but that this is of no consequence due to the short duration of this exceedance and may therefore be ignored. This conclusion is based on NRC guidance provided in the Division of Operating Reactors (DOR) Guidelines which in turn is based, for example, on the short duration of the temperature spike, lower heat transfer coefficient, and the elevation of the steam lines relative to equipment. Thus, we conclude that the Ginna design basis temperature profile, as revised based on LOCA results discussed above, remains valid.

Very truly yours,

John E. Maier

John E. Maier

ATTACHMENT A

COMMENTS ON NRC SAFETY EVALUATION

p. 5 Containment conditions resulting from a main steam break were assessed in a gross fashion in FSAR page 14.2.5-10. That analysis does not, however, comply with current criteria.

p. 5 The LOCA analysis underestimated the effectiveness of the containment fan coolers by a heat removal rate of approximately 25 to 30 million BTU per hour. This resulted from an incorrect set of data being provided by RG&E to the NRC consultant (see also LER 81-022 transmitted by letter dated January 4, 1982 from John E. Maier, RG&E, to Ronald C. Haynes, NRC). We have estimated the impact of correcting the fan cooler heat removal rate to be on the order of a 1 to 2 psi reduction in peak containment pressure. This, therefore, is an additional conservation in the analysis.

p. 6 The SER discussion of the result of a main steam line break should be revised to reflect the sensitivity study presented in Attachment B.

p. 7 The last paragraph of the SER should be revised to reflect the conclusions presented in our letter transmitting this attachment.

p.15 See our comment above regarding FSAR Section 14.2.5.

p.16 The assumption of a constant containment pressure of 14.7 psia throughout the transient will result in an overestimate of the mass and energy release and, therefore, an overestimate of containment pressure and temperature.

p.16 All information provided in conversations was also available on the docket.

p.18 It is our understanding that only accumulator water was (conservatively) set at 272.9°F, not all ECCS flow. This should be clarified. If our understanding is not correct, then Table 1, which indicates RWST temperature, should be revised.

p.21 The containment design pressure is incorrectly stated at the top of the page as being 74 psia; it is 75 psia.

General A number of other conservatisms are discussed in the LOCA evaluation. A more reasonable assessment would not require the level of conservatism employed here.

p.21-25 See comments provided in Attachment B.

Attachment B

Containment Temperature/Pressure Following a Main Steam Line Break

Introduction

The purpose of this study is to provide a reevaluation of the containment conditions following a main steam line break. The first step will be to reconstruct the worst case containment temperature/pressure transients presented by the NRC in Reference 1 for a large steam break. Once the Reference 1 results have been reproduced, the assumptions necessary to reproduce those results can be evaluated. It may then be possible to remove some of the conservatism and calculate a more reasonable result.

Discussion

The containment temperatures and pressures presented in this study were calculated using the CONTEMPT-EI/28B computer code. The results presented in Reference 1 were calculated using CONTEMPT-LT/028. The CONTEMPT-EI/28B code is quite similar to the CONTEMPT-LT/028 code with changes which allow more user flexibility.

Hot Zero Power Case

The highest containment pressure was calculated in Reference 1 to occur for a large steam line break at HZP with failure of one spray train. The input for this case was run using CONTEMPT-EI/28B. Figure 1 and 2 illustrates the results of this run and points taken from Reference 1. The following peak temperature and pressure was obtained.

Reference 1 Case #5	85.8 psia @ 91 sec.
	413° @ 34 sec.
CONTEMPT-EI/28B	83.4 psia @ 99.8 sec.
	403.9° @ 35 sec.

While reproducing this case from the Reference 1 input one inconsistency was noted. Reference 1 states that spray was initiated 35 seconds after the setpoint at 30 psig was reached. In general, this pressure setpoint is reached at approximately 10 sec. Therefore, spray would start at approximately 45 sec. Since the temperature rise is terminated by spray, the peak temperature would occur when spray starts. All curves in Reference 1 illustrate peak temperature at approximately 35 sec. Therefore, it appears that the Reference 1 analysis neglects the time to reach the spray setpoint when actuating spray.

Using the reconstruction of Reference 1 case #5 as the base case, several cases were run to determine the sensitivity of containment temperature and pressure to various parameters. The results of these sensitivities are listed on Table 1 and are discussed below.

- Q/V, where Q is the total energy released to the time of peak containment pressure and is the containment volume, is a parameter associated with the Tagami film heat transfer correlation. This should represent total energy release to the time of peak pressure. A Q/V of ~ 165 results from the energy released to containment up to the time of peak pressure. However, a better approximation of the Reference 1 results can be obtained by reducing this parameter. The effect of reducing this parameter can be seen by comparing #1 and #3 on Table 1. Increasing Q/V results in increasing the film heat transfer coefficient. Changing Q/V from 87 to 165 results in approximately -1.0 psi pressure change and approximately -3.1° change in temperature (#3 temperature - 400.8° @ 35.0 sec.). Therefore, the Q/V term in Tagami may be doubled and still have only a small effect on containment temperature and pressure.

- The Uchida film heat transfer correlation has traditionally been used for steam breaks. When Uchida is used in the Ref. 1 Model a 3.7 psi pressure reduction and a 15.8° temperature reduction results (#1 versus #4 on Table 1).

- Exxon Nuclear Company (ENC) mass and energy release for the most limiting large steam line break (Ref. 3) was used in the evaluation. The ENC mass and energy was normalized to the total mass in the broken steam generator at HZP plus the mass released from the unaffected steam generator until main steam isolation occurs. The normalized ENC mass and energy is illustrated on Figure 3 with the mass and energy release used in the Reference 1 analysis. The mass associated with auxilary feed was not included. The effect of auxiliary feed on peak containment pressure and temperature would be negligible since the mass added during the time frame of interest is a very small fraction of the secondary side inventory (<1%).

  The effect of using the normalized Exxon mass and energy release is a pressure reduction of 3.3 psi and a temperature reduction of 17.4° (#4 and #5 on Table 1).

- A comparison between the RG&E containment model and the containment model used in Ref. 1 is illustrated on Table 2. The major difference between the models is the inclusion of the accumulators and ducting in the RG&E model. The area of the ducting is an assumed value based on values used by other plants, i.e.,

  Palisades = 20,072 @ 0.10 in.
  Indian Point = 22,000 @ 0.1382 in.
  Prairie Island = 22,000 @ 0.1875 in.

  Therefore, an assumption was made that Ginna had 20,000 sq. ft. @ 0.10 in.

  The CONTEMPT codes are sensitive to node spacing. A large spacing will result in lower surface temperatures which will result in removing too much energy from containment. The effect of node spacing can be seen by comparing #6 and #7. The effect of inclusion of accumulators and ducting is also illustrated on Table 1.

- In the process of doing this study it was determined that the heat removal capacity of the fan coolers used in the Reference 1 analysis was the capacity of one fan cooler at a service water temperature of 35°F. This corresponds to maximum cooling capability for one cooler. The minimum capability should be used in this analysis. The minimum capability is associated with the maximum service water temperature (80°F). Reference 4 presents a curve of heat removal versus containment pressure and equipment specifications presents the capacity at 120° and 286°.

  The following illustrates the heat removal capacity used in case #11 of Table 1: The capacities represent the minimum values of Reference 4 and equipment specifications; therefore, the values are conservative.

containment temperature °F	heat removal per fan MBTU/hr	total heat removal (4 fans) MBTU/hr
120	1.575	6.30
286	50.0	200.
308	54.72	218.9
320	56.52	226.0

  The effect of using the appropriate fan cooler capacity can be seen by comparing #10 and #11 on Table 1. This represents a 2.2 psi reduction in containment pressure and a 4.7° reduction in containment temperature.

- The effect of containment volume is illustrated on Table 1. Increasing the volume by 28,000 cu. ft. results in a 1.4 psi reduction in pressure and a 1.9° reduction in temperature. Since the gross volume of containment is approximately 1.13E6; 28,000 cu. ft. represents approximately 2.5% of the gross volume. Calculations show a net volume of approximately 1.037E6 cu. ft. Based on the FSAR the net volume of 9.72E5 cu. ft. represents a conservative small volume containing at least 3% margin. Therefore, a best estimate volume would be between 9.72E5 and 1.037E6 cu. ft. This represents available margin that was not used in this study.

Figures 4 and 5 illustrate the effect of Uchida, ENC mass and energy, and the RG&E containment heat sink model on the worst case containment response presented in Reference 1 (#1, #4, #5, and #11 of Table 1).

Using the RG&E containment heat sink model (volume = 9.72E5), ENC mass and energy release, Uchida correlation and fan cooler capacity of four fans results in:

72.4 psia @ 128.6 sec.
356.1° @ 20.3 sec.

<u>Full Power Case</u>

The highest containment temperature was calculated in Reference 1 to occur for a large steam break occurring at full power with failure of one spray train. The mass and energy release presented in Reference 1 for this case was coupled with the Reference 1 model discussed previously and containment temperature and pressure was calculated using the CONTEMPT-EI/28B code. The following results were obtained:

Reference 1 Case #3	75 psia @ 60 sec. 421° @ 34 sec.
CONTEMPT-EI/28B	73.3 psia @ 59.0 sec. 412.1° @ 35.0 sec.

The mass and energy release presented in Reference 1 was used with the RG&E containment heat sink model (volume = 9.72E5) previously described, Uchida correlation, and fan cooler capacity of four fans. This resulted in the following peak temperature and pressure:

63.2 psia @ 51.8 sec.
374.0° @ 32.0 sec.

The temperature and pressure versus time is illustrated on Figures 6 and 7 together with the reproduction of Reference 1 Case #3 using the CONTEMPT-EI/28B code.

Summary

In summary, the following compares the Reference 1 worst case with the comparable worst case calculated by RG&E as previously described:

case	Reference 1 Results	RG&E Results
Steam Break - HZP	85.8 psia @ 91 sec 413° @ 34 sec	72.4 psia @ 128.6 sec 356.1° @ 20.3 sec
Steam Break - HFP	75 psia @ 60 sec 421° @ 34 sec	63.2 psia @ 51.8 sec 374.0° @ 32.0 sec

For the worst case transients the calculated peak pressure is less than the design pressure for the Ginna containment (60 psig). The temperature is above the design temperature for Ginna containment (286°F). However, the temperature is exceeded only for a short period of time.

The RG&E calculations do not account for revaporization, entrainment, or best estimate containment volume. Inclusion of these effects would result in additional margin to design limits.

1. Revaporization - Reference 2 presents a discussion on revaporization. A temperature response is presented for a large steam line break using the Uchida heat transfer coefficient. When revaporization is used the temperature profile is reduced by approximately 40°. This would also result in a reduction in containment pressure.

2. Entrainment - In reality the steam flowing out of the break would not be dry steam but would contain some moisture. As the moisture content of the steam increases, the energy associated with the steam decreases; therefore, the energy added to containment decreases. This would result in a decrease in containment pressure and temperature. It has been estimated that the decrease in containment pressure and temperature resulting from accounting for entrainment would be similar to the decrease associated with revaporization.

3. Containment Volume - As previously described, a best estimate containment volume would be between 1.037E6 and 9.72E5 cu. ft. Increasing the containment volume used in the analysis would result in a slight pressure decrease.

References

1. NRC letter from D. M. Crutchfield to J. E. Maier, "Systematic Evaluation Program (SEP) for the R. E. Ginna Nuclear Power Plant - Evaluation Report on Topics VI-2.D and VI-3," November 3, 1981.

2. NRC letter from R. Tedesco to R. Mattson, V. Stello, and R. Boyd, "Best Estimate Evaluation for Environmental Qualification of Equipment Inside Containment Following A Main Steam Line Break," February 21, 1978.

3. Exxon Report XN-NF-77-40 Supplement 1, "Plant Transient Analysis for the R. E. Ginna Unit 1 Nuclear Power Plant," March 1980.

4. Rochester Gas and Electric Corporation, R.E. Ginna, FSAR.

TABLE 1
SENSITIVITY STUDY
Hot Zero Power Case

Case	Peak Pressure (psia)	Peak Temperature (psia)	P	T
1. Ref. 1 Model; Q/V = 87	83.4 @ 99.8 sec.	403.9 @ 35.0 sec.	-	-
2. Ref. 1 Model; Q/V = 87, no fan coolers	83.7 @ 99.4 sec.	404.8 @ 35.0 sec.	+0.3	+ 0.9
3. Ref. 1 Model, Q/V = 165, spray @ 45 sec.	82.4 @ 84.4 sec.	408.0 @ 44.6 sec.	-1.0	+ 4.1
4. Ref. 1 Model; Uchida	79.7 @ 100.4 sec.	388.1 @ 34.8 sec.	-3.7	-15.8
5. Ref. 1 Model; Uchida, ENC mass/energy	76.4 @ 129.4 sec.	370.7 @ 32.6 sec.	-3.3	-17.4
6. RGE Model (large spacing), Uchida, ENC mass/energy	73.5 @ 129 sec.	359.7 @ 32.8 sec.	-2.9	-11.0
7. RGE Model (small spacing); Uchida, ENC mass/energy	75.1 @ 129.2 sec.	365.4 @ 32.6 sec.	+1.6	+ 5.7
8. same as 7 with containment volume = 1.0E6	73.7 @ 129.2 sec.	363.5 @ 32.8 sec.	-1.4	- 1.9
9. same as 7 with accumulators	74.9 @ 128.8 sec.	365.4 @ 34.6 sec.	-0.2	- 0.0
10. same as 9 with ducting	74.6 @ 129.0 sec.	360.8 @ 30.0 sec.	-0.3	- 4.6
11. same as 10 with 4 fan coolers	72.4 @ 128.6 sec.	356.1 @ 20.29 sec.	-2.2	-4.7

TABLE 2
CONTAINMENT MODEL

Reference 1		RG&E	
Containment Volume	9.72E5 cu. ft.	Containment Volume	9.72E5 cu. ft.
Insulated Dome and Walls	36,181 sq. ft.	same	36,181 sq. ft.
Uninsulated Dome and Walls	12,474 sq. ft.	same	12,474 sq. ft.
Sump Walls	2,342 sq. ft.	same	2,342 sq. ft.
-		Sump Floor	297
			*2,639 sq. ft.
-		Basement Floor	*7,955 sq. ft.
Refueling Cavity Wall and Floor	6,400 sq. ft.	same	6,400 sq. ft.
Outside Refueling Cavity and S.G. Comp.	21,800 sq. ft.	same	21,800
Operating Floor	9,162 sq. ft.	same	9,162
			30,962
Intermediate Floor	6,170 sq. ft.	same but 2 X Area =	12,340 sq. ft.
1.5 in. Beams	9,174 sq. ft.	same	9,174 sq. ft.
1.0 in. Beams	5,016 sq. ft.	same	5,016 sq. ft.
0.5 in. Beams	8,586 sq. ft.	same	8,586 sq. ft.
Crane Supports	5,756 sq. ft.	same	5,756 sq. ft.
Grating etc.	7,000 sq. ft.	same	7,000 sq. ft.
-		Accumulators	1,756 sq. ft.
-		Ducting @ 0.10 in.	20,000 sq. ft.

*Surface assumed to be in contact with pool water.

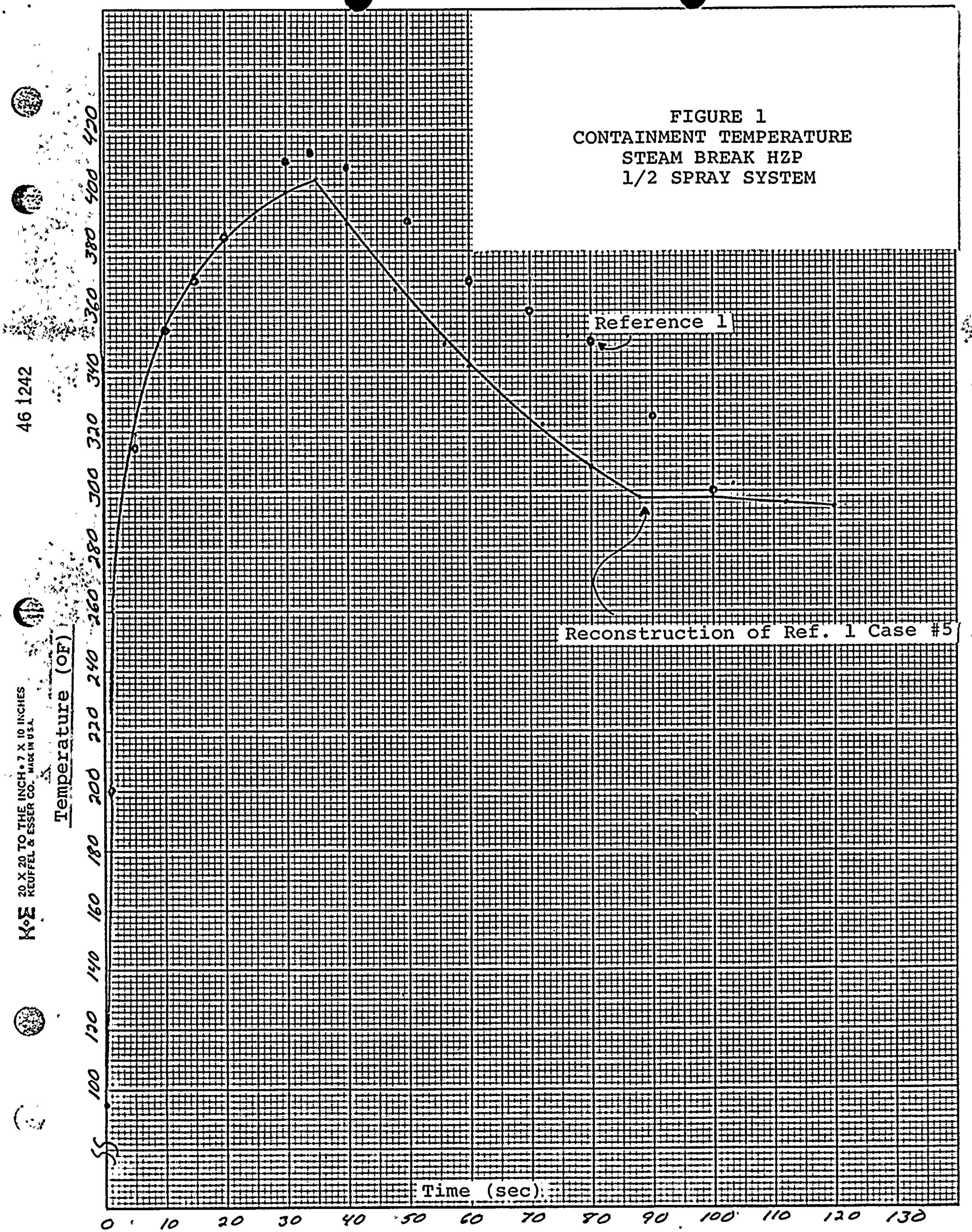
FIGURE 1
CONTAINMENT TEMPERATURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM
Reference 1
Reconstruction of Ref. 1 Case #5
Temperature (OF)
Time (sec)
0
10
20
30
40
50
60
70
80
90
100
110
120
130
100
120
140
160
180
200
220
240
260
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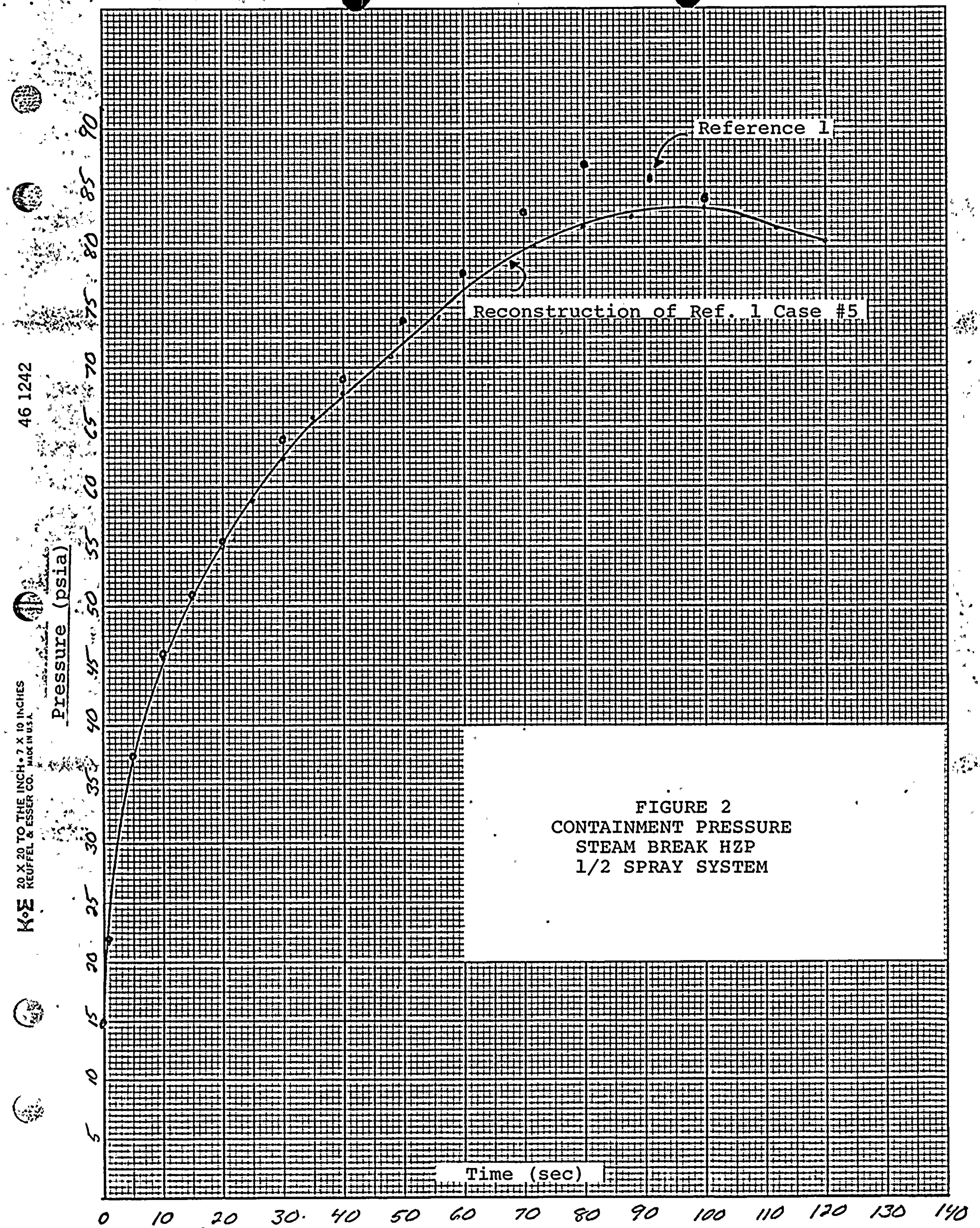

FIGURE 2
CONTAINMENT PRESSURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM

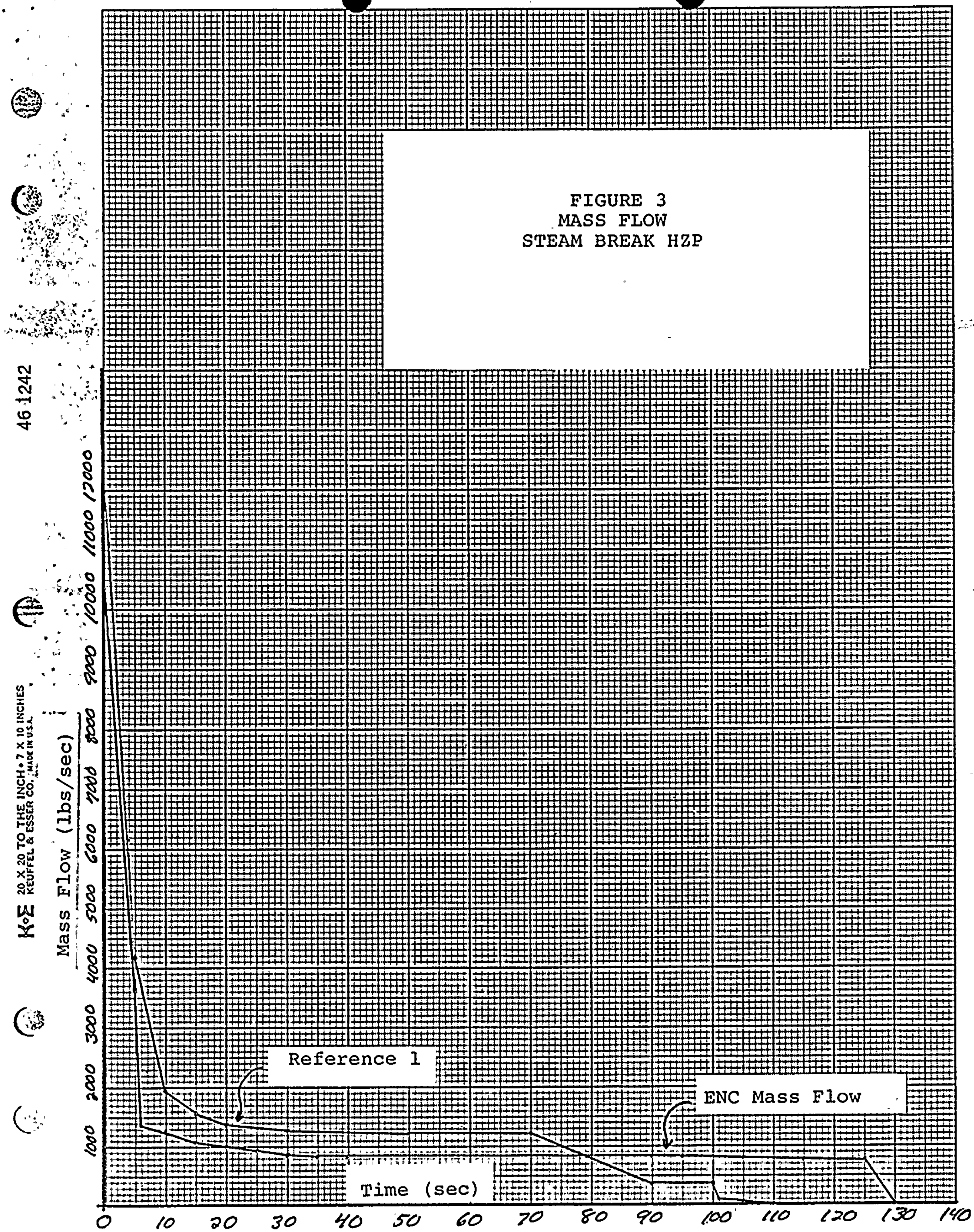

FIGURE 3
MASS FLOW
STEAM BREAK HZP
Reference 1
ENC Mass Flow
Mass Flow (lbs/sec)
Time (sec)
1000
2000
3000
4000
5000
6000
7000
8000
9000
10000
11000
12000
0
10
20
30
40
50
60
70
80
90
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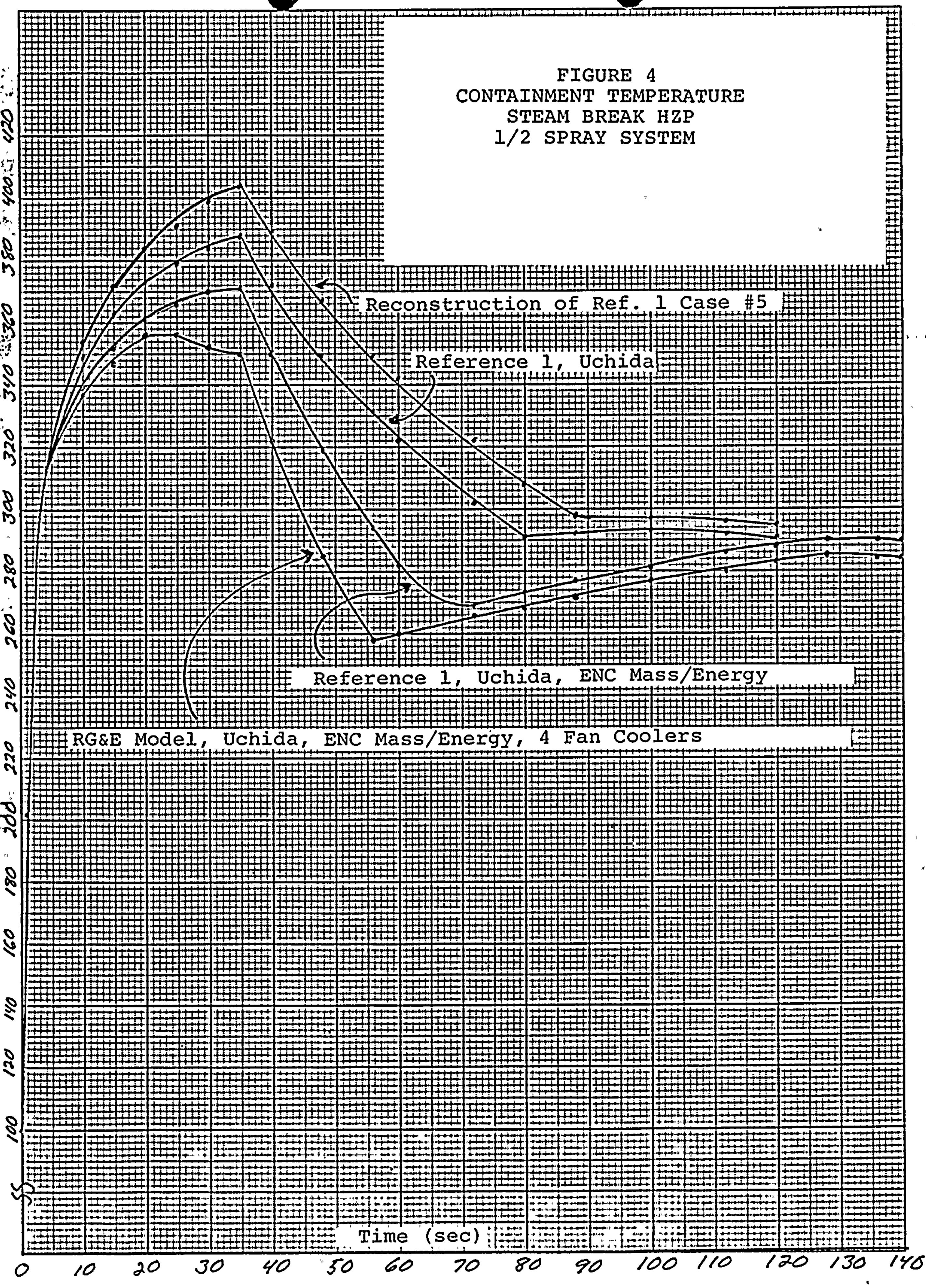

FIGURE 4
CONTAINMENT TEMPERATURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM

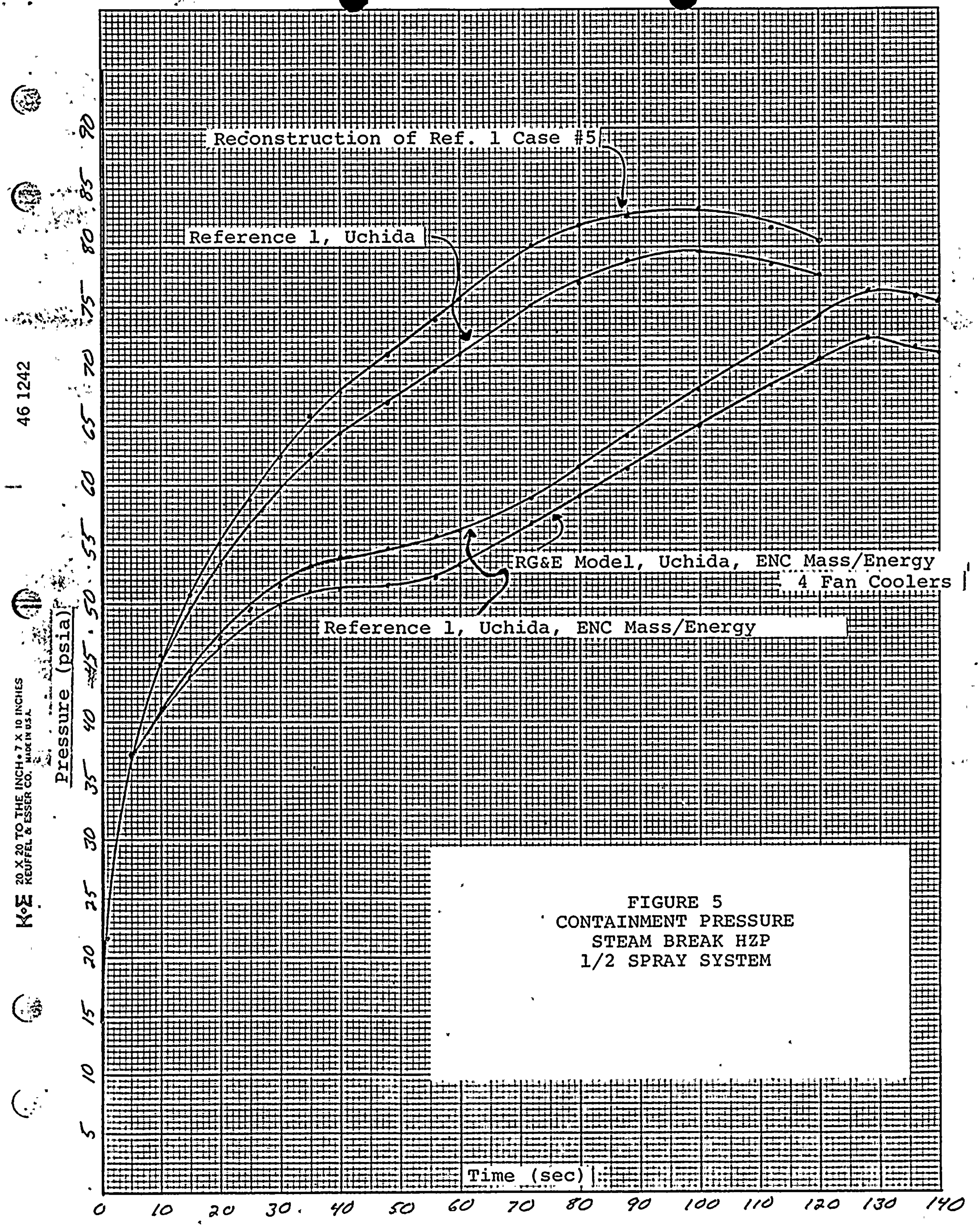

FIGURE 5
CONTAINMENT PRESSURE
STEAM BREAK HZP
1/2 SPRAY SYSTEM

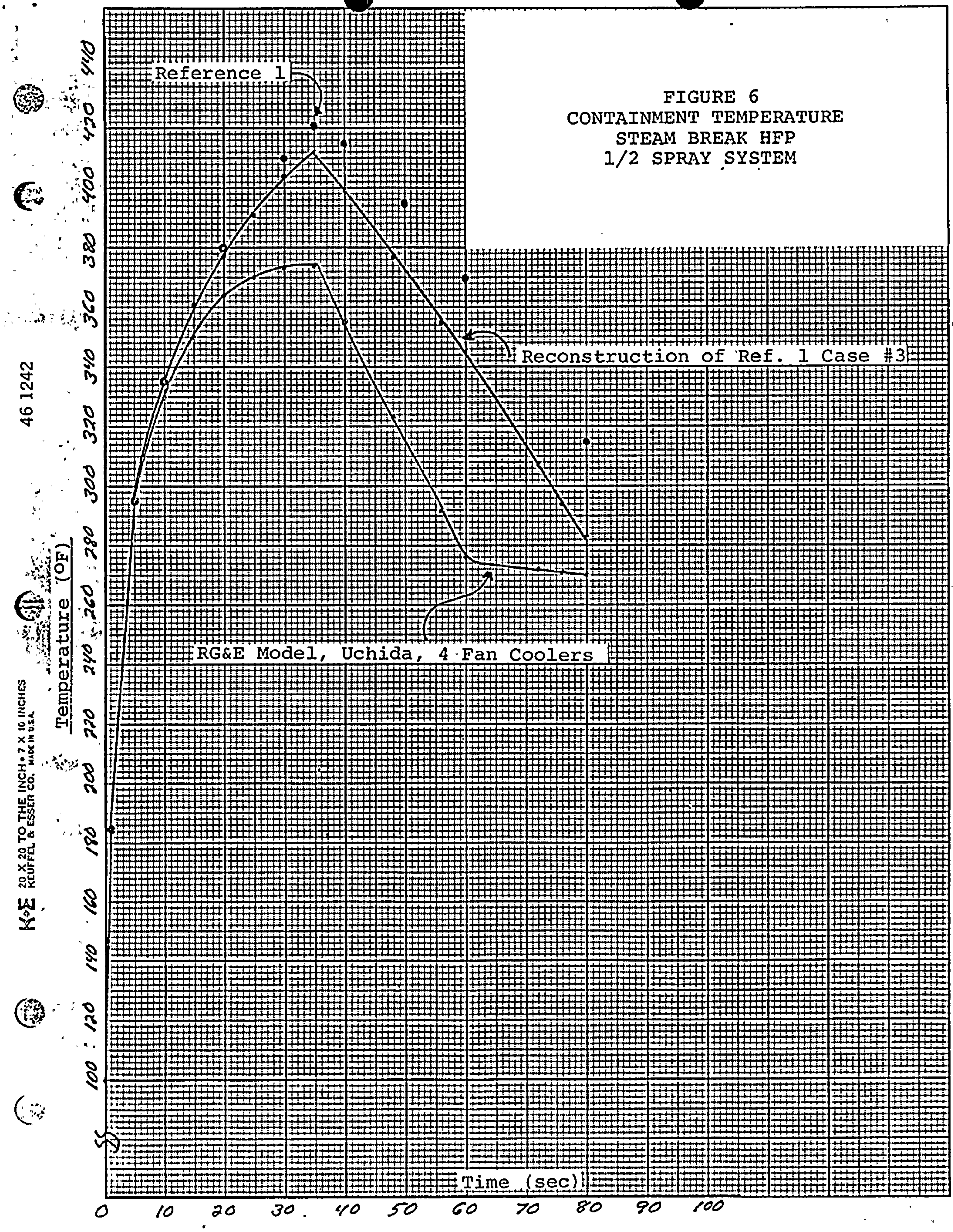
FIGURE 6
CONTAINMENT TEMPERATURE
STEAM BREAK HFP
1/2 SPRAY SYSTEM
Reference 1
Reconstruction of Ref. 1 Case #3
RG&E Model, Uchida, 4 Fan Coolers
Temperature (°F)
Time (sec)
0
10
20
30
40
50
60
70
80
90
100
55
100
120
140
160
180
200
220
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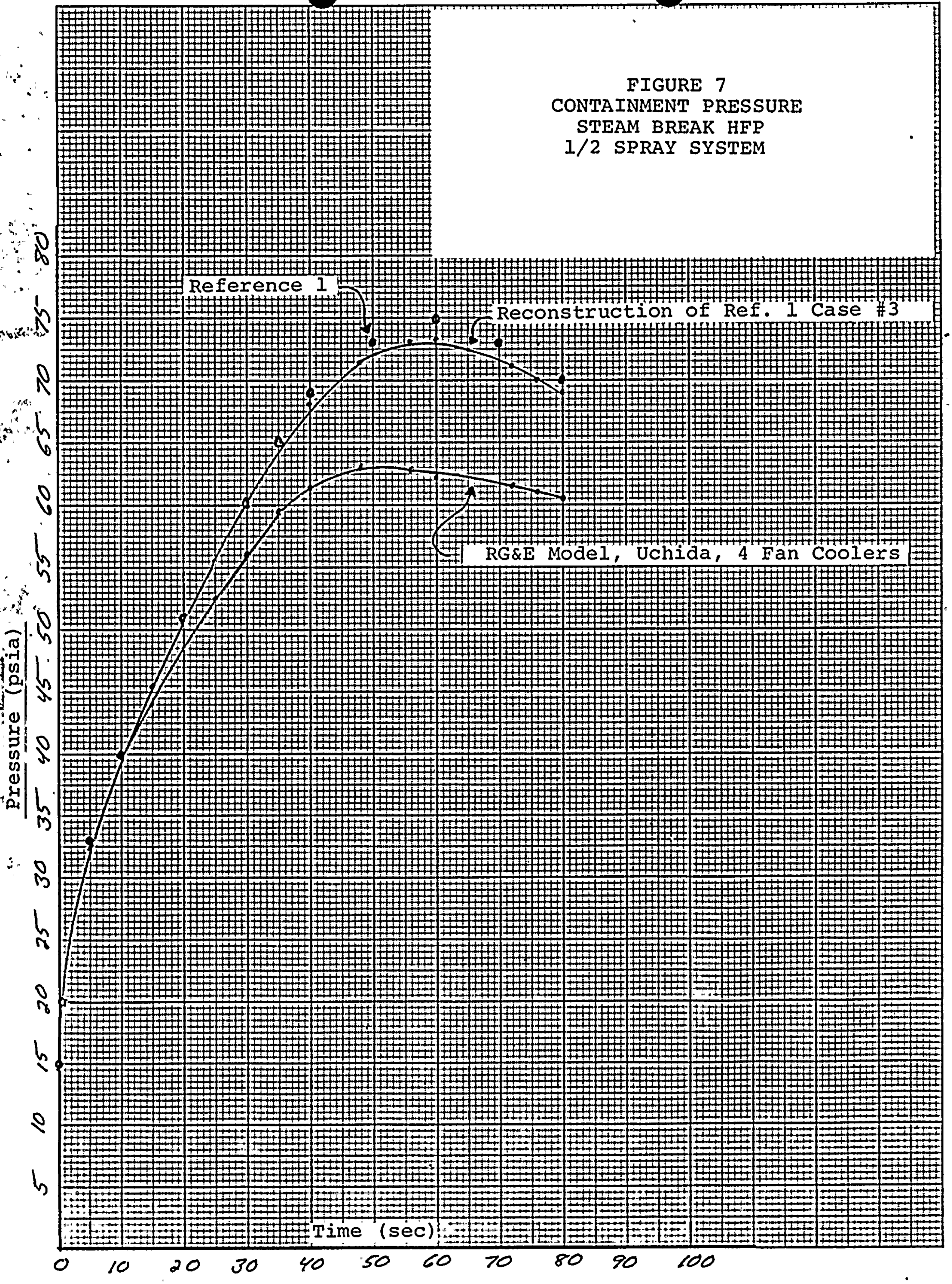

FIGURE 7
CONTAINMENT PRESSURE
STEAM BREAK HFP
1/2 SPRAY SYSTEM