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NUCLEAR PRODUCTION DEPARTMENT

August 13, 1983

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Mr. Harold R. Denton, Director

Dear Mr. Denton:

SUBJECT: Grand Gulf Nuclear Station
Units 1 and 2
Docket Nos. 50-416 and 50-417
License No. NPF-13
File 0260/0272/0756
Response to NRC Letter on Hydrogen Control
AECM-83/0455

Reference: 1. Letter from Mr. A. Schwencer to Mr. J. P. McGaughy, dated July 22, 1983
2. Letter HGN-012, from Mr. S. H. Hobbs to Mr. H. R. Denton, dated August 12, 1983

On June 29, 1983, the Hydrogen Control Owners Group (HCOG), of which Mississippi Power & Light Company (MP&L) is a member, met with members of the Nuclear Regulatory Commission (NRC) staff and presented preliminary information from the HCOG twentieth scale tests. As a result, on July 22, 1983, you transmitted Reference 1 expressing concern about impact of the test results on the licensing of the Grand Gulf Nuclear Station (GGNS). That letter also transmitted a request for additional information with regard to drywell temperature calculations. On July 28, 1983, the HCOG met again with members of the NRC staff. During that meeting, a plan for resolution of the concerns was discussed. This letter, with its attachments, is the MP&L response to questions raised in Reference 1 and the July 28 meeting.

Reference 1 indicated that hydrogen control systems are required to handle a variety of degraded core accidents which generate a metal water reaction of up to 75 per cent of the total active clad at varying hydrogen release rates but which are mitigated and recovered before core melt. MP&L is committed, through HCOG activities, to evaluate a range of such degraded core scenarios. Attachment 1 to this letter summarizes preliminary scoping studies which have been completed to evaluate a limited number of degraded core accident scenarios. This attachment also summarizes the work that will be completed by the HCOG to evaluate the required range of degraded core accidents. Finally, the attachment summarizes the expected results from the planned analyses.

Reference 1 expressed concern over vital equipment survivability due to very high local ambient temperatures resulting from diffusion flames in certain areas above the suppression pool. These high local temperatures were



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defined based on scaled test data from the 1/20th scale tests. Reference 1 also questioned the adequacy of the 1/20th scale tests to provide reliable information on the effects of hydrogen burns in the GGNS containment, and suggested that further testing might be required. MP&L concurs that further testing is needed and will, through the HCOG, conduct 1/4th scale tests to more accurately determine thermal environments associated with the range of degraded core scenarios. Reference 2 provides a commitment from the HCOG to complete these tests. Based on conversations with the staff, MP&L also concurs that early review of the proposed test matrix and experimental design by the NRC is needed and will result in a more expeditious final resolution of the issues that are being investigated by this testing. MP&L will assure that the NRC Staff has the opportunity for such an early review.

With regard to your concern over vital equipment survivability, we believe that the following factors should be weighed in order to place the 1/20th scale test results in proper perspective:

1. The 1/20th scale test was designed to provide flow visualization rather than thermal environment data; as a result, the thermal environment data is quite limited in extent.

2. The thermal environment obtained in the 1/20th scale tests resulted from locating the thermocouples in essentially the hottest location in the test facility since the instrumentation was not only located directly above a sustained diffusion flame, but was located simultaneously in one of the relatively unrestrained up flow areas (i.e., a "hot chimney") which had simulated directly below it two adjacent active spargers.

3. Relatively little thermal gradient data was obtained from the 1/20th scale test and that primarily in the vertical direction. Radial and circumferential thermal gradients although not measured, exist which tend to lower temperatures at essental equipment locations even in the hot chimneys.

4. The 1/20th scale test data overpredicts flame heights which causes a correspondingly hotter thermal environment than exists in full scale. A greater distance over which radiation heat transfer will operate and a greater distance above the flame for mixing of cooler air with the plume rising from the flame will be present when flame heights are more accurately represented. Preliminary results from an intermediate scale single sparger mockup have confirmed that this overprediction of flame height is substantial, perhaps by as much as a factor of two.

5. The 1/20th scale tests did not model containment spray effects which will be twofold. First, sprays will cause direct local cooling of essental equipment which will tend to mitigate the effects of the hot thermal environment. Second, sprays will cool the ambient containment air temperature resulting in entrainment of cooler air in the plume above the diffusion flame which will tend to reduce the gas temperatures in the plume.

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Attachment 2 discusses previous work which has been completed by MP&L to investigate survivability of essential equipment. The attachment also presents additional work which MP&L has completed to evaluate the effects of diffusion flames on equipment survivability. This work demonstrates that representative essential components will survive thermal environments derived from the 1/20 scale test data when realistic representation of mitigating effects such as containment sprays are included in the definition of the thermal environment.

The results from the preliminary scoping studies of hydrogen release rates in conjunction with the expected results from the planned detailed HCOG study provide assurance that the hydrogen ignition system is capable of mitigating accidents involving hydrogen generation. Accidents which result in large cladding coolant reactions approaching 75% will produce sustained hydrogen release rates which are less than or equal to 0.2 lbm/sec. Other accident sequences may produce lower cladding coolant reactions with higher release rates up to 1.0 lbm/sec for short periods of time, i.e., less than 10 minutes. Based on the aggregate of the work completed to date which is summarized in Attachments 1 and 2, MP&L believes that the hydrogen ignition system will assure that essential equipment survives the accident and that containment pressure integrity is maintained for all recoverable accidents.

A brief discussion of HCOG commitments on the 1/4th scale test and the hydrogen release rate calculations as well as substantial technical information regarding scaling methodology and the 1/4th scale facility were submitted in the HCOG letter of August 12, 1983 (Reference 2). MP&L endorses this HCOG letter. That letter in conjunction with this submittal provide what we believe is sufficient information to resolve the concerns raised in Reference 1 and the July 29 meeting.

Attachment 3 provides our response to your request for additional information about the drywell temperature. A revised, more realistic, however still extremely conservative, base case is presented. Preburn temperatures are about 315°F with the maximum base line temperatures for burns reaching 325°F. A response to various questions raised by the Staff is included in this attachment. Additional CLASIX-3 model evaluation and sensitivity analyses are in progress and will be submitted in the near future.

Attachment 4 provides a summary of test program results from the 1/20th scale tests and the 1/5th scale single sparger mockup. It concludes that the results expected from the planned 1/4th scale test will confirm that the GGNS thermal environment will be less severe than that predicted by the 1/20th scale test results.

MP&L believes that this letter along with attachments provides a complete response to the concerns identified in Reference 1 and in the July 28 meeting. Additional CLASIX-3 drywell sensitivity analyses are in progress and will be submitted in the near future. In addition, more equipment thermal response data and equipment location information will be provided in a follow up submittal. This information along with commitments that have been made to complete additional testing, analysis of hydrogen release rates, and analysis

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of equipment survivability should be sufficient to warrant issuance of a full power operating license for the Grand Gulf Nuclear Station.

Yours truly,

Sam H Hobbs

for L. F. Dale
Manager of Nuclear Services

RWE/SHH:sap

Attachment

cc: Mr. J. B. Richard (w/o)
Mr. R. B. McGehee (w/o)
Mr. T. B. Conner (w/o)
Mr. G. B. Taylor (w/o)

Mr. Richard C. DeYoung, Director (w/a)
Office of Inspection & Enforcement
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. J. P. O'Reilly, Regional Administrator (w/a)
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Region II
101 Marietta St., N.W., Suite 2900
Atlanta, Georgia 30303

Attachment 1 to
AECM-83/0455

HYDROGEN RELEASE RATE CALCULATIONS

The initial containment response analysis, which evaluated hydrogen combustion following degraded core accidents, utilized output from the MARCH computer code for hydrogen and steam release rates. The MARCH output was considered to be the best available representation of core degradation and subsequent hydrogen production. Even though the MARCH results represented the best available information, the output required considerable modification to more accurately represent a recoverable degraded core accident instead of a severe core melt accident. The modifications of the MARCH code output which were made for the containment response analyses are described in the CLASIX-3 Containment Response Sensitivity Analysis (Reference 1).

An improved analytical model of a BWR core's performance under degraded core conditions has been developed as part of the IDCOR program. This computer code was developed under the direction of EPRI/NSAC and has been designated as the BWR Core Heatup Code. The Hydrogen Control Owners Group (HCOG) committed in Reference 2 to utilize this code in completing a detailed analysis of possible hydrogen release rates associated with different recoverable accident scenarios.

The detailed modeling used by the BWR Core Heatup Code will be discussed at a meeting between the NRC, HCOG, IDCOR, and EPRI on August 23 and 24, 1983. This meeting will summarize capabilities of the code as well as identify input assumptions.

Several preliminary analyses with varying core injection flows have been completed using the BWR Core Heatup Code. Scenarios have been evaluated to date include unmitigated coolant inventory boiloff, low injection flows corresponding approximately to flow from 1 CRD pump, intermediate injection flow corresponding approximately to flow from the RCIC system, and high injection flow corresponding approximately to core spray flow. Additional limited sensitivity studies were also conducted for the unmitigated coolant inventory boiloff scenario. The results of these preliminary analyses were presented during a meeting between the HCOG and the NRC on June 29, 1983.

The predictions for hydrogen release rate as a function of time for the four scenarios identified above are included as Figures 1-4 and summarized in Table 1. These Figures show that the maximum sustained hydrogen production rate occurs for an intermediate injection rate and results in production of approximately .8 lbm/sec for 10 minutes.

The hydrogen release rate graphs in Figures 1-3 are extremely conservative. The BWR Core Heatup Code runs which predict these release rates also predict substantial core melt fractions associated with these accidents. These core melt fractions are produced by essentially unrecoverable accidents which go beyond the requirements of the degraded core hydrogen control rule. Figure 4 depicts a core spray activation which would not result in substantial core melt and would be recoverable.

Reference 2 discusses a commitment by the HCOG to complete additional analyses of hydrogen release rates for a range of accident scenarios. Scenarios which will be evaluated include constant injection flow rates, multiple boiloff and reflood transients, and carefully orchestrated sequences which result in reaction of the equivalent of 75% active fuel cladding. The code runs will focus on injection flows which preclude core melting.

Experience gained in developing the BWR Core Heatup Code, utilizing the code to support the IDCOR program, and completing the preliminary scoping studies performed to date provides confidence in the results which can be expected from the planned HCOG study. Extremely large fractions of the core zirconium inventory (up to 75% equivalent active fuel cladding) can be oxidized without core melt over very long periods of time at hydrogen release rates of less than .2 lbm/sec for continuous injection. A multiple boiloff and reflood scenario resulting in no core melting and zirconium water reactions approaching the equivalent of 75% of the active fuel cladding can also be orchestrated. This scenario is expected to produce higher release rates up to .8-1 lbm/sec for durations less than 10 minutes. Each boiloff/reflood transient will result in a decrease in the peak hydrogen release rate.

The HCOG also intends to analyze scenarios which result in relatively high sustained hydrogen release rates above .4 lbm/sec but which do not result in reaction of the equivalent of 75% active fuel cladding. These scenarios, for both constant injection and multiple boiloff and reflood transients, will produce higher hydrogen release rates up to 1.0 lbm/sec for periods less than 10 minutes.

Due to the exothermic nature of the zirconium oxidation reaction, the reaction rate which can be sustained without leading to core melt is severely restricted. It is not possible to sustain an oxidation rate which produces greater than .8 lbm/sec for even 10 minutes without producing substantial core melt fractions. Consequently, due to fundamental limitations on reactor core ability to absorb additional energy, high hydrogen release rates simply cannot be sustained within the context of degraded core events which stop short of core melt.

Thus, sustained hydrogen release rates will be low (less than .2 lbm/sec) for relatively long durations or somewhat higher (up to 1.0 lbm/sec) for short durations on the order of several minutes. The hydrogen ignition system is capable of mitigating accidents which result in either type of release rate. Release rates of .2 lbm/sec of hydrogen will result in deflagration type combustion which has much lower severity consequences than the deflagration type combustion analyzed in Reference 1. This is based upon the threshold for creating sustained diffusion flames established by testing discussed in Attachment 4. Higher release rates up to 1.0 lbm/sec will result in steady diffusion flames for less than 10 minutes. As discussed in Attachment 2, representative components of essential equipment have been shown to survive longer duration diffusion flames. The pressure integrity of the containment is not challenged for either low or high hydrogen release rates. MP&L therefore concludes that the hydrogen ignition system is an effective system for mitigating all recoverable accidents involving substantial hydrogen generation.

References

1. CLASIX-3 Containment Response Sensitivity Analysis transmitted to the NRC by letter number HGN-001 from J. D. Richardson to H. R. Denton dated January 15, 1982.
2. Letter number HGN-012 from S. H. Hobbs to H. R. Denton dated August 12, 1983.

Table 1

HYDROGEN SOURCE TERM SUMMARY*
EPRI - BWR HEATUP CODE ANALYSES

| Figure | Reflood Flow | Reflood Equip. | $H_2$ Peak LB/Sec. | Sustained Rate LB/Sec. | Duration of Peak (Min.) |
|---|---|---|---|---|---|
| 1 | 0 gpm boiloff only | None | 0.66 | 0.5 | 15 |
| 2 | 58 gpm | CRD | 0.8 | 0.6 | 10 min. |
| 3 | 650 gpm | RCIC | 1.5 ( 1 min.) | 0.8 | 10 min. |
| 4 | 4750 gpm | Core Spray | 1.35 ( 2 min.) | 0.6 | 7 min. |

*From EPRI slide used at June 29, 1983 HCOG presentation

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FIGURE 1

PURE BOILOFF

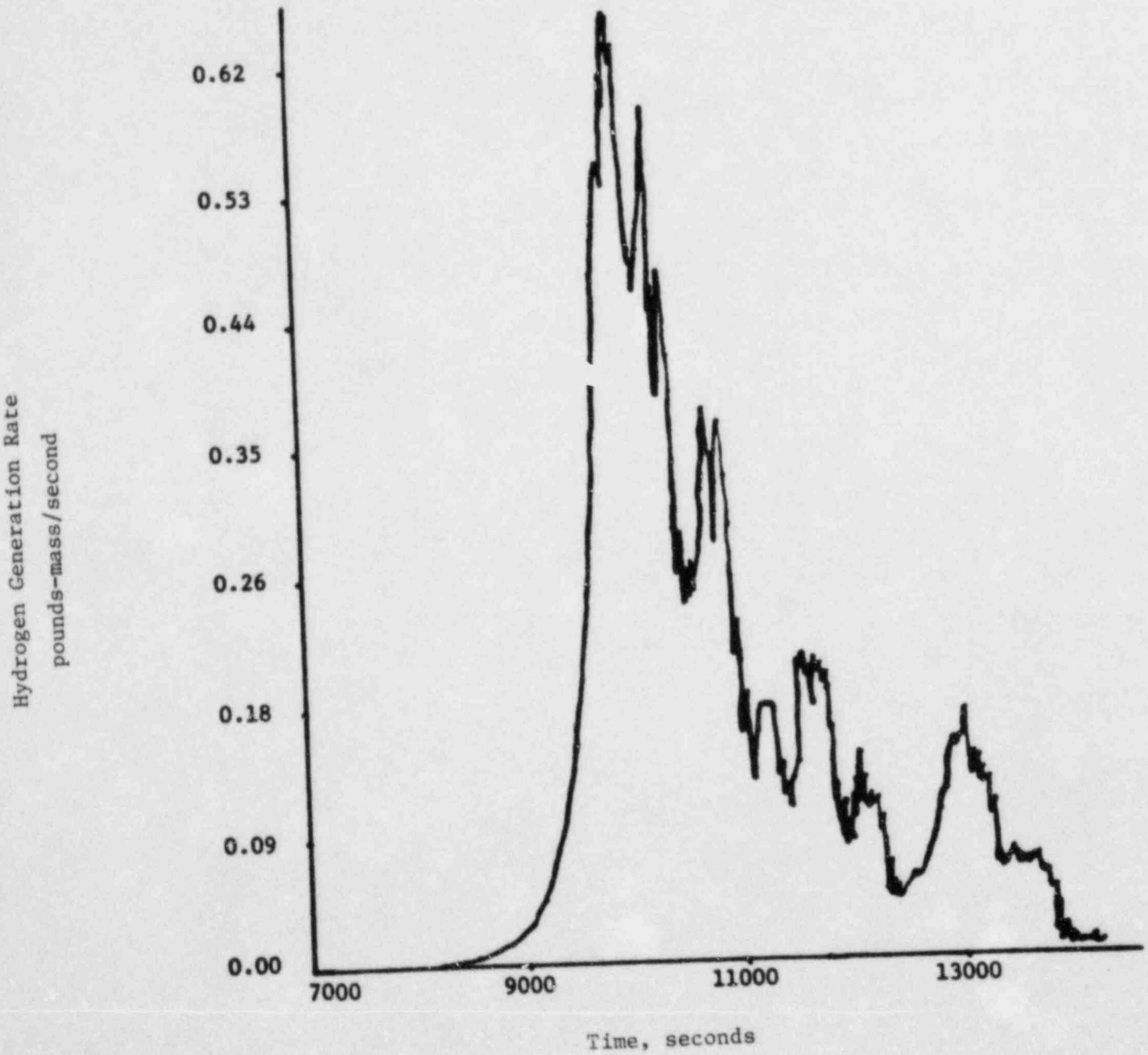


FIGURE 2

CRD Injection

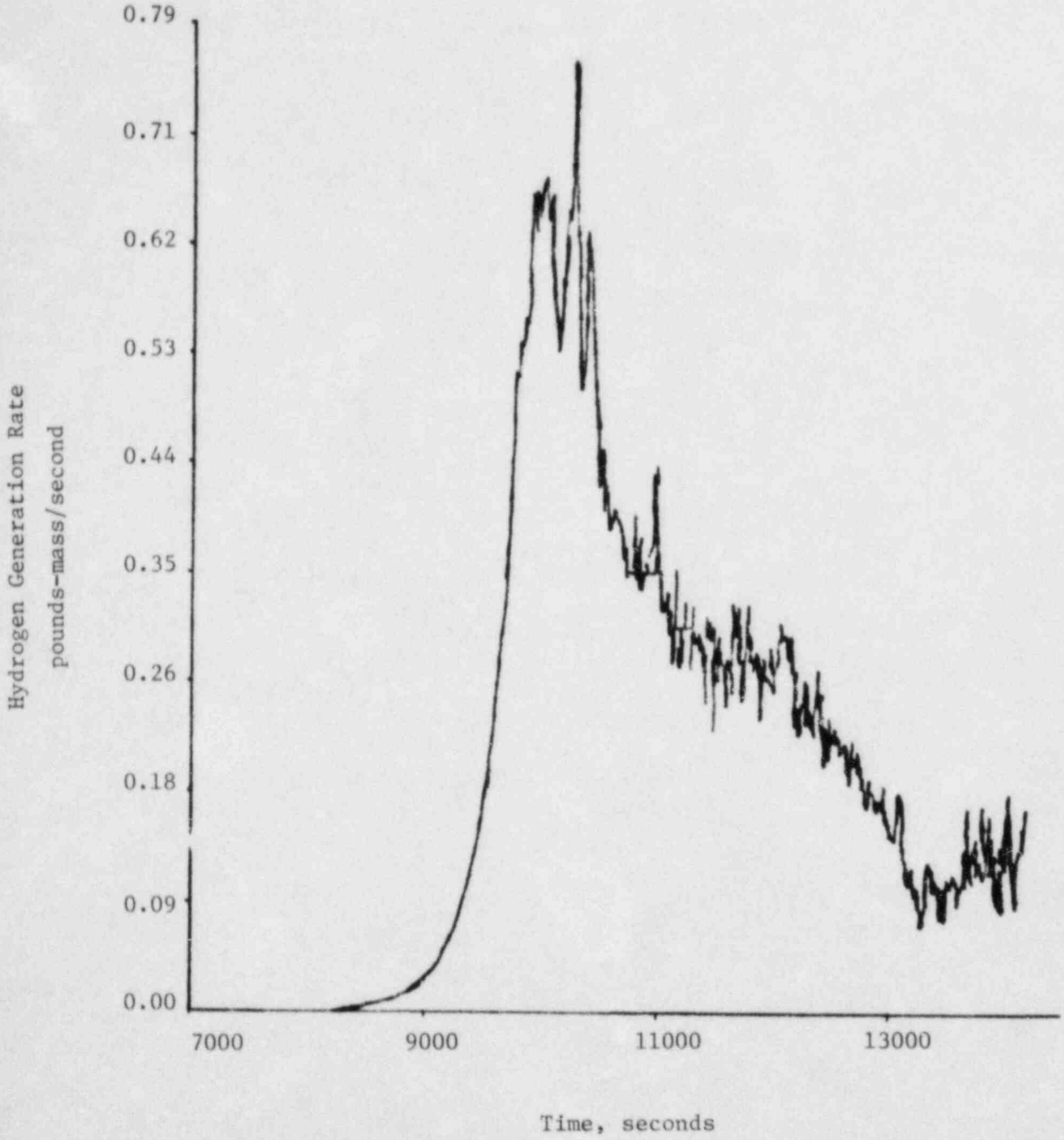


FIGURE 3

RCIC INJECTION

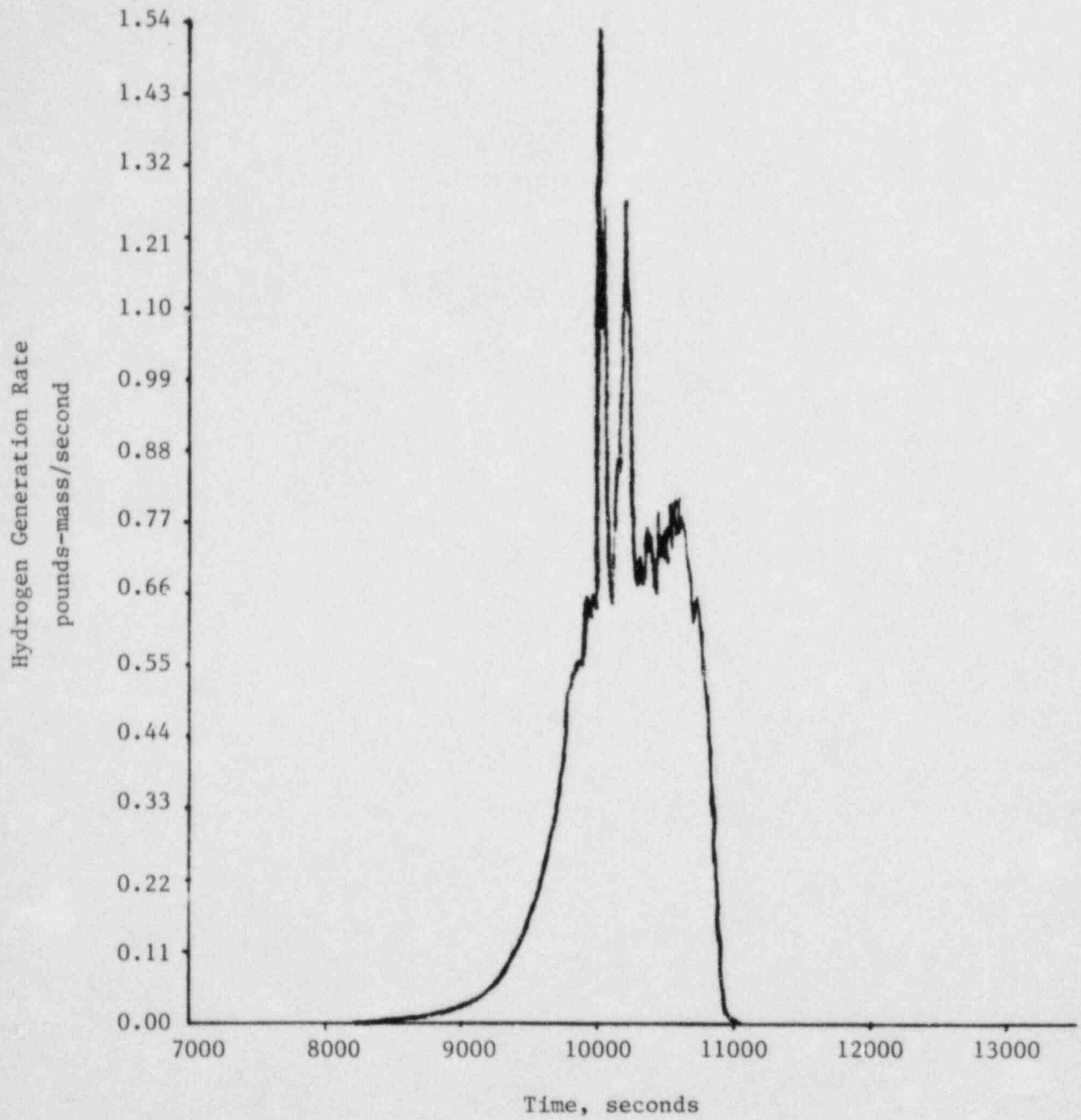
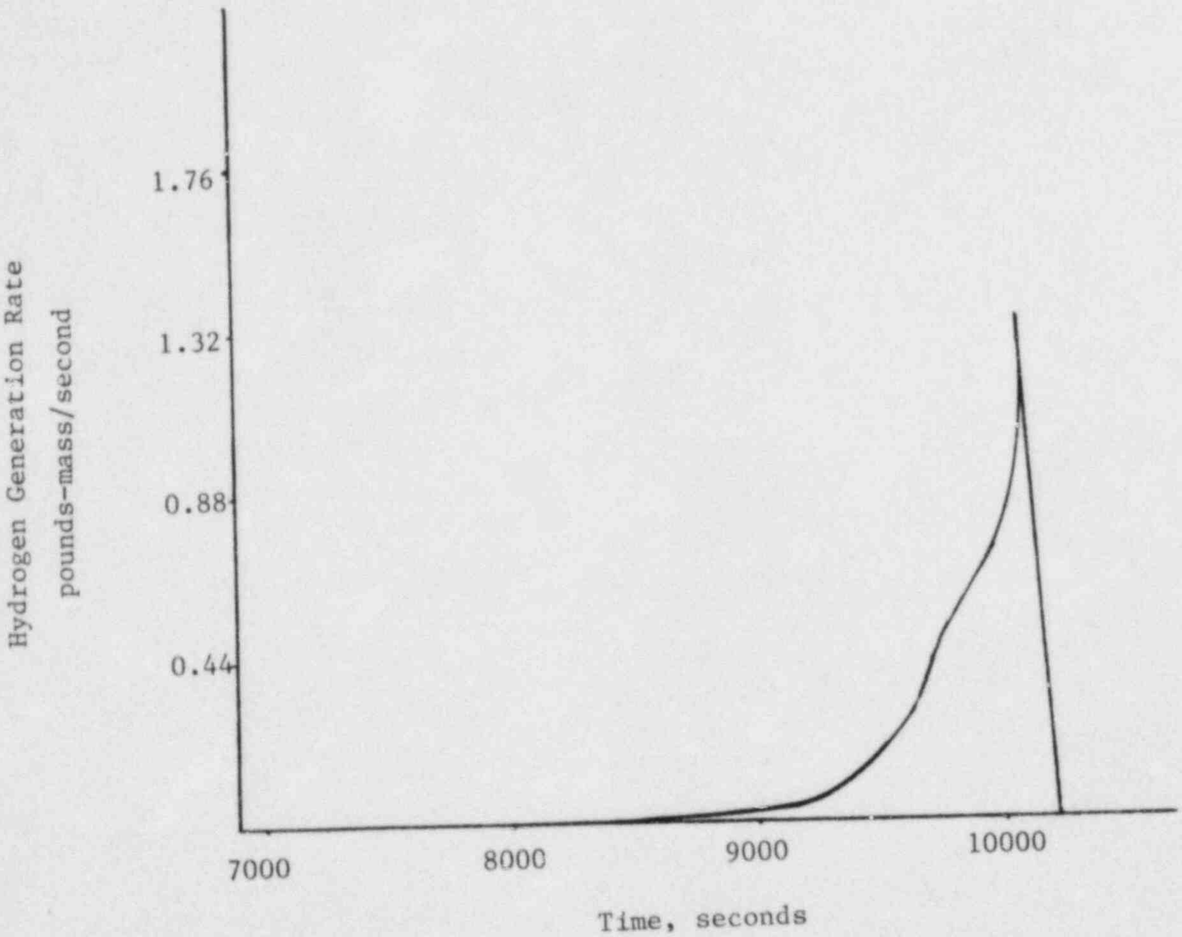


FIGURE 4

CORE SPRAY INJECTION



# EQUIPMENT SURVIVABILITY
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# EQUIPMENT SURVIVABILITY

## I. Analysis for Deflagration Type Combustion

The Mississippi Power & Light (MP&L) Report on Equipment Survivability in Support of the Grand Gulf Nuclear Station Hydrogen Igniter System (transmitted by Reference 1) provided a comprehensive assessment of the ability of essential equipment to withstand the environment produced by hydrogen combustion following a degraded core accident. All essential equipment was conservatively evaluated for deflagration type combustion against the most severe temperature environment, i.e., the environment in the wetwell compartment. The temperature environment was derived using output from the CLASIX-3 computer code. The report documented that all essential equipment will survive the hydrogen combustion events. This conclusion was documented based on heat transfer calculations which showed that the maximum component internal temperatures remain well below the component environmental qualification temperatures. In almost all cases, the equipment surface temperature remains below the environmental qualification temperatures. The information in Reference 1 provides assurance that for the base case hydrogen release rate 1 lbm/sec, essential equipment is capable of surviving deflagration type hydrogen combustion. This evaluation bounds cases for deflagration type burns resulting from lower hydrogen release rates.

In the course of confirmatory testing to evaluate the type of hydrogen combustion which may occur above the suppression pool, the Hydrogen Control Owners Group (HCOG) determined that a potential exists for establishing steady diffusion flames above the pool surface. The testing to date has demonstrated that steady diffusion flames will be established for hydrogen release rates above .4 to .5 lbm/sec. For flow rates below this threshold, the hydrogen will burn essentially in the manner analyzed by CLASIX-3 and the equipment survivability analysis described in Reference 1 bounds the thermal response for all essential equipment.

## II. Preliminary Analyses Using Thermal Environments From 1/20 Scale Test

The confirmatory testing mentioned above was completed in a 1/20th scale full containment model. Portions of the data taken during the test included bulk gas temperatures, radiant heat fluxes and gas velocities. Using well established scaling relations, a full scale thermal environment was developed for the region immediately below the HCU floor. The thermal environment was defined for the highest temperature region which is directly above the steady flame. The test used to define the thermal environment was completed using a constant hydrogen flow rate equivalent to .8 lbm/sec at full scale.

The effects of the experimentally derived thermal environment on the igniter device were investigated using the heat transfer model developed for the survivability analyses contained in Reference 1. The igniter was selected for evaluation because it initially appeared to be the only component which could be directly effected by the experimentally defined thermal environment. The results from the revised heat transfer analysis were presented to the NRC in a meeting with the HCOG on June 29, 1983. Figure 1 shows the temperature response for the surface of the igniter enclosure and for the igniter transformer. The thermal environment was applied to the igniter until the equivalent of 75% of active fuel cladding would have reacted with core coolant.

The HCOG has completed a preliminary, scoping analysis of possible hydrogen release rates for varying reflood rates using the BWR Core Heatup Code. This scoping analysis is described in detail in Attachment 1. The results from this scoping analyses showed that a sustained hydrogen release rate of .8 lbm/sec until the equivalent of 75% of the active fuel cladding has been reached, results in very substantial core melt fractions. These large core melt fractions would lead to non-recoverable accidents and are clearly outside the range of degraded core accidents which stop short of core melt. At the June 29 meeting, the HCOG stated that even allowing a substantial core melt fraction, the maximum equivalent of 25% of the active fuel cladding could be reacted at a sustained release rate of .8 lbm/sec of hydrogen. Certain essential equipment (which is believed to be representative of thermally sensitive components) was analyzed for a period of 18 minutes with a hydrogen release rate of .8 lbm/sec. The BWR Core Heatup Code predicts less than 10 minutes at a sustained release rate of .8 lbm/sec. The other 8 minutes of the transient has a release rate of between .4 and .8 lbm/sec.

Figure 1 shows that the peak temperature reached by the igniter transformer is below 400°F if the transient is terminated when the equivalent of 25% of the active fuel cladding has reacted. The igniter transformer has been certified by the manufacturer as apable of continuing to function up to temperatures of 400°F. Therefore, MP&L concludes that the igniter would survive a transient which does not lead to substantial core melt but does produce a hydrogen release rate of .8 lbm/sec.

The thermal environment which was defined for this analysis is very conservative. The 1/20th scale test data used to develop the temperature profile does not include the effects of containment sprays which are expected to provide substantial cooling. Also, as noted above, the thermal environment has been defined for the highest temperature region in the wetwell directly above the steady diffusion flame. Results from a preliminary single sparger test at a larger scale indicate that the 1/20th scale test results substantially overpredict the resulting flame height. Finally, the duration of the hydrogen release and the associated steady diffusion flame is longer than would result if core melt is avoided. Attachment 1 notes that the .8 lbm/sec release rate cannot be sustained for 10 minutes without substantial core melt.

## III. Analyses Using Modified Thermal Environments From 1/20th Scale Test

In order to provide a more realistic assessment of essential equipment survivability, MP&L has revised the thermal environment discussed above. The revised thermal environment incorporates an estimation of the effects of containment sprays. The 1/20th scale test data shows the bulk gas temperature initially rises, remains constant for a short period of time and then rises to the temperatures used in the first definition of the thermal environment. The final increase in wetwell temperatures is believed to result from heating of the atmosphere in the upper containment. Consequently high temperature gases are being recirculated to the diffusion flame where they are being further heated. In a facility capable of simulating sprays, the upper containment would be cooled by the sprays and relatively hot gases would not be recirculated through the diffusion flames. The revised thermal environment has been defined based on the continued availability of relatively cool atmosphere in the upper containment which is simulated by the data from early in the 1/20th scale tests. This revised thermal environment is summarized in Table 1.

MP&L has evaluated three representative components against this revised thermal profile. The components evaluated were an igniter, a reactor vessel pressure transmitter and a containment isolation valve actuator. Figures 2-4 show the thermal response of the three components. The igniter transformer reaches a temperature of approximately 320°F when 25% of the cladding has reacted which is well within the survivability temperature of the transformer.

The revised thermal environment remains extremely conservative for evaluating the temperature response of equipment. The gas components of the radiant heat fluxes used in defining the thermal environment have been determined to be approximately 700 Btu/hr-ft$^2$ above the correct values. This environment still applies for the region directly above the diffusion flame and below the HCU floor. No credit has been taken for expected horizontal attenuation above or below the HCU floor since most equipment lies outside of the narrow vertical thermal plume from the spargers. The thermal environment used is still based upon overpredictions of flame height in the 1/20th scale tests. Finally, the duration of the diffusion flame is exaggerated due to the core melt fraction which is associated with the length of time at this hydrogen release rate.

## IV. Identification of Essential Equipment

Reference 1 contained a listing of essential equipment which was required to survive hydrogen combustion. At the time that the list of essential equipment in Reference 1 was prepared, four criteria were established for requiring essential equipment to survive hydrogen combustion. These criteria, as identified in Reference 1, are:

1. Systems which must function to mitigate the consequences of the event.
2. Equipment which must maintain the containment pressure boundary.

3. Systems which may be necessary to recover the core.
4. Systems whose function may be required to monitor the course of the event.

The list of essential equipment provided in Reference 1 was broad and did not identify equipment locations since the combustion phenomenon evaluated, i.e., deflagrations, was not location dependent. All equipment in the containment and drywell regardless of location was evaluated against the most severe deflagration type thermal environment which was the wetwell area.

Since the submittal of Reference 1, MP&L has determined that a different type of combustion phenomenon may influence definition of the thermal environment in the drywell. Reference 2 submitted a report by Combustion Explosives, Inc. (COMBEX) which suggested that an inverted diffusion flame may be established at the purge compressor discharge vents into the drywell.

The Hydrogen Control Owners Group (HCOG) during confirmatory testing in a 1/20th scale full containment mock up has also determined that a different type of combustion phenomenon may dominate definition of the thermal environment in the wetwell. This testing, which is described in Attachment 4, has shown that for certain hydrogen flow rates, sustained diffusion flames may be formed above the suppression pool surface.

As a result of these separate findings, MP&L has determined that a need exists to revise the list of essential components provided in Reference 1 and to identify the location and function of each component. Table 2 provides a listing of the drywell essential components and specifies their exact locations. Figures 5 - 8 show approximate locations of equipment in the drywell.

The criteria for establishing a component as essential are very similar to the criteria specified in Reference 1 with some important exceptions. The revised criteria are as follows:

1. Systems and components which must function to mitigate the consequences of the event
2. Systems and components needed for maintaining integrity of the containment pressure boundary

3. Systems and components needed for maintaining the core in a safe condition
4. Systems and components needed for monitoring the course of the accident.

Some of the important exceptions relate to definition of systems and components needed for maintaining the integrity of the containment pressure boundary. Isolation valve actuators are no longer treated as essential unless the valve must perform an active function to recover the core, maintain the core in a safe condition, or mitigate the consequences of the event. Also a narrower definition has been applied to systems and components needed for monitoring the course of the event. Isolation valve position indication switches have been excluded from the revised list as have drywell pressure instruments.

The list of essential equipment inside containment to be provided in a follow up submittal will show that the only equipment which may be effected by diffusion flames above the suppression pool, i.e., below elevation 140', is limited to four items. These items include the igniters, certain air operated isolation valves, the reactor pressure vessel pressure transmitters and the reactor pressure vessel level transmitters.

This attachment provides a survivability evaluation of igniters, air operated isolation valve actuators and pressure transmitters against a diffusion flame thermal environment which should be representative of the full scale environment when the effect of sprays is considered. Attachment 2 demonstrates that all of these are expected to survive these thermal environments. It should be noted that the thermal response of a level transmitter should be directly comparable to the thermal response of a pressure transmitter.

## V. Summary and Conclusions

MP&L and the HCOG recognize that the test data obtained from the 1/20th scale is not sufficient to provide a complete assessment of equipment survivability. Reference 3 provides a commitment from the HCOG to conduct testing in a 1/4 scale Mark III containment mock up. The thermal profiles measured in this facility will be used to provide a complete assessment of equipment survivability.

MP&L has demonstrated that, for deflagration type combustion, all essential equipment should survive sustained hydrogen release rates up to 1 lbm/sec which result in a total hydrogen production equivalent to the reaction of 75% of the active fuel cladding. Using conservative thermal environments derived experimentally, based upon anticipated full scale effects, MP&L has shown that representative components can survive steady diffusion flames with hydrogen release rates between .4 lbm/sec and .8 lbm/sec.

The hydrogen release rate studies described in Attachment 1 to this letter are expected to demonstrate that the duration of hydrogen release rates above .8 lbm/sec will be much less than the duration of the .8 lbm/sec release rate which has been used in the preliminary equipment survivability evaluations completed to date. This result is expected because of limitations on the amount of energy which can be added to the core by the exothermic reaction of zirconium and water. Excessive energy addition will lead to substantial core melt fractions which are outside of the scope of the present rule.

Thus MP&L has shown that all essential equipment can survive deflagration type combustion and that representative components can survive conservative best estimate representations of expected diffusion flame environments. Based upon MP&L's commitment to participate with the HCOG in confirmatory testing, MP&L believes that the information presented to date is sufficient to justify issuance of a full power operating license for the Grand Gulf Nuclear Station.

References

1. Letter number AECM-82/26, dated January 19, 1982, from Mr. L. F. Dale to Mr. H. R. Denton.

2. Letter number AECM-82/25 dated March 2, 1982, from Mr. L. F. Dale to Mr. H. R. Denton.

3. Letter HGN-012, dated August 12, 1983, from Mr. S. H. Hobbs to Mr. H. R. Denton.

Table 1

Revised Thermal Environment Based on
1/20 Scale Test Data From .8 lbm/sec Hydrogen Flows

| | |
|---|---|
| Temperature at HCU Floor | 458°F |
| Temperature at 2.5' Below HCU Floor | 480°F |
| Radiant Heat Flux Below HCU Floor | |
| Gas | 4300 Btu/hr ft² |
| Grating | 470 Btu/hr ft² |
| Radiant Heat Flux Above HCU Floor | |
| Gas | 3000 Btu/hr ft² |
| Grating | 470 Btu/hr ft² |
| Gas Velocity | 26 ft/sec |

Table 2

Drywell Equipment Required
to Survive a Hydrogen Burn

| Description | Approx. Evaluation | Azimuth | Approx. Dist From Center Line Reactor | Ref. Dwg |
|---|---|---|---|---|
| **Hydrogen Igniters** | | | | |
| E61-D106 | 146'-3 7/8" | 0 | 22/-10" | 1 |
| E51-D107 | 145'-7" | 63 | 29'-3" | 1 |
| E61-D108 | 146'-2" | 120 | 29'-8" | 1 |
| E61-D109 | 147'-1" | 180 | 26'-3" | 1 |
| E61-D110 | 145'-7" | 240 | 29-1½" | 1 |
| E61-D111 | 145'-7" | 313 | 25'-1¼" | 1 |
| E61-D112 | 160'-7 7/8" | 0 | 27'-3 3/8" | 1 |
| E61-D113 | 160'11 3/4" | 60 | 29'-8 3/4" | 1 |
| E61-D114 | 160'-4" | 135 | 27'-0 3/8" | 1 |
| E61-D115 | 160'-11½" | 180 | 26'-10" | 1 |
| E61-D116 | 160'-6" | 232 | 26'-1" | 1 |
| E61-D117 | 160'6" | 324 | 26'-4 5/8" | 1 |
| E61-D118 | 179'-0" | 0 | 26'-4 5/8" | 1 |
| E61-D119 | 179'-0" | 65 | 26'-3 3/4" | 1 |
| E61-D120 | 179'-0" | 125 | 26'-3 3/4" | 1 |
| E61-D121 | 179'-0" | 185 | 26'-3 3/4" | 1 |
| E61-D122 | 179'-0" | 245 | 26'-3 3/4" | 1 |
| E61-D123 | 179'-0" | 305 | 26'-3 3/4" | 1 |
| **Isolation Valves (MOV)** | | | | |
| B21-F016 | 141'-3" | 8 | 34'-6" | M1301 |
| E51-F063 | 143'-2" | 0 | 30'-0" | M1301 |
| E51-F076 | 143'-2" | 0 | 30'-0" | |
| G33-F001 | 139'-5" | 3 | 33'-0" | M1301 |
| G33-F252 | 166'-10" | 7 | 33'-0" | M1301 |
| E12-F009 | 124'-7" | 0 | 25'-0" | M1306 |
| **Safety Relief Valves (AOV)** | | | | |
| B21-F022A | 150'-7" | 8 | 31'-0" | M1302 |
| B21-F022B | 150'-7" | 340 | 31'-0" | M1302 |
| B21-F022C | 150'-7" | 20 | 31'-0" | M1302 |
| B21-F022D | 150'-7" | 352 | 31'-0" | M1302 |
| B21-F047A | 154'-0" | 34 | 22'-0" | M1302 |
| B21-F047L | 154'-0" | 53 | 27-6" | M1302 |
| B21-F041D | 154'-0" | 315 | 21'-0" | M1302 |
| B21-F041F | 154'-0" | 288 | 26'-6" | M1302 |
| B21-F041K | 154'-0" | 304 | 27'-0" | M1302 |
| B21-F051A | 154'-0" | 45 | 22'-0" | M1302 |
| B21-F051B | 154'-0" | 272 | 25'-6" | M1302 |
| B21-F051C | 154'-0" | 77 | 26'-0" | M1302 |
| B21-F051D | 154'-0" | 327 | 21'-6" | M1302 |

Table 2 (Cont'd)

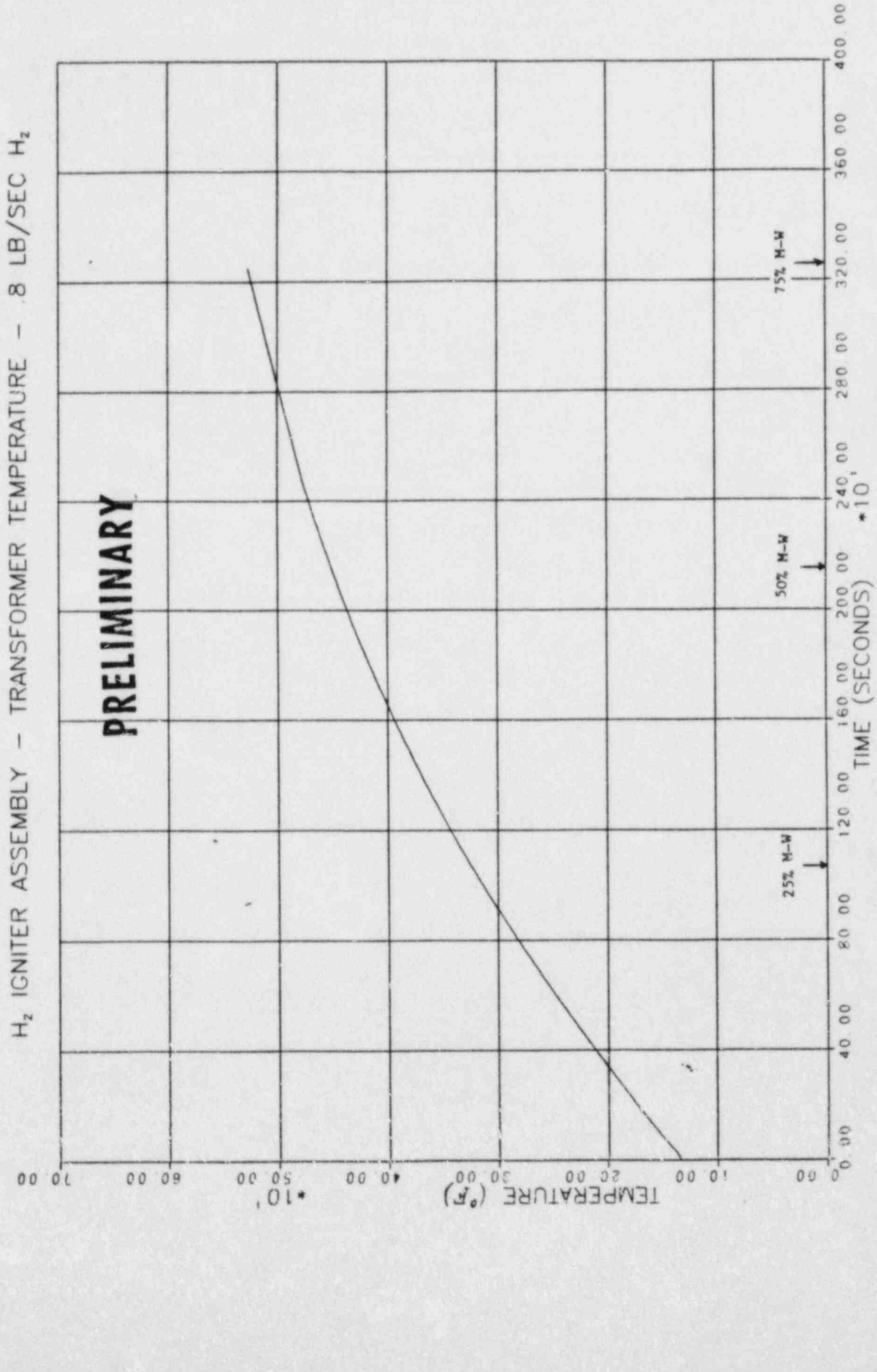
| Description | Approx. Evaluation | Azimuth | Approx. Dist From Center Line Reactor | Ref. Dwg |
|---|---|---|---|---|
| RHR Valves | | | | |
| E12-F041A | 140'-4" | 39 | 30'-0" | |
| E12-F041B | 147'-6" | 219 | 21'-6" | |
| E12-F041C | 148'-2" | 141 | 21'-0" | |
| E12-F006 | 153'-9" | 120 | 19'-0" | |
| E22-F055 | 153'-2" | 30 | 19'-0" | |
| Instrumentation | | | | |
| D21-RE-N048A(Radmonitors) | 161'-10" | 0 | 36'-0" | J1508 |
| D21-RE-N048D | 161'-10" | 183 | 36'-0" | J1508 |
| M71-TE-N008A(Cont. & DW) | 161'-10" | 40 | 36'-0" | J1508 |
| M71-TE-N008B(Temp Monitors) | 161'-0" | 250 | 36'-0" | J1508 |
| M71-TE-N008C | 161'-0" | 135 | 36'-0" | J1508 |
| M71-TE-N008D | 161'-0" | 310 | 36'-0" | J1508 |
| M71-TE-N013A | 94'-6" | 55 | 10'-7" | J1505 |
| M71-TE-N013B | 94'-6" | 225 | 10'-7" | J1505 |
| M71-TE-N013C | 94'-0" | 112 | 10'-3" | J1505 |
| M71-TE-N013D | 94'-6" | 280 | 10'-7" | J1505 |

Table 2 (Cont'd)

Equipment Qualification Temperatures

| Component | NUREG-0588 Qualification Temp | Duration |
|---|---|---|
| Hydrogen Igniters | 330°F | 3 Hours |
| Transformer (Igniter) | 400°F | - |
| Valves (E12-F009, G33-F001, G33-F252, R51-F063, B21-F016, R51-F076) | 240°F | - |
| Valves (B21-F022A, B, C, & D) | 330°F | 1 hour |
| Valves (B21-F047, F041, F051) | 349°F | 4 Days |
| Valve Limit/Position Switches, Excludes MSIVs | 330°F | - |
| Power Cables | 346°F | 3 hours, 20 minutes |
| Control Cables | 346°F | 3 hours, 20 minutes |
| Instrument Cables | 340°F | 6 hours |
| Thermocouple Ext. Wire | 340°F | 5½ hours |
| Terminal Blocks | 340°F | 5½ hours |
| Instrumentation | | |
| D21-RE-N048A, D | 340°F | - |
| M74-TE-N008A, B, C, D N013A, B, C, D | 340°F | 6 hours |

FIGURE 1



H2 IGNITER ASSEMBLY - TRANSFORMER TEMPERATURE - .8 LB/SEC H2
PRELIMINARY
TIME (SECONDS) *10^1
0.00
40.00
80.00
120.00
160.00
200.00
240.00
280.00
320.00
360.00
400.00
25% M-W
50% M-W
75% M-W
TEMPERATURE (°F) *10^1
0.00
10.00
20.00
30.00
40.00
50.00
60.00
70.00

FIGURE 2

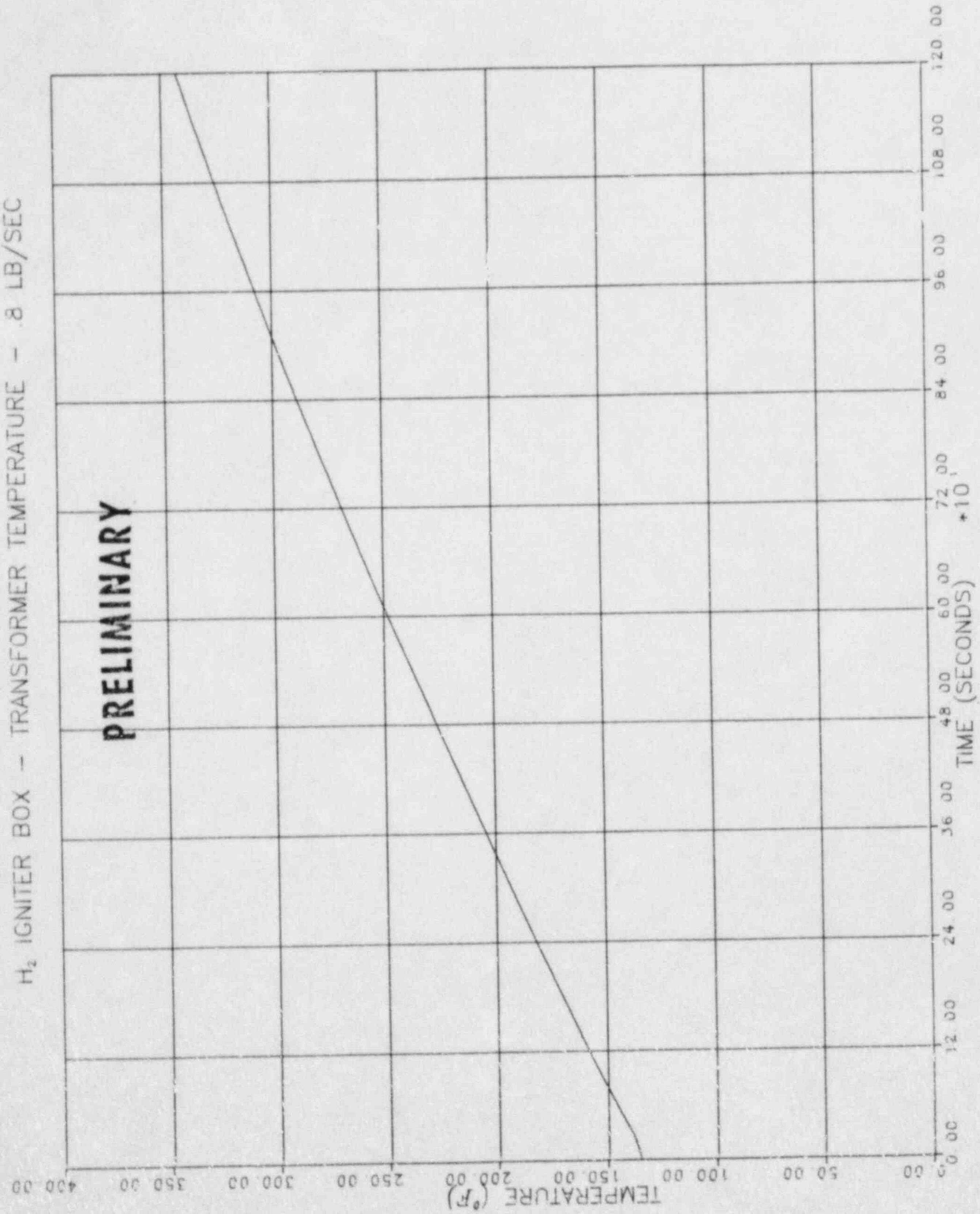

H2 IGNITER BOX - TRANSFORMER TEMPERATURE - .8 LB/SEC
PRELIMINARY
TIME (SECONDS) *10^1
TEMPERATURE (°F)
0.00
12.00
24.00
36.00
48.00
60.00
72.00
84.00
96.00
108.00
120.00
50.00
100.00
150.00
200.00
250.00
300.00
350.00
400.00

FIGURE 3

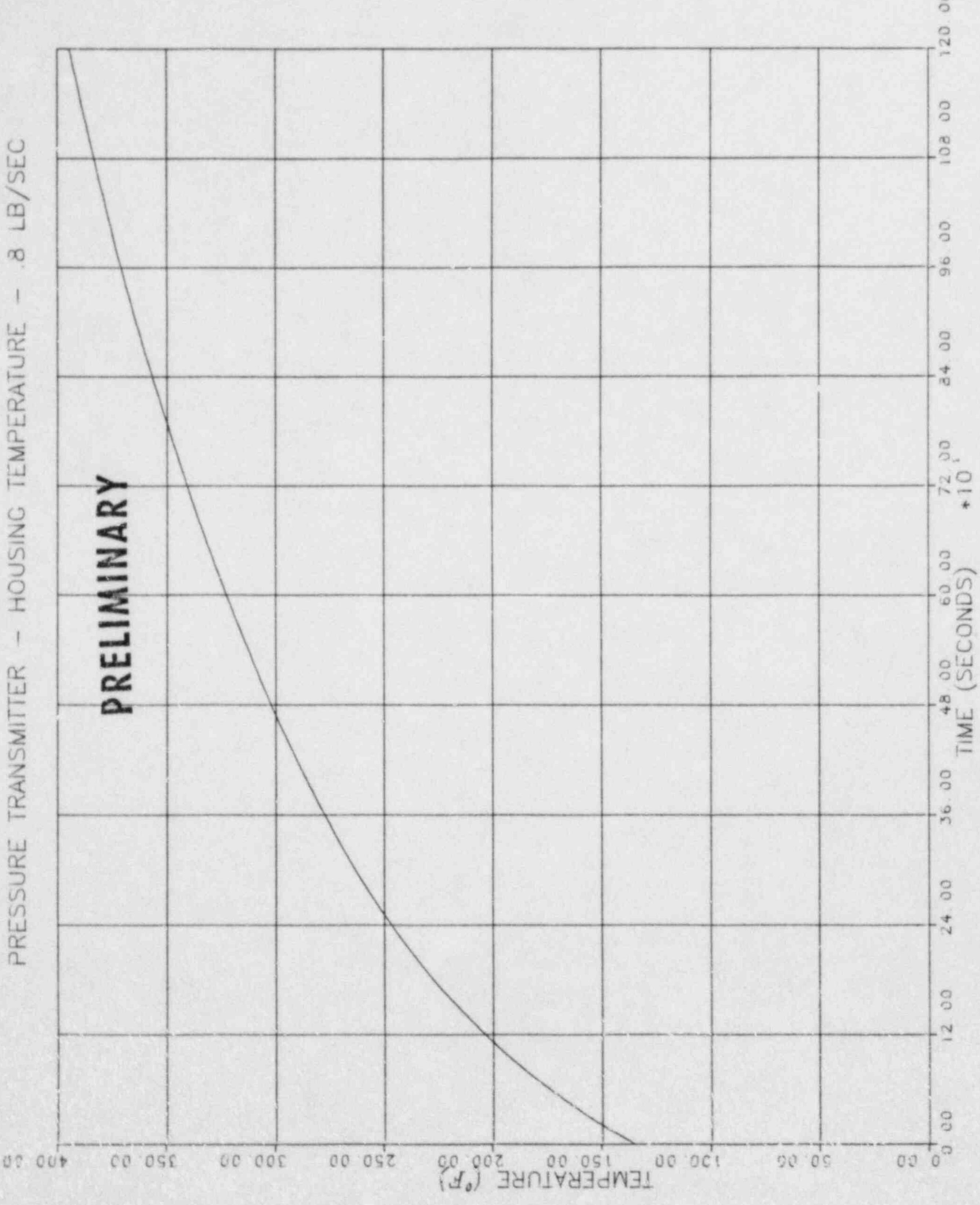
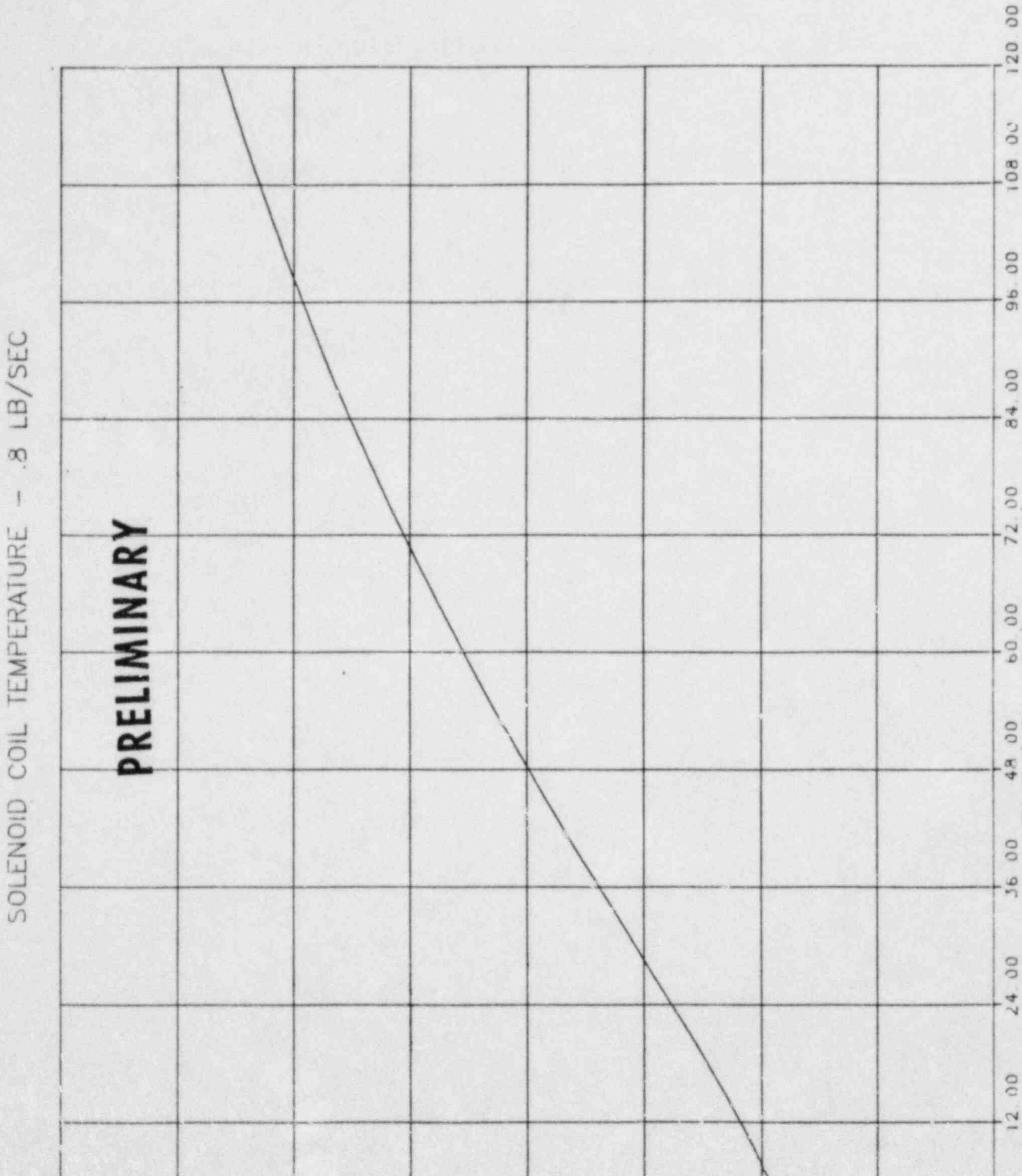
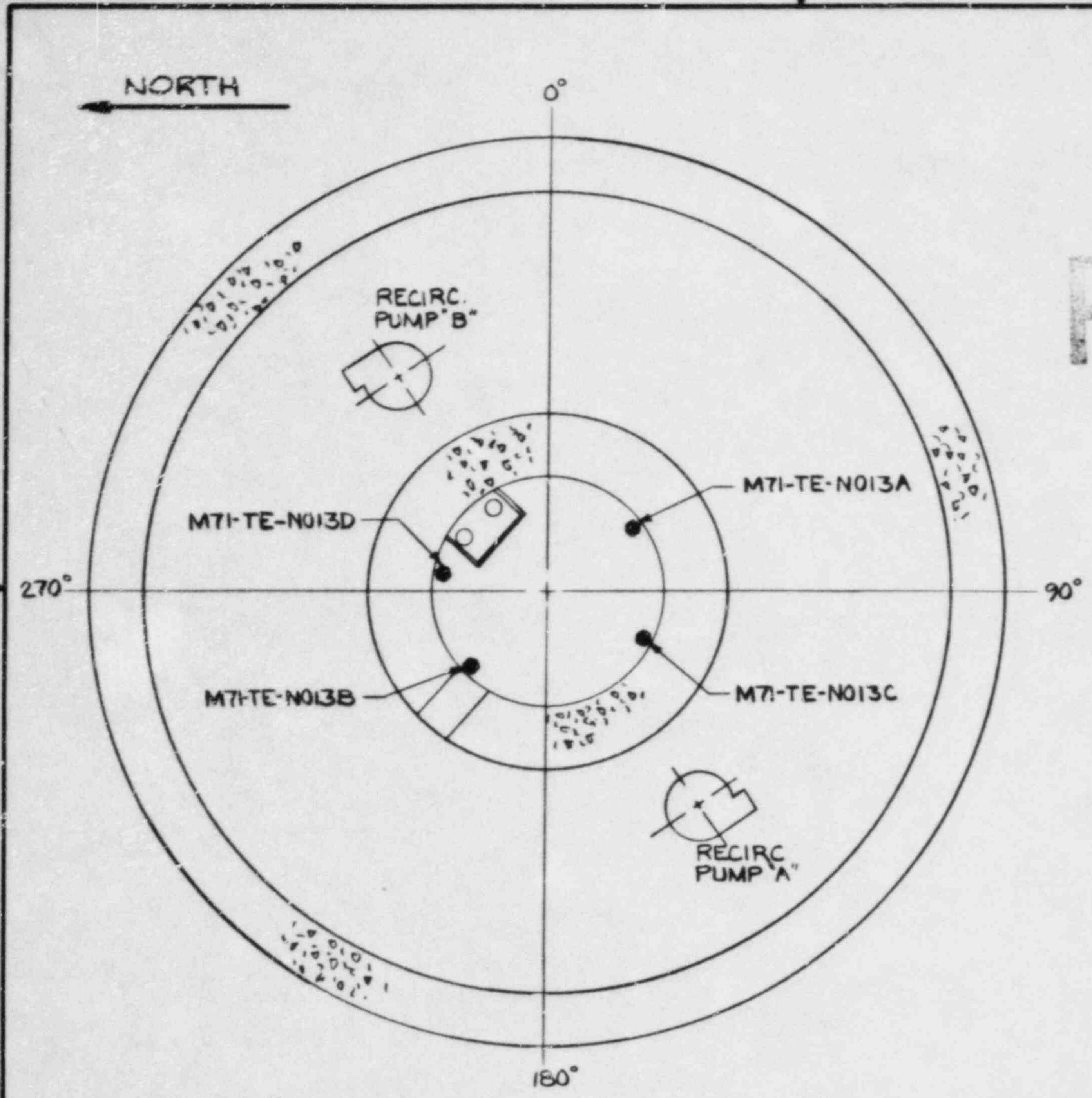

PRESSURE TRANSMITTER - HOUSING TEMPERATURE - .8 LB/SEC
PRELIMINARY
TIME (SECONDS) *10¹
0.00
12.00
24.00
36.00
48.00
60.00
72.00
84.00
96.00
108.00
120.00
TEMPERATURE (°F)
0.00
50.00
100.00
150.00
200.00
250.00
300.00
350.00
400.00

FIGURE 4


SOLENOID COIL TEMPERATURE - .8 LB/SEC
PRELIMINARY
TIME (SECONDS) *10'
0.00
12.00
24.00
36.00
48.00
60.00
72.00
84.00
96.00
108.00
120.00
TEMPERATURE (°F)
0.00
75.00
150.00
225.00
300.00
375.00
450.00
525.00
600.00




LEGEND
● INSTRUMENTATION

PRELIMINARY

FIGURE 5

| NO. | DATE | REVISIONS | BY | CH'K | DES. SUPV. | ENG'R | PROJ. ENGR. |
|---|---|---|---|---|---|---|---|
| △ | | | | | | | |
| △ | | | | | | | |
| △ | | | | | | | |

| SCALE | DESIGNED | DRAWN | CHIEF ENGR. |
|---|---|---|---|
| | | | |

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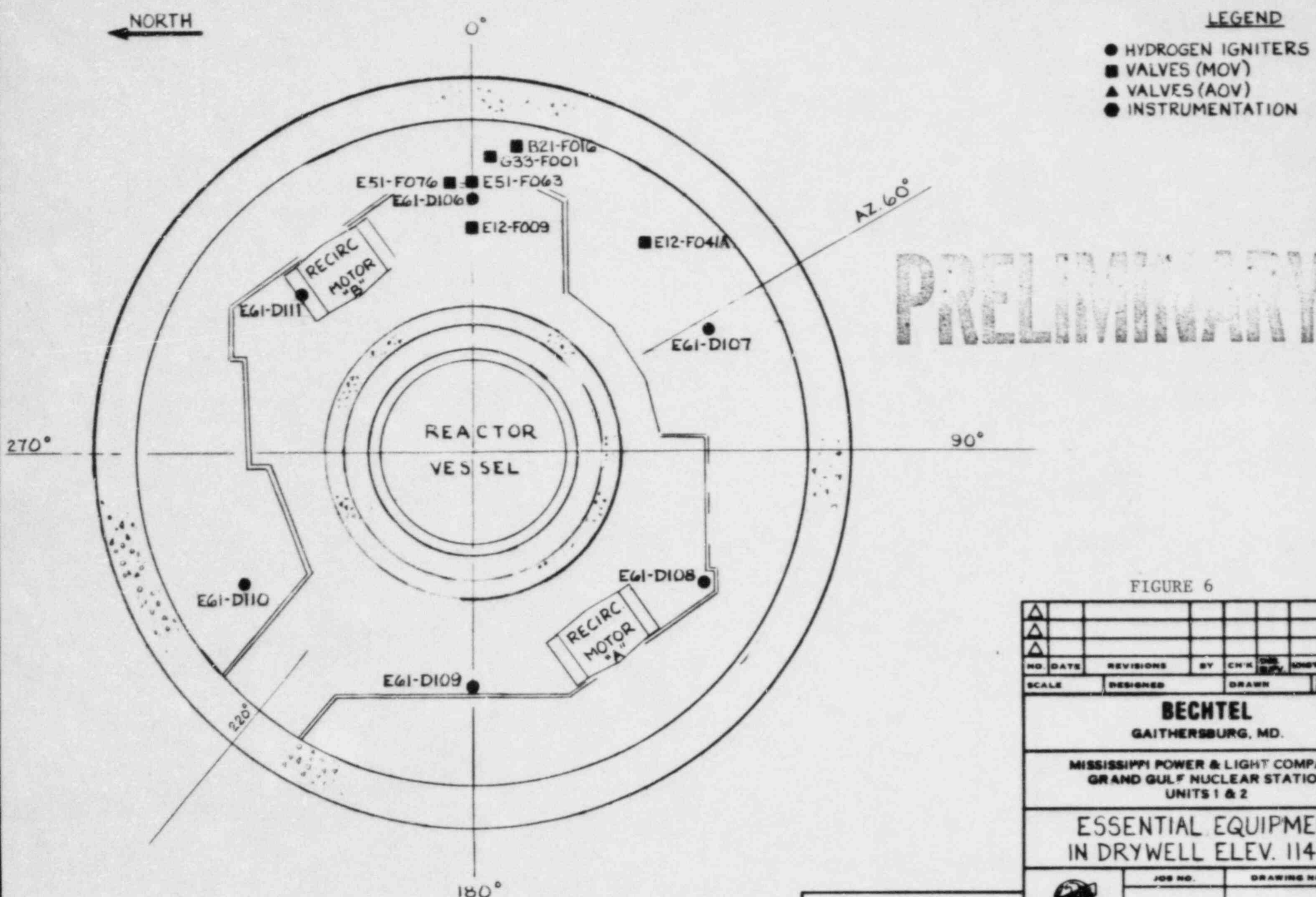
GRAND GULF NUCLEAR STATION
UNITS 1 & 2

ESSENTIAL EQUIPMENT
IN DRYWELL EL. 93'-0"

| MPL No. | | JOB NO. | DRAWING NO. | RE |
|---|---|---|---|---|
| | | | | |

11 X 17




PRELIMINARY

FIGURE 6

| NO. | DATE | REVISIONS | BY | CH'K |
|---|---|---|---|---|
| | | | | |
| | | | | |
| | | | | |

SCALE | DESIGNED | DRAWN | CHIEF ENGR.

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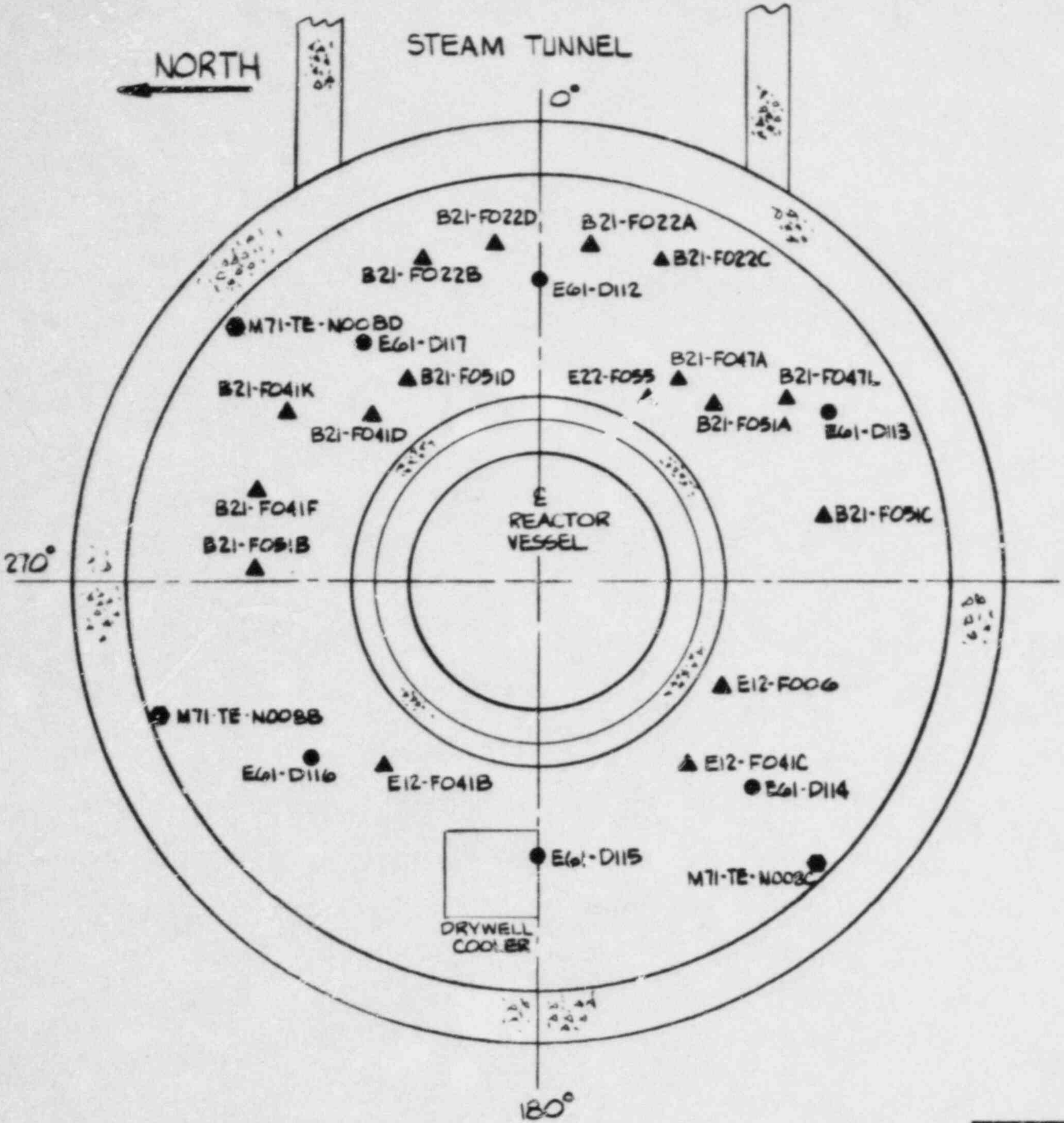
ESSENTIAL EQUIPMENT IN DRYWELL ELEV. 114'-6"

JOB NO. | DRAWING NO.

MPL No.




LEGEND

- ● HYDROGEN IGNITERS
- ■ VALVES (MOV)
- ▲ VALVES (AOV)
- ● INSTRUMENTATION

PRELIMINARY

FIGURE 7

| NO | DATE | REVISIONS | BY | CH'K | DES. SUPV. | ENG'R | PROJ ENGR. |
|---|---|---|---|---|---|---|---|
| △ | | | | | | | |
| △ | | | | | | | |
| △ | | | | | | | |

SCALE | DESIGNED | DRAWN | CHIEF ENGR.

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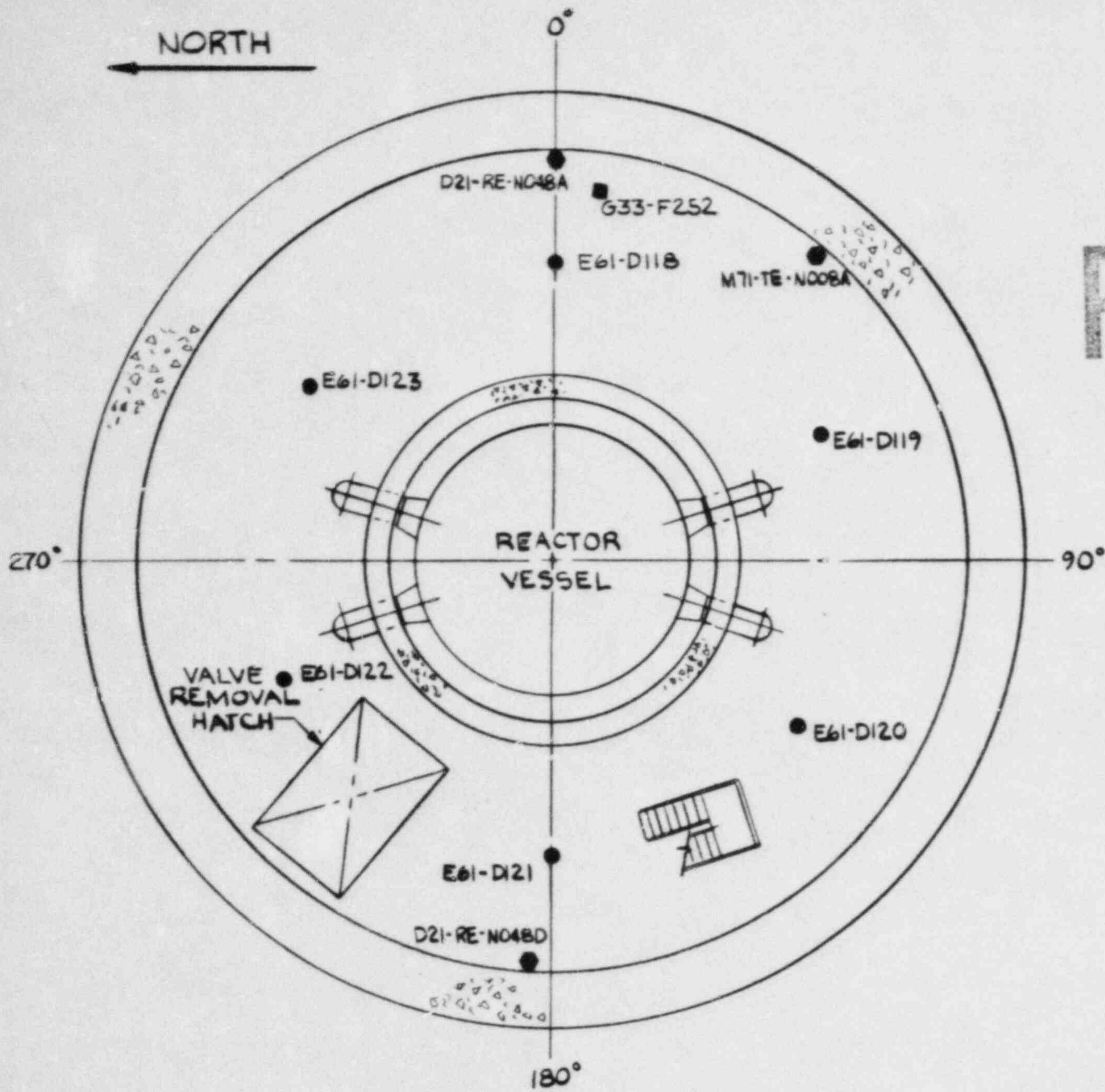
UNITS 1 & 2

ESSENTIAL EQUIPMENT IN DRYWELL EL 147'-7"

JOB NO. | DRAWING NO.

MPL No.




LEGEND
- ● HYDROGEN IGNITERS
- ■ VALVE (MOV)
- ● INSTRUMENTATION

PRELIMINARY

FIGURE 8

| NO. | DATE | REVISIONS | BY | CH'K | DES. SUPV. | ENG'R | PROJ. ENGR. |
|---|---|---|---|---|---|---|---|
| △ | | | | | | | |
| △ | | | | | | | |
| △ | | | | | | | |

SCALE | DESIGNED | DRAWN | CHIEF ENGR.

**BECHTEL**
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MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2

ESSENTIAL EQUIPMENT
IN DRYWELL EL. 161'-10"

JOB NO. | DRAWING NO.

MPL No.

# REVISED CLASIX-3 ANALYSES

## TABLE OF CONTENTS

# TABLES

# LIST OF FIGURES

## 1. Background and Summary of NRC Concerns

As part of the NRC Staff review of the adequacy of the CLASIX-3 computer program, reference (a), the Staff performed comparative analyses with the CONTEMPT-LT computer program, reference (b). Since the CONTEMPT program cannot explicitly represent hydrogen and has other limitations, a direct comparison of CONTEMPT results with those of CLASIX-3 is not appropriate. However, with certain approximations in the input to CONTEMPT, a representation of the Drywell (DW) temperature prior to hydrogen deflagration was calculated by the NRC Staff. Based on the comparison of the results of the two programs, the NRC Staff concluded in reference (c) that there appears to be some degree of non-conformance to the provisions of NUREG-0588 (reference (d)) by CLASIX-3. The material that follows addresses these concerns and concludes that an adequate margin exists based upon the excess conservatisms embodied in the analyses.

## 2. Background of the CLASIX-3 Development

The development of the CLASIX series of computer programs was undertaken as a direct response to the incident at Three Mile Island Unit 2 (TMI-2) in March 1979. The immediate concern generated by this incident was over the ability of other containments, particularly small volume pressure suppression containments such as BWR containments and PWR ice condenser containments, to maintain their integrity under a similar transient. Since containment integrity was the major concern, the assumptions selected in the development of the analytical model were based on providing a conservative estimate of the peak pressure in the containment resulting from a hydrogen deflagration.

Within a totally or partially enclosed volume, the increase in pressure due to a hydrogen deflagration will increase with the amount of hydrogen burned and the rate at which it burns. For a given set of ignition criteria, the amount of hydrogen available for combustion and rate of combustion (units of mass per unit of time) will both increase as the temperature at ignition decreases. Therefore, to provide a conservatively high pressure increase from a hydrogen deflagration, it is appropriate for the temperature in the compartment, prior to ignition, to be conservatively low. To achieve this, a conservatively high rate of heat transfer to passive heat sinks should be provided.

NUREG-0588, reference (d), as cited by the NRC in reference (c), is for the environmental qualification of safety-related electrical equipment. The methodology recommended in NUREG-0588 to determine the environment in the containment as a result of a loss of coolant accident or steam line rupture would be expected to result in conservatively high temperatures. Since the CONTEMPT program is based on the methodology presented in NUREG-0588, it is not unreasonable that CLASIX-3 would predict a lower temperature prior to deflagration than the CONTEMPT-LT program.

# 3. Additional Heat Transfer Option in CLASIX-3

To provide a more conservative thermal environment for the evaluation of equipment survivability during a degraded core accident, a new option for heat transfer to passive heat sinks has been made available in CLASIX-3. The heat transfer model represented by this option is based on a combination of those presented in NUREG-0588, Branch Technical Position CSB 6-1 (reference (e)) and the CONTEMPT program description document. The model programmed and discussed below was developed in consultation with the NRC Staff in an effort to minimize the potential for future modifications.

The condensing heat transfer coefficient is based on the Uchida correlation of reference (f). The tabular values of the coefficient as a function of the mass ratio of air to steam are presented in both the Branch Technical Position and the CONTEMPT program description document. Although the correlation is based on a mixture of air and steam, CLASIX-3 may have vitiated air and hydrogen mixed with the steam. In determining the heat transfer coefficient, the ratio of the mass of non-condensibles to the mass of steam is used in CLASIX-3. The condensing region of heat transfer is defined by the wall surface temperature being below the saturation temperature corresponding to the partial pressure of water vapor in the compartment. Under these conditions, the rate of heat transfer is given by:

$$q = h_U A \left(T_s - T_w\right) \tag{1}$$

where
- $q$ = rate of heat transfer
- $h_U$ = Uchida heat transfer coefficient
- $A$ = area of heat transfer
- $T_s$ = saturation temperature corresponding to the partial pressure of the water vapor
- $T_w$ = wall surface temperature

To provide a smooth transition from the condensing to superheated region, the rate of heat transfer is also evaluated at a constant value for the Uchida coefficient of 2, so that

$$q = 2A\ (T_B - T_W) \tag{2}$$

where $T_B$ = bulk compartment temperature.

The largest value of $q_U$ as determined by equations (1) and (2) is used.

Consistent with NUREG-0588, 92% of the condensing heat transfer is assumed to be derived from condensation and 8% is assumed to be removed directly from the bulk compartment atmosphere. The rate of condensation is

$$\dot{m}_u = 0.92\ q/(h_B - h_f) \tag{3}$$

where $\dot{m}_u$ = rate of condensation
$h_B$ = bulk enthalpy of vapor
$h_f$ = saturated liquid enthalpy corresponding to $T_s$

The condensate is assumed to be immediately removed to the sump so that there is no revaporization of condensate from the walls.

Under superheated conditions with the wall surface temperature above the saturation temperature, the film coefficient is calculated from the same correlation as that used in CONTEMPT. The film coefficient is given by

$$h_c = 0.13[p_f^2\ g\beta_f\Delta T\ C_{pf}\ k_f^2/\mu_f]^{1/3} \tag{4}$$

where $g$ = gravitational acceleration
$h_c$ = heat transfer coefficient
$p_f$ = density of gas region
$\beta_f$ = inverse of the absolute temperature of the film (assumes ideal gas)

$\Delta T$ = temperature difference between wall and bulk gas temperature

$C_{pf}$ = specific heat of gas at constant pressure

$k_f$ = thermal conductivity of gas region

$\mu_f$ = viscosity of gas region

The gas properties are evaluated at the average film temperature

$$(T_w + T_{bulk})/2$$

and the mass weighted average values assigned to the gas.

# 4. Redefined Base Case

Table 1 presents the significant changes between the previous drywell break case of reference (g) and the redefined base case. These differences and their justification are discussed below.

## 4.1 Flow Split

In the original base case, the blowdown went to the drywell until the time when the water level reached the top of active fuel, at which time, the Automatic Depressurization System (ADS) was actuated and 50% of the blowdown would be discharged directly through the spargers into the wetwell side of the suppression pool. Sample calculations have been performed to evaluate the flow split between the drywell and the spargers and the results are presented in Appendix A. Based on these results, a large fraction of the blowdown will be discharged through the spargers for only slight pressurization of the reactor system relative to the drywell. For pressure ratios (Reactor Vessel Pressure/Drywell Pressure) greater than 2 (Reactor Vessel Pressure about 55 psia) approximately 89% of the blowdown will be discharged through the spargers with eight safety valves open. The number becomes approximately 87% with seven ADS valves open. For conservatism, it was assumed for the revised base case that 30% of the blowdown was discharged into the drywell after ADS actuation. To obtain 70% of flow through the spargers, only 3 valves need to be open at a pressure ratio of 2. In fact, the 70/30 flow split can be maintained by 3 valves down to a pressure ratio of 1.2. For a drywell pressure of 25 psia, the reactor vessel pressure need only be 30 psia or 5 psig relative to the drywell to maintain the flow split.

## 4.2 Time of Flow Split

Previously, it was assumed that all ADS valves opened simultaneously at 20 minutes after initiation of the transient. Due to concern over the timing of the opening of the valves and considering that a delayed actuation of ADS would result in a more severe environment for equipment survivability in the drywell, and be more consistent with plant operating procedures, the delay from initiation of the transient to ADS has been increased by 50% to 30 minutes. Considering that only three valves, as discussed above, need to be opened to achieve the assumed flow split, and that plant emergency procedures require ADS actuation when the water level reaches core midplane which is approximately 30 minutes for a stuck open SRV, the 30 minute delay is appropriately conservative.

### 4.3 Heat Transfer Correlation and Condensation

The original analyses were based on a heat transfer correlation that resulted in a conservatively high pressure transient. Since the peak pressures have not posed a challenge to the containment integrity, the emphasis has shifted to equipment survivability. To calculate a conservatively high thermal environment, new options, as discussed above in Section 3, have been made available and were used in this analysis.

### 4.4 Combustible Gas Control System (CGCS) Flow Rate

The original analysis was based on two compressors at a nominal flow rate of 500 cfm each or a total greater than 1000 cfm. Since the original analyses, tests under typical plant conditions indicate each compressor has a capacity of greater than 1050 cfm. The redefined base case considers one compressor operating at a constant flow of 1050 cfm. Additional work is in progress to evaluate the effect of 2 compressors.

### 4.5 Vacuum Breaker Flow Area

Re-evaluation of the vacuum breaker piping arrangement indicated that an incorrect flow area had been used in prior analyses. Although the error has been determined to have had negligible impact, it has been corrected in the redefined base case. Each vacuum breaker penetration is modeled at a flow area of 0.55 $ft^2$. A review of the overall modeling of the vacuum breakers is in progress and will be submitted later.

4.6 Heat Capacity of Concrete

The heat capacity of the concrete in the original analysis was based on a handbook value that was found to be more than a factor of 4 too low. This error has been determined to result in slightly conservative peak pressures but has been corrected in the redefined base case.

4.7 Drawdown

During the original analyses, the critical portion of the transient occurred as a result of hydrogen deflagration in the drywell long after hydrogen generation ceased. Thus, the exact timing of the dump of the upper pool to the suppression pool and the removal of water from the suppression pool to refill the reactor system was not important, provided that they were both complete prior to the hydrogen burn in the drywell. In the present analysis, the pressure in the drywell might have an impact on the temperatures. Therefore, a more mechanistic representation is required. For the redefined base case, it is assumed that the upper pool dump occurs at 30 minutes. It is further assumed that the drawdown (1888 gpm) is initiated at a time (7,361 seconds into the transient) such that a volume (14,040 $ft^3$) equivalent to the volume of the reactor vessel has been removed from the suppression pool continuing at a constant rate, to simulate flow out the break, until the holdup volume inside the weir wall is full. As a test of sensitivity to drawdown, a run was made that filled the vessel to one-third full, corresponding to a steam cooling level. The drawdown (4,680 $ft^3$) of 1888 gpm was initiated at 7658 and terminated at the same time as the base case run.

4.8 Radiant Heat Transfer

Radiant heat transfer from the steam in the atmosphere to the walls is the same in both analyses. An evaluation is in progress to respond to an NRC comment concerning the modeling of the radiant transfer rate.

Table 1

Redefinition of Drywell Break Base Case

| Modification | Original Base Case | Redefined Base Case |
|---|---|---|
| Flow Split | 50/50 | 70/30 |
| Time of Flow Split (sec) | 1200 | 1800 |
| Heat Transfer Correlation | Tagami Based | Uchida Based |
| Compressor Flow | 2 @ 500 cfm each | 1 @ 1050 cfm |
| Vacuum Breaker Flow Area ($ft^2$) Per Breaker | 1.09 | 0.55 |
| Heat Capacity of Concrete ($Btu/ft^3$-F) | 6.24 | 28.8 |
| Drawdown | At Upper Pool Dump | Mechanistic |
| Radiant Heat Transfer | Original | Original |

# 5. Results

## 5.1 Analytical Results of the Redefined Drywell Break Base Case

The results of the CLASIX-3 analysis of the redefined drywell break base case are shown in Figures 1 to 27. Figures 1 to 6 show the results through the end of hydrogen release and Figures 7 to 27 extend the results until a burn occurs in the drywell. Based on the criteria selected for ignition, a burn would not occur in the containment volume. For conservatism and to provide a better comparison with the original base case, the criterion for ignition in the containment was reduced to 7% hydrogen to result in a burn in the containment to closely follow the burn in the drywell. As shown in Figure 21, the hydrogen rapidly increases in the containment as a result of the combustion in the drywell and reaches the 7% hydrogen criterion in the containment.

A comparison of significant results from both the original and redefined base cases are provided in Table 2. There is an apparent anomaly in the drywell pressure in the redefined base case. The slightly higher drywell pressure prior to cessation of hydrogen release in the redefined base case is a result of the treatment of drawdown. As a result of the delay in initiation of drawdown, there is a higher static head in the wetwell and consequently a higher drywell pressure is required to vent. The effect of initiation of drawdown is apparent in the downward trend in the drywell pressure between 7361 seconds and 7807 seconds in Figure 4. (The sharp drop in pressure beyond 7807 seconds in Figure 10 is due to the effect of water spraying out of the drywell break.)

The results beyond the cessation of hydrogen release are not directly comparable because the burn in the containment was initiated at different conditions for the two transients.

The Combustible Gas Control System at the Grand Gulf Nuclear Station, has three basic flow paths, exclusive of the suppression pool, which connect the containment and the drywell. In each flow path there are isolation valves, check valves to prevent flow from the drywell to the containment and, in two of the flowpaths, a compressor, in parallel with the check valves, with suction from the containment and discharge to the drywell. With the compressor in operation, the high pressure discharge will prevent the check valve from operating. However, if the compressor is inoperable and the isolation valves open, the check valves could open if the containment pressure exceeded the drywell pressure.

In the revised base case, the only time the check valves operate to allow flow from the containment to the drywell is during the forced containment burn at the end of the transient. Since the actuation of the check valves bypassing the one inoperable compressor would only serve to mitigate the consequences of the containment burn, it was conservatively assumed that the mode of failure of the second compressor was the failure of the isolation valves associated with the second compressor.

## 5.2 Drywell Temperature - No Flow Split

To evaluate the drywell temperature as a function of the time of the initiation of the flow split, a short transient was analyzed to determine the temperature response of the drywell with all the discharge entering directly into the drywell atmosphere. The temperature and pressure in the drywell are shown in Figures 28 and 29 respectively. After the first temperature peak at about 150 seconds, the temperature rises at a fairly uniform rate. Most of the perturbation from 1800 seconds to just beyond 2000 seconds is a result of the upper pool dump. This is shown dramatically in the pressure trace as the drywell pressure rapidly increases by about 2.5 psi due to the increase in static head.

The main conclusion drawn from this analysis is that actuation of at least some of the SRVs early in the transient is beneficial.

## 5.3 Inverted Diffusion Flame in the Drywell

The results of an inverted diffusion flame in the drywell are shown in Figures 30 to 50. This transient utilized the same ignition criteria for the diffusion flame as those in the drywell sensitivity study [Reference (g)].

The ignition occurred at approximately 5200 seconds into the transient. Prior to this time, the transient is identical to the redefined base case. The peak temperature in the drywell is 355°F compared to 346°F for the original results in Reference (g).

Other than the impact on the drywell temperature, there is negligible impact on other temperatures, pressures or gas concentrations.

## 5.4 Modified Drawdown

The major effect of this modification is to slightly increase the pressure in the drywell over the latter portion of the transient. There are minor increases in some of the peak pressures and temperatures as well. The most significant impact is on the burn that initiates in the drywell near the end of the transient. The reduced drawdown leaves more water in the suppression pool and therefore increases the pressure in the drywell just prior to ignition. The increased pressure will require more mass of hydrogen to remain in the drywell and consequently a larger mass of oxygen is required to reach the ignition criterion. Thus, the drywell burn is more severe and results in slightly more severe consequences from the subsequent burns in the wetwell and containment.

## 5.5 Additional Analyses Planned

Assessment of the work completed for the revised base case is in progress. Several sensitivities will be evaluated including a two compressor case, modified vacuum breaker treatment case, and the impact of a modified source term. The modified source term will be developed to more accurately represent a degraded core accident rather than a severe accident which is beyond the scope of these analyses. It is believed the modified source term, in addition to being more consistent with program objectives, will result in less energy being added to the drywell and thus reduce both the preburn and postburn temperatures.

## Table 2

Comparison of Base Cases

| | | Original | Redefined |
|---|---|---|---|
| Number of | DW | 0 [1] | 0 [1] |
| Burns | WW | 26 [6] | 52 [4] |
| | CT | 0 [1] | 0 [1] |
| Total $H_2$ | DW | 0 [104] | 0 [98] |
| Burned | WW | 1233 [319] | 1517 [1725] |
| (Lbs) | | | |
| | CT | 0 [587] | 0 [484] |
| $H_2$ | DW | 712 [240] | 569 [197] |
| Remaining | WW | 21 [15] | 28 [18] |
| (Lbs) | | | |
| | CT | 629 [114] | 499 [92] |
| Peak T | DW | 296 [707] | 323 [760] |
| (F) | WW | 1110 [2295] | 1192 [2274] |
| | CT | 196 [860] | 176 [898] |
| Peak P | DW | 12.3 [16.3] | 14.1 [14.0] |
| (psig) | WW | 11.9 [31.6] | 11.8 [30.7] |
| | CT | 11.7 [32.1] | 9.1 [30.7] |

The values in [ ] occur as a result of forcing a burn at a $H_2$ concentration of 7% to maximize combustion.

## 6. References

a. "The CLASIX-3 Computer Program for the Analysis of Reactor Plant Containment Response to Hydrogen Release and Deflagration," WCAP-10259 (Proprietary) dated March 1983 and WCAP-10260 (Non-proprietary) dated March 1983.

b. "CONTEMPT-LT - A Computer Program for Predicting Containment Pressure - Temperature Response to a Loss of Coolant Accident," ANCR-1219, UC-78, dated June 1975.

c. NRC letter Docket No.: 50-416, "Mark III BWR Hydrogen Control," dated July 22, 1983.

d. "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," NUREG-0588 dated December 1979.

e. "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," Branch Technical Position CSB 6-1, dated July 1981.

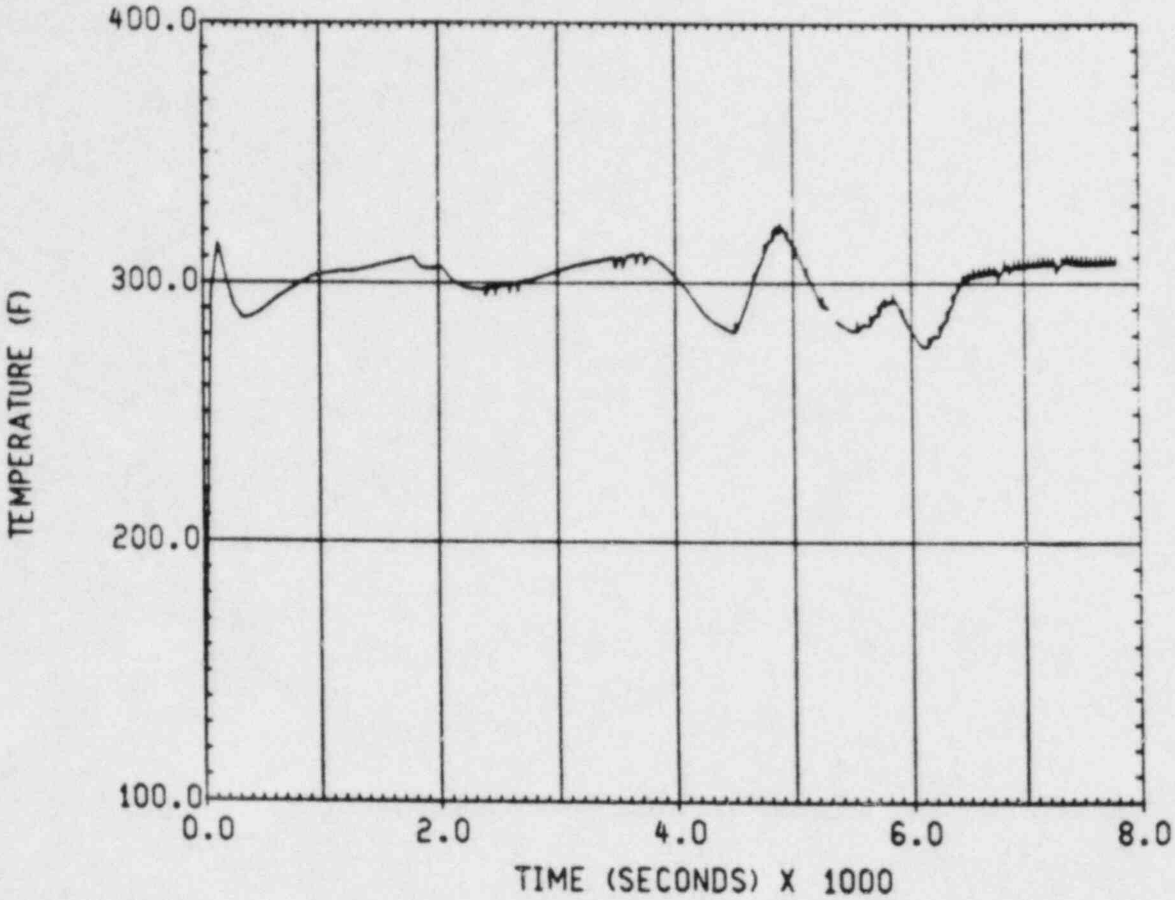
f. H. Uchida, A. Oyama, and T. Toga, "Evaluation of Post-Incident Cooling Systems of Light-Water Power Reactors," Proc. Third International Conference on the Peaceful Uses of Atomic Energy, Volume 13, Session 3.9, United Nations, Geneva (1964).

g. AECM-83/0212 - MP&L submittal of the Drywell Break Sensitivity Study, OPS-38A54 Revision C, dated May 11, 1983 (MP&L Proprietary).

Table 3

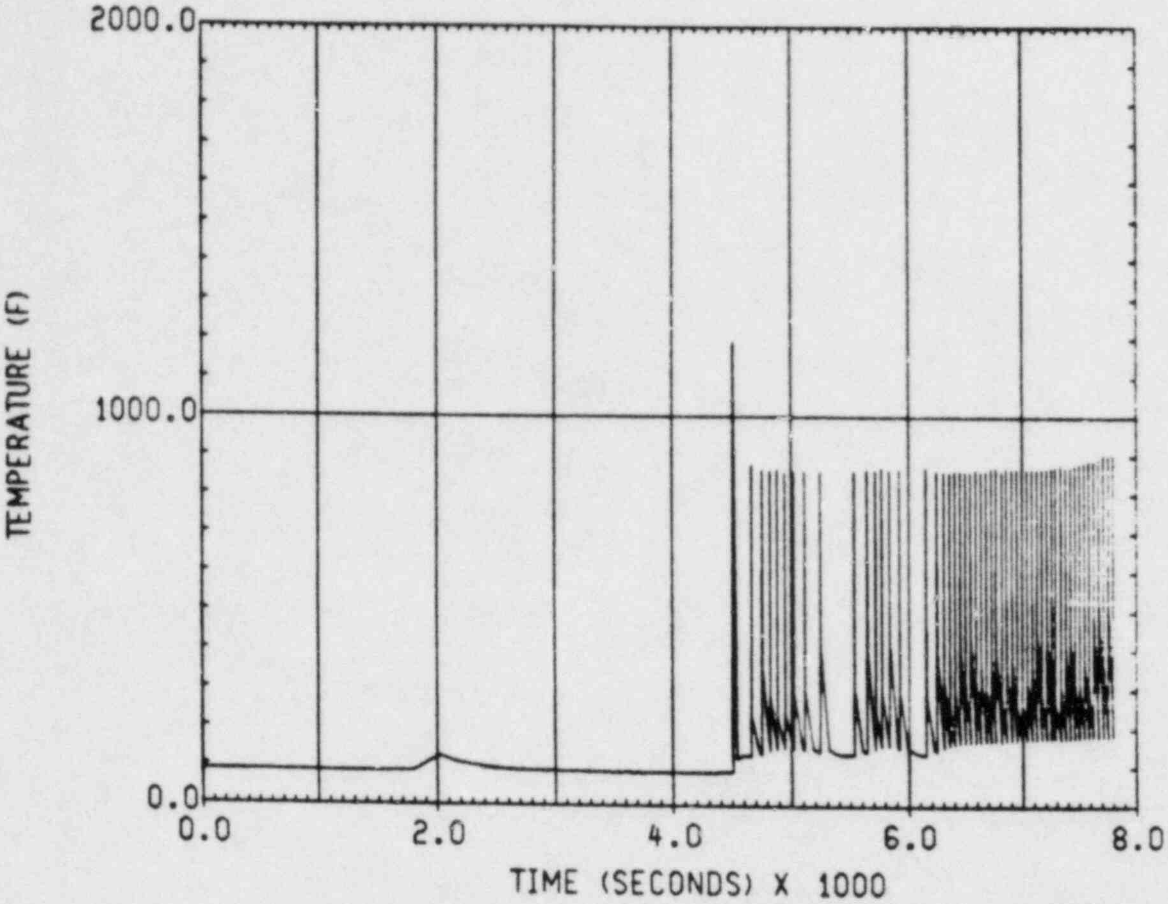
SUMMARY OF SENSITIVITIES

| | DA4 | DC1 Base Case | DC2 Contin Burn | DC3 No Split | DC4 New Draw |
|---|---|---|---|---|---|
| Transient Time | 12,300 | 11010 | 8010 | 2410 | 10010 |
| Mass, Energy, and $H_2$ Flow Split | 50/50 | 70/30 | 70/30 | 100/0 | 70/30 |
| Time of Flow Split | 1200 sec | 1800 s | 1800 s | -- | 1800 s |
| Heat Transfer Correlation | Tagami | Uchida | Uchida | Uchida | Uchida |
| Condensation Factor Used | No | Yes | Yes | Yes | Yes |
| Emissivity Included | Yes | Yes | Yes | Yes | Yes |
| Heat Capacity of Concrete | 6.24 $BTU/ft^3$-F | 28.8 | 28.8 | 28.8 | 28.8 |
| Fan Flow Values | Variable Table | 1050 (CFM) Constant | 1050 Constant | 1050 Constant | 1050 Constant |
| Number of Compressors | 2 | 1 | 1 | 1 | 1 |
| Flow Area Per Vacuum Breaker | 1.09 $ft^2$ | 0.55 | 0.55 | 0.55 | 0.55 |
| Time of Drawdown | 1800 sec | 7361 s | 7361 s | 7361 s | 7658 s |
| Drawdown Volume Into Reactor Vessel | 14040 $ft^3$ | 14040 | 14040 | 14040 | 4680 |
| Continuous Burn | No | No | Yes | No | No |



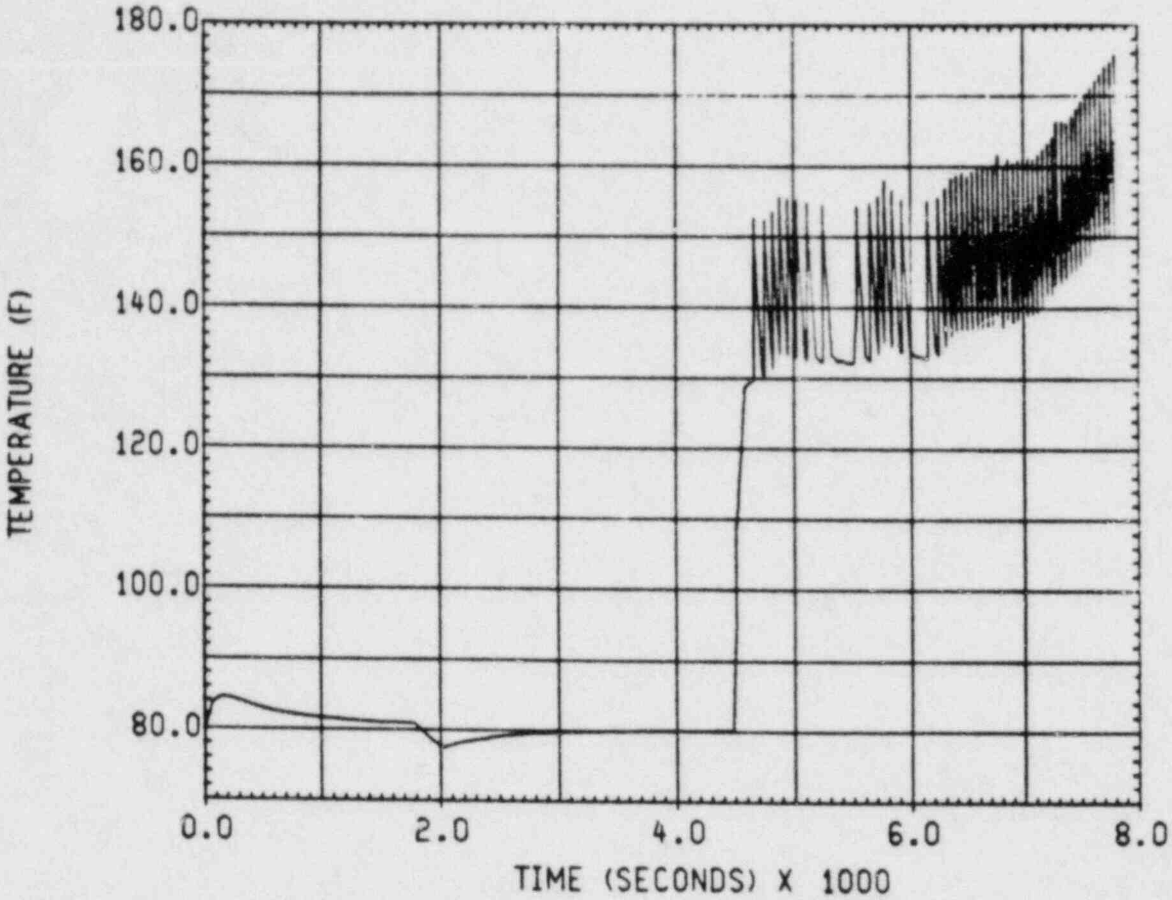

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
DRYWELL TEMPERATURE

FIGURE 1



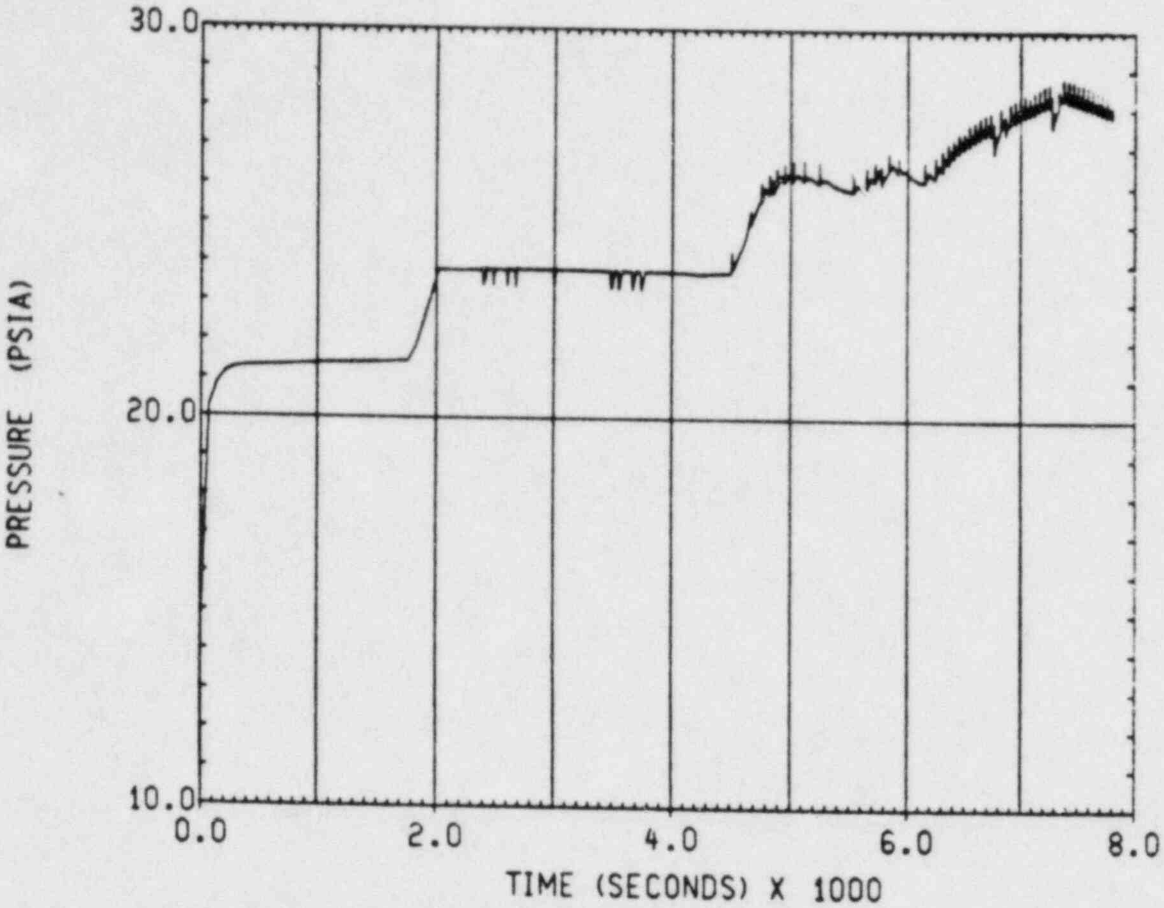

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
WETWELL TEMPERATURE

FIGURE 2



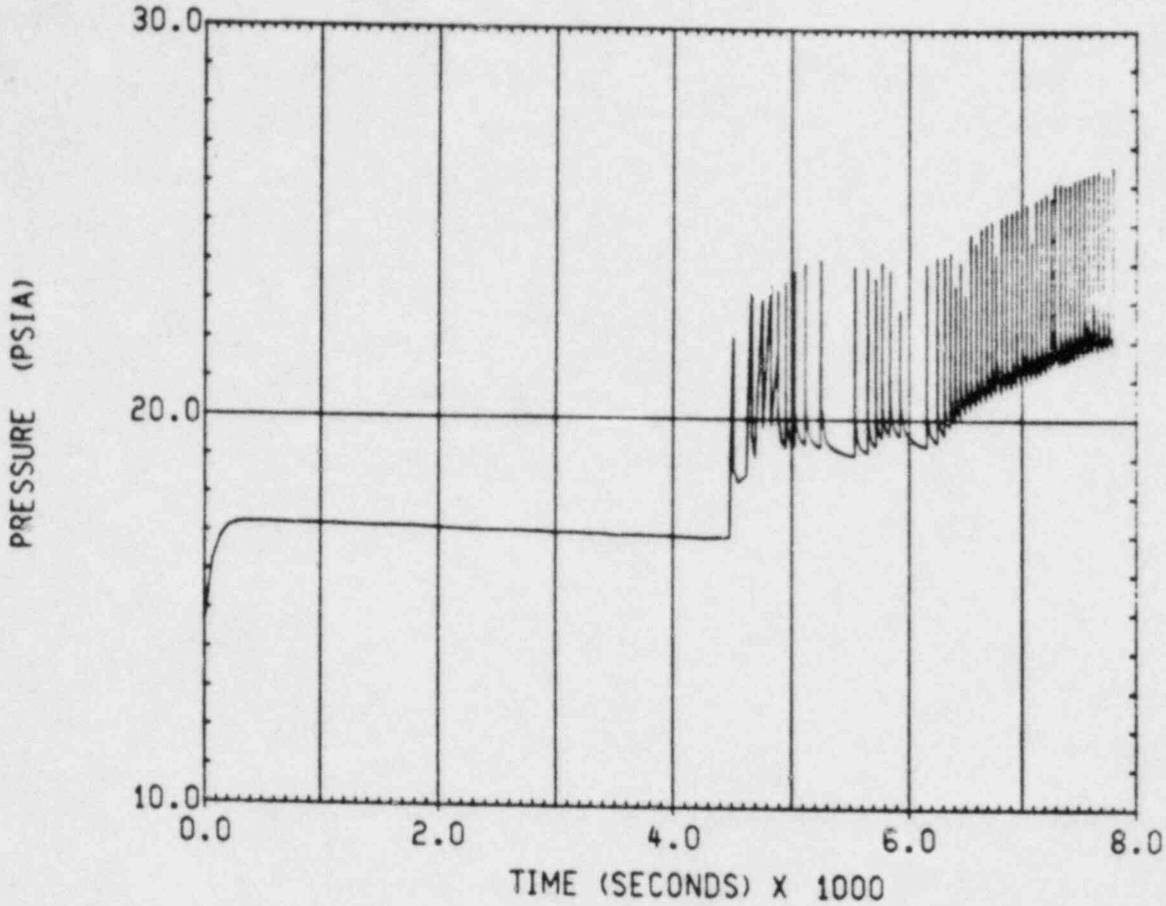

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
CONTAINMENT TEMPERATURE

FIGURE 3



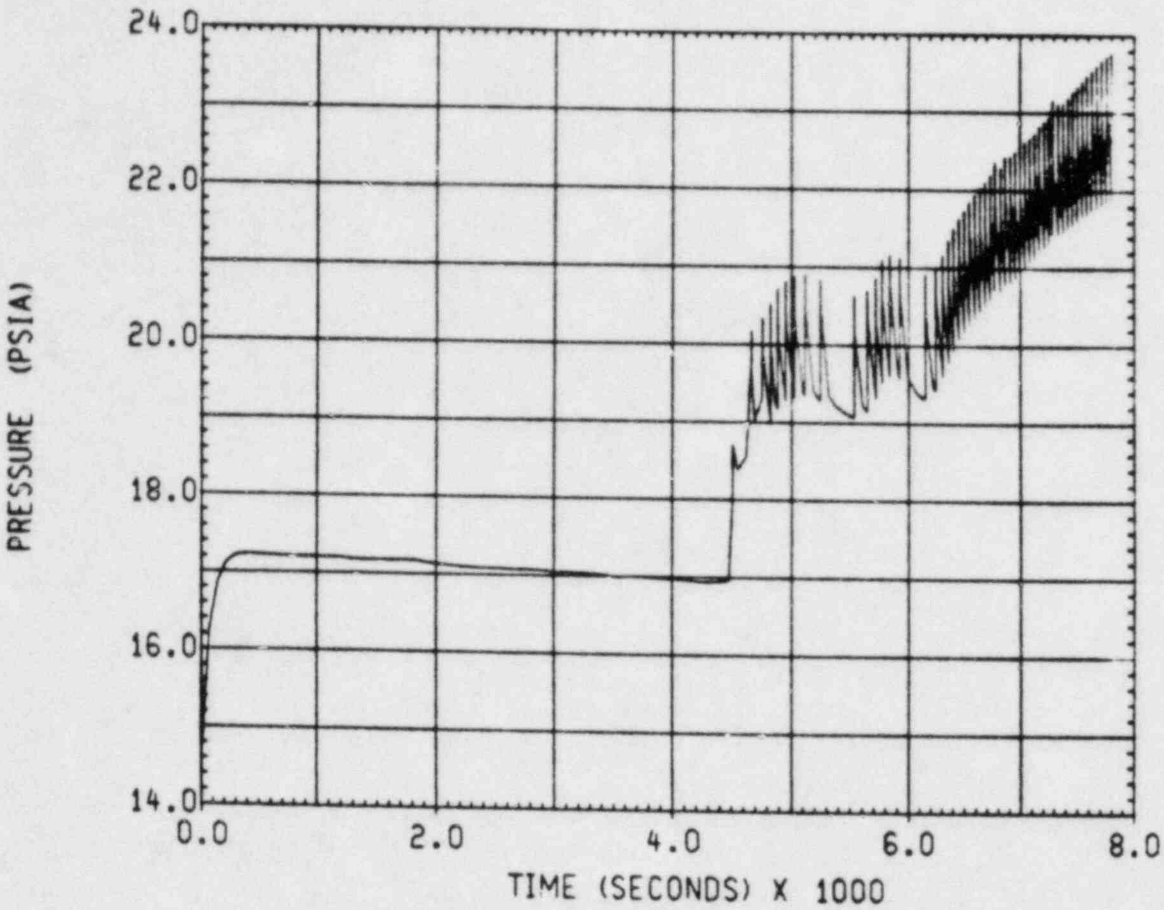

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
DRYWELL PRESSURE

FIGURE 4



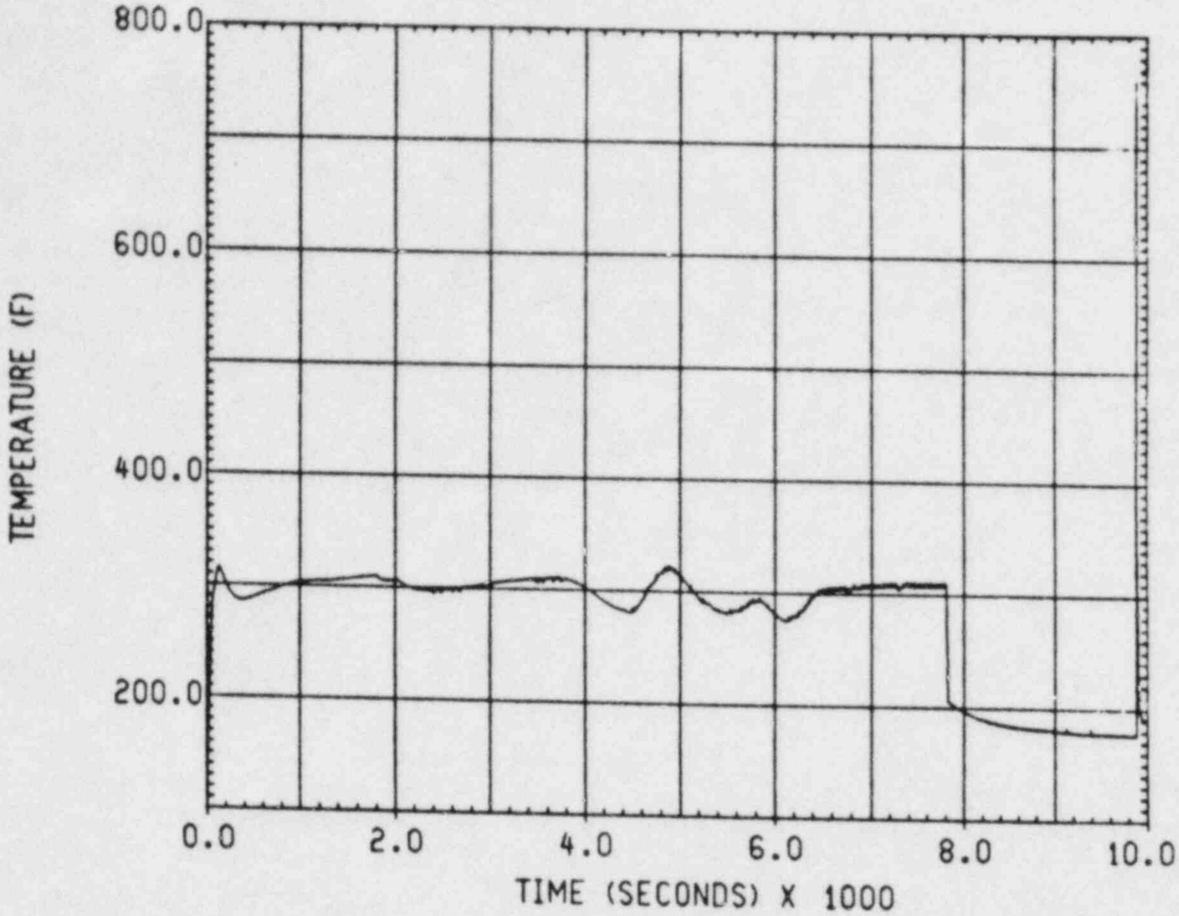

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
WETWELL PRESSURE

FIGURE 5



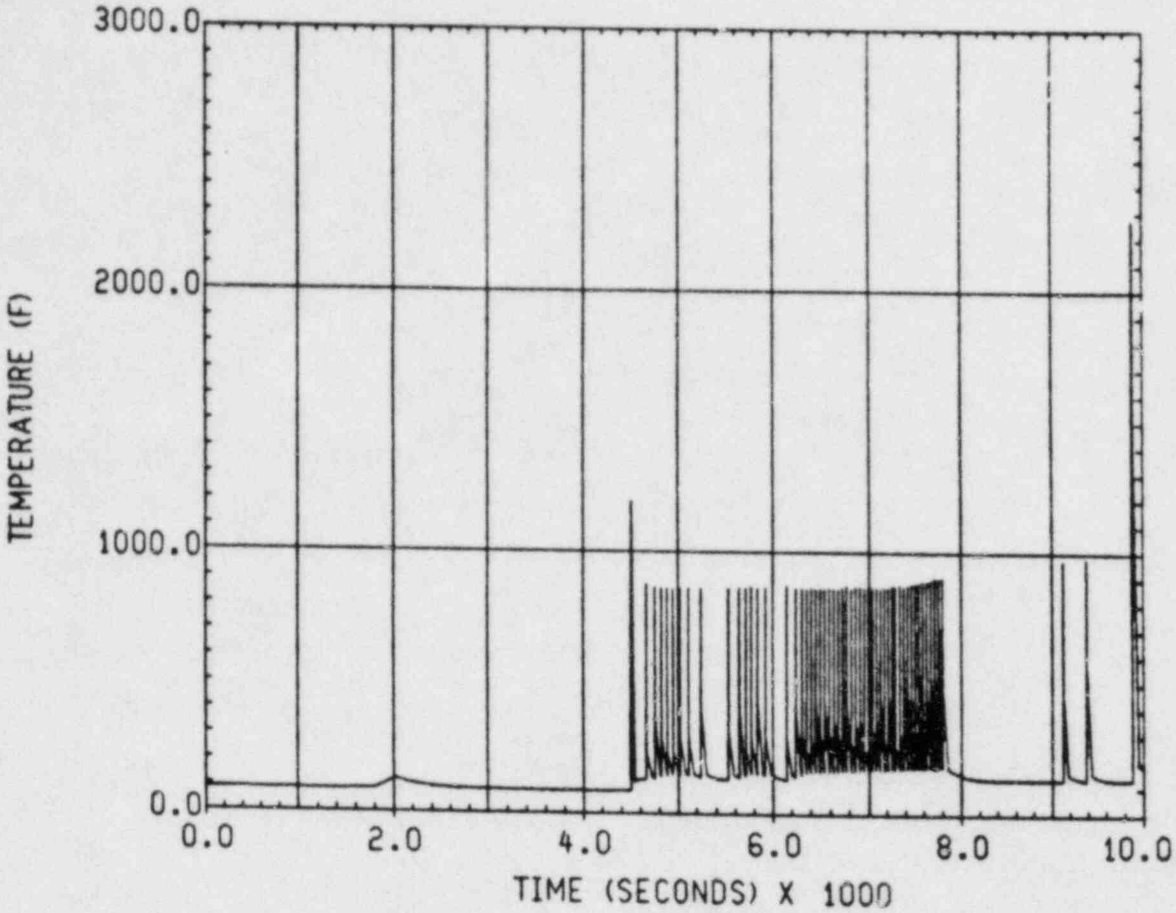

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
CONTAINMENT PRESSURE

FIGURE 6



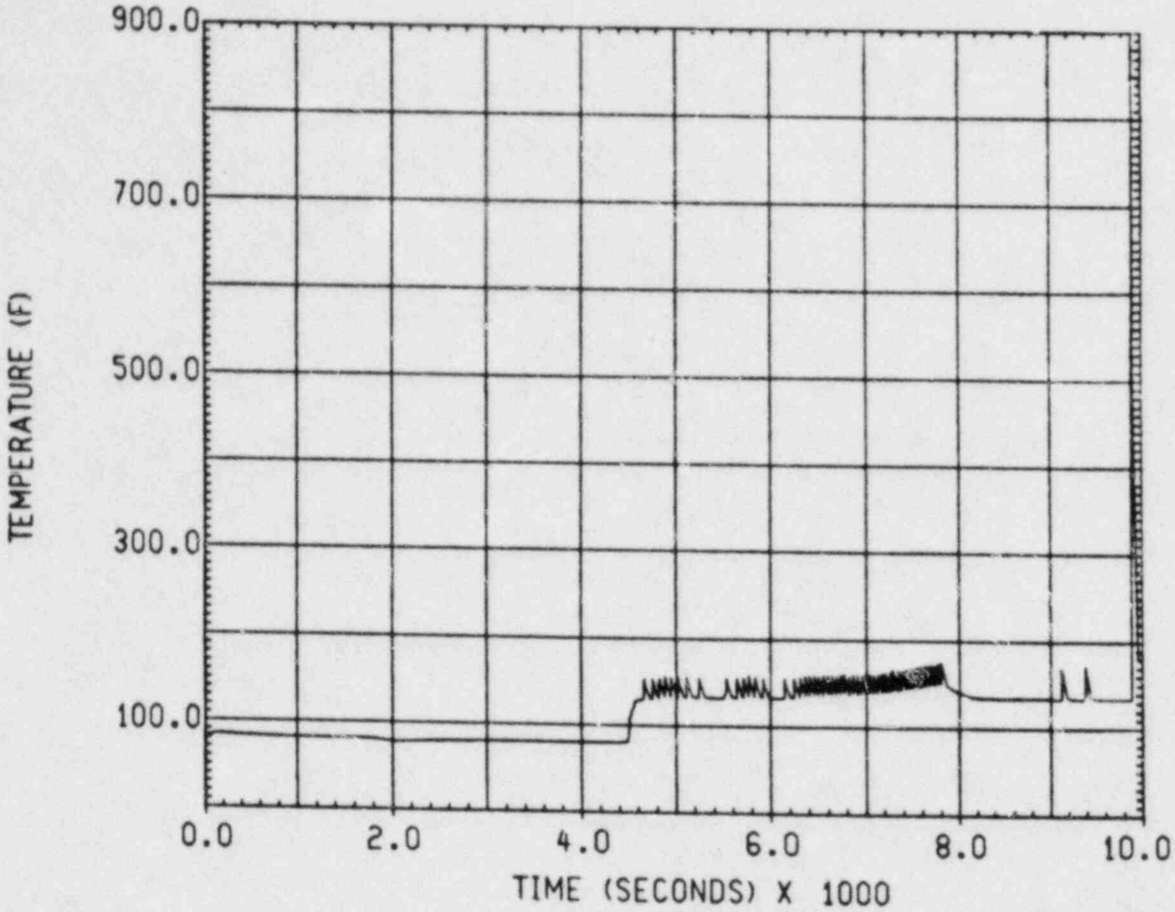

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
DRYWELL TEMPERATURE

FIGURE 7



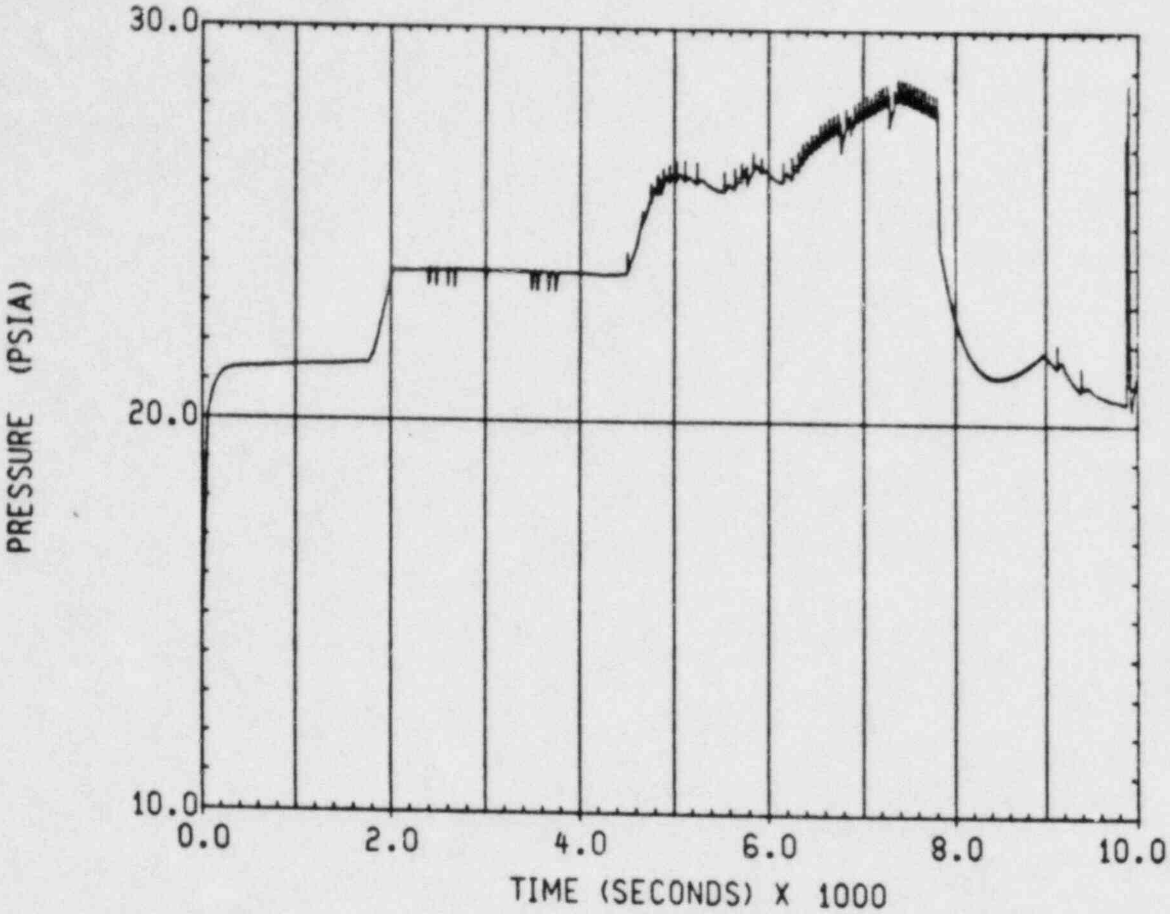

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
WETWELL TEMPERATURE

FIGURE 8



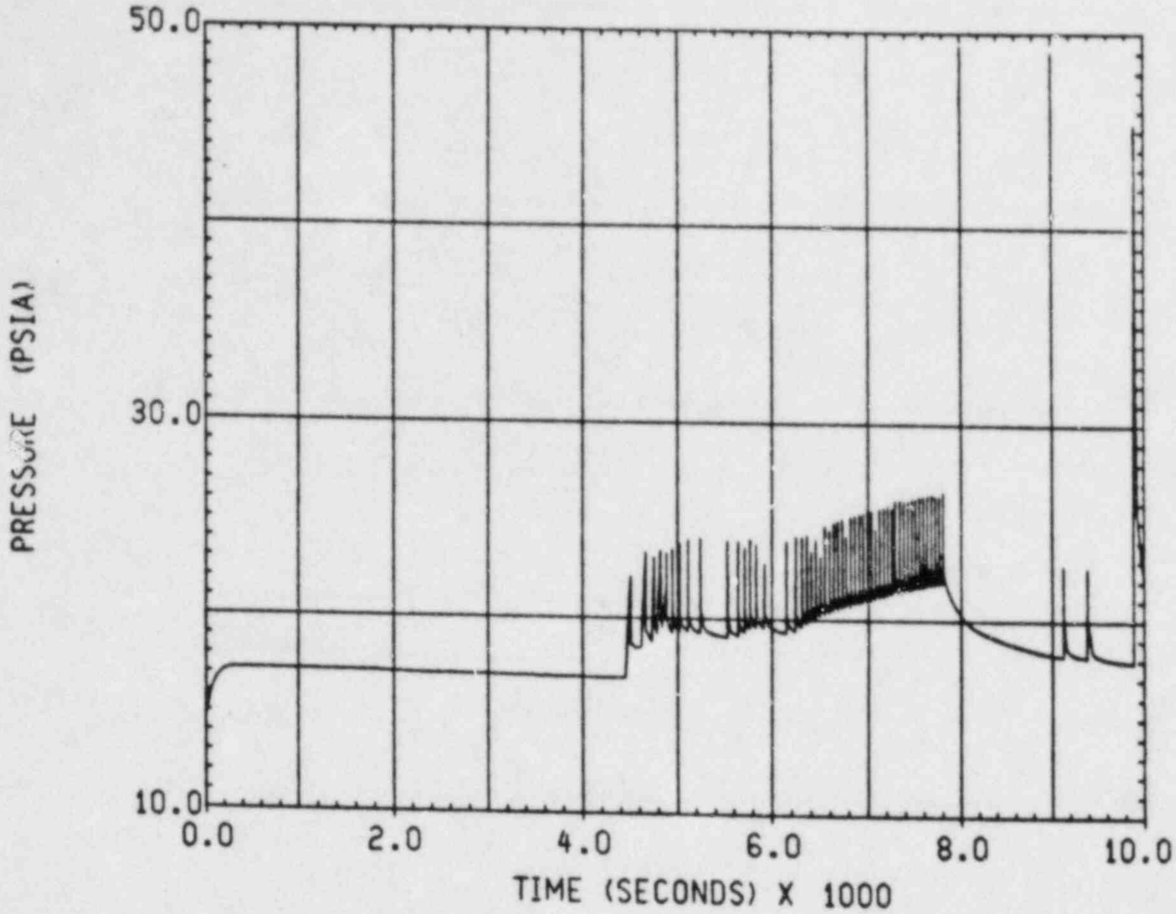

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
CONTAINMENT TEMPERATURE

FIGURE 9



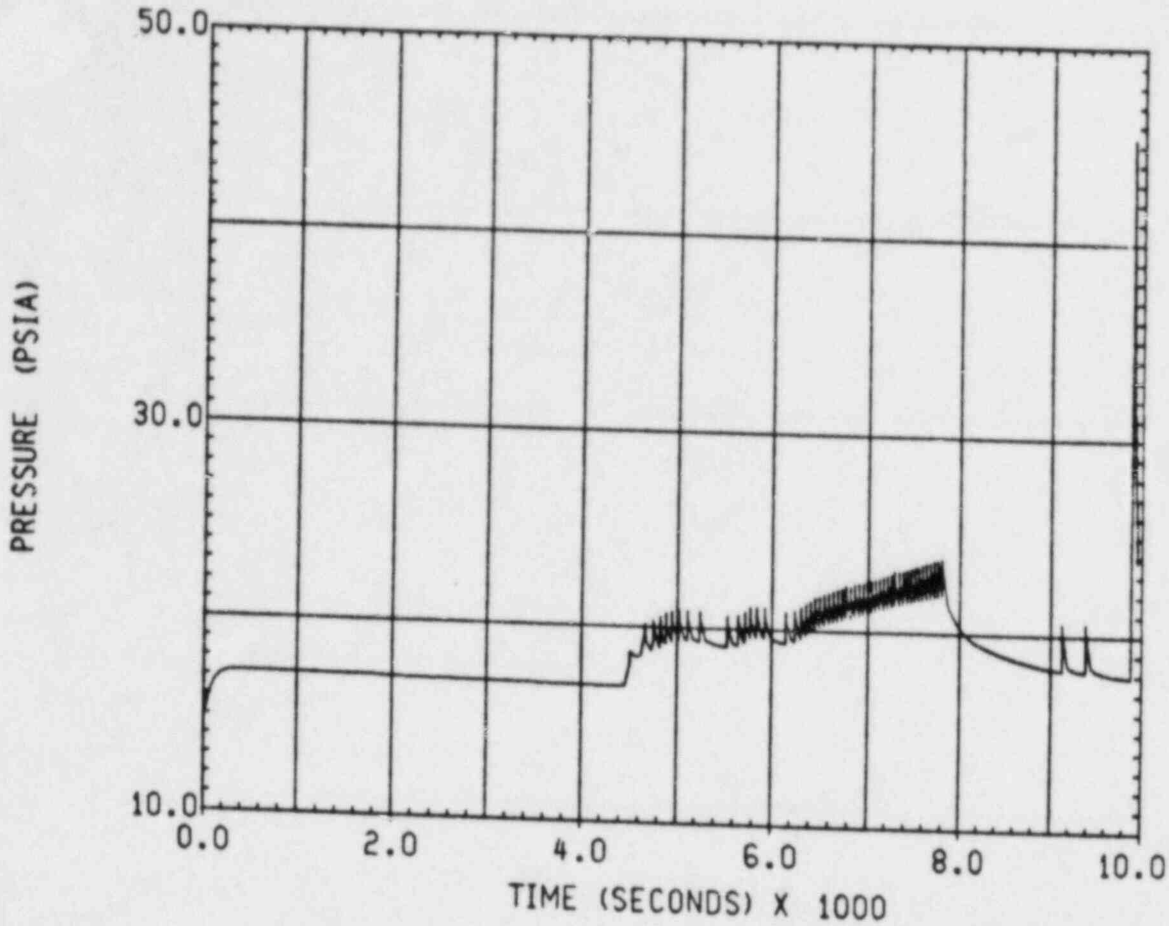

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
DRYWELL PRESSURE

FIGURE 10



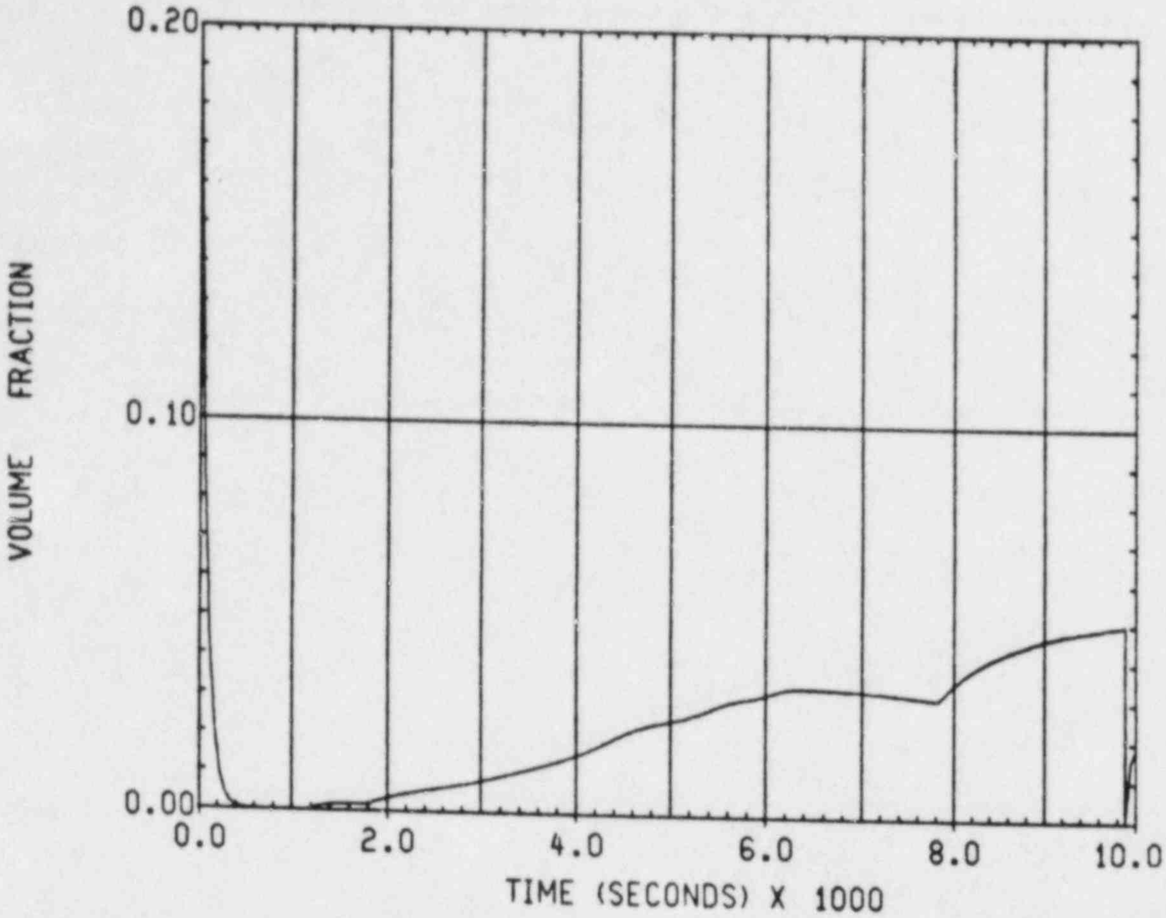

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
WETWELL PRESSURE

FIGURE 11



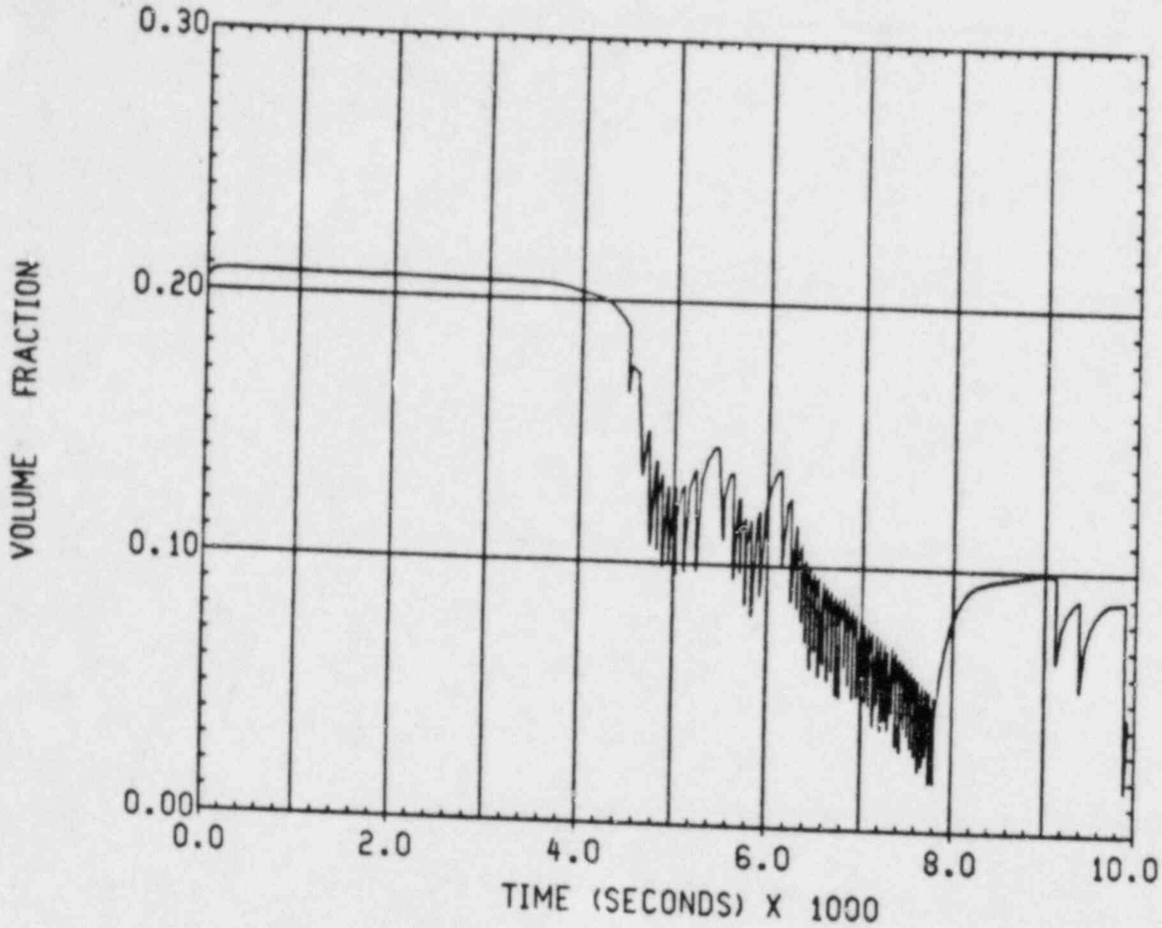

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
CONTAINMENT PRESSURE

FIGURE 12



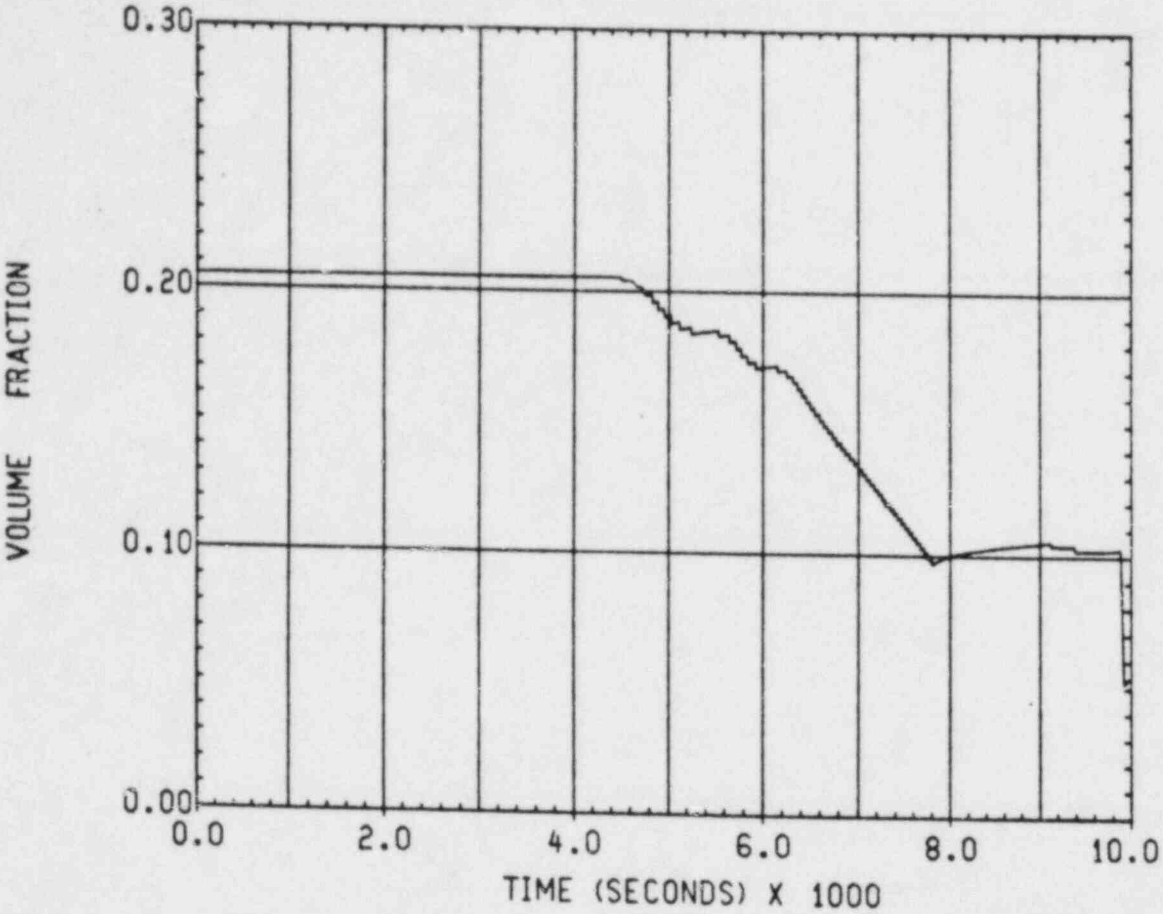

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
DRYWELL O2 GAS CONCENTRATION

FIGURE 13



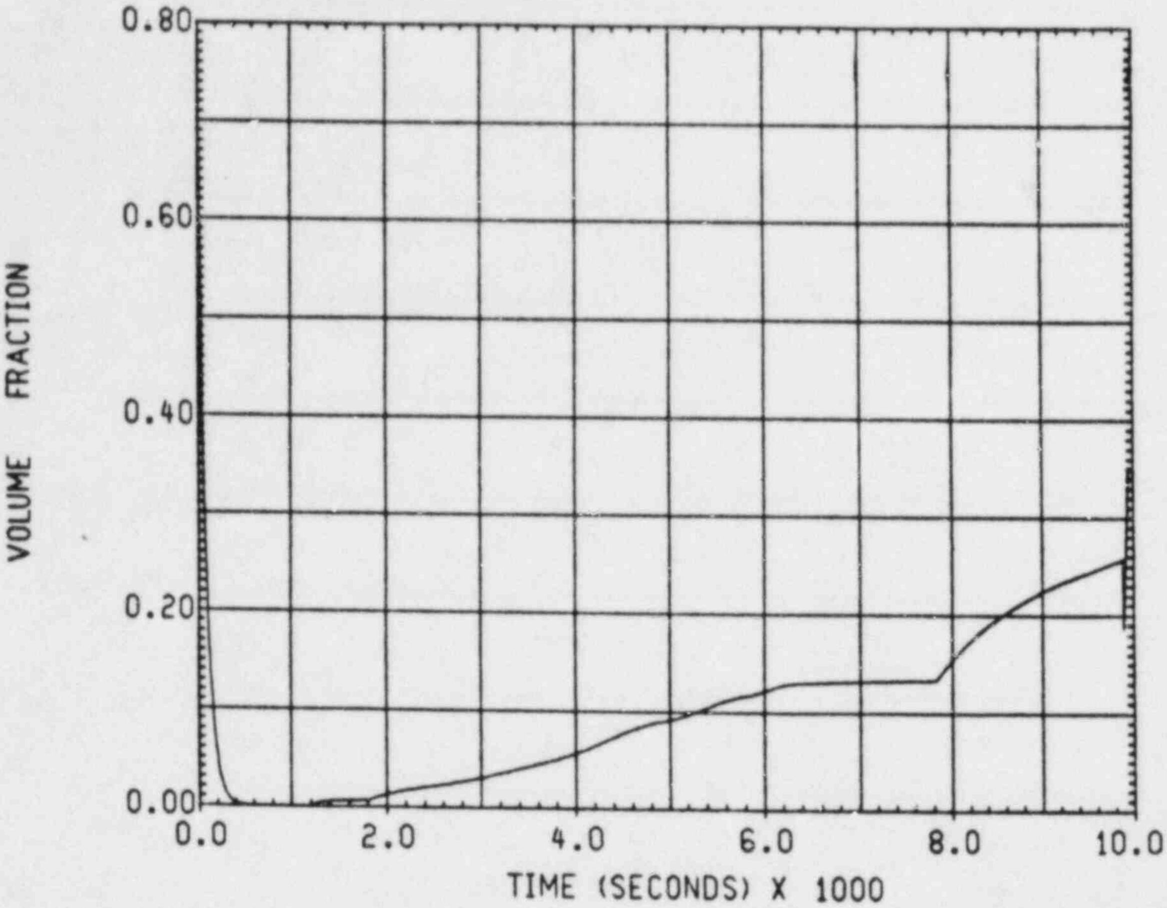

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
WETWELL O2 GAS CONCENTRATION

FIGURE 14



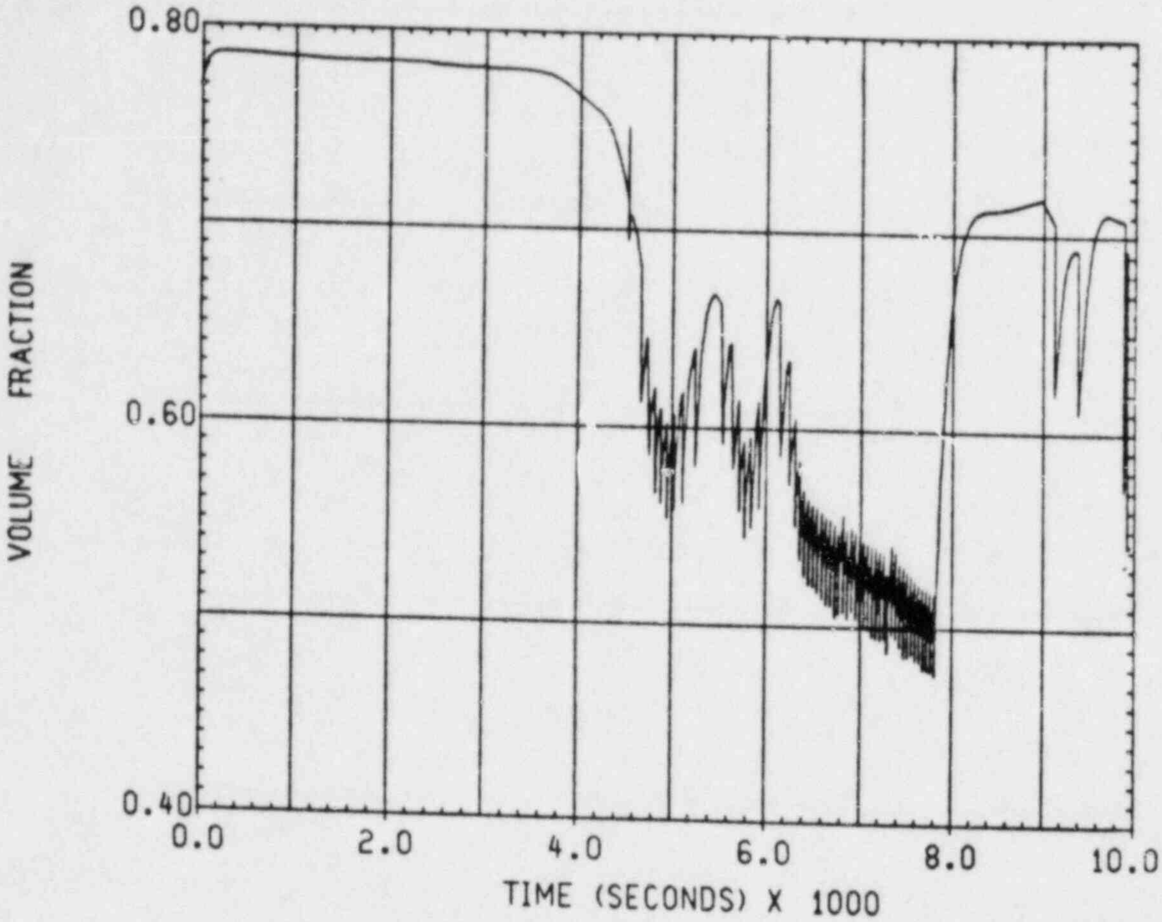

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
CONTAINMENT 02 GAS CONCENTRATION

FIGURE 15



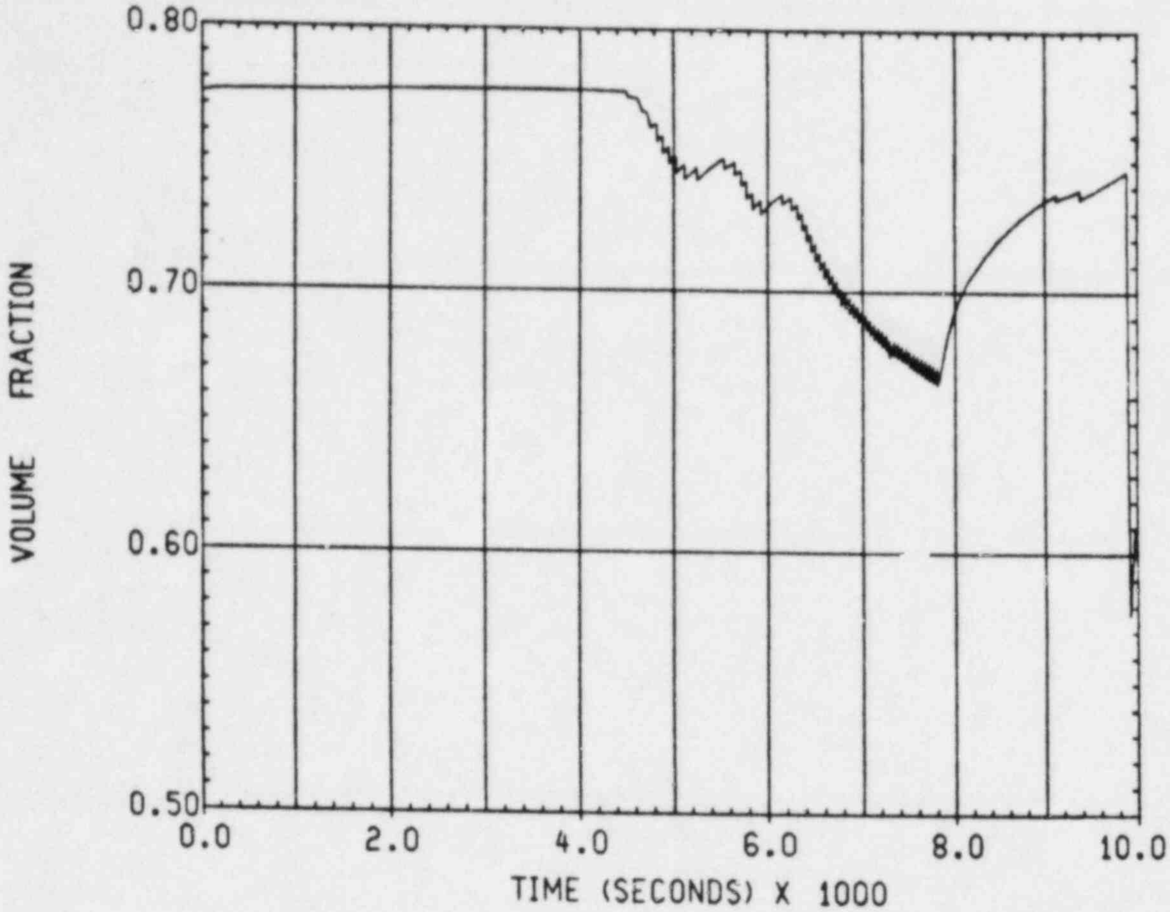

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
DRYWELL N2 GAS CONCENTRATION

FIGURE 16




70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
WETWELL N2 GAS CONCENTRATION

FIGURE 17




70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
CONTAINMENT N2 GAS CONCENTRATION

FIGURE 18

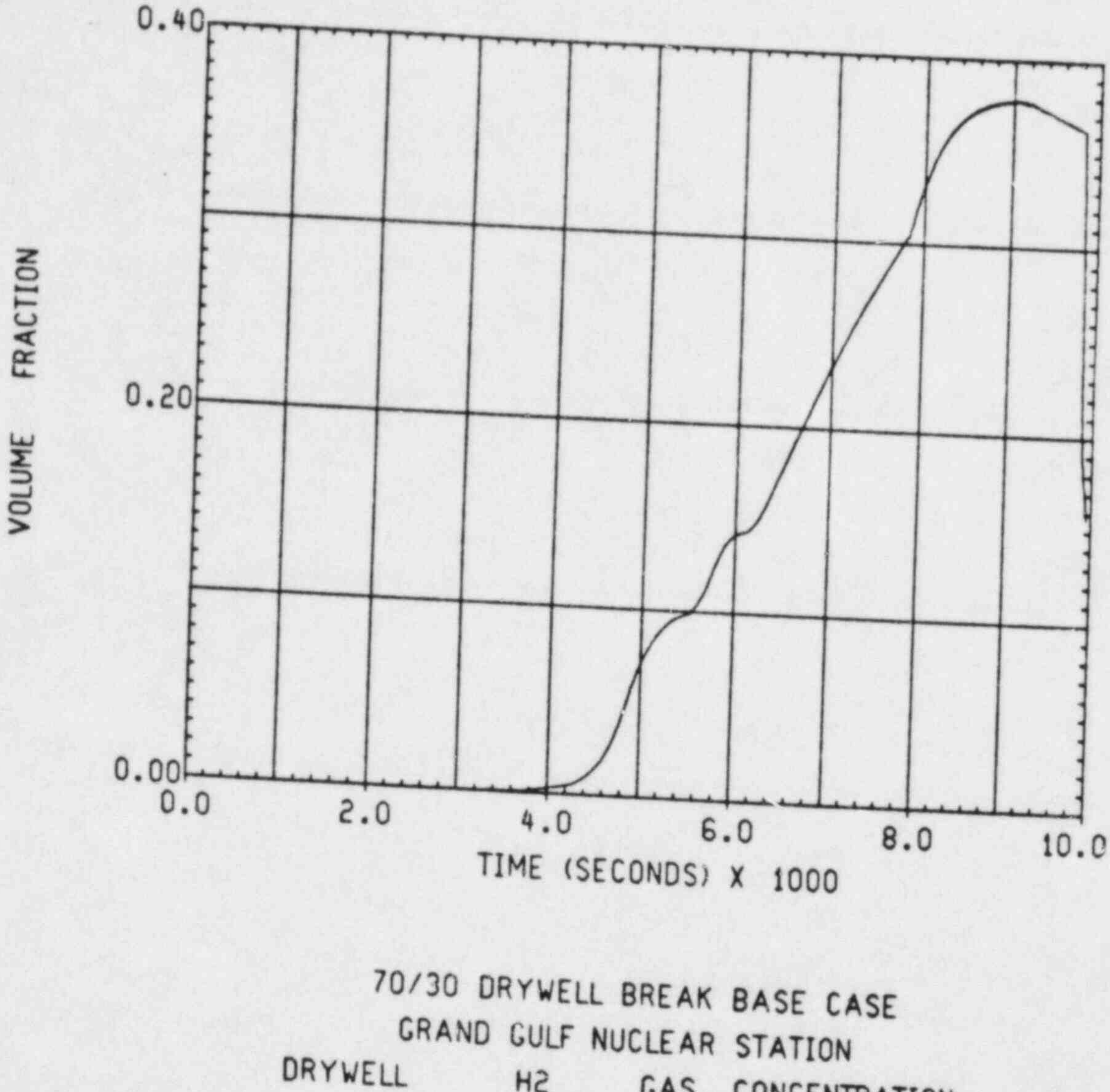
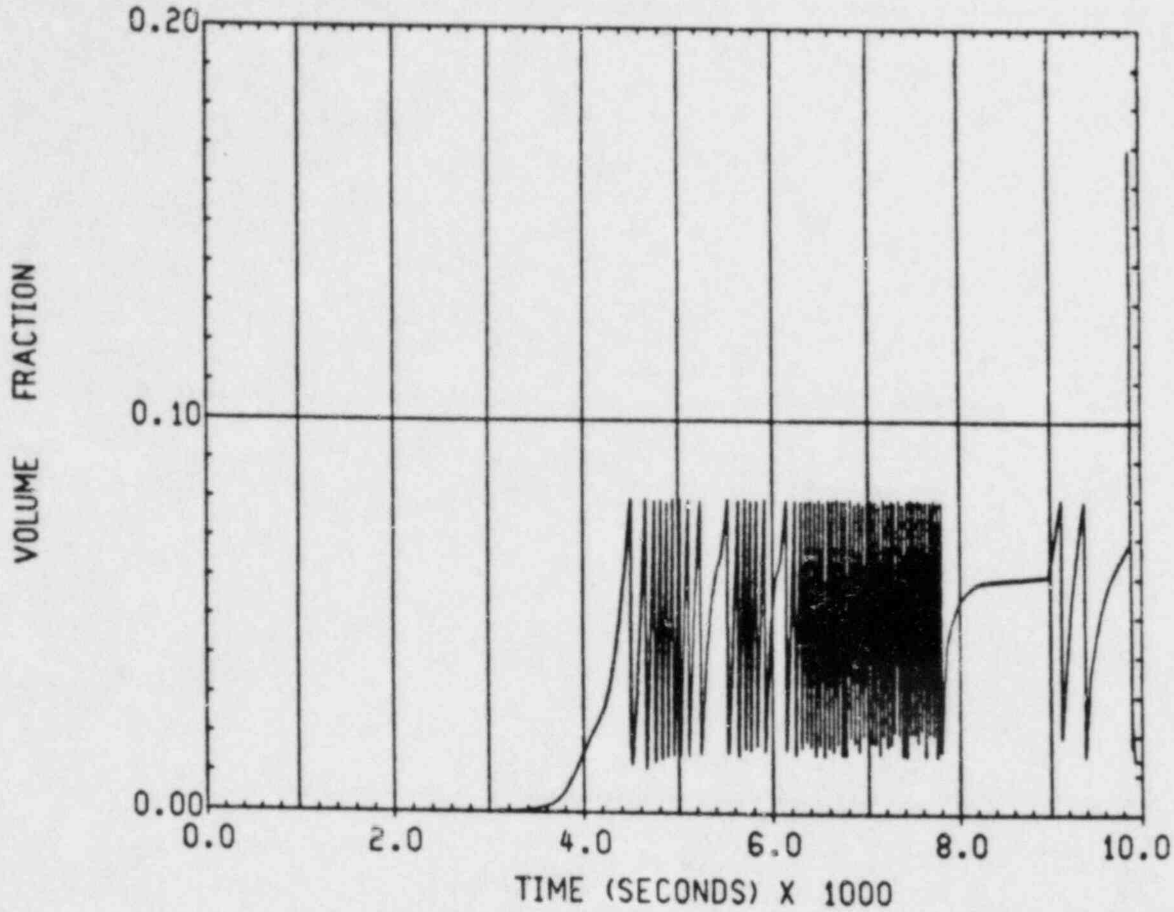
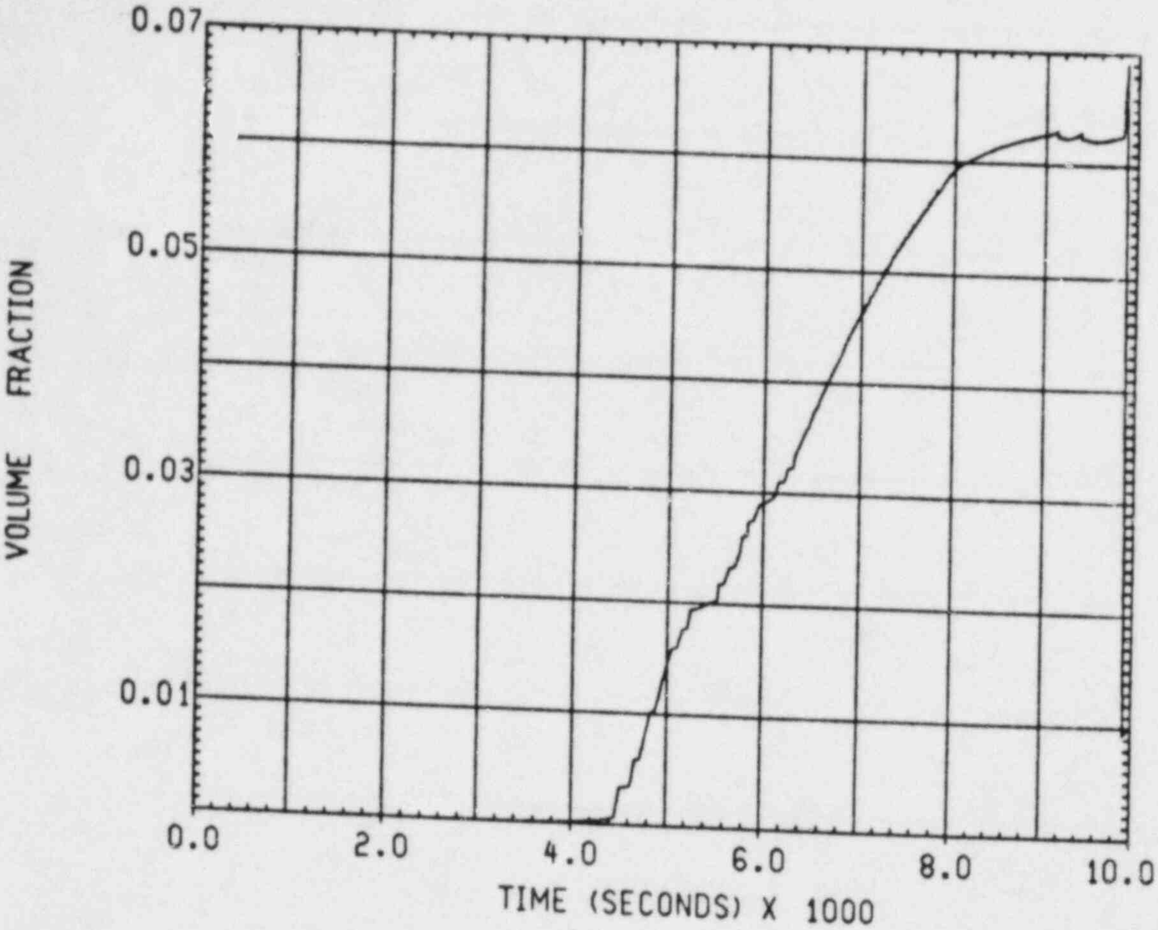

0.40
0.20
0.00
VOLUME FRACTION
0.0
2.0
4.0
6.0
8.0
10.0
TIME (SECONDS) X 1000
70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
DRYWELL H2 GAS CONCENTRATION

FIGURE 19



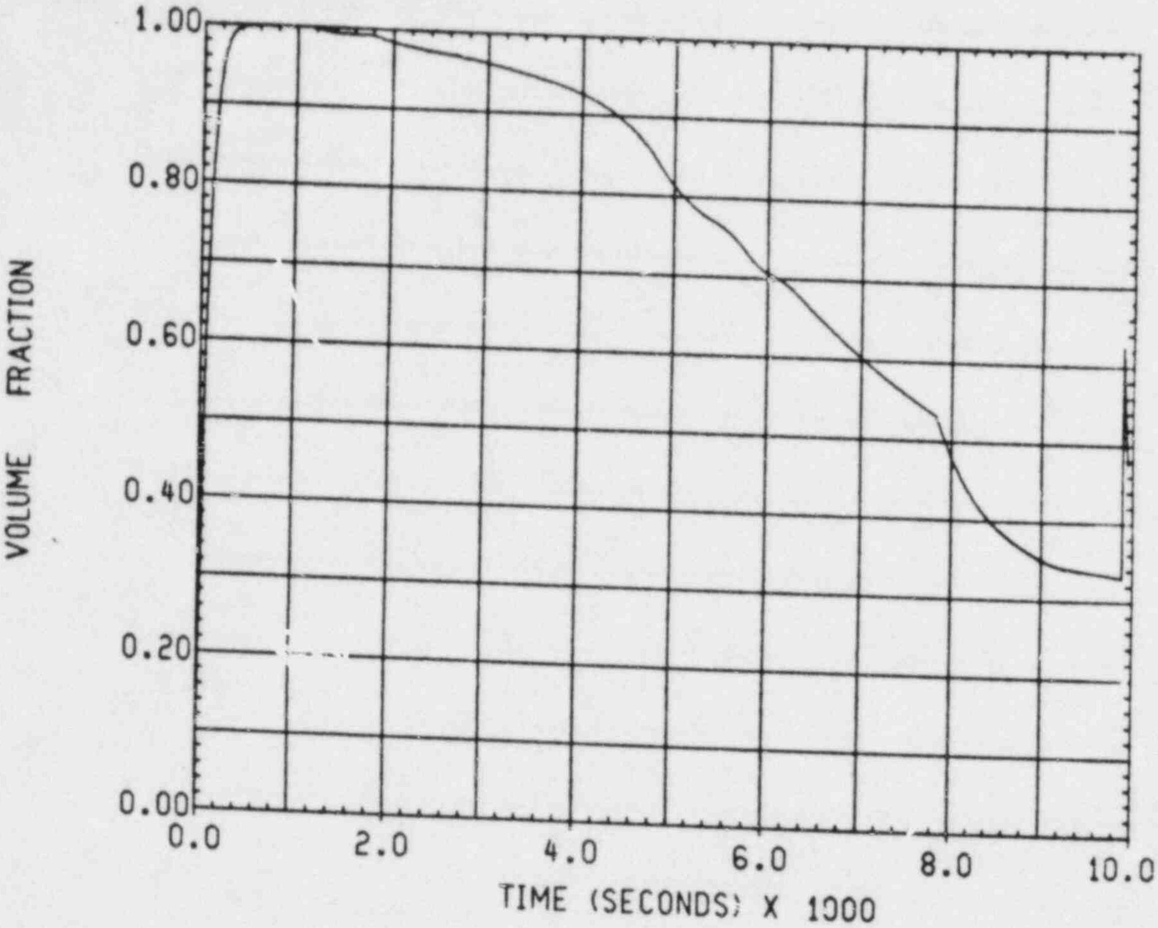

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
WETWELL H2 GAS CONCENTRATION

FIGURE 20



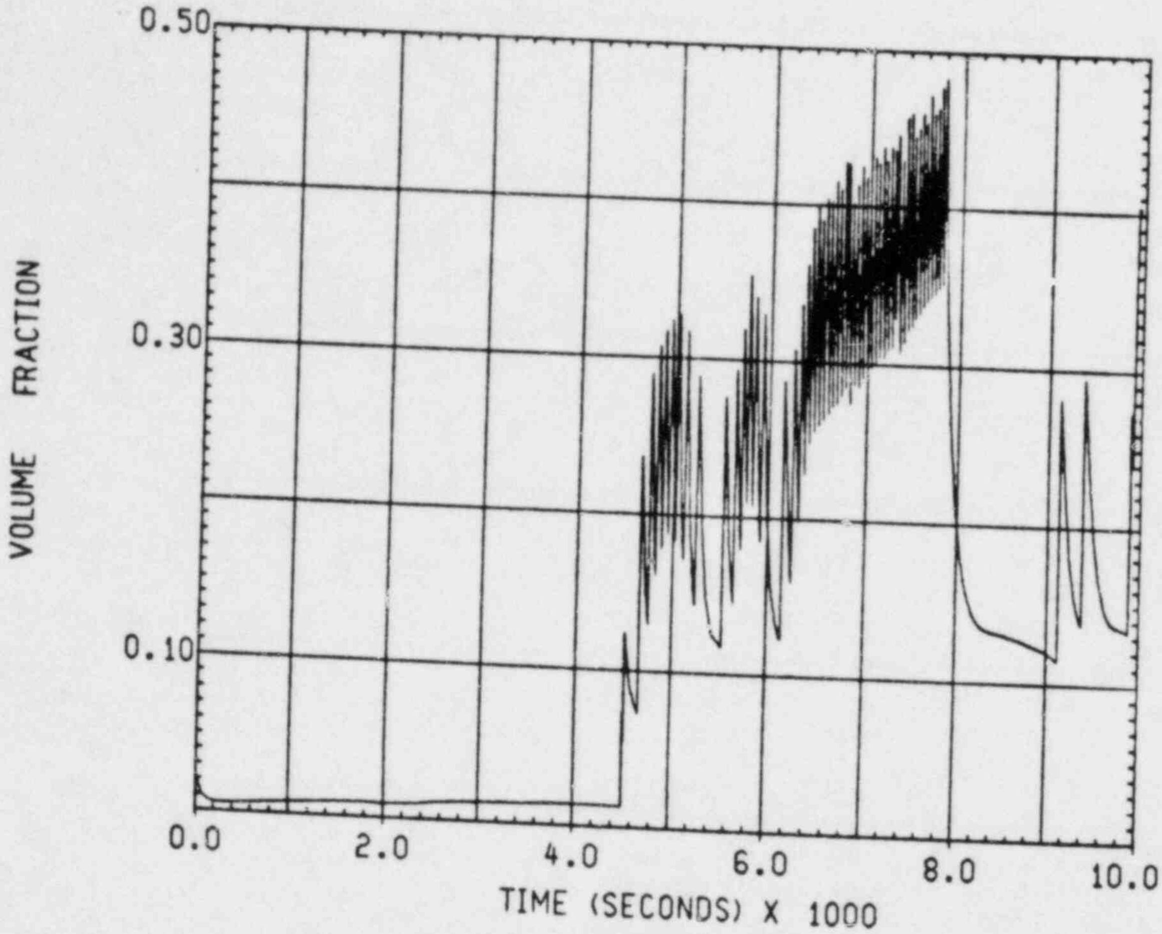

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
CONTAINMENT H2 GAS CONCENTRATION

FIGURE 21



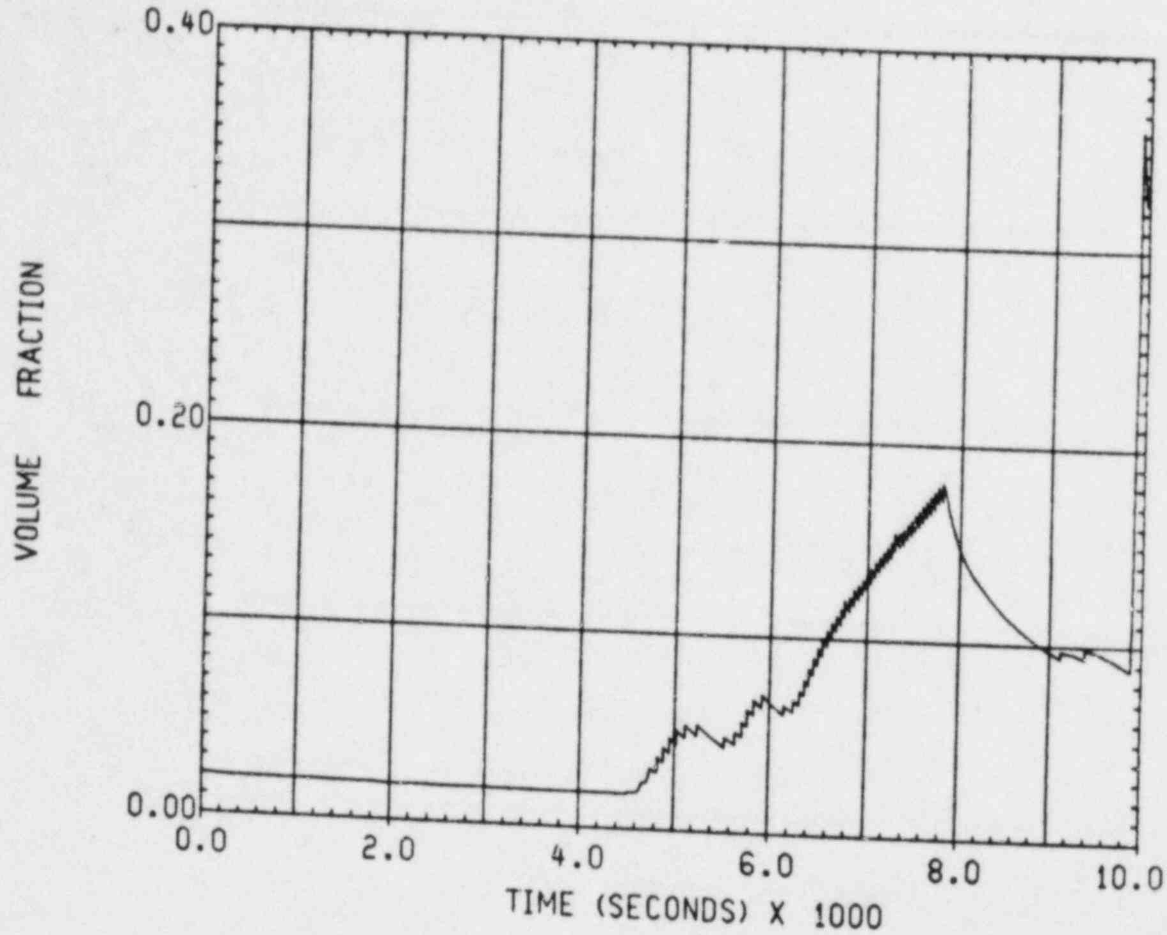

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
DRYWELL STEAM GAS CONCENTRATION

FIGURE 22



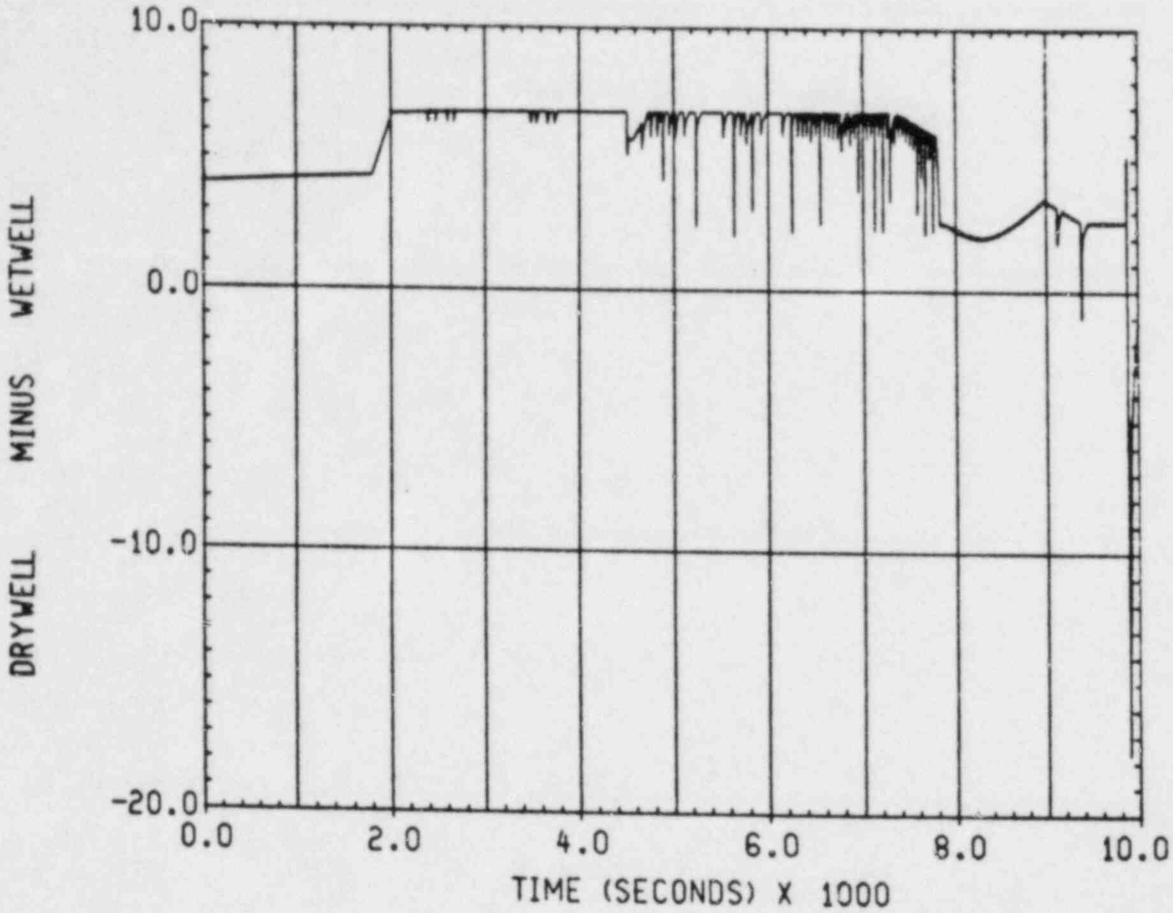

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
WETWELL STEAM GAS CONCENTRATION

FIGURE 23



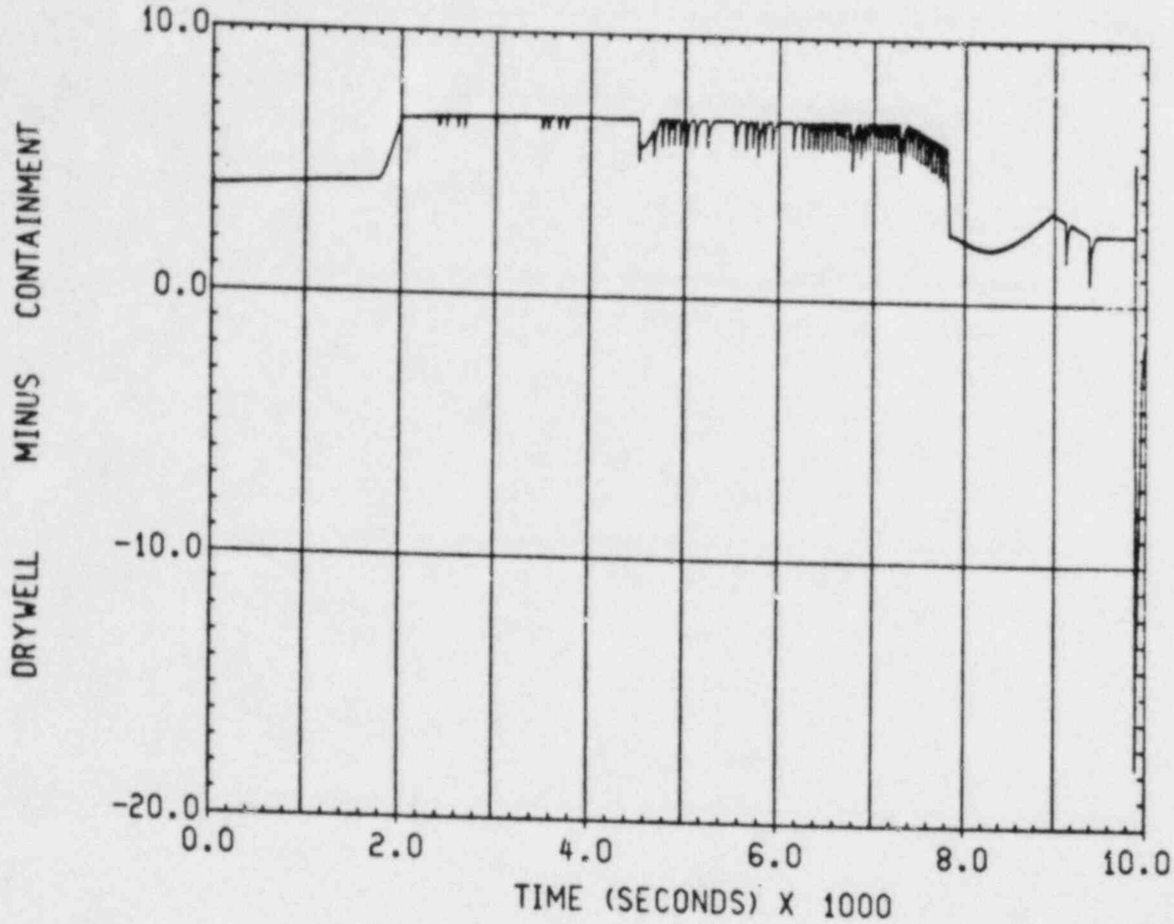

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
CONTAINMENT STEAM GAS CONCENTRATION

FIGURE 24



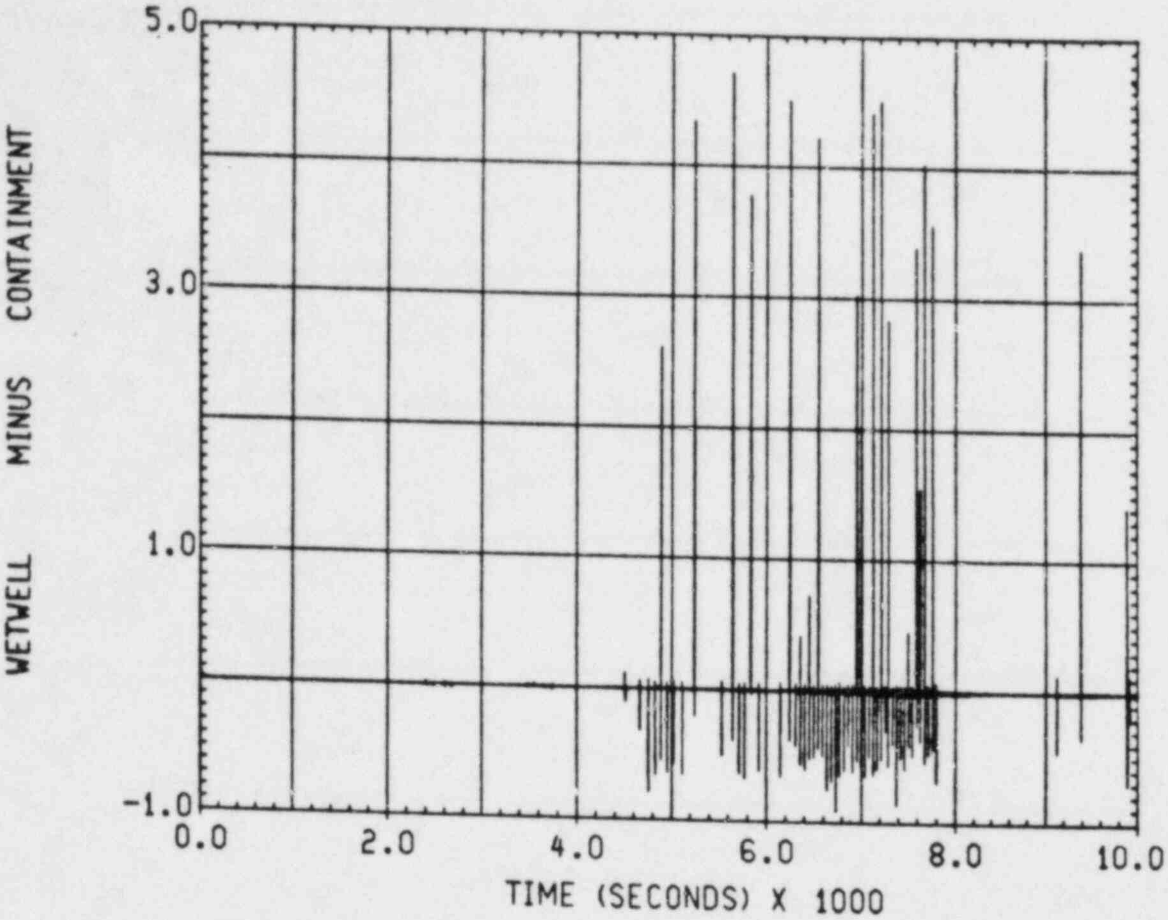

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
DIFFERENTIAL PRESSURE

FIGURE 25



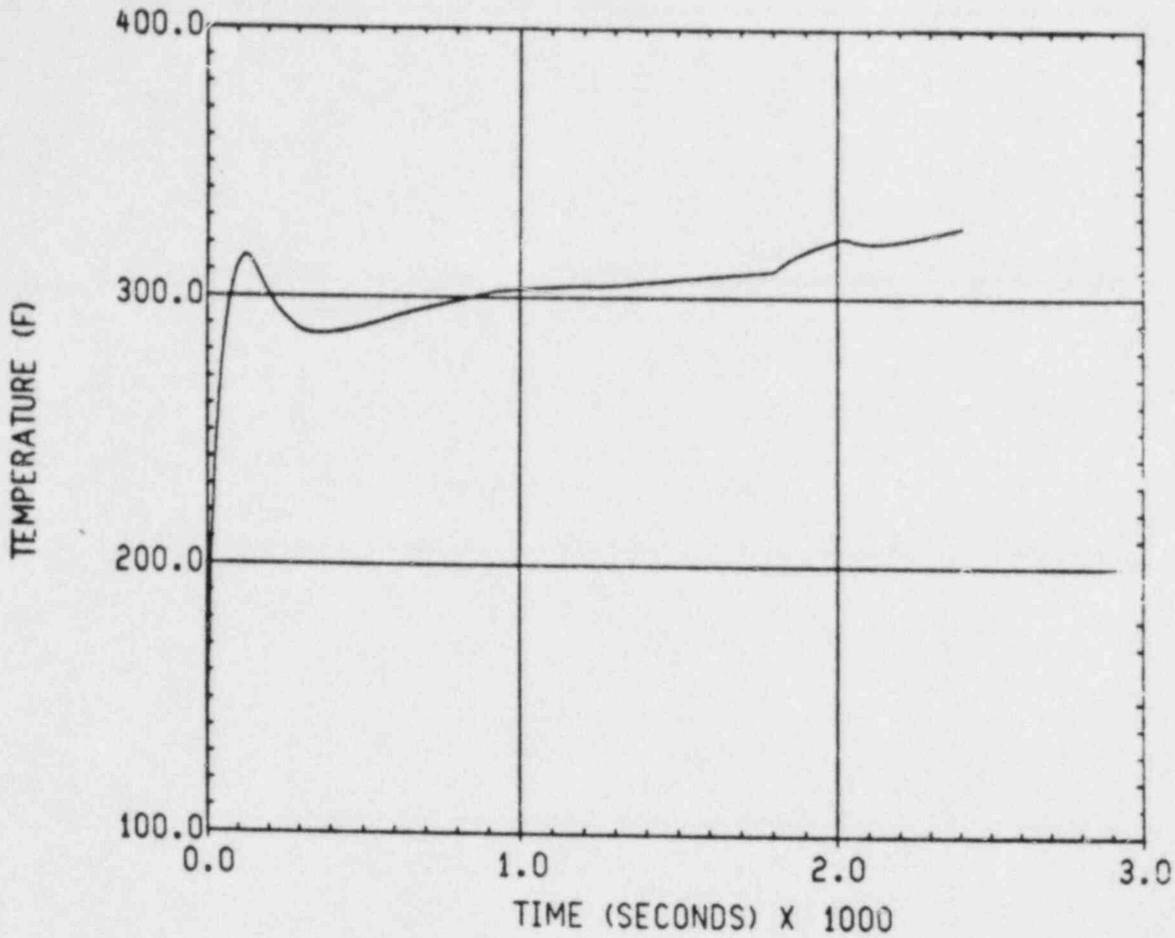

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
DIFFERENTIAL PRESSURE

FIGURE 26



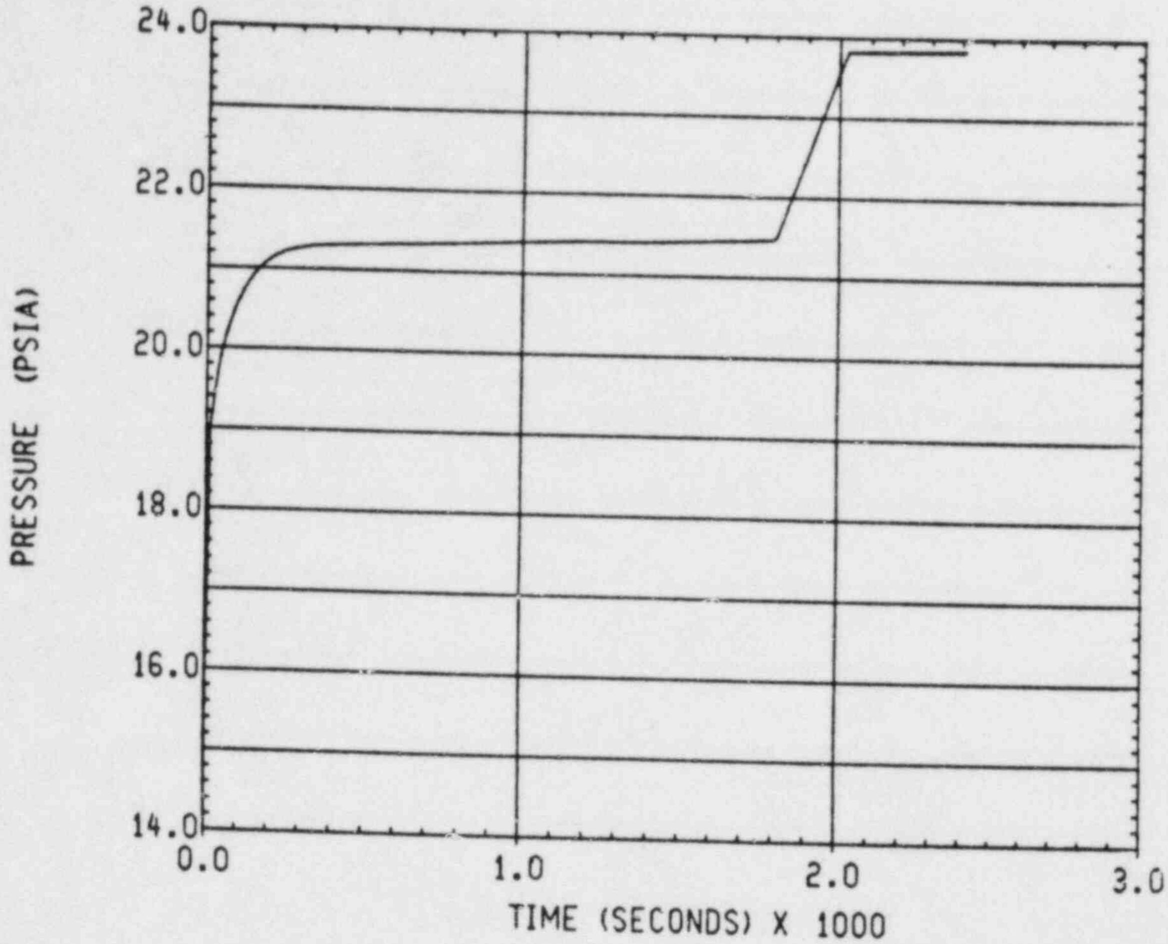

70/30 DRYWELL BREAK BASE CASE
GRAND GULF NUCLEAR STATION
DIFFERENTIAL PRESSURE

FIGURE 27



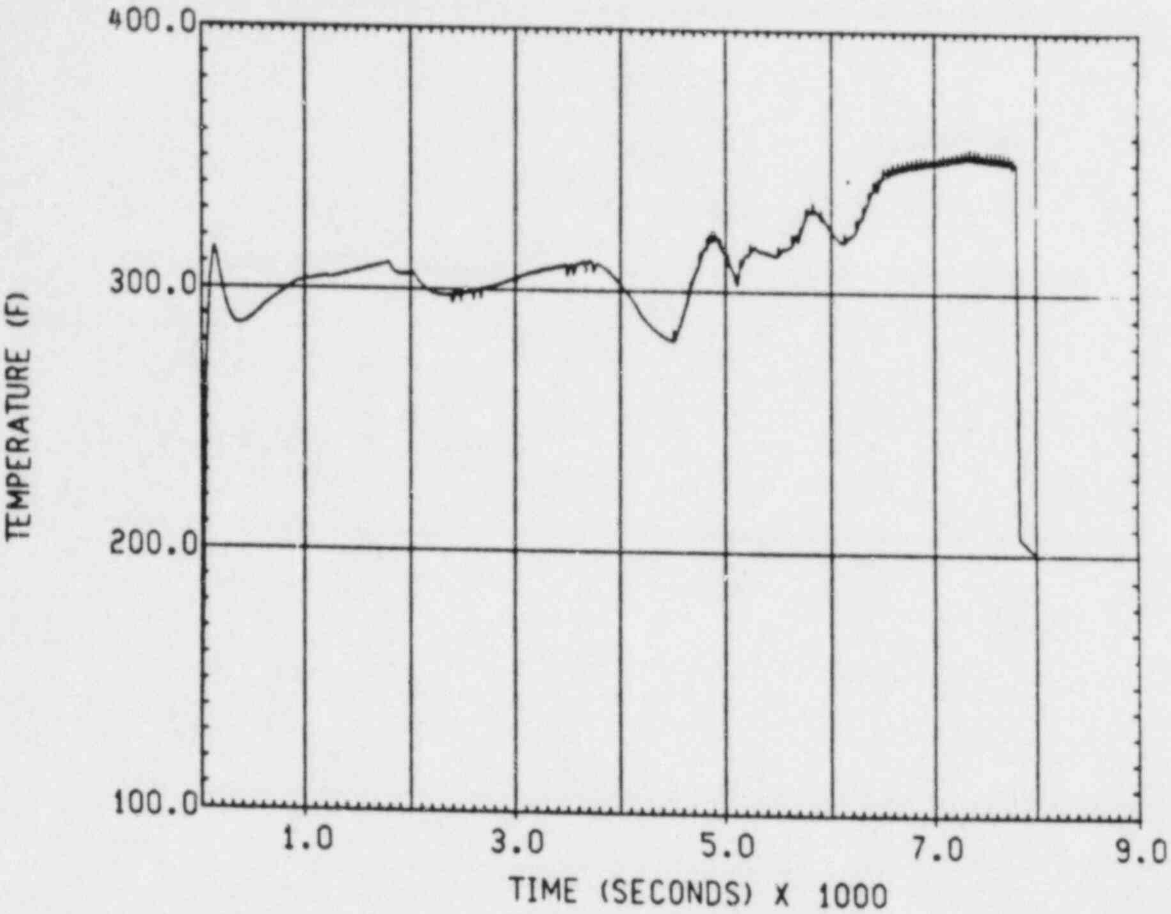

100/0 DRYWELL BREAK
GRAND GULF NUCLEAR STATION
DRYWELL TEMPERATURE

FIGURE 28



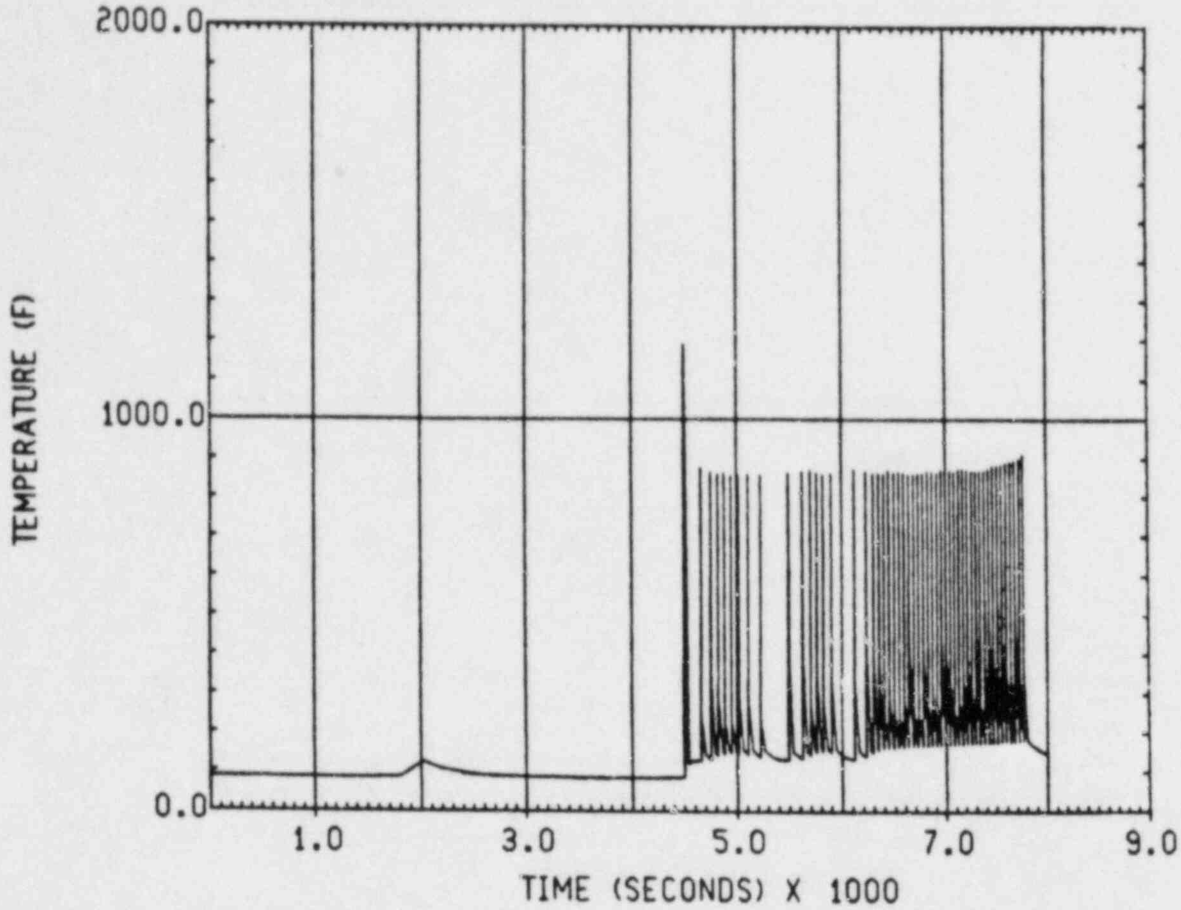

100/0 DRYWELL BREAK
GRAND GULF NUCLEAR STATION
DRYWELL PRESSURE

FIGURE 29



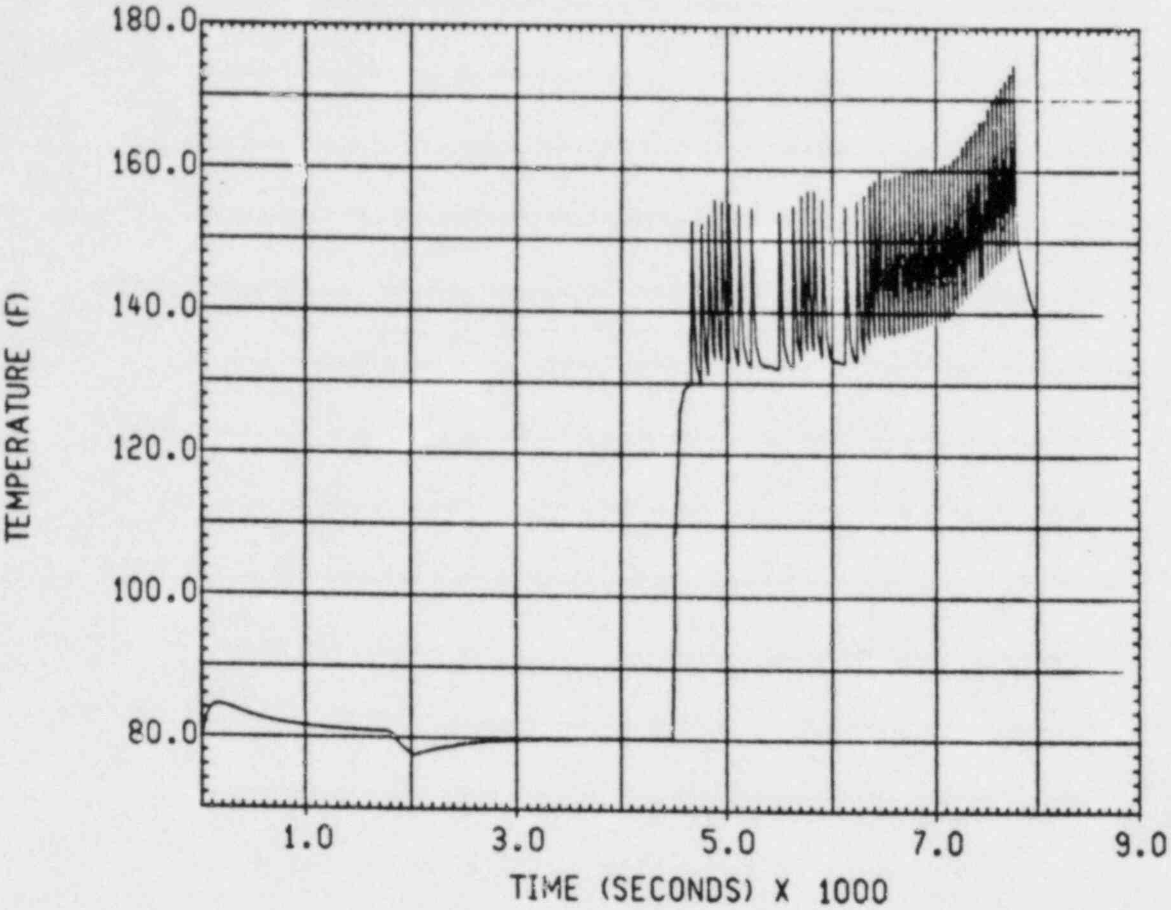

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
DRYWELL TEMPERATURE

FIGURE 30



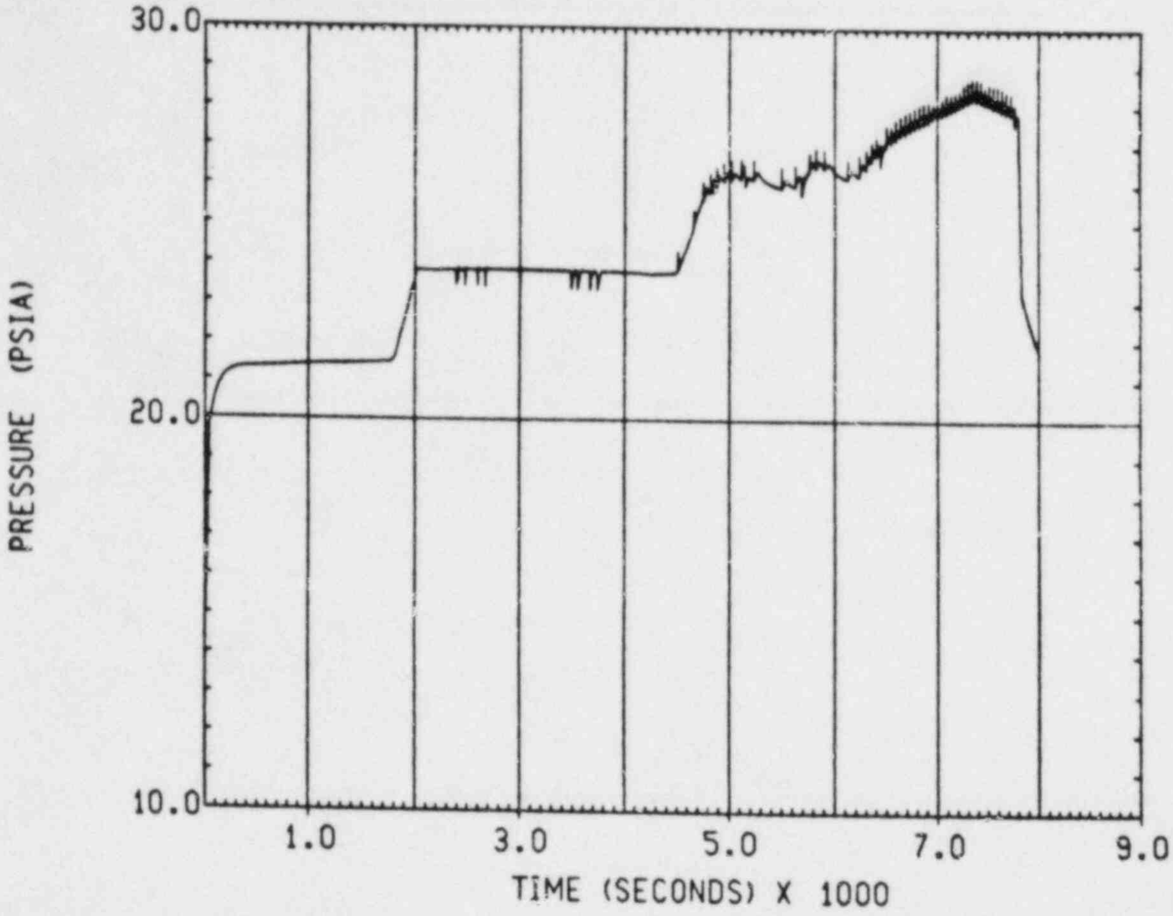

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
WETWELL TEMPERATURE

FIGURE 31



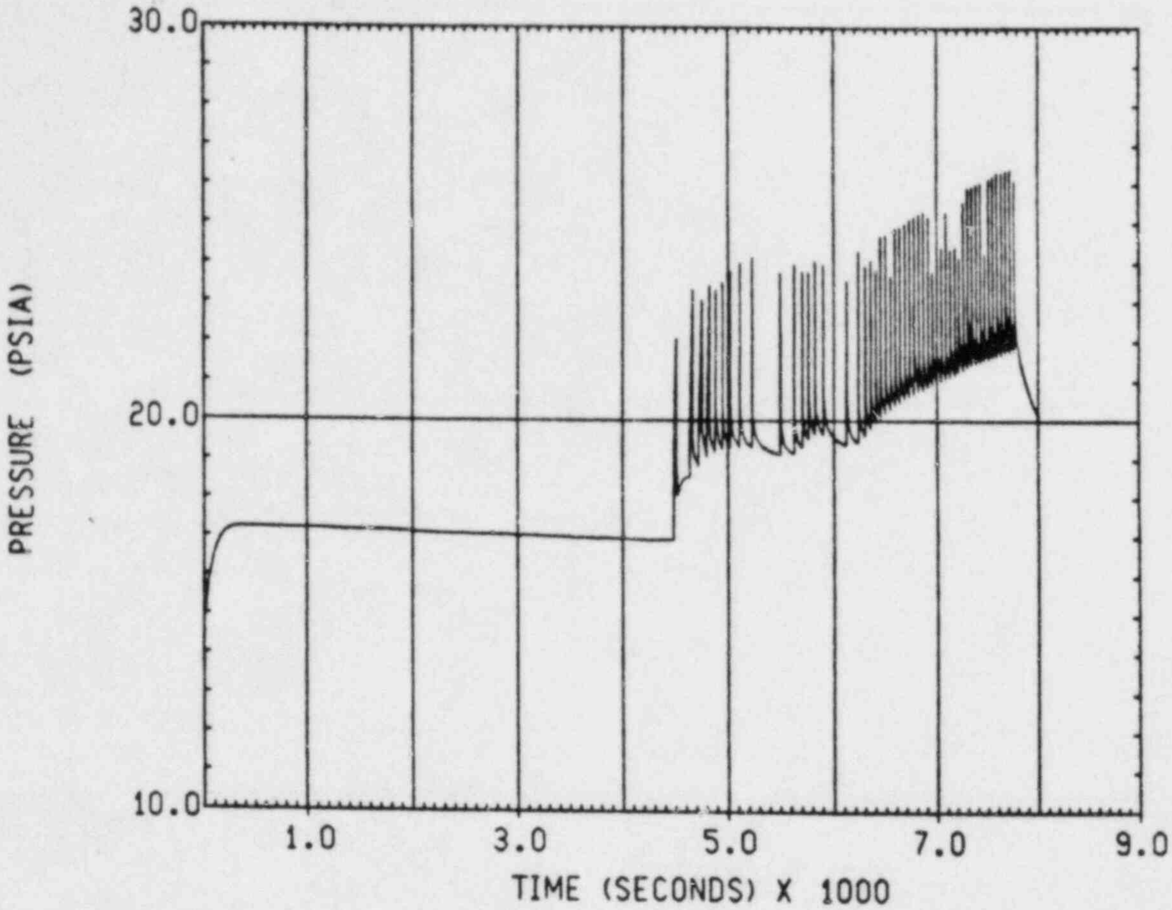

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
CONTAINMENT TEMPERATURE

FIGURE 32



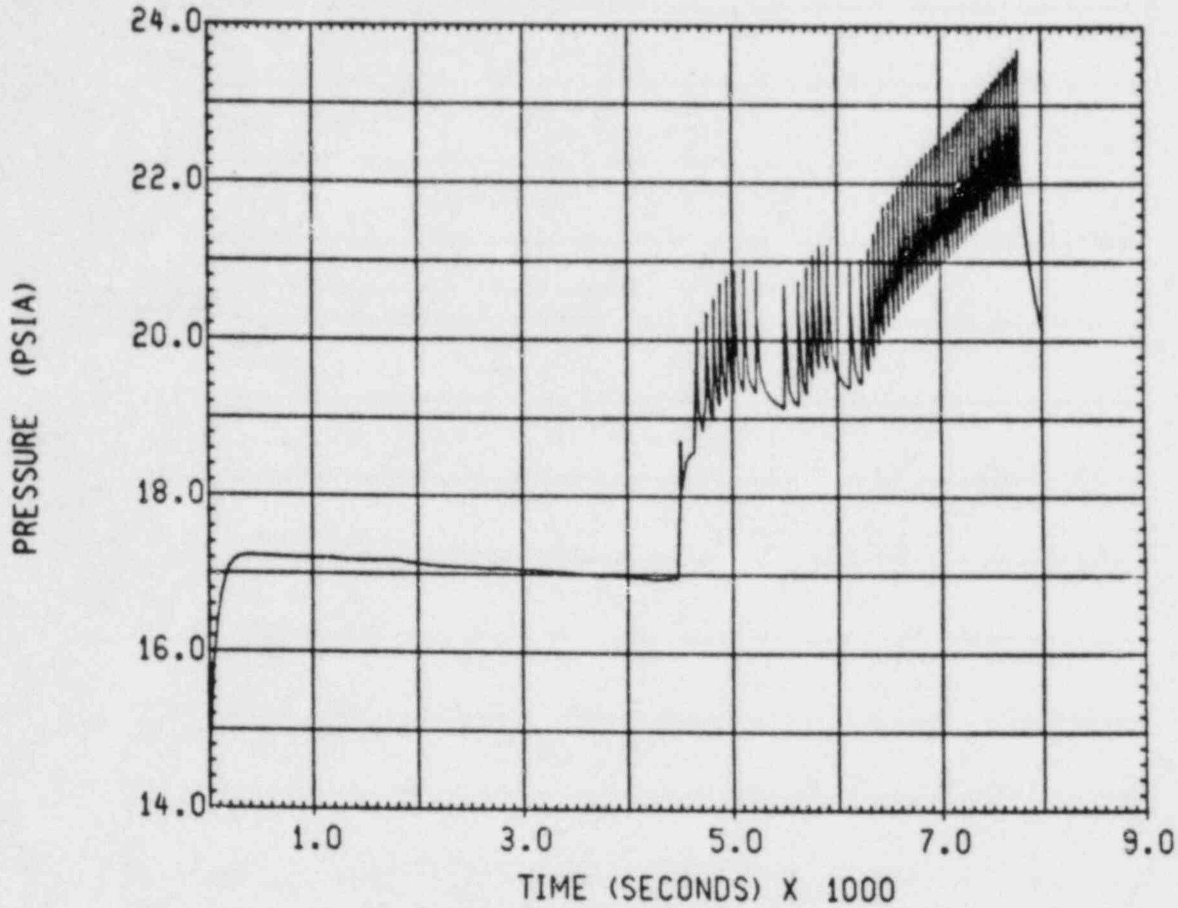

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
DRYWELL PRESSURE

FIGURE 33



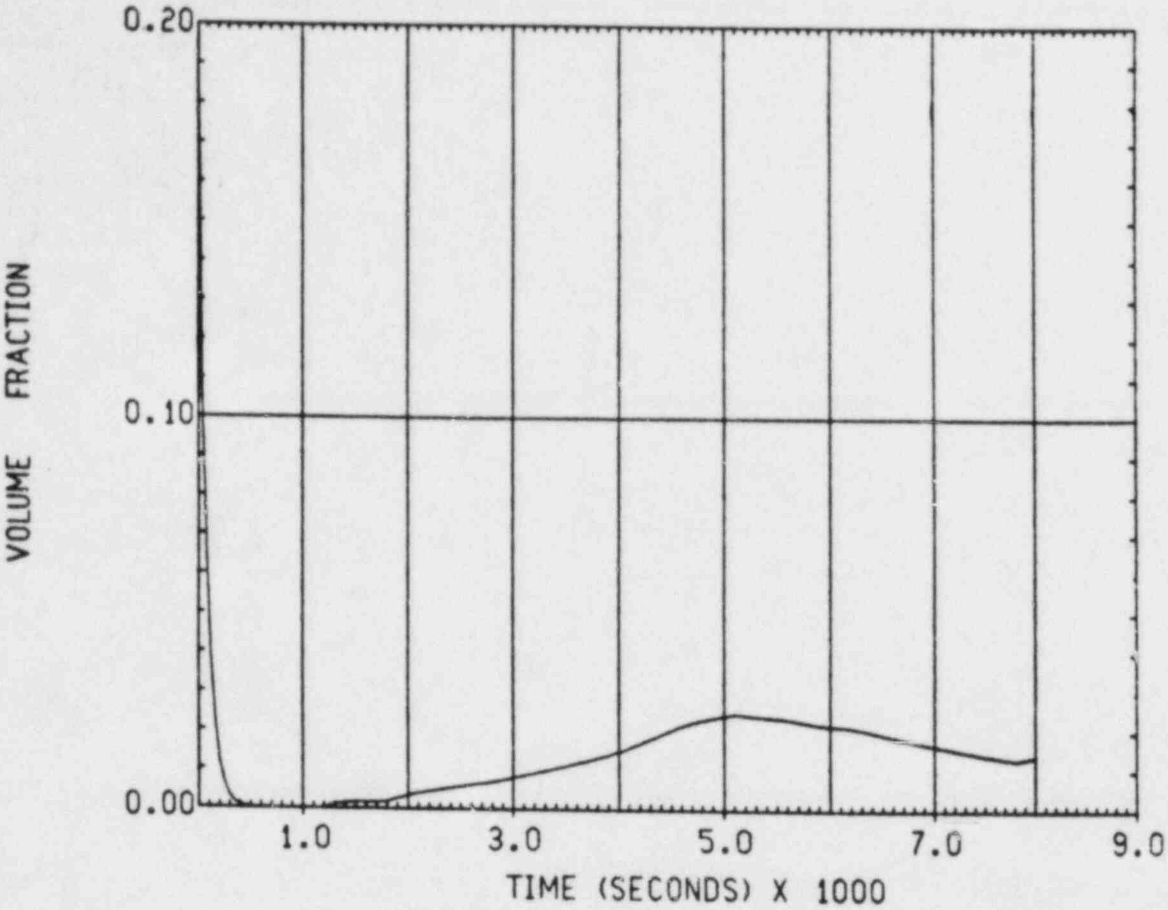

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
WETWELL PRESSURE

FIGURE 34



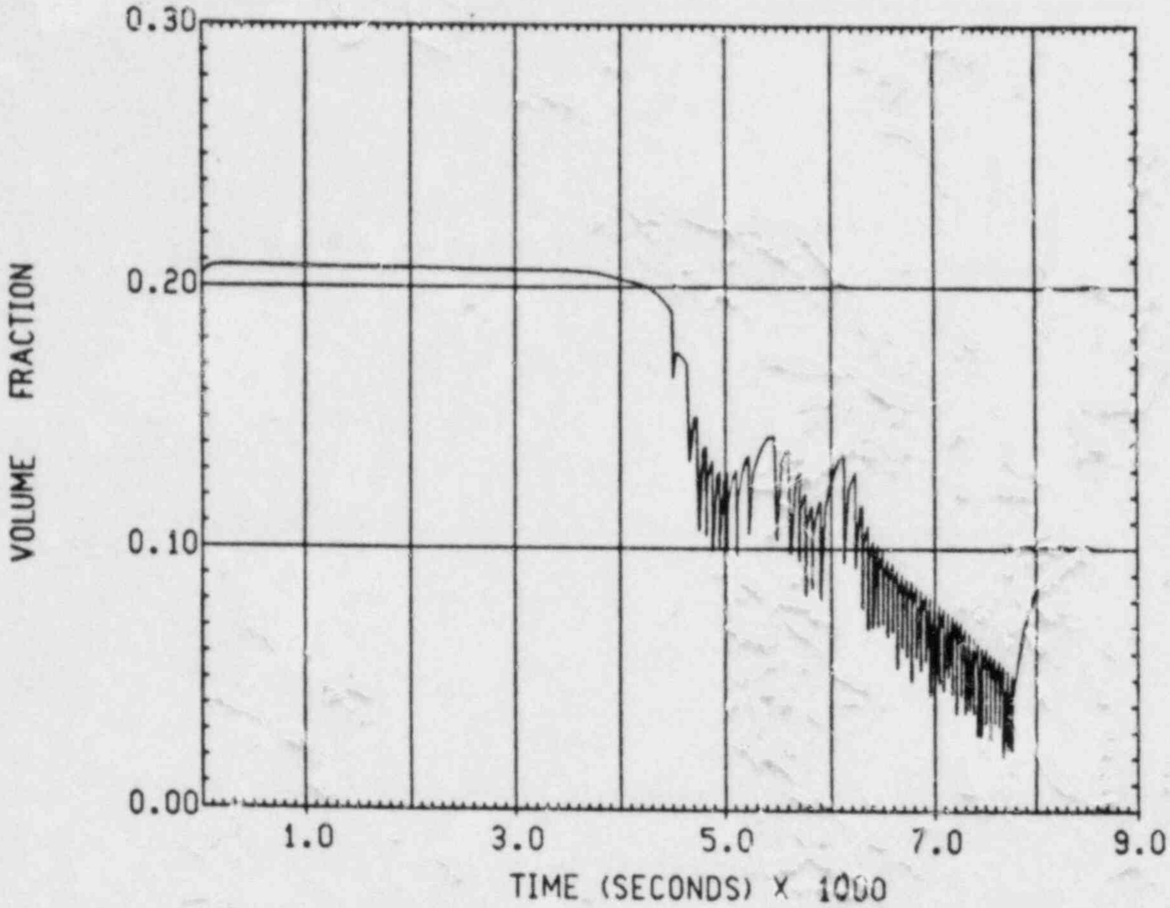

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
CONTAINMENT PRESSURE

FIGURE 35



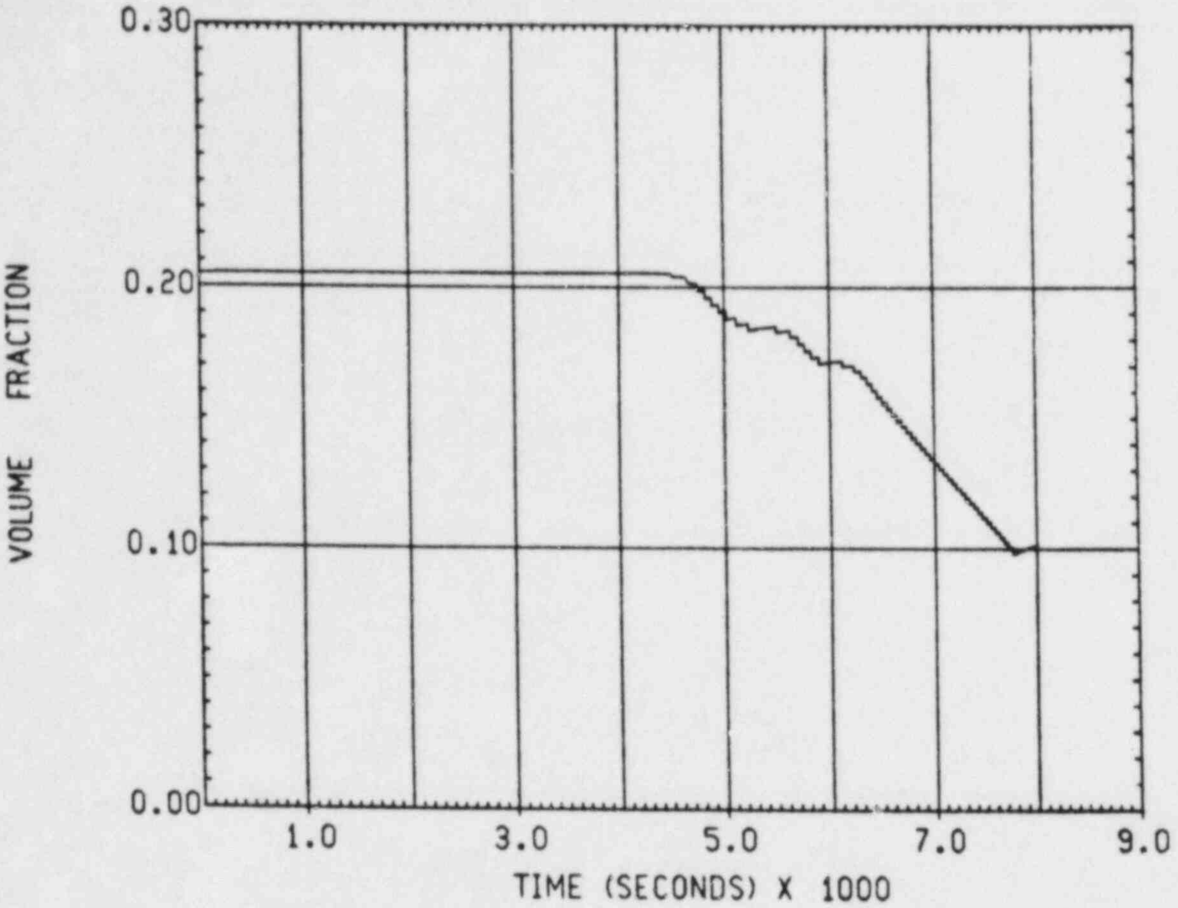

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
DRYWELL O2 GAS CONCENTRATION

FIGURE 36



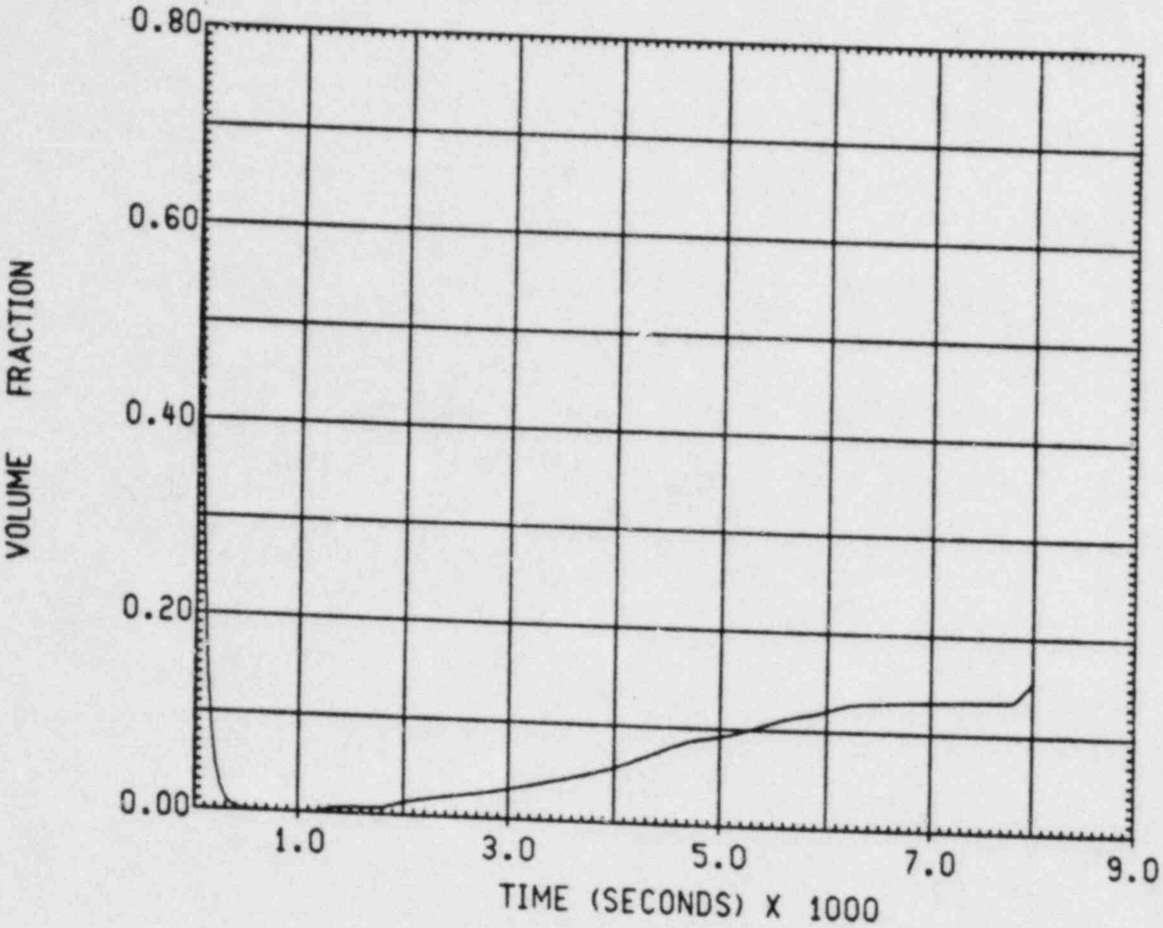

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
WETWELL O2 GAS CONCENTRATION

FIGURE 37



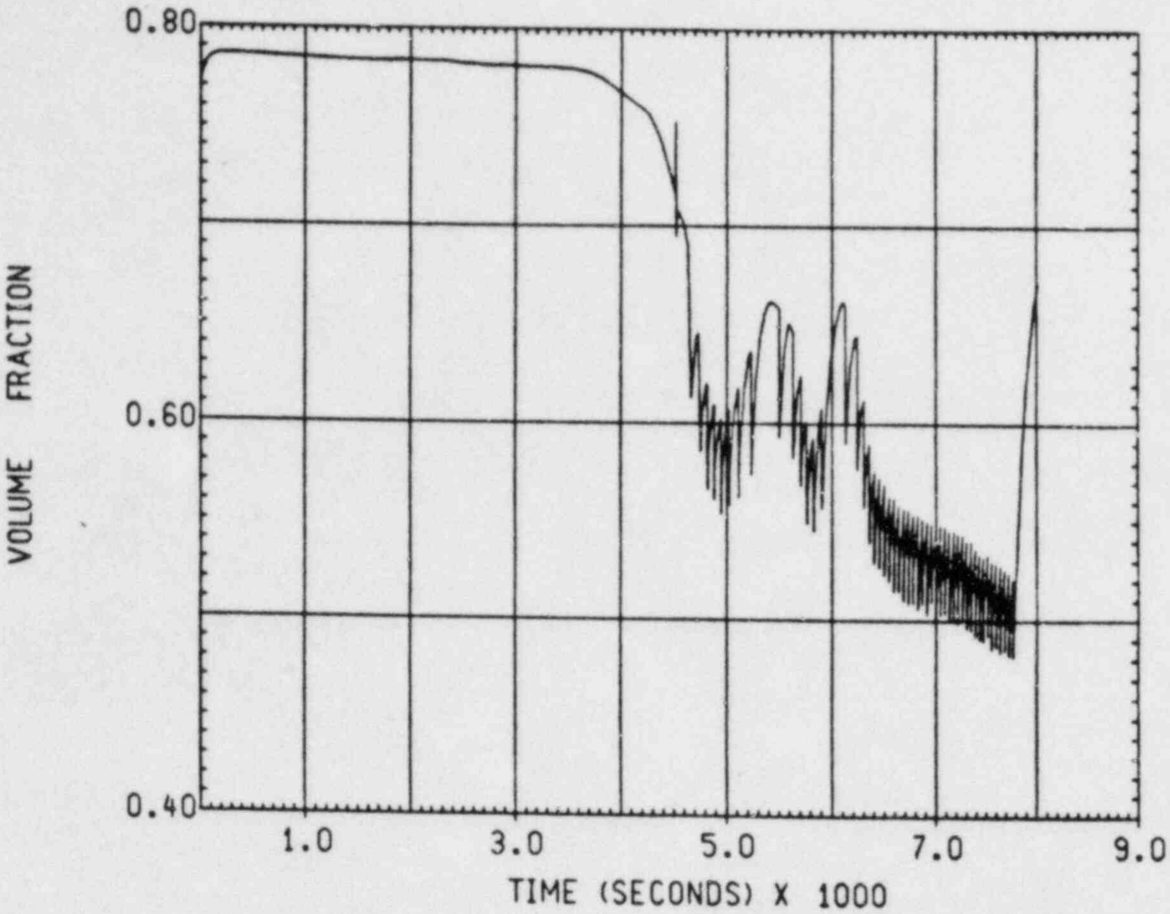

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
CONTAINMENT O2 GAS CONCENTRATION

FIGURE 38



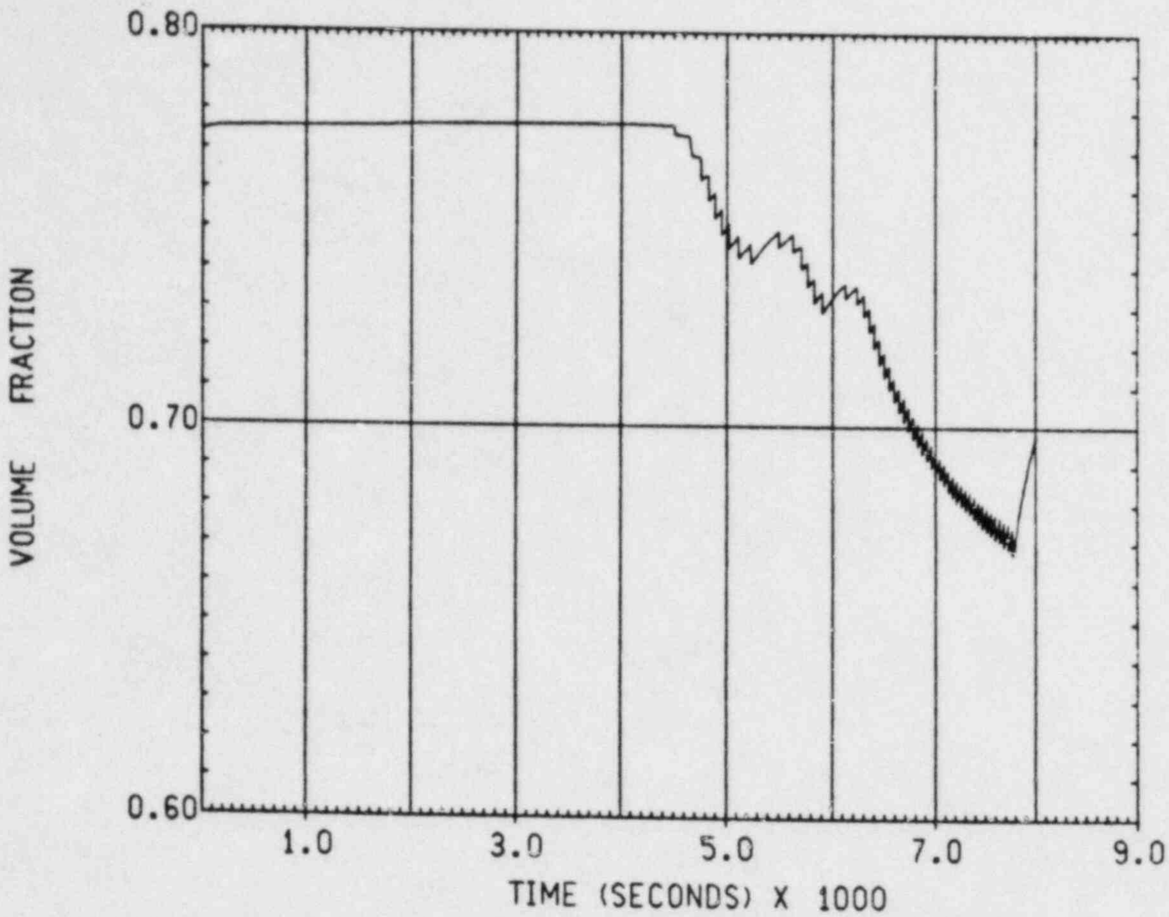

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
DRYWELL N2 GAS CONCENTRATION

FIGURE 39



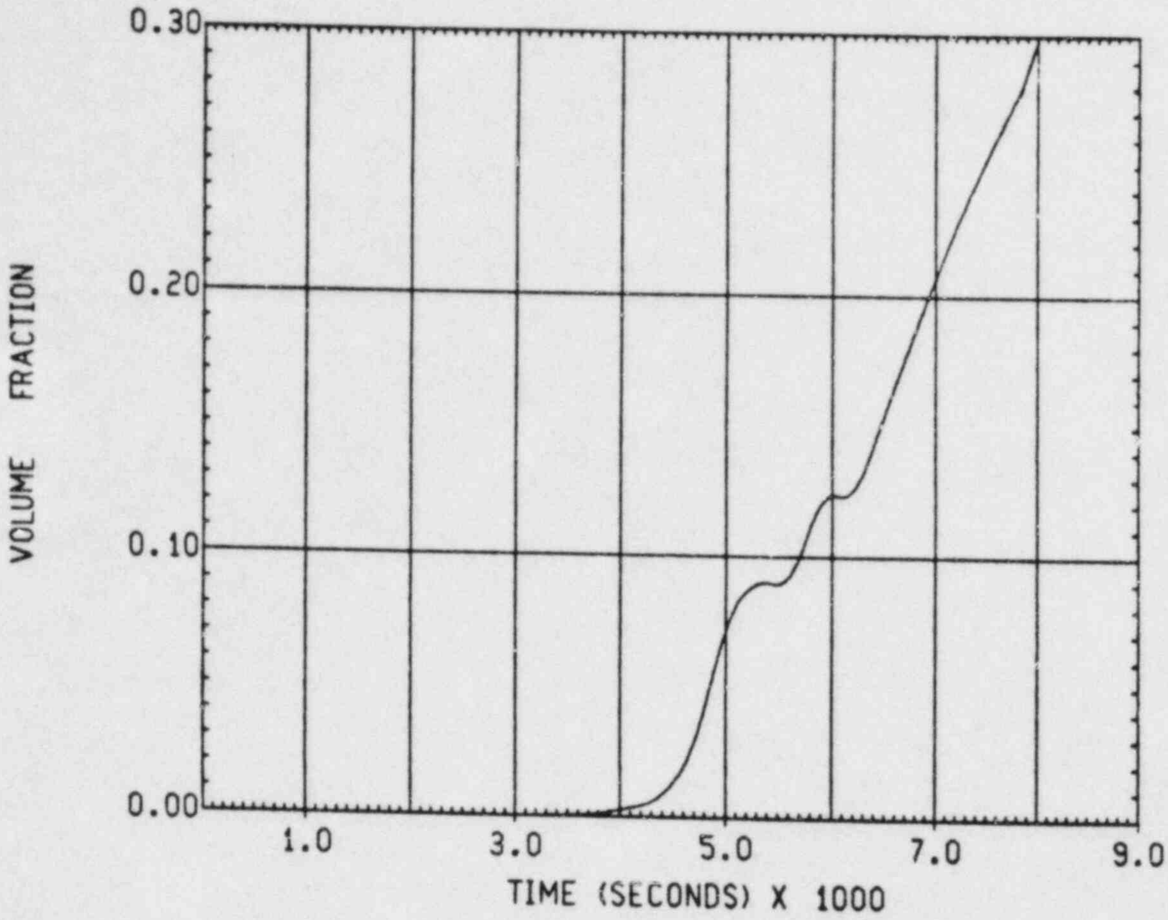

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
WETWELL N2 GAS CONCENTRATION

FIGURE 40



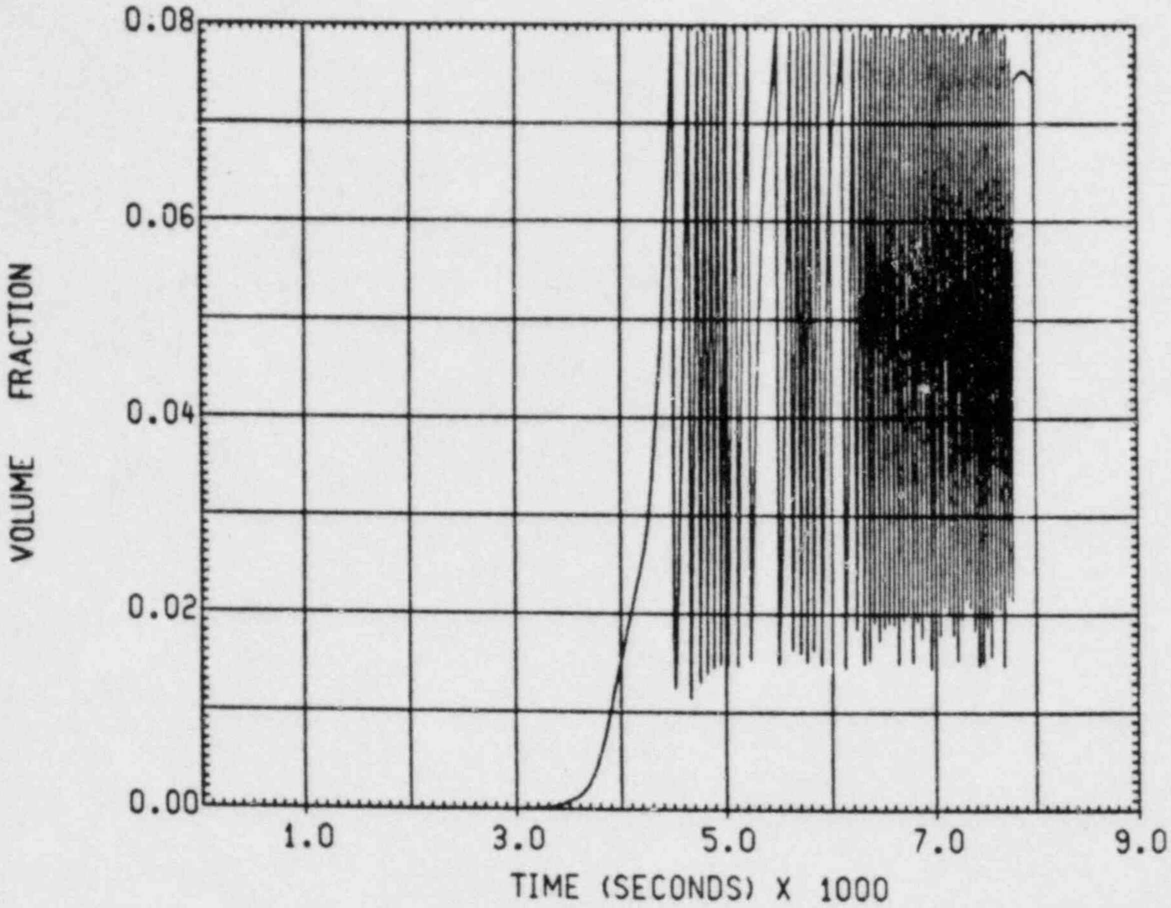

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
CONTAINMENT N2 GAS CONCENTRATION

FIGURE 41



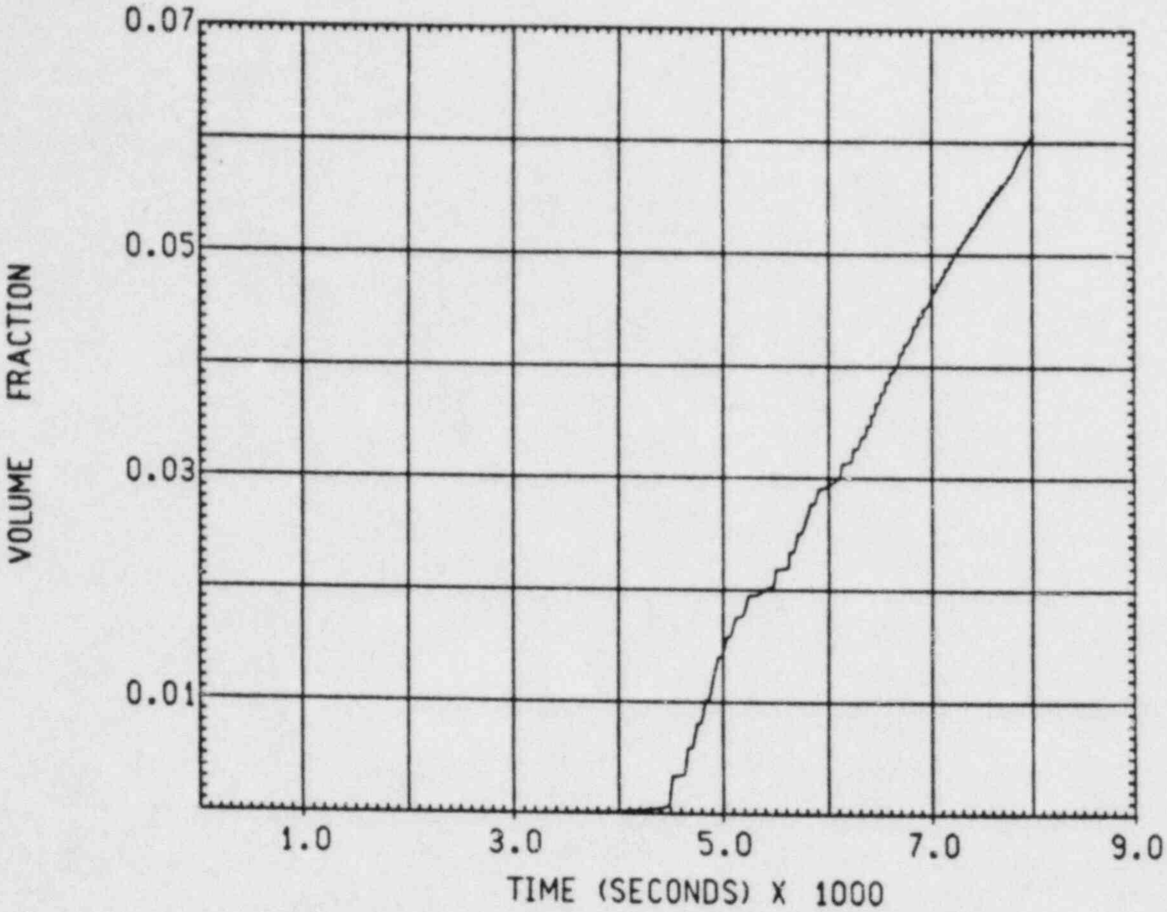

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
DRYWELL H2 GAS CONCENTRATION

FIGURE 42




70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
WETWELL H2 GAS CONCENTRATION

FIGURE 43

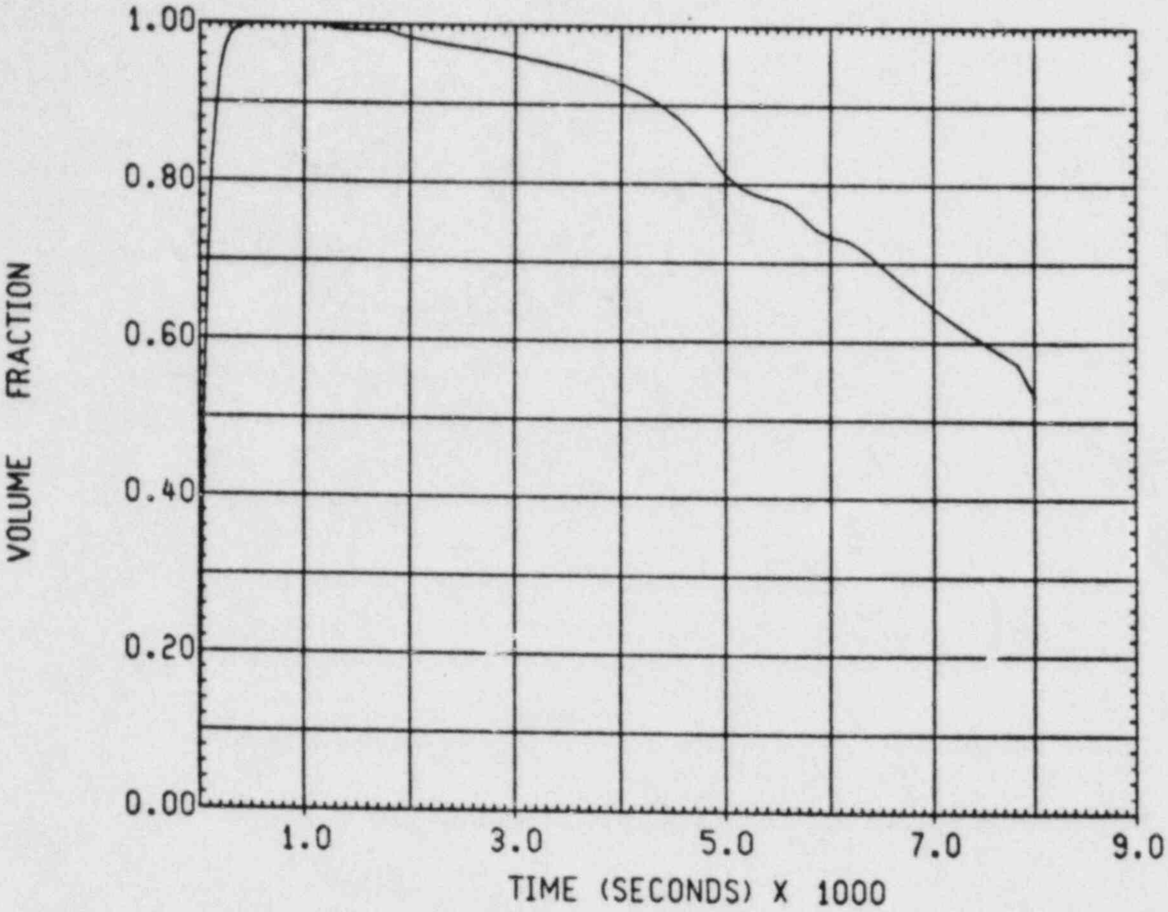



70/30 DRYWELL BREAK CONTINUOUS BURN

GRAND GULF NUCLEAR STATION

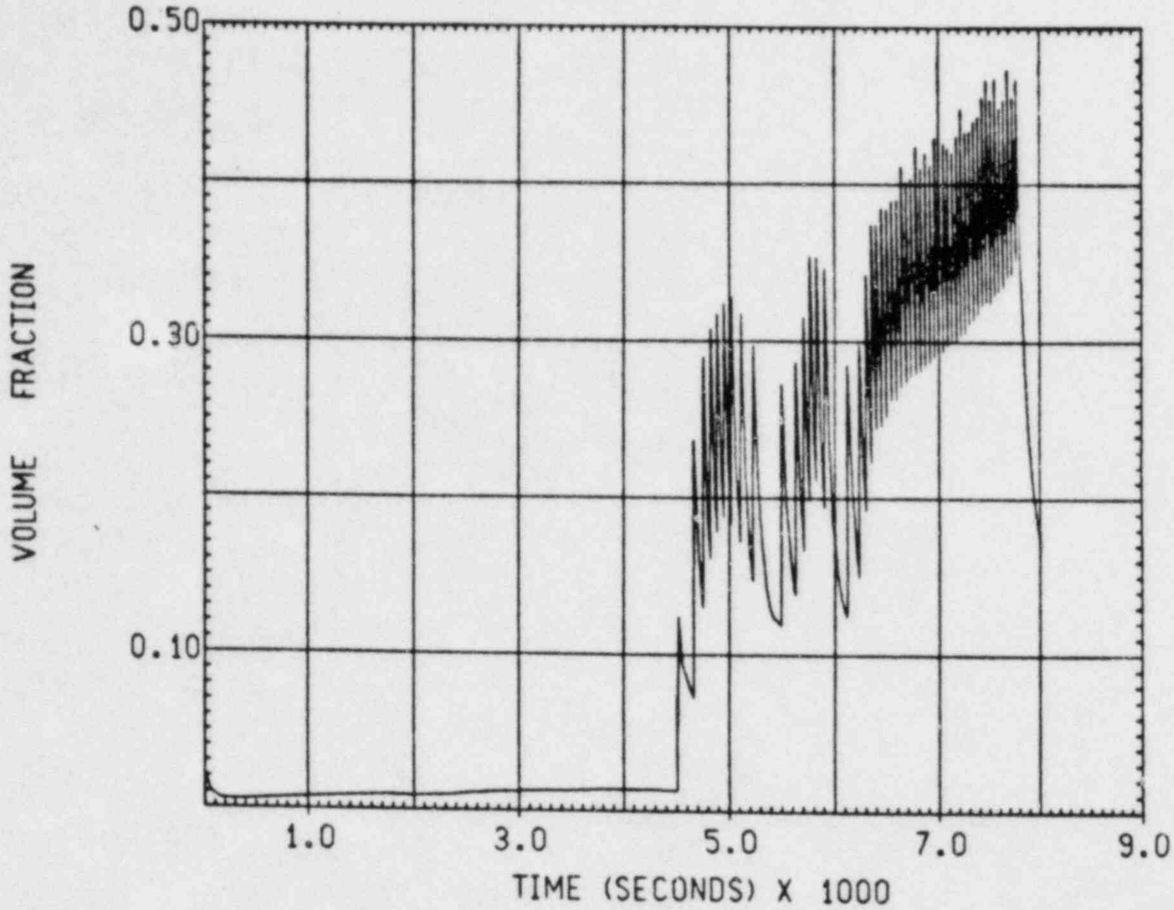
CONTAINMENT H2 GAS CONCENTRATION

FIGURE 44



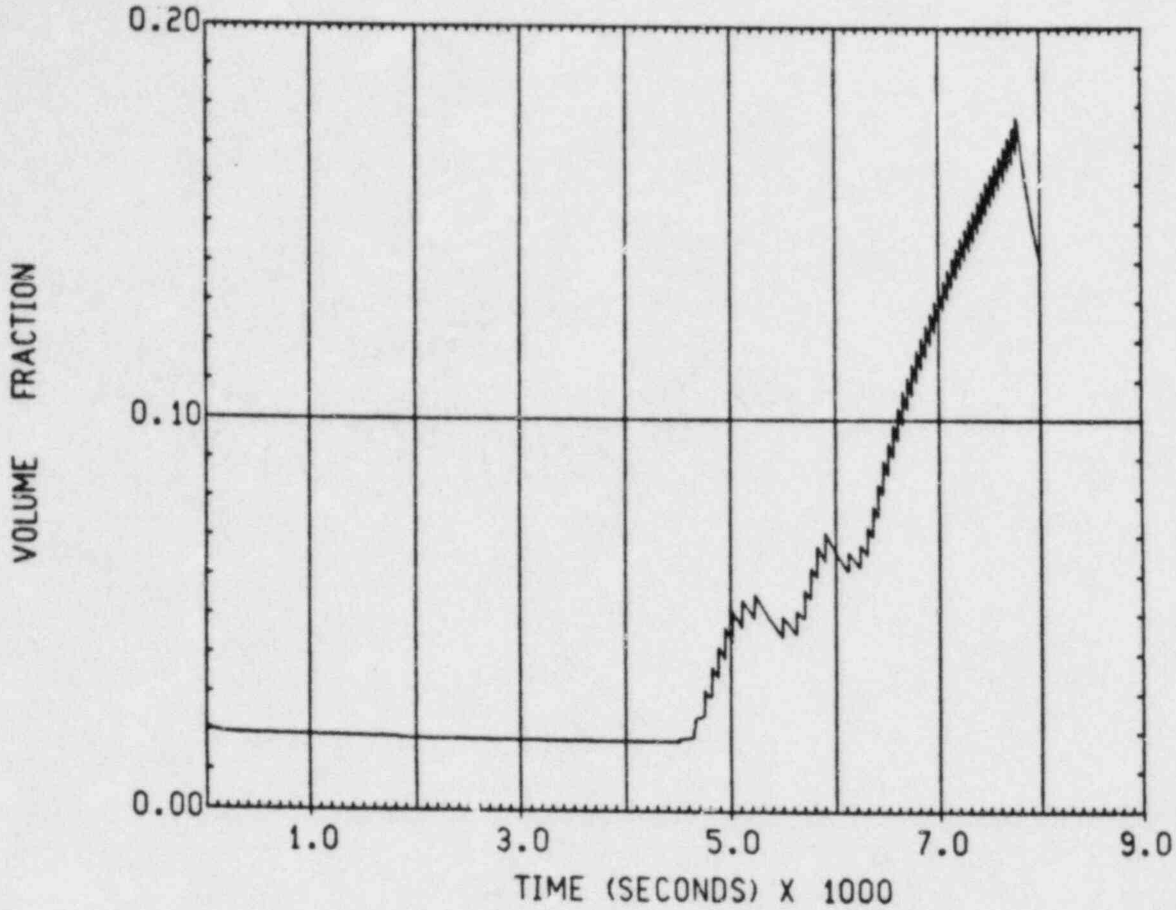

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
DRYWELL STEAM GAS CONCENTRATION

FIGURE 45



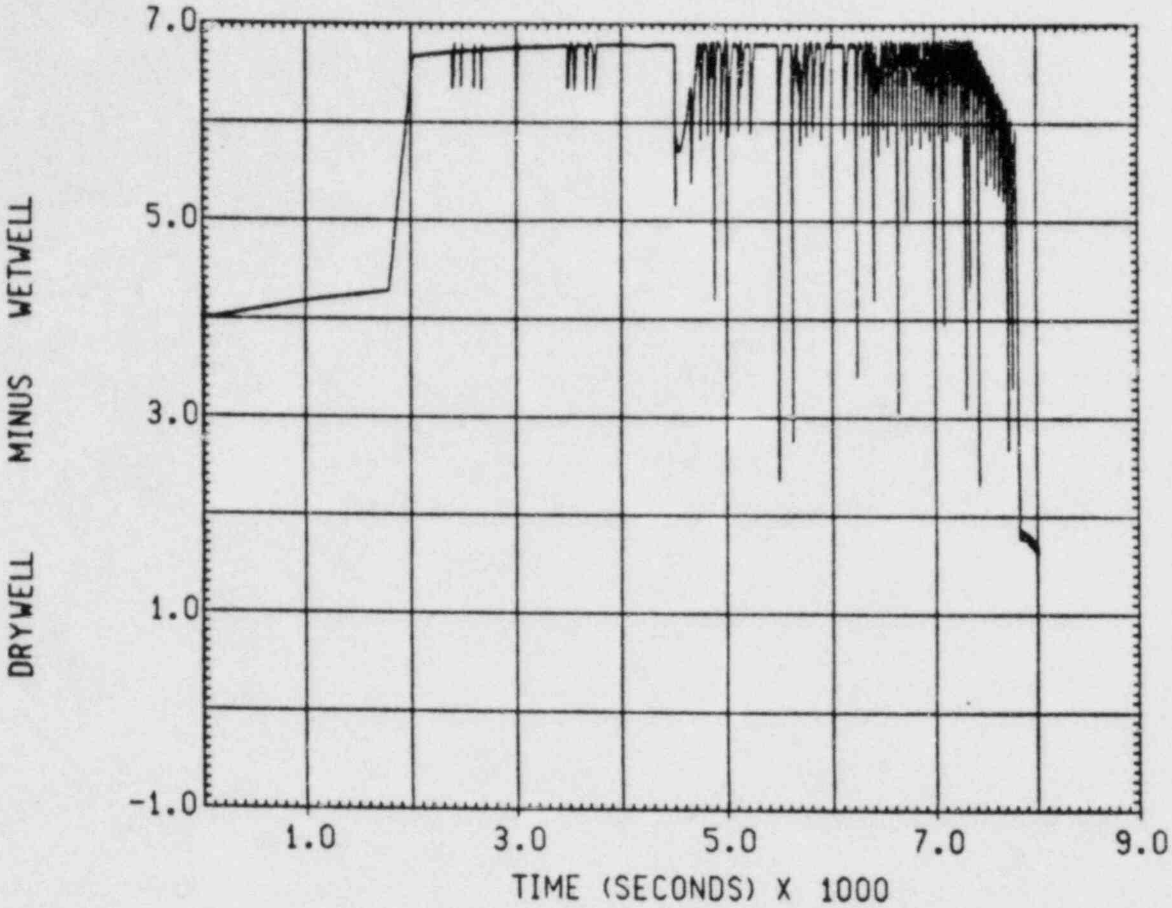

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
WETWELL STEAM GAS CONCENTRATION

FIGURE 46



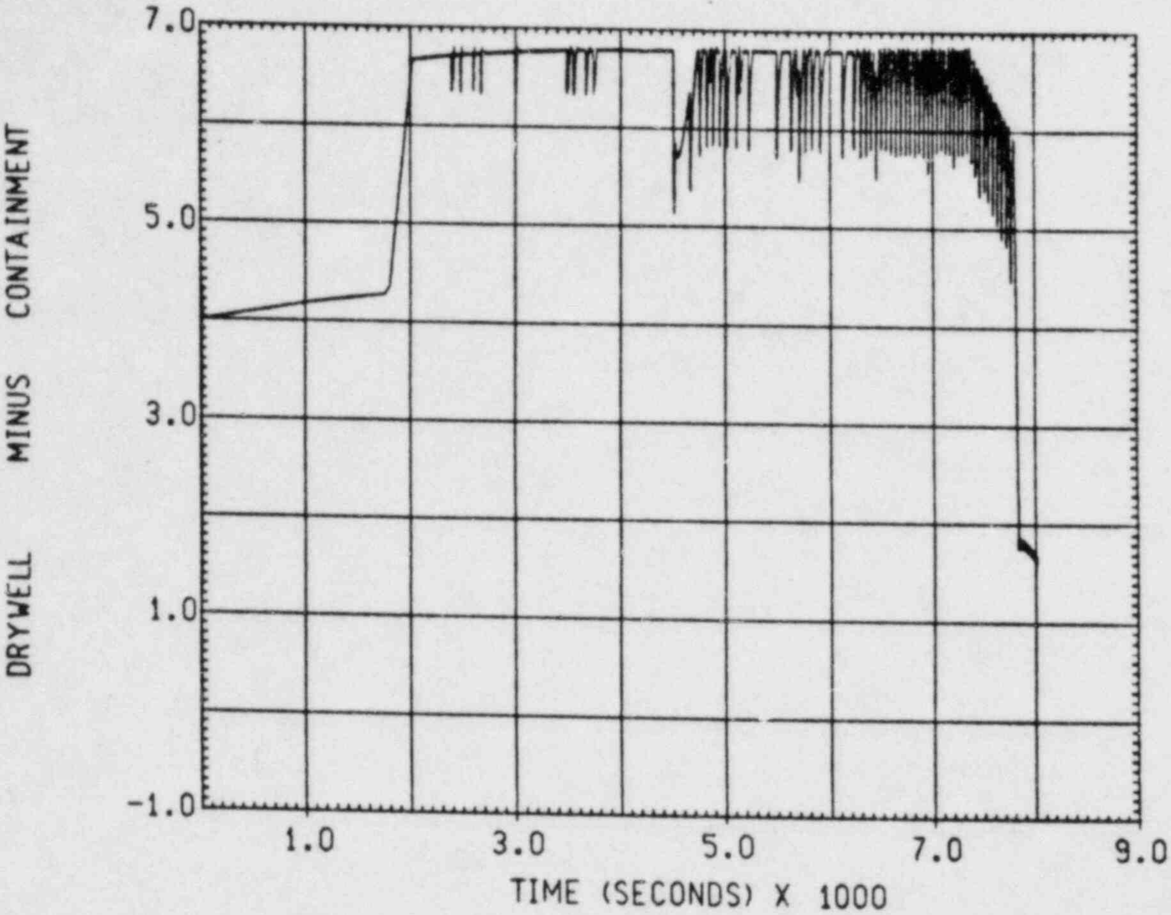

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
CONTAINMENT STEAM GAS CONCENTRATION

FIGURE 47



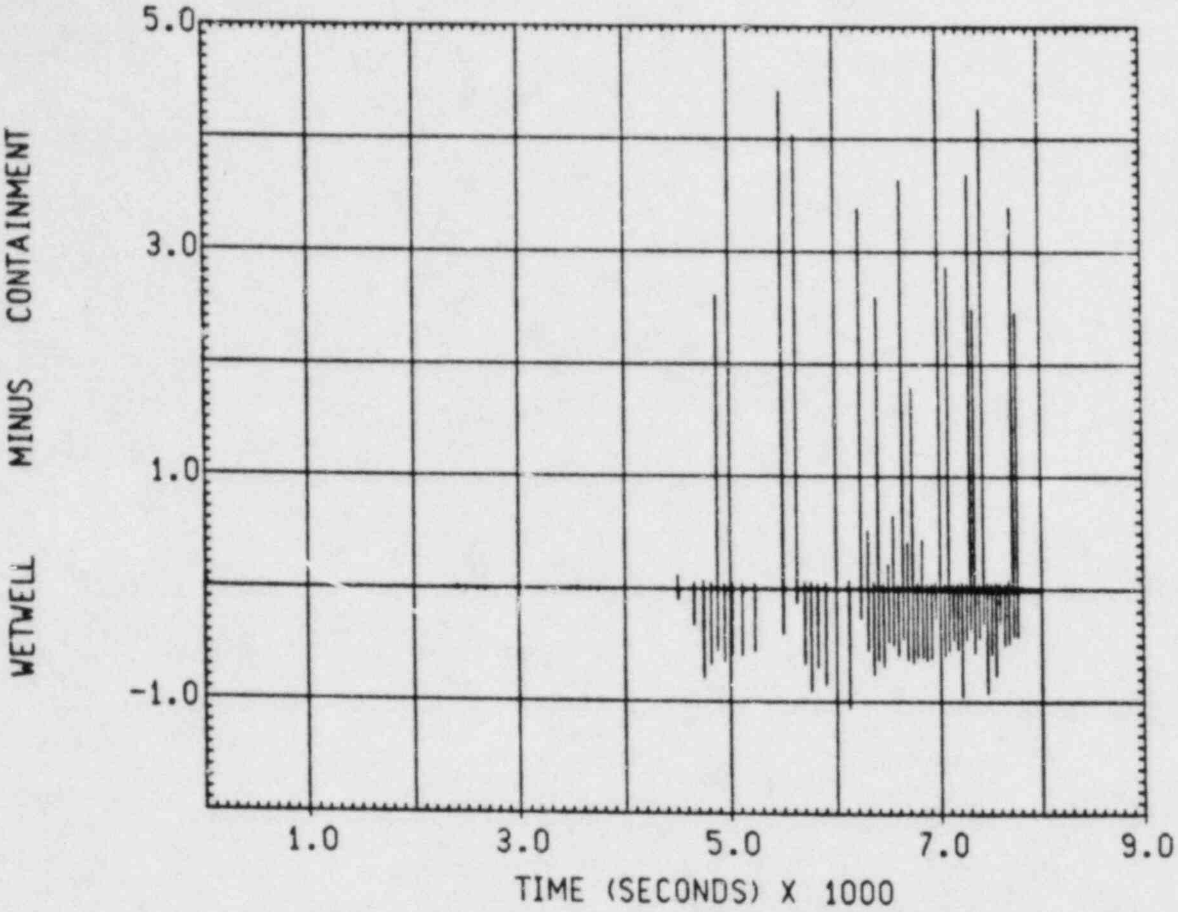

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
DIFFERENTIAL PRESSURE

FIGURE 48



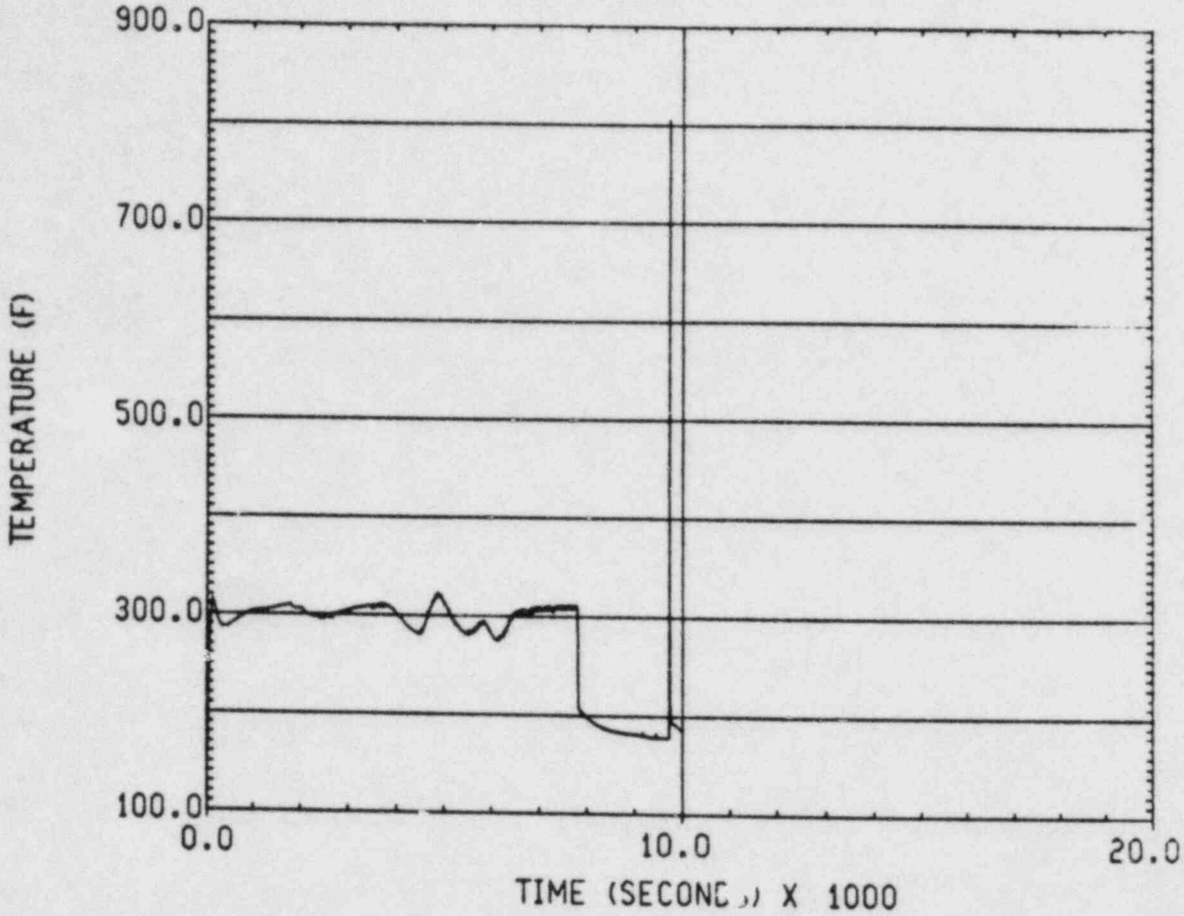

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
DIFFERENTIAL PRESSURE

FIGURE 49



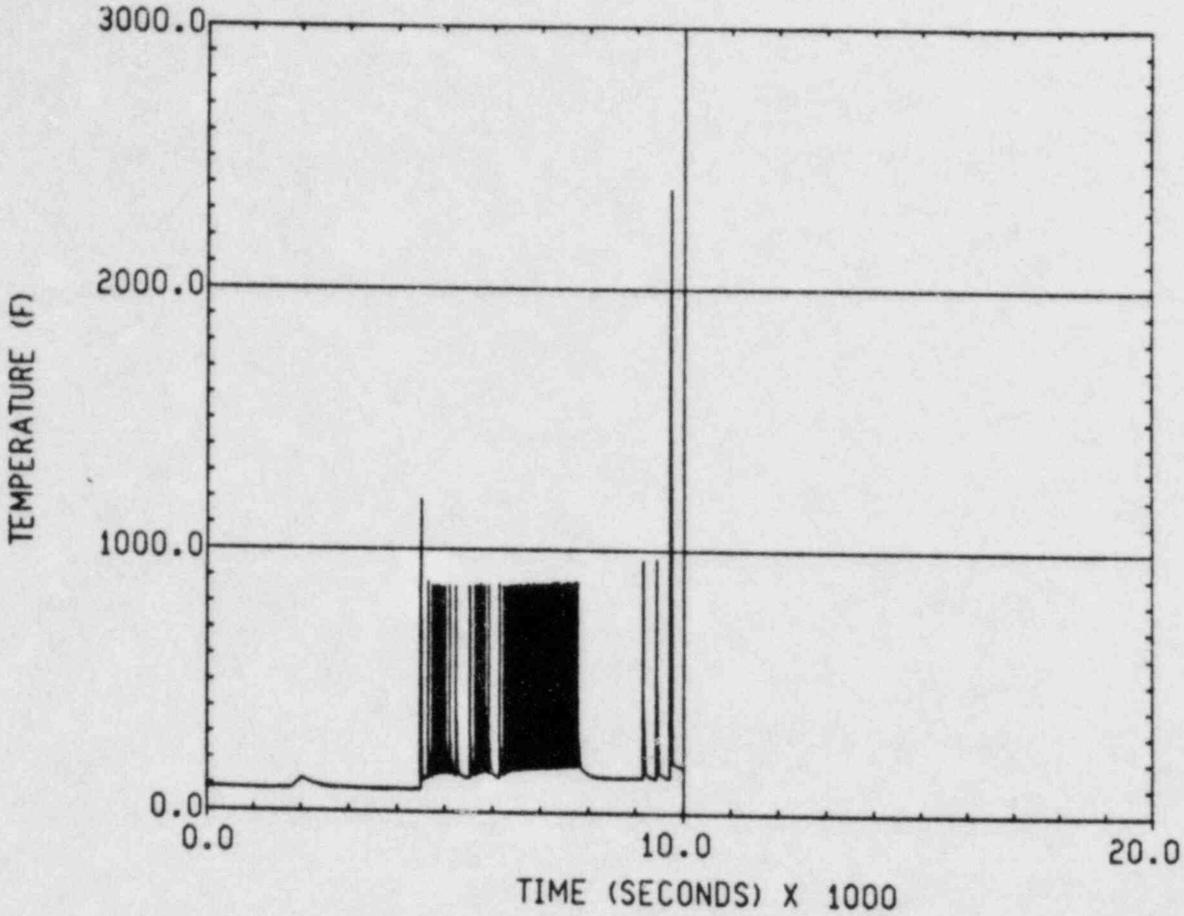

70/30 DRYWELL BREAK CONTINUOUS BURN
GRAND GULF NUCLEAR STATION
DIFFERENTIAL PRESSURE

FIGURE 50



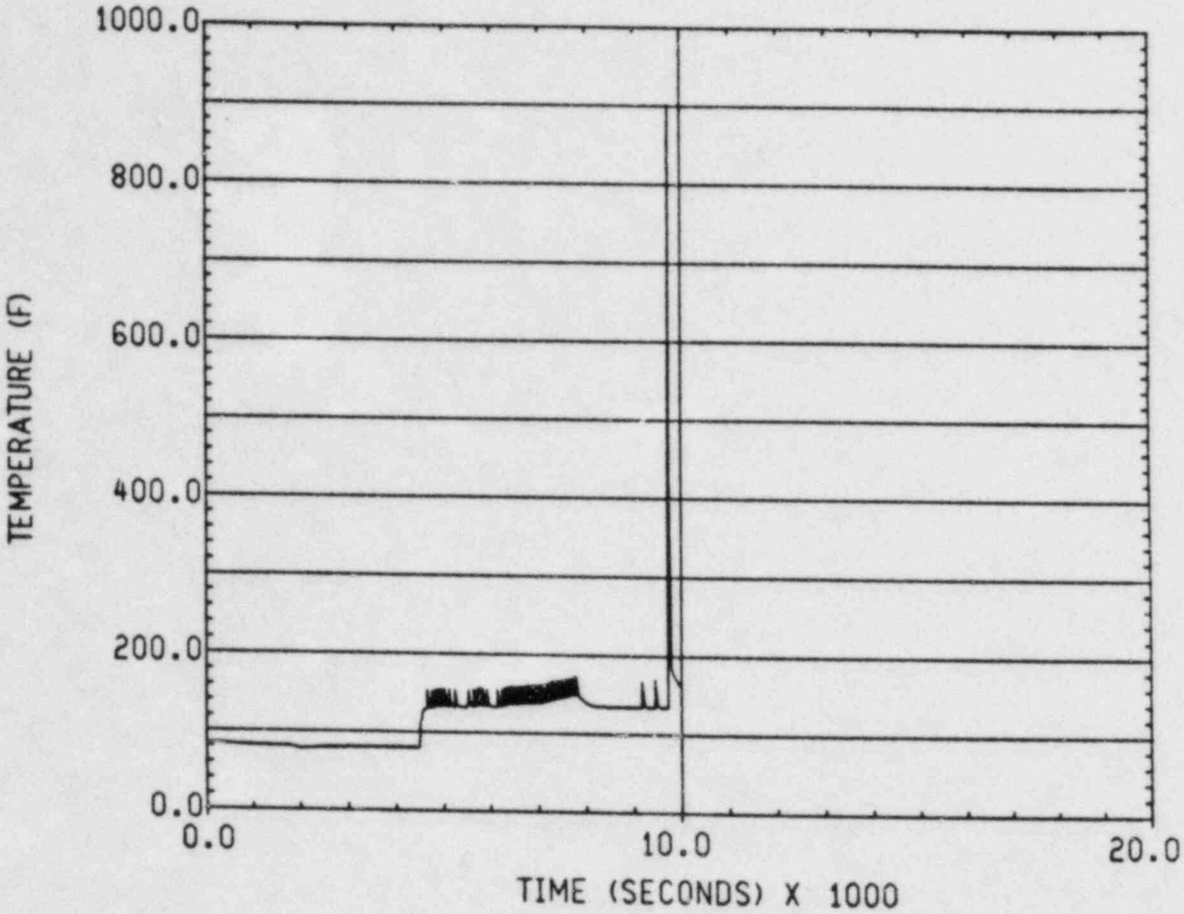

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
DRYWELL TEMPERATURE

FIGURE 51



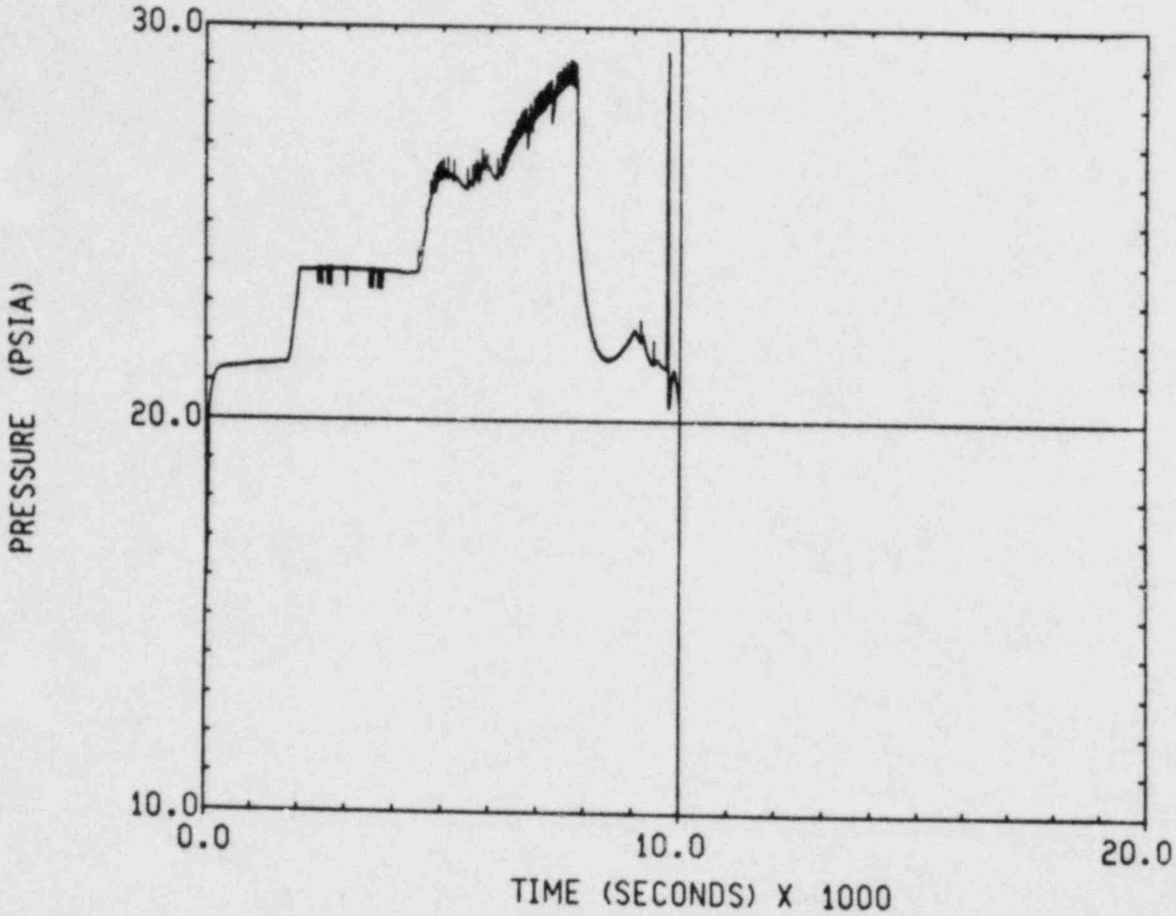

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
WETWELL TEMPERATURE

FIGURE 52



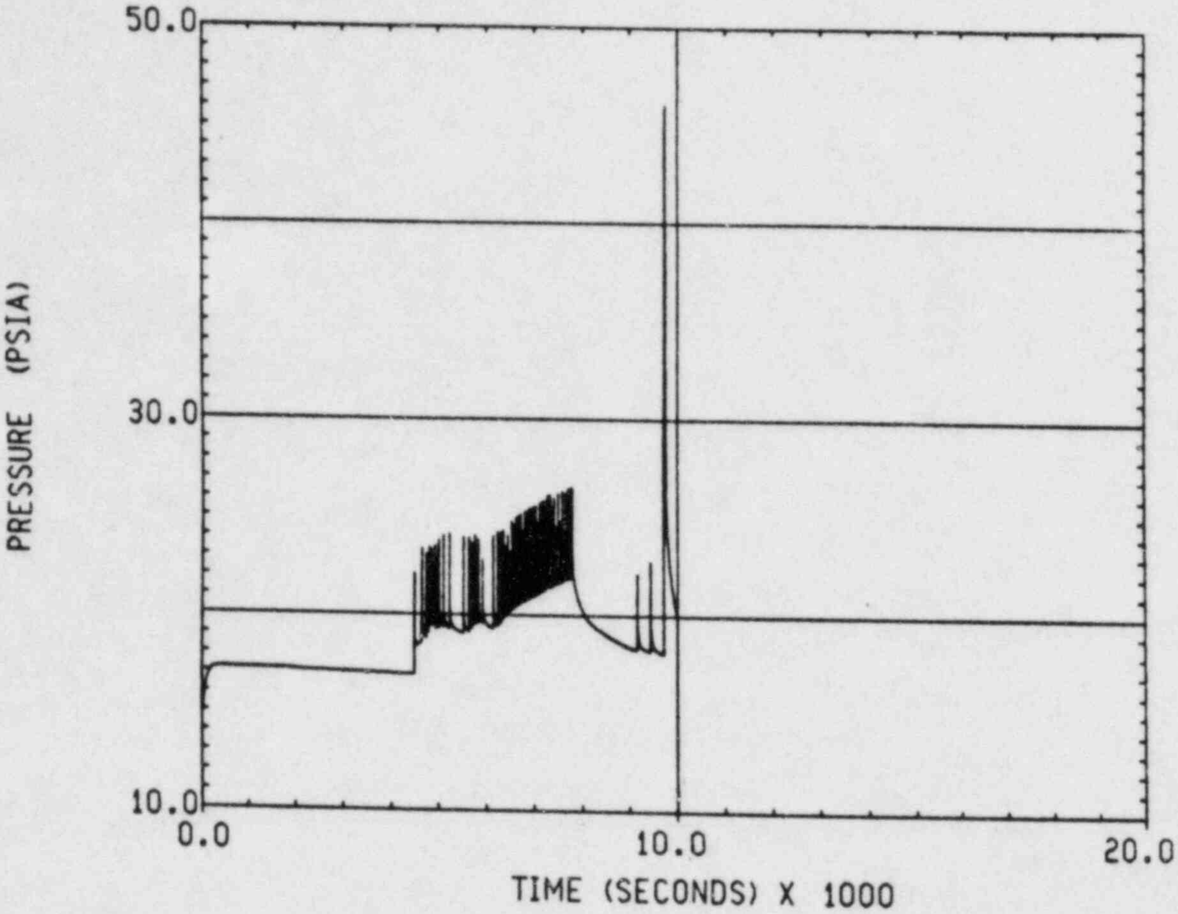

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
CONTAINMENT TEMPERATURE

FIGURE 53



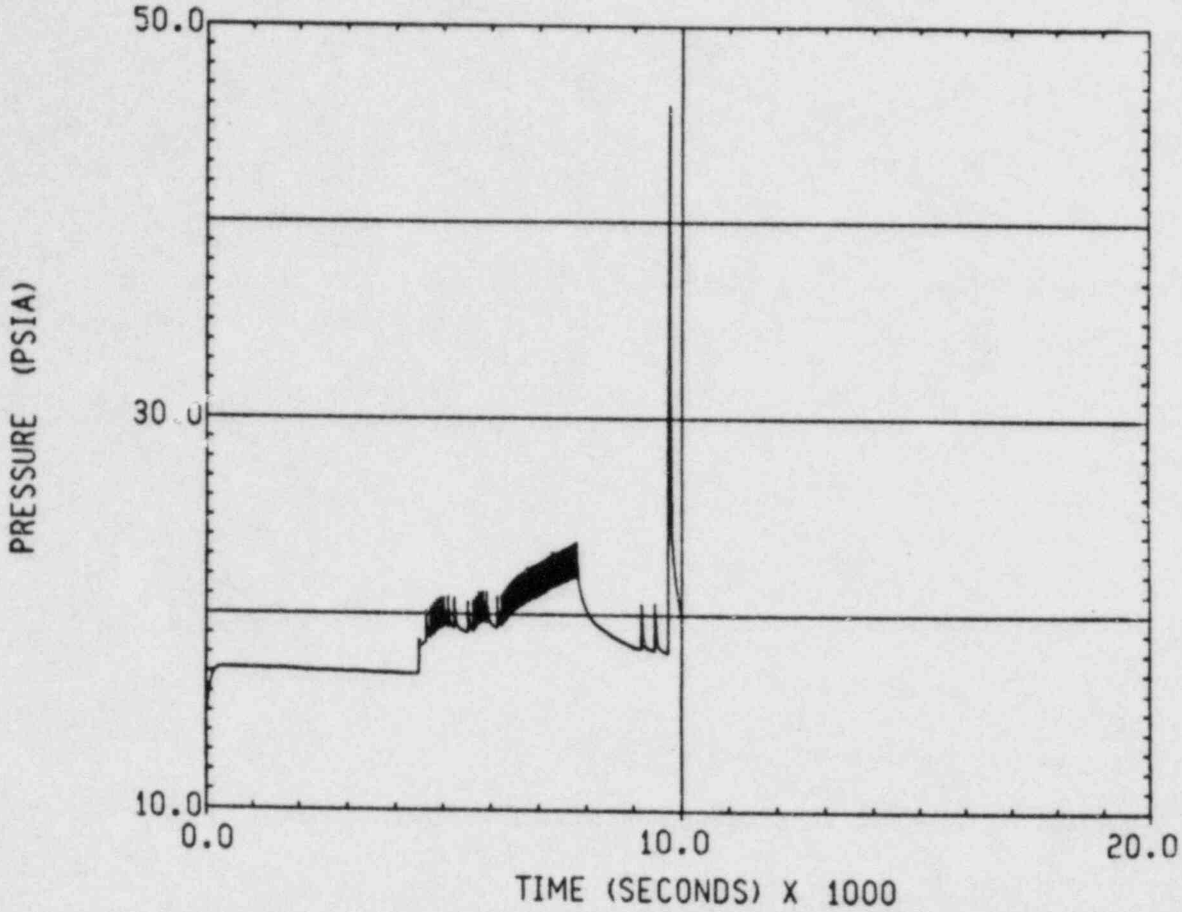

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
DRYWELL PRESSURE

FIGURE 54



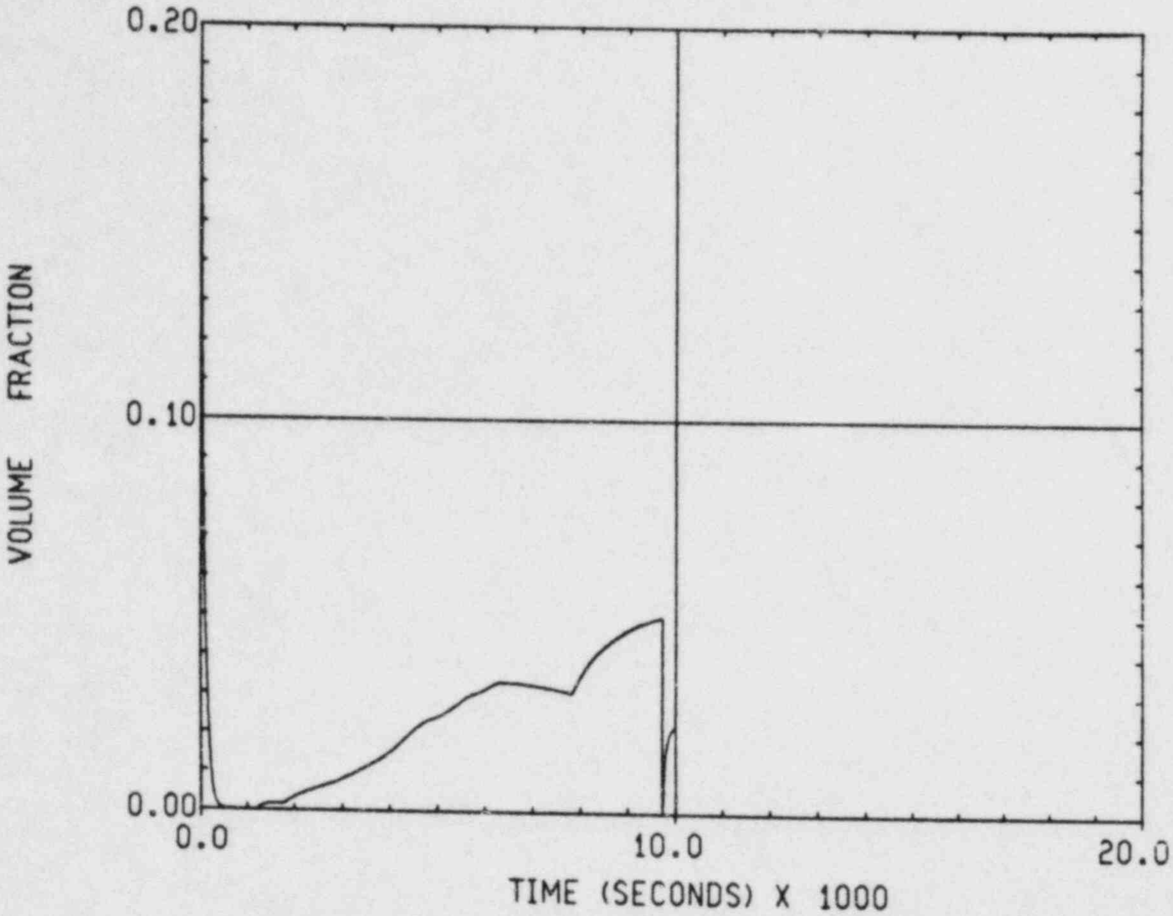

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
WETWELL PRESSURE

FIGURE 55



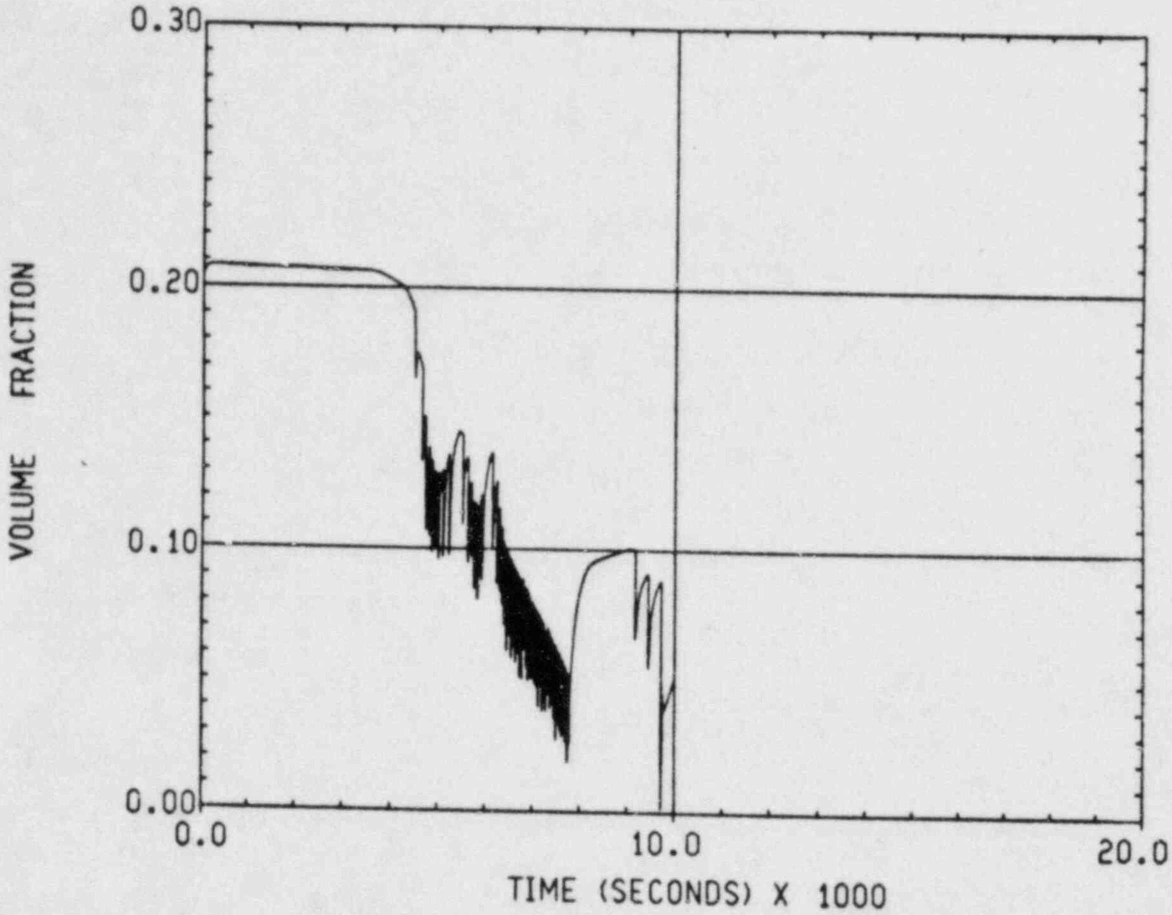

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
CONTAINMENT PRESSURE

FIGURE 56



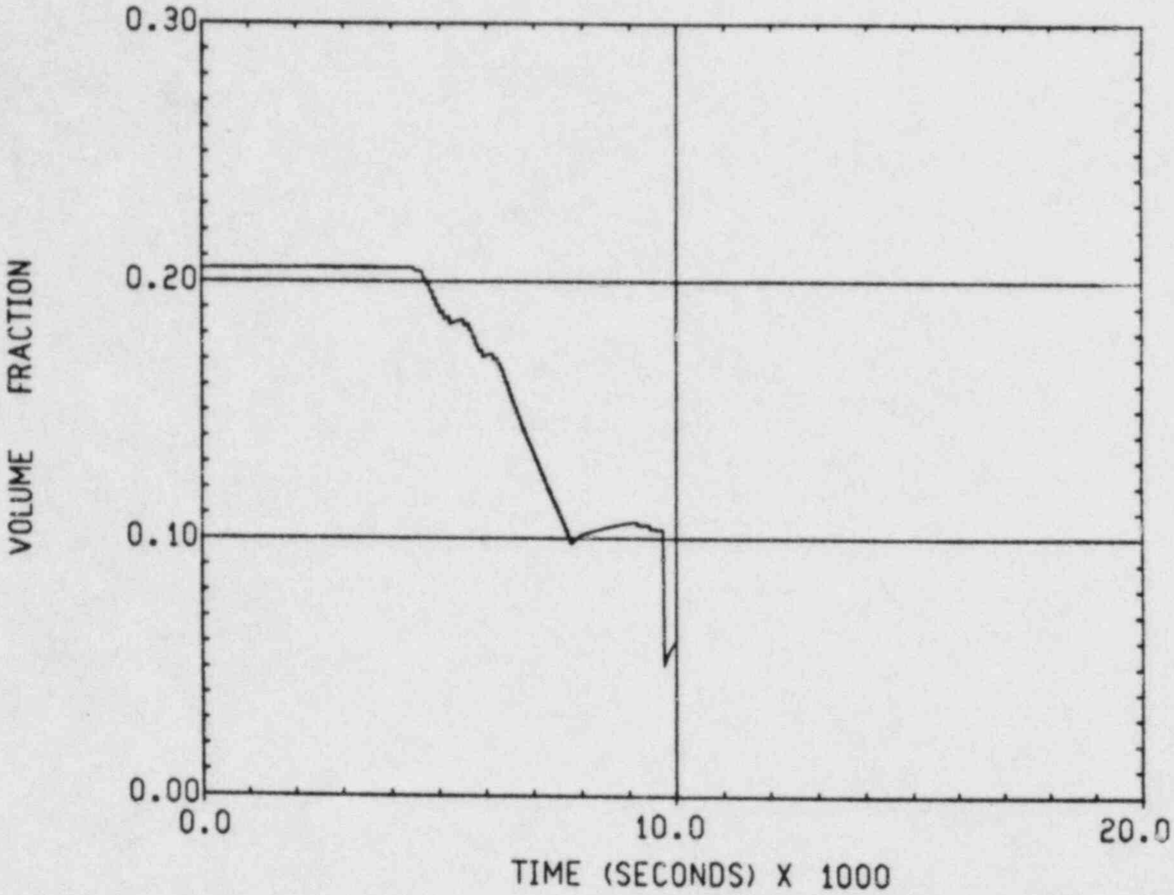

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
DRYWELL O2 GAS CONCENTRATION

FIGURE 57



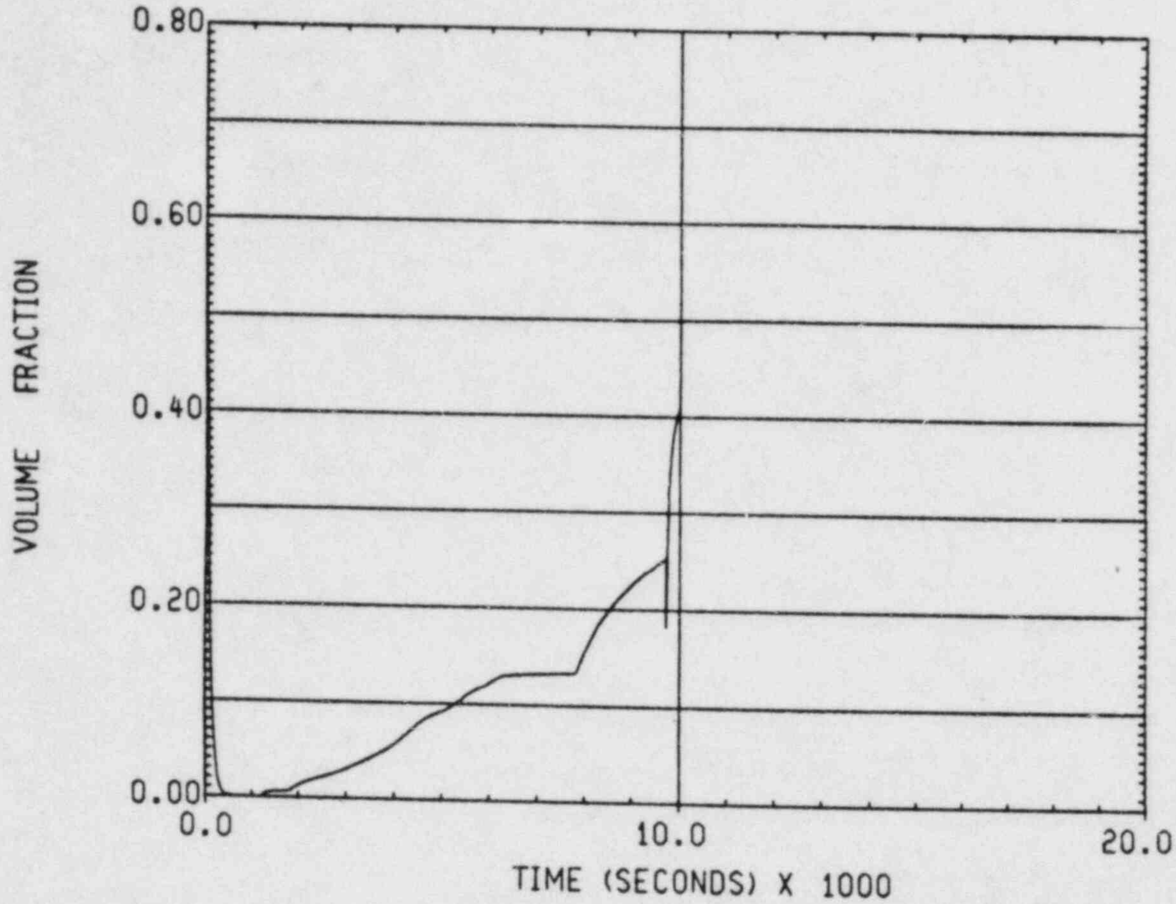

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
WETWELL 02 GAS CONCENTRATION

FIGURE 58



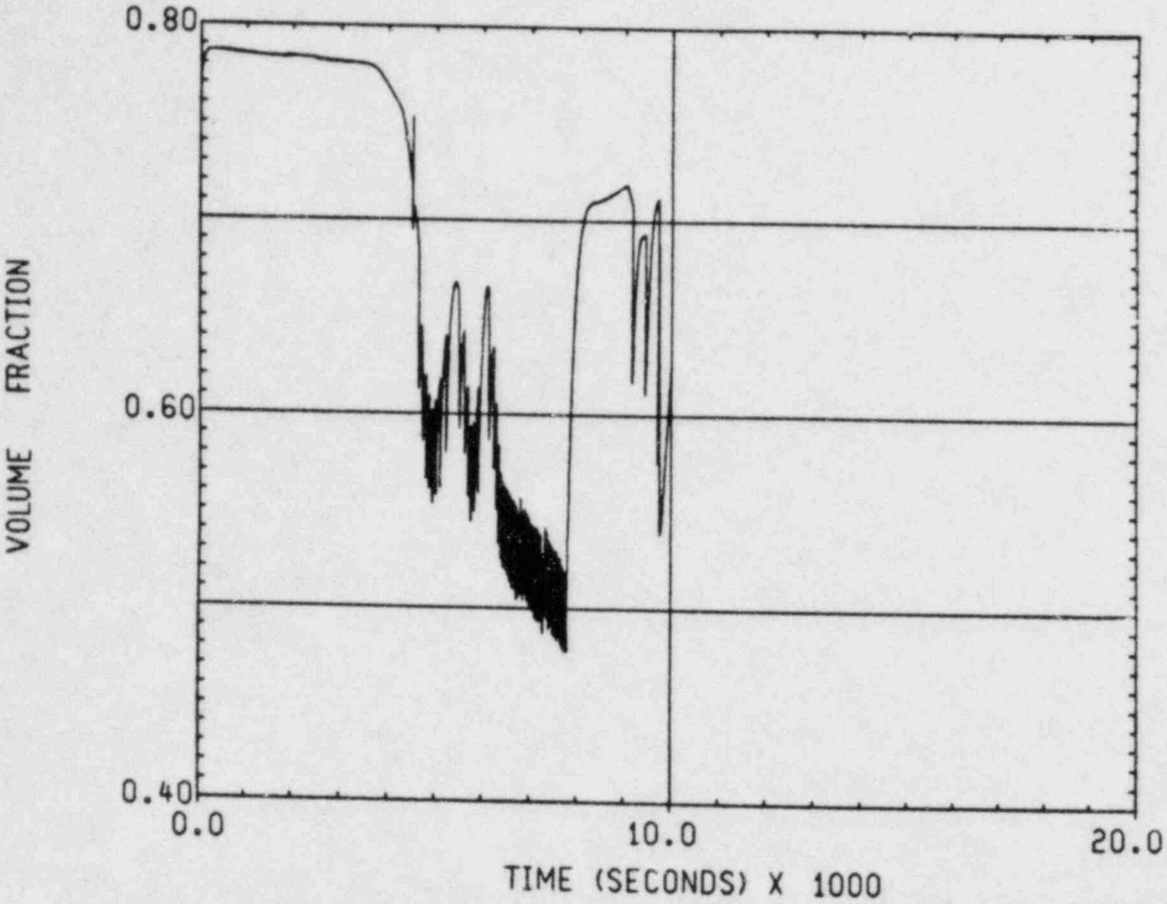

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
CONTAINMENT 02 GAS CONCENTRATION

FIGURE 59



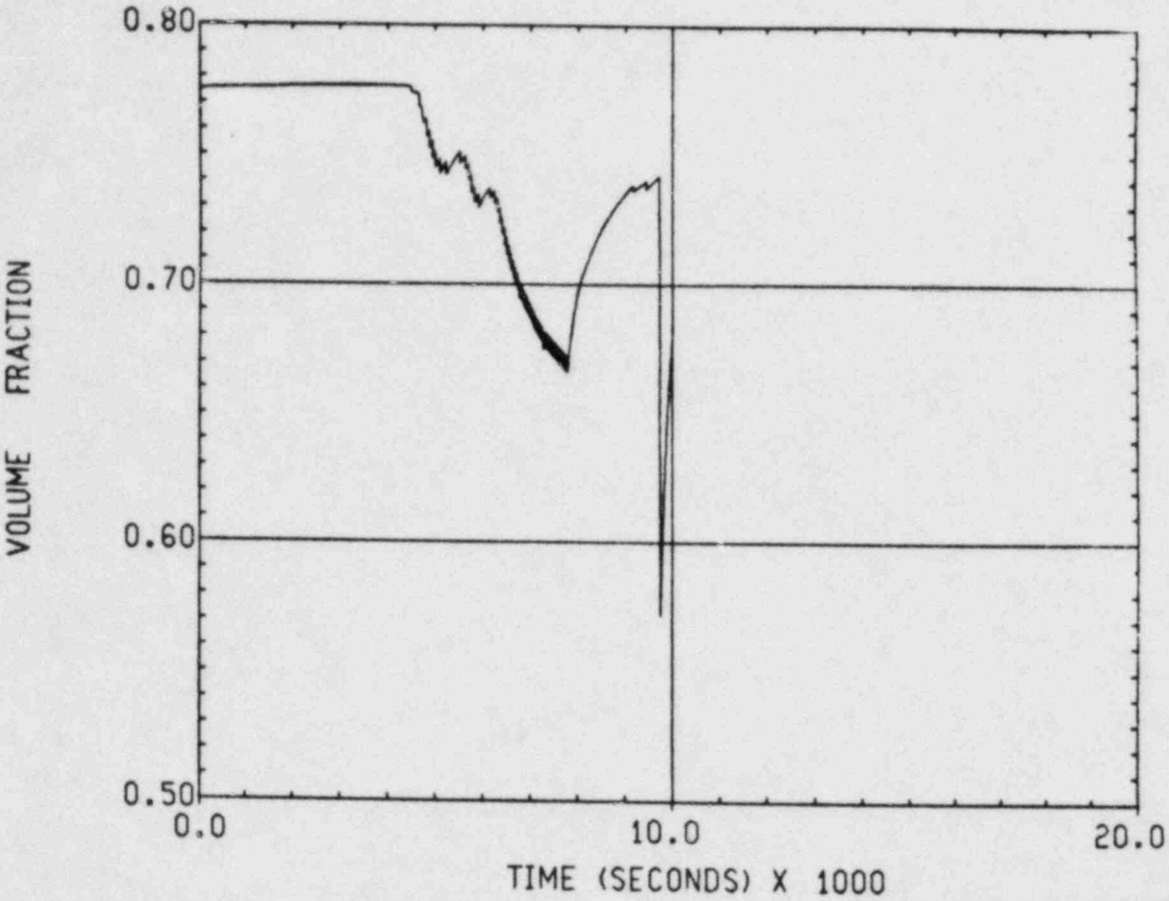

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
DRYWELL N2 GAS CONCENTRATION

FIGURE 60



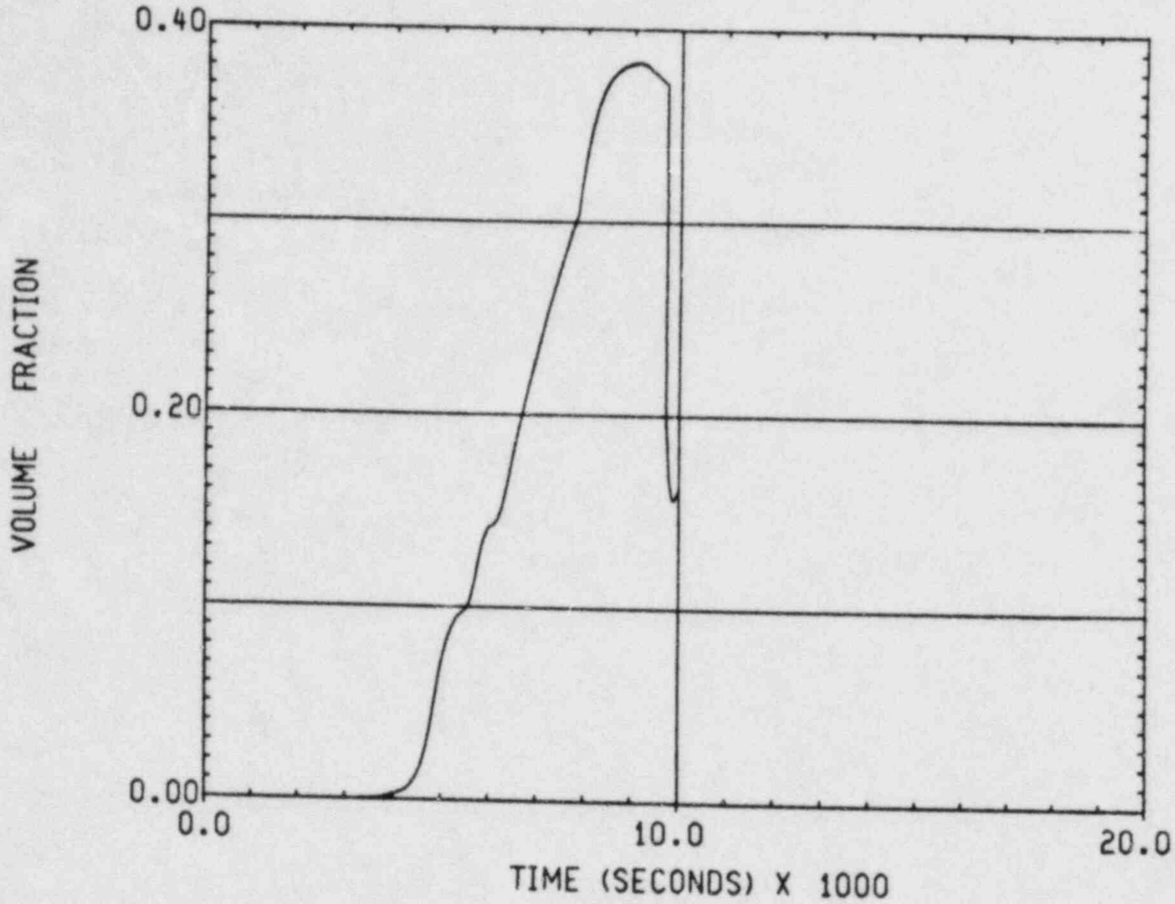

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
WETWELL N2 GAS CONCENTRATION

FIGURE 61



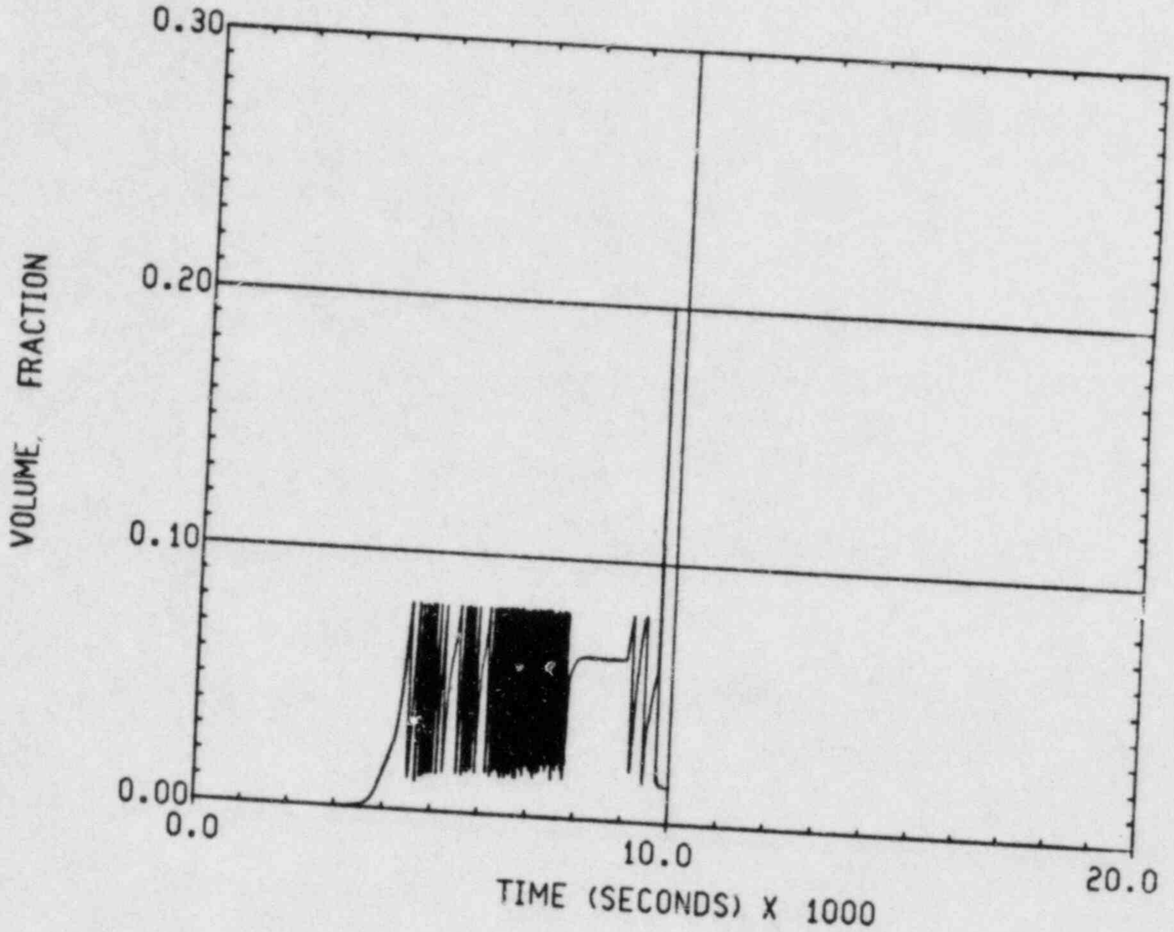

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
CONTAINMENT N2 GAS CONCENTRATION

FIGURE 62



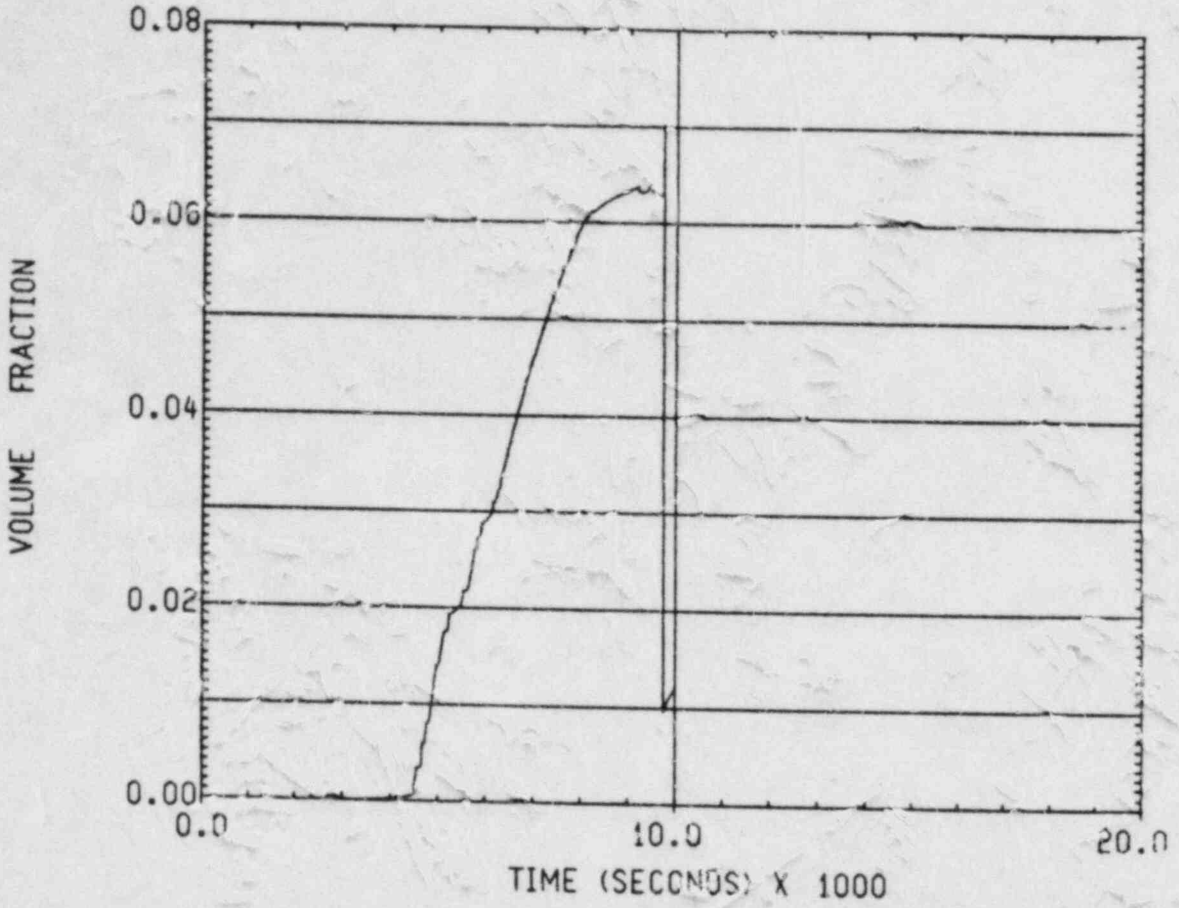

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
DRYWELL H2 GAS CONCENTRATION

FIGURE 63




70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
WETWELL H2 GAS CONCENTRATION

FIGURE 64

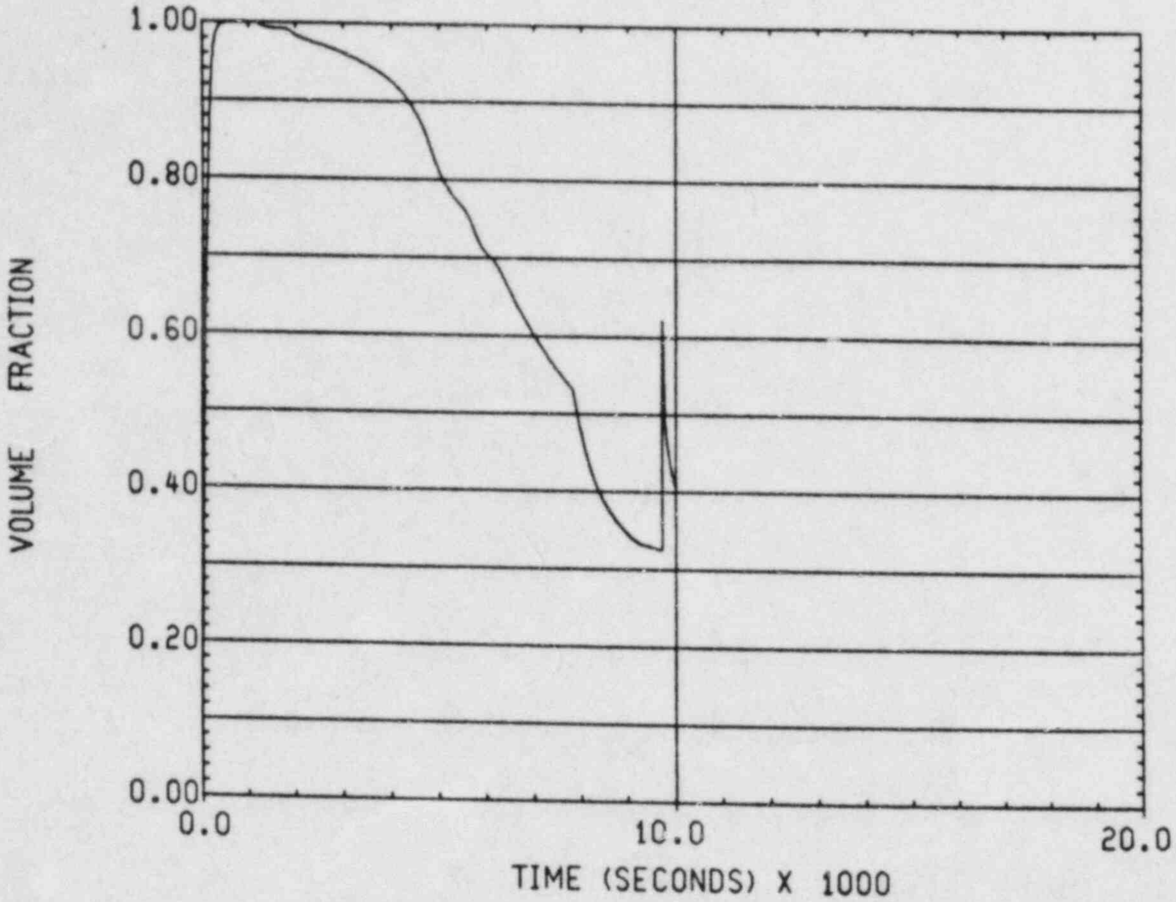



70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN

GRAND GULF NUCLEAR STATION

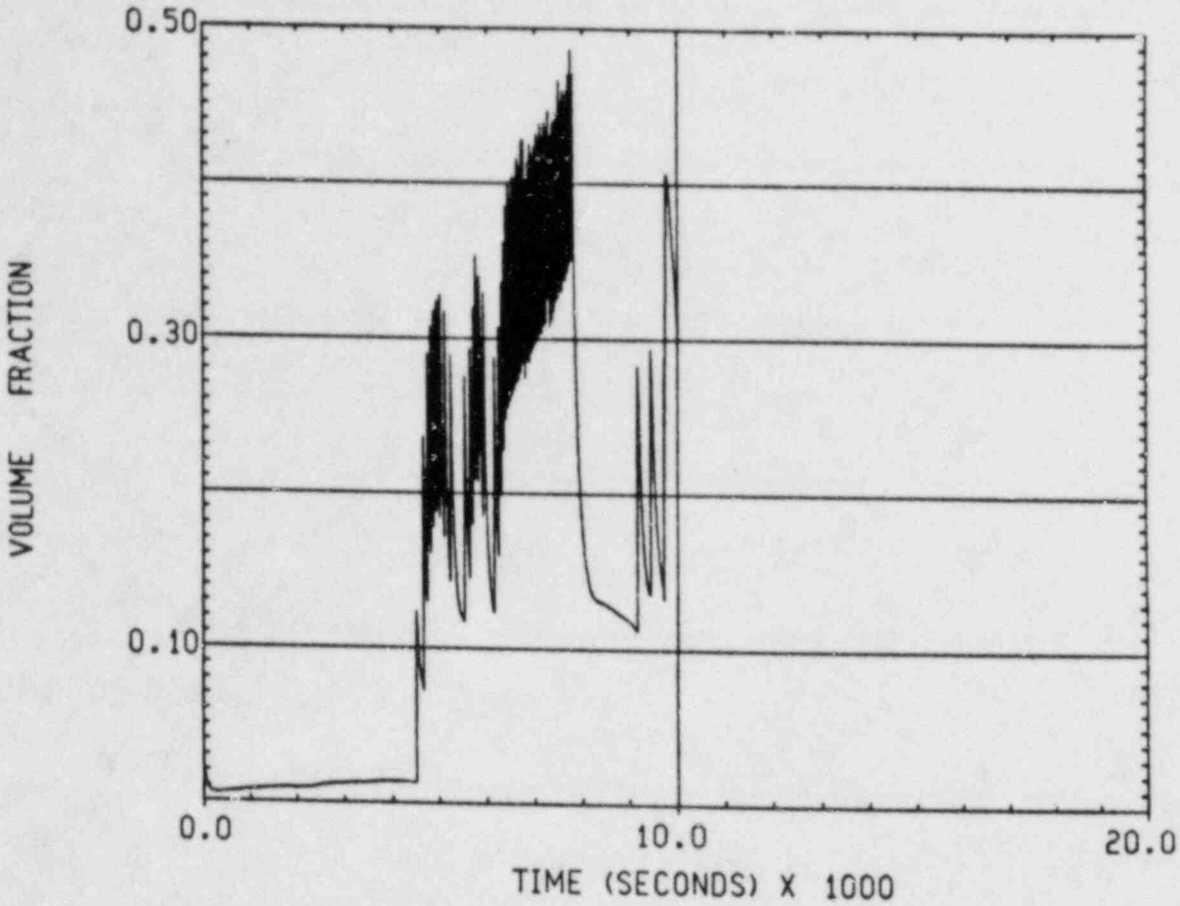
CONTAINMENT H2 GAS CONCENTRATION

FIGURE 65



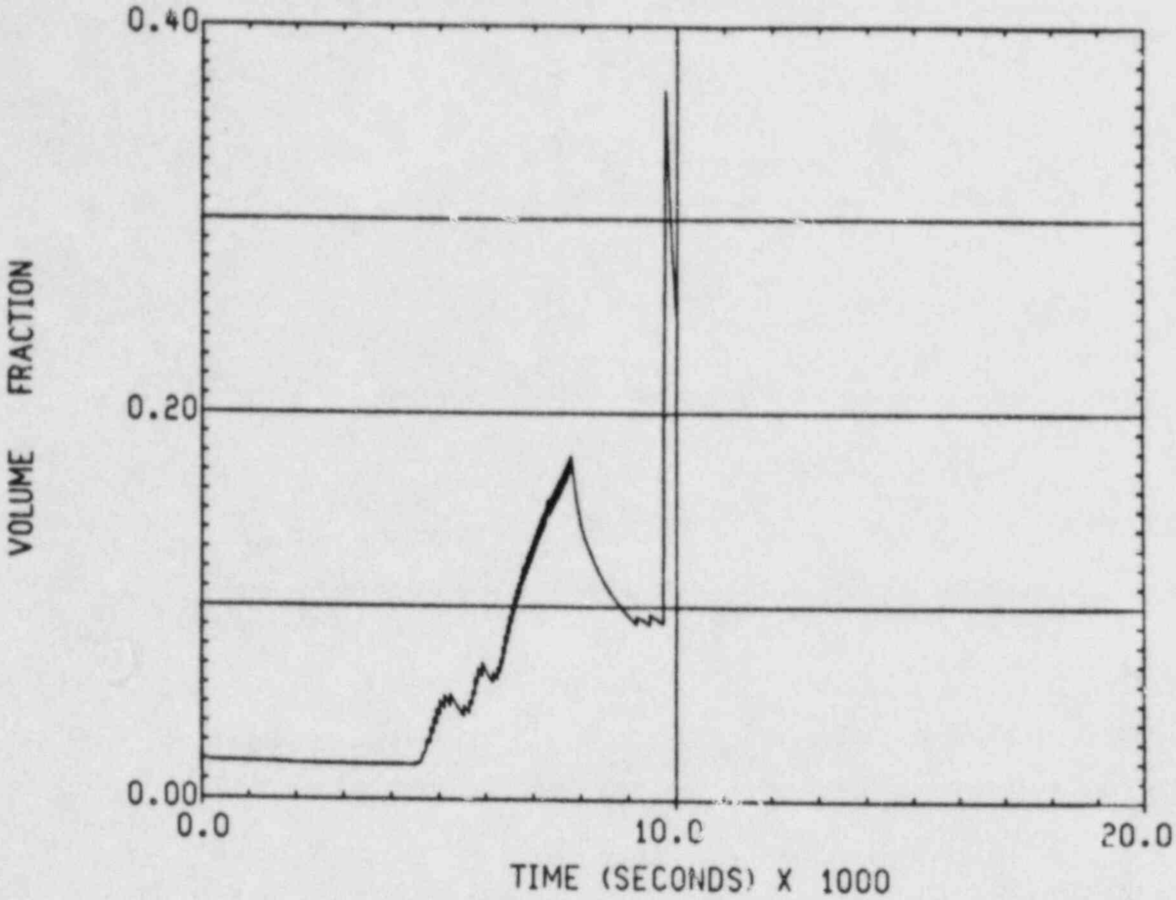

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
DRYWELL STEAM GAS CONCENTRATION

FIGURE 66



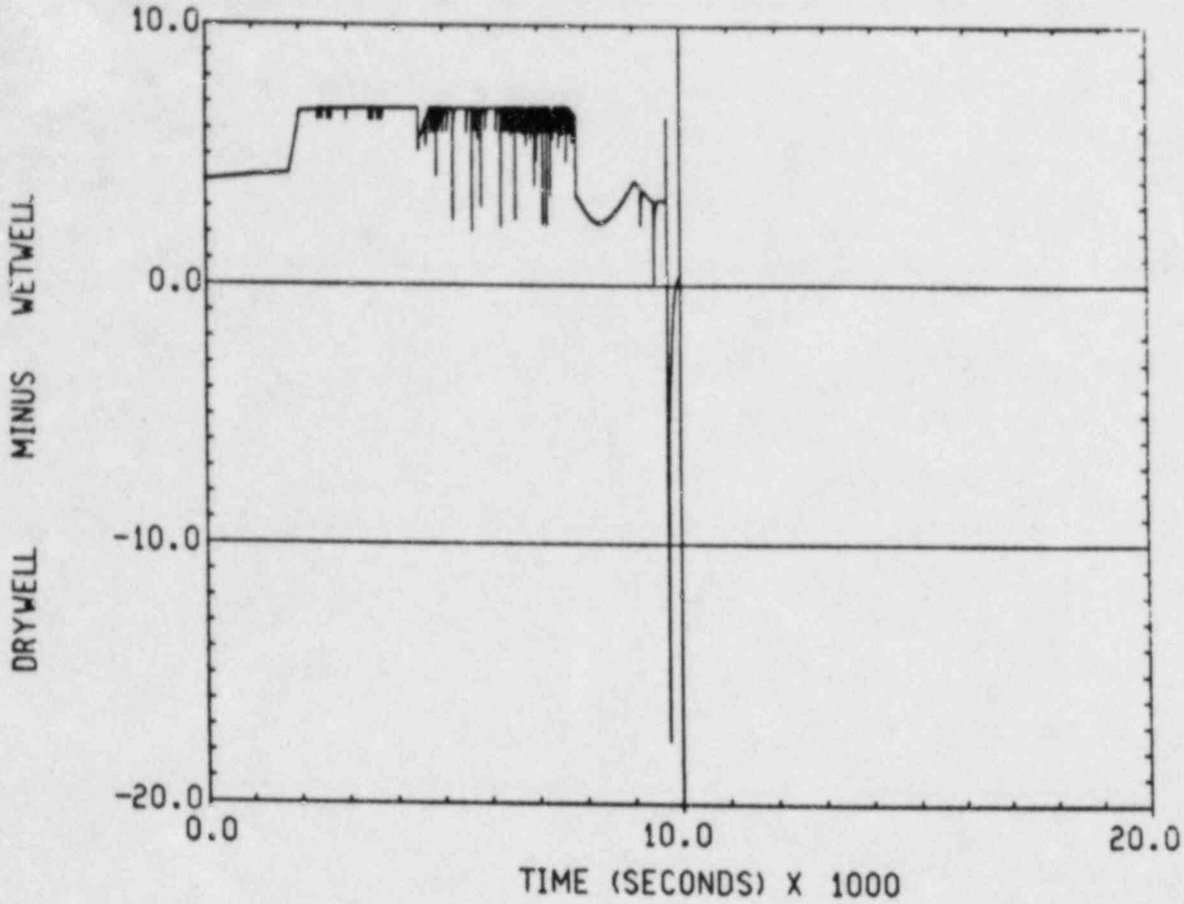

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
WETWELL STEAM GAS CONCENTRATION

FIGURE 67



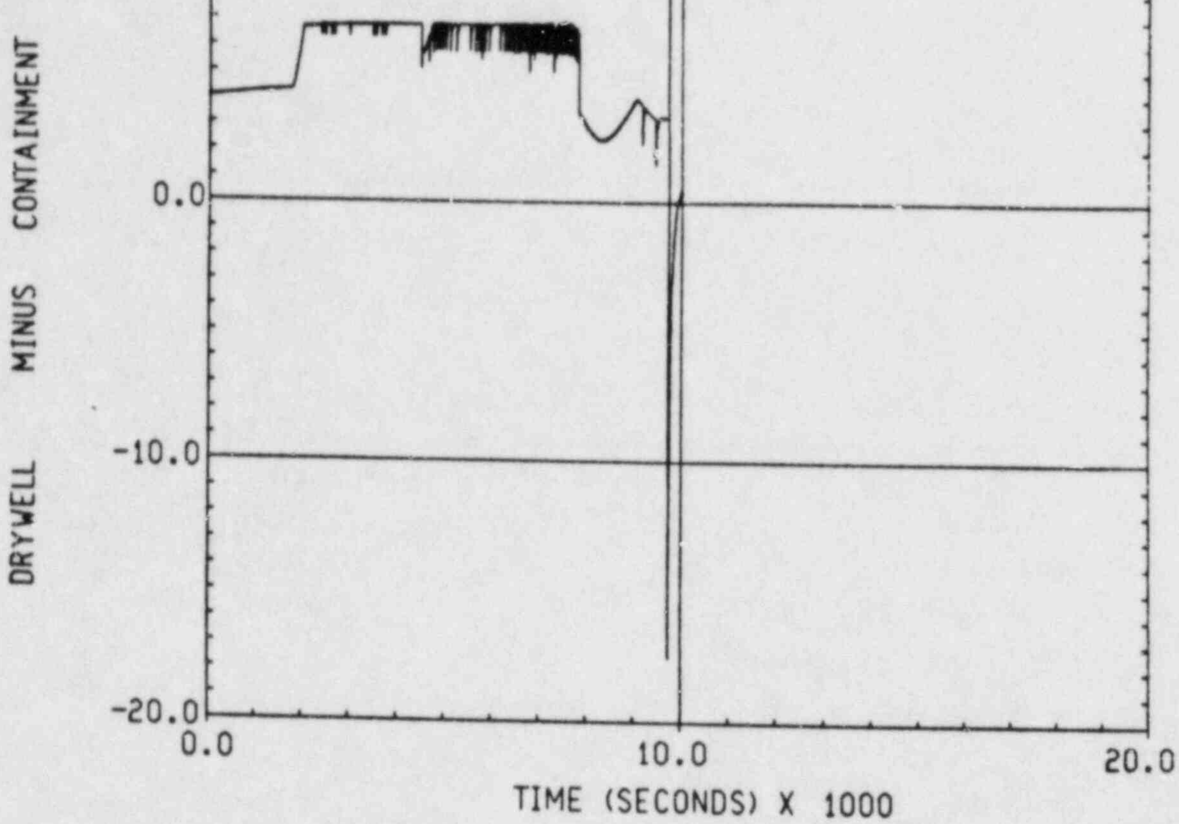

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
CONTAINMENT STEAM GAS CONCENTRATION

FIGURE 68



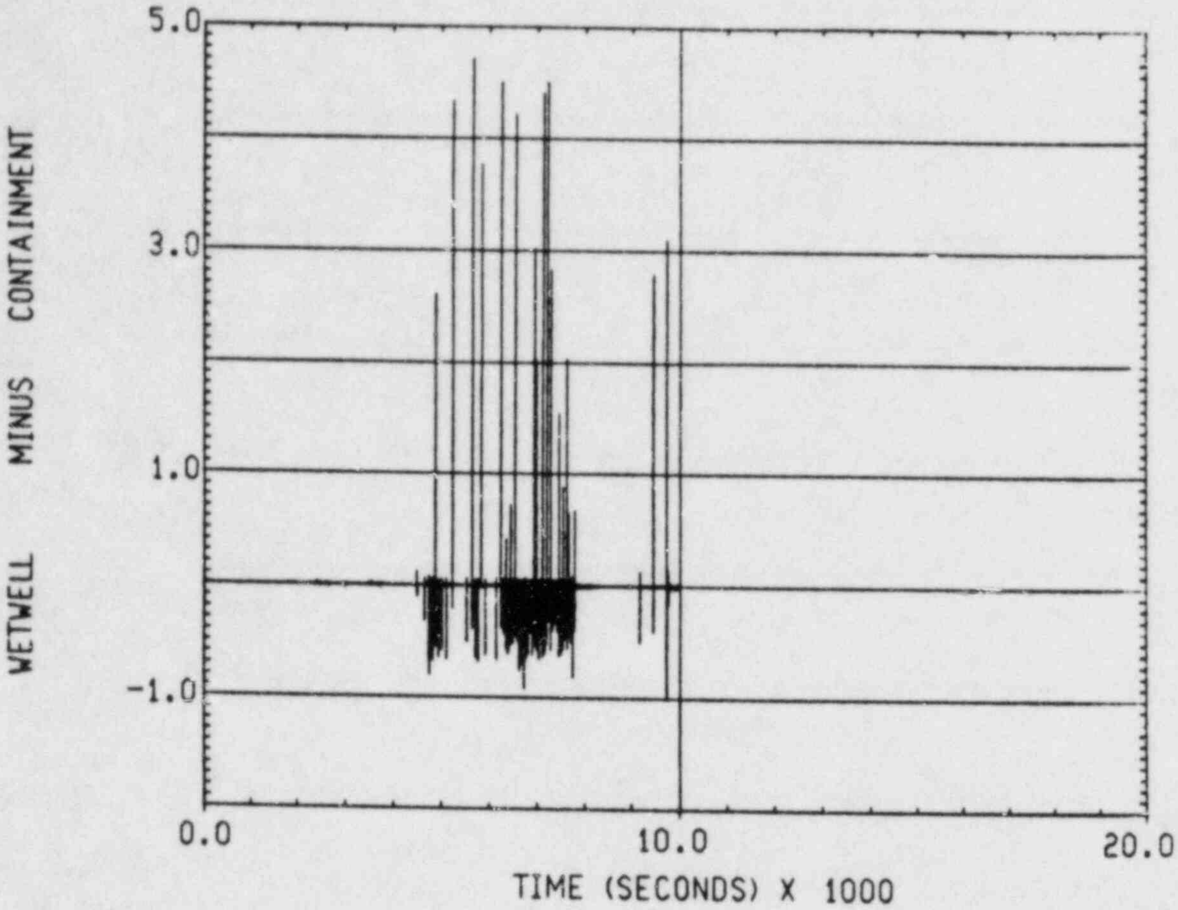

70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
DIFFERENTIAL PRESSURE

FIGURE 69




70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
DIFFERENTIAL PRESSURE

FIGURE 70




70/30 DRYWELL BREAK WITH MODIFIED DRAW DOWN
GRAND GULF NUCLEAR STATION
DIFFERENTIAL PRESSURE

FIGURE 71

## APPENDIX A

### Flow Split Determination

This appendix deals with the evaluation of the blowdown flow split between the Safety Relief Valves (SRVs) and the drywell given a small break LOCA in the drywell. Many assumptions were made in performing this calculation and are discussed below.

For non-choked or subcritical flow, the following equations are used in deriving the flow equations:

$$Q = AV; \; h = \frac{KV^2}{2g}; \quad P = \rho g h$$

where
- A = area
- V = velocity
- K = loss coefficient
- $\rho$ = density
- g = gravitational acceleration
- Q = volumetric flow rate
- h = head
- P = differential pressure

Using the above equations a general flow equation can be derived of the following form:

$$Q = A\sqrt{2\Delta P/\rho k}$$

Given this general flow equation, break flow and safety relief valve flow can be calculated. For break flow, the pressure differential is the difference in reactor pressure and drywell pressure. For SRV flow, the pressure differential is between the reactor and the exit of the SRV spargers at the bottom of the suppression pool. It was assumed that during blowdown the suppression pool level would be located at the center of the first (top) row of vents. From the suppression pool dimensions, the exit of the spargers are located approximately 6'-3-3/4" below the center of the first row of vents. This distance equates to approximately 2.7 psi pressure differential between the drywell and the exit of the spargers.

The density of flow was assumed to be that of steam at 1000°F and reactor pressure. Loss coefficients were assumed to be equal for both SRV and break flows at a dimensionless value of 2. The drywell pressure was assumed to be a constant value of 25 psia.

A break area of 0.163 $ft^2$ was assumed, the same value used in the MARCH TPE run used in the Grand Gulf CLASIX-3 analysis. Nine SRVs were assumed to be open, each with an area equal to the break flow area.

Given these assumptions, the non-choked flow equations for the drywell and the SRVs can be determined.

$$Q_{Drywell} = 0.163 \sqrt{4636.8\, \upsilon_R\, (P_R - 25)}$$

$$Q_{SRV} = 9\ (0.163) \sqrt{32.2\, \upsilon_R\, [144\ (P_R - 25) - 392.45]}$$

where $P_R$ = reactor pressure (psia)

$\upsilon_R$ = specific volume @ $P_R$ and 1000°F ($ft^3$/lbm)

Q = flow rate ($ft^3$/sec)

For choked or critical flow, an equation of the following form was used:

$$Q = K_{choked}\ \ P$$

An assumption was made that choked flow would occur above P upstream/ P downstream = 2. To provide continuity of flow, at a pressure ratio of 2, the choked flow equation was set equal to the non-choked flow equation to determine the value for K choked. For break flow, the pressure ratio equals 2 at a reactor pressure of 50 psia while in the SRVs, choked flow does not occur until $P_R$ equals 55.5 psia.

By equating the choked and non-choked flow equations at these pressures, the following K choked values were obtained.

$$K_{choked,drywell} = 4.62$$

$$K_{choked,SRVs} = 37.45$$

Therefore the choked flow equations become

$$Q_{drywell}\ (P_R \geq 50\ \text{psia}) = 4.62\ P_R$$

$$Q_{SRVs}\ (P_R \geq 55.5\ \text{psia}) = 37.45\ P_R$$

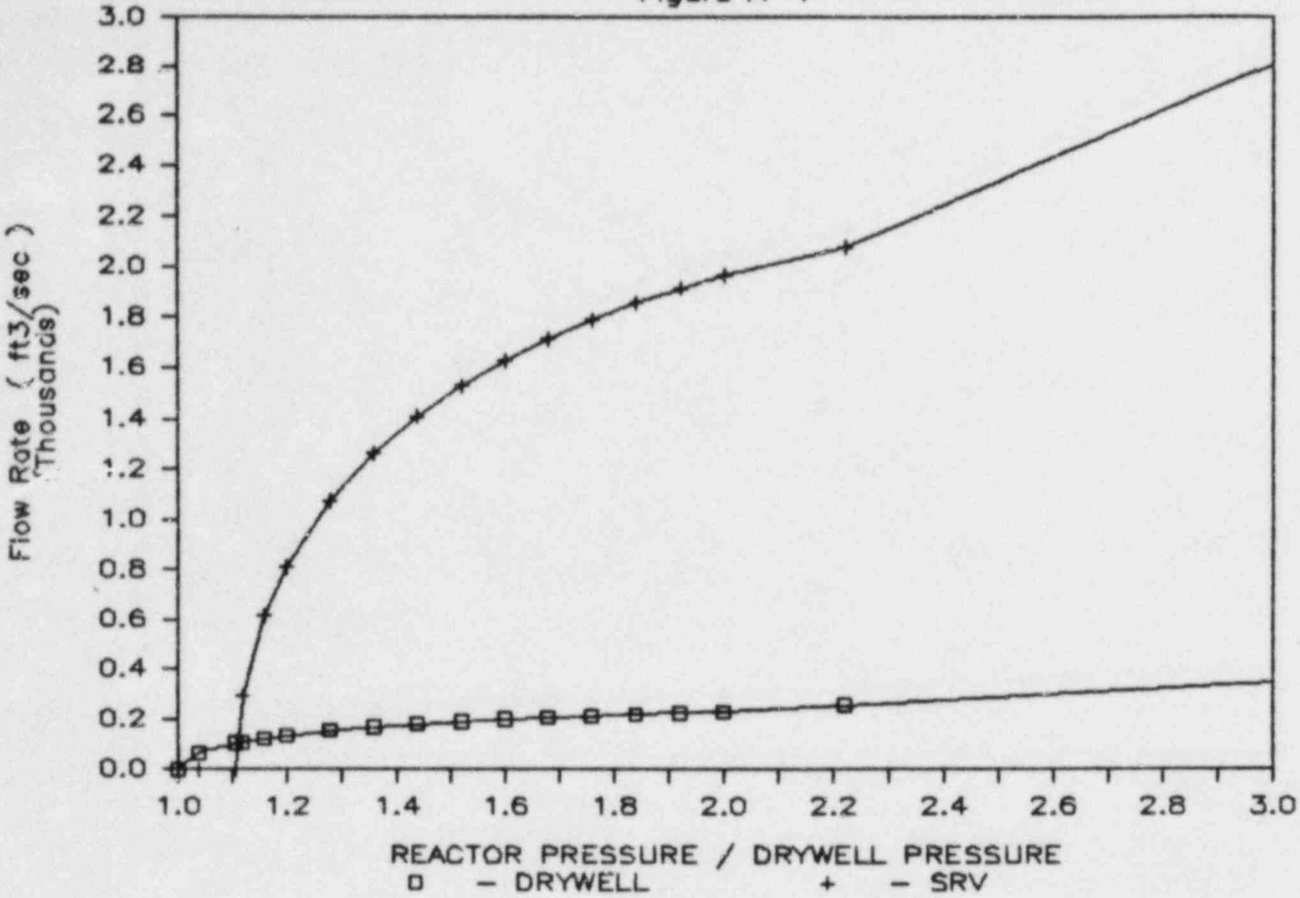
where $P_R$ = reactor pressure (psia)

$Q$ = flow rate ($ft^3$/sec)

Results using the derived equations are shown in Figures A-1 and A-2 for a range of pressures from 25 psia to 75 psia in the reactor vessel. Figures A-1 shows the flow rates from both the drywell break and the SRVs versus a ratio of reactor pressure to dry··ll pressure. Figure A-2 shows the drywell flow percentage versus the same pressure ratio.

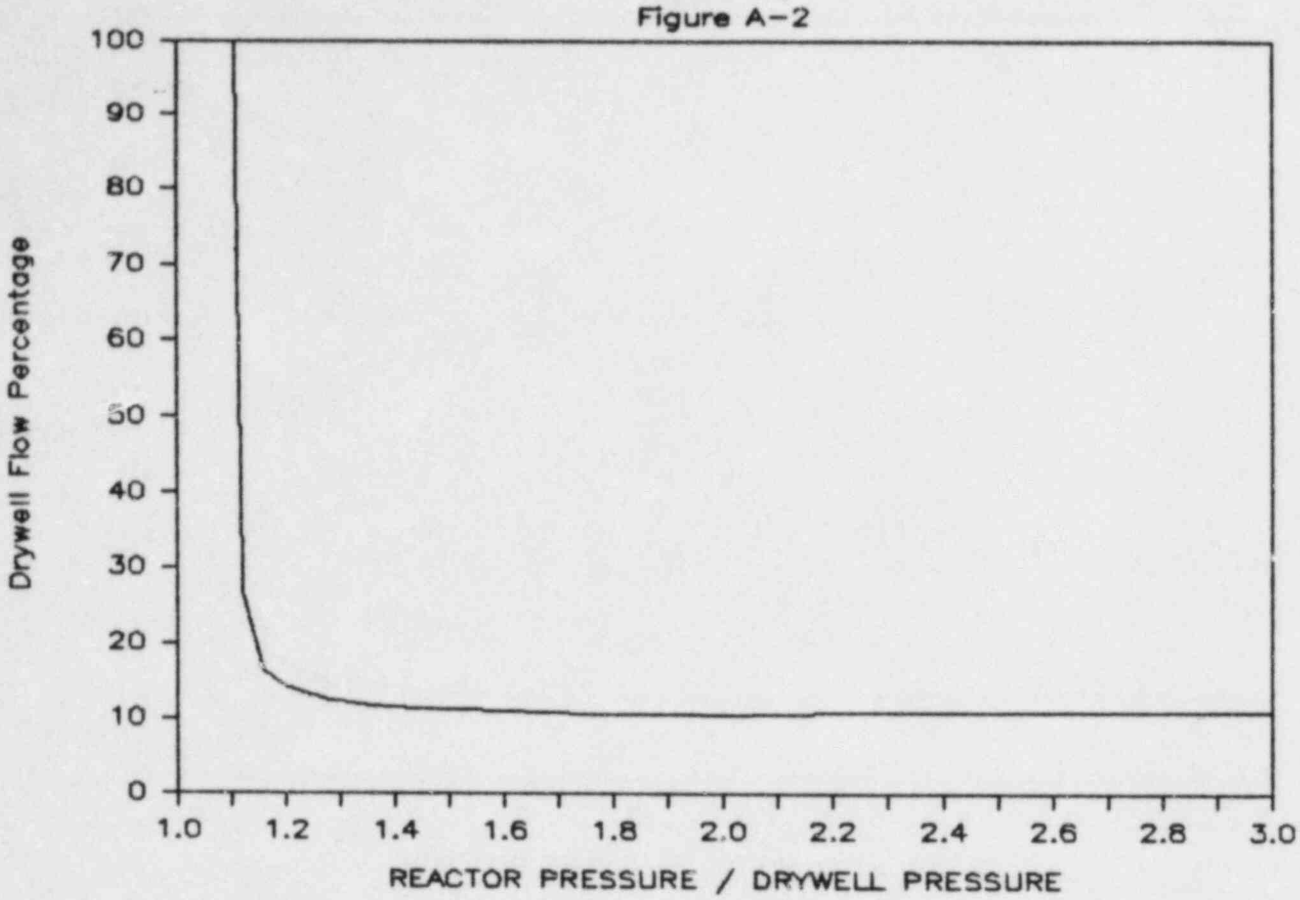
# Grand Gulf Flow Split Calculation

Figure A-1



# Grand Gulf Flow Split Calculation

Figure A-2



# SUMMARY OF TEST PROGRAM RESULTS FROM 1/20th SCALE TESTS AND 1/5th SCALE SINGLE SPARGER MOCK UP

The Hydrogen Control Owners Group (HCOG) has completed a series of hydrogen combustion tests in a 1/20th scale model of a Mark III containment. The preliminary results from tests were presented to the Nuclear Regulatory Commission during a meeting on June 29, 1983.

The tests were conducted to provide a visual record of global hydrogen combustion behavior in a full 360° model of a Mark III containment. The tests also provided data for estimating the thermal environment produced by steady diffusion flames including temperatures and radiant heat fluxes. A total of 41 tests were conducted which included assessing the effects of varying hydrogen release rates, blockages above the pool surface, hydrogen release points, number of operating SRV spargers, suppression pool temperature and heat loss through the containment shell.

The test results showed that steady diffusion flames will be established above the pool surface for hydrogen injection rates greater than .4 lbm/sec. This threshold was defined by initially establishing a stable diffusion flame, then decreasing hydrogen flow to a point at which steady diffusion flames could no longer be sustained. Some variation in this threshold for sustaining diffusion flames as a result of varying pool temperature was observed. For pool temperatures more nearly representative of post accident conditions, i.e., a temperature of 185°F, the threshold for sustaining diffusion flames was .5 lbm/sec.

The sequence of events observed in the 1/20th scale facility relative to hydrogen combustion are fairly consistent. For a nine sparger discharge case (8 ADS plus one SORV) after hydrogen injection commences, a weak upward flame propagation from the lowest igniters is followed almost immediately by a rapid downward propagation to the pool surface. Steady diffusion flames anchored at the surface of the suppression pool directly above the spargers are established almost immediately with the most intense burning occurring in the 312° chimney for the Grand Gulf configuration. (This is the location of the least restricted vertical flow path which for conservatism has simulated below it a sparger associated with an ADS SRV and an immediately adjacent sparger associated with a stuck open relief valve.) The combustion produces strong horizontal air flows above the pool with flow direction dependent upon annular flow blockage geometry. As the oxygen concentration in the facility is depleted, the flames weaken and grow taller. The flames move upward and anchor at the HCU floor grating for a very short period of time before the combustion is terminated by oxygen deprivation, due to prior gas venting from the facility.

The tests showed peak gas temperatures below the HCU floor generally ranged from 560°F to 700°F with one test showing a peak temperature of 836°F. The observed heat fluxes to the HCU floor from below totaled 4000 Btu/hr-$ft^2$ with a convective component of 3,200 Btu/hr-$ft^2$. The thermal environment including peak gas temperatures and total heat fluxes is most influenced by the hydrogen release rate and the number of release locations.

An additional test was conducted to provide greater confidence in the threshold hydrogen flow rate which produces sustained diffusion flames. This test was conducted in a 4' diameter cylindrical tank filled with water and open to the atmosphere. A single sparger was simulated in the tank 4.5 feet below the water level. A 3' high shroud enclosed a 270° section of the tank in order to prevent the wind from dissipating hydrogen rising from the sparger. This shroud had no discernible effect on the supply of oxygen to the pool surface. The igniter for this test was located approximately 6 inches above the pool surface. The complete facility represents roughly a 1/5th scale mock up of a single sparger.

The tests were performed in the same manner as the threshold tests conducted in the 1/20th scale facility. A steady diffusion flame was established by injecting a relatively high hydrogen flow rate. The flow rate was then decreased to a value which was insufficient to sustain a steady diffusion flame. The tests showed that for a full scale flow rate of .4 lbm/sec a steady diffusion flame could not be maintained.

The tests conducted with 1/5th scale single sparger mock up were performed with relatively cold water. No facilities were available to raise the water temperature to post accident temperatures of approximately 185°F. Consequently, the tests in the 1/20th scale at higher pool temperatures could not be repeated in the larger scale facility. However, based upon the close agreement between the 1/20th scale full containment model and the 1/5th scale single sparger model for cool water, the .5 lbm/sec threshold for hot pool releases is appropriate. This threshold will be investigated further during the planned 1/4th scale testing program.

The flame heights generated in the 1/5th scale single sparger mock up were measured and compared to the flame heights measured in the 1/20th scale facility. The flame heights observed in the 1/5th scale model were much lower than the flame heights which would be expected based on the 1/20th scale measurements. The 1/20th scale test data appears to overpredict flame heights by as much as a factor of 2. Consequently, it is anticipated that diffusion flames which will be studied in the 1/4th scale test facility will produce much less severe thermal enviroments than the 1/20th scale test data extrapolated to full scale.