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IDCOR

Program Report

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Technical Report 23.1

GRAND GULF NUCLEAR STATION

INTEGRATED CONTAINMENT ANALYSIS

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GRAND GULF NUCLEAR STATION

IDCOR Task 23.1

Integrated Containment Analysis

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TABLE OF CONTENTS

DRAFT

TABLE OF CONTENTS (Continued)

DRAFT

TABLE OF CONTENTS (Continued)

DRAFT

DRAFT

LIST OF FIGURES

DRAFT

LIST OF FIGURES (C tinued)

DRAFT

LIST OF FIGURES (Continued)

DRAFT

LIST OF FIGURES (Continued)

DRAFT

LIST OF TABLES

DRAFT

LIST OF TABLES (Continued)

DRAFT

1.0 INTRODUCTION

1.1 Statement of the Problem

The objective of this investigation was to calculate the response of the Grand Gulf Nuclear Station (GGNS) primary system and containment to postulated severe accident sequences which have been identified as potentially leading to core degradation and melting. These analyses include evaluations of the thermal-hydraulic response, the release of fission products from degraded fuel, and the transport of the released fission products within the containment. These calculations were performed on a best estimate basis phenomenologically and include assessments of the major uncertainties associated with state-of-the-art modeling. This study includes assessments of the results of a limited set of operator interventions in these sequences and an assessment of the influence of a specific mitigating feature associated with the Grand Gulf Nuclear Station design.

1.2 Relationship to Other Tasks

The primary system and containment response analyses of IDCOR Subtask 23.1 are dependent upon the primary system and containment thermal-hydraulic models developed in Subtasks 16.2 and 16.3 (Executive Analysis Program) and the fission product release and retention models developed in IDCOR Task 11 (Fission Product Transport). The accident sequences used for the analyses along with the operator interventions were developed by considering the dominant accident sequences identified in Subtask 3.2 (Assess Dominant Sequences) and the physical processes occurring during these accidents.

It should be noted that the analyses developed as part of IDCOR Subtasks 16.2 and 16.3 involve the detailed consideration of many different phenomena which are themselves considered in separate IDCOR subtasks. These include: hydrogen generation; distribution and combustion (Subtasks 12.1, 12.2 and 12.3); steam generation (Subtask 14.1); core heatup (Subtask 15.1); debris behavior (Subtask 15.2) and core-concrete interactions (Subtask 15.3).

DRAFT

Operator intervention sequences were developed as part of Subtask 23.1 and applied to the specific accident sequences in the Grand Gulf Nuclear Station design to determine those potential actions which could terminate the accident sequence and result in a safe stable state. These results were used in IDCOR Subtask 22.1 (Safe Stable States) which discusses both the inherent and intervention means of terminating the various core damage sequences considered for the Grand Gulf Nuclear Station design. The mitigative design feature sequence for GGNS was developed via a review of a list of mitigative and preventative design features identified in IDCOR Task 21 (Risk Reduction Potential).

The ultimate structural capability of the containments associated with the reference plants and other typical designs were assessed in IDCOR Subtask 10.1. These analyses define the containment failure pressure and failure mode in this analysis.

Calculations of the rate and amount of fission products released from the containment, for those sequences which result in containment failure, were supplied to IDCOR Subtask 18.1 (Atmospheric and Liquid Pathway Dose) to formulate assessments of the health consequences associated with these postulated accident scenarios. These health consequence analyses were then supplied to IDCOR Subtask 21.1 to evaluate the risk reduction potential for possible mitigating operator actions and containment mitigative design features.

Detailed considerations for each of the related subtasks can be found in the final reports submitted for the specific task. Individual issues are addressed in this report only as required to understand the specific behaviors obtained for the accident sequences considered.

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2.0 STRATEGY AND METHODOLOGY

The basic strategy of this subtask was to analyze accident sequences which have been previously identified as potential contributors to core melt frequency. These analyses consisted of plant thermal hydraulic response and fission product transport calculations for accident sequences which led to core degradation and melting. These analyses model performance of the ECCS systems and the containment engineered safety systems, such as the suppression pool, decay heat removal system, etc.

The MAAP code [2.1] was used to perform the primary system and containment thermal-hydraulic response analyses. This code considers the major physical processes associated with an accident progression, including hydrogen generation, steam formation, debris coolability, debris dispersal, core-concrete interactions, and hydrogen combustion. The FPRAT module for MAAP was adopted from [2.2] to evaluate the fission product release from the fuel. Natural and forced circulation within the primary system is modeled both before and after vessel failure and is integrated with the fission produce release model to determine the transport of vapors and aerosols throughout the primary system and containment. Fission product deposition processes modeled include vapor condensation, steam condensation and sedimentation.

For each of the four GGNS accident scenarios selected for analysis, thermal-hydraulic calculations were performed both with and without selected operator actions during the accident. The "base case" analyses, which assume only minimal operator response during the accident, establish a reference system response during each of the accident scenarios. The "operator action" analyses are branch calculations of the base cases. These operator intervention cases demonstrate the effect of selected operator actions on the progression of an accident, based on the time windows available to the operator to take such action. Additional uncertainty and sensitivity analyses have been performed on several key parameters associated with the accident response. These are reported in Ref. [2.4].

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2.1 References

2.1 "MAAP, Modular Accident Analysis Program User's Manual," Technical Report on IDCOR Tasks 16.2 and 16.3, May 1983.

2.2 "Analysis of In-Vessel Core Melt Progression," Technical Report on IDCOR Subtask 15.1B, September 1983.

2.3 Richard K. McCardell, "Severe Fuel Damage Test 1-1 Quick Look Report," EG&G Idaho, October 1983.

2.4 IDCOR Technical Report on Task 23.4, "Uncertainty and Sensitivity Analyses for the IDCOR Reference Plants," to be published.

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3.0 DESCRIPTION OF MODELS AND MAJOR ASSUMPTIONS

The Modular Accident Analysis Program (MAAP), Ref. [3.1] is used to model the Grand Gulf Nuclear Station (GGNS) response to postulated severe accidents. This code includes containment response, fission product release, and fission product transport. In addition, both the thermal hydraulic response and the fission product behavior are modeled for the reactor building which surrounds the primary containment.

3.1 Plant Specific Information

The Grand Gulf Nuclear Station (GGNS) is a two unit boiling water reactor located in Claiborne County, Mississippi, on the east side of the Mississippi River approximately 25 miles south of Vicksburg and 37 miles north-northeast of Natchez, Mississippi. The two units are nearly identical; both will be operated by Mississippi Power & Light Company (MP&L). Unit 1 is scheduled to go into commercial operation in early 1985; Unit 2 is scheduled to do so several years later. Each unit is designed with a core thermal output of 3833 MWth, a gross electrical power output of 1306 MWe, and a net electrical output of 1250 MWe. Each unit is powered by a BWR-6 water reactor, designed and supplied by General Electric Company. Each reactor is housed in a steel-lined reinforced concrete Mark III containment building.

3.1.1 Nuclear System

The primary system consists of the equipment and instrumentation necessary to produce, contain, and control the steam power required by the turbine-generator. Principal components of the system are the reactor pressure vessel (RPV) and internals, reactor water recirculation system, and the main steam system. Other important systems include the condensate and main feedwater systems which close the primary system flow loop by condensing the steam and water exhausted by the turbines and pumping this condensate back into the RPV. The reactor vessel houses the reactor core, contains the heat, produces steam within its boundaries, and serves as one of the fission product barriers during normal operation and in the event of fuel failure.

DRAFT

The core is composed of 800 fuel assemblies, each containing 62 fuel rods and two hollow water rods. These fuel rods are sealed Zircaloy-2 tubes, which are loaded with UO_2 fuel pellets, with the Zircaloy-2 cladding providing both structural support and a fission product barrier between the fuel and the primary system water. The remaining reactor pressure vessel internal components support and align the fuel and provide the water circulation flow paths to distribute coolant to the fuel. Upper vessel internals also furnish moisture removal for the steam generated within the core, to minimize the moisture content of the exiting steam. The major internal components consist of the core, the shroud top grid, core plate, steam generator and dryer, jet pumps, control rods, and control rod drives.

The reactor water recirculation system provides a forced continuous internal circulation of coolant water through the core. Four main steam lines direct steam to the balance of the plant. During an abnormal event occurring during power operation, main steam isolation valves (MSIVs) on each of these lines provide isolation of the reactor vessel from the balance of the plant. If their closure is required, a set of 20 safety/relief valves (SRVs) provide reactor vessel overpressure protection, with their discharge being directed to the suppression pool.

The majority of the primary system data used in this analysis came from the Grand Gulf FSAR [3.2]. This information includes initial conditions, pressures, temperatures, flow rates, enthalpies, masses, system pressure setpoints, control logic, and other parameters. A plant parameter file for MAAP was prepared based on these data; it appears in Appendix A.1.

3.1.2 Containment

The reactor vessel is housed in the containment building. This structure is designed to condense the steam (pressure suppression) and contain the fission products which may be released as a result of a Loss of Coolant Accident (LOCA). The Mark III containment is a steel-lined reinforced concrete structure, with a cylindrical shape, topped with a hemispherical dome. The containment foundation is a thick, circular reinforced concrete slab. Major elements of this pressure-suppression design are an inner volume and an

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outer volume, separated by a large heat capacity suppression pool. The inner region, the drywell, is a cylindrical volume containing the reactor pressure vessel, which is supported by a hollow concrete cylinder called the pedestal. The drywell and outer containment volumes communicate via horizontal vent openings located below the suppression pool surface. A water seal between the inner and outer volumes is accomplished by the drywell weir wall. The pool which provides for steam suppression during postulated LOCA events. The outer containment volume consists of the annular space above the suppression pool and the dome. The upper containment pool, located in the outer containment volume, provides a post-LOCA source of makeup water to the suppression pool. Containment sprays, also located in the outer compartment, provide an additional means of rapidly removing possible post-accident steam and/or fission products from the outer containment atmosphere. In addition to these features, hydrogen igniters are located in both drywell and outer containment volumes to control hydrogen accumulations following postulated severe accidents.

The GGNS BWR-6/Mark III design, like that of other nuclear plants, is based on a defense-in-depth principle. Thus, if an abnormal event were to occur, backups to the normal systems are designed to maintain the integrity of the fuel cladding, the reactor pressure vessel, and the containment barriers. These backup systems perform two general functions: core cooling and containment pressure control. Those systems which perform the first function include the reactor core isolation cooling (RCIC) system, the high pressure and low pressure emergency core cooling systems (ECCS), the automatic depressurization system (ADS), and the standby liquid control (SLC) system. The containment pressure control function is accomplished via the suppression pool makeup system, the drywell purge system, the post-LOCA vacuum breakers, the suppression pool cooling and containment spray modes of the residual heat removal (RHR) system, and the hydrogen ignition system.

MAAP input data, including initial conditions, heat transfer coefficients, exposed surface areas, and flow areas between volumes are based on information from the Grand Gulf FSAR [3.2], and architect/engineer drawings. These data appear in the MAAP parameter file listed in Appendix A.1.

3.2 Modular Accident Analysis Program (MAAP)

Within the IDCOR Program, the phenomenological models developed in Tasks 11, 12, 14 and 15 have been incorporated into an integrated analysis package in Subtask 16.3, while Subtask 16.2 provides a computer code (MAAP) to analyze the major degraded core accident scenarios for both Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). The MAAP code is designed to provide realistic assessments for severe core damage accident sequences using first principle models for the major phenomena that govern the accident progression, the release of fission products from the fuel matrix, the transport of these fission products and their deposition within the primary system and containment. The following sections describe the primary system and containment nodalization and include a description of the safety systems modeled in MAAP.

3.2.1 MAAP Nodalization

The BWR primary system nodes are illustrated in Figure 3.1 and include the lower plenum, downcomer, core, and upper plenum. Also indicated are the flow entry locations for CRD flow, feedwater, HPCS, RCIC, LPCI and LPCS as well as the standby liquid control system (SLCS). The SLCS is only modeled as an additional water source since MAAP does not have a neutronics model. Individual mass and energy equations are written for each of these nodes using the water addition locations and the appropriate connecting flow paths. The primary system model also represents the main steam isolation valves and the main steam safety and relief valves. The latter exhaust into the suppression pool.

Modeling of the primary system is used to determine if a given sequence (1) leads to core uncovery, (2) results in core damage, (3) yields Zircaloy clad oxidation and hydrogen formation, (4) leads to core melt and vessel failure, or (5) can be recovered before vessel failure. The code predicts the times of these occurrences. The transient response to the spectrum of accident scenarios considered requires the specification of pump curves, valve set points, system logic, etc. With the specification of the accident sequence, the primary system model determines the vessel water

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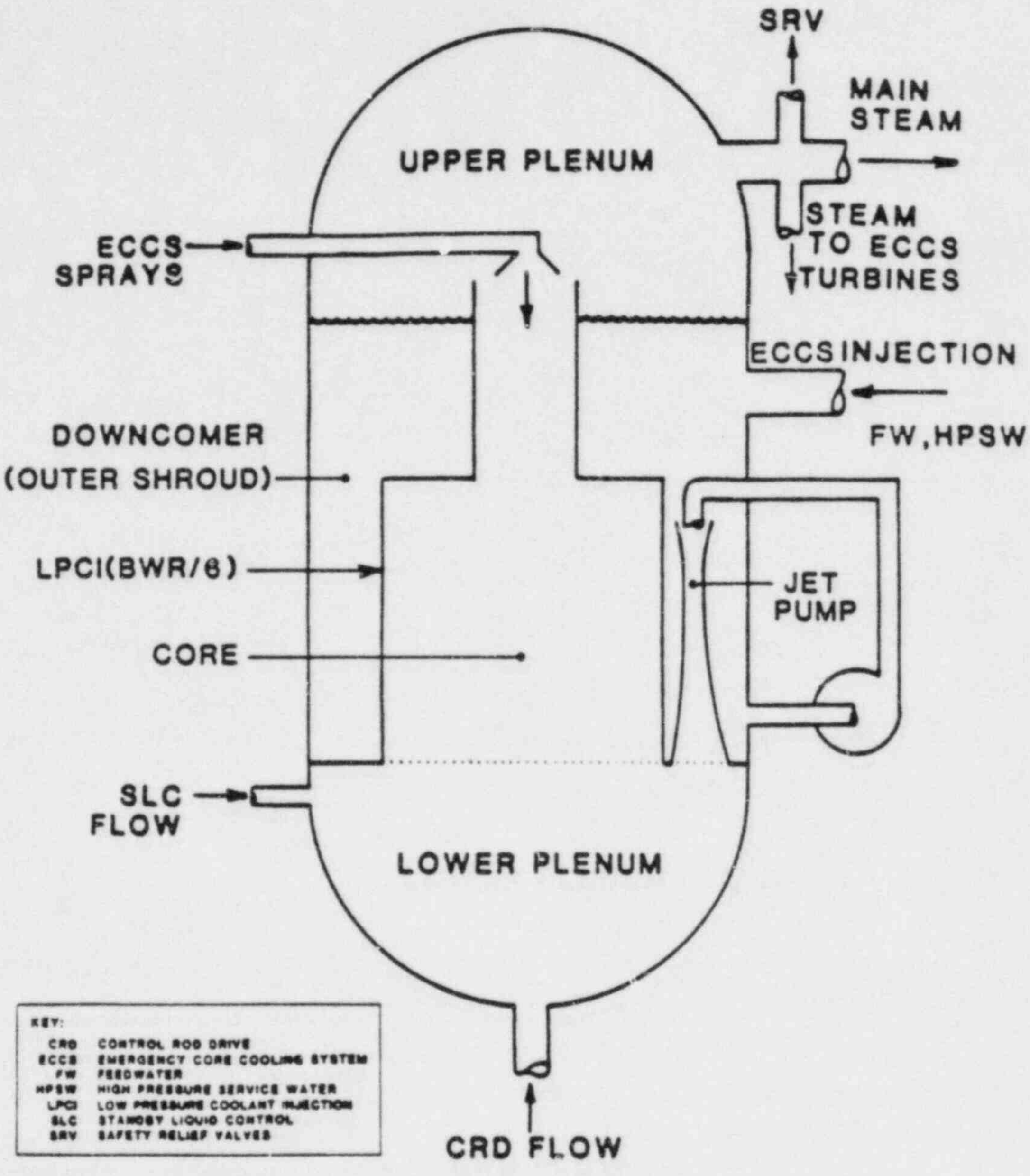

Fig. 3.1 BWR primary system.

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inventory, including the boiled-up level in the core, to evaluate the potential for core uncovery. If the collapsed water level decreases below the top of the core, the HEATUP subroutine calculates the temperature increases for the fuel and cladding. Steam cooling and the oxidation of the Zircaloy clad and channels are determined by the appropriate rate laws and oxygen starvation. The model accounts for the cooling effect of CRD flow. If available, this flow can limit core damage for long-term heat removal failure events.

The Mark III (Grand Gulf) containment nodalization scheme, as shown in Fig. 3.2, separates the containment into five compartments: the pedestal, the drywell, wetwell, Compartment A (annulus above the wetwell), and Compartment B (above the operating deck) regions. MAAP evaluates the behavior of the various compartments during the entire progression of the accident sequence by calculating the mass and energy flow rates between these compartments.

Individual compartment (region) pressures and gas temperatures are derived from the mass and energy balances. MAAP models the transport of all material throughout the containment due to drainage, vaporization, condensation and mass addition to assess the potential for cooling core debris. Separate water and corium temperatures are calculated for each containment compartment.

3.2.2 Grand Gulf Systems Modeled in MAAP

In general, MAAP characterizes the response of the primary system, the containment, and many of the balance of plant systems to user specified event sequences. Figure 3.3 illustrates the plant systems modeled in the code including the various water sources available and the valve line-ups which would allow this water to be injected into either the primary system and/or containment during a postulated sequence. Particular systems of importance include, the control rod drive (CRD) flow from the condensate storage tank, main steam lines, MSIVs, turbine bypass, feedwater, reactor core isolation cooling (RCIC), high pressure core spray (HPCS), low pressure coolant injection (LPCI) and other RHR system modes, low pressure core spray (LPCS), standby liquid control system (SLCS), and high pressure service water (HPSW). In addition to these plant systems, MAAP nodalizes both the primary system and

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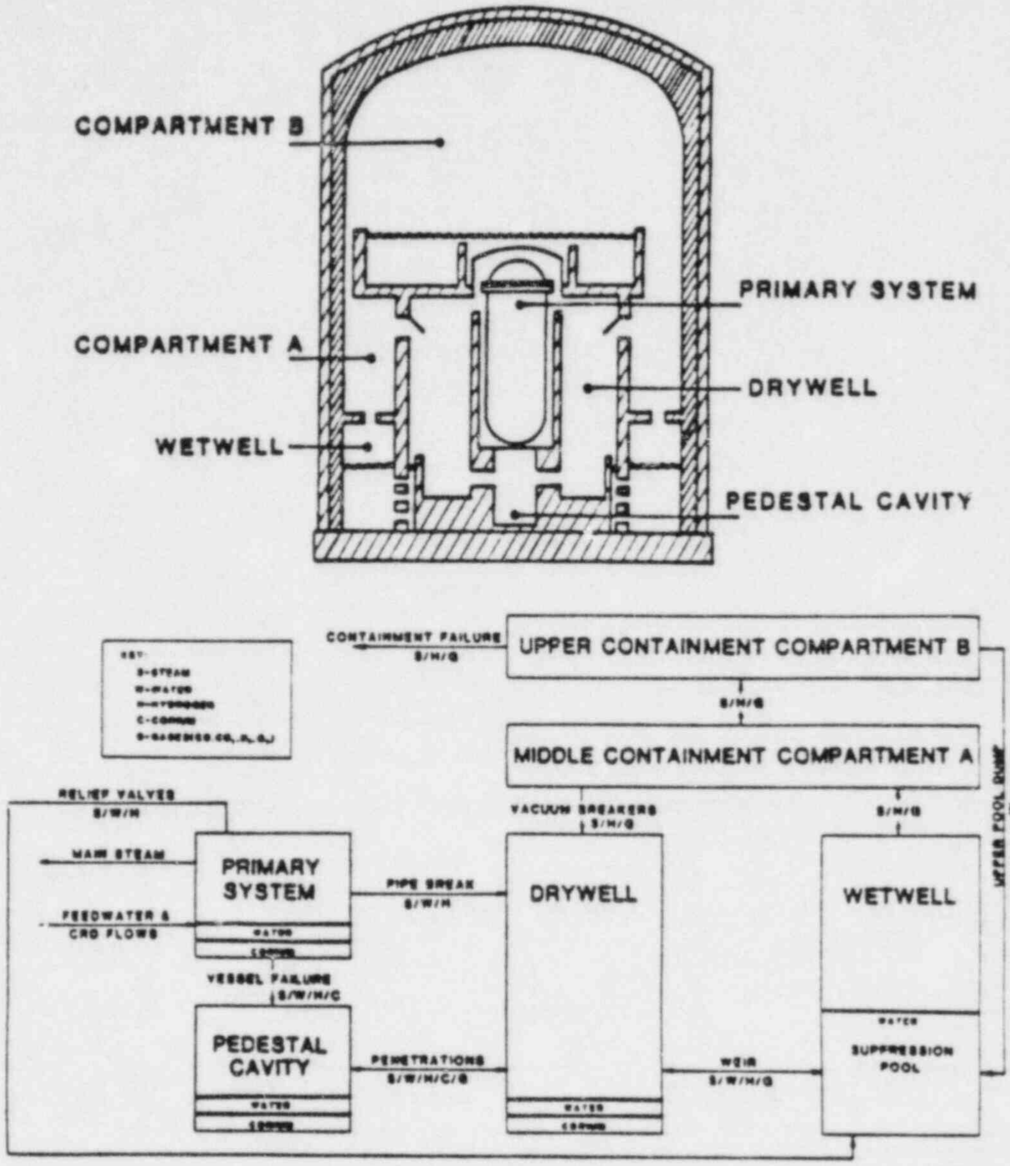

Fig. 3.2 Schematic representation of Grand Gulf Mark III containment and MAAP nodalization.

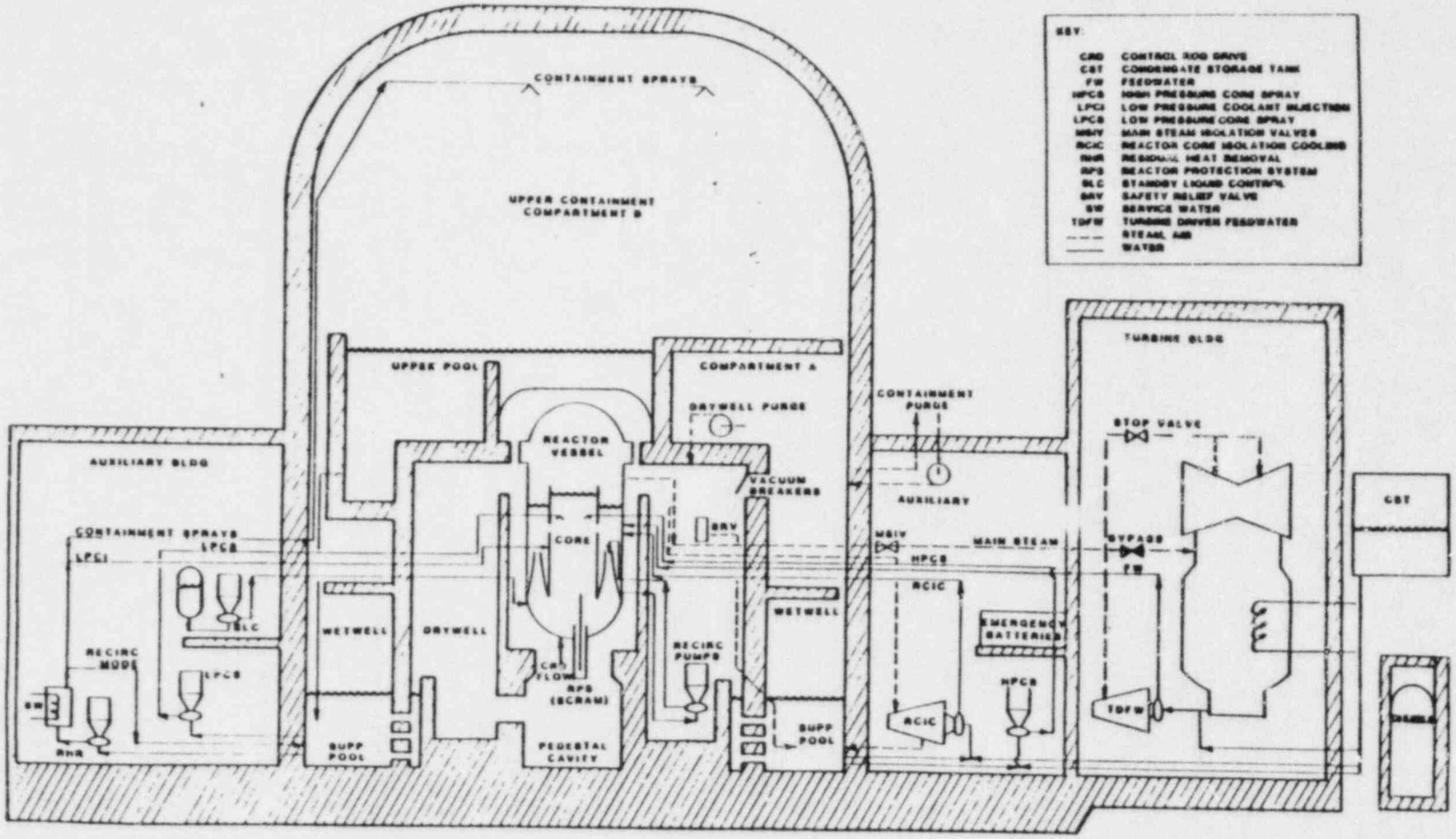

Fig. 3.3 Schematic representation of Grand Gulf safety and other systems.

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containment to model their response to postulated core damage and recovery scenarios.

3.2.3 Fission Product Release from Fuel

The FPRAT module in MAAP, as adapted from Ref. [3.3] was used to calculate the release rates of fission products from the fuel matrix. These rates are dependent upon the fuel temperature history during heatup and upon characteristics of the atmosphere within the vessel which effect saturation of the chemical species as discussed in IDCOR Task 11.1 [3.4]. Fuel temperature histories for the thirty regions in the core are tracked to determine the release characteristics for the fission products and inert materials. The initial inventories of the various fission products were obtained from Ref. [3.5] and are given in Table 3.1.

The gas flow through each node is assumed to be saturated with the vapor of each constituent. If the flow cools as it is transported to higher nodes, the gas cools and creates aerosols of each species to remain saturated. This flow provides the aerosol and vapor source for the upper plenum. For the regions in which blockage has occurred, it is assumed that sufficient flow exists to remove the volatile fission products as saturated vapor. Once this flow is determined, the removal of the remaining less volatile species is evaluated based upon saturation of this calculated flow. The required FPRAT input for MAAP is given in the parameter file in Appendix A.1.

The calculations consider evaporation and condensation characteristics of chemical species. Several key assumptions consistent with the recommendations of IDCOR Subtask 11.1 were made regarding the physical and chemical forms of released fission products. These are:

1. Cesium and iodine combine to form CsI upon entry to the fission product release pathway. The excess cesium forms CsOH. Both chemical species exhibit similar physical behavior, hence the source rate for the Cs,I fission product group is assumed to be the sum of the Cs and I release rates. As stated above, it is assumed to be liberated in vapor form.

DRAFT

Table 3.1

INITIAL INVENTORIES OF FISSION PRODUCTS AND STRUCTURAL MATERIALS RELEASED FROM THE FUEL

Fission Products	Initial Inventory (kg)
Kr	27.3
Xe	412
Cs	220
I	17.7
Te	37.1
Sr	66.7
Ru	183
Mo	252
Sn	1190
Mn	268

DRAFT

2. Tellurium is assumed to be released as vaporized TeO_2.

3. Inert aerosol generation rate is the combined release rates for volatile structure material (Mn and Sn).

4. Strontium and ruthenium represent their respective nonvolatile fission product groups as defined in WASH-1400. They are also calculated to be released as vapor which quickly forms aerosols when they exit the core.

5. Release of the volatile fission products (Cs, I, Te) and the noble gases (Xe and Kr) is allowed to continue until complete, even if the vessel has failed first.

3.2.4 Description of the Natural Circulation Model

Substantial quantities of fission products are released during core degradation, but before vessel failure. Gas flow through the primary system determines the aerosol transport and deposition throughout the reactor vessel. Following vessel failure, most fission products remain within the primary system and subsequently heat the adjacent structures. As the structure and gas temperatures increases, density differences within the primary system would result in natural circulation flows that could distribute both heat and mass throughout the primary system.

The natural circulation model determines flows within the primary system, and includes descriptions for fission product heat generation, material vaporization, condensation and deposition. Also, the nodalization allows for a representation of the structural heatup in each node as well as the heat losses from these nodes to the containment environment. The circulation for the BWR system after vessel failure is graphically represented in Fig. 3.4. As illustrated, the throat area for the jet pumps controls the circulation rate and the containment pressurization/depressurization influences the flow from the primary system.

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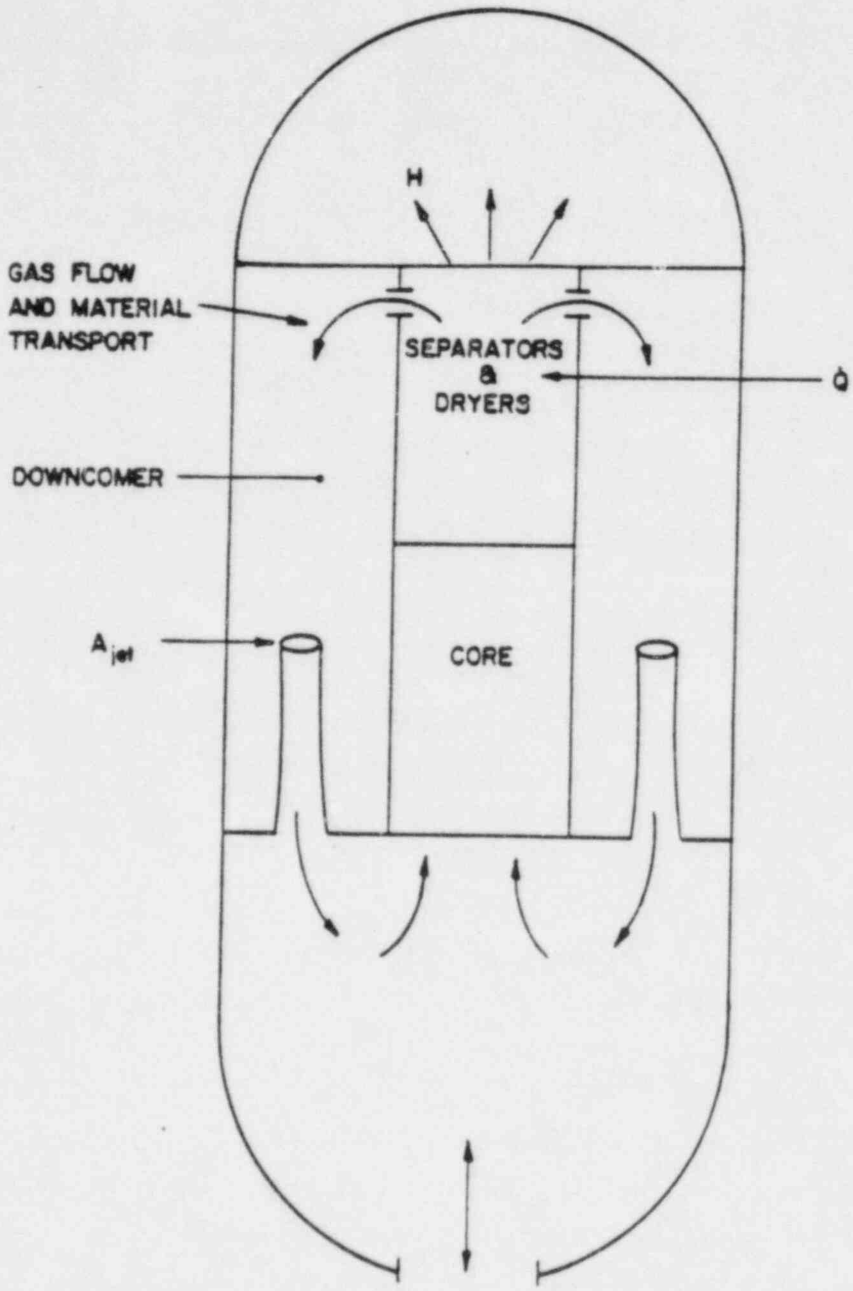

Fig. 3.4 BWR natural circulation model.

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Since natural circulation flows are driven by the gas density differences between various regions, and since the volatile fission products are dense vapors, calculation of the gaseous flows within the primary system must account for the gas mixture properties in the various nodes. In addition, with the reflective insulation used on the Grand Gulf reactor vessel, the heat losses from the vessel must also include the magnitude of heat losses as a function of the primary system temperature and the potential for oxidation of the stainless steel layers in the reflective insulation.

These analyses have been coupled with models for aerosol deposition and heatup to evaluate the primary system flows after reactor vessel failure. Such assessments provide the rate and amount of material lost from the primary system as a result of the subsequent heatup of primary system structures. In this analysis, the difference between the primary system and containment pressurization determines the flows between these two systems which govern the release of fission products to the containment environment.

3.2.5 Aerosol Deposition

IDCOR Task 11.3, Ref. [3.6], applied state-of-the-art fission product behavior models to produce the RETAIN code, which describes the aerosol agglomeration and removal processes based upon an assumed log-normal distribution [3.6]. Both vapor and aerosol forms of fission products are considered. MAAP represents the aerosol removal rate due to settling as a function of the aerosol cloud density [3.5]. This is consistent with the general behavior predicted by detailed descriptions, such as RETAIN, and more importantly, is in good agreement with the results of large scale experiments. MAAP models physical mechanisms for vapor condensation on structures and aerosol retention due to steam condensation in addition to gravitational settling. These removal processes substantially reduce the magnitude of the release to the environment.

The primary system and containment nodalization for fission product transport are the same as those used for the thermal hydraulic calculations. The specific transport paths were earlier illustrated in Fig. 3.2 for the primary system and containment.

DRAFT

The key assumptions in the aerosol modeling are:

1. Cesium and iodine are assumed to be released as CsI with excess cesium as CsOH.

2. The decontamination factor associated with the wetwell suppression pool is estimated to be 1000 for releases through the spargers and 600 for releases through the horizontal vents [3.8].

3. Prior to vessel failure any fission products that may enter the drywell (such as from a LOCA pathway from the primary system) are available to enter Compartment A via the slight design-basis drywell leakage. These pathways are assumed to be closed off following vessel failure due to plugging by aerosols [3.9].

4. Fission products reaching the SRV discharge lines were treated as having reached the suppression pool.

5. Hygroscopic aerosols, such as cesium hydroxide, are assumed to accumulate an equilibrium concentration of water as determined by the steam partial pressure and temperature.

6. Release of volatile fission products (Cs, I, Te) and the noble gases (Xe and Kr) is allowed to continue until complete, even if the vessel has already failed.

3.2.6 Fission Product and Aerosol Release from Core-Concrete Attack

The release of aerosols due to core-concrete attack was determined using a model based on the concrete ablation rates from MAAP. The mass of low volatility fission products and inert aerosols released from core debris is based upon a vapor stripping model assuming the melt constituents follow Raoult's law. This calculation is dependent upon the amount of gas sparging through the core debris, the molar concentration of fission products in the

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core debris, the vapor pressure of the chemical species of interest, and the temperature of the core debris.

The key assumptions are:

1. The masses of CO_2 and water vapor released per cubic meter ablated for the limestone concrete used at Grand Gulf are 572 kg and 130 kg respectively.

2. Stripping only occurs when the corium is calculated to be molten.

3. The gases released by the downward attack pass through the molten pool and cause stripping. Gases generated by sidewall attack are assumed to bypass the pool.

4. The predominant form of Sr is SrO, of Ru is elemental Ru, and of La is La_2O_3.

5. Inert aerosols of CaO may be generated during core-concrete attack. This chemical form is used as a surrogate for the various concrete melt constituents that could be added to the corium pool.

6. Deposition of fission products in the SRV discharge lines was neglected.

7. Concrete aerosol generation was not incorporated into the overall fission product removal calculations but was used to make an assessment of the extent of plugging of the drywell to compartment A pathway.

3.3 References

3.1 "MAAP, Modular Accident Analysis Program," Technical Report on IDCOR Subtasks 16.2 and 16.3, 1983.

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3.2 Final Safety Analysis Report, Grand Gulf Nuclear Station, Mississippi Power and Light Company, 1979.

3.3 "Analysis of In-Vessel Core Melt Progression," Technical Report on IDCOR Subtask 15.1B, September 1983.

3.4 "Estimation of Fission Product and Core-Material Source Characteristics," Technical Report on IDCOR Subtasks 11.1, 11.4, and 11.5, 1983.

3.5 Radionuclide Release Under Specific LWR Accident Conditions -- Volume III BWR, Mark III Design, Battelle Columbus Laboratories, 1984.

3.6 "Fission Product Transport in Degraded Core Accidents," Technical Report on IDCOR Subtask 11.3, December 1983.

3.7 "Aerosol Deposition Model," FAI report to be published.

3.8 Personal Communication, K. Holtzclaw (GE) to R. Henry (FAI), March 1984.

3.9 H. A. Morewitz, "Leakage of Aerosols from Containment Buildings," Health Physics, Vol. 42, No. 2, pp. 195-207, 1982.

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4.0 PLANT RESPONSE TO SEVERE ACCIDENTS

This section provides the results of plant thermal-hydraulic analyses of four base case accident sequences, using the MAAP code. The accident scenarios are specific cutsets of each accident sequence and, as such, are not necessarily representative of all cutsets of these sequences. The accident scenarios are defined below, followed by descriptions of the reactor coolant system response and the containment response. The time dependence of the most significant MAAP-generated thermal-hydraulic parameters associated with each scenario are presented in Appendix B. The plant parameters utilized to characterize Grand Gulf in these analyses are listed in Appendix A.

The base sequences are:

1. T_1QUV - Transient with failure of injection.

2. AE - A large LOCA with failure of injection.

3. T_{23}QW - Transient followed by loss of containment heat removal.

4. T_{23}C - Transient followed by failure of the reactor to scram and standby liquid control (without operator action to reduce power level).

The T_1QUV was analyzed both with and without manual activation of the ADS in order to determine if this action would play a significant role in the overall containment response and fission product release.

The sequences analyzed in this section are low probability core damage events. The sequences exclude all, or nearly all, operator actions that could prevent or significantly delay core melt or that could mitigate its consequences. Operator actions which would prevent the accident are considered in the determination of the sequence probabilities. Those which would mitigate the consequences are not considered. This approach was taken to produce results which bound or are at the high end of the range of possible consequences for the four selected sequences. Generally, only minimal operator actions to control selected plant systems are assumed for these events. For example, it is assumed that the operators regulate low pressure injection to maintain water level at the high level trip.

As a result of the minimal operator response models employed in this analysis, the results presented here do not represent what would be realistically expected to occur for the specified equipment failures and are extremely improbable. The more probable expected plant response to the specified equipment failures is evaluated in Section 5. This later section includes in the sequence definition some examples of actions which the operator would be expected to take in accordance with the Emergency Procedure Guidelines. As a result of these actions the operator is able to terminate the event prior to core melt or significantly mitigate its consequences. Section 5 considers only some examples of the many actions available to the operator to prevent or mitigate the accident.

A major objective of excluding mitigating operator actions in this analysis and allowing the events to progress unchecked was to provide the added perspective of defining the time windows available for operator intervention. The results clearly demonstrate that the operator has an extensive time period to implement primary or alternative actions that will successfully terminate or mitigate postulated severe accidents.

The following subsections discuss plant response for each severe accident sequence analyzed. In these analyses the containment ultimate pressure capacity is based on the evaluation contained in the IDCOR Task 10.1 report [4.1], Containment Structure Capability of Light Water Nuclear Power Plants. The ultimate pressure capability was calculated to be 71.3 psia with the defined failure condition (twice the elastic strain) occurring at the "transition" between the cylindrical and spherical parts of the containment. (It should be noted that a detailed assessment of penetration behavior under high strain conditions was not part of the analysis.)

A containment break size of 0.1 ft^2 (1.5 ft^2 for TC) is assumed because it permits depressurization of containment enabling airborne fission products to be transported out the break. This assumption is consistent with

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the concept of yield leading to rupture resulting in diminishing yield as the containment depressurizes.

4.1 Plant Response to the T_1QUV Accident

4.1.1 Sequence Description

The T_1QUV accident is assumed to occur during full-power operation. It is initiated by a loss of off-site power event (T_1). During the accident, all systems not automatically transferred to the emergency busses are assumed to be unavailable. Thus, both the main feedwater and main condenser systems are assumed to be unavailable (Event Q) for the entire accident. The accident also specifies that neither the high-pressure nor the low-pressure emergency core cooling systems (ECCS) are available at any time during the accident (Events U and V, respectively). The faults in these makeup systems are taken to be such that the systems are unavailable in any of their modes of operation. In addition, the control rod drive (CRD) flow to the reactor pressure vessel (RPV) is modeled as being lost due to the accident initiator. Thus, for this event, no water makeup to the RPV occurs; and, neither primary system nor containment heat removal is assumed available. All other plant systems, including emergency diesels, are modeled to be available. No credit is taken for any operator action other than to energize the containment igniter system at the accident initiation and to manually depressurize the vessel when the water level drops to the RPV Level 2 LOCA setpoint. The T_1QUV base case accident chronology is provided on Table 4.1.

4.1.2 Primary System and Containment Response

The loss of off-site power, the loss of feedwater, the turbine stop valve (TSV) closures, and the turbine bypass valve (TBV) closures are modeled to occur simultaneously. Loss of off-site power and the TSV closures actuate a reactor scram which is modeled to bring the reactor subcritical by 7.8 sec. The core power remains at decay heat levels for the remainder of the event. The TSV and TSB closures cause a RPV pressure excursion which is relieved by the safety/relief valves (SRV). Steam released from the RPV through the SRVs is routed to the suppression pool (SP), where it is quenched. By 95 sec,

DRAFT

Table 4.1

GRAND GULF NUCLEAR STATION

T_1QUV - BASE CASE

ACCIDENT CHRONOLOGY

Time	Event
0.0 sec	Initiating Event: Loss of off-site power; Loss of main feedwater; TSV/TBV closures
7.8 sec	Reactor scram completed
95 sec	RPV Level 2 LOCA setpoint reached
26.0 min	RPV Level 1 LOCA setpoint reached; Vessel depressurization manually initiated
26.5 min	DW purge system actuates
28.0 min	Core begins to uncover
57.0 min	SPMU actuates
2.0 hr	Fuel melting begins
2.35 hr	High DW pressure LOCA setpoint reached
2.35 hr	Core plate failure followed by vessel failure
47.0 hr	Containment failure

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sufficient RPV water inventory has been lost through the cycling SRVs to lower the RPV water level to the RPV Level 2 LOCA setpoint. At the Level 2 setpoint, signals are automatically generated to trip off the recirculation pumps and to actuate the high pressure core spray (HPCS) and reactor core isolation cooling (RCIC) systems. Since both HPCS and RCIC are unavailable, the RPV water level continues to drop, reaching the RPV Level 1 LOCA setpoint at 26 min. At this point, it is assumed that manual depressurization of the vessel is initiated. In addition, permissive signals are automatically generated for the drywell (DW) purge, the suppression pool makeup (SPMU), the low pressure core spray (LPCS), and low pressure coolant injection (LPCI) systems. The DW purge system is modeled to actuate after a programmed 30 sec delay. Then, the DW purge compressors pressurize the DW atmosphere to the 1.89 psig High DW Pressure LOCA setpoint by 2.35 hours into the event. The SPMU system actuates the upper containment pool dump following a programmed 30 min delay. Since neither LPCS nor the LPCI are available, the RPV level continues to fall, and the core begins to uncover at 28 min.

Temperatures in the uncovered fuel regions begin to rise, and begin to reach 2000°F in about .5 hour after core uncovery. The cladding oxidation rate increases rapidly above the 2000°F fuel temperature point. Oxidation of the Zircaloy cladding increases the fuel heatup rate and thus tends to promote further cladding oxidation. Cladding oxidation within a channel is limited, however, by refreezing of molten cladding in lower, cooler portions of the channel. The steam trying to enter the channel is diverted around the blockage, thus preventing further oxidation and hydrogen formation within the channel (see Ref. [4.2]). Hydrogen generated by the Zircaloy-steam reaction in the core is released to the wetwell via the cycling SRVs. The amount released in the vessel is insufficient to cause burning. The maximum release rate being approximately 0.05 lb/sec.

Fuel melting is predicted to begin at 2.0 hr. Molten fuel is modeled to relocate to the core plate. By 2.35 hr, sufficient molten core material is accumulated on the core plate (20% of total) to cause it to fail. The core debris then flows to the bottom of the RPV, initiates thermal attack of the vessel wall and fails the vessel at a welded penetration. Following vessel failure, the molten core debris is discharged onto the pedestal floor.

Due to the depressurized state at the time of vessel failure, no core debris is dispersed from the pedestal onto the drywell floor. The discharge of molten core debris from the vessel is followed by the lower plenum water. Some of this water spills from the pedestal to the drywell. After vessel failure, about 50,000 lb of water remains in the lower downcomer region of the vessel.

Following vessel failure, the pedestal and drywell volumes are filled with steam; and, the air in these compartments is exhausted through the SP vents into the outer containment compartments. The pressures and temperatures in the drywell and in the outer containment are shown on Figs. 4.1 through 4.4. Drywell leakage flow paths bypassing the suppression pool are modeled to plug with aerosols. These aerosols are released from the vessel when it fails and from the core-concrete interaction in the pedestal. All flow exiting the drywell to the outer containment is afterward forced to pass through the suppression pool. Within ten minutes after vessel failure, the core debris bed in the pedestal is cooled to below concrete ablation temperatures by the lower plenum water; an ablation depth of 0.3 ft is predicted to this point in the accident sequence. The core debris temperature and concrete penetration depth in the pedestal are provided on Figs. 4.5 and 4.6. The debris remains quenched until its blanket of water is boiled away, which occurs at 4.0 hr into the event. Corium within the pedestal re-heats, renewing its attack on the pedestal concrete floor and walls at about five hours.

The thermal decomposition of the pedestal concrete floor and walls causes significant ablation (see Fig. 4.6), and produces large volumes of carbon dioxide and steam. As these two gases pass through the partially molten corium debris bed in the pedestal, they oxidize the zirconium in the bed to produce hydrogen gas and elemental carbon. The hydrogen production resulting from the core-concrete interaction in the pedestal raises the hydrogen concentration to ignitable levels; and within minutes after vessel failure, the igniters start the hydrogen burning. The igniters, which are powered by the emergency bus, provide for an almost continuous controlled burn-off of all combustible gases being evolved during the accident. The burnoff prevents the accumulation of high concentrations of combustible gases. By about 13 hr, 100% of the zirconium has been oxidized. About 1800 lb of

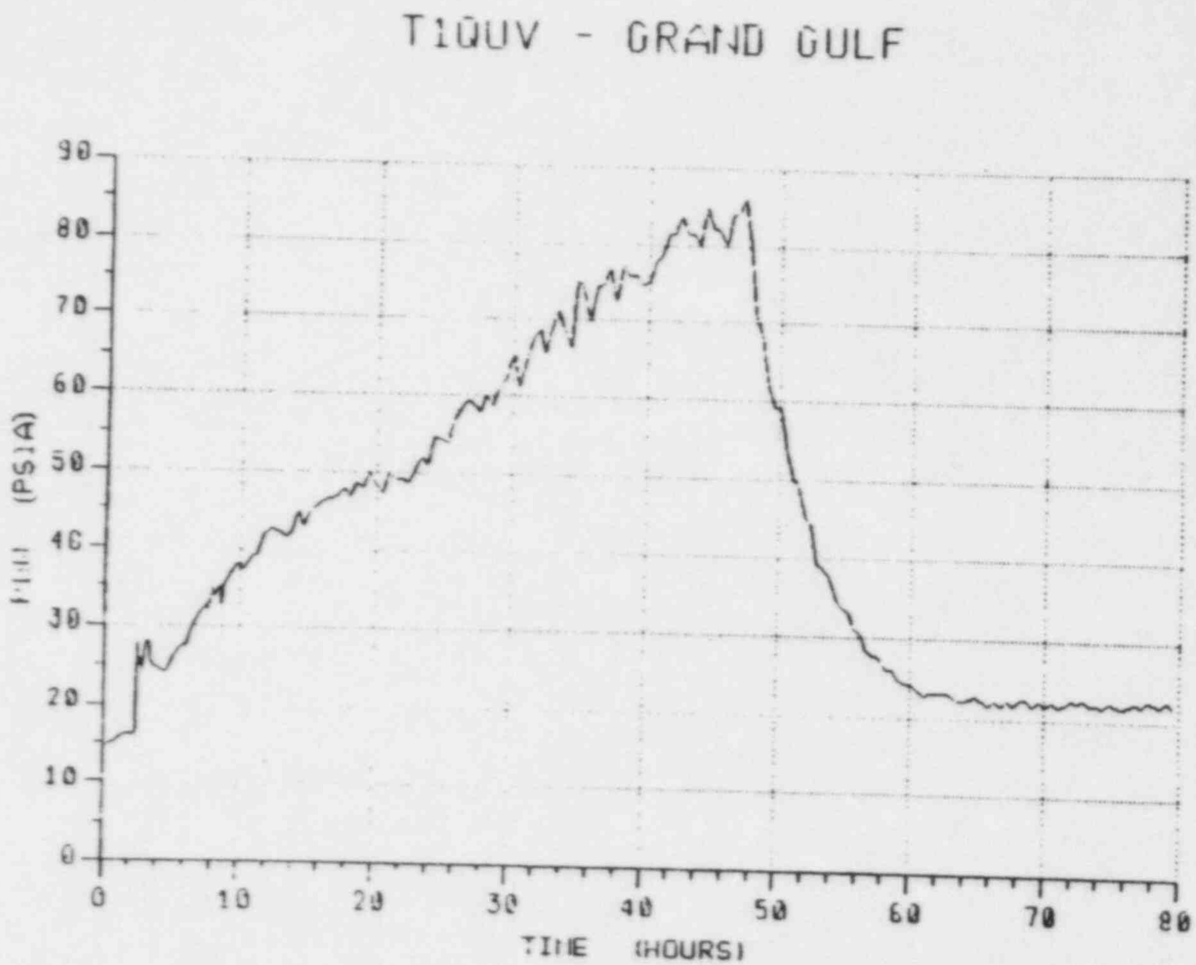

Fig. 4.1 Pressure in the drywell.

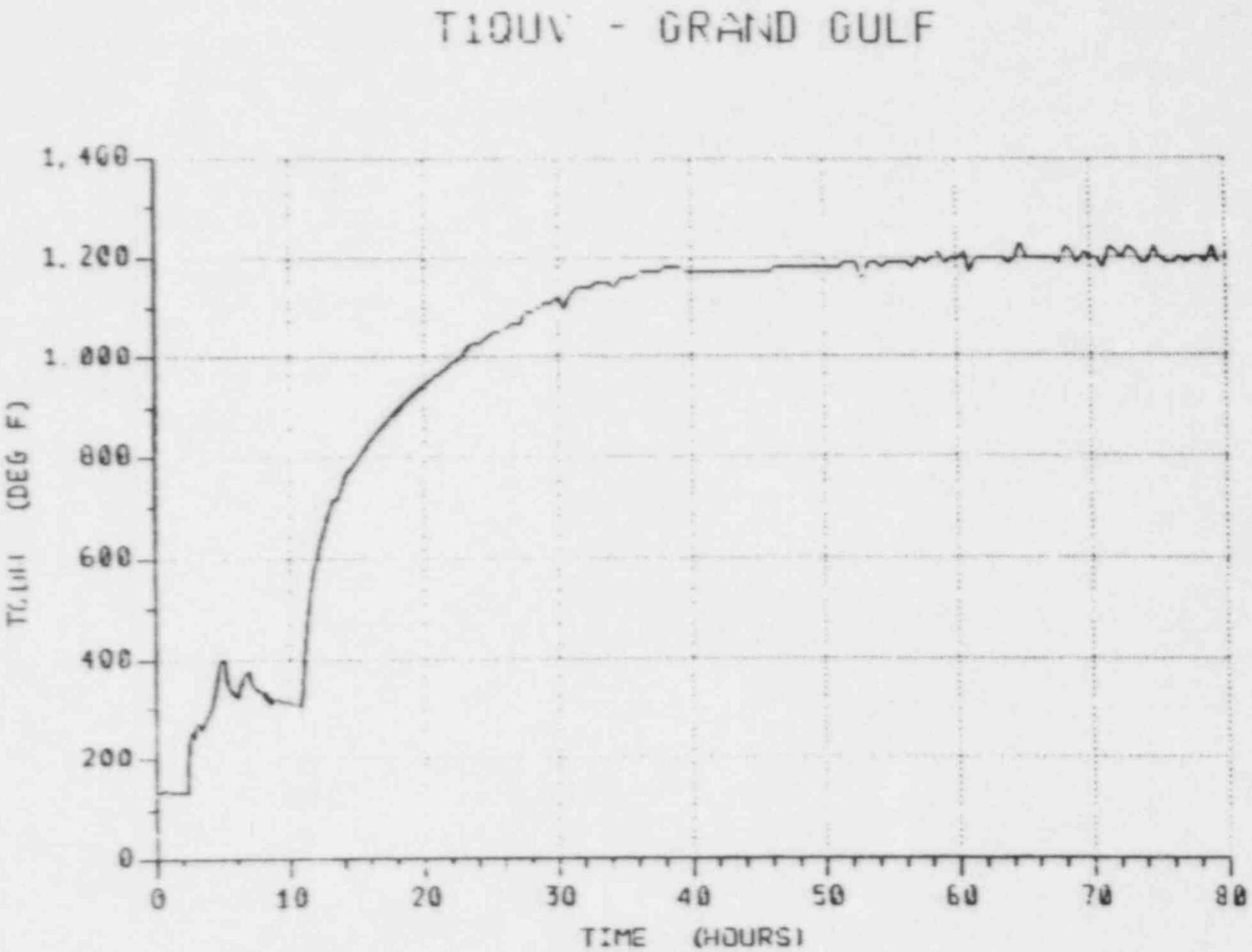

Fig. 4.2 Temperature of gas in the drywell.

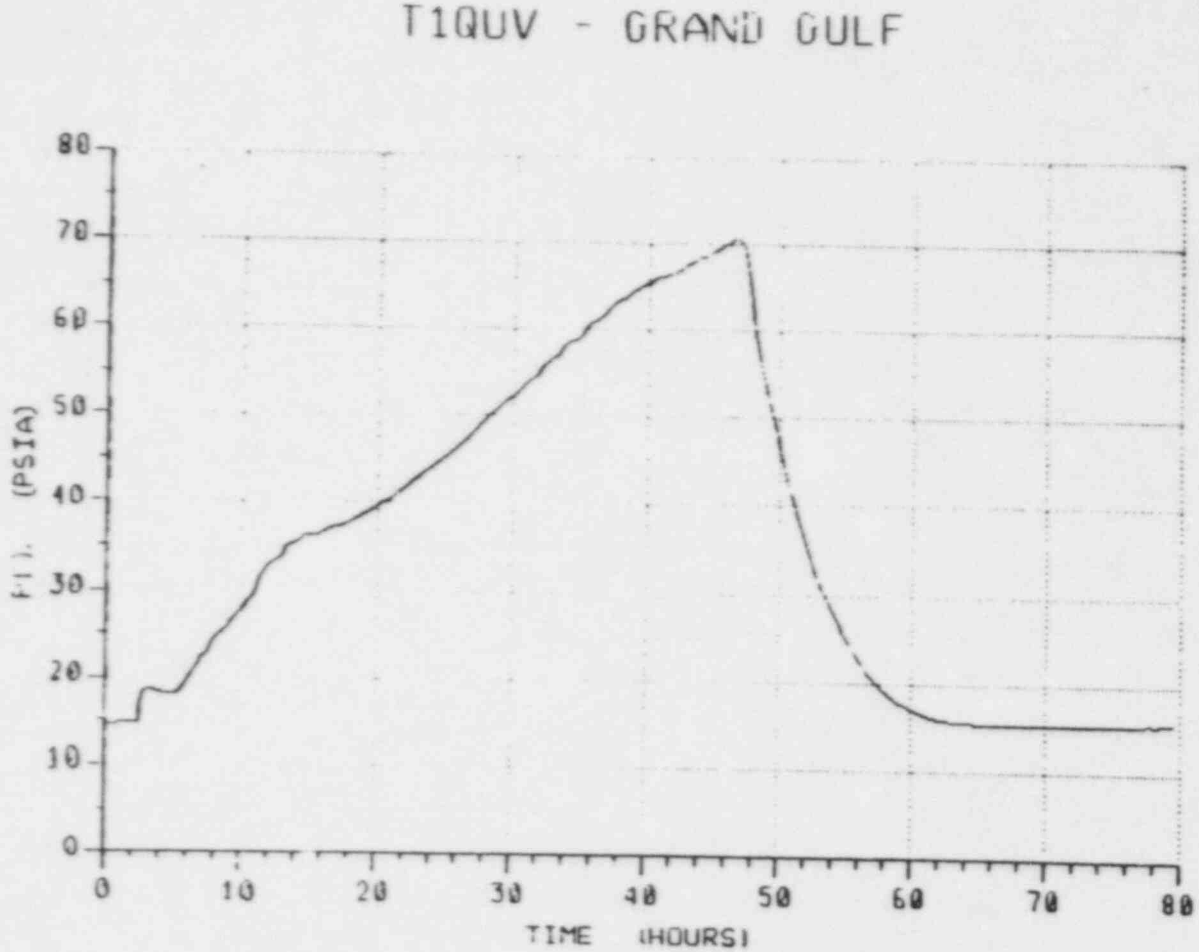

Fig. 4.3 Pressure in Compartment B.

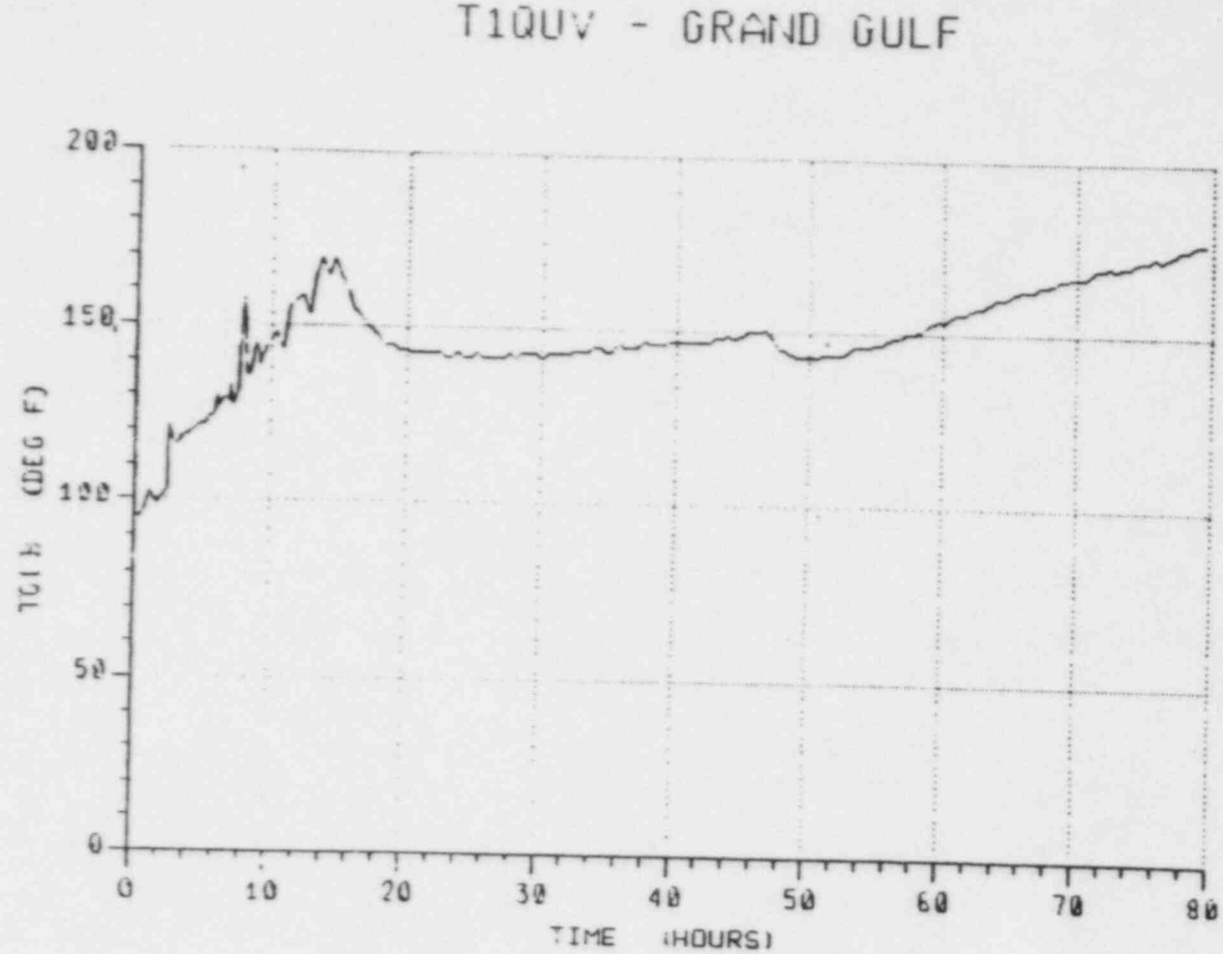

Fig. 4.4 Temperature of gas in Compartment B.

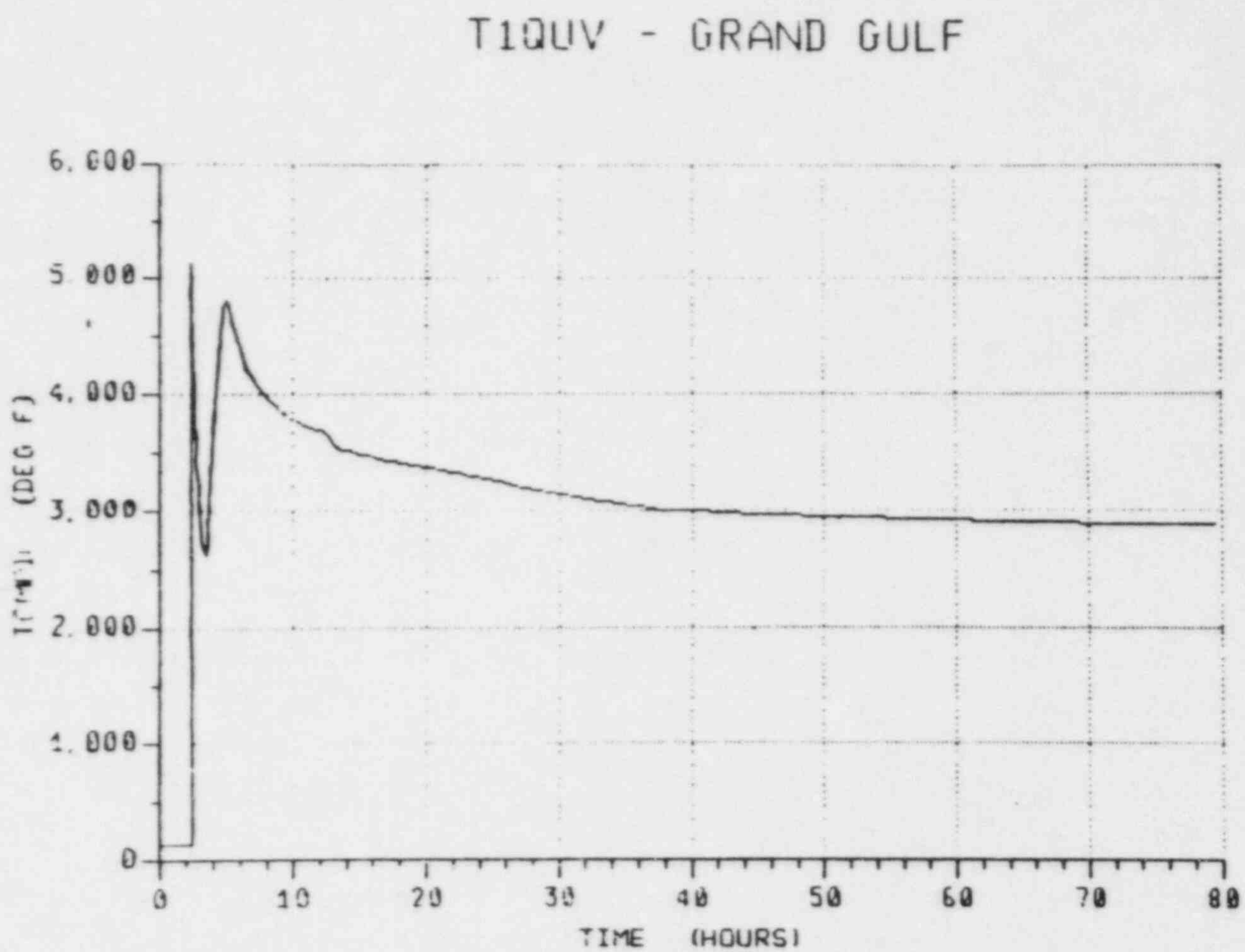

Fig. 4.5 Average corium temperature in the pedestal.

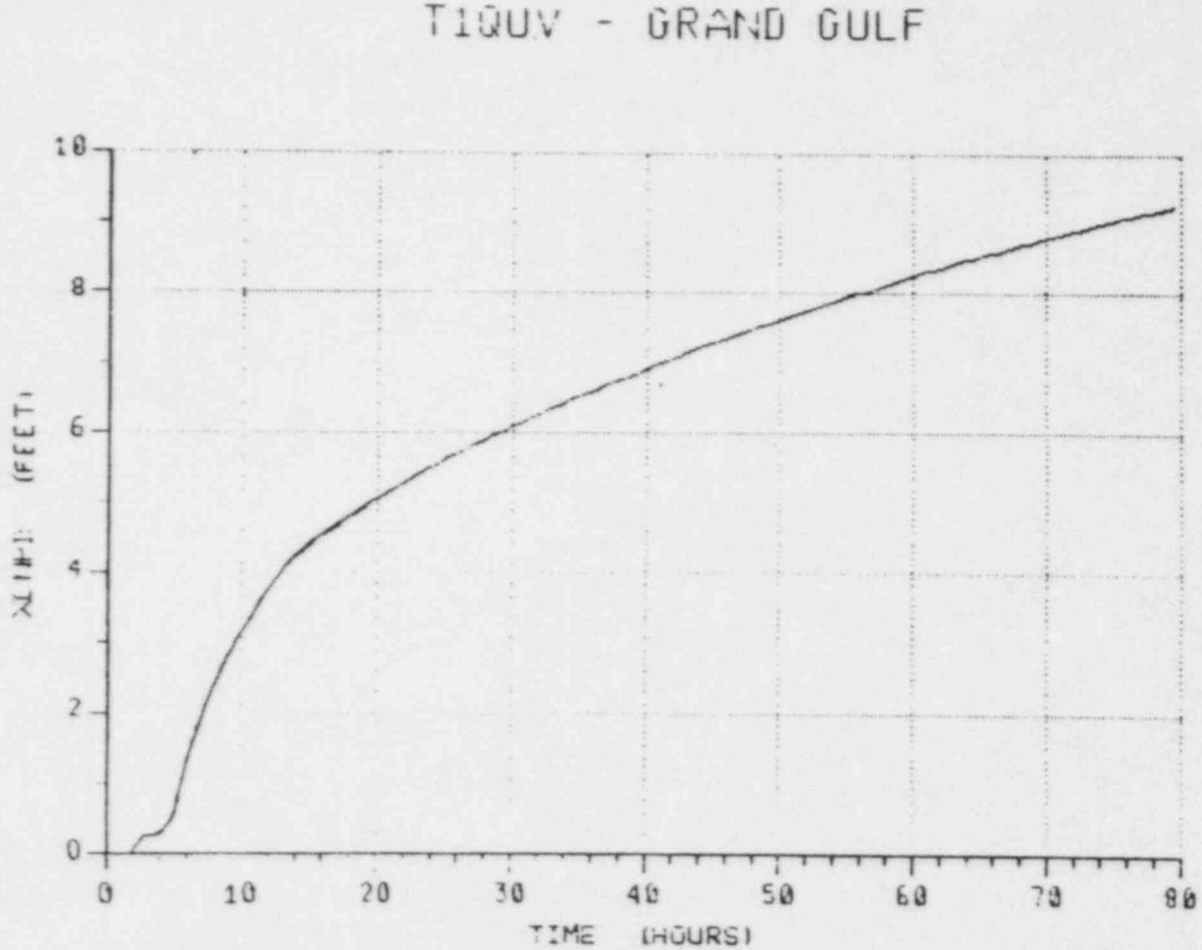

Fig. 4.6 Concrete ablation depth in the pedestal.

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hydrogen have been produced and burned to this point. Afterward, the endothermic reactions of elemental carbon with steam and with carbon dioxide begin in the corium debris bed, and hydrogen and carbon monoxide are evolved. At about 15 hours, when the oxygen concentration falls below a combustible level in all containment compartments, burning ceases. At about 38 hr, the corium inventory of elemental carbon is exhausted and combustible gas production ceases; steam and carbon dioxide gas production continues. A total of 3000 lb of hydrogen and 75,000 lb of carbon monoxide is calculated to be produced during this accident. Note, however, that less than 100 lb of the hydrogen came from in-vessel production.

Primarily because the primary system was depressurized prior to vessel failure, debris did not disperse from the pedestal to the drywell. Consequently, the temperature and pressure in the drywell behave as shown in Figs. 4.1 and 4.2. Note the rapid pressure rise in the drywell after vessel failure to about 26 psia due to debris entering the pedestal. The drywell temperature rise following vessel failure is due to the corium/concrete attack in the pedestal.

At 47 hr into the event, the GGNS containment reaches 71.3 psia (see Fig. 4.3). The containment is assumed to fail at this pressure at a location just below the junction between the cylinder and the dome [4.1]*. The failure cause is overpressurization by noncondensable gases. A containment breach area of 0.1 ft^2 was selected for modeling the containment depressurization. For this containment failure size and location, the containment depressurizes to within about 0.5 psid of atmospheric pressure in about 10 hours. The suppression pool remains intact following the containment failure event. As can be seen in Fig. 4.7, the pool temperature is less than 200°F at the time of the containment failure. Note that the suppression pool remains subcooled throughout the accident. Appendix B includes additional plots of results for this sequence.

*This is consistent with the analyses reported in Ref. [4.1] which only addressed the ultimate capacity. Consequently, failure modes were not addressed, specifically the effects on penetrations under large deflections.

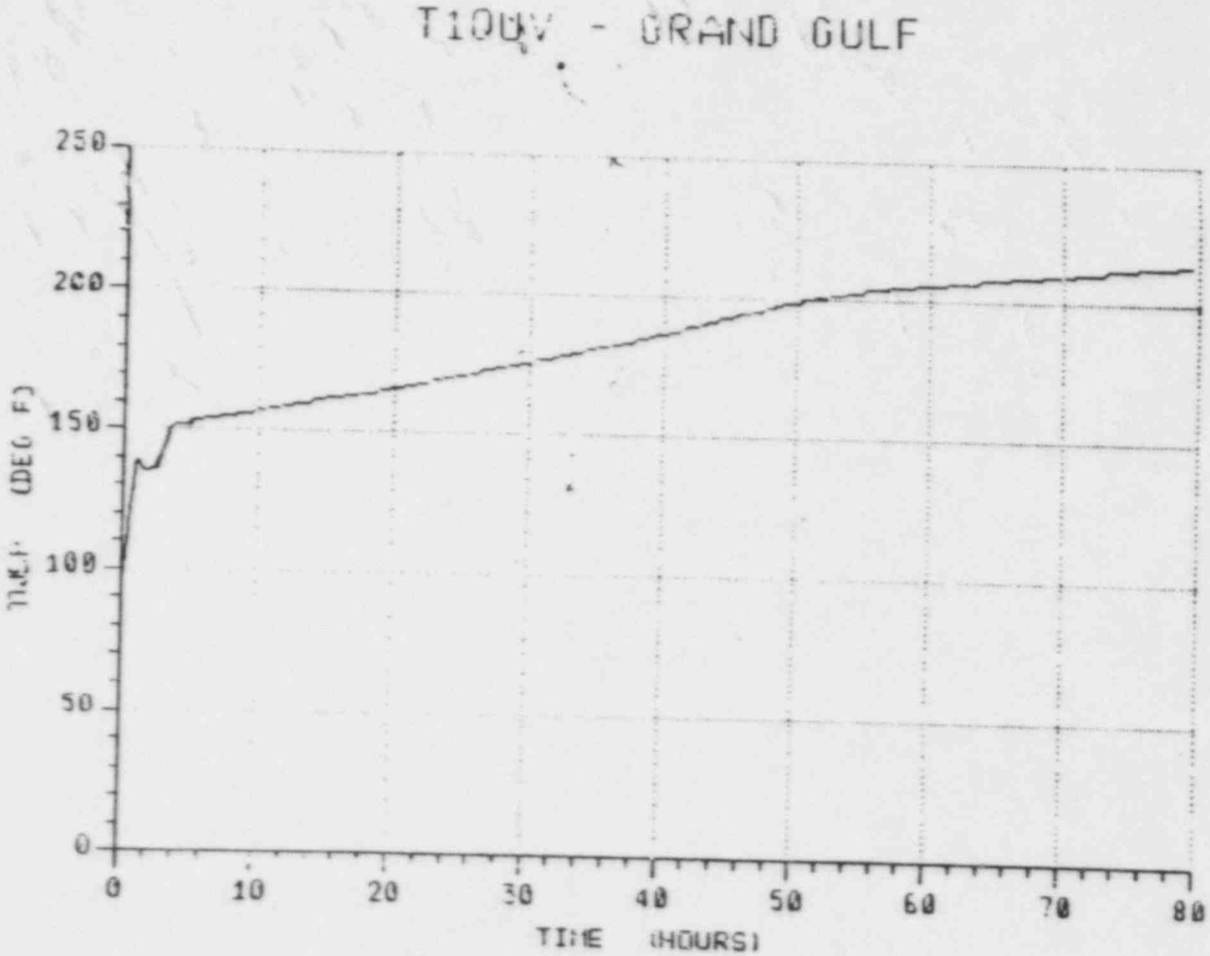

Fig. 4.7 Temperature of the suppression pool.

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4.1.3 Manual Depressurization Sensitivity Analysis

In order to assess the sensitivity of the accident response to the assumption of manual vessel depressurization prior to vessel failure, the T_1QUV accident scenario was reanalyzed without vessel depressurization. No major variations in the sequence resulted, although some of the details differed. Key differences between this analysis and the base case are shown in Table 4.2.

For the most part, differences from the base case prior to vessel failure are small, and are due core degradation occurring at an elevated or reduced pressure. The only significant differences are the longer time to vessel failure, the increased in-vessel hydrogen production, and the higher primary system gas temperatures. The first two are due to the slower boiloff of primary system water, and the latter is due to the higher hydrogen generation rates.

Following vessel failure, most of the molten core debris exiting the vessel is dispersed from the pedestal to the drywell, in contrast. No such dispersion occurs into the base case. Despite this difference, Table 4.2 shows that the difference in drywell pressurization from the dispersal is not large between the two cases. Since the core debris in the drywell is well-dispersed, the heat losses are too large for the debris to reach concrete ablation temperatures. The gas and structural temperatures in the drywell rise more quickly than in the base case, however.

There is less concrete attack in the pedestal than in the base case due to the smaller corium inventory in the pedestal. This results in a slower ablation rate, less noncondensable gas generation, and a longer time to containment failure. In summary, while there are minor differences in the accident progression, these would not substantially alter the overall accident response.

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Table 4.2

EFFECTS OF DEPRESSURIZATION IN THE T_1QUV ACCIDENT

Quantity	Depressurization at 0.43 hr	No Depressurization Until Vessel Failure
Core Uncovery Time, hr	0.47	0.62
Vessel Failure Time, hr	2.35	3.4
Containment Failure Time, hr	47.0	60
In-Vessel Hydrogen Production, lb	10	430
Mass of Core Debris in Dry-well Following Vessel Failure, lb	0	48,000
Pressure in Drywell Follow-ing Vessel Failure, psia	26	45
Gas Temperature in Drywell at Vessel Failure, °F	370	550
Concrete Ablation in Pedestal at 50 hr	7.6	7.2
Total Hydrogen Produced, lb	3,000	3,200
Total Carbon Monoxide Produced, lb	75,000	66,000

4.2 Plant Response to the AE Accident

4.2.1 Sequence Description

The AE accident is assumed to occur during full-power operation. This accident is a large-break loss of coolant accident (LOCA). It is initiated by a 3.14 ft^2 liquid line break (Event A) in the suction side of the recirculation loop. The accident sequence specifies that neither the high-pressure nor the low-pressure emergency core cooling systems (ECCS) are available at any time during the accident (Event E). The faults in these makeup systems are taken to be such that the systems are unavailable in any of their modes of operation. Thus, for this event, the only water makeup to the reactor pressure vessel (RPV) is due to the control rod drive (CRD) flow; neither the primary system nor containment heat removal is assumed available. All other plant systems are modeled to be available. No credit is taken for any operator action other than to start the containment igniter system at the accident initiation. The AE accident chronology is provided on Table 4.3.

4.2.2 Primary System and Containment Response

The loss of coolant through the primary system break causes a rapid depressurization of the RPV and a rapid pressurization of the drywell (DW). The DW pressure reaches the 1.73 psig and 1.89 psig high drywell pressure LOCA setpoints by 0.2 sec into the accident. The former generates a reactor scram signal; the latter generates actuation signals for the high pressure core spray (HPCS), the low pressure core spray (LPCS), and the low pressure coolant injection (LPCI) systems. The reactor scram is modeled to bring the reactor subcritical by 3.9 sec. The core power remains at decay heat levels for the remainder of the event. Since the HPCS, LPCS and LPCI are assumed unavailable, the RPV level drops to the RPV Level 2 LOCA setpoint. At this point, 5.2 sec into the event, the recirculation pumps are signaled to trip off and the reactor core isolation cooling (RCIC) system is signaled to start. The recirculation pump trips are completed by 5.6 sec; RCIC is assumed unavailable. The RPV water level falls to the RPV Level 1 LOCA setpoint at 6.5 sec. At this point, the main feedwater system trips off and the main steam isolation valves (MSIV) close. In addition, permissive signals are generated for

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Table 4.3

GRAND GULF NUCLEAR STATION

AE BASE CASE

ACCIDENT CHRONOLOGY

Time	Event
0.0 sec	Initiating Event: A large break in suction side of a recirculation loop
0.2 sec	High DW pressure LOCA setpoints reached
3.9 sec	Reactor scram completed
5.2 sec	RPV Level 2 LOCA setpoint reached
6.5 sec	RPV Level 1 LOCA setpoint reached; MSIVs close; Main feedwater pumps trip off
45.0 sec	Core begins to uncover
11.6 min	DW purge system actuates
30.4 min	SPMU actuates
1.1 hr	Fuel melting begins
1.4 hr	Core plate failure followed by vessel failure
22.3 hr	CST drained and CRD flow to vessel ceases
58.0 hr	Containment failure

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the suppression pool makeup (SPMU) and the drywell (DW) purge systems. The SPMU system releases the upper containment pool following a programmed 30 minute delay. DW purge actuation is delayed until other permissives are satisfied. Without sufficient water inventory makeup, the core begins to uncover at 45 sec.

Temperatures in the uncovered fuel regions begin to rise and begin to reach 2000°F at about 13 min. The cladding oxidation rate increases rapidly above this point. Oxidation of the Zircaloy cladding, in turn, increases the fuel heatup rate and tends to promote further cladding oxidation. Since the boiloff time is short for the large-break LOCA response, in-vessel Zircaloy oxidation is minimal.

Fuel melting is predicted to begin at 1.1 hr, and relocates to the core plate. By 1.4 hr, sufficient core material is calculated to have fallen onto the RPV core plate to cause it to fail. The core debris then falls to the bottom of the RPV and thirty seconds later, vessel failure occurs at a welded RPV penetration point.

At vessel failure, the molten fraction of the lower plenum core debris falls onto the pedestal floor followed by the saturated lower plenum water. A small steam spike occurs at this point, causing a pressure rise in the pedestal and drywell to about 26 psia. Since the vessel was depressurized prior to failure, no debris is dispersed from the pedestal to the drywell. Drywell leakage flow paths bypassing the suppression pool are modeled to plug with aerosols. These aerosols are released from the vessel when it fails and from the core-concrete interaction in the pedestal. All flow exiting the drywell to the outer containment is afterward forced to pass through the suppression pool.

The debris attacks the concrete until it is cooled below concrete ablation temperatures by the lower plenum water at about two hours. The concrete is ablated to a depth of 2.5 inches up to this time. The remaining water in the pedestal, plus that continually added by the CRD flow, is boiled away while slowly quenching the debris, as can be seen on Figs. 4.8 and 4.9. By about seven hours, the debris and the water are at about the same

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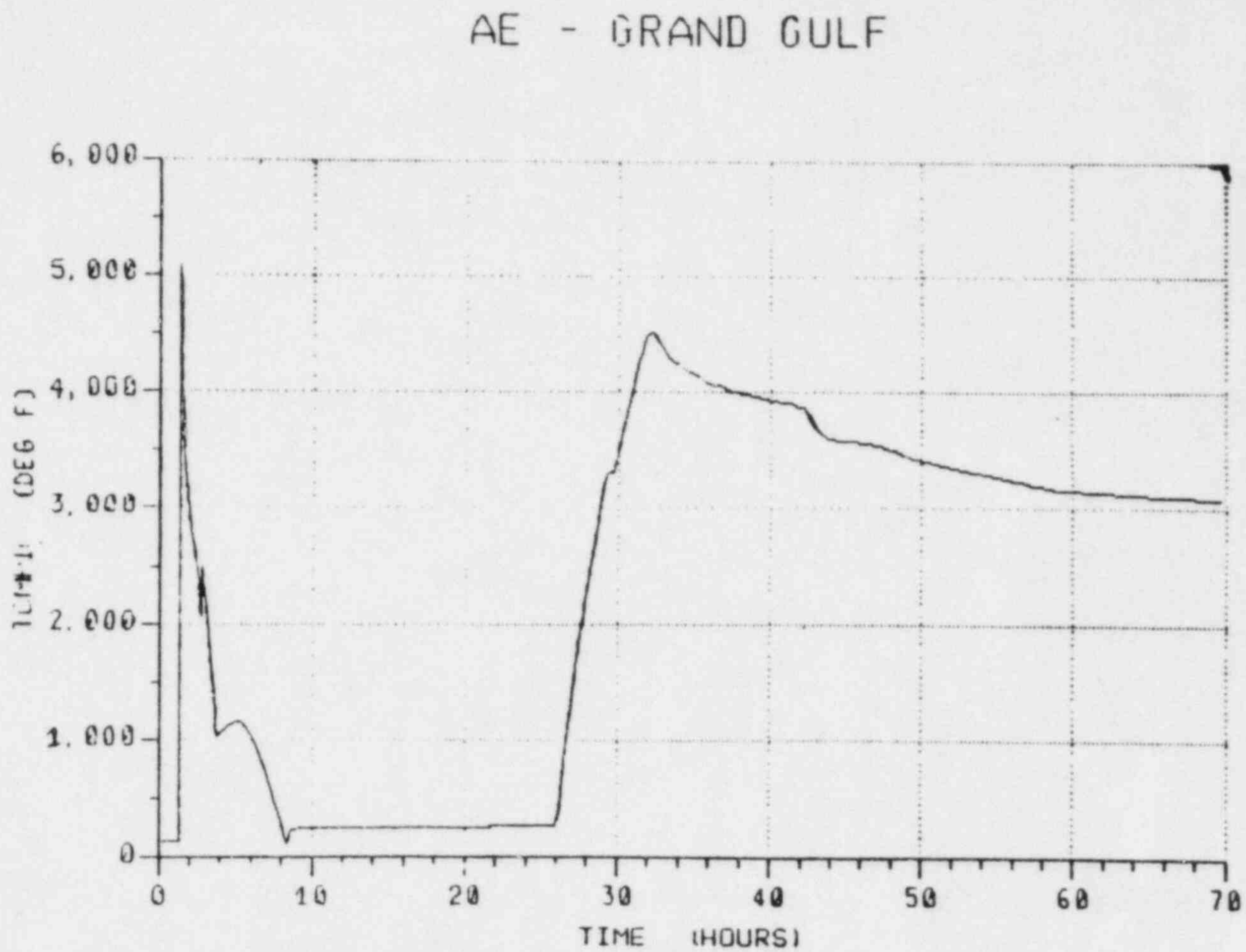

Fig. 4.8 Average corium temperature in the pedestal.

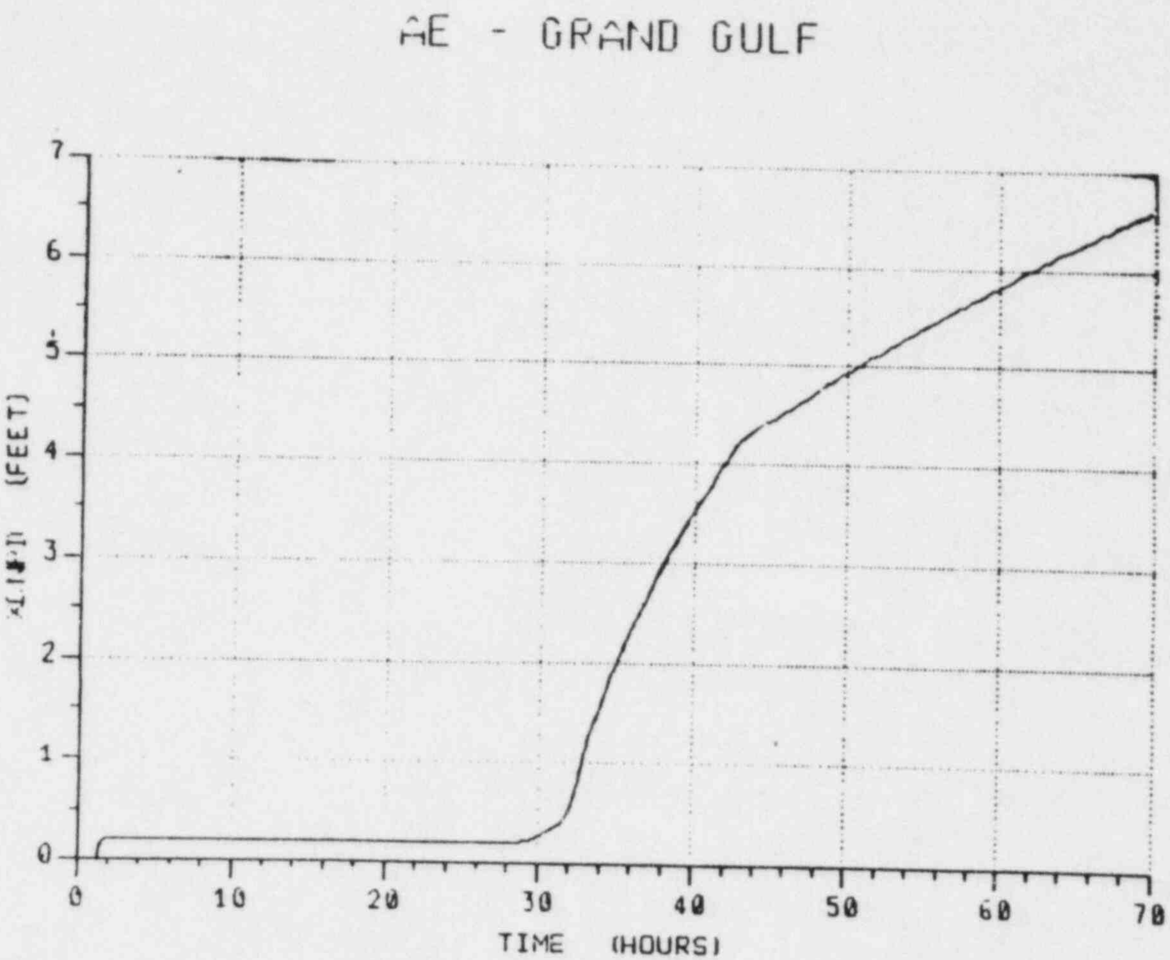

Fig. 4.9 Concrete ablation depth in the pedestal.

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temperature. From this point on, the continuing CRD flow into the pedestal refills it to the pedestal doorstep level. Excess water spills into the drywell. The CRD flow keeps the debris quenched until the CST runs out of water at 22.3 hours. Without replenishment, the pedestal water boils away and, by 26 hours the debris begins to reheat. Concrete ablation in the pedestal resumes at 30 hours.

The thermal decomposition of the pedestal concrete floor and walls produces large volumes of carbon dioxide and steam. As these two gases pass through the partially molten corium debris bed in the pedestal, they oxidize the zirconium in the bed to produce hydrogen gas and elemental carbon. The igniters provide for an almost continuous controlled burn-off of all combustible gases being evolved during the accident. The first burn begins at about 35 hours; thereafter, their continuous burn-off prevents high concentrations of combustible gases from occurring. By 43 hr, 100% of the zirconium has been oxidized. At this point, the endothermic reactions of elemental carbon with steam and with carbon dioxide begin in the corium debris bed. Hydrogen and carbon monoxide are evolved in these reactions. At about 45 hours, when the oxygen concentration falls below a combustible level in all containment compartments, burning ceases and the containment becomes self-inerted.

Drywell temperatures rise to about 900°F after the core debris-concrete interaction resumes in the pedestal, as shown on Fig. 4.10. The suppression pool water temperature, shown on Fig. 4.11, reaches saturation due to the large amount of steam generated by quenching the debris in the pedestal prior to dryout. Temperatures in compartment B remain relatively low due to the cooling effect of the suppression pool (as shown in Fig. 4.12).

At 58 hours into the event, the GGNS containment reaches 71.3 psia (see Fig. 4.13). The containment is modeled to fail at this pressure at a location just below the junction between the cylinder and the dome. The cause is overpressurization by steam and by noncondensable gases. A containment breach area of 0.1 ft^2 was selected for modeling the containment depressurization. For this containment failure size and location, the containment depressurizes to within 0.5 psid of atmospheric pressure in about 10 hours. And, the suppression pool remains intact following the containment failure event.

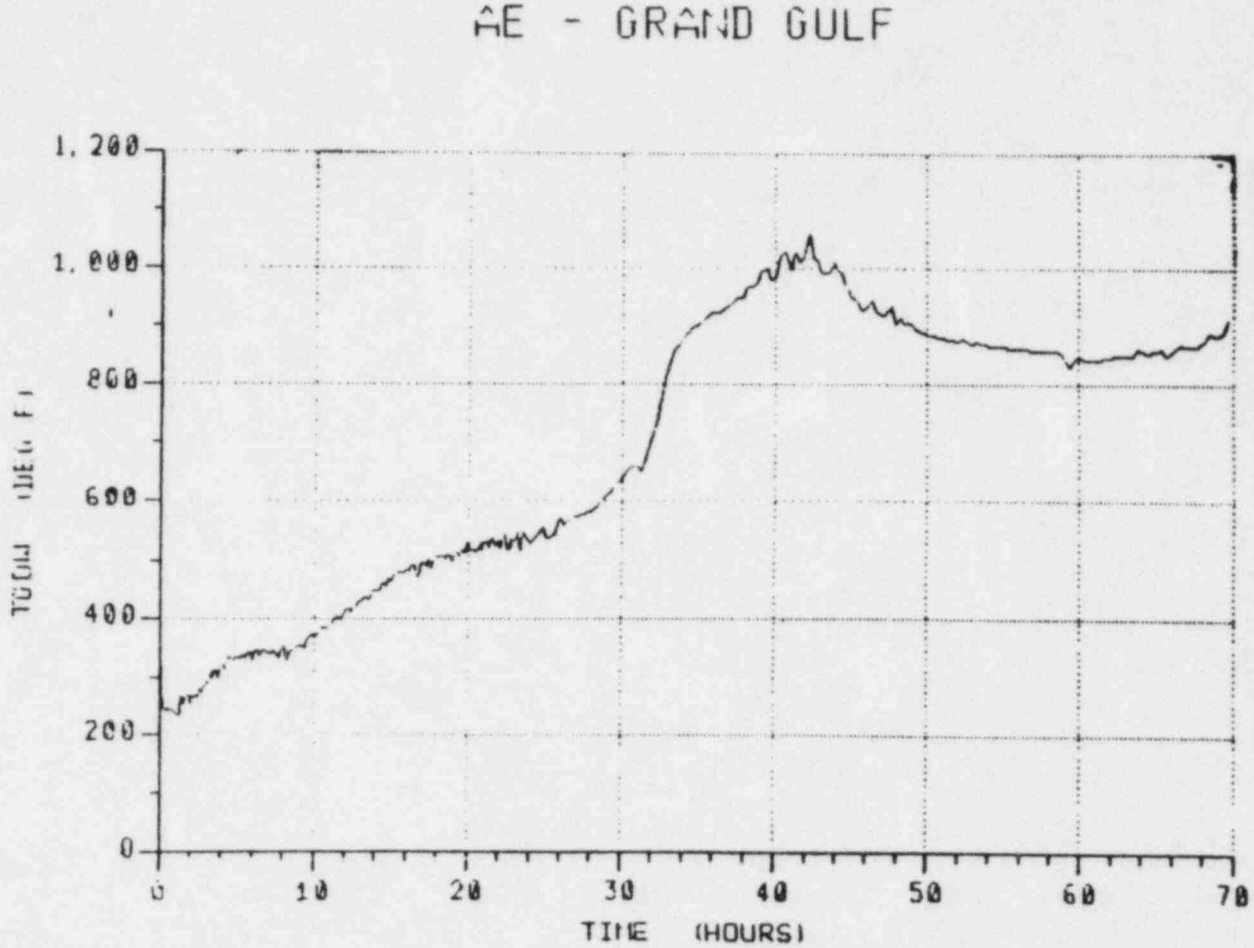

Fig. 4.10 Temperature of gas in the drywell.

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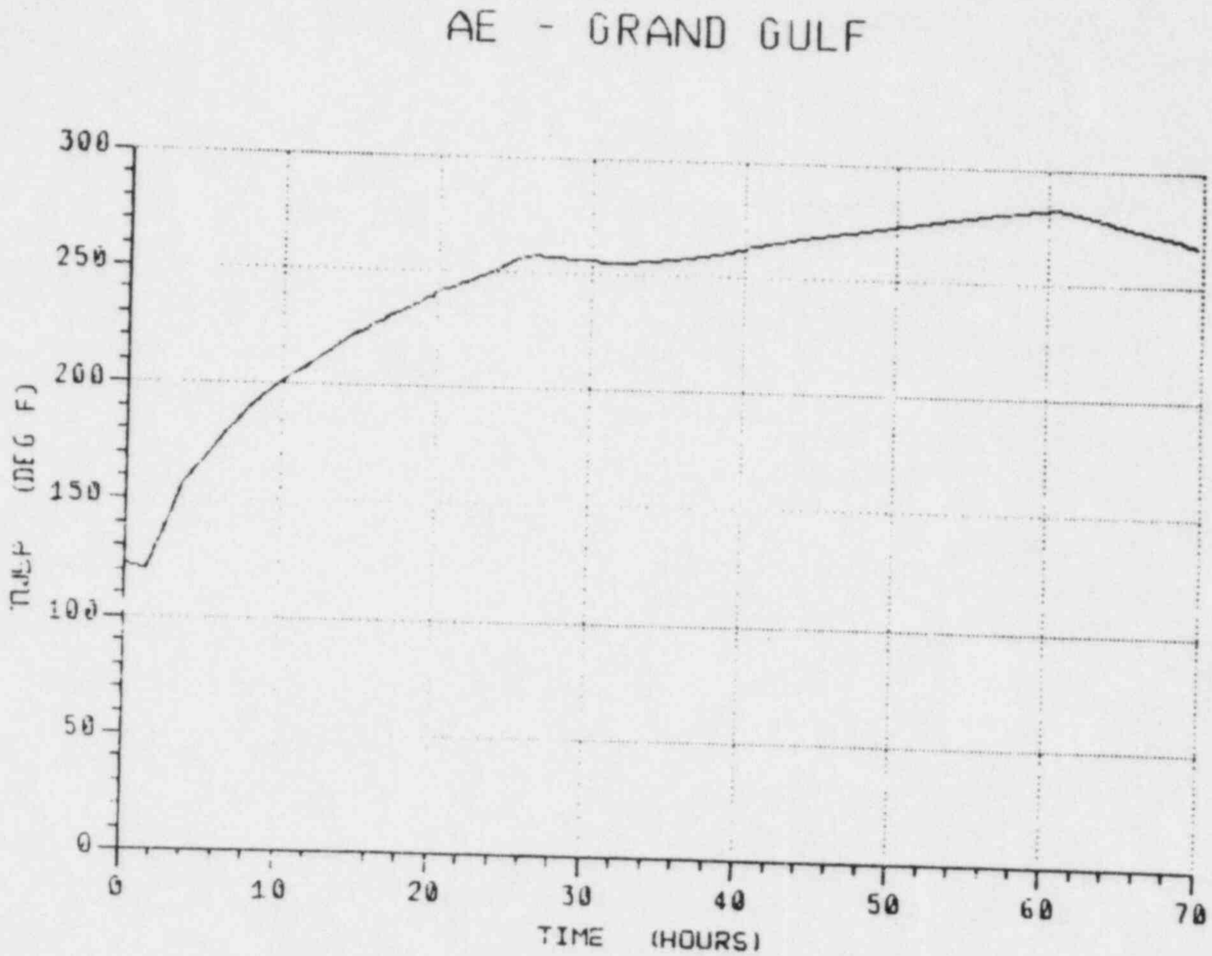

Fig. 4.11 Temperature of the suppression pool.

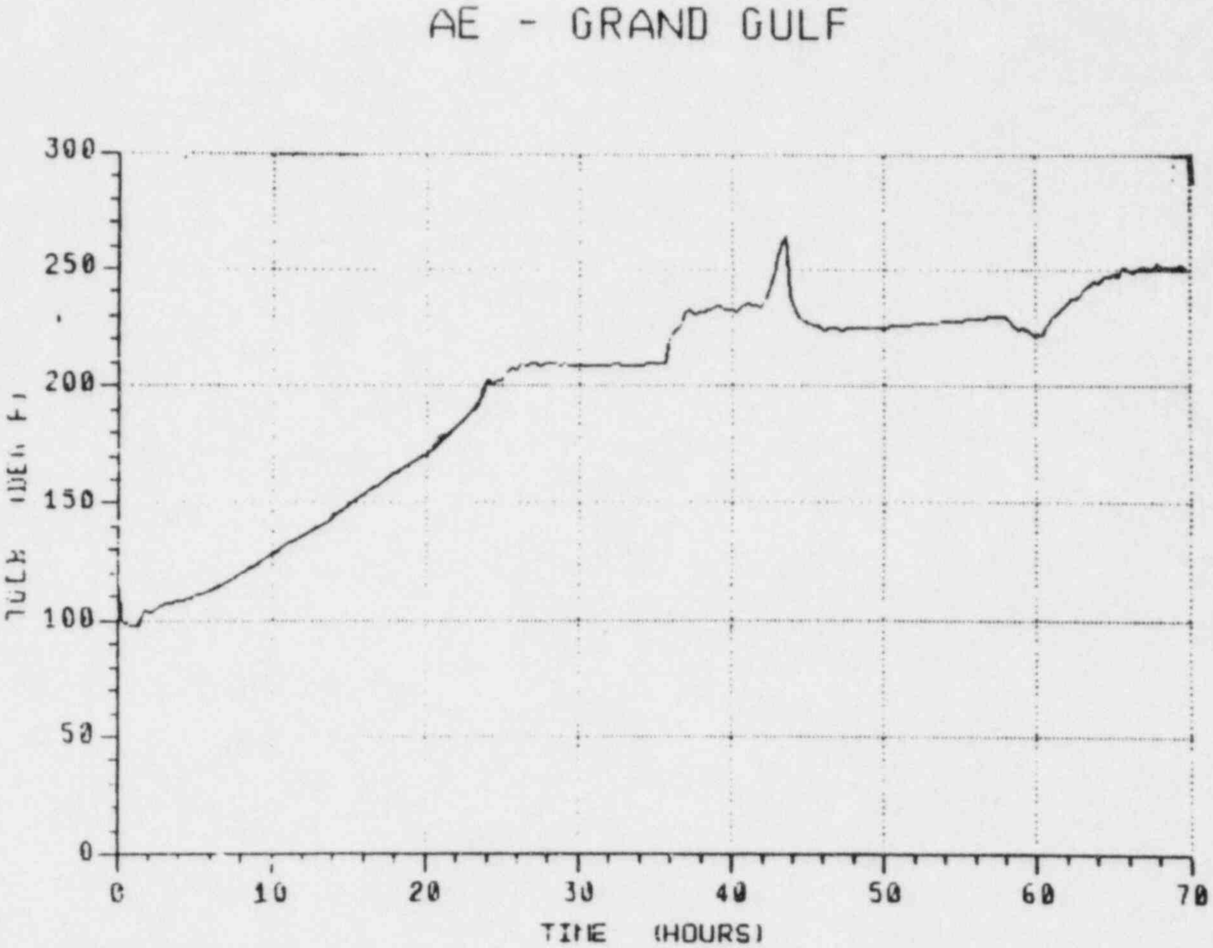

Fig. 4.12 Temperature of gas in Compartment B.

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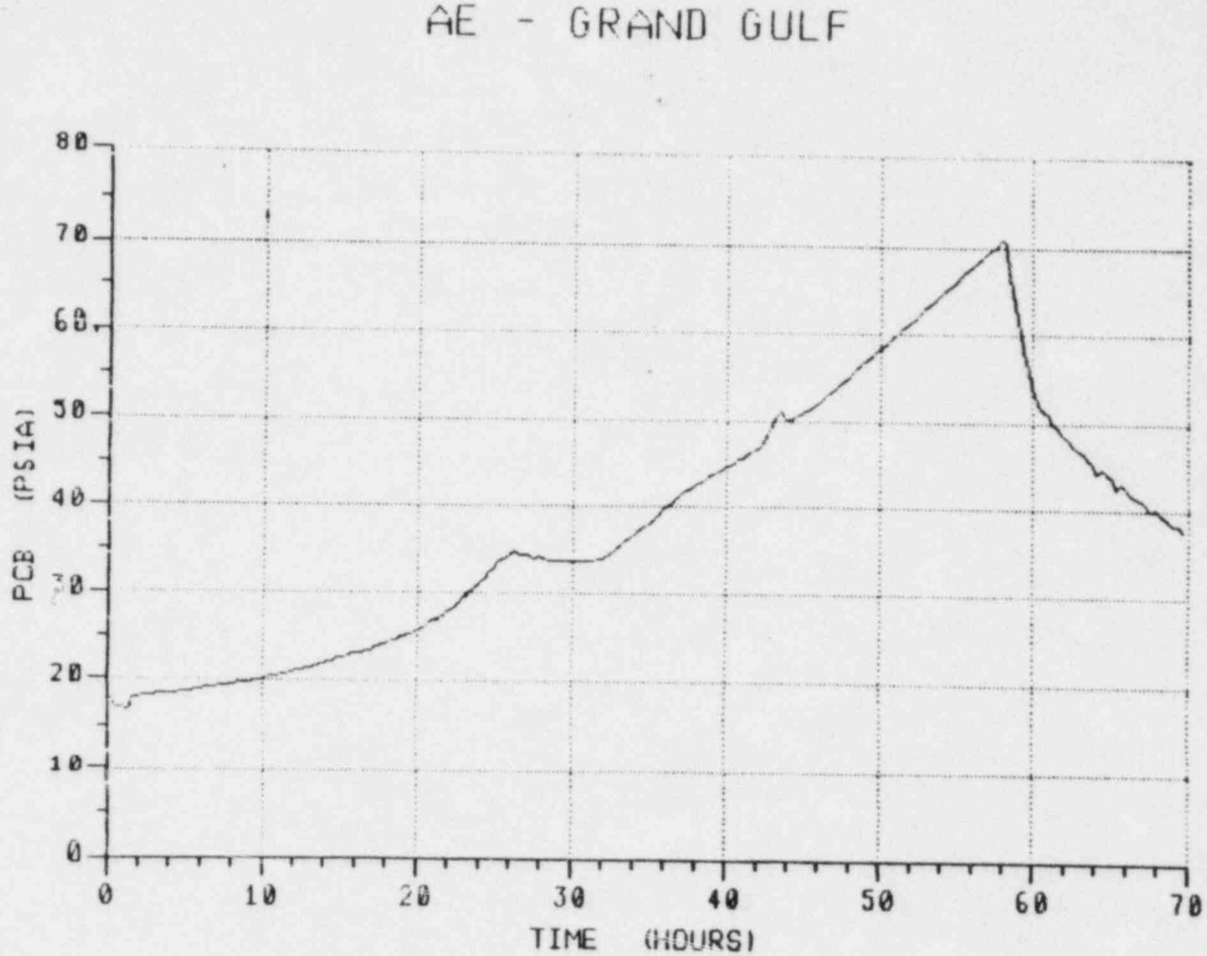

Fig. 4.13 Pressure in Compartment B.

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Since the pool temperature is nearly 280°F at the time of the containment failure, about 2% of the pool inventory is calculated to boil away within 10 hrs following failure. Appendix B includes additional plots of results for this sequence.

4.3 Plant Response to the $T_{23}QW$ Accident

4.3.1 Sequence Description

The $T_{23}QW$ accident is assumed to occur during full-power operation. It is initiated by inadvertent main steam isolation valve (MSIV) closures (Event T_{23}). The main feedwater and main condenser are assumed to be unavailable (Event Q) for the entire accident. The accident sequence also specifies that containment heat removal is not available for the entire accident (Event W). Control rod drive (CRD) flow to the reactor pressure vessel (RPV) is modeled to be available. All other plant systems are assumed to be available. However, all emergency core cooling systems (ECCS) are assumed to fail on containment failure. No credit is taken for any operator action other than to start the containment igniter system at the accident initiation and to manually depressurize the RPV when the suppression pool temperature exceeds 145°F. The $T_{23}QW$ accident chronology is provided on Table 4.4.

4.3.2 Primary System and Containment Response

The initiating event, which is inadvertent closure of the MSIV, causes a reactor pressure vessel (RPV) pressure excursion which is relieved by the safety relief valves (SRV). The exiting RPV steam is routed to the suppression pool (SP), where it is quenched. The MSIV closures actuate a reactor scram which is modeled to bring the reactor subcritical by 3.7 sec into the event. The core power remains at decay heat levels for the remainder of the event. At 2.35 hours into the accident the suppression pool temperature exceeds 145°F and an operator intervention occurs to manually initiate ADS.

At 4.1 hr, steam pressurization of the containment building causes a high drywell (DW) pressure LOCA signal. This signal is a permissive signal

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Table 4.4

GRAND GULF NUCLEAR STATION

$T_{23}QW$ - BASE CASE

ACCIDENT CHRONOLOGY

Time	Event
0 sec	Initiating event: MSIV closures; Loss of main feedwater
3.7 sec	Reactor scram completed
28 sec	RPV Level 2 LOCA setpoint reached
1.0 min	HPCS and RCIC systems begin operating
1.1 hr	HPCS and RCIC systems transfer suction from CST to SP
2.35 hr	Suppression pool temperature exceeds 145°C, manual ADS
4.1 hr	High DW pressure LOCA setpoint reached; DW purge system actuates; LPCS and LPCI actuate (but cannot provide makeup without RPV depressurization)
4.6 hr	SPMU actuates
6.3 hr	RCIC pump fails on high suction temperature
22.4 hr	CST empties
23.5 hr	High wetwell pressure setpoint reached; Containment sprays actuate
40.0 hr	Containment failure; All ECCS assumed to fail
48.8 hr	Core begins to uncover
54.1 hr	Fuel melting begins
56.2 hr	Core plate failure followed by vessel failure

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for the DW purge system, the SP makeup (SPMU) system, and the automatic depressurization system (ADS); it is an actuation signal for the low pressure core spray (LPCS) and low pressure coolant injection (LPCI) systems. The DW purge system actuates after a 30 sec time delay and the SPMU system actuates the upper containment pool dump following a programmed 30 min delay. The RPV water inventory is maintained by the HPCS and RCIC systems. The high DW pressure LOCA signal is modeled to switch the HPCS and RCIC systems' level control logic to maintain the RPV water level about the RPV Level 8 setpoint.

Because of the assumed unavailability of containment cooling, the SP temperature rises during most of this event (Fig. 4.14). One exception to this trend occurs at 4.6 hr, when the SP makeup system releases relatively cold upper containment pool water into the SP. After the upper pool dump, the SP water temperature continues to rise again. When the SP temperature reaches 200°F at 6.3 hr, the RCIC pump is modeled to fail due to high bearing temperatures. After the loss of the RCIC, the HPCS and the CRD flow continue to maintain adequate RPV inventory. Driven by the steam produced in the core, the containment pressure reaches the 9 psig containment spray actuation pressure setpoint at 23.5 hr into the event. Note that the accident definition assumes that the RHR heat exchangers are unavailable. Thus, the operation of containment sprays removes no heat from the containment; it merely homogenizes temperatures in the outer containment. The effect of this homogenization can be observed in Figs. 4.14, 4.15 and 4.16: the suppression pool temperature decreases, the outer containment air temperatures increase, and, consequently, the outer containment pressure increases slightly. The latter pushes water from the wetwell to the drywell side of the suppression pool and results in a large spill of suppression pool water onto the drywell and pedestal floors. This water plays a key role in quenching the core debris after the vessel fails. At that time, trains A and B of the residual heat removal (RHR) system automatically switch into their spray mode. At 22.4 hr the CST empties and the CRD flow ceases. From this point on, only the HPCS is available to maintain inventory.

At 40 hr into the event, the GGNS containment pressure reaches 71.3 psia. The containment is modeled to fail at this pressure at a location just below the junction between the cylinder and the dome. The failure cause is

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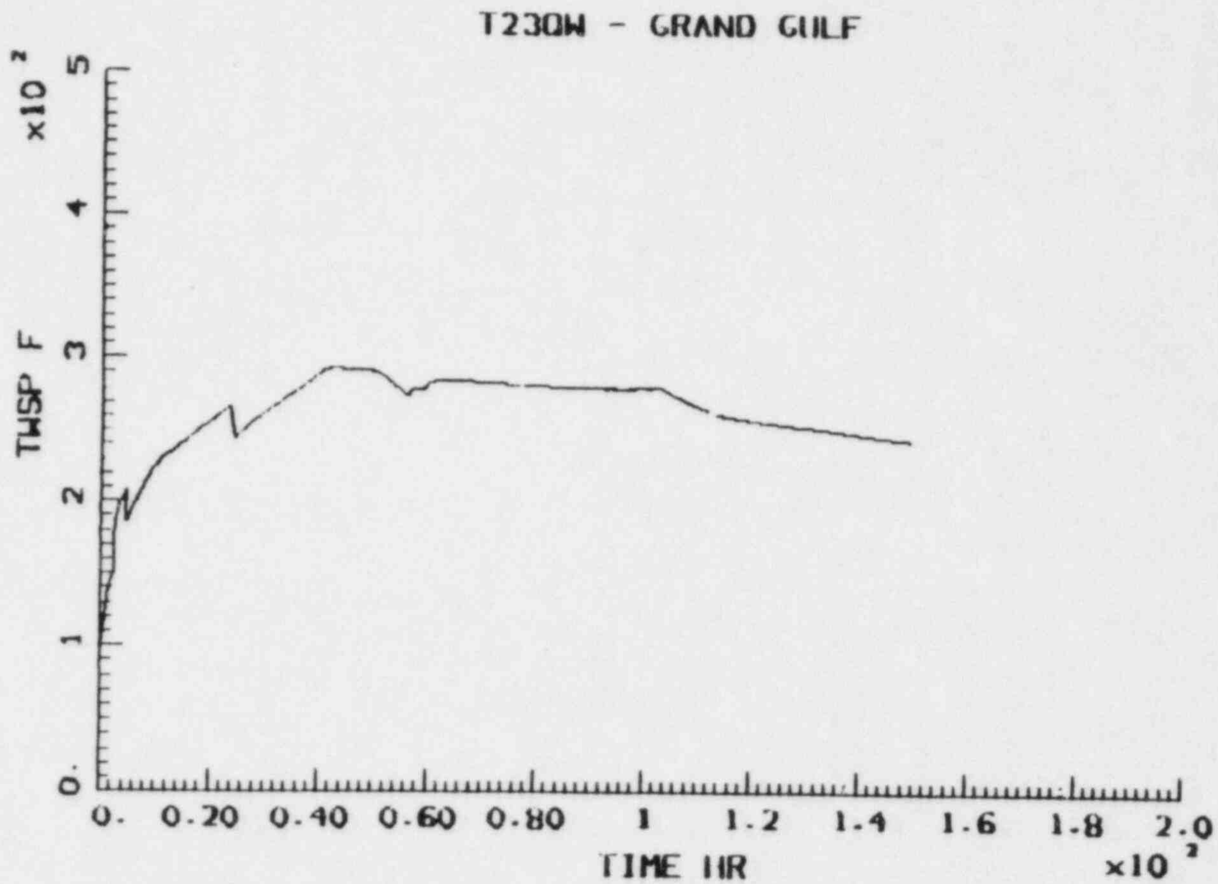

Fig. 4.14 Temperature of the suppression pool.

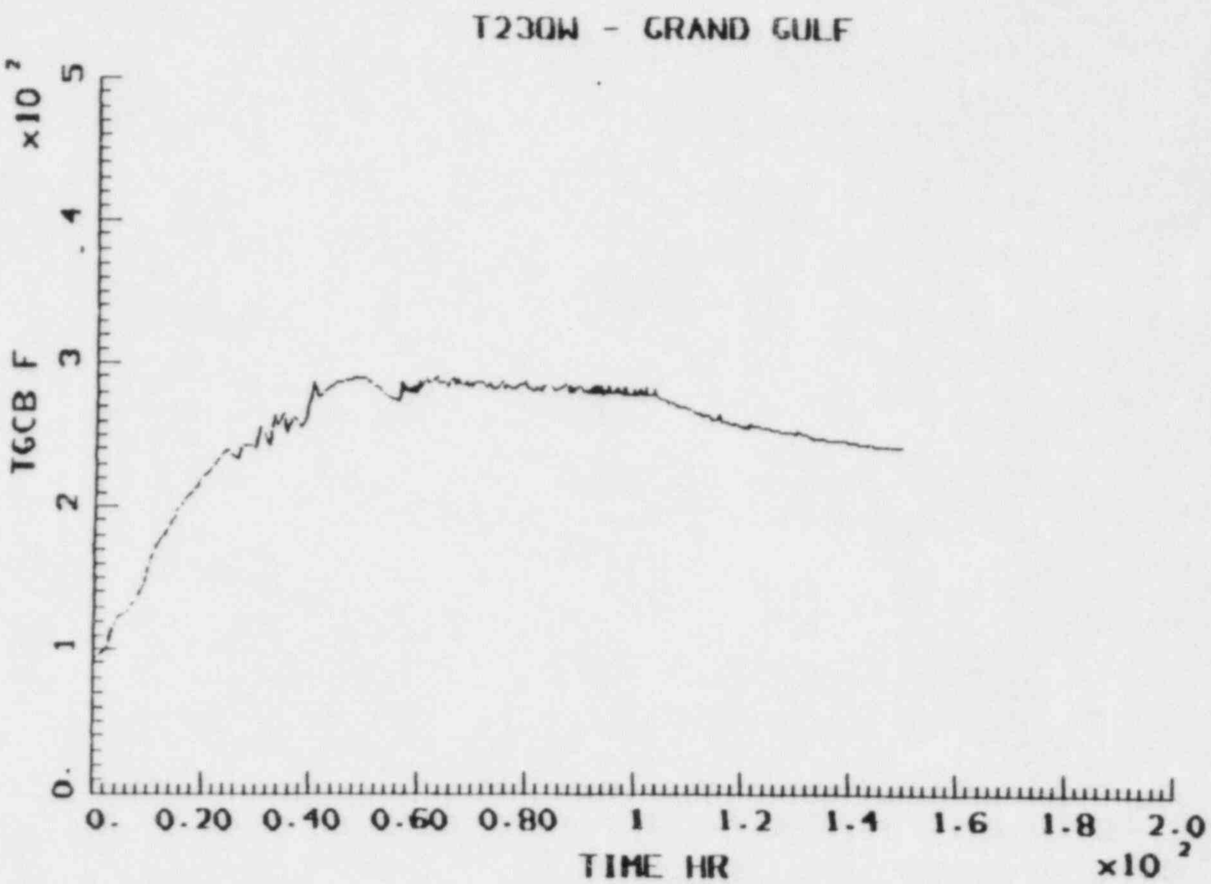

Fig. 4.15 Temperature of gas in Compartment B.

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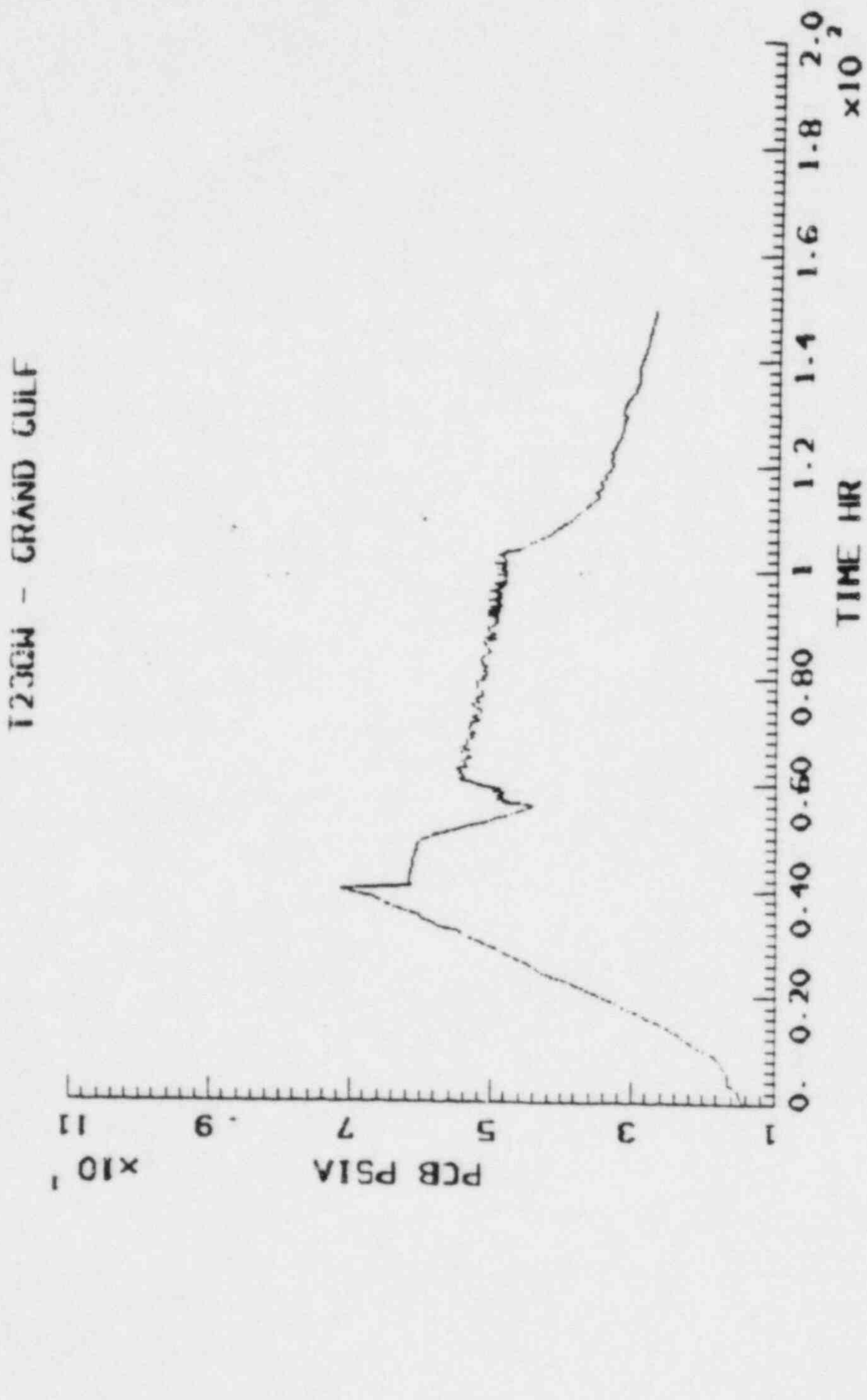

Fig. 4.16 Pressure in Compartment B.

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steam overpressurization. A containment breach area of 0.1 ft^2 was modeled. For this containment failure size and location, the containment depressurizes to within about 0.5 psid of atmospheric pressure in about 10 hours. The suppression pool remains intact following the containment failure event. Suppression pool boiloff maintains an elevated containment pressure after the containment fails. Gas temperatures in all outer containment compartments are relatively constant at about 300°F after containment failure. The drywell air temperature is shown on Fig. 4.17.

In order for the $T_{23}QW$ sequence to result in core damage, it is necessary that all systems supplying or capable of supplying water to the RPV fail at or before containment failure. A realistic mechanism which could cause such a simultaneous failure has not been identified. The accounting of containment failure location, pressure, fluid flow loading, and ECCS pump suction temperature [4.1], pressure, and NPSH limitations [4.2] indicates that at least one GGNS ECCS train should survive a containment failure event. However, for this analysis, the conservative assumption that all ECCS equipment fails on containment failure was made.

Without vessel makeup, the RPV water level falls. The decrease is relatively slow in comparison with the T_1QUV and AE events, since decay heat levels in the $T_{23}QW$ accident are relatively low. Core uncovery takes place about 8 hours after containment failure, and fuel heatup begins thereafter. Fuel temperatures in the uncovered region of the core begin rising above 2000°F at 51 hr. The clad oxidation rate increases rapidly above the 2000°F fuel temperature point. Since the oxidation of the Zircaloy fuel cladding is an exothermic reaction, its occurrence increases the fuel heatup rate and thus tends to promote further cladding oxidation. About 5% of the total Zircaloy was oxidized at vessel failure.

Fuel melting is predicted to begin at about 54 hr into the event. After melting, the fuel moves to the core plate. By 56.2 hr, sufficient core material is calculated to have fallen onto the RPV core plate to cause it to fail. The core debris then falls to the bottom of the RPV and, about 30 sec later, vessel failure occurs at a welded RPV penetration. At vessel failure,

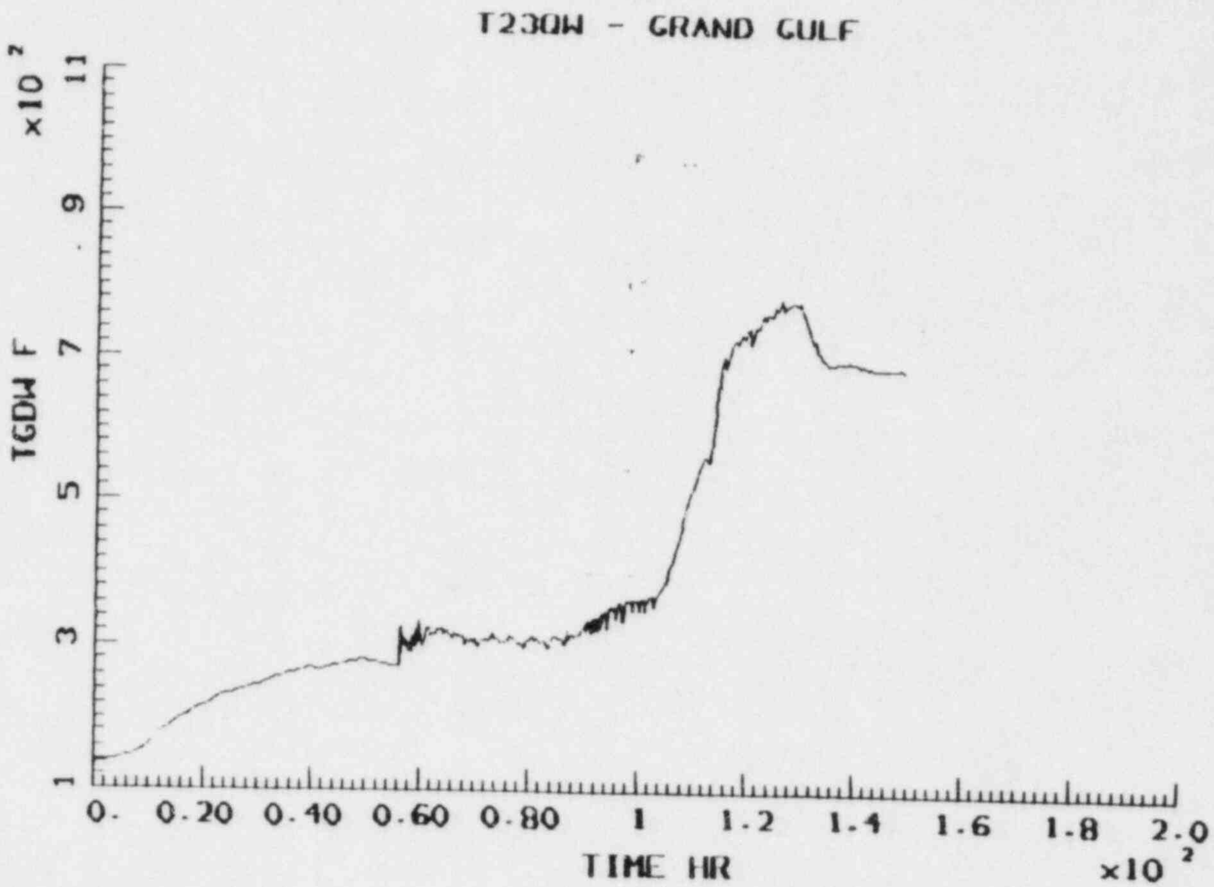

Fig. 4.17 Temperature of gas in the drywell.

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the molten fraction of the lower plenum core debris falls onto the pedestal floor followed by the flashing high-pressure lower plenum water.

Since the containment failure size was 0.1 ft^2, the suppression pool remains saturated at about 280°F, passing the steam entering it through to the upper compartment. The containment pressure remains high, gradually diminishing as the heat load diminishes, as shown in Fig. 4.16. The gas temperatures in all of the containment compartments are relatively constant at about 300°F during the period of interest. The drywell temperature variation is shown on Fig. 4.17.

Since the containment has such large amounts of steam, it is effectively inerted when the hydrogen leaving the vessel enters the wetwell (prior to vessel failure) and the drywell (after vessel failure). Hence, no burning is predicted to occur. For the same reason, any noncondensable gases that may be generated at very late times (beyond 100 hours) from core debris-concrete attack would not burn. The average corium temperature and penetration depth histories are shown in Figs. 4.18 and 4.19. Appendix B includes additional plots of results for this sequence.

4.4 Plant Response to the $T_{23}C$ Accident

4.4.1 Sequence Description

The $T_{23}C$ accident is assumed to occur during full-power operation. It is initiated by inadvertent main steam isolation valve (MSIV) closures (Event T_{23}). The accident sequence specifies that the control rod drive (CRD) system fails to automatically bring the reactor subcritical (Event C). This analysis assumes that no control rods were inserted into the core. All other plant systems are assumed to be available. No credit is taken for any operator action other than to start the containment igniter system at the accident initiation and to manually initiate ADS when the suppression pool temperature exceeds 145°F. The $T_{23}C$ accident chronology is provided on Table 4.5.

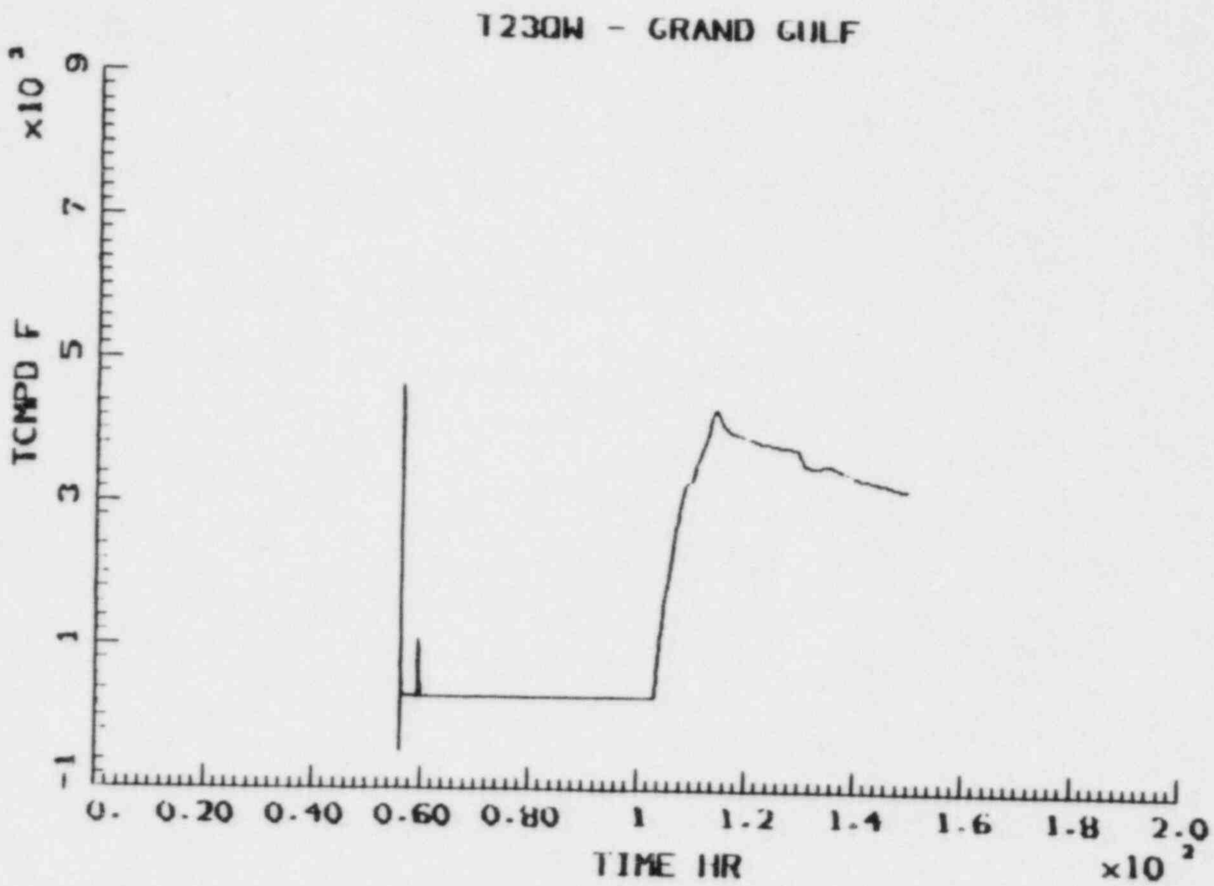

Fig. 4.18 Average corium temperature in the pedestal.

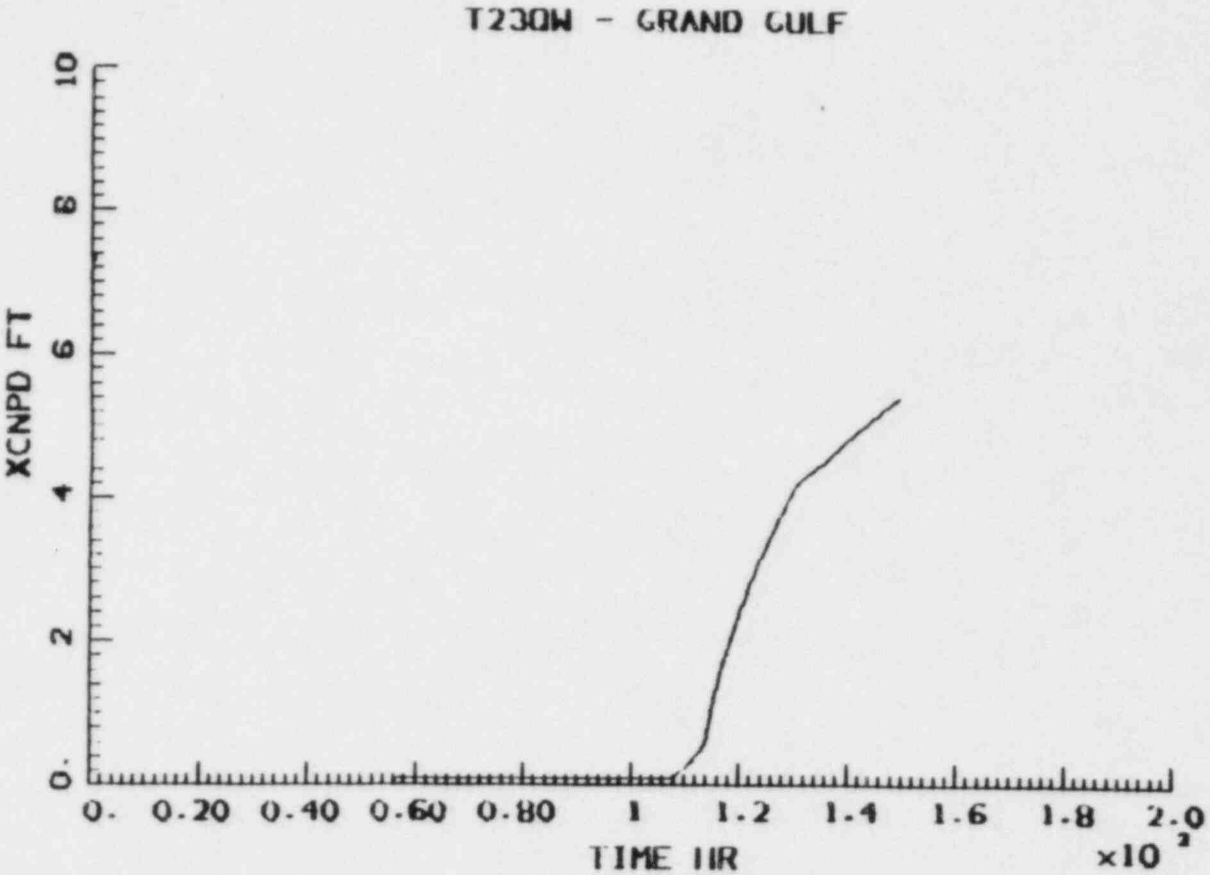

Fig. 4.19 Concrete ablation depth in the pedestal.

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Table 4.5

GRAND GULF NUCLEAR STATION

$T_{23}C$ BASE CASE

ACCIDENT CHRONOLOGY

Time	Event
0 sec	Initiating events: MSIV closures; Failure to scram; Loss of main feedwater
33 sec	RPV Level 2 LOCA setpoint reached
49 sec	HPCS begins operating
52 sec	RCIC begins operating
4.5 min	HPCS/RCIC systems transfer suction from CST to SP
8 min	ADS manually initiated
18.3 min	RCIC pump fails on high suction temperature
23.0 min	High DW pressure LOCA setpoint reached; Post-LOCA DW vacuum breakers open
23.6 min	Drywell purge system actuates
23.8 min	LPCS and LPCI actuate
26.2 min	High wetwell pressure setpoint reached
33.8 min	Containment sprays actuate
53.1 min	SPMU actuates
1.0 hr	Containment failure and subsequent ECCS failure
1.3 hr	Core begins to uncover
3.0 hr	Fuel melting begins
3.8 hr	Core plate failure followed by vessel failure

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4.4.2 Primary System and Containment Response

The MSIV closures are modeled to actuate a reactor scram which fails to insert the control rods into the core. Despite this failure to scram, the core power is modeled to decrease from its initial full-power level to about 20% of full power level within seconds. This power reduction simulates the thermal-hydraulic reactivity feedback effects which are expected to occur as a result of the initiating MSIV closure event, the resultant recirculation and feedwater trips, and the ensuing high pressure core spray (HPCS) and reactor core isolation cooling (RCIC) systems actuations. The estimate of 20% of full power is based on the assumption that the core power will equilibrate at a level which just equals the power needed to boil all incoming coolant flow. In addition, core power is assumed to linearly decrease from 18% to 6% of full power as the downcomer water level decreases from 7.2 ft above the active core to the top of the jet pumps. Decay heat power levels are assumed for uncovered fuel nodes. The $T_{23}C$ core power history is provided in Fig. 4.20.

The MSIV closures cause a reactor pressure vessel (RPV) pressure excursion which is relieved by the SRVs. The vessel remains at the SRV relief setpoint pressure. The exiting RPV steam is routed to the suppression pool (SP), where it is quenched. By 33 sec into the event, sufficient RPV water inventory has been lost through the cycling SRVs to drop the RPV water level to the RPV Level 2 LOCA setpoint. At that point, signals are automatically generated to actuate the HPCS and RCIC systems. The HPCS begins injecting water into the RPV at 49 sec; the RCIC begins at 52 sec. These systems maintain RPV inventory between RPV Levels 2 and 8. At 4.5 min, suction for these systems is transferred from the condensate storage tank (CST) to the SP on a high SP water level signal. At 8 min, when the suppression pool temperature reaches 145°F, the RPV is manually depressurized according to emergency procedure guidelines. Because the core power generation rate is much greater than the decay heat level, the SP water temperature rises very rapidly. When the SP temperature reaches 200°F at 18.3 min, the RCIC pump is assumed to fail due to high bearing temperatures. The HPCS is unable to maintain sufficient RPV inventory at SRV setpoint pressures and at a 20% of full power level. As a result, the RPV water level decrease to a new equilibrium state. These can be seen in Fig. 4.21.

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T23C - GRAND GULF

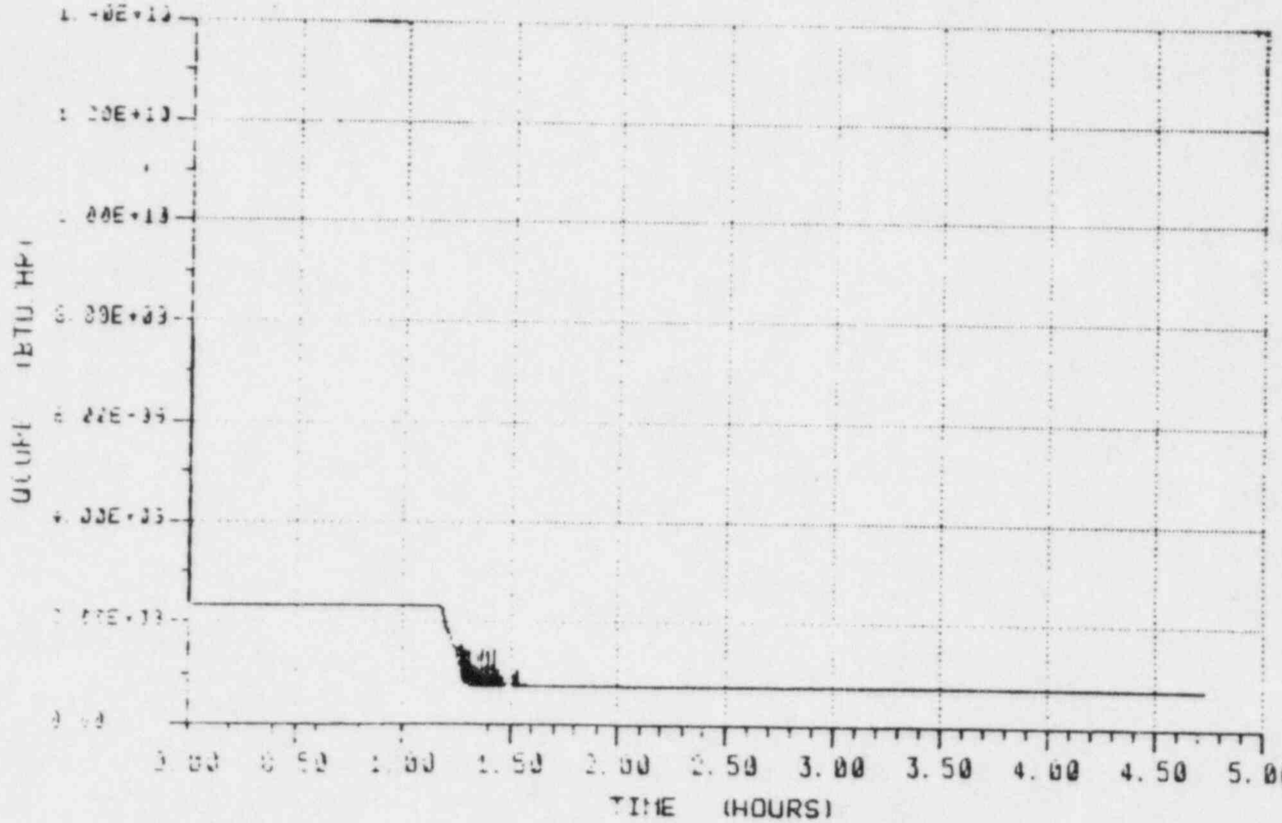

Fig. 4.20 Average core power.

T23C - GRAND GULF

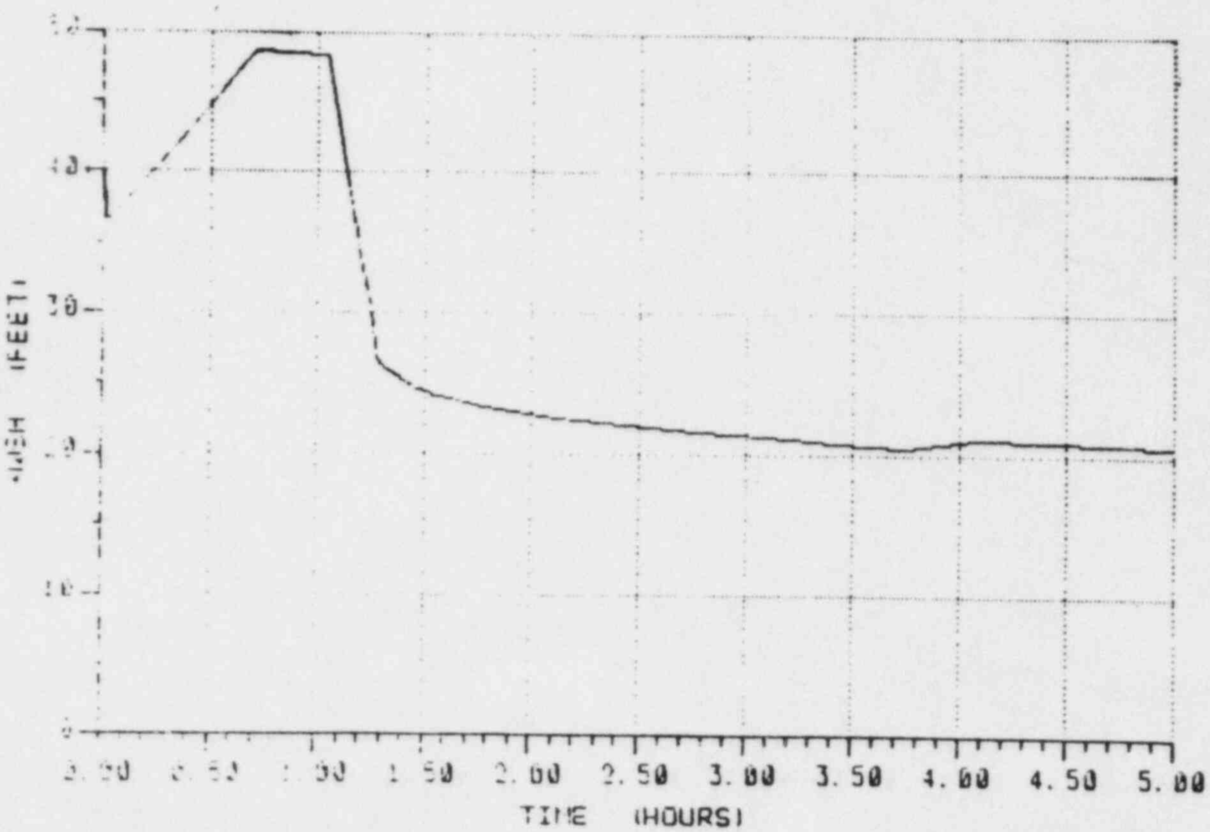

Fig. 4.21 Reactor vessel water level.

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The SP reaches saturation conditions and is no longer able to completely quench the steam exiting the RPV through the cycling SRVs; a steam-pressurization of the containment ensues. The rising suppression pool water temperature and the resulting rise in pressures and temperatures in both the drywell and outer containment can be seen in Figs. 4.22 through 4.25. The rising pressure actuates the 1.89 psig high DW pressure LOCA signal at 23.0 min. This signal is a permissive signal for the DW purge system, the post-LOCA DW vacuum breakers, and the SP makeup (SPMU) system; it is an actuation signal for the low pressure core spray (LPCS) and low pressure coolant injection (LPCI) systems. Since the post-LOCA DW vacuum breaker permissive requiring a 0.87 psid drywell vacuum relative to the wetwell is already satisfied, the vacuum breakers open immediately. The DW purge system actuates after a 30 sec time delay and the SPMU system actuates the upper containment pool dump following a programmed 30 min delay. The continuing HPCS injection maintains RPV level. At 26.2 min into the event, the containment pressure reaches the 9 psig containment spray actuation pressure setpoint. At that time, trains A and B of the residual heat removal (RHR) system automatically switch into their spray mode and eight minutes later begin to spray SP water into the upper containment volume. Since the containment spray water cooling requires manual alignment, which was not modeled in this analysis, the containment spray system is unable to effect a containment pressure reduction.

At 53.1 min into the event, the SPMU system releases, as designed, approximately half of the upper containment pool volume into the suppression pool. This action brings the suppression pool to a subcooled state. Consequently, the containment steam pressurization ceases and, in fact, reverses. The former is due to the renewed ability of the suppression pool to quench the SRV steam discharge. The rapid outer containment depressurization is due to the action of the containment sprays which draw suction from the suppression pool. Within 15 minutes of the upper pool release, the continued core power generation reheats the suppression pool to a saturated state and outer containment pressurization resumes. The additional pool inventory begins to spill onto the drywell and pedestal floors at that time. This spill has a large mitigative effect if this accident proceeds beyond vessel failure. At 1.0 hrs into the accident, only minutes after the renewed pressurization, the containment is modeled to fail at this pressure at a failure location just

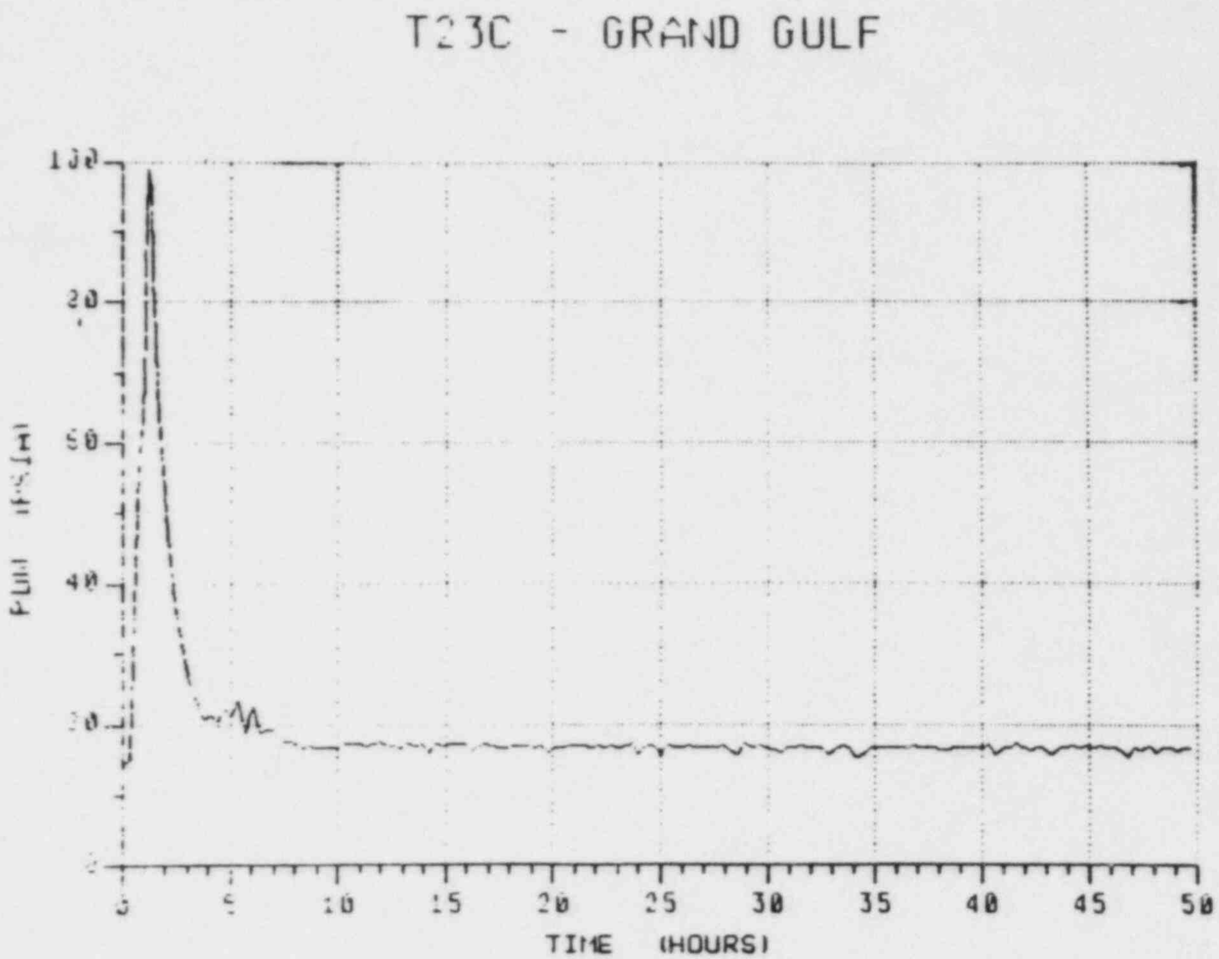

Fig. 4.22 Pressure in the drywell.

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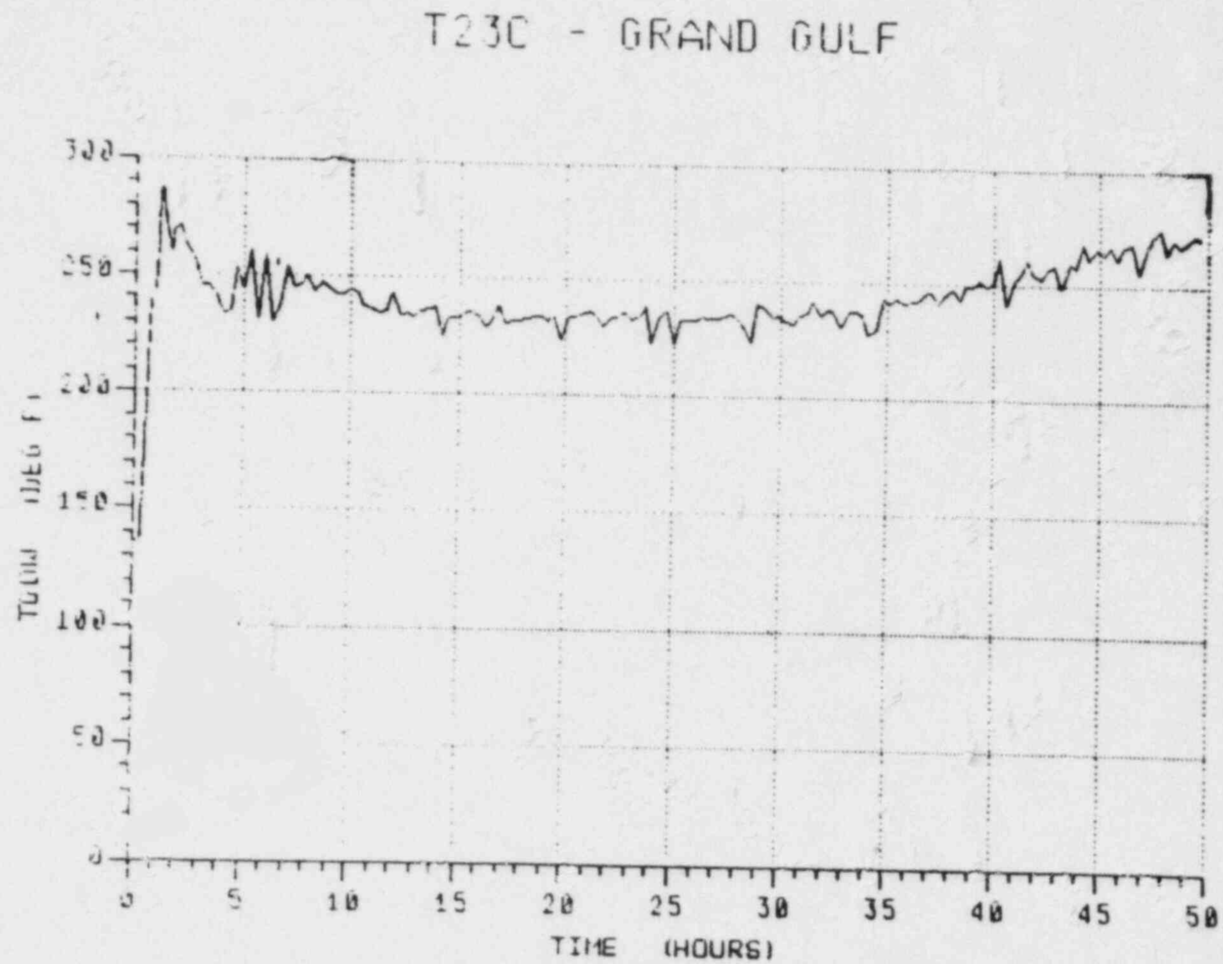

Fig. 4.23 Temperature of gas in the drywell.

T23C - GRAND GULF

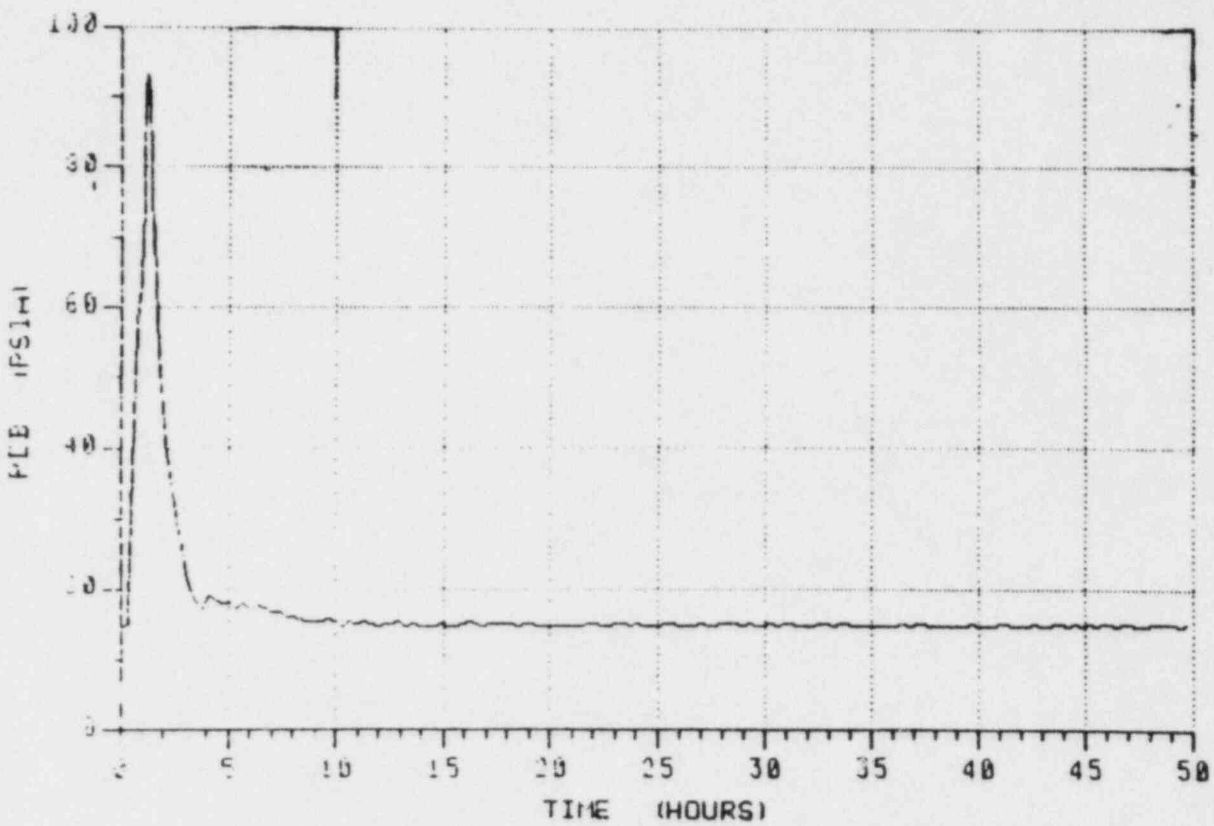

Fig. 4.24 Pressure in Compartment B.

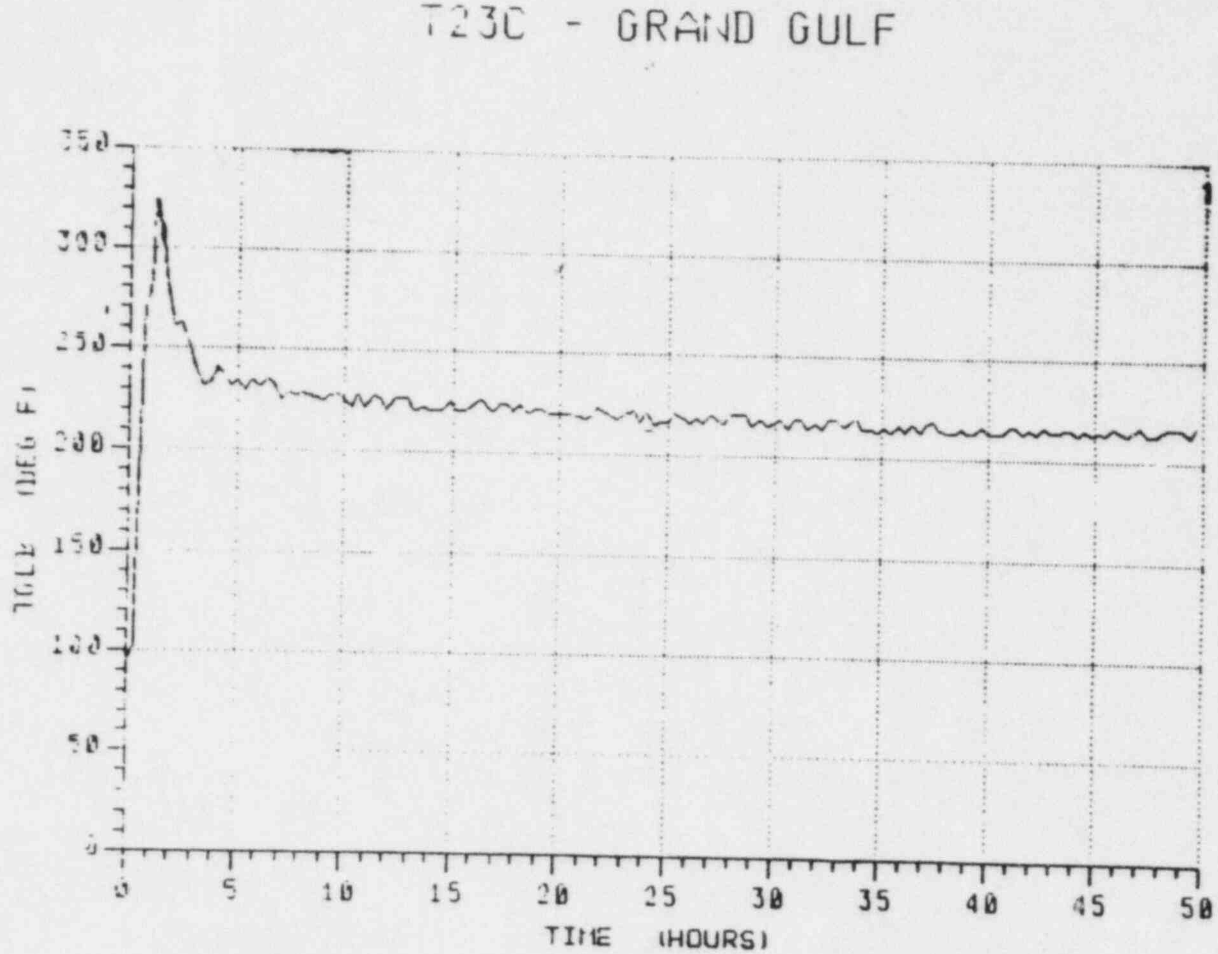

Fig. 4.25 Temperature of gas in Compartment B.

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below the junction between the cylinder and the dome. The failure cause is steam overpressurization. A containment breach of 1.5 ft^2 was modeled.

In order for the $T_{23}C$ sequence to result in core damage, it is necessary that all systems supplying or capable of supplying water to the RPV fail at or before containment failure. A realistic mechanism which could cause such a simultaneous failure has not been identified. The accounting of containment failure location, pressure, fluid flow loading, and ECCS pump suction temperature, pressure, and NPSH limitations indicates that at least one GGNS ECCS train should survive a containment failure event. However, for this analysis, the conservative assumption that all ECCS equipment fails on containment failure was made. The CRD flow was assumed to continue, at the rate of approximately 90 gpm.

Given that all ECCS fail on containment failure, the RPV water level begins to fall sharply as shown in Fig. 4.21. As the water level continues to fall, the power level decreases to 6% of full power. As a fuel node is uncovered, its power level is modeled to decrease to its decay heat level.

Fuel temperatures in the uncovered regions of the core begin rising above 2000°F at about 1.9 hr. The oxidation of the Zircaloy fuel cladding by steam increases rapidly above the 2000°F point. About 530 lb of hydrogen is produced in the vessel.

Fuel melting is predicted to begin at 3.0 hr. After melting, fuel moves from the core to the core plate. By 3.8 hr, sufficient core material is calculated to have fallen onto the RPV core plate to cause it to fail. The core debris then falls to the bottom of the RPV; shortly thereafter, the vessel fails at a welded penetration. At vessel failure, the molten fraction of the lower plenum core debris falls onto the pedestal floor followed by the lower plenum water.

Since the vessel had been depressurized previously, the debris does not disperse from the pedestal to the drywell upon vessel failure. Furthermore, the remainder of the core material gradually enters the pedestal from the vessel and also stays in the pedestal. The debris attacks the pedestal concrete as it is being quenched (see Fig. 4.26) until about three inches of concrete have been ablated. Once the core debris bed in the pedestal is cooled to below concrete ablation temperatures by the lower plenum water, it remains quenched since its blanket of water is boiled away. As can be seen from Fig. 4.27, this would not occur for a very long time, if ever. Consequently, no appreciable quantities of noncondensable gases are generated.

Subsequent to vessel failure steam flows steadily from the pedestal, to the drywell, to the suppression pool at a rate of roughly 2×10^6 ft^3/hr. The flow is due to the fact that the CRD water is continuing to quench the debris in the pedestal, and producing steam.

No hydrogen burning was predicted to occur in this sequence. By the time the hydrogen produced from Zircaloy oxidation in the core reached the wetwell, all of the oxygen had been depleted from the wetwell atmosphere, as well as from the upper containment atmosphere. Furthermore, there are no appreciable quantities of hydrogen or carbon monoxide generated from core debris-concrete attack. Appendix B includes additional plots of results for this sequence.

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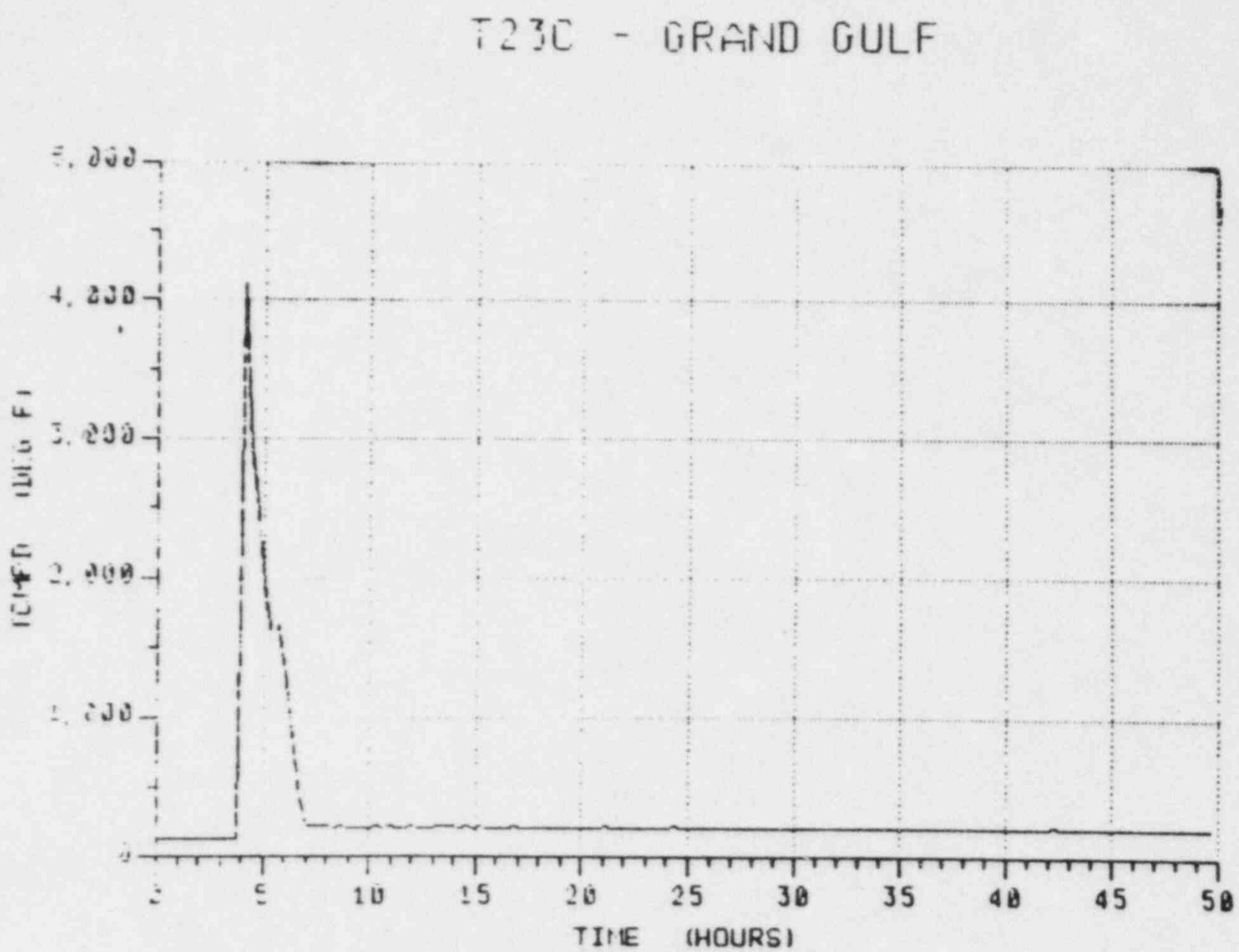

Fig. 4.26 Average corium temperature in the pedestal.

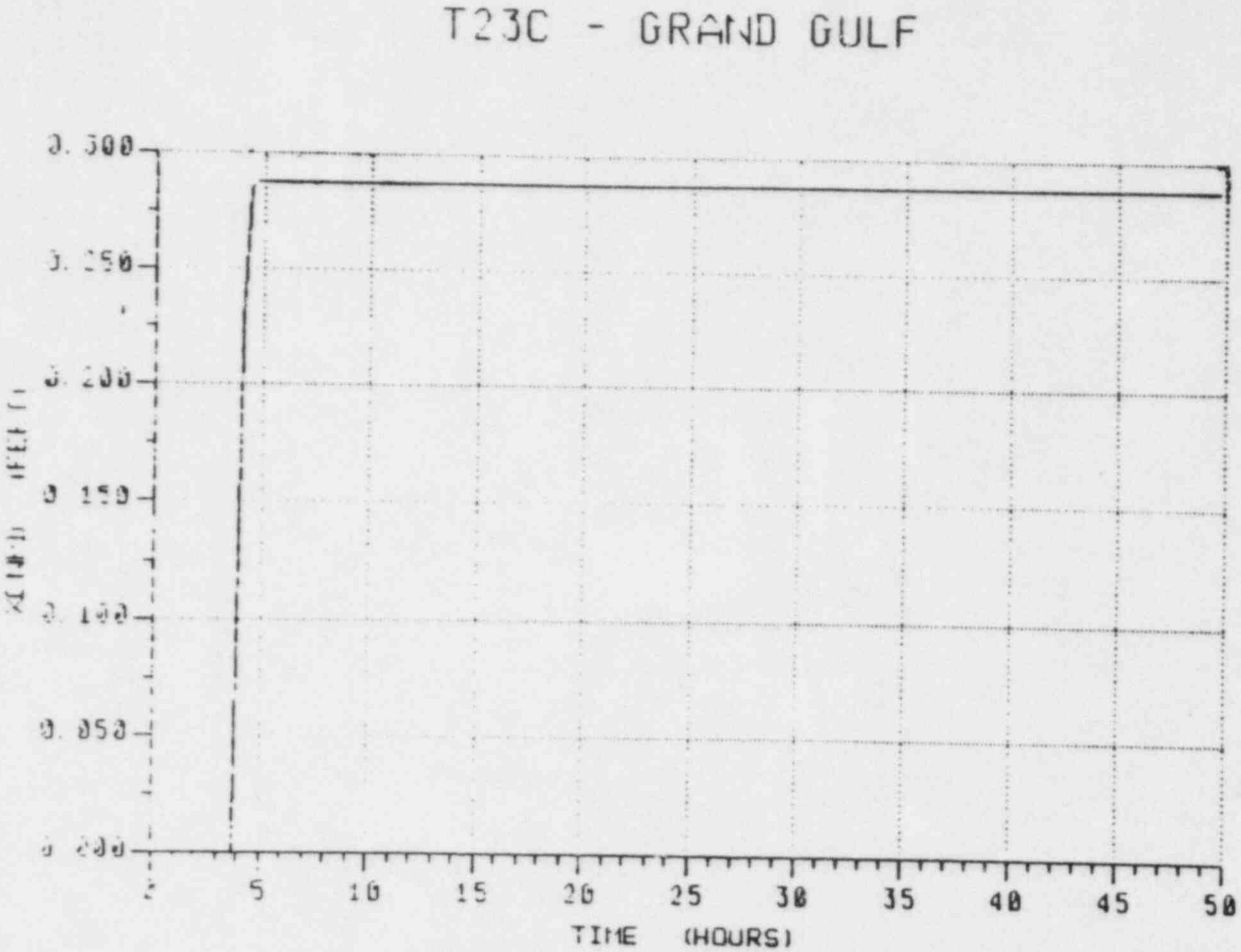

Fig. 4.27 Concrete ablation depth in the pedestal.

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Grand Gulf Section 5 to be supplied later.

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6.0 FISSION PRODUCT RELEASE, TRANSPORT AND DEPOSITION

6.1 Introduction

The phenomena of fission product release from the fuel matrix, its transport within the primary system, their release from the primary system into the containment, their deposition within the containment and the subsequent release of some fission products from the containment are treated through the use of MAAP [6.1]. Release of fission products from the fuel matrix and their transport to the top of the core are treated by a subroutine in MAAP which is based on the FPRAT code [6.2]. Transport of fission products outside the core boundaries is determined by the natural and forced convection flows modeled in MAAP with the gravitational sedimentation described in Ref. [6.3] and other deposition processes described in Ref. [6.4]. Fission product behavior is considered for the best estimate transport, deposition and relocation processes. Influence of surface reactions between chemically active substances like cesium hydroxide and other uncertainties are considered in subtask 23.4. The best estimate calculation, assuming cesium iodide and cesium hydroxide are the chemical state of cesium and iodine, is discussed below.

6.2 Modeling Approach

Evaluations of the dominant chemical species in Ref. [6.5] show the states of the radionuclides (excluding noble gases) which dominate the public health risk to be cesium iodide and cesium hydroxide, tellurium oxide and strontium oxide. These and others are considered in the code when calculating the release of fission products from the fuel matrix. Vapors of these dominant species form dense aerosol clouds in the upper plenum, in some cases approaching 100 g/m^3 for a very short time, which agglomerate and settle onto surfaces. Depending upon the chemical compound and gas temperature, these deposited aerosols can be either solid or liquid. At the time of reactor vessel failure, some material remains suspended as airborne aerosol or vapor and would be discharged from the primary system into the containment. The rate of discharge is determined by the gaseous flow between the primary system and containment which is sequence specific. (It should be noted that some

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fission products can be discharged into the containment before vessel failure through relief valves or through breaks in the primary system. This is also sequence specific.) This set of inter-related processes are treated in MAAP and essentially result in a release of all airborne aerosol and vapor from the primary system into containment immediately following vessel failure.

As a result of the dense aerosols formed when fission products are released from the fuel, considerable deposition occurs within the primary system prior to vessel failure. For some accident sequences, the primary system may be at an elevated pressure at the time of core slump and reactor vessel failure. Resuspension of these aerosol deposits during the primary system blowdown is assessed in Ref. [6.6] in terms of the available experimental results and basic models. It is concluded that resuspension immediately following reactor vessel failure would not be significant, less than 1% of the deposited materials, even for depressurizations initiated from the nominal operating pressure. For delayed containment failure, this small fraction of material is depleted by in-containment mechanisms.

Therefore, a major fraction of the volatile fission products are retained within the primary system following vessel failure, the distribution being determined by the MAAP calculations prior to vessel failure. Natural circulation through the primary system after vessel failure is analyzed using MAAP which allows for heat and mass transport in various nodes of the reactor vessel and the steam generators including heat losses from the primary system as dictated by the reflective insulation. Material transport is due to aerosols and vapors as governed by the heatup of structures due to radioactive decay of deposited fission products. This heatup is principally determined by the transport of cesium iodide and cesium hydroxide by the natural circulation flows. In this regard, the vapor pressure of cesium hydroxide is applied to both the cesium iodide and cesium hydroxide chemical species. In essence, this assumes that the solution of cesium iodide and cesium hydroxide has a vapor pressure close to that of cesium hydroxide, which is a conservatism in the calculations. In carrying out these calculations, the pressurization of the primary system is dependent upon the pressurization of the containment and the heating within the primary system. These determine the in- and out-flows between the primary system and containment.

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Deposition within the containment is calculated using thermal hydraulic conditions determined by MAAP. The major aerosol sources are the releases prior to vessel failure (sequence specific), the airborne aerosols and vapors transferred from the primary system at the time of vessel failure, the subsequent releases from the primary system due to long term heatup, and concrete attack. At the time of containment failure, the remaining airborne aerosol and vapor can be released to the environment. Assessments of the potential for resuspension of deposited aerosols following containment failure [6.6] show this to be negligible.

6.3 Sequences Evaluated

The use of MAAP in the manner indicated above leads to the release fractions shown in Tables 6.1 through 6.5. Four sequences are analyzed, including: transient with failure of injection (T_1QUV); large LOCA with failure of injection (AE); transient followed by loss of containment heat removal (T_{23}QW); and transient with failure to scram (T_{23}C). Thermal-hydraulic behavior for these sequences is described in Section 4. In this section it is shown that, for T_{23}QW and T_{23}C, the containment fails before the core is uncovered. Hence, the cesium and iodine are still in the fuel matrix.

6.3.1 T_1QUV Sequence

As indicated in Table 6.1, two percent of the volatile fission product inventory is swept from the vessel to the suppression pool via the SRV lines prior to vessel failure. Of the remainder, 2% is still in the fuel matrix, 95% is in the upper plenum area, 1% is in the downcomer.

During the time between vessel breach and containment failure, revaporization and relocation of material within the primary system occurs, due to the continuing natural circulation flows. Some material continually flows to the pedestal and drywell as vapor, and from there some of the material flows to the suppression pool. After about a day, the drywell is hot enough that revaporization begins there, and flow to the suppression pool is increased. The pool itself is highly effective in scrubbing the fission

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Table 6.1

DISTRIBUTION OF CsI IN PLANT AND ENVIRONMENT
(FRACTION OF CORE INVENTORY)

	At Vessel Failure			
	$T_{23}QW$	$T_{23}C$	AE	T_1QUV
RPV	.90	.68	.98	.98
Drywell	0.0	0.0	.02	0.0
Suppression Pool	.10	.32	0.0	.02
Primary Containment	5.3×10^{-5}	2.2×10^{-5}	0.0	0.0
Environment	3.2×10^{-5}	2.6×10^{-4}	0.0	0.0

	At Containment Failure			
	$T_{23}QW$	$T_{23}C$	AE	T_1QUV
RPV	1.00	1.00	.91	.46
Drywell	0.0	0.0	.03	.20
Suppression Pool	0.0	0.0	.06	.34
Primary Containment	0.0	0.0	0.0	0.0
Environment	0.0	0.0	0.0	0.0

	Ultimate Distribution			
	$T_{23}QW$	$T_{23}C$	AE	T_1QUV
RPV	.50	.26	.90	.33
Drywell	.12	.05	.03	.02
Suppression Pool	.38	.69	.07	.645
Primary Containment	2.1×10^{-4}	1.1×10^{-4}	5.11×10^{-4}	7.3×10^{-4}
Environment	2.6×10^{-4}	7.6×10^{-4}	$< 1 \times 10^{-5}$	7.3×10^{-5}

DRAFT

Table 6.2

T_1QUV FISSION PRODUCT RELEASE

Assumptions
Containment Failure Location - Compartment B, 237' 9"
Containment Failure Size - .1 ft^2

Fission Product Group	Release Fraction to Environment
Cs, I	7.3×10^{-5}
Te, Sb	3.2×10^{-5}
Sr, Ba	$< 1 \times 10^{-5}$
Ru, Mo	$< 1 \times 10^{-5}$

DRAFT

Table 6.3

AE FISSION PRODUCT RELEASE

Assumptions
Containment Failure Location - Compartment B, 237' 9"
Containment Failure Size - .1 ft^2

Fission Product Group	Release Fraction to Environment
Cs, I	$< 1 \times 10^{-5}$
Te, Sb	1.1×10^{-5}
Sr, Ba	$< 1 \times 10^{-5}$
Ru, Mo	$< 1 \times 10^{-5}$

DRAFT

Table 6.4

$T_{23}QW$ FISSION PRODUCT RELEASE

Assumptions
Containment Failure Location - Compartment B, 237' 9"   Containment Failure Size - .1 ft^2

Fission Product Group	Release Fraction to Environment
Cs, I	2.6×10^{-4}
Te, Sb	2.2×10^{-4}
Sr, Ba	$< 1 \times 10^{-5}$
Ru, Mo	$< 1 \times 10^{-5}$

DRAFT

Table 6.5

$T_{23}C$ FISSION PRODUCT RELEASE

Assumptions
Containment Failure Location - Compartment B, 237' 9"
Containment Failure Size - 1.5 ft^2

Fission Product Group	Release Fraction to Environment
Cs, I	7.6×10^{-4}
Te, Sb	7.5×10^{-4}
Sr, Ba	$< 1 \times 10^{-5}$
Ru, Mo	$< 1 \times 10^{-5}$

products. A decontamination factor of 600 is associated with passage from the drywell to the pool through the vents [6.7].

Table 6.1 also shows the volatile fission product inventories in the various compartments at the time of containment failure. Only the airborne material in the upper compartment and that portion of the material still to be revolatized in the vessel that would not be scrubbed in the suppression pool is available for release to the environment. As can be seen in Table 6.2, the release fractions to the environment for this case are low. Long term releases subsequent to containment failure occur but at extremely slow rates.

Considerable concrete ablation takes place in the pedestal following vessel failure and subsequent flowing of molten core debris into the pedestal. By 24 hr the ablation depth is more than 5 ft.

6.3.2 AE Sequence

The use of MAAP leads to the release fractions shown in Tables 6.1 and 6.3. The thermal-hydraulic analysis is described in Section 4.2.

Table 6.1 shows the distribution of cesium and iodine through the various regions, at vessel failure and 70 hr, when the calculation was terminated. Due to the very low steam flow in the vessel after the initial LOCA blowdown, nearly all of the material is initially deposited in the upper plenum. Hence, very little material enters the suppression pool through the break (less than 1 kg by the time of vessel breach). At the time of vessel breach, only about 1 kg is airborne. This material can leave the vessel. The deposited material (about 229 kg) remains in the vessel at this time.

Following vessel failure, the remainder of the volatile fission products are released from the fuel as it melts. This material, along with that already deposited, moves around the vessel, being deposited, heating up, revaporizing, moving to cooler regions and redepositing, etc. Drywell pressurization from the very hot gases in the pedestal cavity prevents materials from escaping the vessel until containment failure at 58 hr. As can be inferred from Table 6.1 about 1% of cesium and iodine are relocated from the

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vessel to the suppression pool during the period following containment failure. Of this, only one part in 600 escapes the pool to the outer containment [6.7].

Release fractions to the environment are very low, as can be seen in Table 6.3. As for the T_1QUV sequence, however considerable concrete ablation occurs, although it does not occur for the first 30 hr of the event. By 50 hr the ablation depth is approximately 5 ft.

6.3.3 T_{23}QW Sequence

The use of MAAP leads to the release fractions shown in Tables 6.1 and 6.4. The thermal-hydraulic analysis was described in Section 4.3.

Table 6.1 shows the distribution of the volatile fission products (cesium and iodine) through the various regions, at vessel failure and at 150 hr when the calculation was terminated. At vessel failure, nearly all of the volatiles (90%) in the vessel are deposited in the upper structures. The remainder (10%) are in the suppression pool. Only negligible quantities are present elsewhere. The decontamination factor associated with passage through the SRVs and spargers, and subsequent pool scrubbing, is 1000 [6.7].

Since the containment is already failed prior to core uncovery there is no rapid depressurization as in the T_1QUV and AE sequences. Furthermore, there is no large scale concrete attack in the pedestal. Thus the ultimate fission product distribution is such that the release to the environment is very small, as indicated in Table 6.4.

6.3.4 T_{23}C Sequence

The use of MAAP leads to the release fractions shown in Tables 6.1 and 6.5. The MAAP thermal-hydraulic analysis is described in Section 4.4.

Table 6.1 shows the distribution of cesium and iodine through the various regions both at vessel failure and at 50 hr, when the calculation was terminated. At vessel failure 139 kg are deposited in the upper plenum, 10 kg

DRAFT

are in the downcomer, 14 kg are in the core region, and 76 kg have left the vessel through the SRVs to the suppression pool. Only negligible quantities are present elsewhere. The decontamination factor associated with passage through the SRVs and spargers is 1000 [6.7].

The fission products tend not to exit the vessel but rather transfer their heat to gas and structures and move about the primary system. The reflective insulation is very effective in transferring a considerable portion of the heat to the drywell as temperatures rise.

Since the containment is already failed prior to core uncovery there is no rapid depressurization. Furthermore, there is no large scale concrete attack in the pedestal. Thus the ultimate fission product distribution is such that the release to the environment is very small, as indicated in Table 6.5.

6.4 References

6.1 MAAP - Modular Accident Analysis Program, User's Manual, August, 1983.

6.2 IDCOR Technical Report 15.1B, "Analysis of In-Vessel Core Melt Progression," Vol. IV (User's Manual) and Modeling Details for the Fission Product Release and Transport Code (FPRAT), September, 1983.

6.3 Draft IDCOR Technical Report, "FAI Aerosol Correlation," July, 1984.

6.4 IDCOR Technical Report on Task 11.3, "Fission Product Transport in Degraded Core Accidents," December, 1983.

6.5 IDCOR Technical Report on Tasks 11.1, 11.4 and 11.5, "Estimation of Fission Product and Core-Material Source Characteristics," October, 1982.

6.6 IDCOR Technical Report on Task 11.6, "Resuspension of Deposited Aerosols Following Primary System or Containment Failure," July, 1984.

6.7 K. Holtzclaw, Personal Communication, 1984.

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7.0 SUMMARY OF RESULTS

As outlined in Section 2 of this report, the IDCOR Subtask 23.1 Integrated Containment Analysis of the Grand Gulf Nuclear Station (GGNS) consisted of base case accident analyses and operator action case accident analyses. The accident sequences selected for analysis represent a majority of previously-assessed risk and demonstrate a variety of initiating events, a variety of system failures combinations, and a diversity of accident phenomenology. The primary system and containment thermal-hydraulic response analyses and fission product transport were performed via the MAAP code. Fission product release was performed via the FPRAT code which has been integrated into MAAP. Detailed descriptions of each of these analyses are provided in Sections 4 through 7 of this report, respectively. This section of the report summarizes the major results of each of these analyses.

7.1 Base Case Analyses

The base case analyses establish a reference system response during these accidents by assuming a minimum of operator intervention during the accident progression. As such, these analyses do not realistically account for the mitigative response of the trained operating staff and, thus, should not be considered as representative of realistic plant response analyses. The base case fission product transport results are summarized on Table 7.1. A discussion of these results follows.

Accidents involving demand-type failures of all automatically-actuated high and low pressure reactor pressure vessel (RPV) makeup systems, namely those accident sequences containing events UV or E, result in core damage unless an appropriate operator response is taken. For accidents which involve relatively small RPV coolant inventory loss rates and decay power levels, such as T_1QUV and T_{23}PQE, the core is predicted to begin to uncover within about half an hour of the initiating event. Within about one hour, significant fuel cladding degradation is predicted, and fuel melting is calculated to begin about two hours after the initiating event. Vessel

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Table 7.1

SUMMARY OF FRACTIONAL RADIONUCLIDE RELEASES TO THE ENVIRONMENT

Accident	Fission Product Group				
	Xe and Kr	Cs and I	Te	Sr and Ba	Ru and Mo
T_1QUV	1.0	7.3 E-5	3.2 E-5	$< 1 \times 10^{-5}$	$< 1 \times 10^{-5}$
AE	1.0	$< 1 \times 10^{-5}$	1.1 E-5	$< 1 \times 10^{-5}$	$< 1 \times 10^{-5}$
$T_{23}C$	1.0	7.6 E-4	7.5 E-4	$< 1 \times 10^{-5}$	$< 1 \times 10^{-5}$
$T_{23}QW$	1.0	2.6 E-4	2.2 E-4	$< 1 \times 10^{-5}$	$< 1 \times 10^{-5}$
BWR-4	0.6	5.0 E-3*	4.0 E-3	6.0 E-4	6.0 E-4

*Iodine release fraction is 0.8 E-4.
Cesium release fraction is 5.0 E-3.

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failure will follow within another half-hour. For accidents with large RPV inventory loss rates, such as AE, these events occur sooner. For the large-break LOCA case analyzed, the AE accident, fuel melting was predicted to occur within 0.7 hours of the initiating event and was closely followed by vessel failure.

Accidents involving successful RPV makeup but inadequate containment cooling, such as $T_{23}QW$ and $T_{23}C$, will result in containment failure unless appropriate operator action is taken. Previous studies have postulated that all ECCS injection into the RPV will fail on containment failure. With this assumption, and without appropriate operator action, fuel melting will inevitably follow. The results of this study indicate that the assumption that all ECCS equipment fails on containment failure has no mechanistic basis and thus is extremely conservative. Without the containment-failure-induced ECCS failure assumption, many of the previously-postulated dominant GGNS accidents sequences do not lead to core melt and, thus, can no longer be considered risk significant. The $T_{23}QW$ and $T_{23}C$ sequences are all among these accidents.

The mass of hydrogen produced via steam oxidation of fuel cladding in the core was calculated to be significantly lower than that prescribed by the NRC for interim rule on hydrogen control for Mark III containments. The MAAP predictions demonstrate that less than about 10% fuel cladding oxidation prior to fuel melting for severe GGNS accidents. The NRC cladding oxidation rule specifies a 75% cladding reaction. Even if the accidents were to progress unmitigated to vessel failure, the maximum fraction of cladding oxidized is predicted at only 35%. Judicious misaction is necessary to generate cladding reactions of higher magnitudes. Specifically, a low vessel makeup flow or an orchestrated termination and restart of emergency core cooling would be necessary. The rate of hydrogen production calculated for the GGNS severe accident analyses is also substantially lower than those used in previous studies. The maximum average sustained rate observed in the MAAP calculations was less than 0.5 lb/sec lasting for about less than twenty minutes.

For accidents which proceed beyond vessel failure, the molten core debris is calculated to fall onto the pedestal floor. No core debris is

DRAFT

calculated to exit the pedestal volume. Thus, concrete attack is limited to the pedestal floor and walls. Without core-debris cooling, substantial erosion of the pedestal floor and walls is calculated to occur.

Three containment failure modes were observed in the GGNS Mark III containment analysis. They were overpressurization by steam, by noncondensable gases, and/or by hydrogen combustion. The dominant failure mode was found to be accident dependent. All three modes result in long-delayed containment failure events for the GGNS accidents analyzed, the MAAP code predicts no steam explosions large enough to fail either the reactor pressure vessel or the containment. Thus, no prompt containment failures were observed. It is noteworthy to state that the containment failure times predicted in this study are long compared to those of previous studies. This is primarily due to the higher ultimate containment capacity (56.6 psig) used in this study.

For the GGNS Mark III design, the suppression pool was observed to the exert a dominant influence on the accident progression. There are a number of reasons that the suppression pool displays this behavior. First, overpressurization of the containment by steam can occur only if the suppression pool is heated to high temperatures or if the suppression pool is by-passed. The former requires a substantial energy deposition and inadequate suppression pool heat removal. The latter has been evaluated to be a very low probability occurrence. Secondly, the suppression pool controls the temperature of the noncondensable gases which are calculated to be evolved in sequences heading to core degradation, core melting and core-concrete attack. By cooling these gases, as they enter the outer containment volume, the suppression pool substantially slows the rate of pressurization within the containment building. Thirdly, for accident sequences which have proceeded past vessel failure, the suppression pool water can, in general, be supplied to the debris to provide either temporary or potentially long term debris bed cooling. Lastly, it is significant to recognize that the suppression pool can retain substantial quantities of noninert fission product material which would be released by the fuel during a core meltdown event. With the location of the suppression pool in the Mark III design, these materials cannot be exhausted through a containment breach without first being highly decontaminated by the suppression pool.

Fission product release and transport calculations were performed with FPRAT and MAAP for the T_1QUV, $T_{23}QW$, AE, and $T_{23}C$ base case sequences. A summary of the final airborne fission product releases to the environment for the accident sequences analyzed are presented in Table 7.1. The BWR-4 release category from the Reactor Safety Study is also presented for comparison. The data presented on this table shows that for the accidents analyzed the fractional fission product releases to the environment were generally significantly less severe than those associated with the BWR-4 release category. Since the accidents analyzed represent a majority of public health risk, the present analysis indicates that the risk associated with the operation of GGNS is substantially lower than that previously assessed.

The lower fission product release terms produced in this study as compared to previous studies are principally due to the higher suppression pool decontamination factor and the relatively late containment failure time. Other factors which were found to influence the amount of fission product escaping the containment system during the severe accident scenarios analyzed were the duration of the melt releases, the time of the vessel failure, the fission product transport pathway, and the assumed fraction of fission product resuspension at the time of containment failure. A specific finding of these analyses is that accidents which involve rapid core heatups or which display a high RPV pressure until the vessel failure result in rapid releases of volatile fission products from the fuel immediately after the vessel fails. Another finding is that nonvolatile fission product release rates due to core-concrete interaction are small beyond about 20 hours after vessel failure. Lastly, the majority of fission product retention was calculated to occur in the suppression pool and in the drywell.

7.2 Operator Action Analyses

The major results of the operator action case thermal-hydraulics analyses are summarized in Section 5. They demonstrate that a safe stable state can be achieved in the vessel if injection can be restored prior to core plate failure. There are many means available to the operator for providing sufficient makeup flow to the reactor vessel. The time available for aligning and actuating these RPV makeup systems prior to core damage and/or fuel melting was evaluated in the base case analyses to be accident dependent. Once actuated, the operator case analyses indicate that these systems are capable of reflooding the core within minutes. These analyses also demonstrate that given the existence of a safe stable state for the core, a safe stable state for the containment can be achieved by restoring adequate containment cooling. Peak containment temperatures and pressures occur from minutes to hours after such restoration, depending on the core heat level and on the mode and magnitude of containment heat removal.

Debris coolability and the maintenance of containment integrity was demonstrated as possible via the restoration of an emergency core cooling system to flood the pedestal and a containment cooling system to cool the suppression pool.

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8.0 CONCLUSIONS

Based on the results of the severe accident analyses performed in this study, a number of conclusions can be drawn regarding the progression and consequences of such severe accidents for plant designs similar to that of the Grand Gulf Nuclear Station.

The analytical tools employed in this study, namely MAAP, is a viable means of analyzing both the thermal-hydraulic and the radiological response of the Grand Gulf Nuclear Station primary system and containment to severe accident scenarios.

The most significant conclusions which can be drawn from this integrated containment analysis of the Grand Gulf Nuclear Station are itemized below. The first refers to the analytical tools used in this study. The next set are thermal-hydraulic related conclusions. And, the last and probably most significant conclusion relates to the radiological results of this study.

- The MAAP code is a viable means of analyzing both the thermal-hydraulic and the radiological response of the Grand Gulf Nuclear Station primary system and containment to severe accident scenarios.

- For accidents postulated to lead to core damage, fuel melting, and/or containment failure, there are sufficient time and means available to the operating staff to place the plant into a safe stable state.

- Containment failure should no longer be considered a cause for the failure of all ECCS flow to the reactor vessel. Thus, containment failure should no longer be considered a cause for core melt.

- The mass and rate of hydrogen calculated to be produced in the vessel prior to fuel melting is substantially less than that predicted by previous studies.

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- If successful fuel cooling is delayed beyond the point of significant core damage and/or vessel failure, the core debris coolability is possible.

- The suppression pool exerts a dominant thermal-hydraulic and radiological influence on the containment response to a severe accident.

- The GGNS Mark III containment failure modes are overpressurization via steam, noncondensable gas generation, and/or hydrogen combustion. Containment failure times are long compared to previous studies. No prompt containment failures due to steam explosions or steam spiking were calculated.

- The overall containment response is much more sensitive to whether continuous hydrogen combustion occurs than to the details of how incomplete combustion progresses within the containment.

- Through continuous burning of the containment combustible gas, the GGNS containment hydrogen igniters can significantly delay containment failure during a severe accident.

- Decontamination of the fission product releases by the suppression pool and their condensation and gravitational settling in the drywell were found to be the two most important fission product removal mechanisms.

- The public health consequences of the severe accidents are substantially less than those of previous assessments.

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APPENDIX A

Grand Gulf Parameter File

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GULFFP.DAT;14                    6-JUL-1984 14:29               Page 1

**MARK III BWR PLANT PARAMETER VALUES-- TYPICAL OF GRAND GULF          !
**SI UNITS (M-KG-SEC-DEGK)                                             !
** 7-22-83                                                             !
**                                                                     !
*PRIMARY SYSTEM                                                       PS
01    13.521D0  AFLCOR FLOW AREA OF REACTOR CORE                      PS
** ALSH IS CALCULATED BY TAKING THE VOLUME OF WATER IN THE LOWER
** DOWNCOMER+JET PUMP AND DIVIDING BY (ZTOAF-ZBJET)
02    8.57D0    ALSH   FLOW AREA IN LOWER SHROUD                      PS
03    5.695D0   AFLBYP CORE BYPASS FLOW AREA                          PS
** AUSH IS CALCULATED BY TAKING THE VOLUME OF WATER IN THE UPPER
** DOWNCOMER ABOVE TOAF AND DIVIDING BY THE WATER HEIGHT ABOVE TOAF
04    2.644D1   AUSH   FLOW AREA IN UPPER SHROUD                      PS
05    1.116D5   HCRD   SPECIFIC ENTHALPY OF FLOW IN CRD TUBES         PS
06    9.248D5   HFW    SPECIFIC ENTHALPY OF FEEDWATER                 PS
07    1.65697D5 MU2PS  TOTAL MASS OF UO2 IN CORE                      PS
08    8.0D2     NASS   NUMBER OF FUEL ASSEMBLIES IN REACTOR CORE      PS
09    6.2D1     NPINS  NUMBER OF FUEL RODS IN A FUEL ASSEMBLY         PS
10    1.93D2    NCRD   NUMBER OF CRD TUBES                            PS
11    4.5D0     NQFPS  SENSIBLE ENERGY STORED IN FUEL (FULL POWER SECONDS)PS
12    4.0D0     TDMSIV DELAY TIME FOR MSIV CLOSURE                    PS
13    3.5D0     TDSCRM DELAY TIME FOR FULL SCRAM                      PS
14    5.976D7   TIRRAD TOTAL EFFECTIVE IRRADIATION TIME FOR CORE      PS
** ALL PUMP CURVES ASSOCIATE THE FIRST FLOW RATE WITH THE FIRST PRESSURE
** FOR THAT SPECIFIC PUMP
15    7.D-3     WVCRDI  CRD FLOW RATE   PUMP CURVE FOR CRD FLOW       PS
16    1.12D-2   WVCRDI  CRD FLOW RATE      PPS VS WVCRDI              PS
17    1.12D-2   WVCRDI  CRD FLOW RATE      M3/S                       PS
18    1.12D-2   WVCRDI  CRD FLOW RATE                                 PS
19    1.12D-2   WVCRDI  CRD FLOW RATE                                 PS
20    1.12D-2   WVCRDI  CRD FLOW RATE                                 PS
21    1.12D-2   WVCRDI  CRD FLOW RATE                                 PS
22    1.12D-2   WVCRDI  CRD FLOW RATE                                 PS
23    6.894D6   PCRD   PPS FOR CRD PUMP                               PS
24    1.0134D5  PCRD   PPS FOR CRD PUMP                               PS
25    1.0134D5  PCRD   PPS FOR CRD PUMP                               PS
26    1.0134D5  PCRD   PPS FOR CRD PUMP                               PS
27    1.0134D5  PCRD   PPS FOR CRD PUMP                               PS
28    1.0134D5  PCRD   PPS FOR CRD PUMP                               PS
29    1.0134D5  PCRD   PPS FOR CRD PUMP                               PS
30    1.0134D5  PCRD   PPS FOR CRD PUMP                               PS
31    3.333D3   WFWMAX MAXIMUM FEEDWATER FLOW RATE (RUN OUT FLOW)     PS
32    6.85D2    WBPMAX MAXIMUM TURBINE BYPASS FLOW RATE               PS
33    1.63D-1   NXCORE EXIT CORE QUALITY AT TIME ZERO                 PS
34    5.264D0   XDCORE REACTOR CORE DIAMETER TO INNER SHROUD WALL     PS
35    2.206D1   XHRV   INTERIOR HEIGHT OF REACTOR VESSEL              PS
36    3.188D0   XRRV   INTERIOR RADIUS OF REACTOR VESSEL              PS
37    41.01D0   ZBJET  ELEVATION AT BOTTOM OF JET PUMPS               PS
38    38.77D0   ZBRDT  ELEVATION AT BOTTOM OF CRD TUBES               PS
39    50.44D0   ZBSEP  ELEVATION AT BOTTOM OF STEAM SEPARATORS        PS
40    37.41D0   ZBV    ELEVATION AT BOTTOM OF REACTOR VESSEL          PS
41    42.73D0   ZCPL   ELEVATION AT CORE PLATE                        PS
42    45.48D0   ZTJET  ELEVATION AT TOP OF JET PUMPS                  PS
43    1.33D0    AJET   TOTAL AREA OF JET PUMPS                        PS
44    46.74D0   ZTOAF  ELEVATION AT TOP OF ACTIVE FUEL                PS
45    52.65D0   ZTSEP  ELEVATION AT TOP OF STEAM SEPARATORS           PS
46    51.91D0   ZWNORM ELEVATION AT NORMAL SHROUD WATER LEVEL         PS
47    41.82D0   ZLOCA  ELEVATION AT BREAK                             PS
48    .2919D0   ALOCA  AREA OF BREAK                                  PS
49    52.325D0  ZWL8   ELEVATION AT LEVEL 8 TRIP                      PS
50    0.0D0     NOT USED
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51   51.26D0      ZSCRAM LOW WATER LEVEL SCRAM                                    PS
52   7.4435D6     PSCRAM HIGH PRESSURE SCRAM SETPOINT                             PS
53   .20D0        FQATWS ATWS CONSTANT POWER ASSUMPTION                           PS
54   1.2D3        TDSLC  TIME FOR SCRAM WITH SLC                                  PS
55   0.D0         TIRR(1) TIME VS. FRACTION OF TOTAL FLOW FOR RECIRC PUMP         PS
56   2.D0         TIRR(2)                                                         PS
57   4.D0         TIRR(3)                                                         PS
58   6.D0         TIRR(4)                                                         PS
59   8.D0         TIRR(5)                                                         PS
60   10.D0        TIRR(6)                                                         PS
61   15.012D0     TIRR(7)                                                         PS
62   40.D0        TIRR(8)                                                         PS
63   1.D0         FWRR(1)                                                         PS
64   .67D0        FWRR(2)                                                         PS
65   .45D0        FWRR(3)                                                         PS
66   .30D0        FWRR(4)                                                         PS
67   .20D0        FWRR(5)                                                         PS
68   .135D0       FWRR(6)                                                         PS
69   .050D0       FWRR(7)                                                         PS
70   0.D0         FWRR(8)                                                         PS
71   1.18393D5    HSLC   INLET ENTHALPY OF SLC                                    PS
72   0.D0         PSLC(1) PRESSURE POINTS FOR SLC FLOW CURVE                      PS
73   7.93D6       PSLC(2)                                                         PS
74   7.93D6       PSLC(3)                                                         PS
75   7.93D6       PSLC(4)                                                         PS
76   7.93D6       PSLC(5)                                                         PS
77   7.93D6       PSLC(6)                                                         PS
78   7.93D6       PSLC(7)                                                         PS
79   7.93D6       PSLC(8)                                                         PS
80   2.713D-3     WVSLC(1) SLC FLOW RATE AT PSLC(1)    -- M3/S                    PS
81   2.713D-3     WVSLC(2)                                                        PS
82   2.713D-3     WVSLC(3)                                                        PS
83   2.713D-3     WVSLC(4)                                                        PS
84   2.713D-3     WVSLC(5)                                                        PS
85   2.713D-3     WVSLC(6)                                                        PS
86   2.713D-3     WVSLC(7)                                                        PS
87   2.713D-3     WVSLC(8)                                                        PS
88   .2D0         TDRPT DELAY TIME FOR RECIRC PUMP TRIP                           PS
89   47.16D0      ZLMSIV LOW WATER LEVEL FOR MSIV CLOSURE                         PS
90   49.92D0      ZLRPT  LOW WATER LEVEL FOR RECIRC PUMP TRIP                     PS
91   7.858D6      PHRPT  HIGH VESSEL PRESSURE FOR RECIRC PUMP TRIP                PS
92   1.1327D5     PDWSCM HIGH DRYWELL PRESSURE FOR SCRAM                          PS
93   .032D0       FENRCH NORMAL FUEL ENRICHMENT                                   PS
94   2.D4         EXPO   AVERAGE BURNUP IN MWD/TONNE                              PS
95   .6D0         FCR    PRODUCTION OF U239 TO ABSORBTION IN FUEL                 PS
96   1.3D0        FFAF   RATIO OF FISSILE ABSORBTION TO TOTAL FISSION             PS
97   5.D-1        FQFR1  FISSION POWER FRACTION OF U235 AND PU241                 PS
98   4.2D-1       FQFR2  FISSION POWER FRACTION OF PU239                          PS
99   8.D-2        FQFR3  FISSION POWER FRACTION OF U238                           PS
100  .3051D0      XPCRDT PITCH OF CRD TUBES                                       PS
101  .2755D0      XDCRDT OUTER DIAMETER OF CRD TUBE                               PS
102  58.D0        NINST  NUMBER OF INSTRUMENT TUBES                               PS
103  .0044D0      XTHCRD THICKNESS OF CRD TUBE WALL                               PS
104  .0508D0      XDINST OUTER DIAMETER OF INSTRUMENT TUBE                        PS
105  .0818D0      XDRIVE LOWER CRD DRIVE OUTER DIAMETER                           PS
106  1.004D-3     VWCRD  SPECIFIC VOLUME OF CRD WATER                             PS
107  1.004D-3     VWCSI  SPECIFIC VOLUME OF SLC WATER                             PS
108  5.8167D4     MEQPS  MASS OF UPPER PLENUM HEAT SINK                           PS
109  1.016D3      AEQPS  SURFACE AREA OF UPPER PLENUM HEAT SINK                   PS
110  .241D0       XTRV   THICKNESS OF LOWER VESSEL HEAD                           PS
111  0.D0         TIFWCD TIME SINCE MSIV CLOSURE SIGNAL VS. FEEDWATER             PS
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112  0.DO            COASTDOWN MASS FLOW RATE                                PS
113  0.DO                                                                    PS
114  0.DO                                                                    PS
115  0.DO                                                                    PS
116  0.DO                                                                    PS
117  0.DO                                                                    PS
118  0.DO                                                                    PS
119  0.DO        WFWCD                                                       PS
120  0.DO                                                                    PS
121  0.DO                                                                    PS
122  0.DO                                                                    PS
123  0.DO                                                                    PS
124  0.DO                                                                    PS
125  0.DO                                                                    PS
126  0.DO                                                                    PS
127  5.86D6      PLMSIV LOW RPV PRESSURE FOR MSIV CLOSURE                    PS
128  53.9DO      ZMSL   ELAVATION AT CENTER LINE OF MAIN STEAM LINE          PS
**                                                                           HE
*CIRC
01   0.DO        ACSHS(1) CORE + LOWER PLENUM
**                        CARBON STEEL-HEAT SINK HEAT TRANSFER AREA
02   140.DO      ACSHS(2) UPPER PLENUM
03   0.DO        ACSHS(3) DOWNCOMER
04   0.DO        ACSHS(4)
05   0.DO        ACSHS(5)
06   50.D3       MCS(1)   CORE + LOWER PLENUM CARBON STEEL MASS
07   100.D3      MCS(2)   UPPER PLENUM
08   350.D3      MCS(3)   DOWNCOMER
09   0.DO        MCS(4)
10   0.DO        MCS(5)
11   0.DO        MHS(1)   CORE + LOWER PLENUM HEAT SINK MASS
12   100.D3      MHS(2)   UPPER PLENUM
13   0.DO        MHS(3)   DOWNCOMER
14   0.DO        MHS(4)
15   0.DO        MHS(5)
16   0.DO        ACSX(1)  CORE + LOWER PLENUM CARBON STEEL TO DRYWELL
**                        HEAT TRANSFER ARE
17   0.DO        ACSX(2)  UPPER PLENUM
18   240.DO      ACSX(3)  DOWNCOMER
19   0.DO        ACSX(4)
20   0.DO        ACSX(5)
21   0.DO        AHSX(1)  CORE + LOWER PLENUM HEAT SINK TO DRYWELL
**                        HEAT TRANSFER AREA
22   140.DO      AHSX(2)  UPPER PLENUM
23   0.DO        AHSX(3)  DOWNCOMER
24   0.DO        AHSX(4)
25   0.DO        AHSX(5)
26   100.DO      AGCS(1)  CORE + LOWER PLENUM GAS TO CARBON STEEL
**                        HEAT TRANSFER AREA
27   5.D3        AGCS(2)  UPPER PLENUM
28   240.DO      AGCS(3)  DOWNCOMER
29   0.DO        AGCS(4)
30   0.DO        AGCS(5)
31   0.DO        AGHS(1)  CORE + LOWER PLENUM GAS TO HEAT SINK
**                        HEAT TRANSFER AREA
32   140.DO      AGHS(2)  UPPER PLENUM
33   0.DO        AGHS(3)  DOWNCOMER
34   0.DO        AGHS(4)
35   0.DO        AGHS(5)
36   8.0DO       XL(1)    CORE + LOWER PLENUM LENGTH
37   5.DO        XL(2)    UPPER PLENUM LENGTH
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38   10.D0      XL(3)     DOWNCOMER LENGTH
39   0.D0       XL(4)
40   0.D0       XL(5)
41   11.D0      AG(1)     CORE + LOWER PLENUM FLOW AREA
42   11.D0      AG(2)     UPPER PLENUM FLOW AREA
43   10.D0      AG(3)     DOWNCOMER FLOW AREA
44   0.D0       AG(4)
45   0.D0       AG(5)
46   5.D0       DH(1)   HYDRAULIC DIAMETER FOR CORE REGION
47   .15D0      DH(2)   HYDRAULIC DIAMETER FOR UPPER PLENUM
48   .4D0       DH(3)   HYDRAULIC DIAMETER FOR DOWNCOMER
49   0.D0       DH(4)
50   0.D0       DH(5)
51   0.D0       QCO     RPV CONVECTION LOSSES AT TIME ZERO
52   8.D0       FINPLT NUMBER OF LAYERS IN REFLECTIVE INSULATION
**                                                                          HE
*HEATUP                                                                     HE
01   3.81D0     XZFUEL LENGTH OF ACTIVE FUEL                                HE
02   5.21D-3    XRFUEL RADIUS OF FUEL PELLET                                HE
03   8.13D-4    XTCLAD THICKNESS OF CLADDING                                HE
04   5.033D4    MZRCAN TOTAL MASS OF ZR IN ASSEMBLY CAN                     HE
05   1.7D4      MBCR   TOTAL MASS OF CONTROL BLADES IN REACTOR CORE         HE
06   3.048D-3   XZRCAN CAN WALL THICKNESS                                   HE
** NODE 1,1 IS BOTTOM-CENTER, 1,10 IS TOP-CENTER, 2,1 IS SECOND RADIAL
** RING OUT FROM CENTER AT THE BOTTOM OF THE CORE ,ETC
07   6.680D-1   FPEAK(1,1) PEAKING FACTOR FOR NODE (1,1)                    HE
08   7.910D-1   FPEAK(2,1) PEAKING FACTOR FOR NODE (2,1)                    HE
09   4.700D-1   FPEAK(3,1) PEAKING FACTOR FOR NODE (3,1)                    HE
15   1.145D0    FPEAK(1,2) PEAKING FACTOR FOR NODE (1,2)                    HE
16   1.466D0    FPEAK(2,2) PEAKING FACTOR FOR NODE (2,2)                    HE
17   9.290D-1   FPEAK(3,2) PEAKING FACTOR FOR NODE (3,2)                    HE
23   1.019D0    FPEAK(1,3) PEAKING FACTOR FOR NODE (1,3)                    HE
24   1.343D0    FPEAK(2,3) PEAKING FACTOR FOR NODE (2,3)                    HE
25   8.960D-1   FPEAK(3,3) PEAKING FACTOR FOR NODE (3,3)                    HE
31   1.029D0    FPEAK(1,4) PEAKING FACTOR FOR NODE (1,4)                    HE
32   1.281D0    FPEAK(2,4) PEAKING FACTOR FOR NODE (2,4)                    HE
33   8.67D-1    FPEAK(3,4) PEAKING FACTOR FOR NODE (3,4)                    HE
39   1.223D0    FPEAK(1,5) PEAKING FACTOR FOR NODE (1,5)                    HE
40   1.414D0    FPEAK(2,5) PEAKING FACTOR FOR NODE (2,5)                    HE
41   9.430D-1   FPEAK(3,5) PEAKING FACTOR FOR NODE (3,5)                    HE
47   1.235D0    FPEAK(1,6) PEAKING FACTOR FOR NODE (1,6)                    HE
48   1.373D0    FPEAK(2,6) PEAKING FACTOR FOR NODE (2,6)                    HE
49   9.03D-1    FPEAK(3,6) PEAKING FACTOR FOR NODE (3,6)                    HE
55   1.198D0    FPEAK(1,7) PEAKING FACTOR FOR NODE (1,7)                    HE
56   1.269D0    FPEAK(2,7) PEAKING FACTOR FOR NODE (2,7)                    HE
57   8.09D-1    FPEAK(3,7) PEAKING FACTOR FOR NODE (3,7)                    HE
63   1.235D0    FPEAK(1,8) PEAKING FACTOR FOR NODE (1,8)                    HE
64   1.243D0    FPEAK(2,8) PEAKING FACTOR FOR NODE (2,8)                    HE
65   7.11D-1    FPEAK(3,8) PEAKING FACTOR FOR NODE (3,8)                    HE
71   1.331D0    FPEAK(1,9) PEAKING FACTOR FOR NODE (1,9)                    HE
72   1.107D0    FPEAK(2,9) PEAKING FACTOR FOR NODE (2,9)                    HE
73   5.53D-1    FPEAK(3,9) PEAKING FACTOR FOR NODE (3,9)                    HE
79   7.40D-1    FPEAK(1,10)PEAKING FACTOR FOR NODE (1,10)                   HE
80   5.64D-1    FPEAK(2,10)PEAKING FACTOR FOR NODE (2,10)                   HE
81   2.69D-1    FPEAK(3,10)PEAKING FACTOR FOR NODE (3,10)                   HE
87   0.3D0      XCHIM  UNHEATED FUEL LENGTH AT TOP OF CORE                  HE
88   1.D-7      XIZROX INITIAL CLADDING OXIDE THICKNESS                     HE
**                                                                          ES
**                                                                          ES
*ENGINEERED SAFEGUARDS                                                      ES
01   1.D0       NLPCI1 NUMBER OF LPCI PUMPS IN LOOP 1                       ES
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02    1.0D0       NLPCI2 NUMBER OF LPCI PUMPS IN LOOP 2                        ES
03    1.0D0       NLPCI3 NUMBER OF LPCI PUMPS IN LOOP 3 (INJECTION ONLY)       ES
04    1.D0        NLPCSP NUMBER OF LPCS PUMPS                                  ES
05    0.0D0       NOT USED
06    1.4D1       VMNCST MIN. WATER VOLUME IN CONDENSATE STORAGE TANK          ES
**                       FOR HPCI AND RCIC SUCTION SWITCH OVER                 ES
07    1.008D-3  VWCST  SPECIFIC VOLUME OF CST WATER                            ES
** ALL PUMP CURVES ARE ARRANGED SO THAT THE FIRST FLOW ENTRY CORRESPONDS
** TO THE FIRST PRESSURE ENTRY
24    2.09D6      PLPCI(1)     PUMP CURVES FOR ECCS -- LPCI                    ES
25    2.D6        PLPCI(2)     PPS-PDW VS. VOLUMETRIC FLOW                     ES
26    1.896D6     PLPCI(3)                                                     ES
27    1.641D6     PLPCI(4)                                                     ES
28    1.462D6     PLPCI(5)                                                     ES
29    1.1651D6    PLPCI(6)                                                     ES
30    .841D6      PLPCI(7)                                                     ES
31    .4964D6     PLPCI(8)                                                     ES
32    0.0D0       WVLPCI(1)                                                    ES
33    .1262D0     WVLPCI(2)                                                    ES
34    .1893D0     WVLPCI(3)                                                    ES
35    .3155D0     WVLPCI(4)                                                    ES
36    .3786D0     WVLPCI(5)                                                    ES
37    .4417D0     WVLPCI(6)                                                    ES
38    .5048D0     WVLPCI(7)                                                    ES
39    .5641D0     WVLPCI(8)                                                    ES
40    3.584D6     PLPCS(1)     LPCS PUMP CURVE                                 ES
41    3.378D6     PLPCS(2)                                                     ES
42    3.06D6      PLPCS(3)                                                     ES
43    2.889D6     PLPCS(4)                                                     ES
44    2.67D6      PLPCS(5)                                                     ES
45    2.392D6     PLPCS(6)                                                     ES
46    2.068D6     PLPCS(7)                                                     ES
47    1.572D6     PLPCS(8)                                                     ES
48    0.D0        WVLPCS(1)                                                    ES
49    .1262D0     WVLPCS(2)                                                    ES
50    .2524D0     WVLPCS(3)                                                    ES
51    .3155D0     WVLPCS(4)                                                    ES
52    .3786D0     WVLPCS(5)                                                    ES
53    .4417D0     WVLPCS(6)                                                    ES
54    .5048D0     WVLPCS(7)                                                    ES
55    .5742D0     WVLPCS(8)                                                    ES
56    9.892D6     PHPCS(1)      HPCS PUMP CURVE                                ES
57    8.886D6     PHPCS(2)                                                     ES
58    7.521D6     PHPCS(3)                                                     ES
59    6.749D6     PHPCS(4)                                                     ES
60    5.667D6     PHPCS(5)                                                     ES
61    4.226D6     PHPCS(6)                                                     ES
62    2.296D6     PHPCS(7)                                                     ES
63    0.0D0       PHPCS(8)                                                     ES
64    0.D0        WVHPCS(1)                                                    ES
65    .1262D0     WVHPCS(2)                                                    ES
66    .2524D0     WVHPCS(3)                                                    ES
67    .3155D0     WVHPCS(4)                                                    ES
68    .3786D0     WVHPCS(5)                                                    ES
69    .4417D0     WVHPCS(6)                                                    ES
70    .5048D0     WVHPCS(7)                                                    ES
71    .5742D0     WVHPCS(8)                                                    ES
72    10.341D6    PRCIC(1)     RCIC PUMP CURVE                                 ES
73    10.340D6    PRCIC(2)                                                     ES
74    6.894D6     PRCIC(3)                                                     ES
75    3.447D6     PRCIC(4)                                                     ES
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76   2.758D6    PRCIC(5)                                                    ES
77   2.068D6    PRCIC(6)                                                    ES
78   4.144D5    PRCIC(7)                                                    ES
79   4.137D5    PRCIC(8)                                                    ES
80   0.D0       WVRCIC(1)                                                   ES
81   .0505D0    WVRCIC(2)                                                   ES
82   .0505D0    WVRCIC(3)                                                   ES
83   .0505D0    WVRCIC(4)                                                   ES
84   .0505D0    WVRCIC(5)                                                   ES
85   .0505D0    WVRCIC(6)                                                   ES
86   .0505D0    WVRCIC(7)                                                   ES
87   0.D0       WVRCIC(8)                                                   ES
88   50.D0      ZLHPCI LOW WATER INITIATION FOR HPCI                        ES
89   1.D10      PSHPCI HIGH DRYWELL PRESSURE SET POINT FOR HPCI             ES
90   1.D10      TDHPCI TIME DELAY FOR HPCI                                  ES
91   1.D10      PHHPCI MINIMUM PRESSURE FOR HPCI TURBINE                    ES
92   49.92D0    ZLHPCS LOW WATER INITIATION FOR HPCS                        ES
93   1.144D5    PSHPCS HIGH DRYWELL PRESSURE SET POINT FOR HPCS             ES
94   27.D0      TDHPCS TIME DELAY FOR HPCS                                  ES
95   47.16D0    ZLLPCI LOW WATER INITIATION FOR LPCI                        ES
96   1.144D5    PSLPCI HIGH DRYWELL PRESSURE SET POINT FOR LPCI             ES
97   40.D0      TDLPCI TIME DELAY FOR LPCI                                  ES
98   1.D10      PLLPCI LOW VESSEL PRESSURE PERMISSIVE FOR LPCI              ES
99   47.16D0    ZLLPCS LOW WATER INITIATION FOR LPCS                        ES
100  1.144D5    PSLPCS HIGH DRYWELL PRESSURE SET POINT FOR LPCS             ES
101  37.D0      TDLPCS TIME DELAY FOR LPCS                                  ES
** THE NEXT PARAMETER IS A LOCA PERMISSIVE SIGNAL AND IF ONE DOES NOT
** EXIST THEN ENTER VERY LARGE NUMBER (1.D10 PA)
102  1.D10      PLLPCS LOW VESSEL PRESSURE PERMISSIVE FOR LPCS              ES
103  49.92D0    ZLRCIC LOW WATER INITIATION FOR RCIC                        ES
104  1.0D10     PSRCIC HIGH DRYWELL PRESSURE SET POINT FOR RCIC             ES
105  30.D0      TDRCIC TIME DELAY FOR RCIC                                  ES
106  5.15D5     PHRCIC MINIMUM VESSEL PRESSURE FOR RCIC TURBINE             ES
107  1.70D5     HCST   ENTHALPY OF CST                                      ES
108  498.D0     WSWHX  SERVICE WATER FLOW RATE (KG/S) THRU EACH RHR HTX     ES
109  .0119D0    ASRV1  FLOW AREA OF RELIEF VALVE TYPE #1                    ES
110  .0119D0    ASRV2  FLOW AREA OF RELIEF VALVE TYPE #2                    ES
111  .0119D0    ASRV3  FLOW AREA OF RELIEF VALVE TYPE #3                    ES
112  .0119D0    ASRV4  FLOW AREA OF RELIEF VALVE TYPE #4                    ES
** IF THE AREA OF GROUP #5 IS INPUT AS A NEGATIVE NUMBER THEN THE VALVE
** WILL DISCHARGE DIRECTLY INTO THE DRYWELL, IF POSITIVE IT WILL
** DISCHARGE INTO THE SUPPRESSION POOL
113  .0D0       ASRV5  FLOW AREA OF RELIEF VALVE TYPE #5                    ES
114  1.0D0      NSRV1  NUMBER OF TYPE #1 RELIEF VALVES                      ES
115  1.0D0      NSRV2  NUMBER OF TYPE #2 RELIEF VALVES                      ES
116  9.0D0      NSRV3  NUMBER OF TYPE #3 RELIEF VALVES                      ES
117  9.0D0      NSRV4  NUMBER OF TYPE #4 RELIEF VALVES                      ES
118  0.D0       NSRV5  NUMBER OF TYPE #5 RELIEF VALVES                      ES
119  0.D0       NADS1  NUMBER OF ADS VALVES IN GROUP 1                      ES
120  1.D0       NADS2  NUMBER OF ADS VALVES IN GROUP 2                      ES
121  3.D0       NADS3  NUMBER OF ADS VALVES IN GROUP 3                      ES
122  4.D0       NADS4  NUMBER OF ADS VALVES IN GROUP 4                      ES
123  7.1220D6   PSRV1  PRESSURE SETPOINT FOR #1 RELIEF VALVE                ES
124  7.398D6    PSRV2  PRESSURE SETPOINT FOR #2 RELIEF VALVE                ES
125  7.674D6    PSRV3  PRESSURE SETPOINT FOR #3 RELIEF VALVE                ES
126  7.743D6    PSRV4  PRESSURE SETPOINT FOR #4 RELIEF VALVE                ES
127  1.D10      PSRV5  PRESSURE SETPOINT FOR #5 RELIEF VALVE                ES
128  47.16D0    ZLADS  LOW WATER LEVEL FOR INITIATION OF ADS                ES
129  114.37D3   PSADS  HIGH DRYWELL PRESSURE SET POINT FOR ADS              ES
130  115.D0     TDADS  TIME DELAY FOR ADS ACTUATION                         ES
** LPCI,LPCS,HPCS HAVE NPSH REQUIRMENTS
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** HPCI AND RCIC WILL TRIP OFF ON USER SUPPLIED TEMPERATURE OF SUPP POOL
131  373.D0     TCHPCI INLET TEMP LIMIT FOR HPCI                                ES
132  31.4D0     ZCLHPS PUMP CENTER LINE ELAVATION FOR HPCS                      ES
133  29.3D0     ZCLLPI PUMP CENTER LINE ELAVATION FOR LPCI                      ES
134  30.5D0     ZCLLPS PUMP CENTER LINE ELAVATION FOR LPCS                      ES
135  366.3D0    TCRCIC INLET TEMP LIMIT FOR RCIC                                ES
136  305.D0     TWSW   SERVICE WATER TEMP (RHR HEAT EXCHANGERS,TCOLD)           ES
137  13.D0      TDDG1  HPCS LOAD DELAY TIME FOR DIESEL                          ES
138  13.D0      TDDG2  LPCI LOAD DELAY TIME FOR DIESEL                          ES
139  13.D0      TDDG3  LPCS LOAD DELAY TIME FOR DIESEL                          ES
140  2.3D-4     XDDROP SRRAY DROPLET DIAMETER FOR CONTAINMENT SPRAYS            ES
141  19.6D0     XHSPWW SPRAY FALL HEIGHT IN WETWELL                             ES
142  10.D0      XHSPDW SPRAY FALL HEIGHT IN DRYWELL                             ES
** THE HPSW SYSTEM CAN BE USED TO MODEL ANY INJECTION MODE SUCH AS
** SERVICE WATER OR FIRE WATER, THE SYSTEM IS TOTALLY DEFINED BELOW
143  1.837D5    HWHPSW ENTHALPY OF HIGH PRES SERVICE WATER (MARK I CI)      ES
144  1.009D-3   VWHPSW SPEC VOL OF HIGH PRES SERVICE WATER (MARK I CI)      ES
145  6.525D5    PHPSW(1) PPS VS. VOLUMETRIC FLOW FOR HPSW CORE INJECTION    ES
146  6.524D5    PHPSW(2)                       (MARK I CORE INJECTION)      ES
147  6.523D5    PHPSW(3)                                                    ES
148  6.522D5    PHPSW(4)                                                    ES
149  6.521D5    PHPSW(5)                                                    ES
150  6.520D5    PHPSW(6)                                                    ES
151  6.519D5    PHPSW(7)                                                    ES
152  0.D0       PHPSW(8)                                                    ES
153  0.D0       WVHPSW(1)                                                   ES
154  .757D0     WVHPSW(2)                                                   ES
155  .757D0     WVHPSW(3)                                                   ES
156  .757D0     WVHPSW(4)                                                   ES
157  .757D0     WVHPSW(5)                                                   ES
158  .757D0     WVHPSW(6)                                                   ES
159  .757D0     WVHPSW(7)                                                   ES
160  .757D0     WVHPSW(8)                                                   ES
161  1.144D5    PDWSPR DRYWELL PRES SET PT FOR MARK III CONTAINMNT SPRAYS ES
162  1.6339D5   PWWSPR WETWELL DPRES SET PT FOR MARK III CONTAINMNT SPRAYSES
163  600.D0     TDSPR  TIME DELAY FOR MARK III CONTAINMENT SPRAYS           ES
164  7.38D5     PDSRV1 DEAD BAND FOR CLOSURE OF SRV#1                       ES
165  9.45D5     PDSRV2 DEAD BAND FOR CLOSURE OF SRV#2                       ES
166  1.151D6    PDSRV3 DEAD BAND FOR CLOSURE OF SRV#3                       ES
167  5.17D5     PDSRV4 DEAD BAND FOR CLOSURE OF SRV#4                       ES
168  0.D0       PDSRV5 DEAD BAND FOR CLOSURE OF SRV#5                       ES
** 8 POINTS ARE USED HERE TO DEFINE THE RCIC AND HPCI TURBINE STEAM FLOW
185  8.22D6     PTURRI(1)  PPS-PWW VS. STEAM FLOW TO RCIC TURBINE           ES
186  1.34D6     PTURRI(2)                                                   ES
187  1.34D6     PTURRI(3)                                                   ES
188  1.34D6     PTURRI(4)                                                   ES
189  1.34D6     PTURRI(5)                                                   ES
190  1.34D6     PTURRI(6)                                                   ES
191  1.34D6     PTURRI(7)                                                   ES
192  1.34D6     PTURRI(8)                                                   ES
193  4.83D0     WSTRCI(1)                                                   ES
194  1.56D0     WSTRCI(2)                                                   ES
195  1.56D0     WSTRCI(3)                                                   ES
196  1.56D0     WSTRCI(4)                                                   ES
197  1.56D0     WSTRCI(5)                                                   ES
198  1.56D0     WSTRCI(6)                                                   ES
199  1.56D0     WSTRCI(7)                                                   ES
200  1.56D0     WSTRCI(8)                                                   ES
201  2.737D5    PHTURH HIGH TURBINE EXHAUST PRESSURE FOR HPCI               ES
202  1.72D5     PHTURR HIGH TURBINE EXHAUST PRESSURE FOR RCIC               ES
203  4.916D5    PCFAIL CONTAINMENT FAILURE PRESSURE                         ES
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** THE SHUT OFF HEAD SHOULD APPEAR IN THE PUMP CURVE DEFINITION FOR ECCS
** THE NEXT TWO PARAMETERS ARE PERMISSIVE SIGNALS FOR TRIPPING SYSTEMS
204  1.D10      PHLPCI HIGH VESSEL PRESSURE TRIP FOR LPCI                   ES
205  1.D10      PHLPCS HIGH VESSEL PRESSURE TRIP FOR LPCS                   ES
206  34.237D0   ZHISP  HIGH SUPP. POOL LEVEL TRIP FOR HP SUCTION            ES
207  47.16D0    ZLSPR  NOT USED
** ALL OF THE HEAT EXCHANGER DATA MAY BE OMITTED WITH THE EXCEPTION
** OF NTUHX1,NTUHX2,NHX1,NHX2
208  0.D0       NTHX   NUMBER OF TUBES IN RHR HTX                           ES
209  0.D0       NBHX   NUMBER OF BAFFLES IN RHR HTX                         ES
210  0.D0       XIDTHX TUBE ID FOR RHR HTX                                  ES
211  0.D0       XTTHX  TUBE WALL THICKNESS FOR RHR HTX                      ES
212  0.D0       XTCHX  TUBE CENTER TO CENTER SPACING FOR RHR HTX            ES
213  0.D0       XSHX   SHELL LENGTH FOR RHR HTX                             ES
214  0.D0       RGFOUL FOULING FACTOR FOR RHR HTX                           ES
215  0.D0       KTHX   THERMAL CONDUCTIVITY FOR TUBE WALL (RHR HTX)         ES
216  0.D0       XBCHX  BAFFLE CUT LENGTH FOR RHR HTX                        ES
217  0.D0       XIDSHX SHELL ID FOR RHR HTX                                 ES
218  0.D0       XSTHX  BUNDLE TO SHELL GAP LENGTH FOR RHR HTX               ES
** NTU VALUES NOT NEEDED IF ABOVE INFORMATION IS DEFINED
219  1.2D0      NTUHX1 NTU FOR RHR HTX #1                                   ES
220  1.2D0      NTUHX2 NTU FOR RHR HTX #2                                   ES
221  2.D0       NHX1   NUMBER OF RHR LOOP #1 HTX                            ES
222  2.D0       NHX2   NUMBER OF RHR LOOP #2 HTX                            ES
223  2.D0       EHX    TYPE OF RHR HTX(1=STRAIGHT TUBE,2=U TUBE)            ES
224  14.4D3     TDBATT BATTERY OPERATION TIME FOR STATION BLACK-OUT         ES
** THE FOLLOWING ARE NPSH CURVES AND THE FIRST ENTRY FOR THAT SYSTEM
** CORRESPONDS TO THE FIRST FLOW RATE LISTED ABOVE FOR THAT PUMP
225  .518D0     ZHDHPS HPCS NPSH FOR GIVEN FLOWS                            ES
226  .518D0            (METERS)                                             ES
227  .518D0                                                                 ES
228  .549D0                                                                 ES
229  .61D0                                                                  ES
230  .762D0                                                                 ES
231  1.372D0                                                                ES
232  2.226D0                                                                ES
233  2.073D0     ZHDLPI LPCI NPSH FOR GIVEN FLOW                            ES
234  .671D0                                                                 ES
235  .335D0                                                                 ES
236  .335D0                                                                 ES
237  .335D0                                                                 ES
238  .335D0                                                                 ES
239  .366D0                                                                 ES
240  .396D0                                                                 ES
241  2.457D0     ZHDLPS LPCS NPSH FOR GIVEN FLOW                            ES
242  1.524D0                                                                ES
243  .854D0                                                                 ES
244  .854D0                                                                 ES
245  .854D0                                                                 ES
246  .854D0                                                                 ES
247  .854D0                                                                 ES
248  .854D0                                                                 ES
249  31.4D0     ZCLRCI PUMP CENTER LINE ELEVATION FOR RCIC                  ES
250  28.35D0    ZCLHPI PUMP CENTER LINE ELEVATION FOR HPCI                  ES
251  .0093D0    ACVENT AREA OF CONTAINMENT VENT                             ES
252  0.0D0      ZCFAIL ELEVATION OF CONTAINMENT VENT IN WETWELL (MII ONLY)ES
253  28.35D0    ZSRVD  AVERAGE ELEVATION OF SRV DISCHARGE IN SUPP POOL      ES
254  0.D0       TGDWHX(1)  COOLING CURVE FOR DRYWELL COOLERS
255  0.D0       TGDWHX(2)  TEMP IN DRYWELL VS. HEAT LOSS RATE (J/S)
256  0.D0       TGDWHX(3)
257  0.D0       TGDWHX(4)
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258  0.D0        TGDWHX(5)
259  0.D0        TGDWHX(6)
260  0.D0        TGDWHX(7)
261  0.D0        TGDWHX(8)
262  0.D0        QGDWHX(1)   HEAT LOSS RATE FOR DRYWELL COOLERS (J/S)
263  0.D0        QGDWHX(2)
264  0.D0        QGDWHX(3)
265  0.D0        QGDWHX(4)
266  0.D0        QGDWHX(5)
267  0.D0        QGDWHX(6)
268  0.D0        QGDWHX(7)
269  0.D0        QGDWHX(8)
**                                                                       DW
**                                                                       DW
*DRYWELL                                                                 DW
01   5.D-1       RELHDW RELATIVE HUMIDITY IN DRYWELL                     DW
02   7650.2D0    VOLDW  VOLUME OF DRYWELL                                DW
03   30.71D0     ZDWF   ELEVATION AT DRYWELL FLOOR                       DW
04   318.4D0     ADWF   AREA OF DRYWELL FLOOR                            DW
05   35.77D0     ZWDWWW ELEVATION OF WEIR WALL BETWEEN DRYWELL AND WETWELL DW
06   24.D0       NIGDW  NUMBER OF IGNITERS IN THE DRYWELL                DW
07   9.27D0      XIGDW  AVERAGE DISTANCE FROM IGNITER TO CEILING         DW
08   318.4D0     ACHDW  CHARACTERISTIC FLOOR AREA FOR BURN CALCULATION   DW
**                                                                       WW
**                                                                       WW
*WETWELL                                                                 WW
01   28.35D0     ZWWF   ELEVATION AT WETWELL FLOOR                       WW
02   5.07D-2     AVB    FLOW AREA THROUGH VACUUM BREAKERS                WW
03   3.0D0       NVB    NUMBER OF VACUUM BREAKERS                        WW
04   6.D3        PSETVB PRESSURE SETPOINT FOR VACUUM BREAKERS            WW
05   6.D3        PDVB   DEAD BAND FOR VACUUM BREAKERS                    WW
06   7.9856D3    VOLWW  TOTAL VOLUME OF WETWELL (PLUS SUPP POOL)         WW
07   6.D-1       RELHWW RELATIVE HUMIDITY IN WETWELL                     WW
08   6.D0        NIGWW  NUMBER OF IGNITERS IN THE WETWELL                WW
09   .6D0        XIGWW  AVERAGE DISTANCE FROM IGNITER TO CEILING         WW
10   619.3D0     ACHWW  CHARACTERISTIC FLOOR AREA FOR BURN CALCULATION   WW
11   0.D0        AWWF   AREA OF WETWELL FLOOR (MARK II)                  WW
**                                                                       PD
**                                                                       PD
*PEDESTAL                                                                PD
01   3.269D1     APDF   AREA OF PEDESTAL FLOOR                           PD
02   3.951D0     APDVT  AREA OF PEDESTAL-DRYWELL OPENING                 PD
03   2.678D2     VOLPD  VOLUME OF PEDESTAL                               PD
04   31.72D0     ZWPDDW ELEVATION OF WIER BETWEEN PED AND DRYWELL        PD
05   28.80D0     ZPDF   ELEVATION AT PEDESTAL FLOOR                      PD
06   5.D-1       RELHPD RELATIVE HUMIDITY IN PEDESTAL                    PD
07   0.D0        NIGPD  NUMBER OF IGNITERS IN THE PEDSTAL                PD
08   0.D0        XIGPD  AVERAGE DISTANCE FROM IGNITER TO CEILING         PD
09   32.69D0     ACHPD  CHARACTERISTIC FLOOR AREA FOR BURN CALCULATION   PD
10   0.D0        XWPDVT WIDTH OF PEDESTAL DOOR (MARK II ONLY)            PD
**    NOTE:  THE NEXT PARAMETER-ADCPD-CAN BE USED TO MODEL THE NORMAL
**     LEAK AREA BETWEEN THE DRYWELL AND COMPARTMENT A OF A MARK III
11   0.0043D0    ADCPD  AREA OF PEDESTAL DOWNCOMERS
12   0.D0        NDCPD  NUMBER OF PEDESTAL DOWNCOMERS
13   2.D0        XHPDDW DISTANCE BETWEEN UPPER AND LOWER VENTS FOR
**                      PED-DRYWELL NATURAL CIRCULATION
**                                                                       SP
**                                                                       SP
*SUPPRESSION POOL(MARKIII ONLY)                                          SP
01   5.15D1      ASPDW  AREA OF DRYWELL SIDE OF SUPPRESSION POOL         SP
02   6.193D2     ASPPC  AREA OF CONTAINMENT SIDE OF SUPPRESSION POOL     SP
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03   4.5D1      NVT1    NUMBER OF VENTS OF TYPE #1 -- TOP                        SP
04   4.5D1      NVT2    NUMBER OF VENTS OF TYPE #2 -- MID                        SP
05   4.5D1      NVT3    NUMBER OF VENTS OF TYPE #3 -- BOTTOM                     SP
06   7.1D-1     XDIAVT  DIAMETER OF ONE SUPPRESSION POOL VENT                    SP
07   33.449D0   ZLLSP   ELEVATION OF SUPP. POOL LOW LEVEL SETPOINT               SP
08   32.16D0    ZVT1    ELEVATION OF TOP OF VENT TYPE #1                         SP
09   30.89D0    ZVT2    ELEVATION OF TOP OF VENT TYPE #2                         SP
10   29.62D0    ZVT3    ELEVATION OF TOP OF VENT TYPE #3                         SP
**                                                                               IN
**                                                                               IN
*INITIAL CONDITIONS                                                              IN
01   3.833D9    QPOWER CORE POWER                                                IN
02   7.17D6     PPSO    INITIAL PRESSURE IN PRIMARY SYSTEM                       IN
03   1.0D5      PPDO    INITIAL PRESSURE IN PEDESTAL                             IN
04   1.0D5      PDWO    INITIAL PRESSURE IN DRYWELL                              IN
05   1.0D5      PWWO    INITIAL PRESSURE IN WETWELL                              IN
06   34.01D0    ZSPDWO  INIT.ELEV. OF WATER LEVEL IN DW SIDE OF SUPP.POOL        IN
07   34.01D0    ZSPWWO  INIT.ELEV. OF WATER LEVEL IN PC SIDE OF SUPP.POOL        IN
08   3.3D2      TPDO    INITIAL TEMPERATURE IN PEDESTAL                          IN
09   3.3D2      TDWO    INITIAL TEMPERATURE IN DRYWELL                           IN
10   3.08D2     TWWO    INITIAL TEMPERATURE IN WETWELL                           IN
11   3.08D2     TWSPO   INITIAL TEMPERATURE OF SUPPRESSION POOL WATER            IN
12   51.91D0    ZWSHO   INITIAL ELEVATION OF WATER IN THE SHROUD                 IN
13   2.05D6     MWCBO   MASS OF WATER IN UPPER POOL (MARKIII ONLY)               IN
14   896.6D0    VCSTO   VOLUME OF WATER IN CONDENSATE STORAGE TANK               IN
15   297.D0     TAMB    AMBIENT TEMPERATURE                                      IN
16   1.D5       PAMB    AMBIENT PRESSURE                                         IN
**                                                                               CC
**                                                                               CC
*CONTROL CARDS                                                                   CC
01   3          IBWR    CONTAINMENT TYPE (MARK 1,2, OR 3 )                       CC
02   46         IRSTW   UNIT NUMBER TO WRITE RESTART FILE (MAIN)                 CC
03   47         IHUW    UNIT NUMBER TO WRITE RESTART FILE (HEATUP)               CC
04   40         IPOUT   UNIT NUMBER TO WRITE PROGRAM OUTPUT FILE                 CC
05   41         IPLT1   FIRST PLOT FILE (5 FILES)                                CC
06   500        IPTSMX  MAXIMUM NUMBER OF PLOTTED POINTS                         CC
07   4          IPTSPK  MAXIMUM NUMBER OF PLOT POINTS TRACED FOR FULL            CC
**                      SCALE SPIKE                                              CC
08   80         IPTSAV  NUMBER OF POINTS SAVED FOR NON-CHANGING PLOT             CC
09   1          ISUMM   SUMMARY DATA(0=ALL EVENTS,1=SHORTER LIST)                CC
10   48         ISUM    SUMMARY FILE NUMBER                                      CC
11   1          IRUNG   1=1ST ORDER R-K,2=2ND ORDER R-K                          CC
12   1          IFREEZ  1= DO FREEZE FRONT CALC. (0=NO CALC.)                    CC
13   5          INPGRP  NUMBER OF TRACE GAS TYPES (FISSION PRODUCTS)             CC
14   1          IRET    1=WRITE RETAIN FILE (0=NO FILE)                          CC
15   49         IFPPLT  RETAIN PLOT FILE UNIT NUMBER                             CC
**                                                                               TD
**                                                                               TD
*TIMING DATA                                                                     TD
01   20.D0      TDMAX   MAXIMUM ALLOWED TIME STEP                                TD
02   1.D-3      TDMIN   MINIMUM ALLOWED TIME STEP                                TD
03   5.D-2      FMCHMX  MAXIMUM MASS CHANGE (%) FOR INTEGRATION                  TD
04   5.D-2      FUCHMX  MAXIMUM GAS TEMP CHANGE FRACTION FOR INTEGRATION         TD
05   4.D2       MAXMST  MAXIMUM MASS OF STEAM CHANGE PER TIME STEP IN PS         TD
**                                                                               CA
**                                                                               CA
*COMPTA (MARKIII-MIDDLE WETWELL COMPARTMENT)                                     CA
01   41.25D0    ZCAF    ELEVATION OF HCU DECK                                    CA
02   1.1589D4   VOLCA   VOLUME OF COMPARTMENT A                                  CA
03   6.D-1      RELHCA  RELATIVE HUMIDITY IN COMPT. A                            CA
04   325.D0     ACAF    AREA OF COMPT. A FLOOR                                   CA
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05   348.DO       ACACB  FLOW AREA BETWEEN COMPT. A AND COMPT. B                    CA
06   139.DO       AWWCA  FLOW AREA BETWEEN WETWELL AND COMPT. A                     CA
07   41.25DO      ZWCAWW CURB HEIGHT ON MIDDLE DECK                                 CA
08   58254.DO     PPUR(1) DRYWELL PURGE PRESSURE VS FLOW (M**3/KG)                  CA
09   87140.DO     PPUR(2)                                                           CA
10   1.05D5       PPUR(3)                                                           CA
11   1.128D5      PPUR(4)                                                           CA
12   1.162D5      PPUR(5)                                                           CA
13   1.162D5      PPUR(6)                                                           CA
14   1.162D5      PPUR(7)                                                           CA
15   1.162D5      PPUR(8)                                                           CA
16   .555DO       WVPUR(1)                                                          CA
17   .515DO       WVPUR(2)                                                          CA
18   .453DO       WVPUR(3)                                                          CA
19   .401DO       WVPUR(4)                                                          CA
20   .330DO       WVPUR(5)                                                          CA
21   .330DO       WVPUR(6)                                                          CA
22   .330DO       WVPUR(7)                                                          CA
23   .330DO       WVPUR(8)                                                          CA
24   2.DO         NPURP  NUMBER OF DRYWELL PURGE PUMPS                              CA
25   47.16DO      ZLPUR  LOW WATER (LOCA) SIGNAL FOR DRYWELL PURGE                  CA
26   1.1437D5     PDWPUR HIGH DRYWELL PRESSURE (LOCA) FOR DRYWELL PURGE             CA
27   6.D3         PDPUR  PRESSURE DIFFERENTIAL SET POINT FOR DRYWELL PURGE          CA
28   30.DO        TDPUR  TIME DELAY FOR DRYWELL PURGE                               CA
29   45.DO        NIGCA  NUMBER OF IGNITERS IN THE COMPT A                          CA
30   8.34DO       XIGCA  AVERAGE DISTANCE FROM IGNITER TO CEILING                   CA
31   6.DO         NIGBCA NUMBER OF IGNITERS IN WETWELL SEEN BY COMPT. A             CA
32   325.DO       ACHCA  CHARACTERISTIC FLOOR AREA FOR BURN CALCULATION             CA
**                                                                                  CB
**                                                                                  CB
*COMPTB (MARKIII-UPPER WETWELL COMPARTMENT)                                         CB
01   63.69DO      ZCBF   ELEVATION OF OPERATING DECK                                CB
02   23766.5DO    VOLCB  VOLUME OF COMPT. B                                         CB
03   6.D-1        RELHCB RELATIVE HUMIDITY IN COMPT.B                               CB
04   64.DO        ZWCBWW CURB HEIGHT ON UPPER DECK                                  CB
05   302.2DO      AWCB   WATER AREA ON CB DECK                                      CB
06   7.5834D4     PCPUR(1) PRESSURE VS FLOW FOR CONTAINMENT PURGE                   CB
07   0.DO         PCPUR(2)                                                          CB
08   0.DO         PCPUR(3)                                                          CB
09   0.DO         PCPUR(4)                                                          CB
10   0.DO         PCPUR(5)                                                          CB
11   0.DO         PCPUR(6)                                                          CB
12   0.DO         PCPUR(7)                                                          CB
13   0.DO         PCPUR(8)                                                          CB
14   1.416DO      WVCPUR(1)                                                         CB
15   1.416DO      WVCPUR(2)                                                         CB
16   1.416DO      WVCPUR(3)                                                         CB
17   1.416DO      WVCPUR(4)                                                         CB
18   1.416DO      WVCPUR(5)                                                         CB
19   1.416DO      WVCPUR(6)                                                         CB
20   1.416DO      WVCPUR(7)                                                         CB
21   1.416DO      WVCPUR(8)                                                         CB
22   1.03D3       VOLUPD   VOLUME OF WATER IN UPPER POOL DUMP                       CB
23   47.16DO      ZLUPD    LOW WATER (LOCA) SIGNAL FOR UPPER POOL DUMP              CB
24   1.1437D5     PDWUPD   HIGH DRYWELL PRESSURE (LOCA) FOR UPPER POOL DUMP         CB
25   225.DO       TDDUMP   TOTAL TIME FOR UPPER POOL DUMP                           CB
26   1.8D3        TDUPD    TIME DELAY FOR UPPER POOL DUMP                           CB
27   56.25DO      ZCBF1    ELEVATION OF UPPER POOL FLOOR                            CB
28   1031.DO      VLAUPD   VOLUME OF WATER REMAINING IN UPPER POOL AFTER            CB
**                         UPPER POOL DUMP                                          CB
29   18.DO        NIGCB  NUMBER OF IGNITERS IN THE COMPT B                          CB
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30    .3D0          XIGCB  AVERAGE DISTANCE FROM IGNITER TO CEILING                CB
31    16.D0         NIGBCB NUMBER OF IGNITERS IN COMPT A SEEN IN COMPT B            CB
32    1121.9D0      ACHCB  CHARACTERISTIC FLOOR AREA FOR BURN CALCULATION          CB
**                                                                                 HS
**                                                                                 HS
*HTSINKS                                                                           HS
** REFER TO DRAWING IN VOL II OF MAAP USERS MANUAL ON MARKIII HEAT SINKS
01    186.6D0       AHS1     AREA OF WALL #1                                       HS
02    1074.8D0      AHS2     AREA OF WALL #2                                       HS
03    492.3D0       AHS3     AREA OF WALL #3                                       HS
04    883.8D0       AHS4     AREA OF WALL #4                                       HS
05    2735.1D0      AHS5     AREA OF WALL #5                                       HS
06    3411.0D0      AHS6     AREA OF WALL #6                                       HS
07    371.1D0       AHS7     AREA OF WALL #7                                       HS
08    289.6D0       AHS8     AREA OF WALL #8                                       HS
09    2.077D0       KHS1     THERMAL CONDUCTIVITY OF WALL #1                       HS
10    2.077D0       KHS2     THERMAL CONDUCTIVITY OF WALL #2                       HS
11    2.077D0       KHS3     THERMAL CONDUCTIVITY OF WALL #3                       HS
12    2.077D0       KHS4     THERMAL CONDUCTIVITY OF WALL #4                       HS
13    2.077D0       KHS5     THERMAL CONDUCTIVITY OF WALL #5                       HS
14    2.077D0       KHS6     THERMAL CONDUCTIVITY OF WALL #6                       HS
15    2.077D0       KHS7     THERMAL CONDUCTIVITY OF WALL #7                       HS
16    2.077D0       KHS8     THERMAL CONDUCTIVITY OF WALL #8                       HS
17    1.753D0       XHS1     THICKNESS OF WALL #1                                  HS
18    1.524D0       XHS2     THICKNESS OF WALL #2                                  HS
19    1.524D0       XHS3     THICKNESS OF WALL #3                                  HS
20    1.067D0       XHS4     THICKNESS OF WALL #4                                  HS
21    1.067D0       XHS5     THICKNESS OF WALL #5                                  HS
22    .762D0        XHS6     THICKNESS OF WALL #6                                  HS
23    .61D0         XHS7     THICKNESS OF WALL #7                                  HS
24    1.295D0       XHS8     THICKNESS OF WALL #8                                  HS
25    0.D0          XLHSI1    INNER LINER THICKNESS FOR WALL #1                    HS
26    0.00634D0     XLHSI2    INNER LINER THICKNESS FOR WALL #2                    HS
27    0.00634D0     XLHSI3    INNER LINER THICKNESS FOR WALL #3                    HS
28    0.00634D0     XLHSI4    INNER LINER THICKNESS FOR WALL #4                    HS
29    0.00634D0     XLHSI5    INNER LINER THICKNESS FOR WALL #5                    HS
30    0.00634D0     XLHSI6    INNER LINER THICKNESS FOR WALL #6                    HS
31    0.D0          XLHSI7    INNER LINER THICKNESS FOR WALL #7                    HS
32    0.00634D0     XLHSI8    INNER LINER THICKNESS FOR WALL #8                    HS
33    0.00634D0     XLHSO1    OUTER LINER THICKNESS FOR WALL #1                    HS
34    0.D0          XLHSO2    OUTER LINER THICKNESS FOR WALL #2                    HS
35    0.00634D0     XLHSO3    OUTER LINER THICKNESS FOR WALL #3                    HS
36    0.D0          XLHSO4    OUTER LINER THICKNESS FOR WALL #4                    HS
37    0.D0          XLHSO5    OUTER LINER THICKNESS FOR WALL #5                    HS
38    0.D0          XLHSO6    OUTER LINER THICKNESS FOR WALL #6                    HS
39    0.0612D0      XLHSO7    OUTER LINER THICKNESS FOR WALL #7                    HS
40    0.00634D0     XLHSO8    OUTER LINER THICKNESS FOR WALL #8                    HS
41    2300.D0       DHS1      DENSITY OF WALL #1                                   HS
42    2300.D0       DHS2      DENSITY OF WALL #2                                   HS
43    2300.D0       DHS3      DENSITY OF WALL #3                                   HS
44    2300.D0       DHS4      DENSITY OF WALL #4                                   HS
45    2300.D0       DHS5      DENSITY OF WALL #5                                   HS
46    2300.D0       DHS6      DENSITY OF WALL #6                                   HS
47    2300.D0       DHS7      DENSITY OF WALL #7                                   HS
48    2300.D0       DHS8      DENSITY OF WALL #8                                   HS
49    880.D0        CPHS1     SPECIFIC HEAT FOR WALL #1                            HS
50    880.D0        CPHS2     SPECIFIC HEAT FOR WALL #2                            HS
51    880.D0        CPHS3     SPECIFIC HEAT FOR WALL #3                            HS
52    880.D0        CPHS4     SPECIFIC HEAT FOR WALL #4                            HS
53    880.D0        CPHS5     SPECIFIC HEAT FOR WALL #5                            HS
54    880.D0        CPHS6     SPECIFIC HEAT FOR WALL #6                            HS
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55  880.D0      CPHS7   SPECIFIC HEAT FOR WALL #7                           HS
56  880.D0      CPHS8   SPECIFIC HEAT FOR WALL #8                           HS
**ALL OF THESE EQUIPMENT HEAT SINKS ARE LOCATED IN GAS VOL. OF COMPARTMENT
57  0.D0        MEQPD   MASS OF EQUIPMENT IN PEDESTAL                       HS
58  151000.D0   MEQDW   MASS OF EQUIPMENT IN DRYWELL                        HS
59  0.D0        MEQWW   MASS OF EQUIPMENT IN WETWELL                        HS
60  342462.D0   MEQCA   MASS OF EQUIPMENT IN COMPT A                        HS
61  1.9581D6    MEQCB   MASS OF EQUIPMENT IN COMPT B                        HS
62  0.D0        AEQPD   AREA OF EQUIPMENT IN PEDESTAL                       HS
63  4153.D0     AEQDW   AREA OF EQUIPMENT IN DRYWELL                        HS
64  0.D0        AEQWW   AREA OF EQUIPMENT IN WETWELL                        HS
65  9.7177D3    AEQCA   AREA OF EQUIPMENT IN COMPT A                        HS
66  1189.2D0    AEQCB   AREA OF EQUIPMENT IN COMPT B                        HS
67  50.D0       HTOUTW  HEAT TRANSFER COEFF. AT OUTER WALL                  HS
68  0.D0        RGAP    INNER LINER TO WALL GAP RESISTANCE #1               HS
69  0.D0        RGAP    INNER LINER TO WALL GAP RESISTANCE #2               HS
70  0.D0        RGAP    INNER LINER TO WALL GAP RESISTANCE #3               HS
71  0.D0        RGAP    INNER LINER TO WALL GAP RESISTANCE #4               HS
72  0.D0        RGAP    INNER LINER TO WALL GAP RESISTANCE #5               HS
73  0.D0        RGAP    INNER LINER TO WALL GAP RESISTANCE #6               HS
74  0.D0        RGAP    INNER LINER TO WALL GAP RESISTANCE #7               HS
75  0.D0        RGAP    INNER LINER TO WALL GAP RESISTANCE #8               HS
76  0.D0        RGAP    OUTER LINER TO WALL GAP RESISTANCE #1               HS
77  0.D0        RGAP    OUTER LINER TO WALL GAP RESISTANCE #2               HS
78  0.D0        RGAP    OUTER LINER TO WALL GAP RESISTANCE #3               HS
79  0.D0        RGAP    OUTER LINER TO WALL GAP RESISTANCE #4               HS
80  0.D0        RGAP    OUTER LINER TO WALL GAP RESISTANCE #5               HS
81  0.D0        RGAP    OUTER LINER TO WALL GAP RESISTANCE #6               HS
82  0.D0        RGAP    OUTER LINER TO WALL GAP RESISTANCE #7               HS
83  0.D0        RGAP    OUTER LINER TO WALL GAP RESISTANCE #8               HS
84  0.D0        MEQWWS  MASS OF EQUIP. HEAT SINK WETWELL (SUBMERGED)        HS
85  0.D0        AEQWWS  AREA OF EQUIP. HEAT SINK WETWELL (SUBMERGED)        HS
86  1.45D2      AHSPS1  AREA OF RPV IN GAS SPACE OF VESSEL                  HS
87  1.75D2      AHSPS2  AREA OF RPV IN DOWNCOMER REGION                     HS
88  6.5D0       AHSPS3  AREA OF RPV IN LOWER PLENUM                         HS
89  2.24D5      MRPV1   RPV WALL MASS IN UPPER DOME REGION                  HS
90  2.22D5      MRPV2   RPV WALL MASS IN DOWNCOMER REGION                   HS
91  9.D4        MRPV3   RPV WALL MASS IN LOWER PLENUM REGION                HS
**                                                                          HS
**                                                                          HS
*MODEL PARAMETERS FOR BWR
01  .005D0      FRCOEF  FRICTION COEFFICIENT FOR CORIUM IN VEAIL            MO
02  2.0D-1      FMAXCP  FRACTION OF TOTAL CORE MASS WHICH MUST MELT
**                      TO FAIL THE CORE PLATE
03  50.D0       HTBLAD  FUEL CHANNEL TO CONTROL BLADE HEAT TRANS. COEFF     MO
04  300.D0      HTFB    FILM BOILING HEAT TRANS. COEFF.                     MO
05  0.D0        FBLOCK  FUEL CHANNEL BLOCKAGE PARAMETER                     MO
**                      0=BLOCKAGE AT TZOOFF,1=NO BLOCKAGE                  MO
06  2300.D0     TZOOFF  OXIDATION CUT-OFF TEMPERATURE                       MO
07  .3D0        FACPF   FRACTION OF CORE PLATE AREA THAT FAILS              MO
08  5.D0        CDBPD   FLAME BUOYANCY DRAG COEFFICIENT IN THE PEDESTAL     MO
09  5.D0        CDBDW   FLAME BUOYANCY DRAG COEFFICIENT IN THE DRYWELL      MO
10  5.D0        CDBWW   FLAME BUOYANCY DRAG COEFFICIENT IN THE WETWELL      MO
11  5.D0        CDBCA   FLAME BUOYANCY DRAG COEFFICIENT IN COMPARTMENT A    MO
12  5.D0        CDBCB   FLAME BUOYANCY DRAG COEFFICIENT IN COMPARTMENT B    MO
13  .10D0       XCNREF  CORIUM REFERENCE THERMAL BOUNDARY LAYER THICKNESS   MO
14  1.D3        HTCMCR  CORIUM-CRUST HEAT TRANSF. COEFF. USED IN DECOMP     MO
15  0.05D0      XCMX    MINIMUM CORIUM THICKNESS ON DRYWELL FLOOR AND PED   MO
**                      FLOOR (MARK II ONLY)                                MO
16  0.01D0      XDCMSP  PARTICLE SIZE (DIAMETER) FOR CORIUM AS IT FALLS     MO
**                      INTO SUPPRESSION POOL (MARK II ONLY)                MO
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17   983.D0      TCFLAM   CRITICAL FLAME TEMPERATURE                              MO
18   1.53D0      FCHTUR   CHURN-TURBULENT CRITICAL FLOW PARAMETER                 MO
19   3.7D0       FDROP    DROPLET CRITICAL FLOW PARAMETER                         MO
20   3.D0        FFLOOD   FLOODING FLOW PARAMETER                                 MO
21   1.D0        FSPAR    PARAMETER FOR BOTTOM-SPARGED STEAM VOID FRACTION        MO
22   2.D0        FVOL     PARAMETER FOR VOLUME SOURCE VOID FRACTION MODEL         MO
23   5.D-1       TTENTR   ENTRAINMENT EFFECTIVE EMPTYING TIME                     MO
24   .9D0        EW       EMISSIVITY OF WATER                                     MO
25   .85D0       EWL      EMISSIVITY OF WALL                                      MO
26   .85D0       ECM      EMISSIVITY OF CORIUM                                    MO
27   .6D0        EG       EMISSIVITY OF GAS                                       MO
28   .85D0       EEQ      EMISSIVITY OF EQUIPMENT                                 MO
29   0.5D0       FOVER    FRACTION OF CORE SPRAY FLOW ALLOWED TO BYPASS COREMO
30   1.D0        NPE      NUMBER OF PENETRATIONS FAILED IN LOWER HEAD             MO
31   2.D0        FCDCDW   DOWNCOMER PERIMETER PER METER FROM PEDESTAL DOOR        MO
**                        (MARK II ONLY)                                          MO
32   0.14D0      FCHF     COEFFICIENT FOR CHF CORRELATION IN PLSTM                MO
33   .75D0       FCDBRK   DISCHARGE COEFFICIENT FOR PIPE BREAK                    MO
34   .33D0       FENTR    NUMBER TO MULTIPLY KUTATELADZE CRITERION BY TO          MO
**                        REPRESENT DIFFICULTY (GT 1.D0) OR EASE (LT 1.D0)        MO
**                        FOR MATERIAL TO BE BLOWN OUT OF CAVITY                  MO
35   1.00        SCALU    SCALING FACTOR FOR ALL BURNING VELOCITIES               MO
36   1.00        SCALH    SCALING FACTOR FOR HT COEFFICIENTS TO PASSIVE           MO
**                        HEAT SINKS                                              MO
37   2.0D0       FUMIN    CLADDING SURFACE MULTIPLIER                             MO
**                                                                                MO
*CONCRETE PROPERTIES                                                              CN
01   56.D0       MOLWCN   MOLECULAR WEIGHT OF CONCRETE                            CN
02   1743.D0     TCNMP    MELTING TEMPERATURE OF CONCRETE                         CN
03   .8D6        LHRCN    REACTION ENERGY FOR CONCRETE DECOMPOSITION              CN
04   65.D0       DCFWCN   FREE WATER DENSITY IN CONCRETE                          CN
05   65.D0       DCCWCN   COMBINED WATER DENSITY IN CONCRETE                      CN
06   572.D0      DCC2CN   CO2 DENSITY IN CONCRETE                                 CN
07   1.D6        LHCN     LATENT HEAT TO MELT CONCRETE                            CN
**
*FISSION PRODUCTS                                                                 FI
01   .028D0      FQP(1) PERCENT POWER IN FISSION PRODUCT GROUP 1                  FI
02   .176D0      FQP(2) PERCENT POWER IN FISSION PRODUCT GROUP 2                  FI
03   .019D0      FQP(3) PERCENT POWER IN FISSION PRODUCT GROUP 3                  FI
04   0.D0        FQP(4) PERCENT POWER IN FISSION PRODUCT GROUP 4                  FI
05   0.D0        FQP(5) PERCENT POWER IN FISSION PRODUCT GROUP 5                  FI
06   439.3       MFP(1) MASS OF FISSION PRODUCT GROUP 1 -NOBLES                   FI
07   237.7       MFP(2) MASS OF FISSION PRODUCT GROUP 2 -CS+I                     FI
08   37.1        MFP(3) MASS OF FISSION PRODUCT GROUP 3 -TE                       FI
09   178.7       MFP(4) MASS OF FISSION PRODUCT GROUP 4 -SR                       FI
10   435.0       MFP(5) MASS OF FISSION PRODUCT GROUP 5 -RU                       FI
11   1190.D0     MSMO(1) MASS OF SN IN CORE REGION                                FI
12   268.D0      MSMO(2) MASS OF MN IN CORE REGION                                FI
13   0.0D0       FDSP(1)SPRAY REMOVAL LAMDA FOR FP GROUP 1                        FI
14   0.0D0       FDSP(2)SPRAY REMOVAL LAMDA FOR FP GROUP 2                        FI
15   0.0D0       FDSP(3)SPRAY REMOVAL LAMDA FOR FP GROUP 3                        FI
16   0.0D0       FDSP(4)SPRAY REMOVAL LAMDA FOR FP GROUP 4                        FI
17   0.0D0       FDSP(5)SPRAY REMOVAL LAMDA FOR FP GROUP 5                        FI
18   600.D0      FDFSP(1) DRYWELL VENTS DECON. FACTOR FOR FP GROUP 1              FI
19   600.D0      FDFSP(2) DRYWELL VENTS DECON. FACTOR FOR FP GROUP 2              FI
20   600.0D0     FDFSP(3) DRYWELL VENTS DECON. FACTOR FOR FP GROUP 3              FI
21   600.0D0     FDFSP(4) DRYWELL VENTS DECON. FACTOR FOR FP GROUP 4              FI
22   600.0D0     FDFSP(5) DRYWELL VENTS DECON. FACTOR FOR FP GROUP 5              FI
23   1000.D0     FDFRV(1) SRV DECON. FACTOR FOR FP GROUP 1                        FI
24   1000.D0     FDFRV(2) SRV DECON. FACTOR FOR FP GROUP 2                        FI
25   1000.0D0    FDFRV(3) SRV DECON. FACTOR FOR FP GROUP 3                        FI
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26   1000.0D0  FDFRV(4) SRV DECON. FACTOR FOR FP GROUP 4        FI
27   1000.0D0  FDFRV(5) SRV DECON. FACTOR FOR FP GROUP 5        FI
**  28 - 31 ARE FOUR CONSTANTS IN HENRY-EPSTEIN MODEL
**                                                              FI
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APPENDIX B

Supplemental Plots for the Base Accident Sequences

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SUPPLEMENTAL PLOTS FOR SEQUENCE T_1QUV

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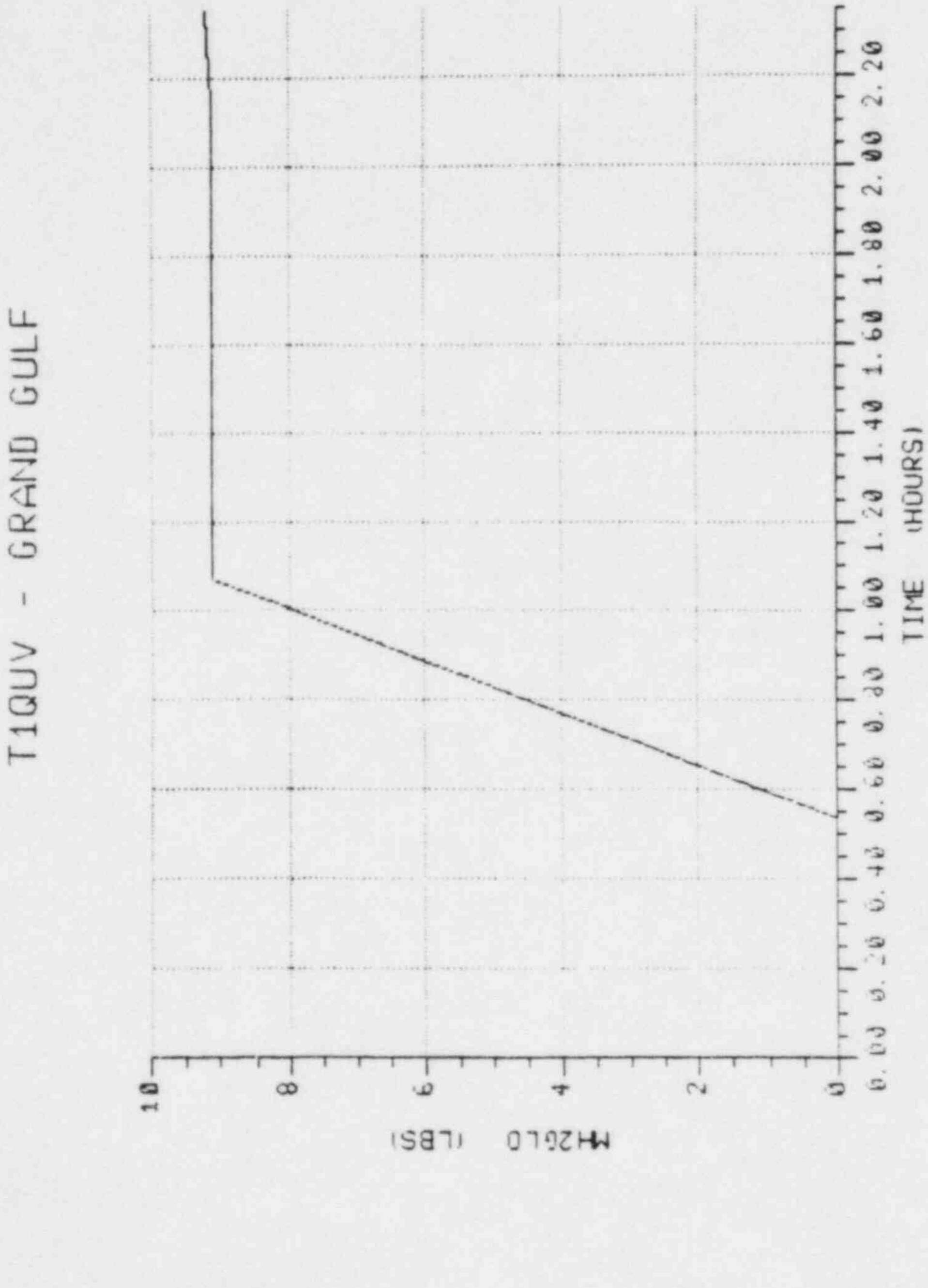

Fig. B.1 Total H_2 generated.

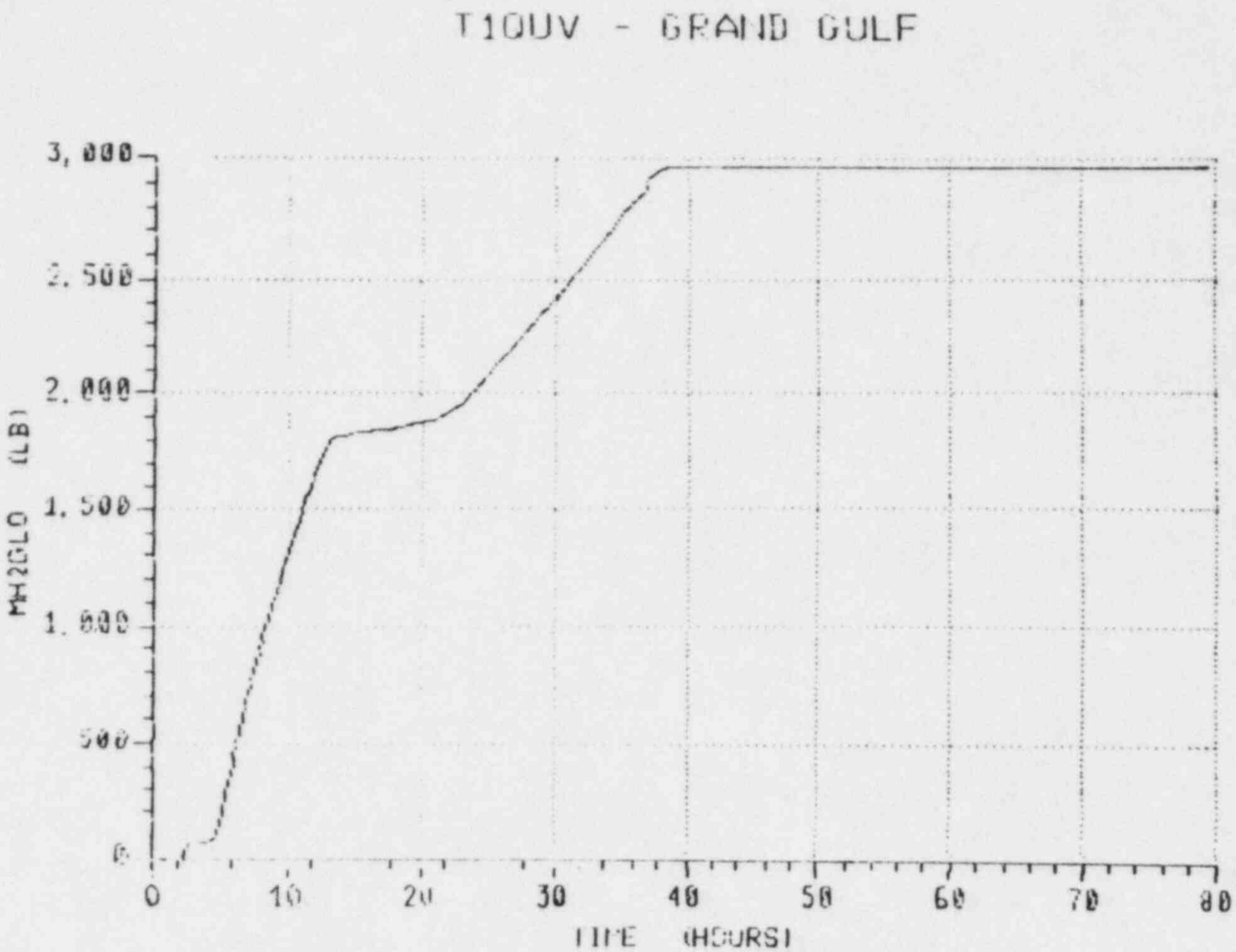

Fig. B.2 Total H_2 generated.

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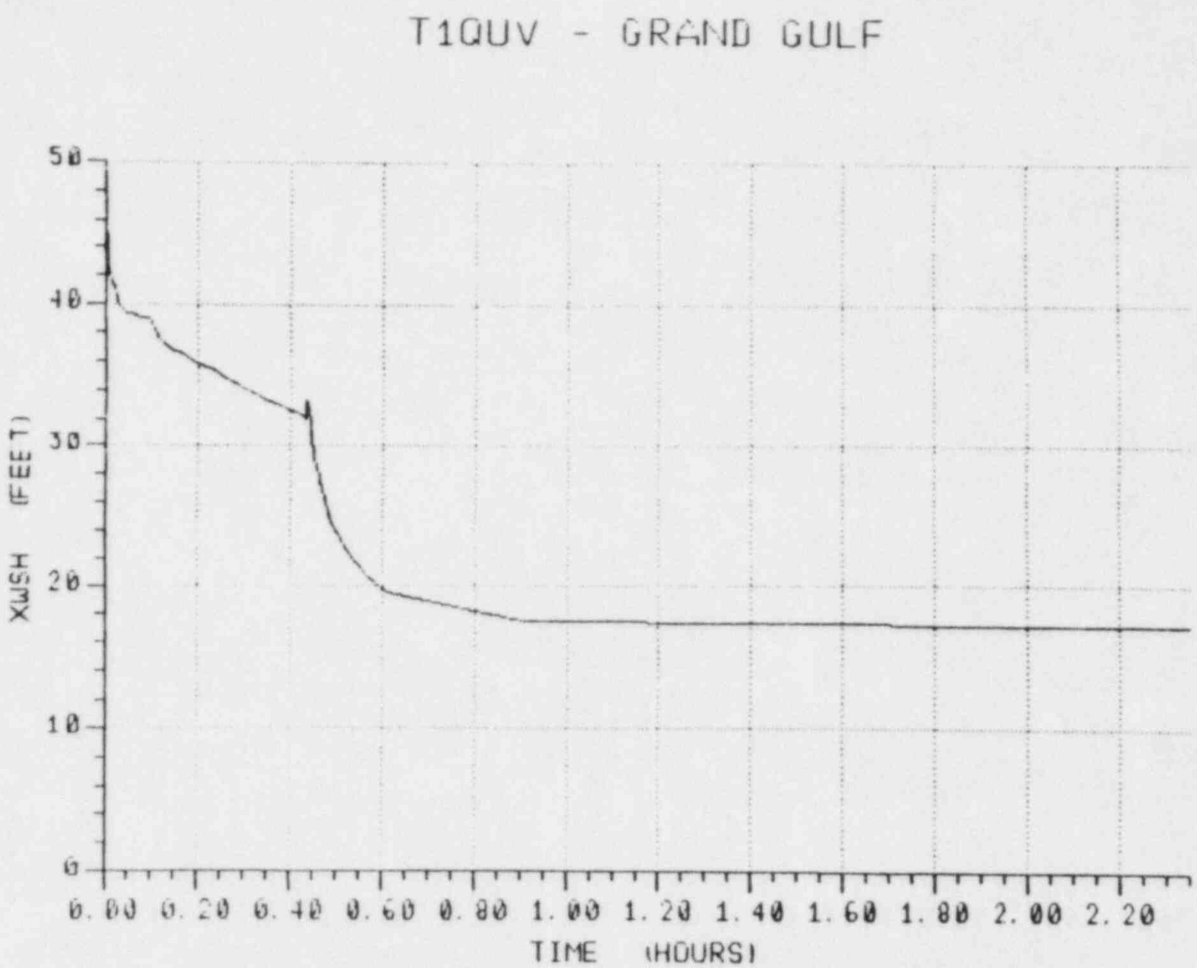

Fig. B.3 Reactor vessel water level.

T1QUV - GRAND GULF

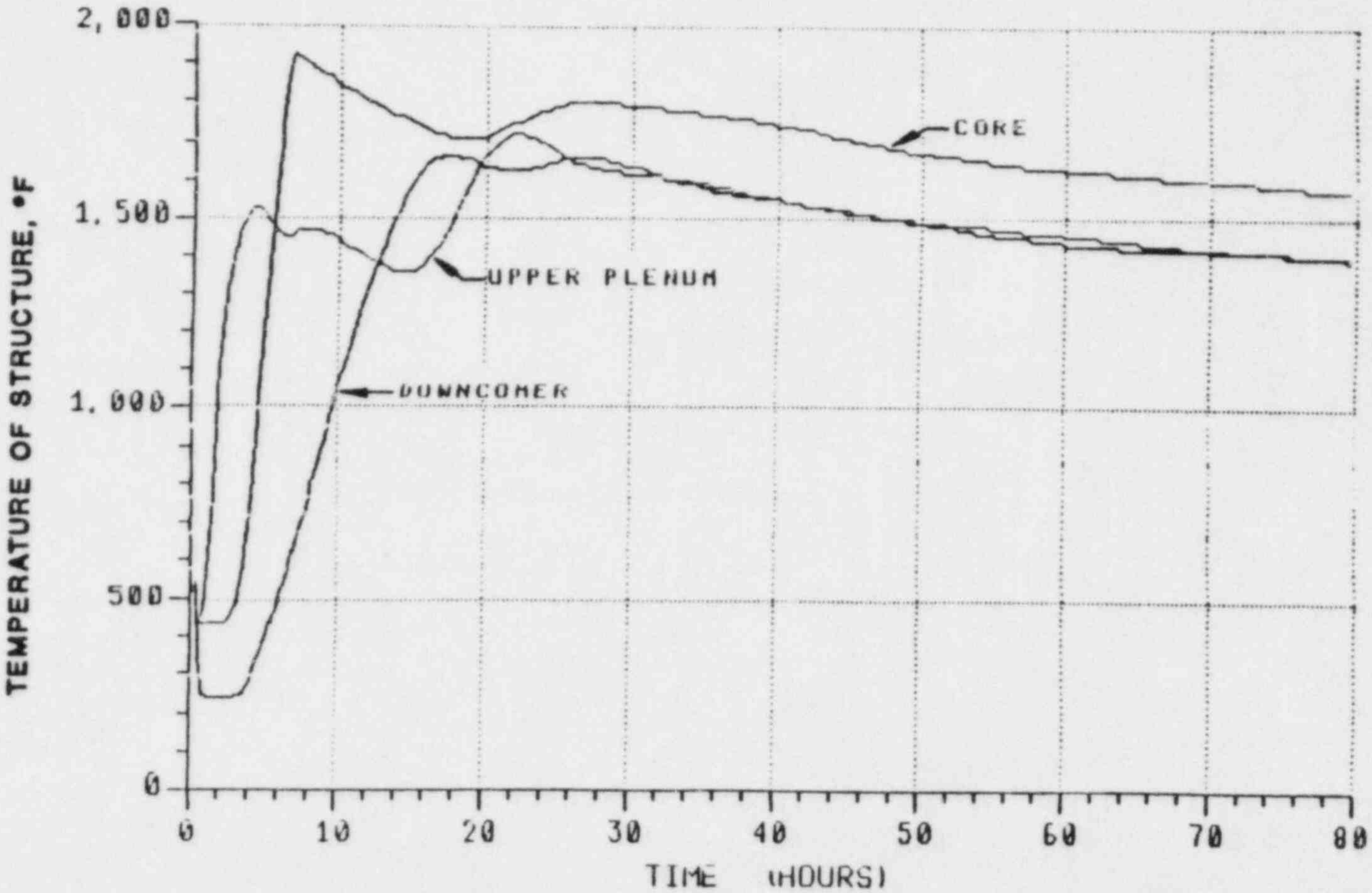

Fig. B.4 Temperature of structure, °F.

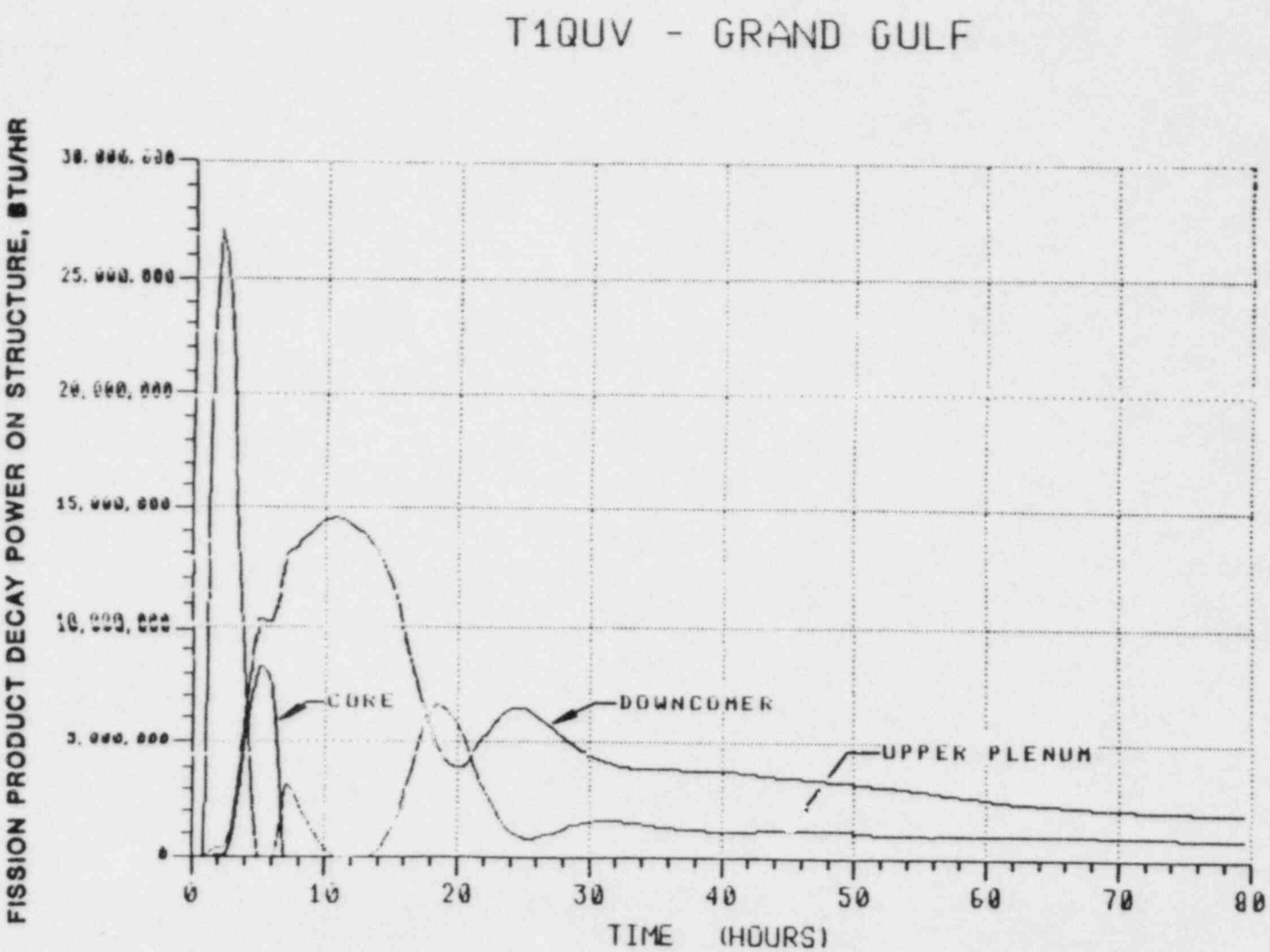

Fig. B.5 Fission product decay power on structure, Btu/hr.

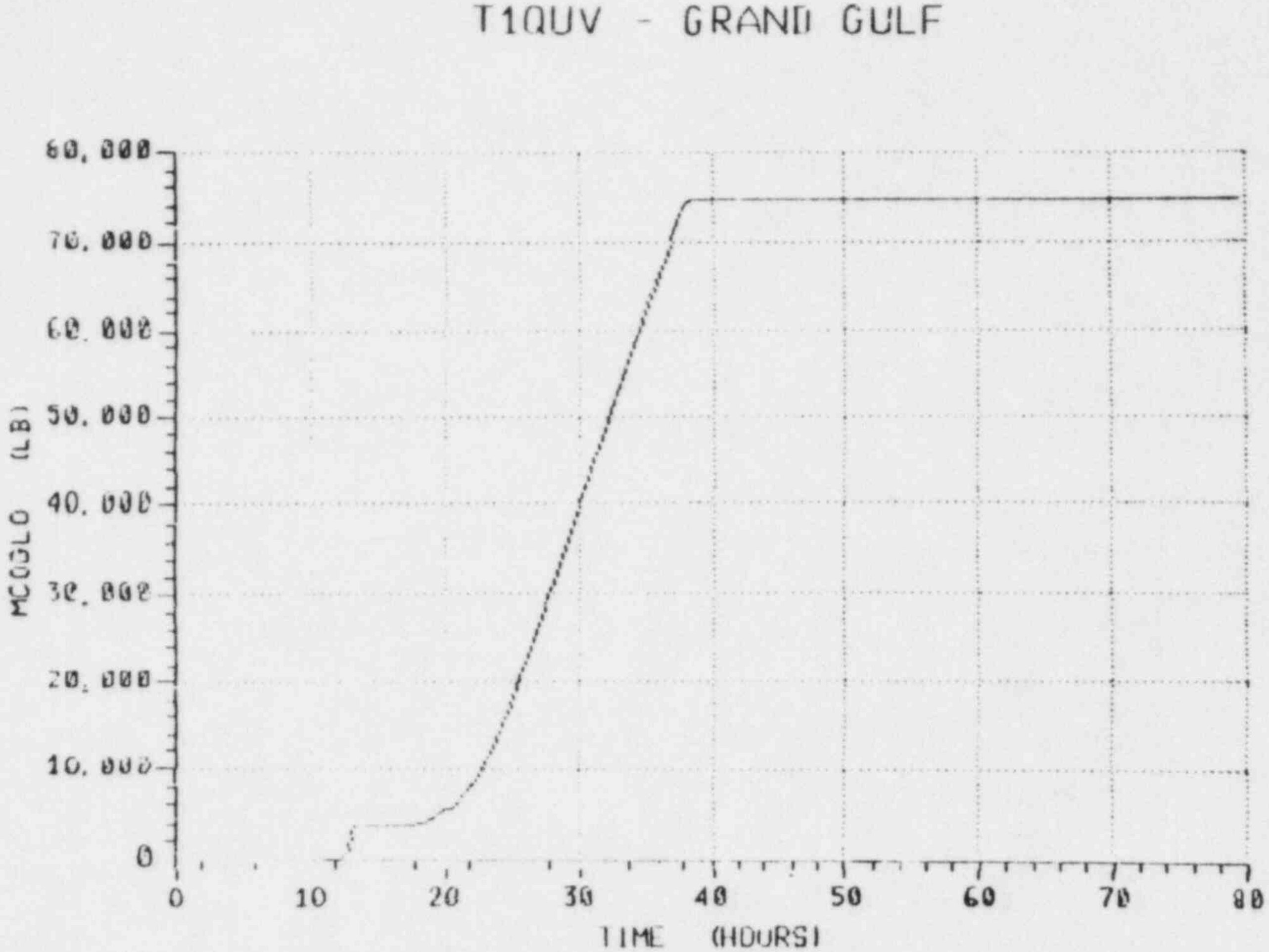

Fig. B.6 Total CO generated.

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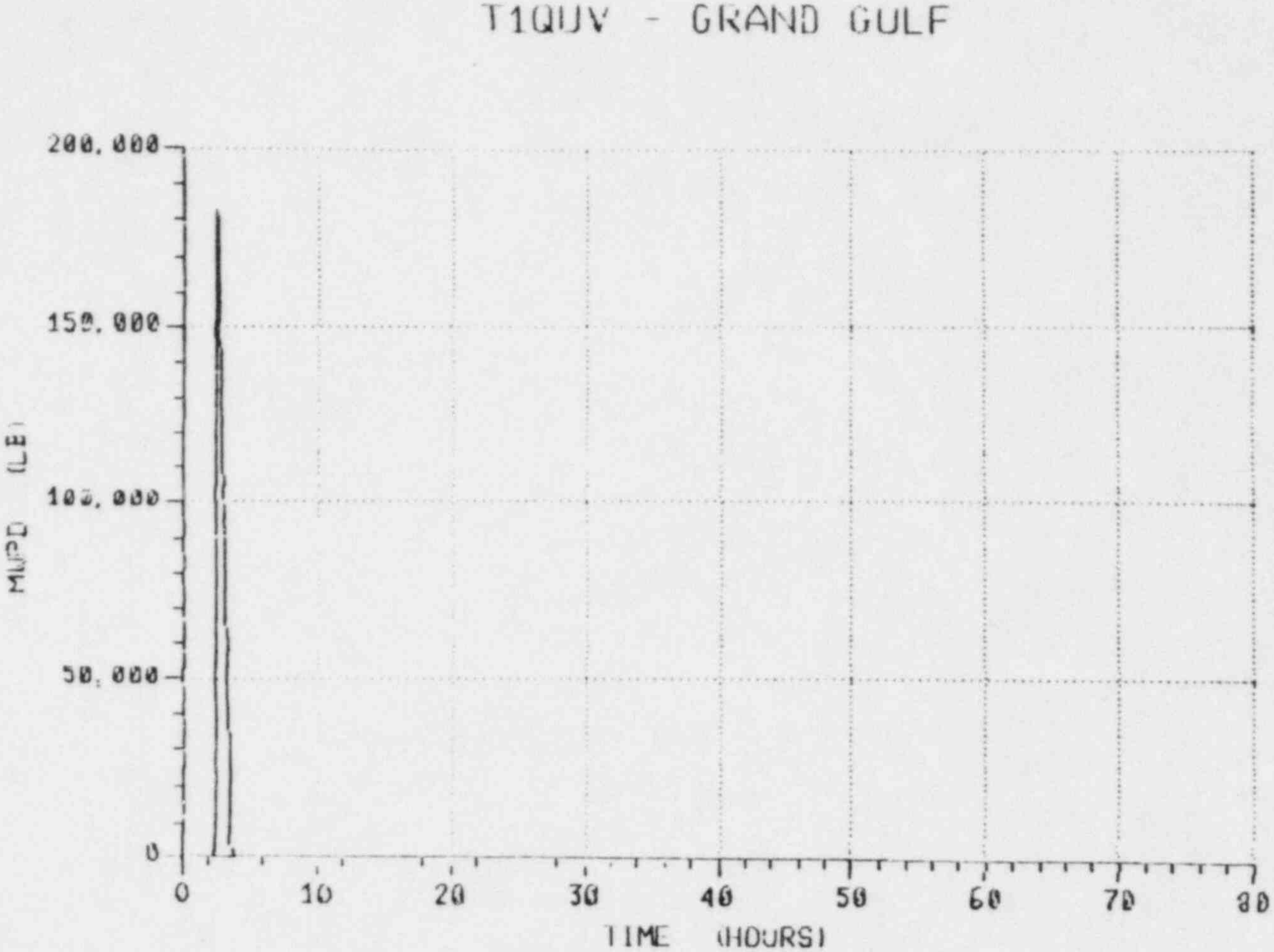

Fig. B.7 Mass of water in the pedestal.

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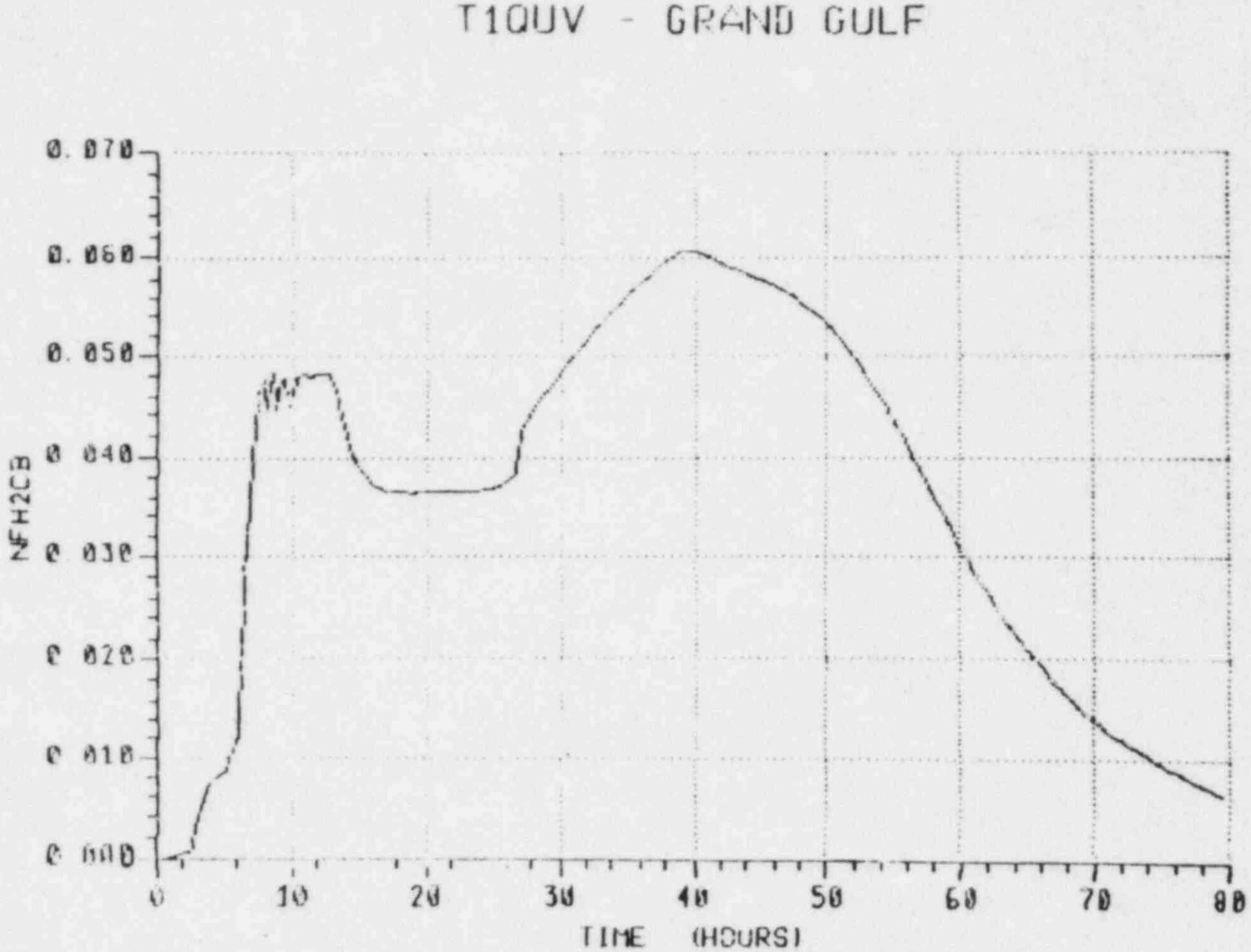

Fig. B.8 Mole fraction of H_2 in Compartment B.

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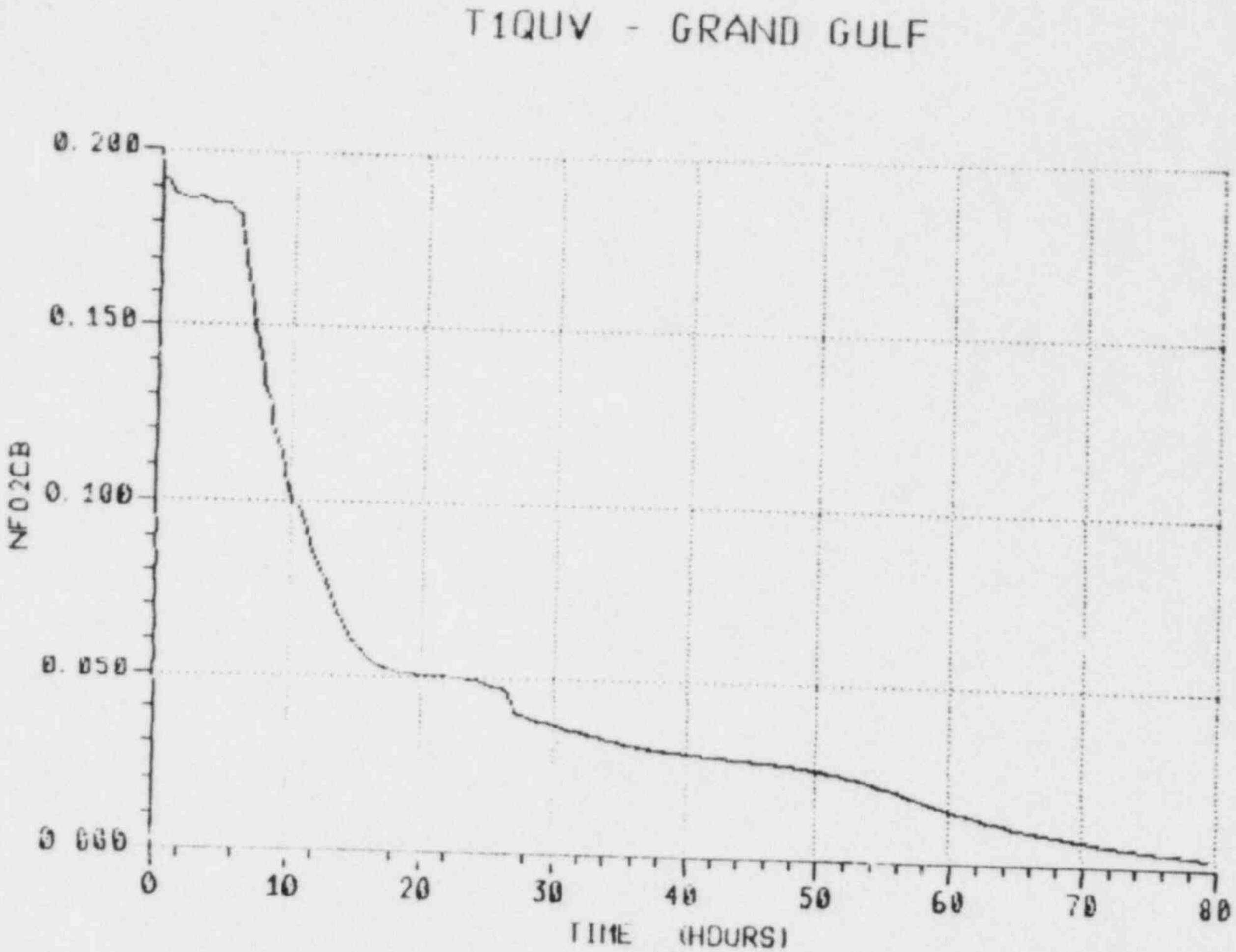

Fig. B.9 Mole fraction of O_2 in Compartment B.

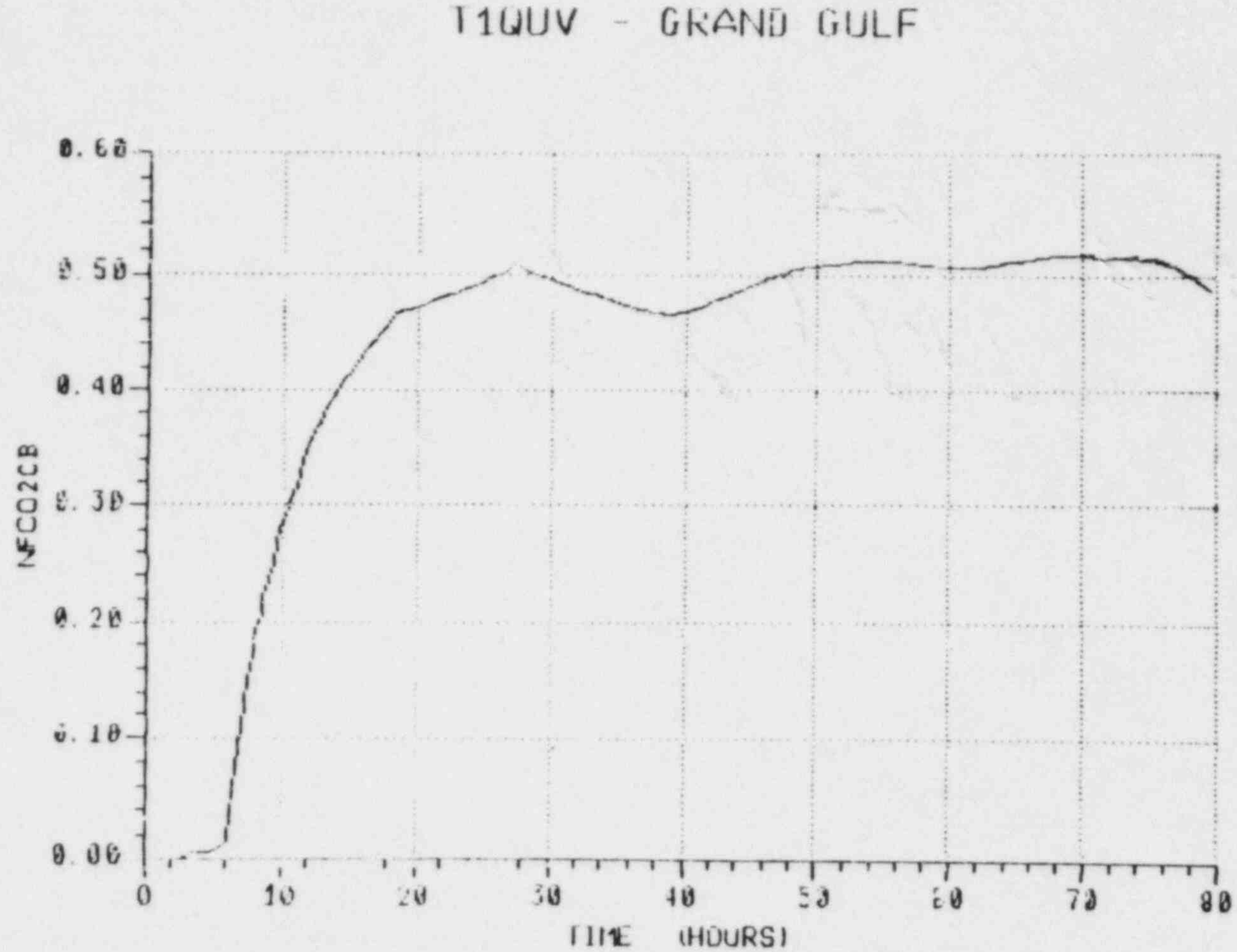

Fig. B.10 Mole fraction of CO_2 in Compartment B.

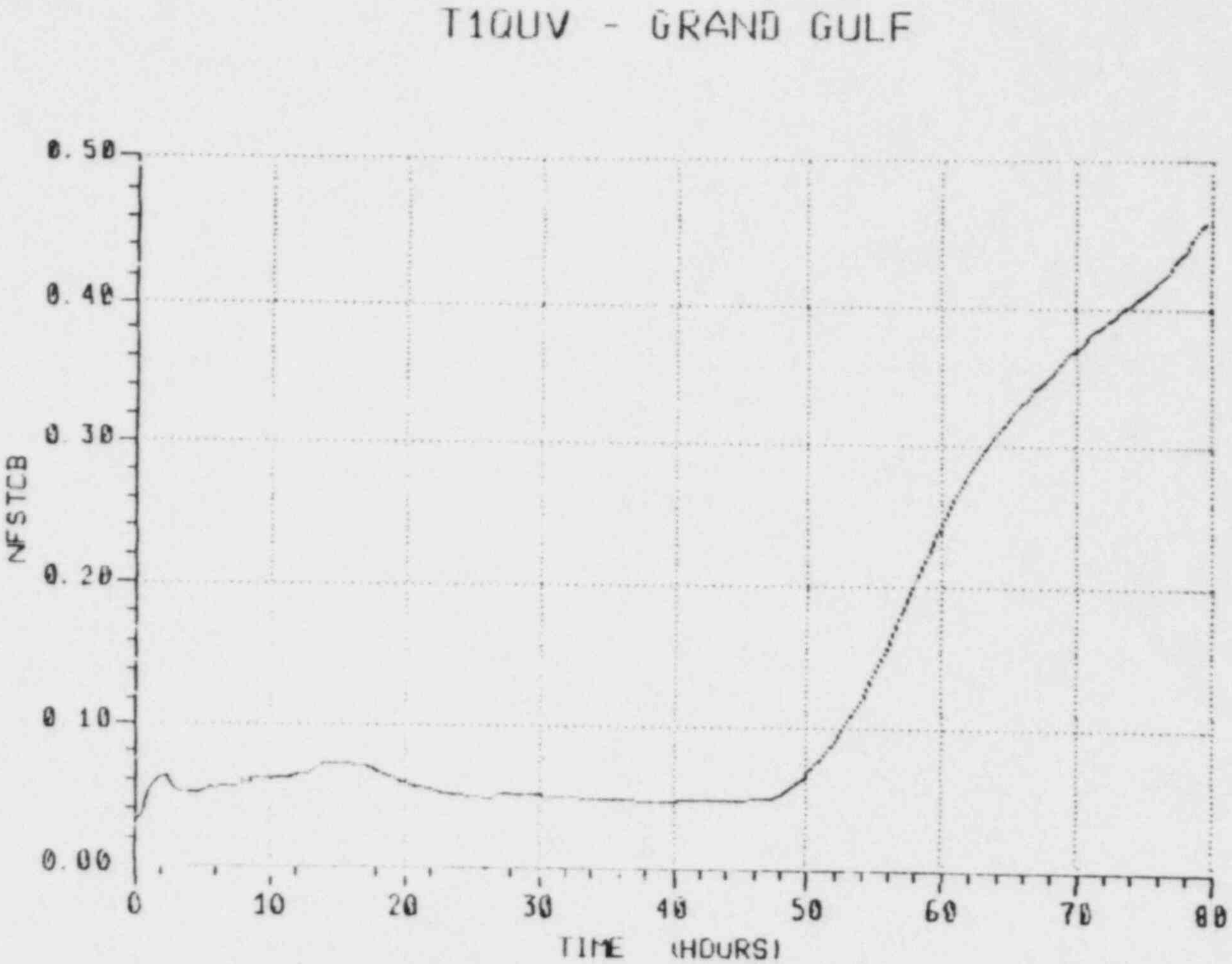

Fig. B.11 Mole fraction of steam in Compartment B.

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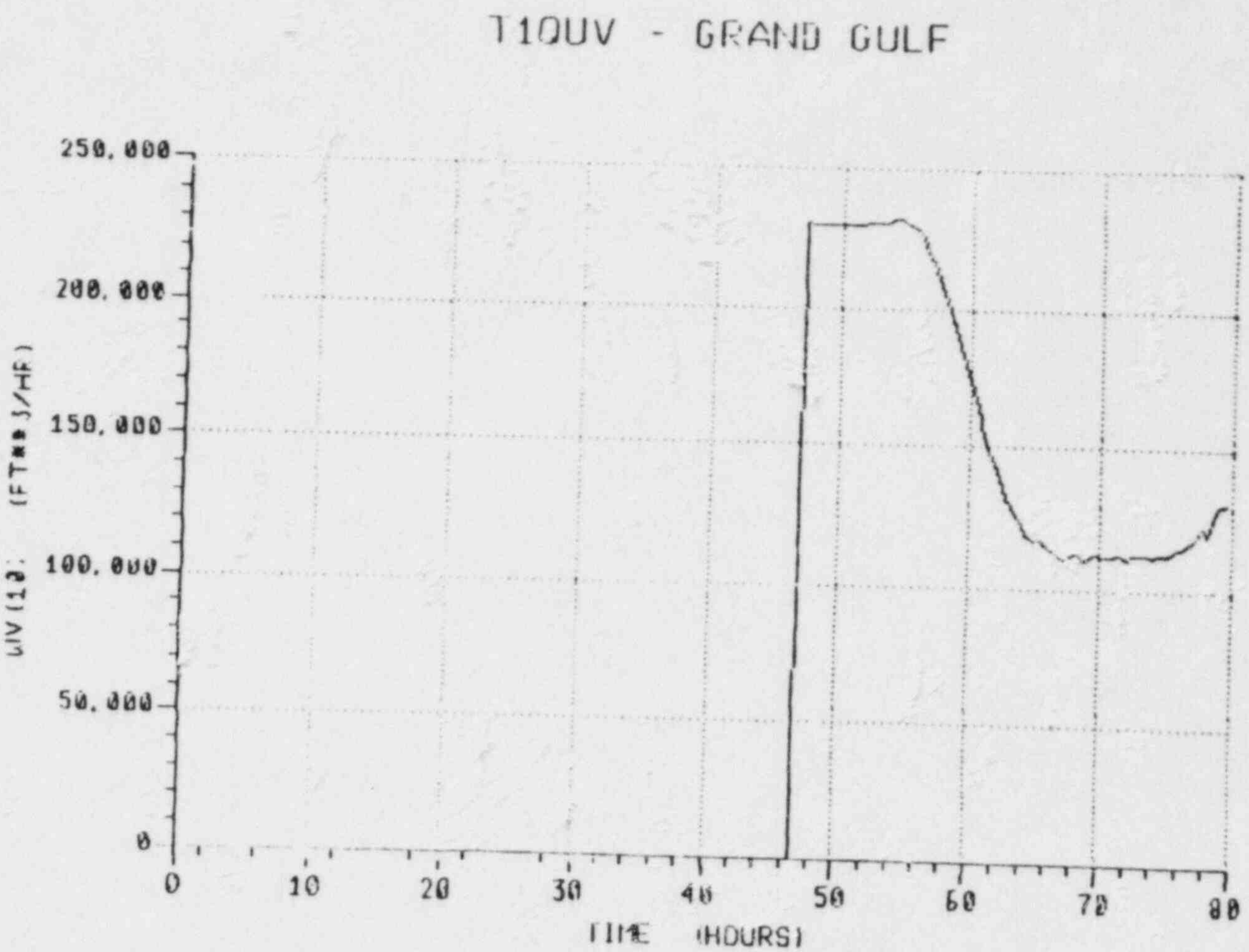

Fig. B.12 Volumetric flow out of containment.

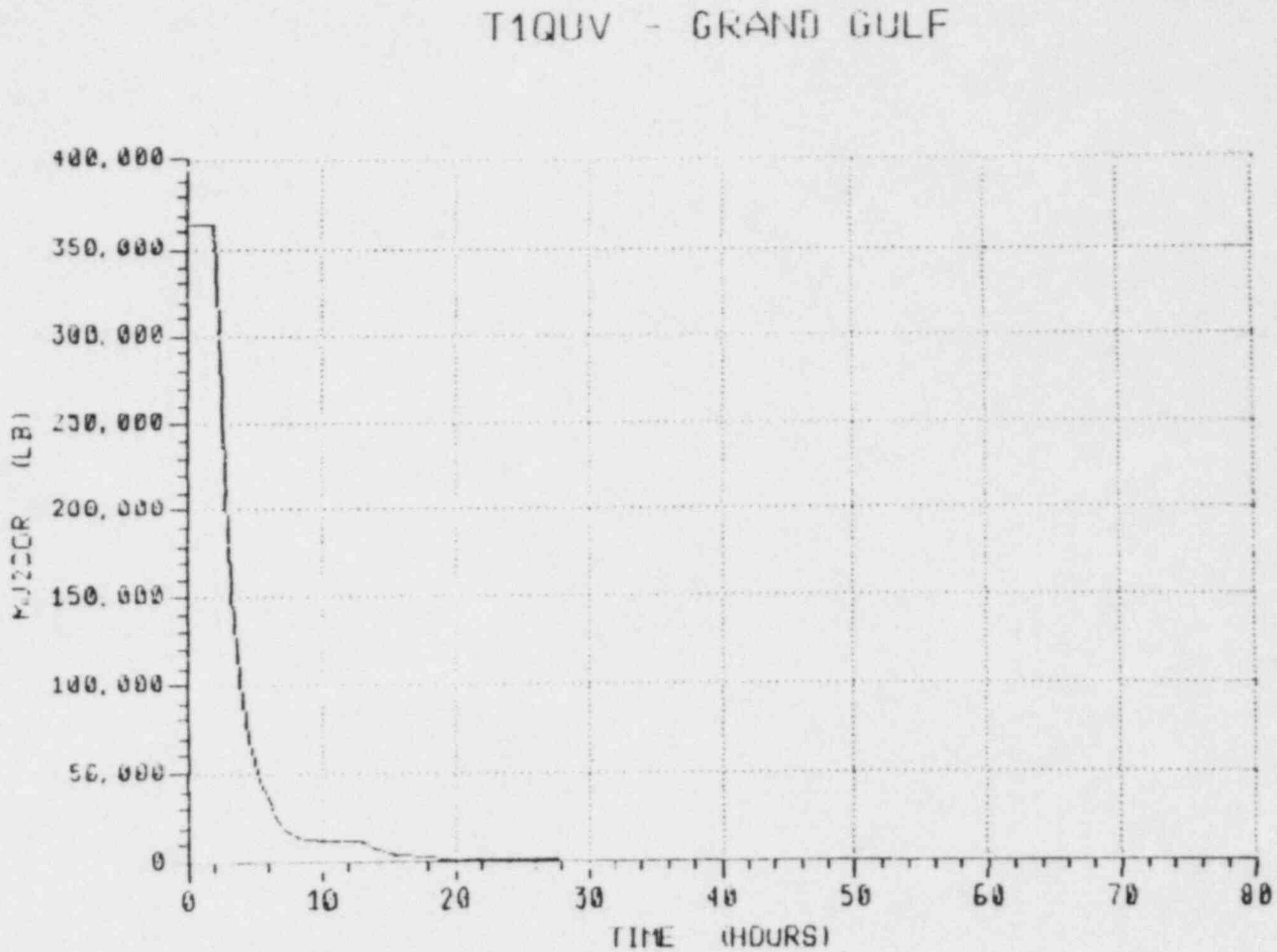

Fig. B.13 Mass of UO_2 in core region.

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SUPPLEMENTAL PLOTS FOR SEQUENCE AE

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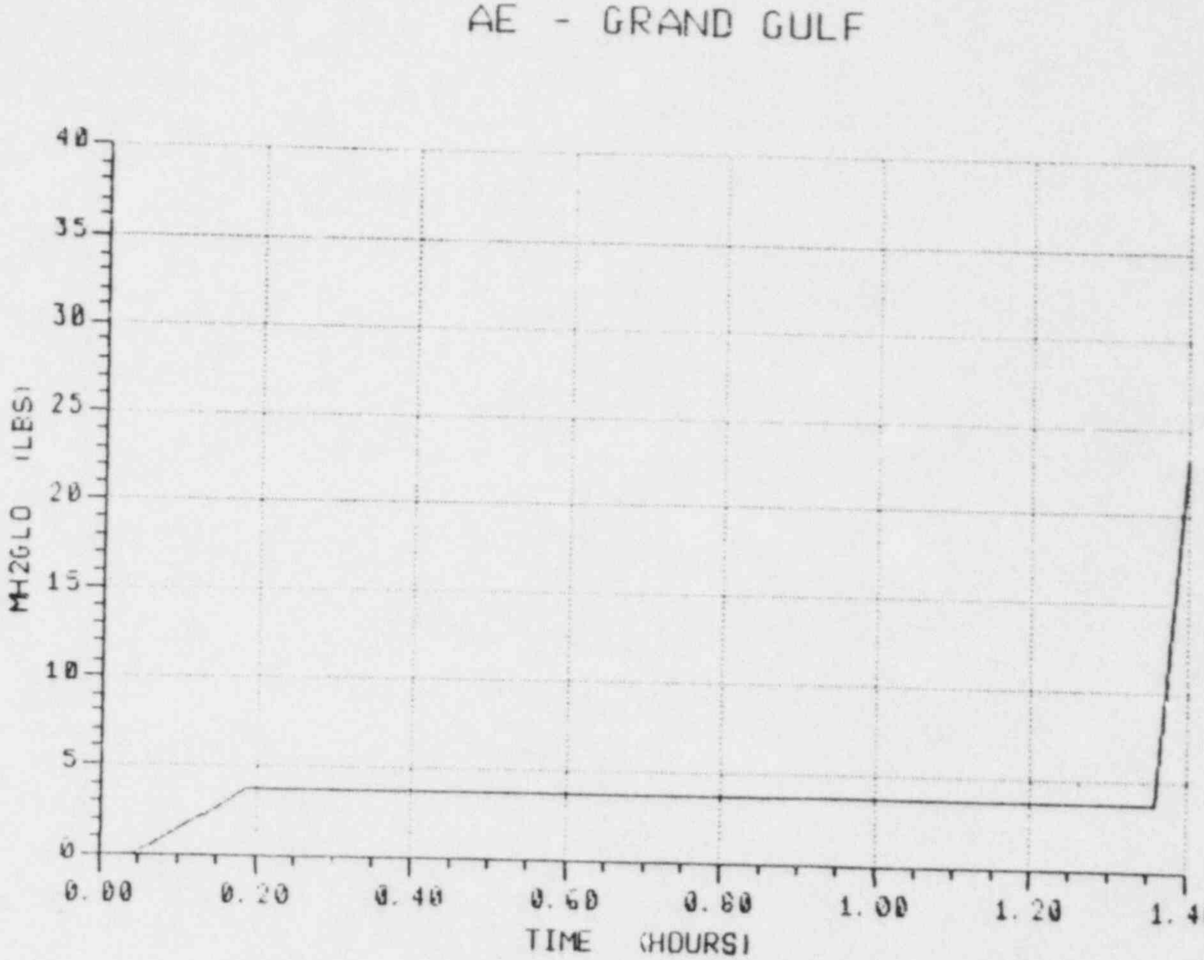

Fig. B.14 Total H_2 generated.

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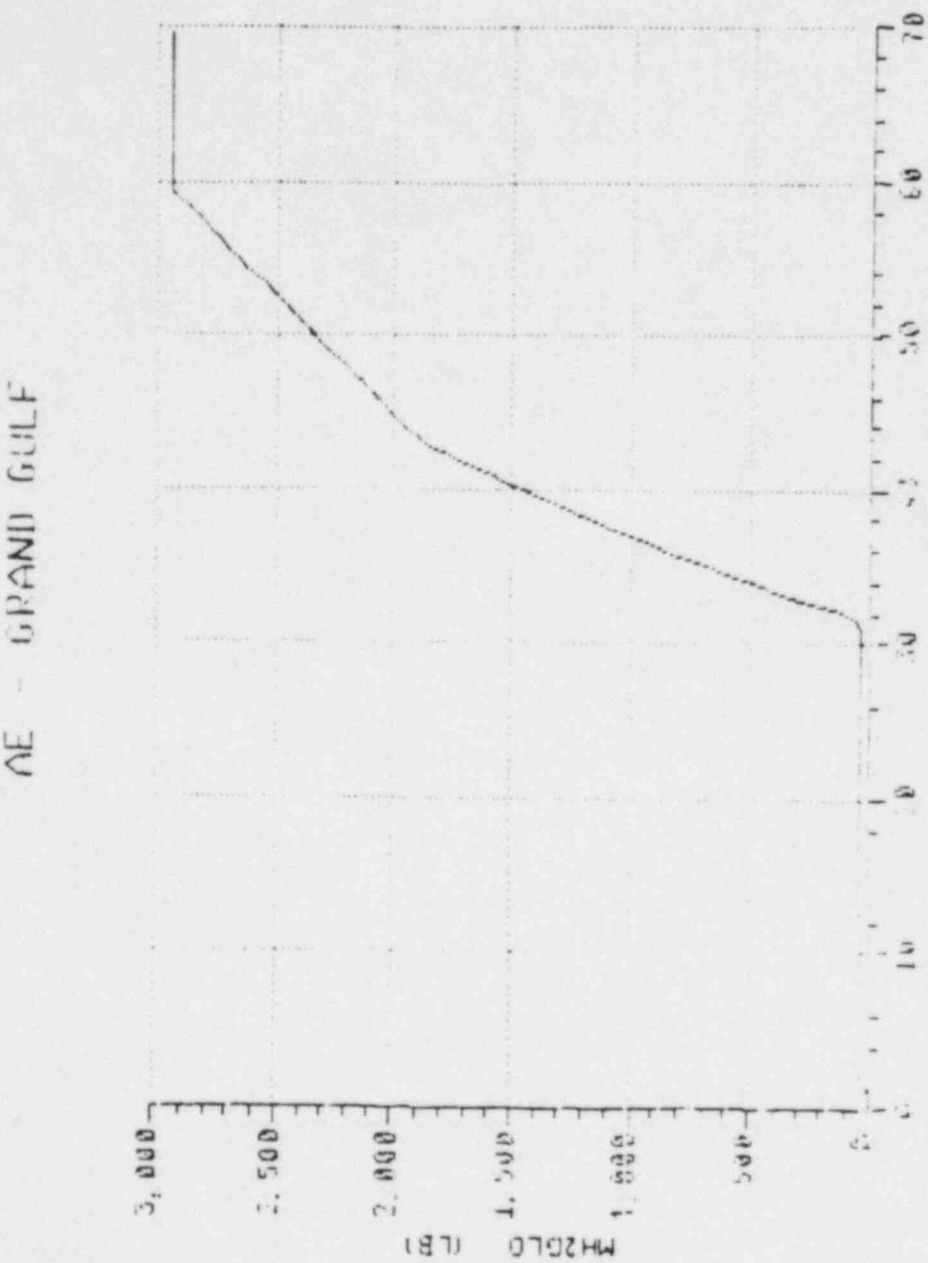

Fig. B.15 Total H_2 generated.

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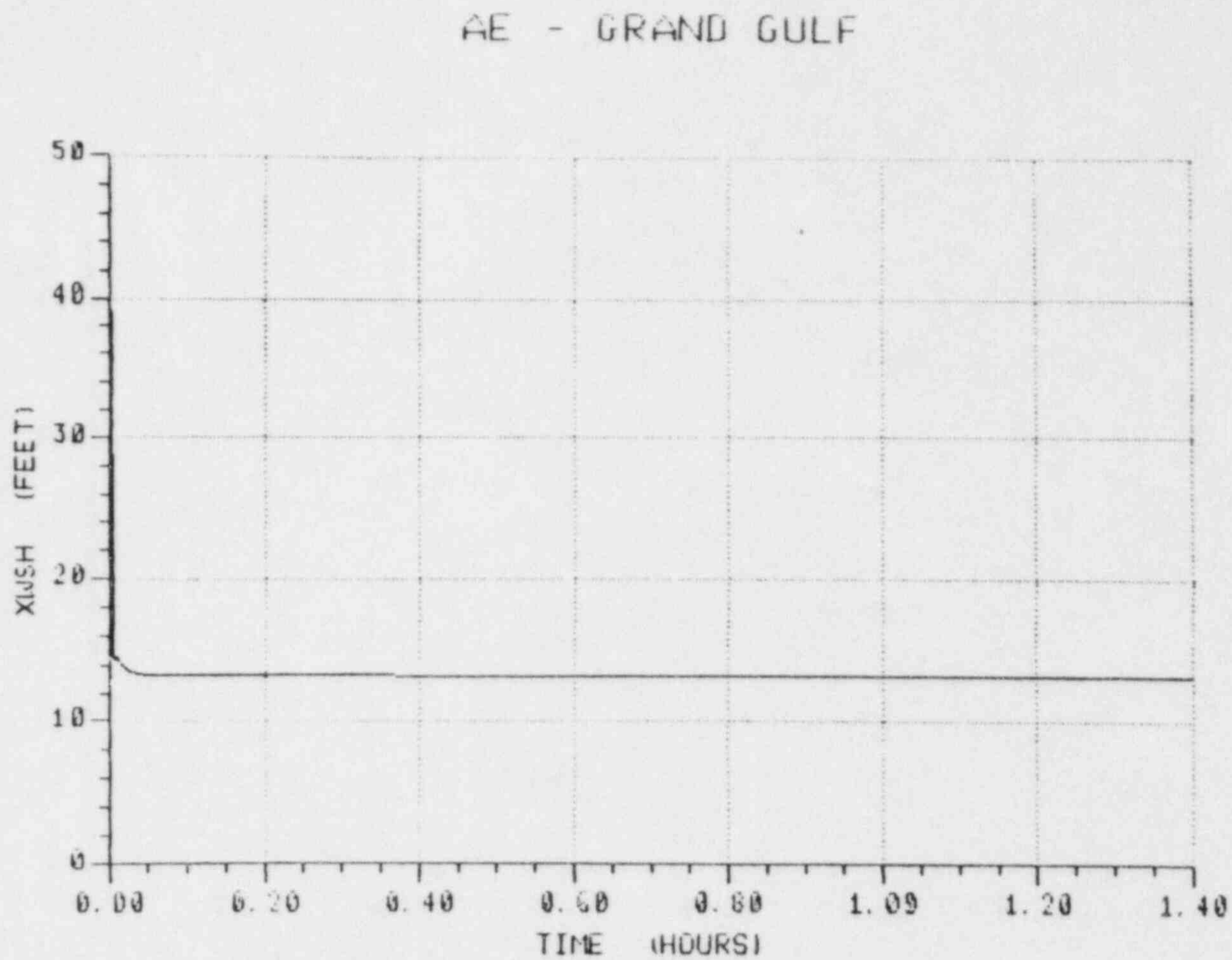

Fig. B.16 Reactor vessel water level.

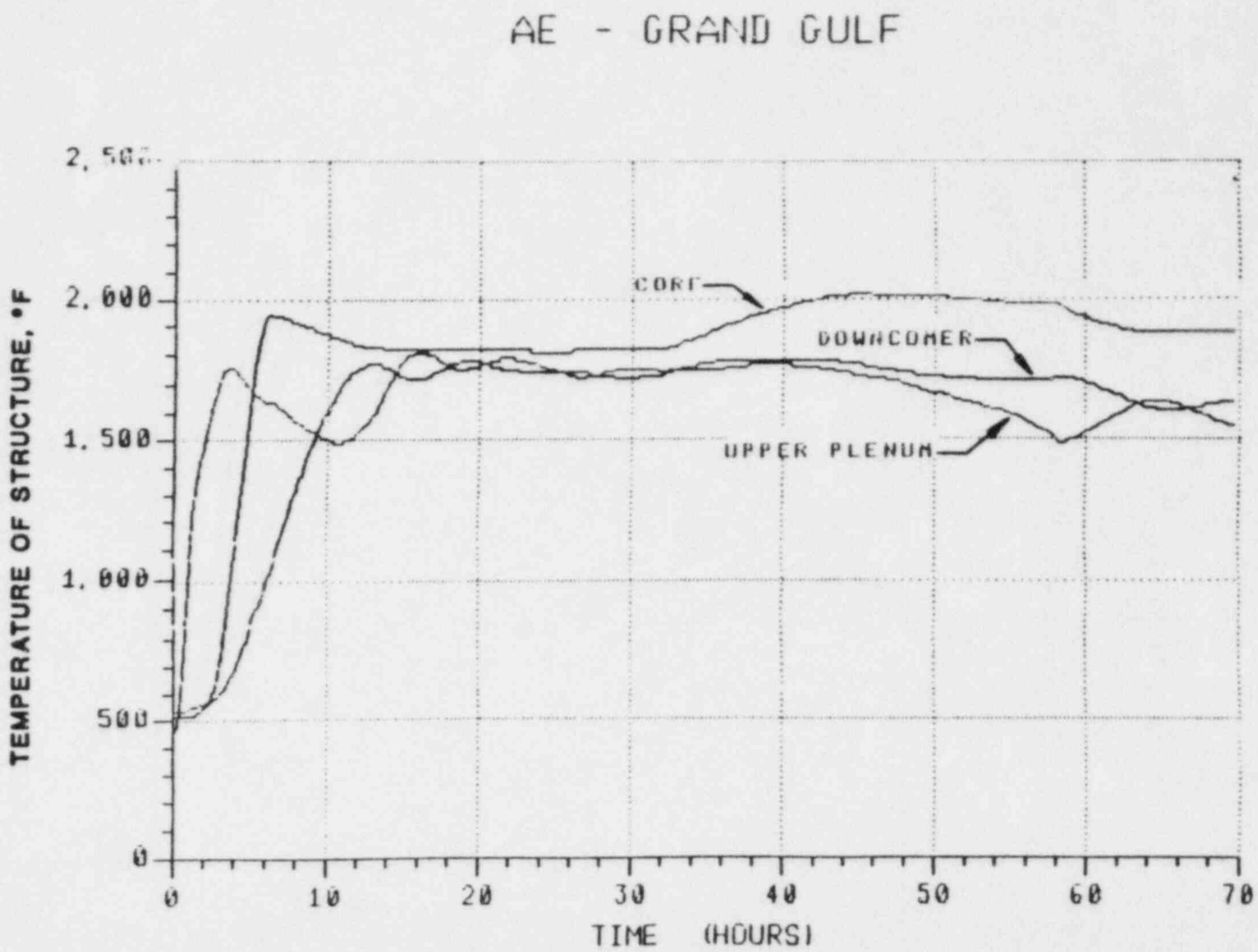

Fig. B.17 Temperature of structure, °F.

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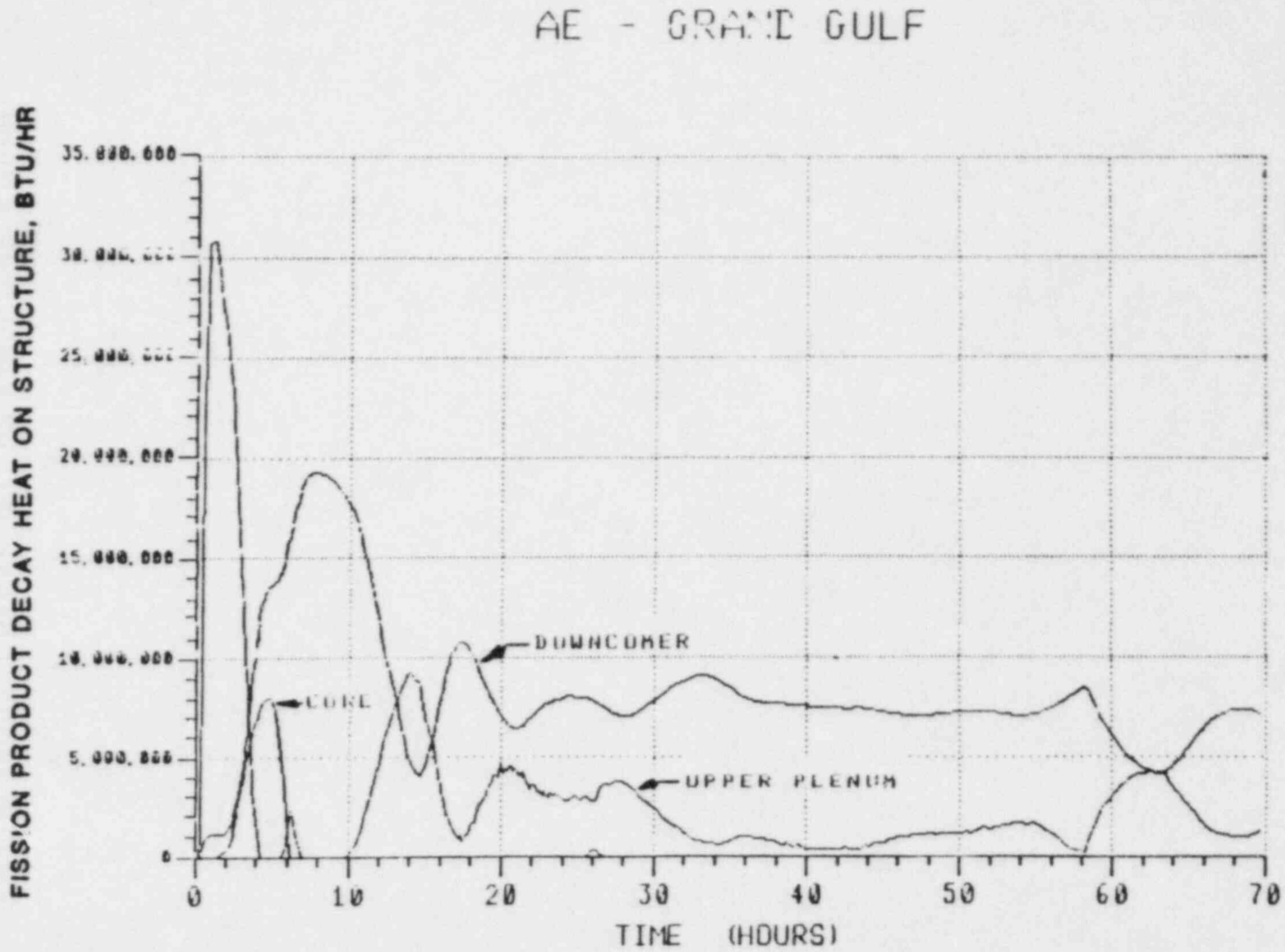

Fig. B.18 Fission product decay heat on structure, Btu/hr.

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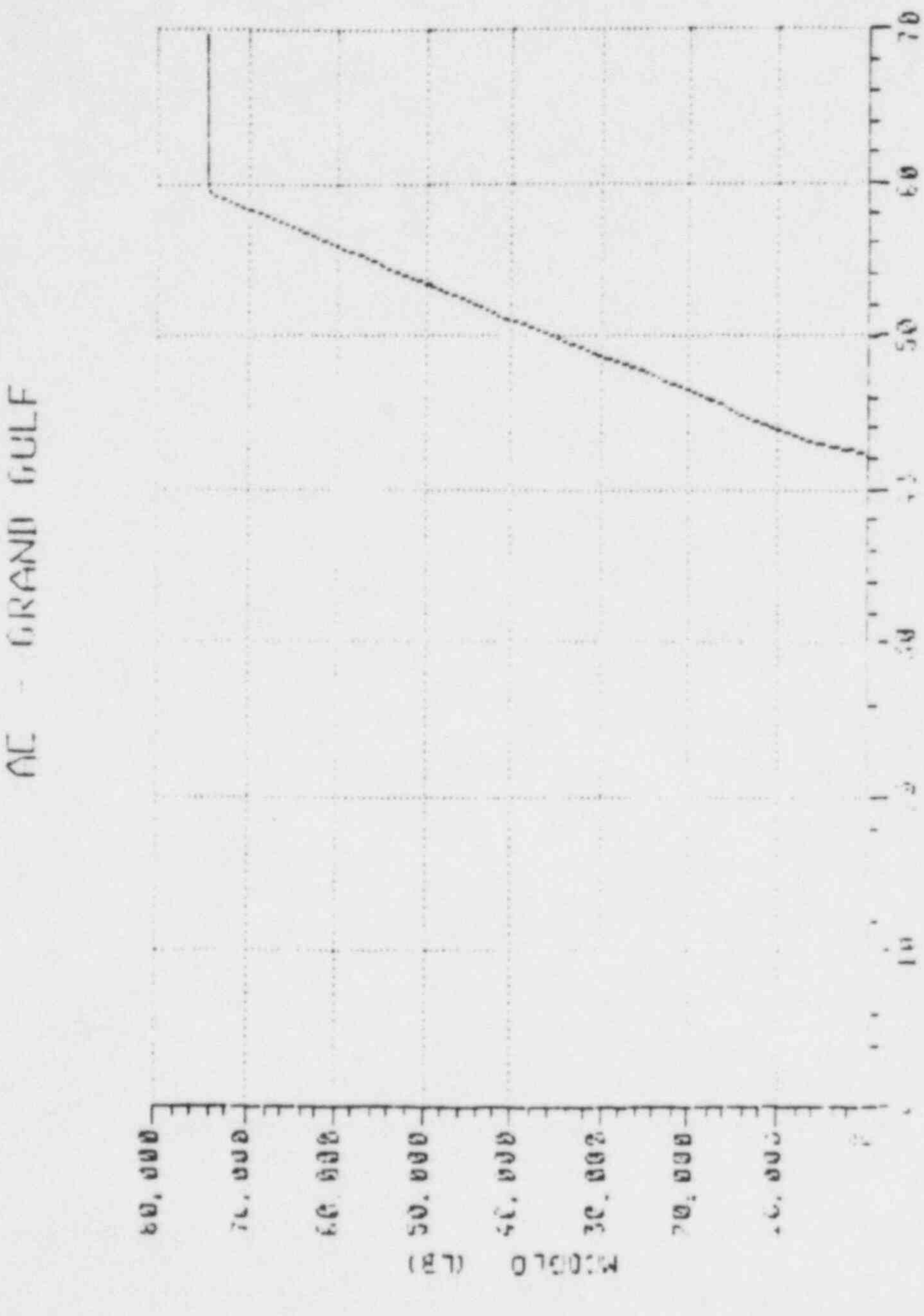

Fig. B.19 Total CO generated.

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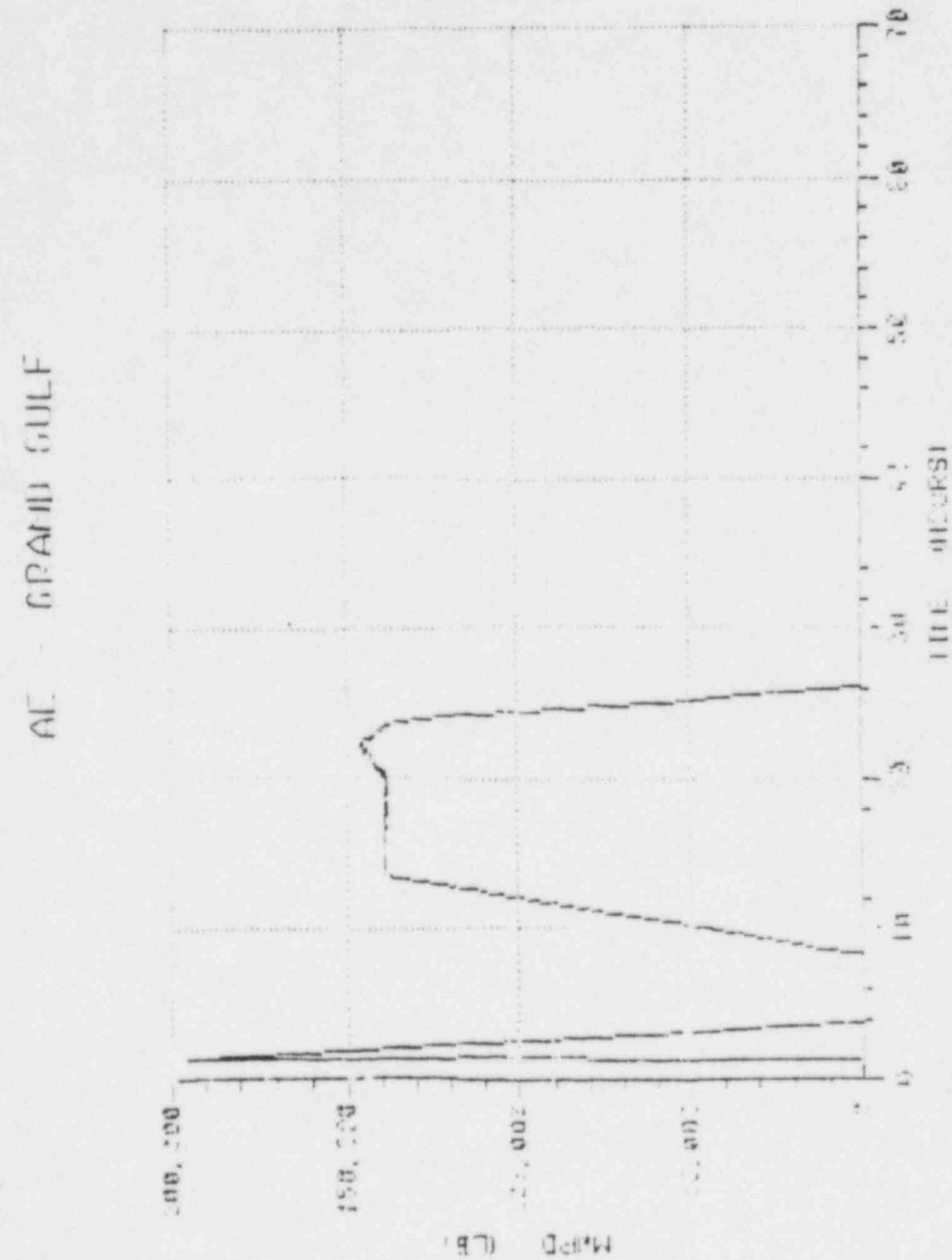

Fig. B.20 Mass of water in the pedestal.

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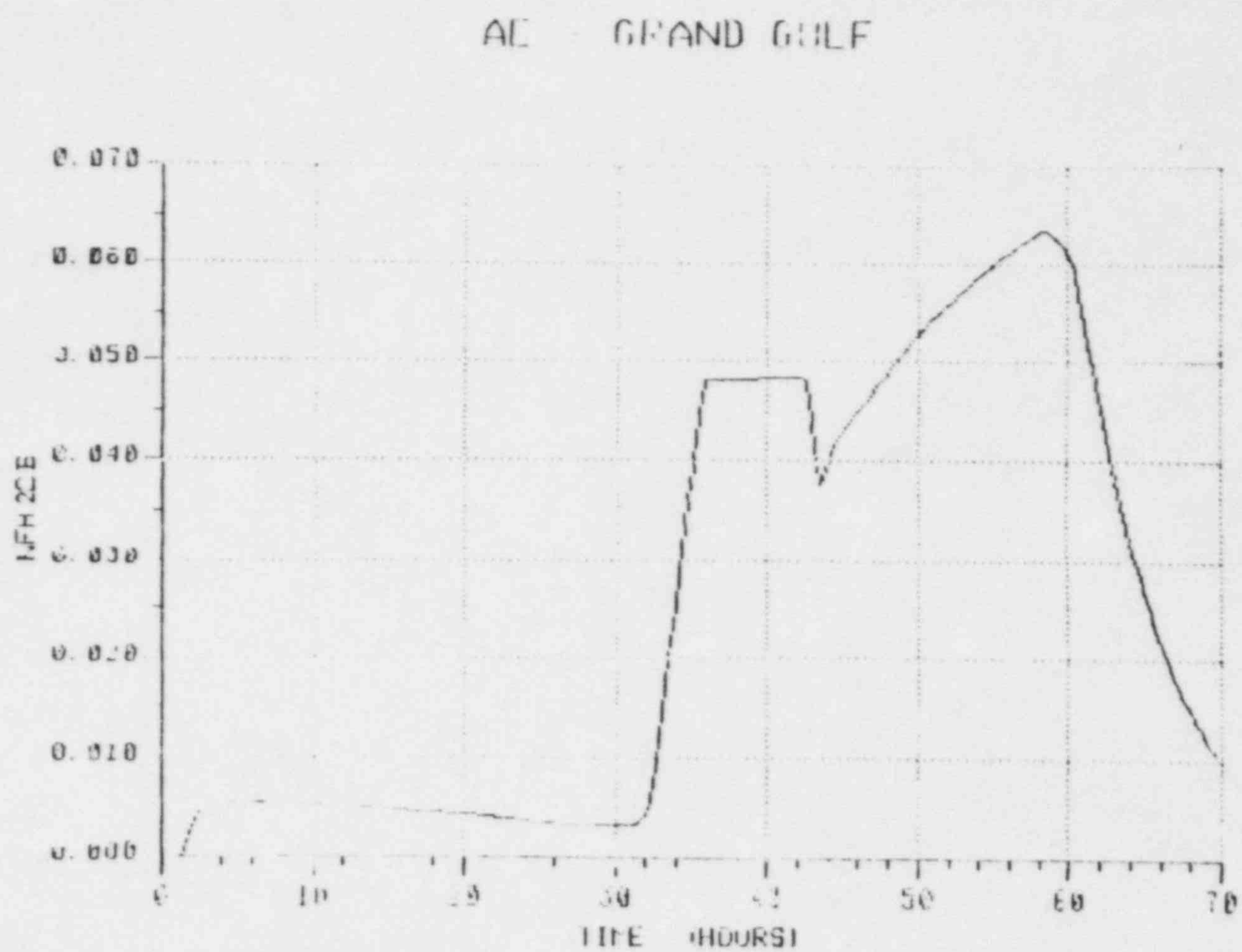

Fig. B.21 Mole fraction of H_2 in Compartment B.

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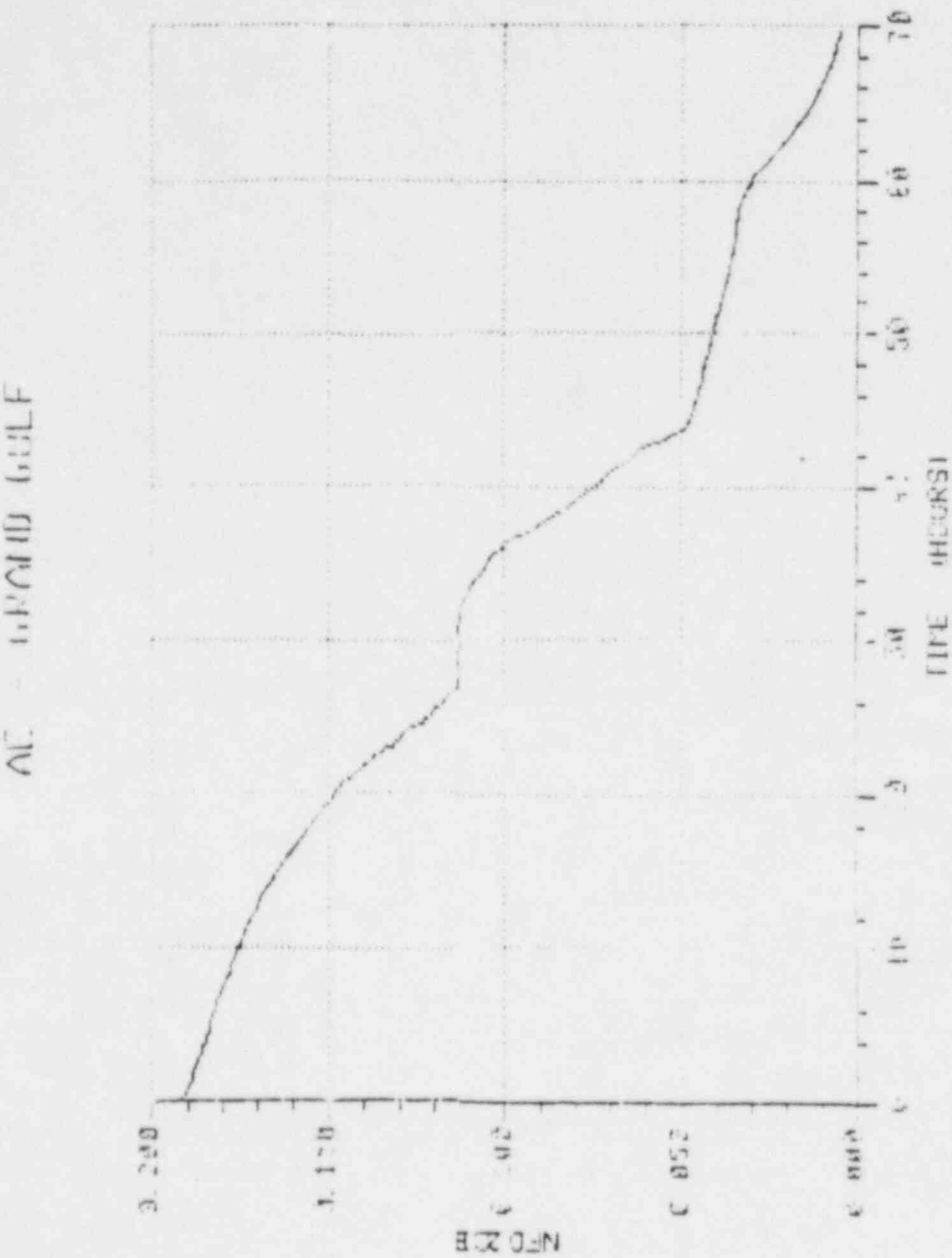

Fig. B.22 Mole fraction of O_2 in Compartment B.

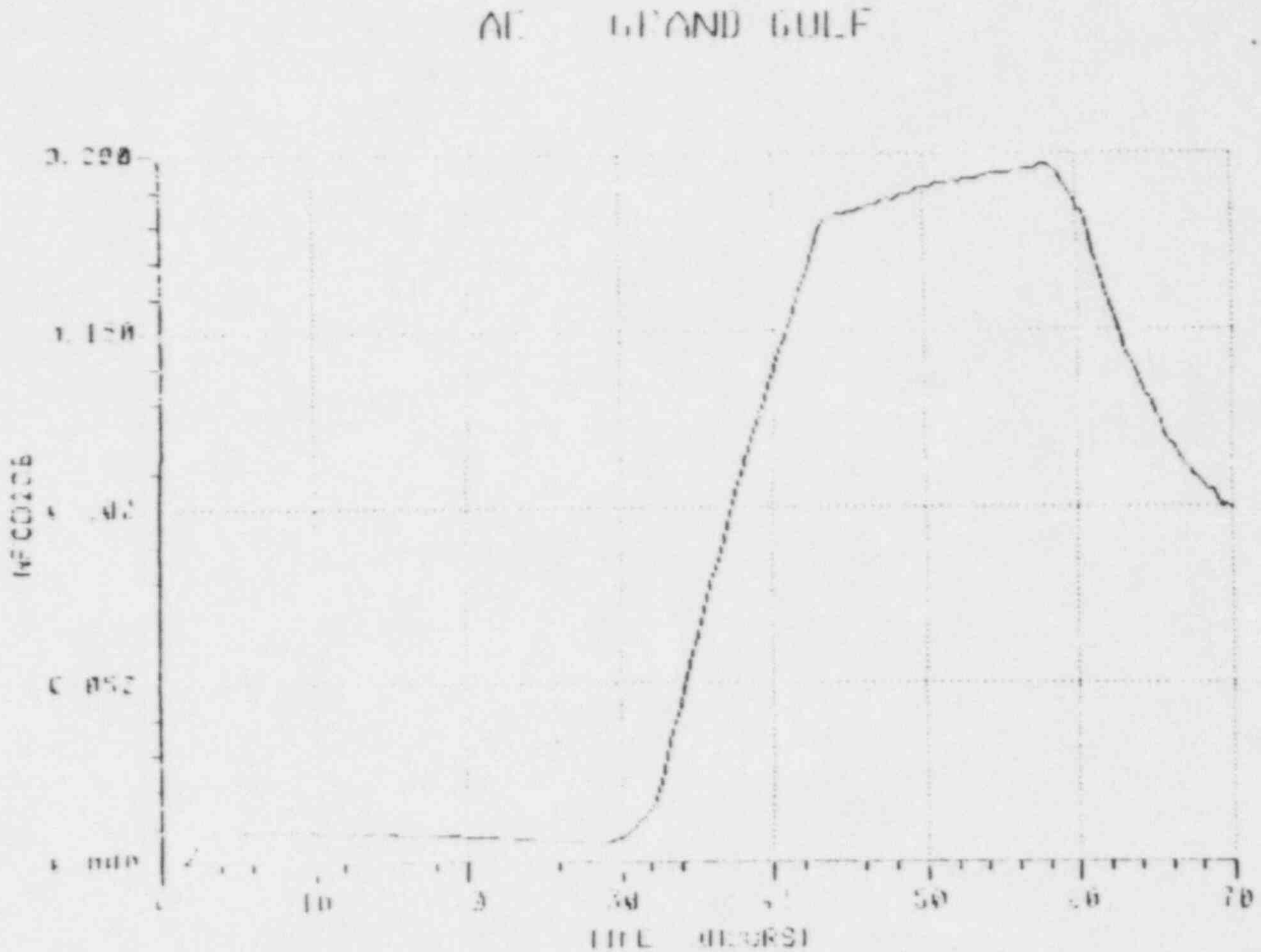

Fig. B.23 Mole fraction of CO_2 in Compartment B.

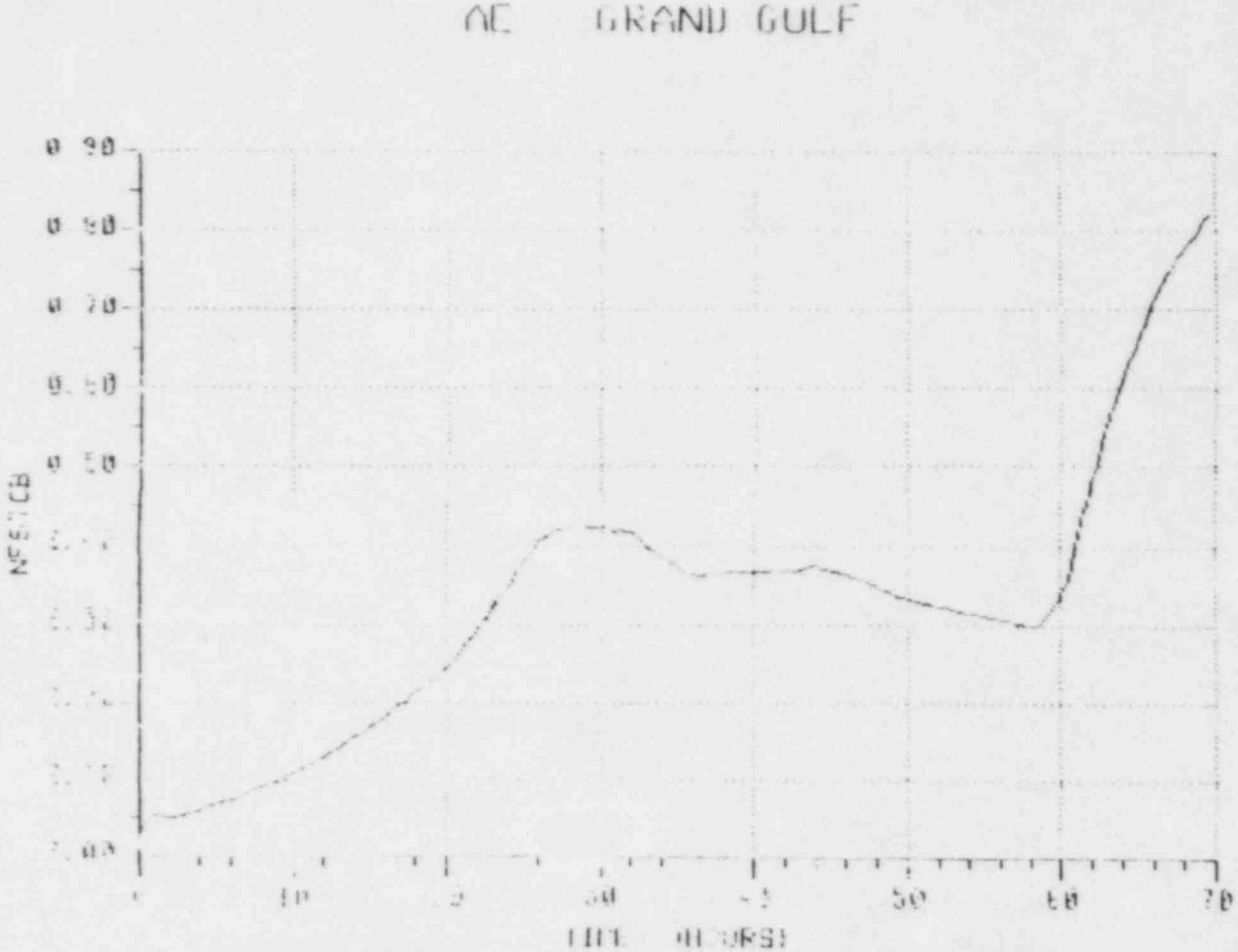

Fig. B.24 Mole fraction of steam in Compartment B.

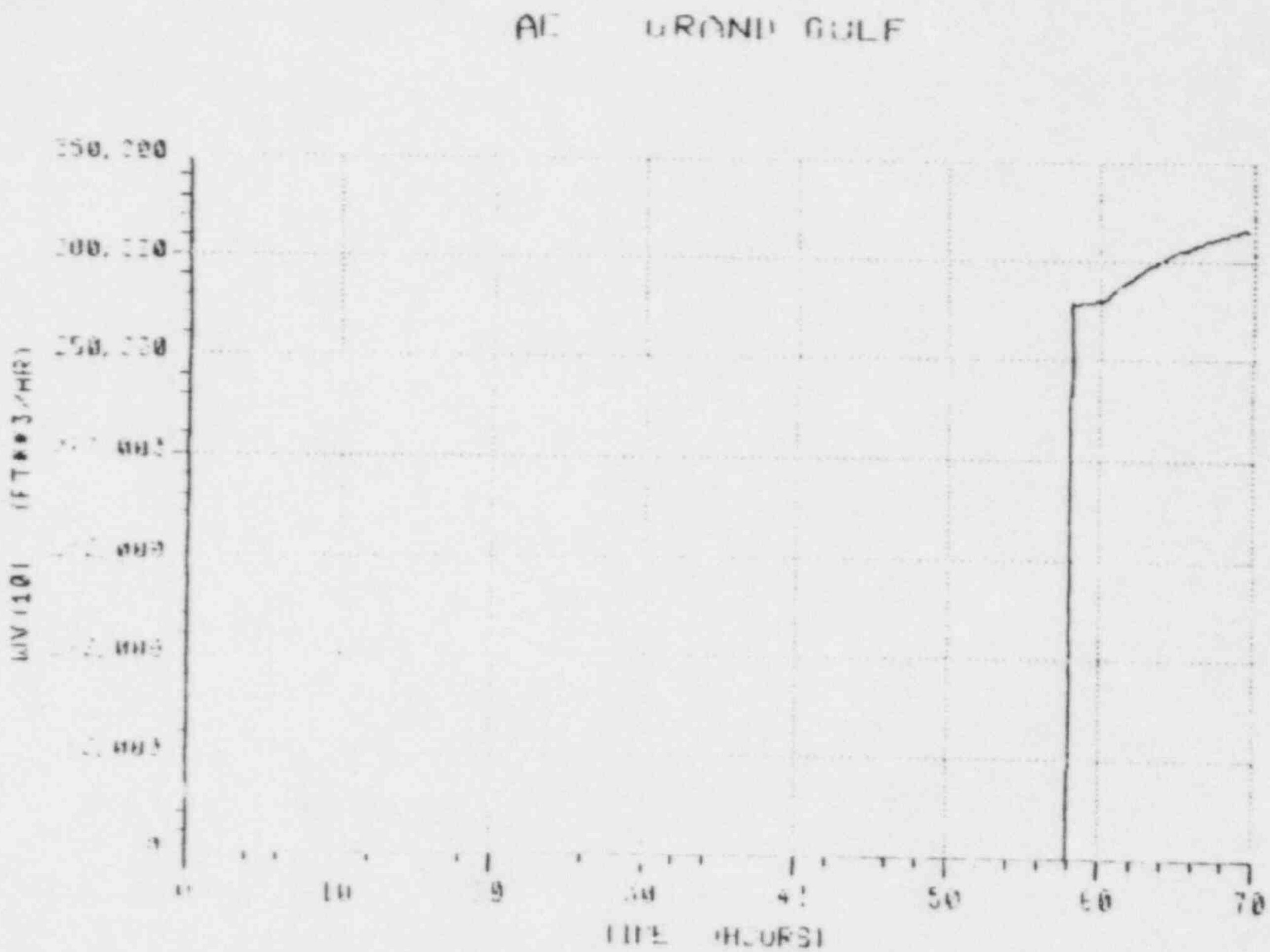

Fig. B.25 Volumetric flow out of containment.

DRAFT

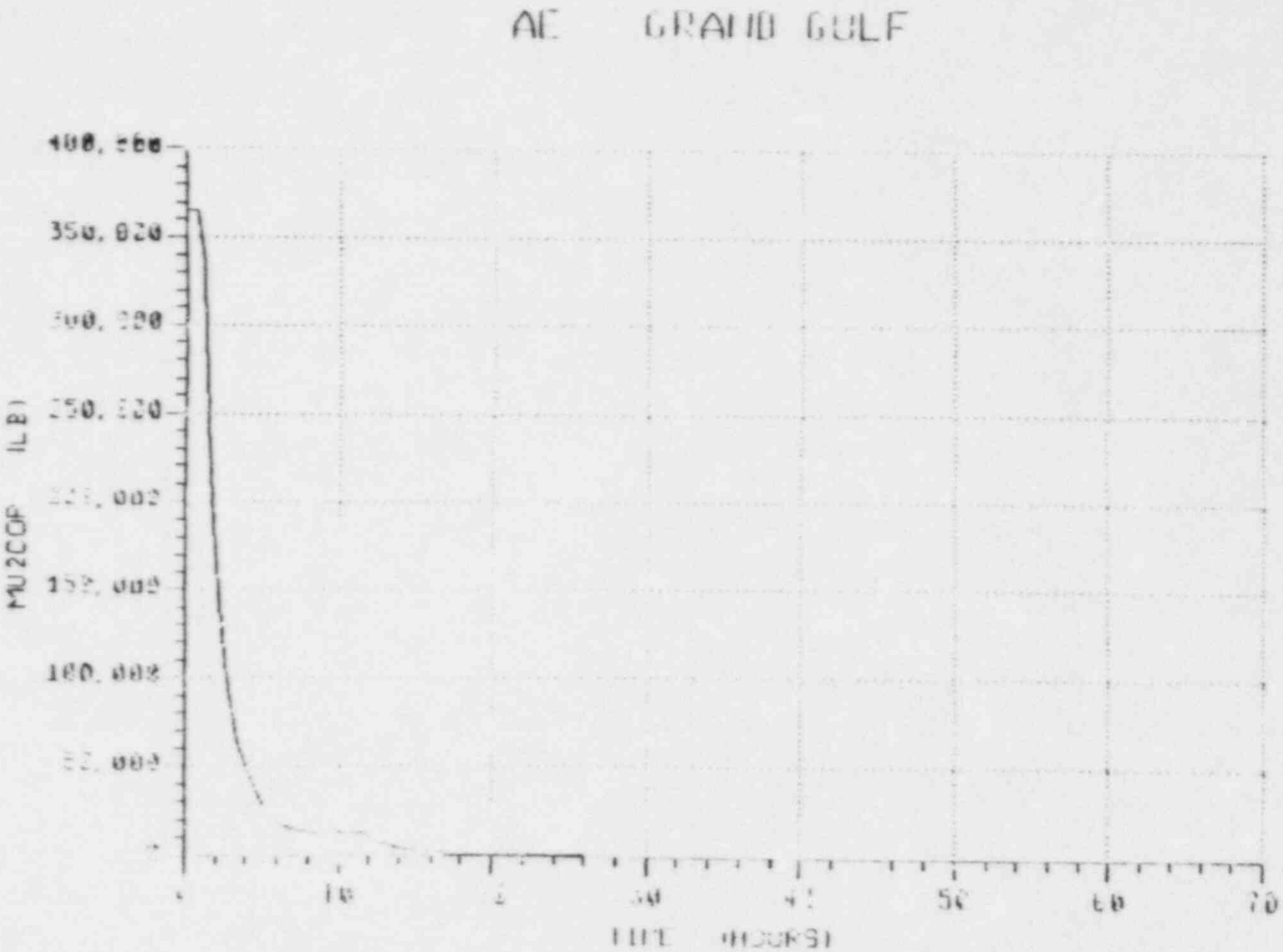

Fig. B.26 Mass of UO_2 in core region.

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SUPPLEMENTAL PLOTS FOR SEQUENCE $T_{23}QW$

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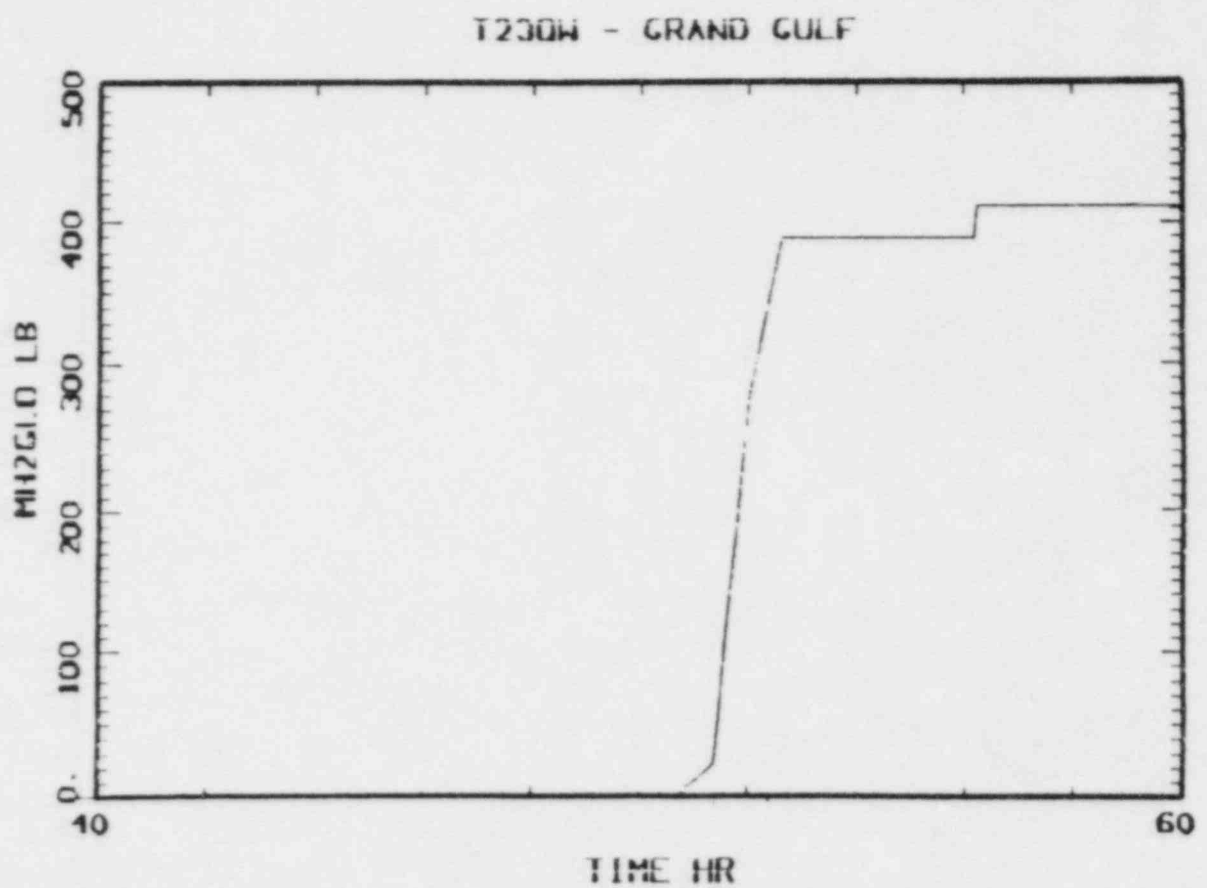

Fig. B.27 Total H_2 generated.

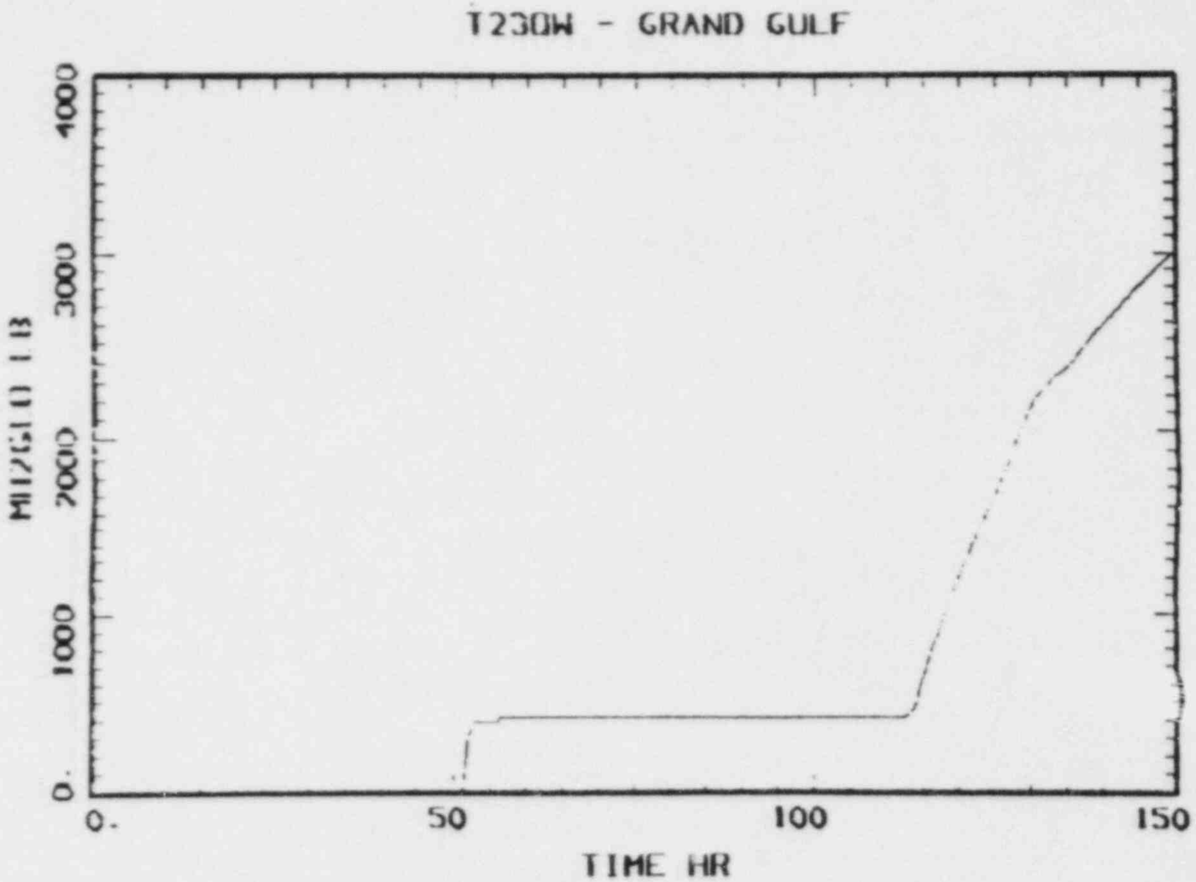

Fig. B.28 Total H_2 generated.

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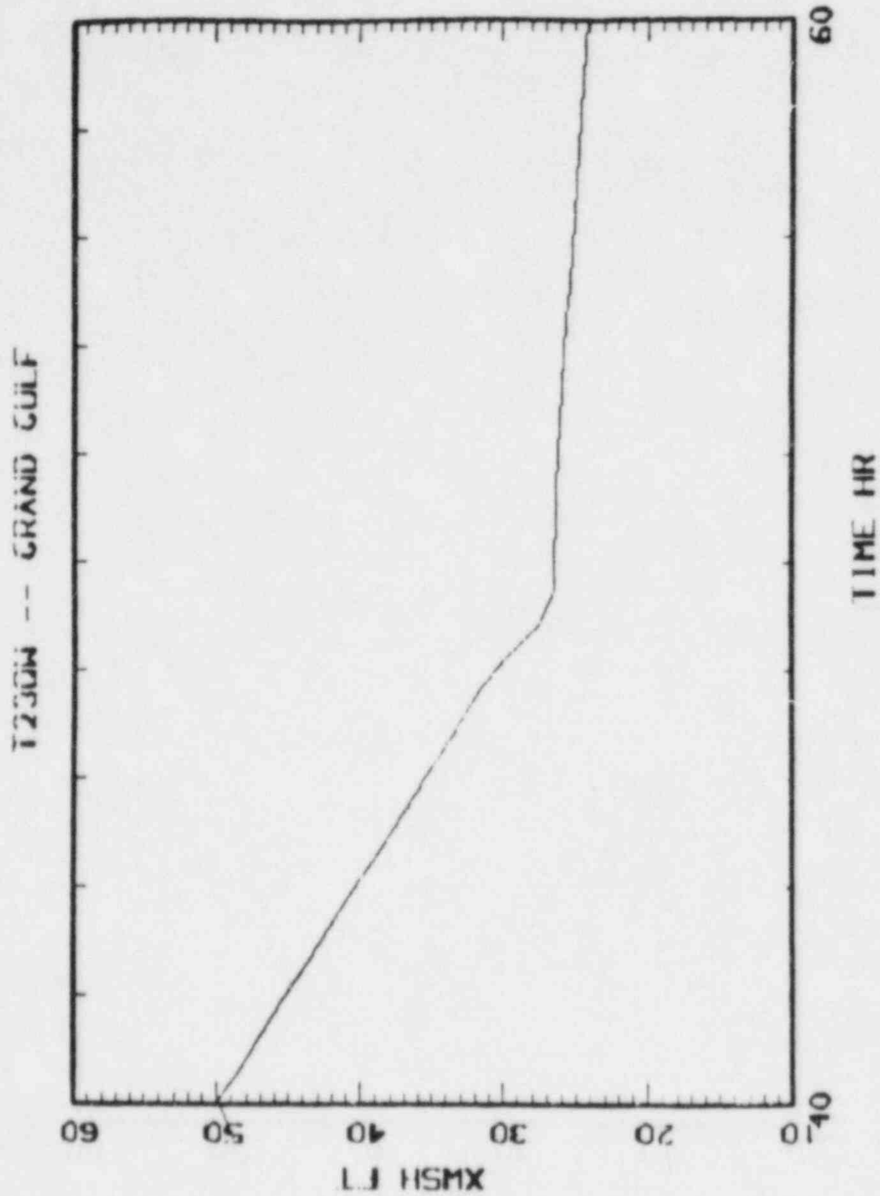

Fig. B.29 Reactor vessel water level.

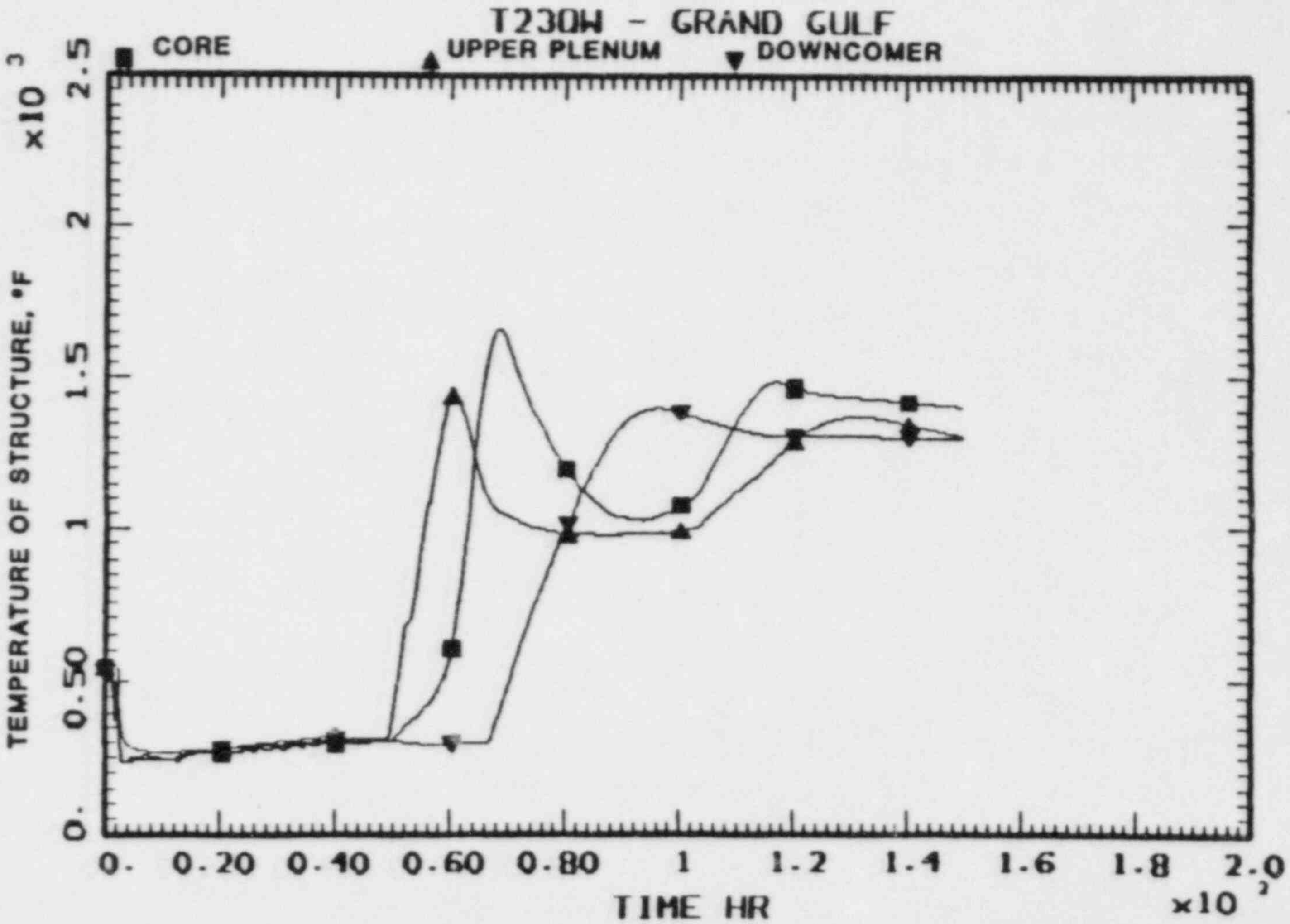

Fig. B.30 Temperature of structure, °F.

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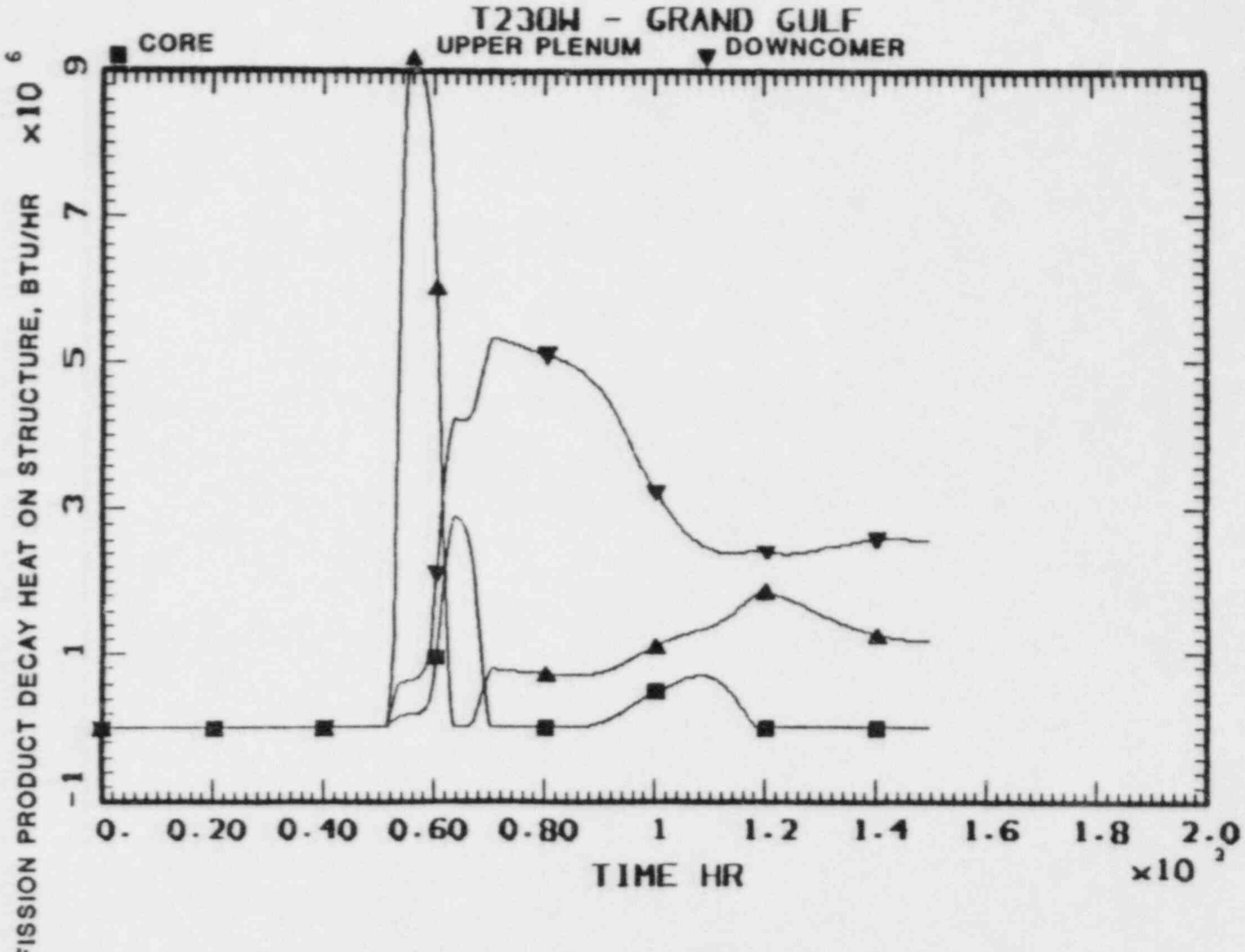

Fig. B.31 Fission product decay heat on structure, Btu/hr.

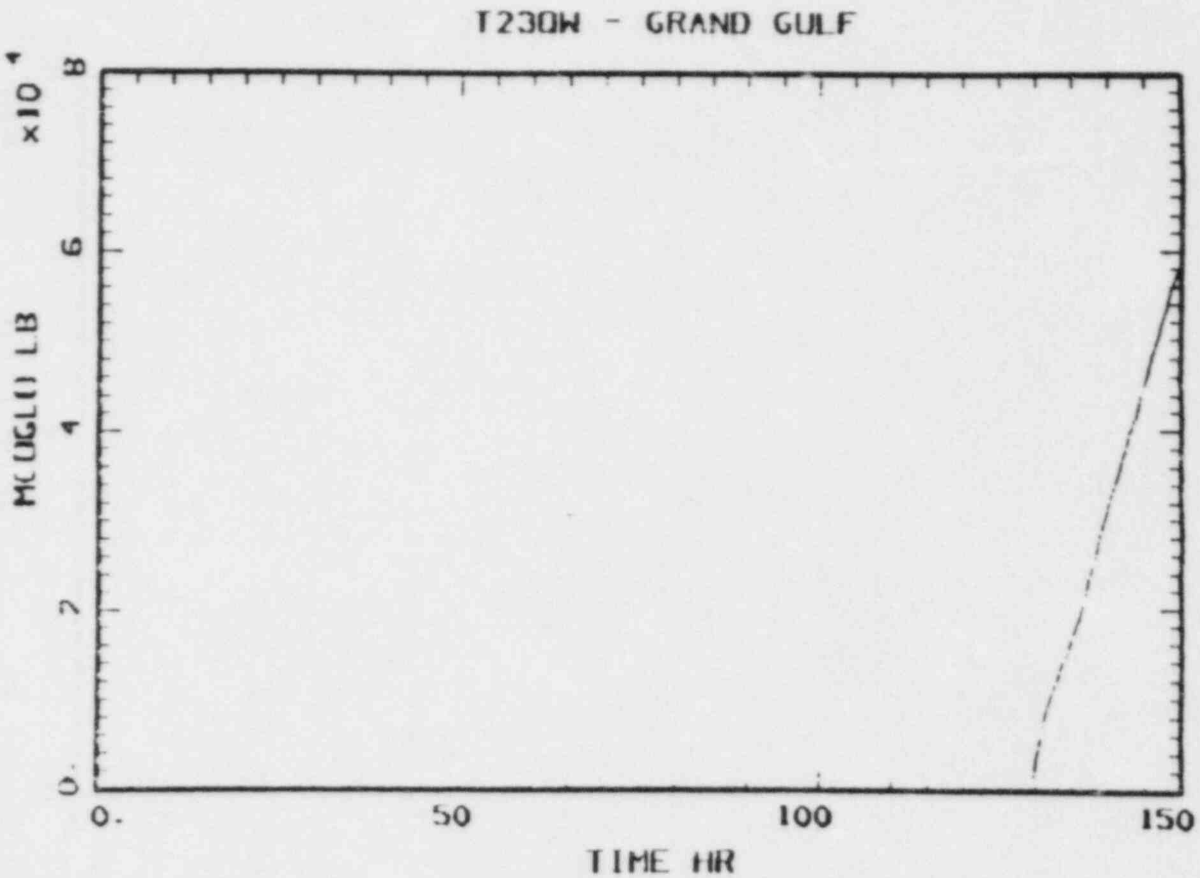

Fig. B.32 Total CO generated.

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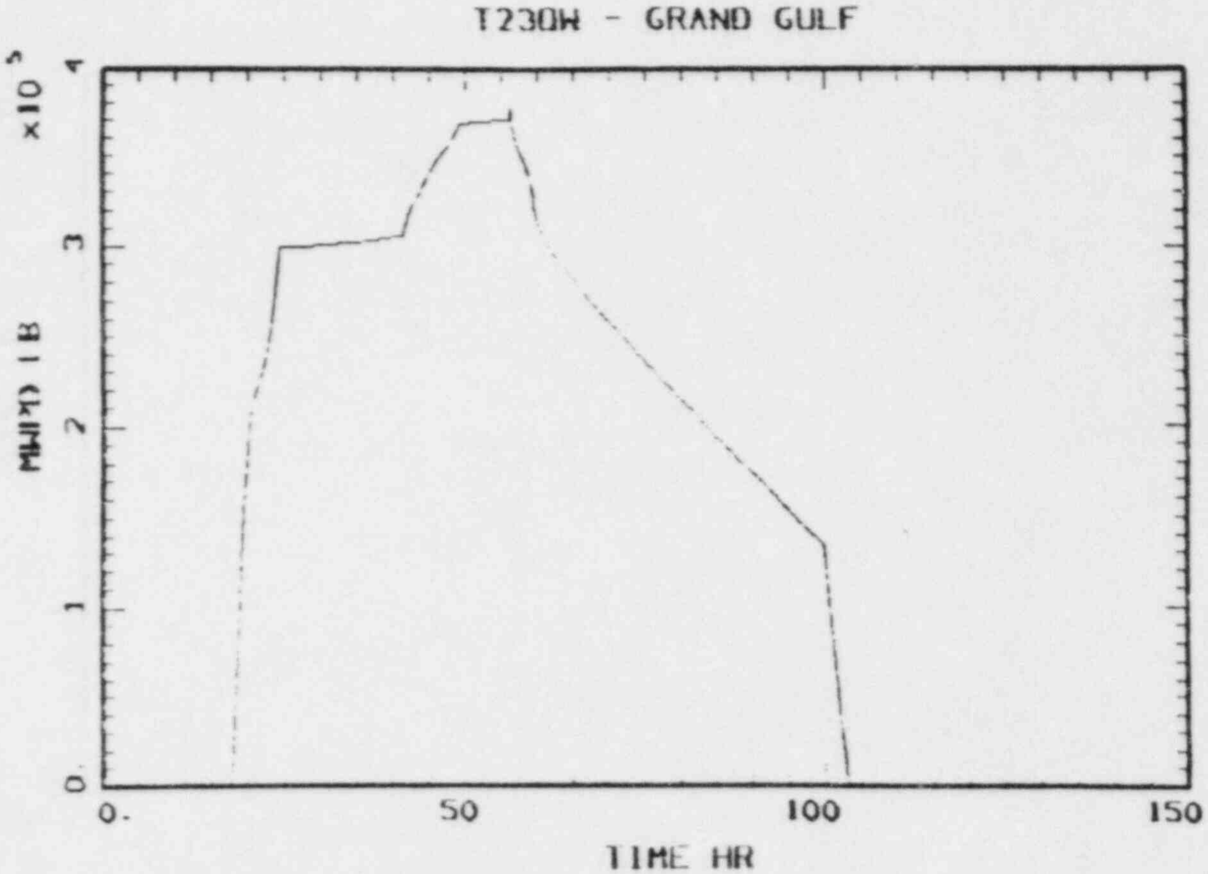

Fig. B.33 Mass of water in the pedestal.

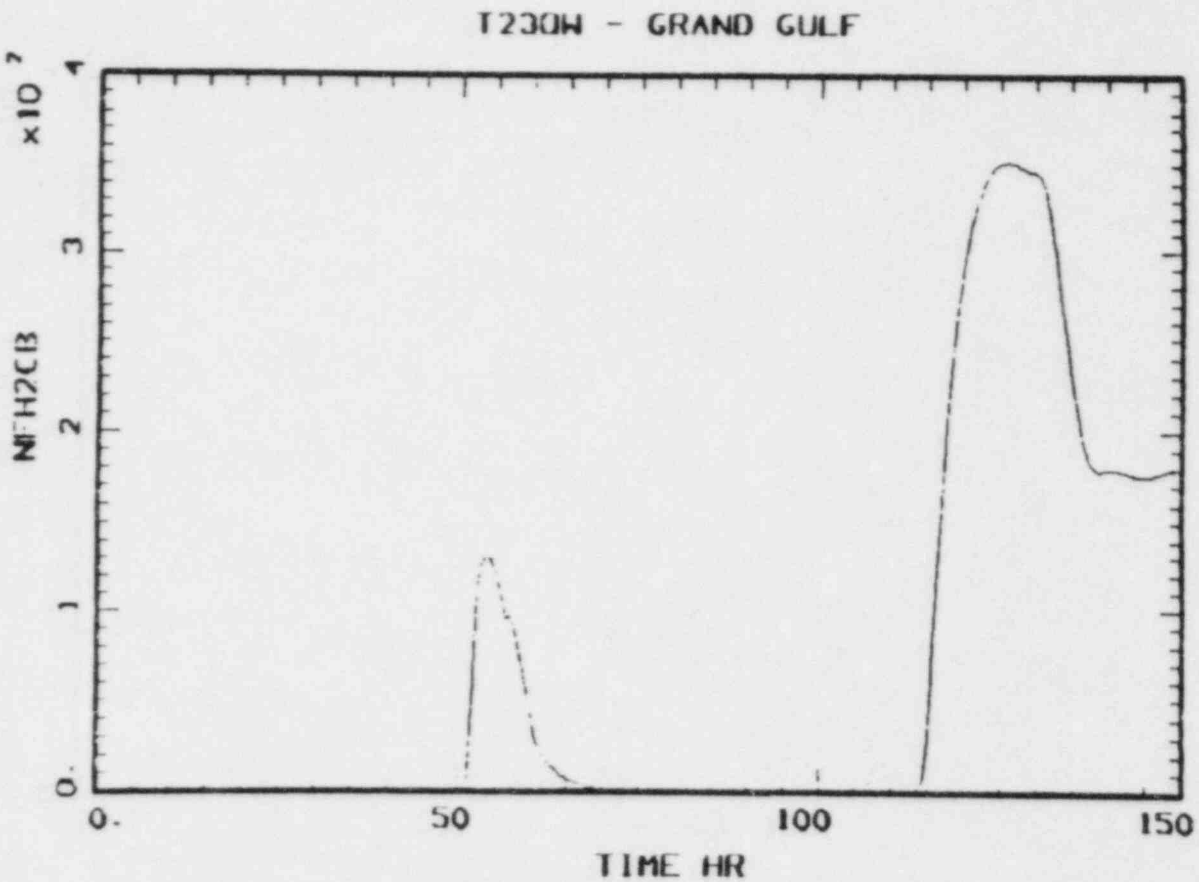

Fig. B.34 Mole fraction of H_2 in Compartment B.

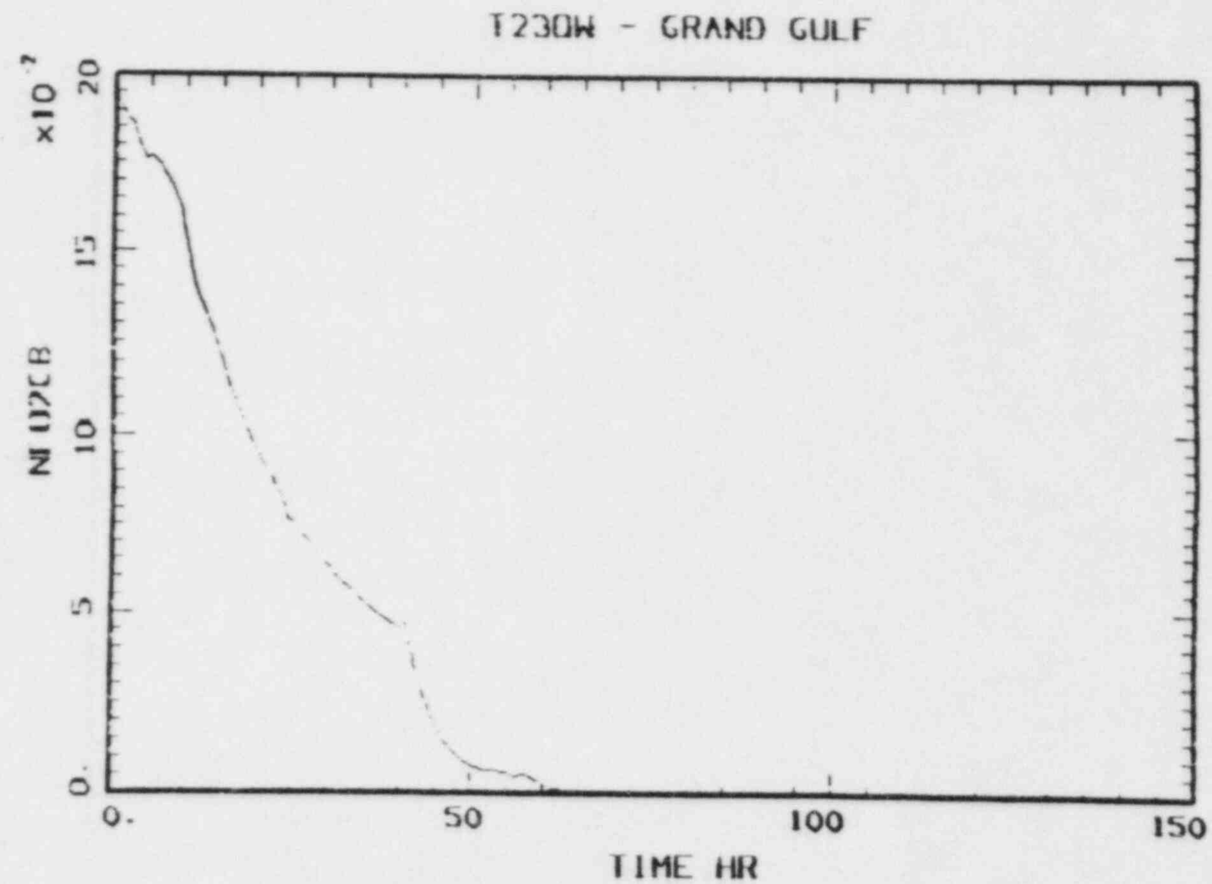

Fig. B.35 Mole fraction of O_2 in Compartment B.

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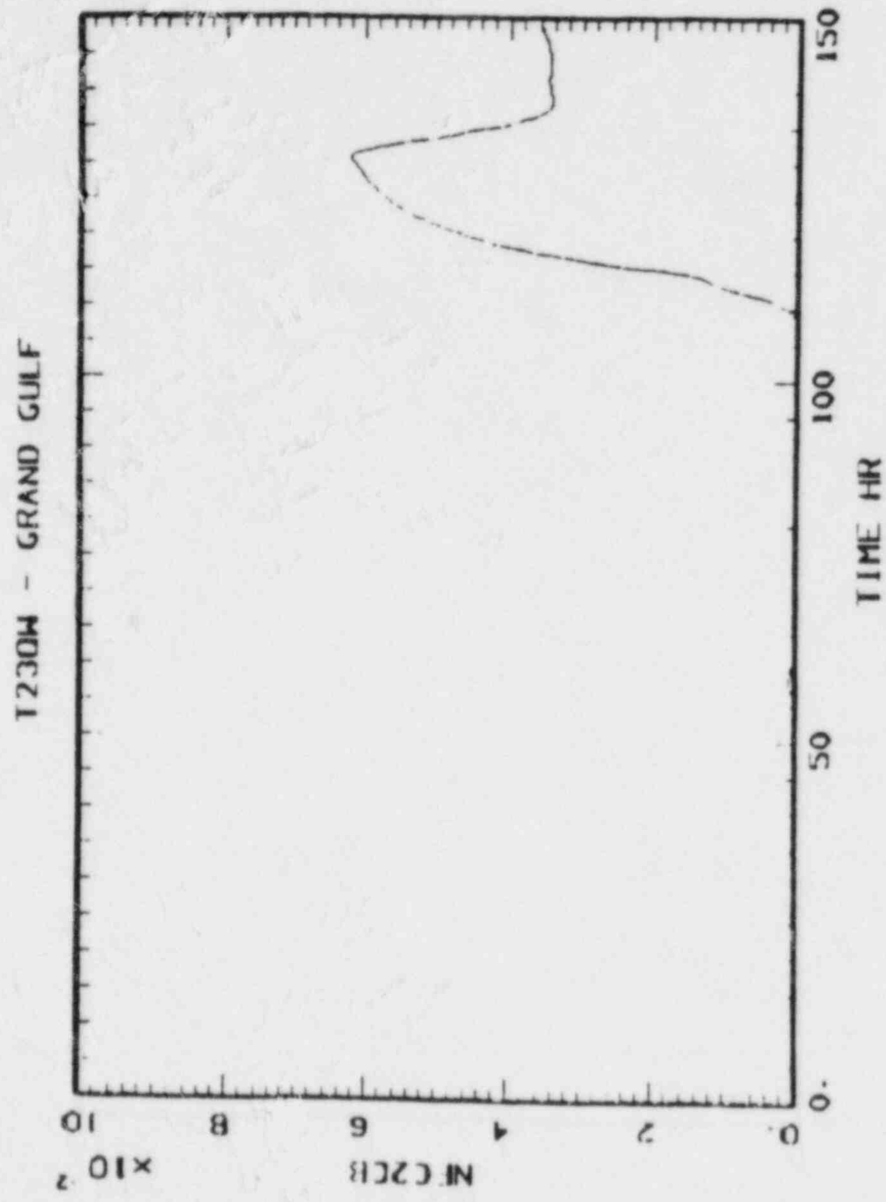

Fig. B.36 Mole fraction of CO_2 in Compartment B.

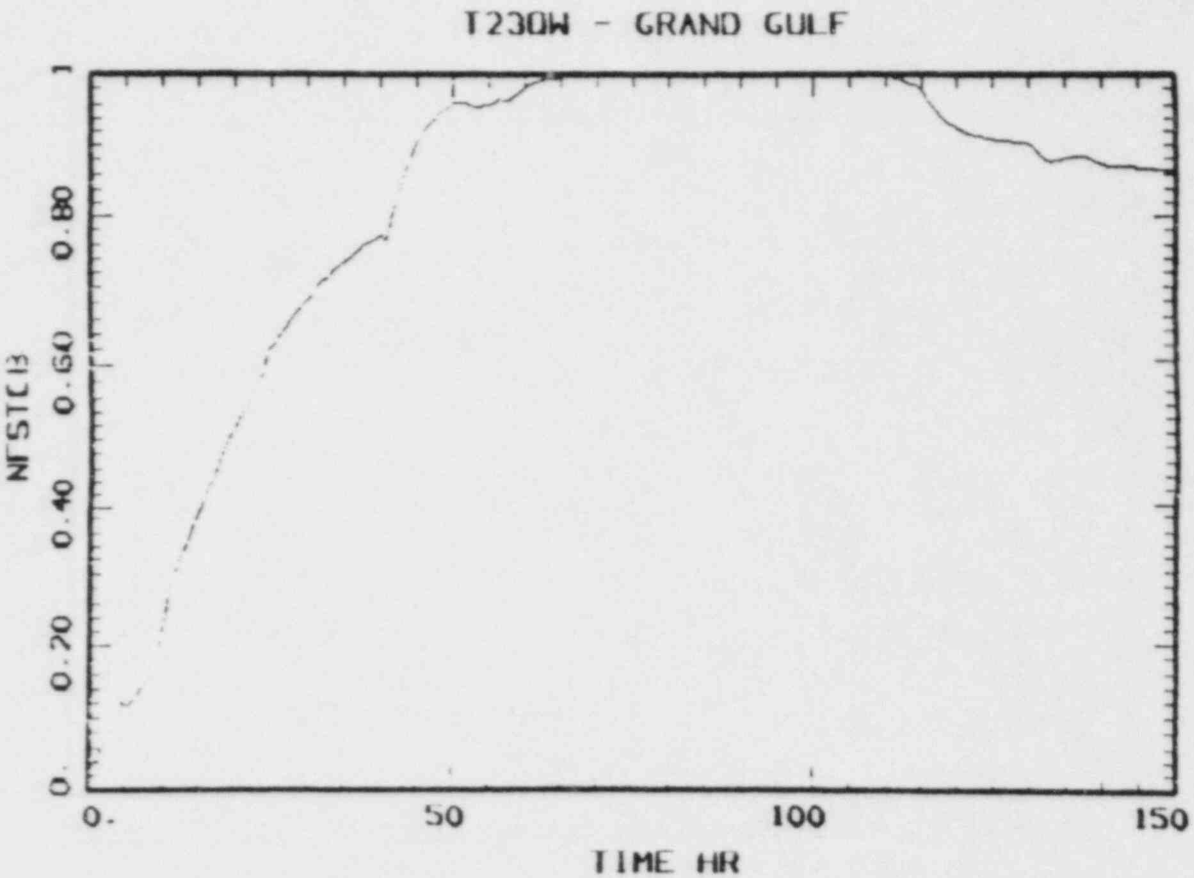

Fig. B.37 Mole fraction of steam in Compartment B.

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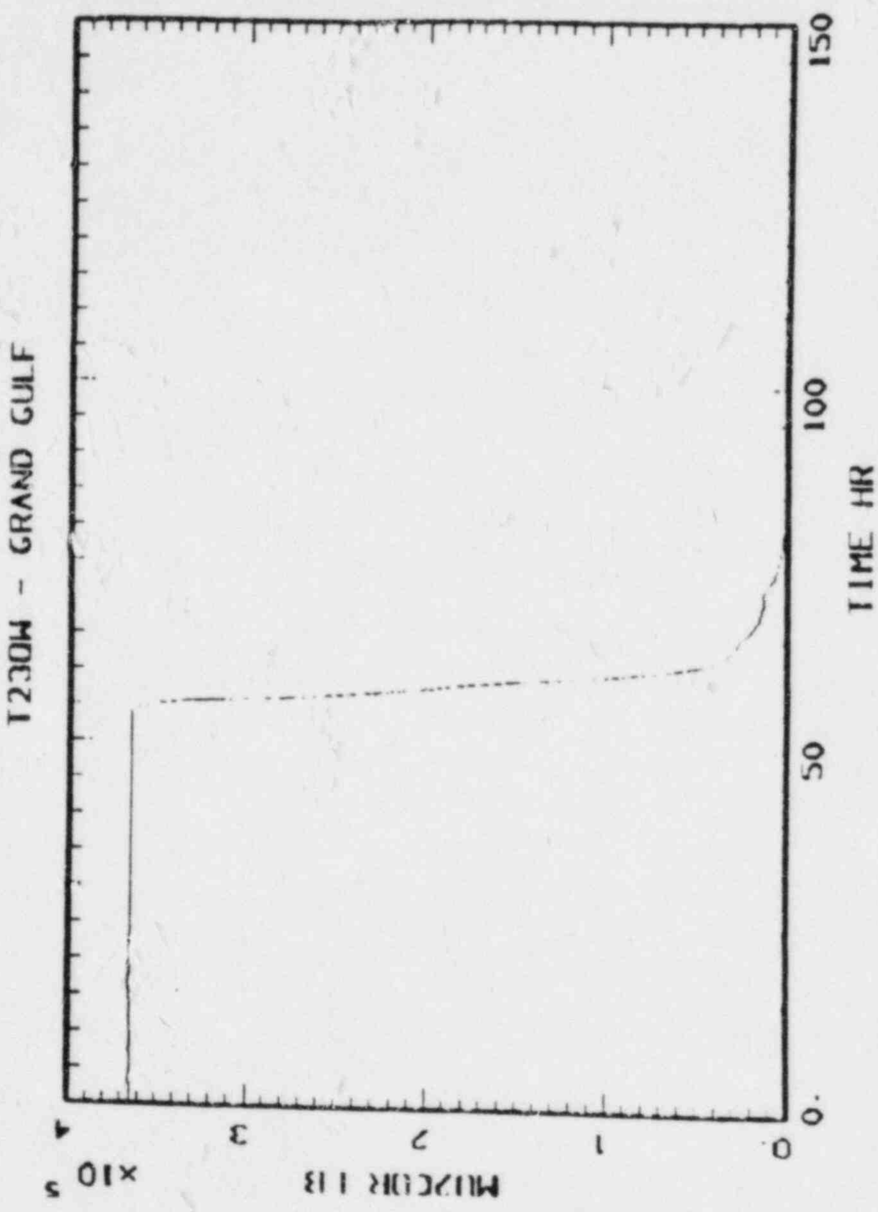

Fig. B.38 Mass of UO_2 in core region.

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SUPPLEMENTAL PLOTS FOR SEQUENCE $T_{23}C$

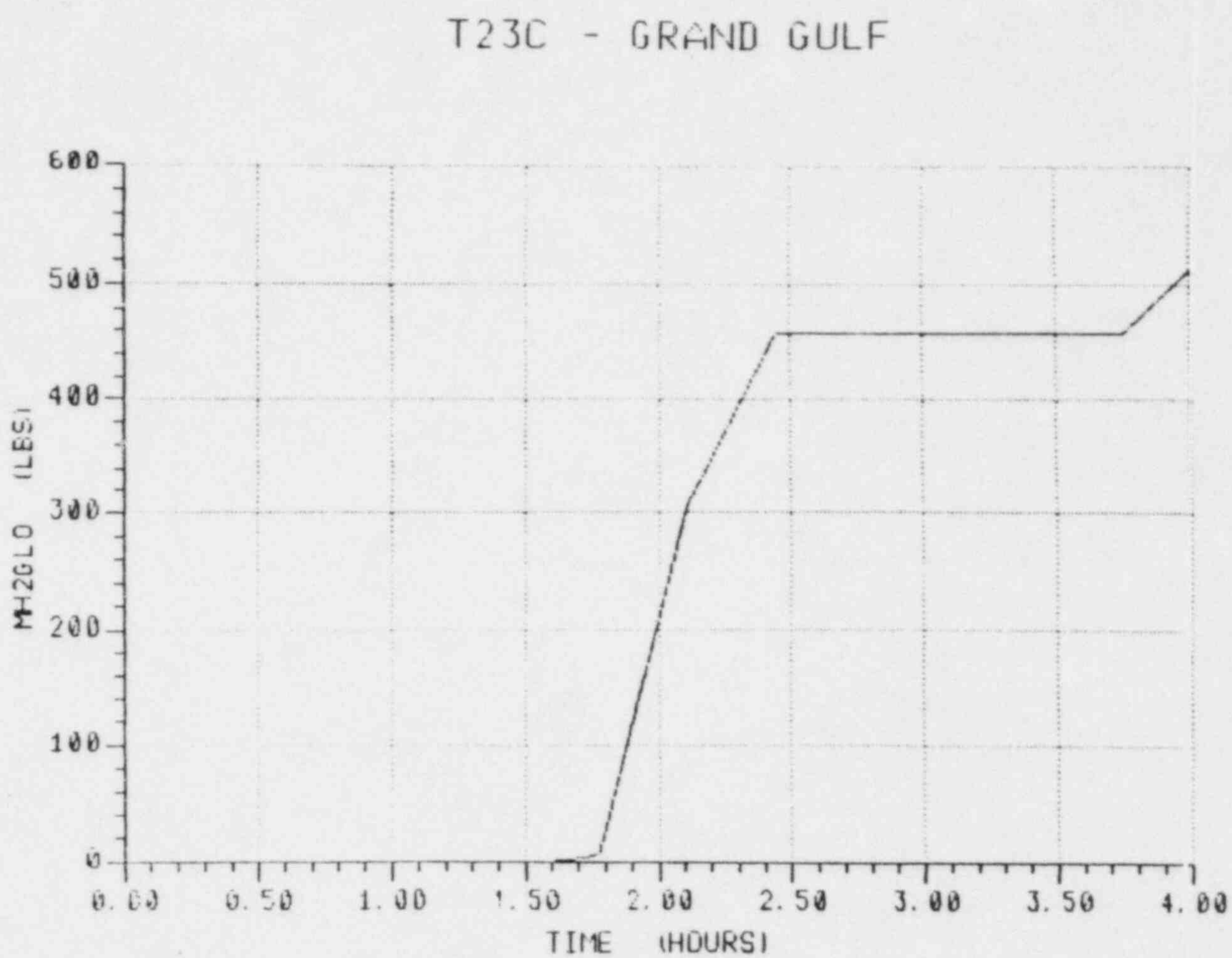

Fig. B.39 Total H_2 generated.

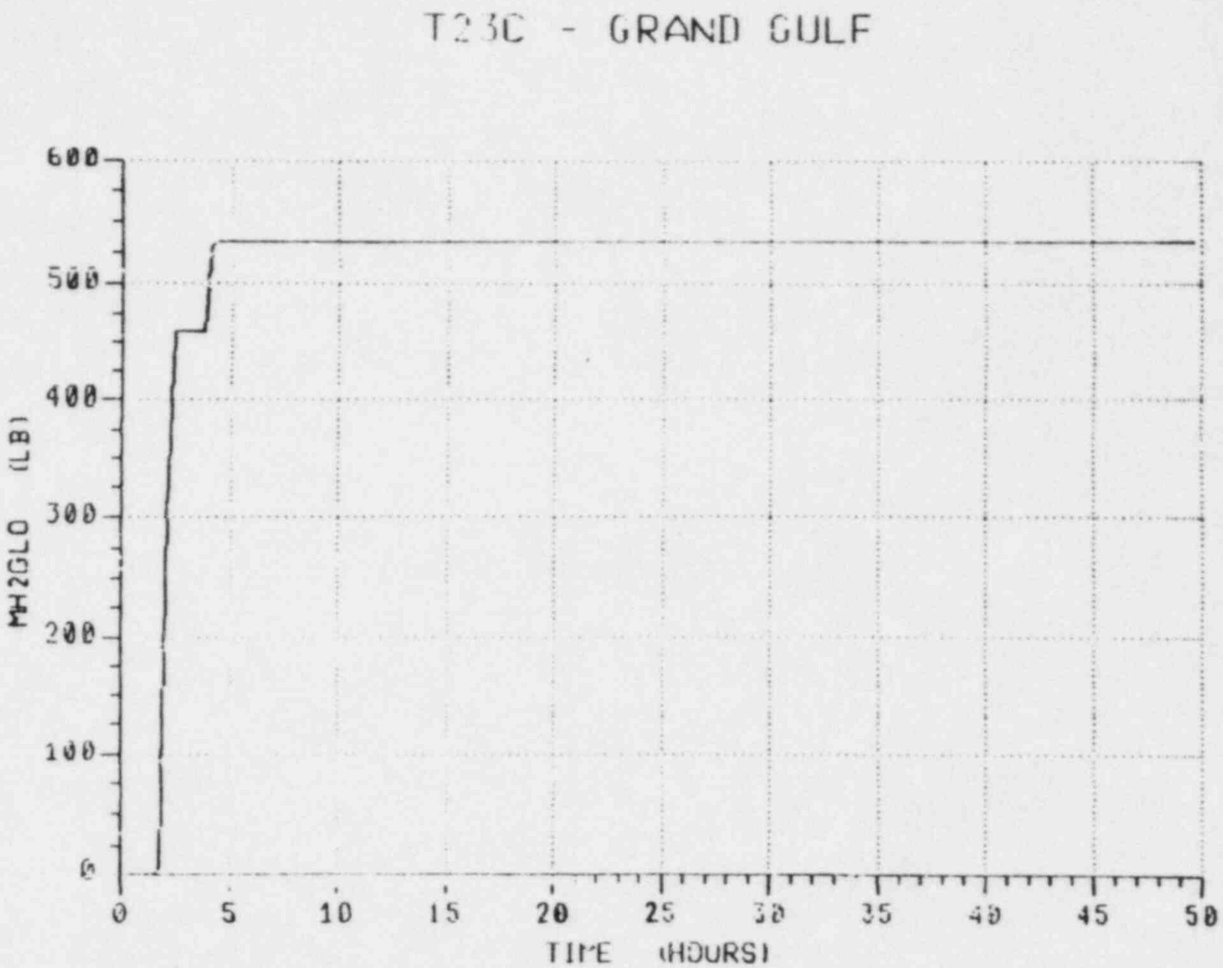

Fig. B.40 Total H_2 generated.

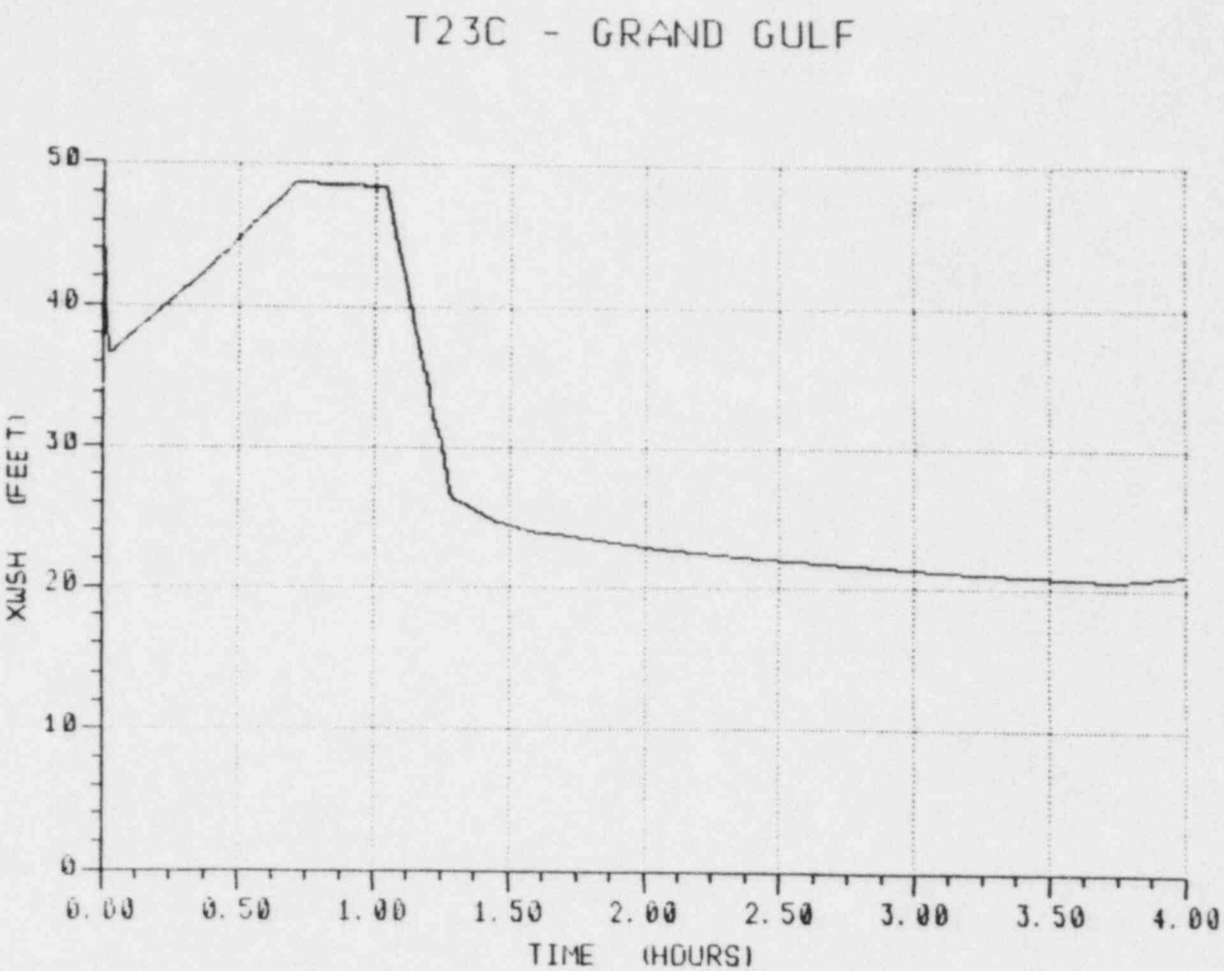

Fig. B.41 Reactor vessel water level.

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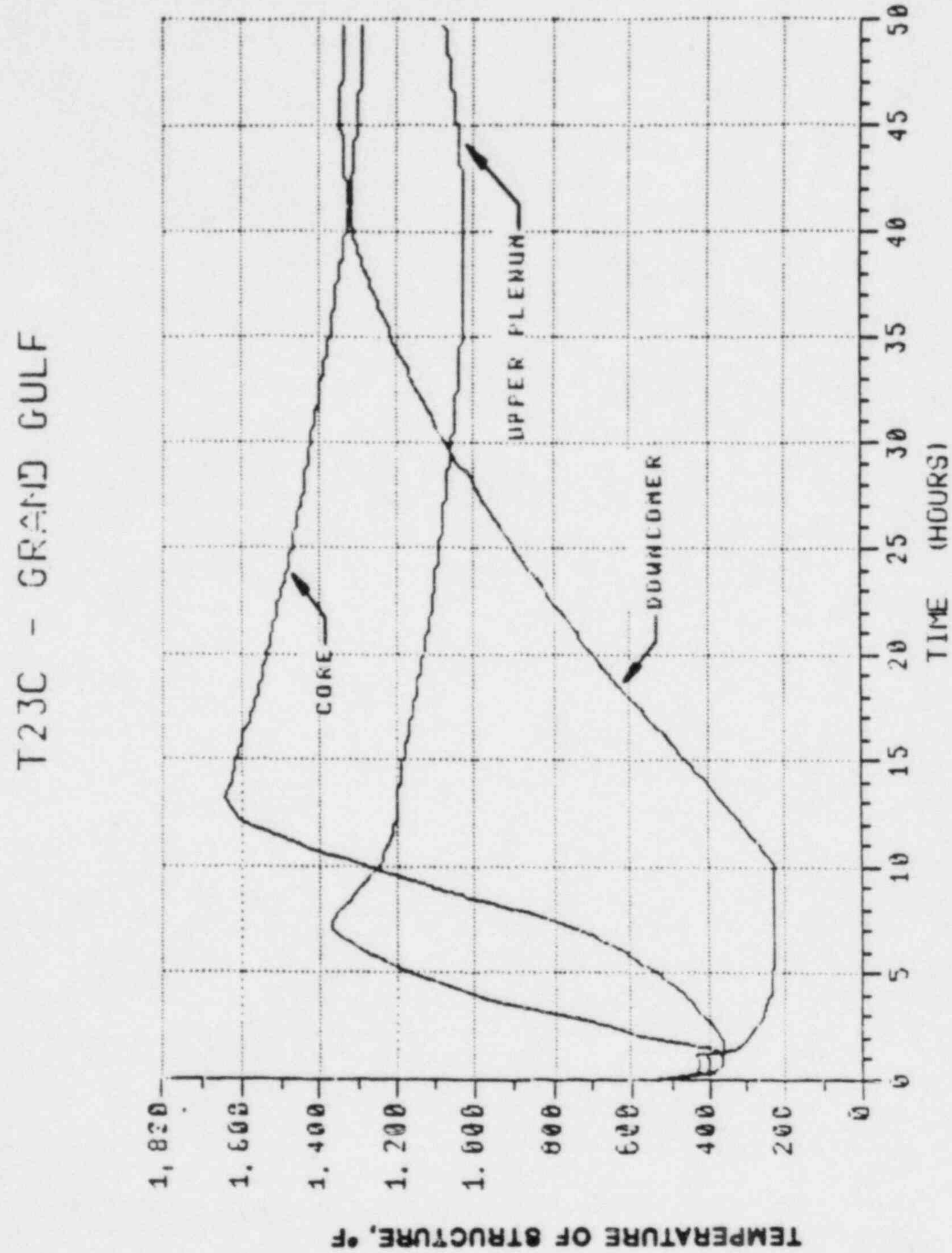

Fig. B.42 Temperature of structure, °F.

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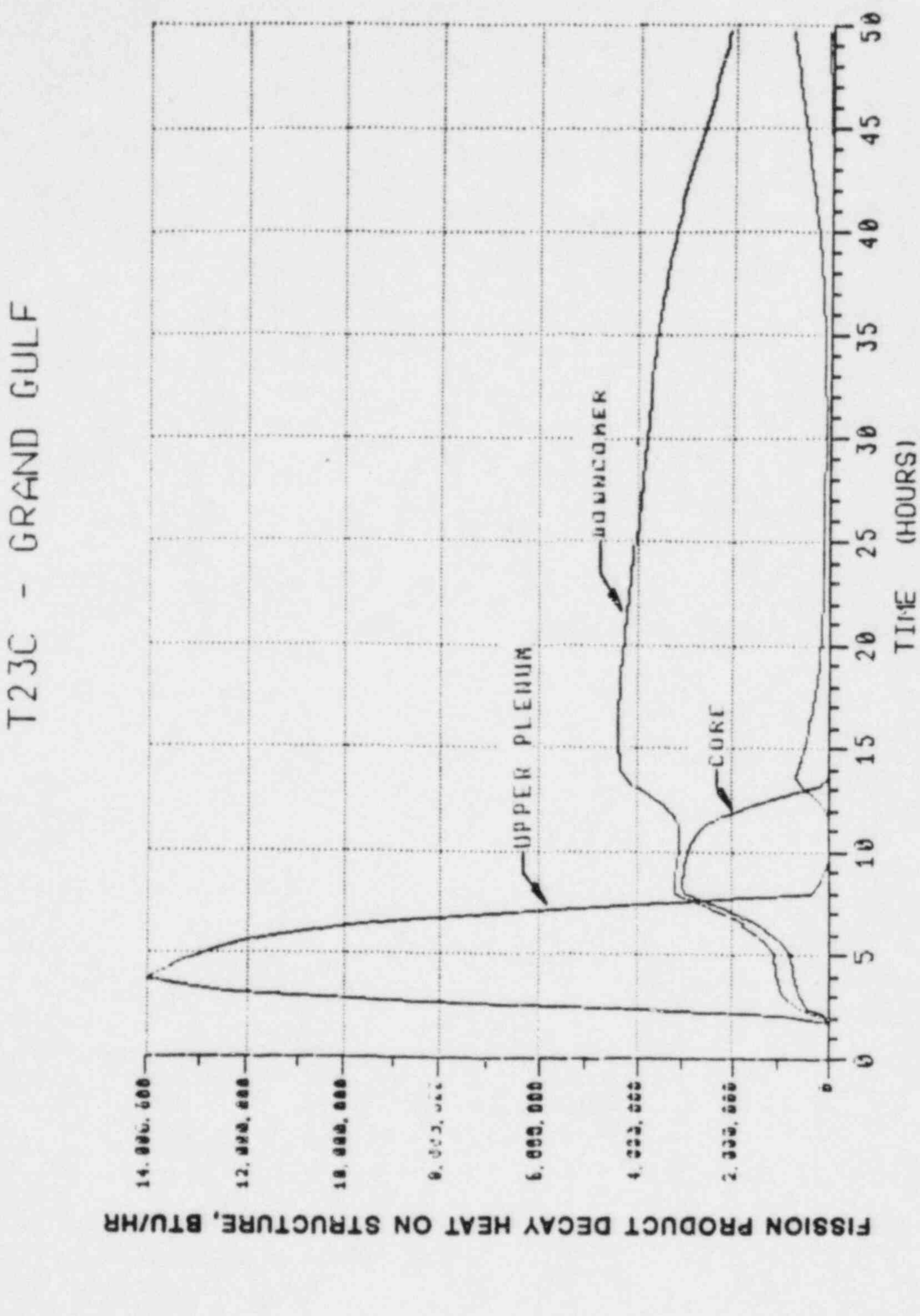

Fig. B.43 Fission product decay heat on structure, Btu/hr.

T23C - GRAND GULF

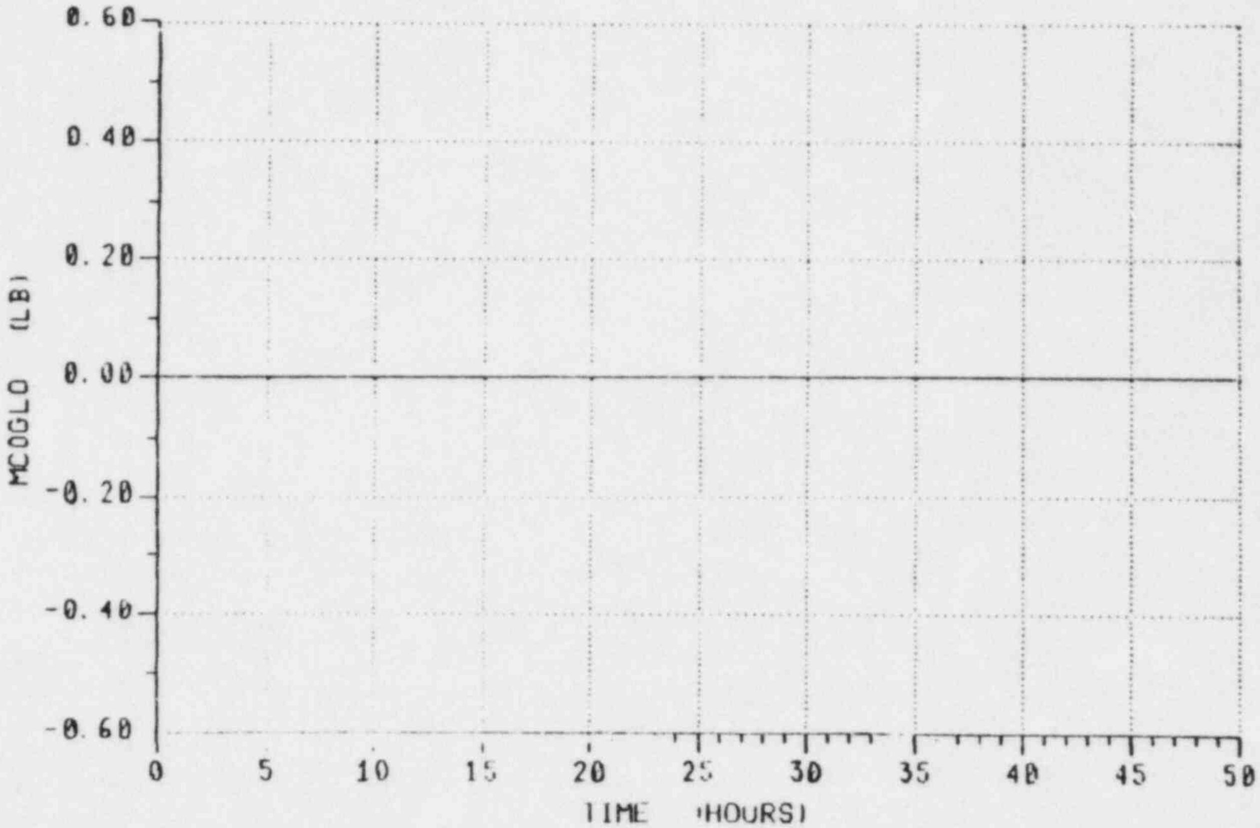

Fig. B.44 Total CO generated.

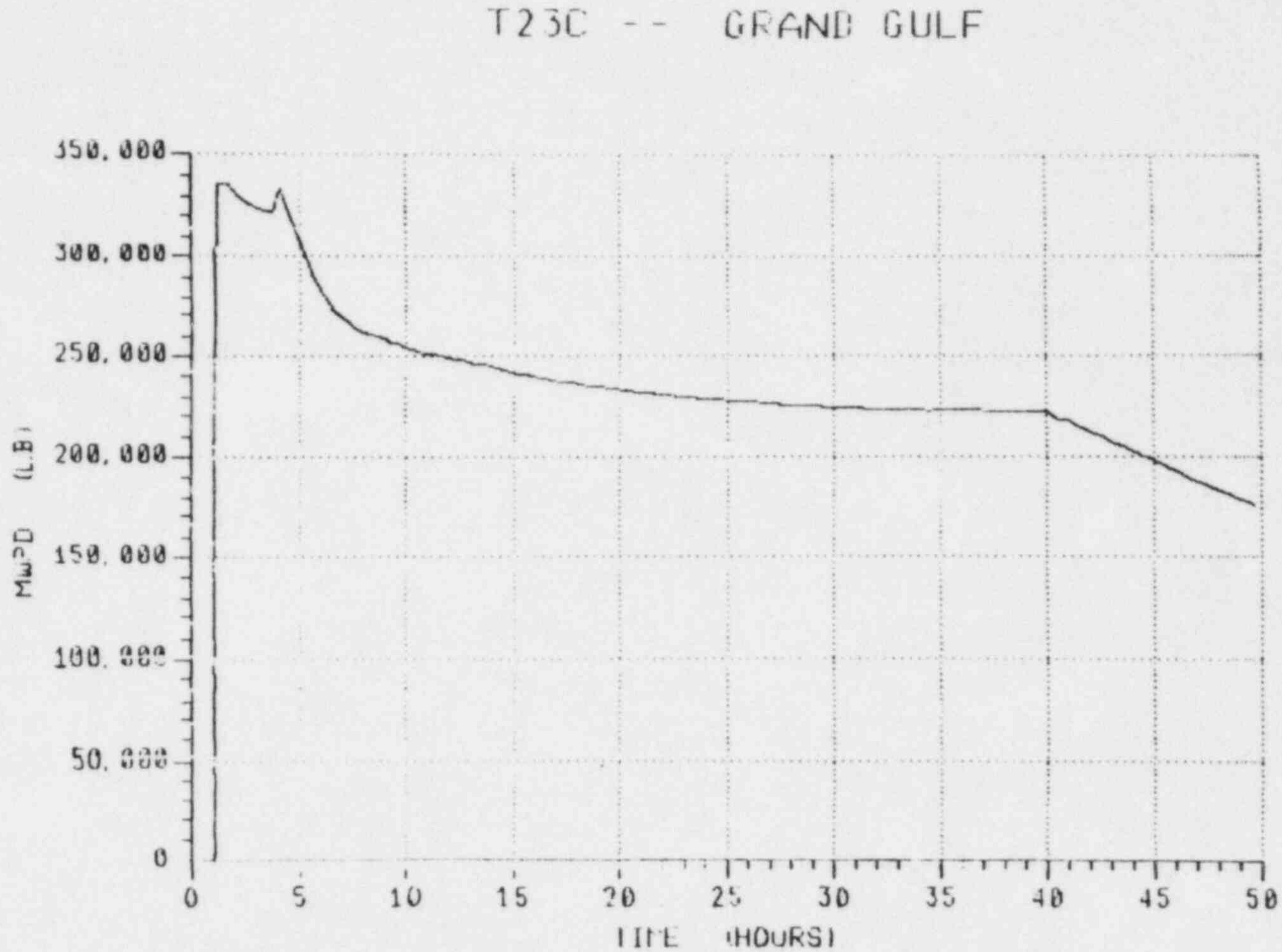

Fig. B.45 Mass of water in the pedestal.

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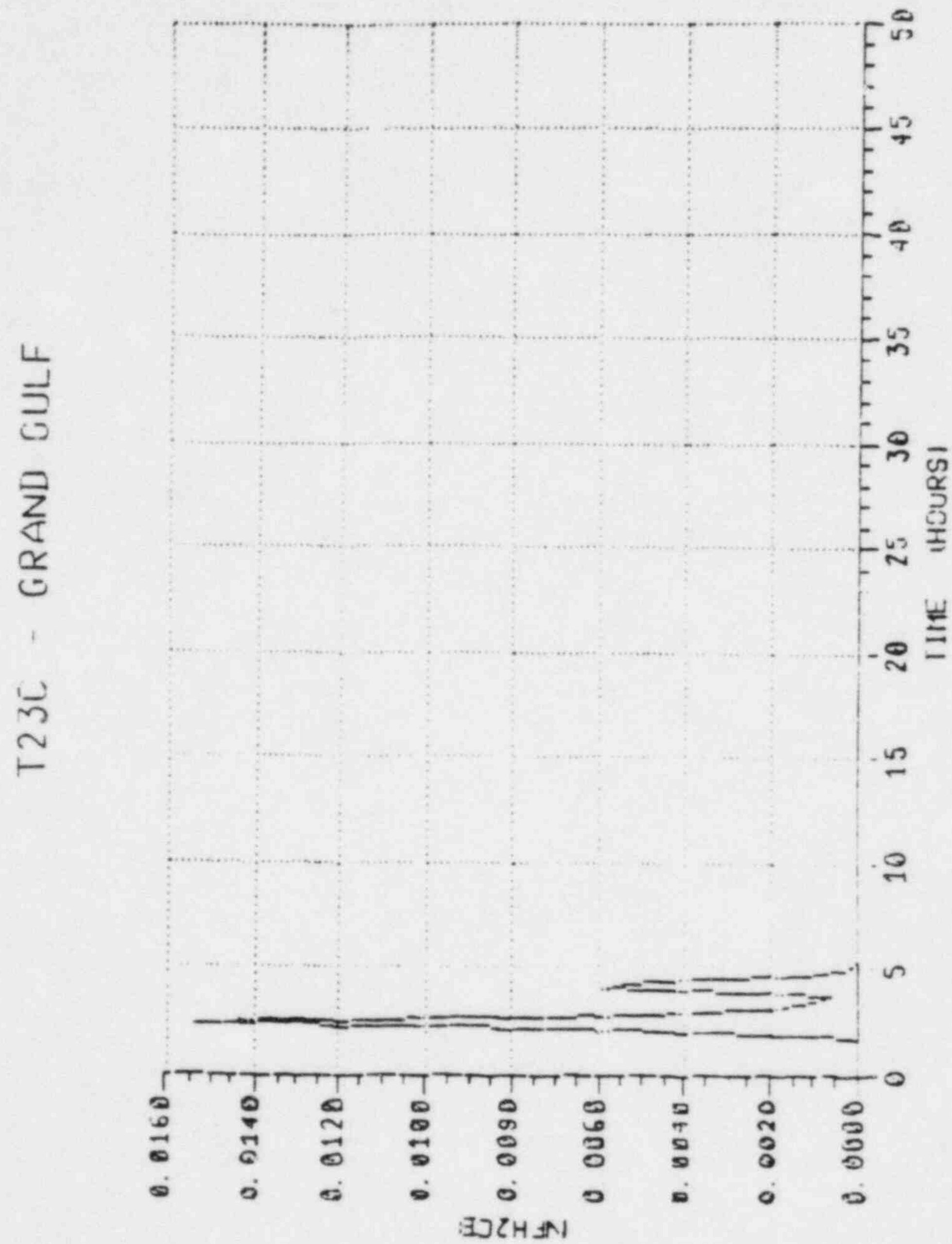

Fig. B.46 Mole fraction of H_2 in Compartment B.

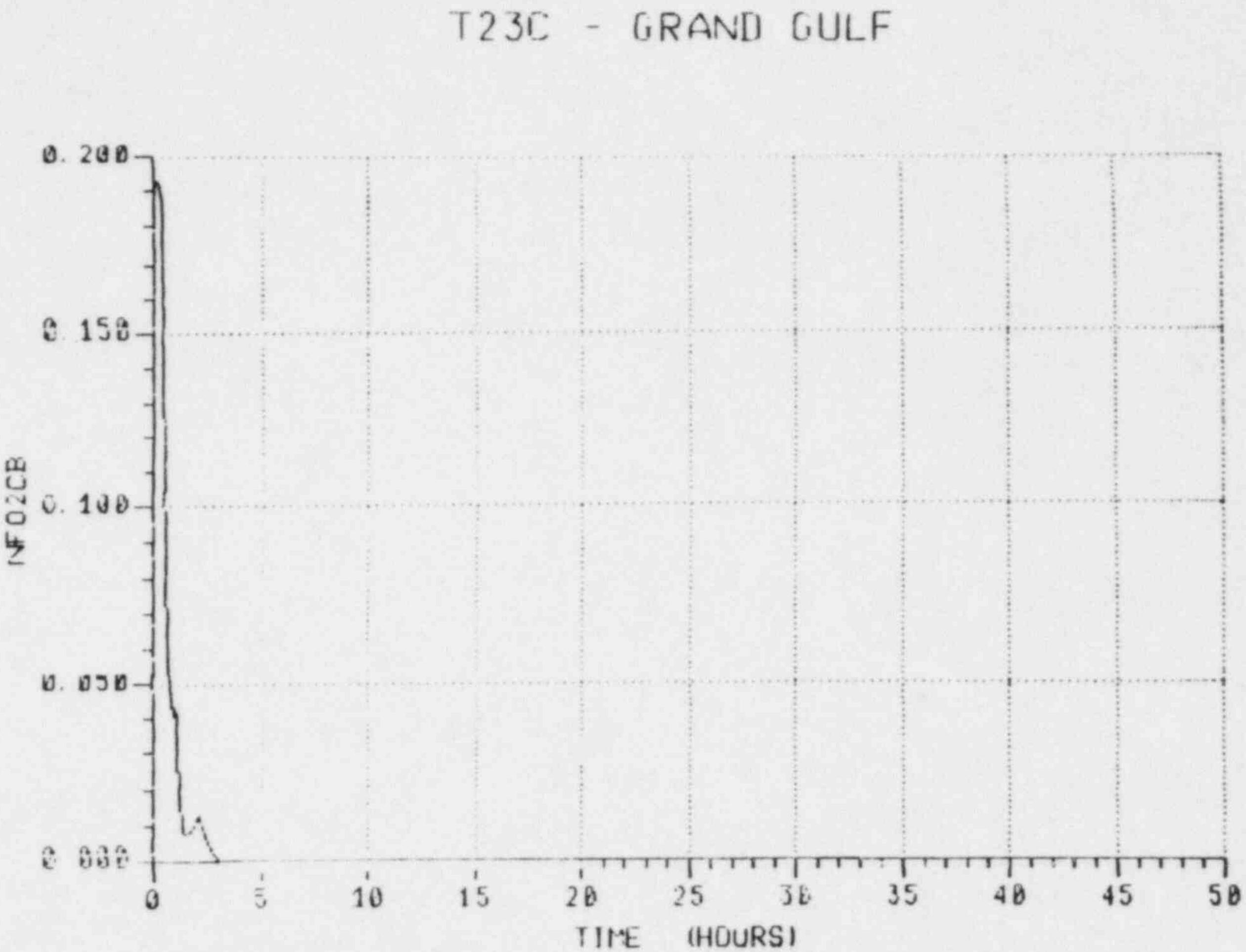

Fig. B.47 Mole fraction of O_2 in Compartment B.

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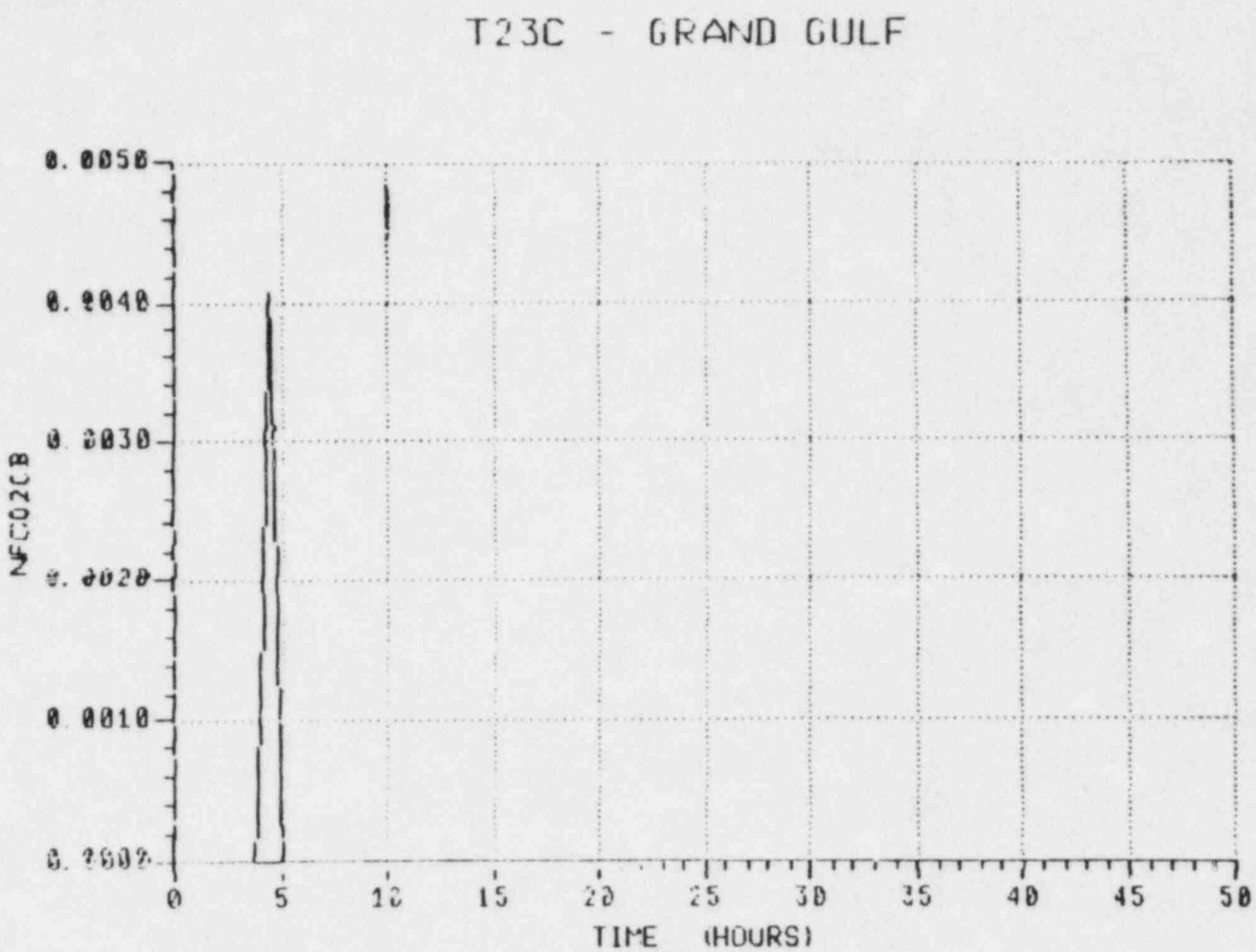

Fig. B.48 Mole fraction of CO_2 in Compartment B.

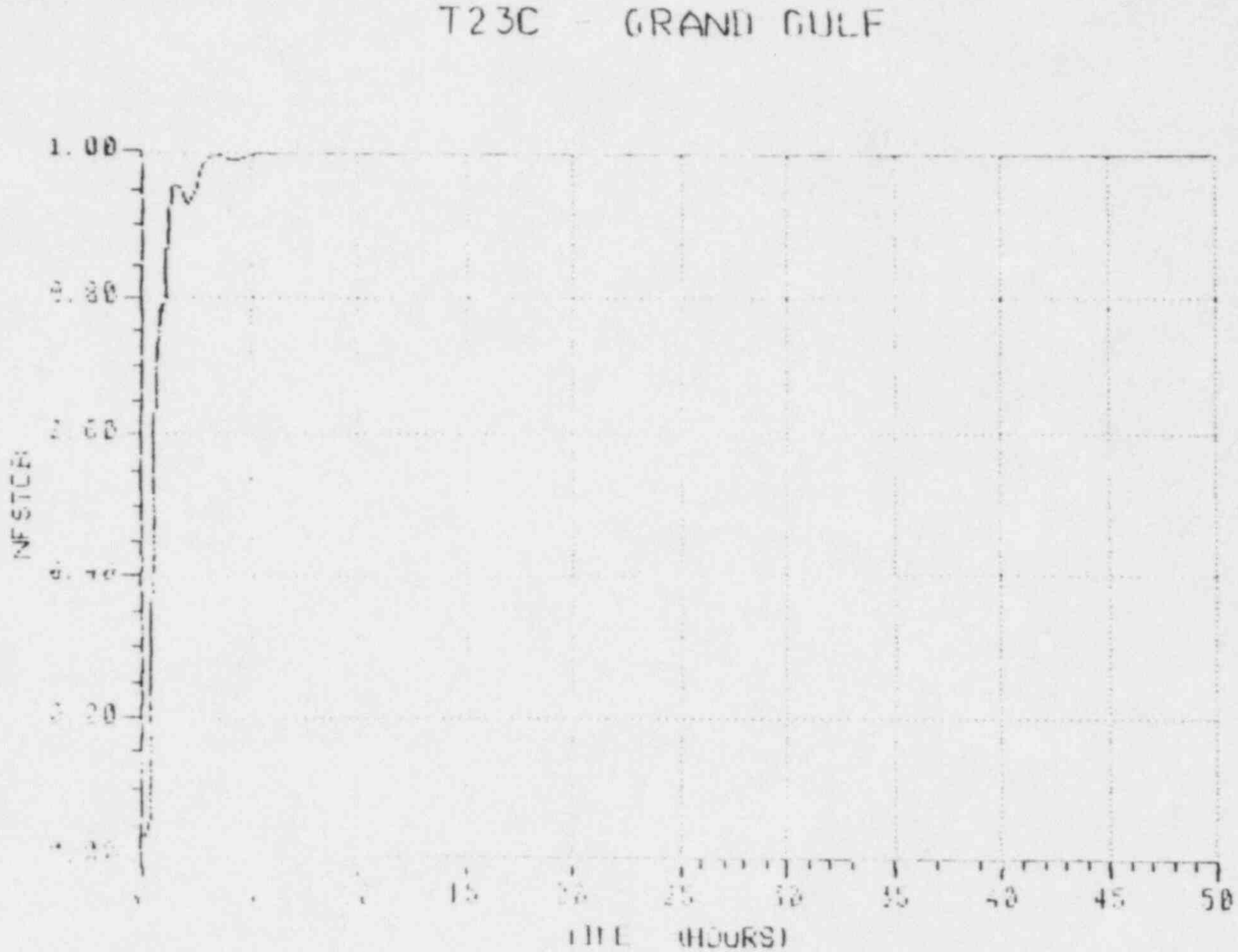

Fig. B.49 Mole fraction of steam in Compartment B.

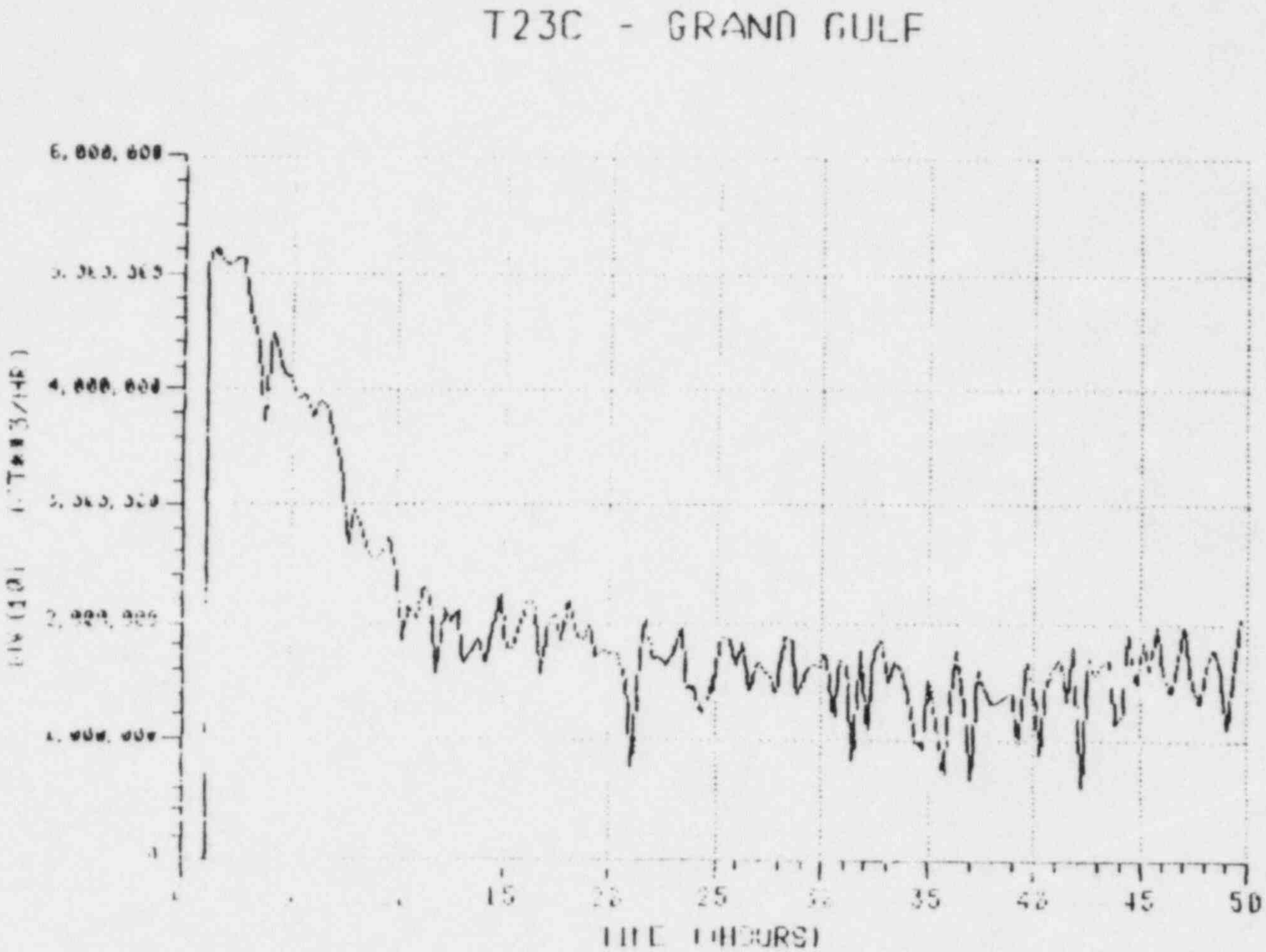

Fig. B.50 Volumetric flow out of containment.

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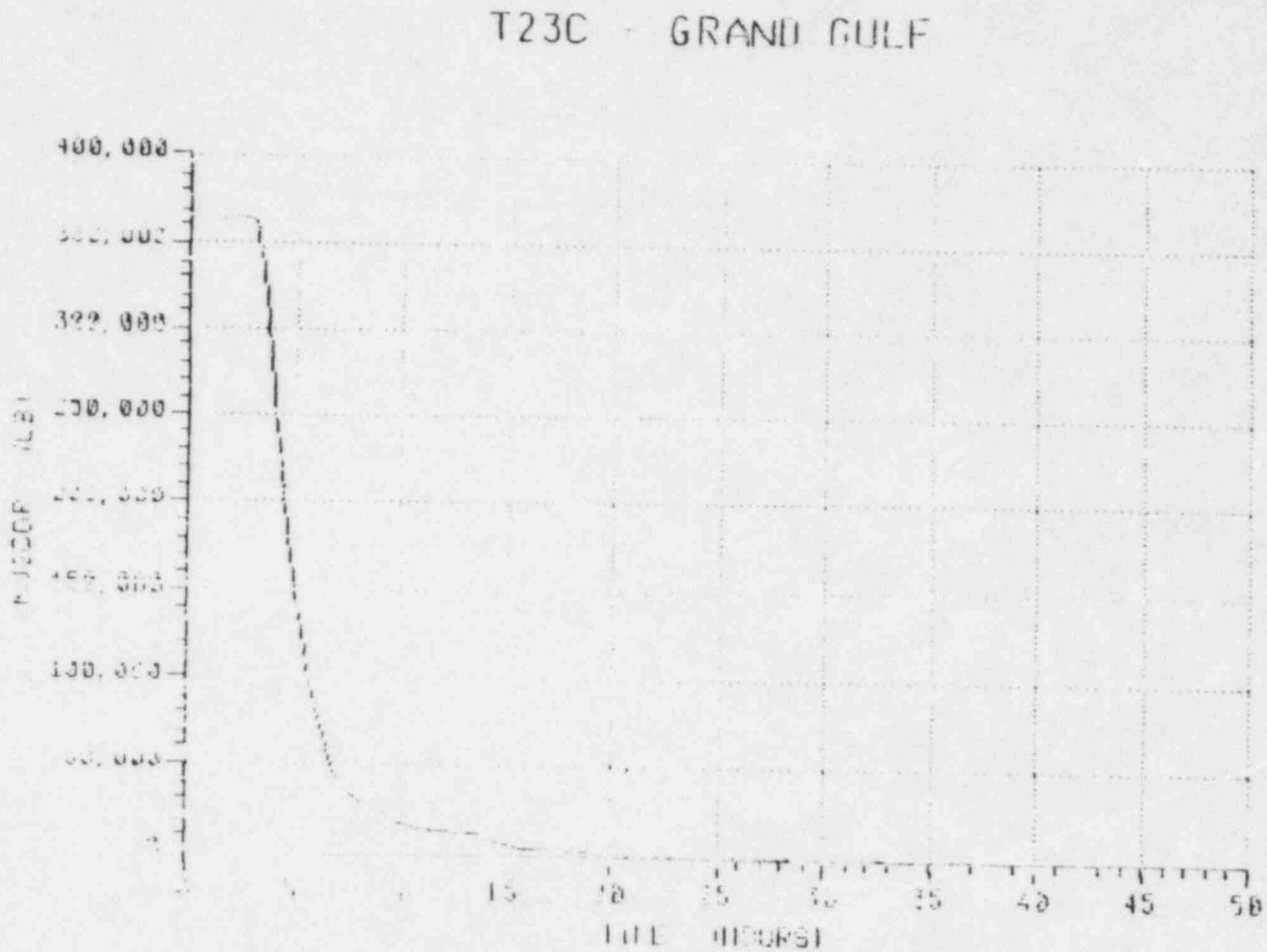

Fig. B.51 Mass of UO_2 in core region.