IDCOR

Program Report

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Technical Report 23.1

PEACH BOTTOM ATOMIC POWER STATION

INTERGATED CONTAINMENT ANALYSIS

8-7-84

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1.0 INTRODUCTION

1.1 Statement of the Problem

The main objective of this investigation is to calculate the response of the Peach Bottom Atomic Power Station (PBAPS) primary system and containment for selected postulated low probability, severe accident sequences representative of those which have been identified in Task 3.2 as dominant sequences potentially leading to core degradation and melting. This response is addressed on a best-estimate phenomenological basis. The study includes assessments of the effects of selected examples of operator interventions on the progression of these sequences.

This analysis is not intended to be a Probabilistic Risk Assessment in that no assessment of the probabilities of Peach Bottom systems or operator failures is included. However, the accident sequences were defined based on the WASH-1400 BWR analyses which identified those sequences most likely to lead to core melting. The results of these analyses indicate the time windows available for operators to implement mitigative actions. The effects of selected actions on accident progression are addressed. No attempt was made to model the variety of operator actions prescribed in the PBAPS emergency procedures. This approach is sufficient to demonstrate the effects that simple, individual actions would have on accident progression.

The results of the containment analysis are incorporated into an assessment of the fission product release and deposition within the various regions of the primary and secondary containment structures. For those sequences in which containment integrity is violated, the release of fission products to the surrounding environment is calculated. The influence of a few existing systems with operator action is described in Section 5.

1.2 Relationship to Other Tasks

The primary system and containment response analyses of IDCOR Subtask 23.1 are carried out with the Modular Accident Analysis Program. This includes models developed in IDCOR subtasks 11, 12, 14, 15 and 16 for thermal-

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hydraulic behavior as well as fission produce release transport and deposition within both the primary system and containment. The accident sequences analyzed were developed by considering the dominant core melt accident sequences presented in Subtask 3.2, Assess Dominant Sequences. Selected primary containment failure modes were chosen to demonstrate the radionuclide transport phenomena for the best-estimate analyses.

The ultimate structural capability of containments associated with the reference plants and other typical designs was assessed in IDCOR Subtask 10.1. This task defined the containment failure pressure and location assumed in Subtask 23.1 analyses for those sequences resulting in containment failure on overpressure.

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 to formulate assessments of the health consequences associated with the assumed accident scenarios. These health consequence analyses were then supplied to IDCOR Subtask 21.1 to evaluate effects on perceived risk.

Also, a few examples of operator interventions were analyzed to demonstrate their effects on the severe accident sequences analyzed for Peach Bottom -- that operator actions can terminate the accident sequence and achieve a safe stable state. The operator actions selected considered IDCOR Subtask 22.1, Safe Stable States, which discusses some of the inherent and intervention means of terminating the various core damage sequences.

It was not the intent of Task 23.1 to address the likelihood of occurrence of the particular sequences and operator actions, but rather to assume these situations and analyze the accompanying containment challenges and release of fission products utilizing the models developed within the IDCOR program.

Finally, it should be noted that the analyses developed as part of IDCOR Subtask 16.2 and 16.3 involve the detailed consideration of many different phenomena which are themselves considered in separate IDCOR subtasks.

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These include: hydrogen generation, distribution and combustion (12.1, 12.2 and 12.3), steam generation (14.1), core heatup (15.1), debris behavior (15.2), and core-concrete interactions (15.3).

Detailed considerations for each of the related subtasks can be found in the final reports submitted for the specific task. Individual issues will only be addressed in this report as required to understand the specific behavior obtained for the accident sequences considered and the specific design characteristics of Peach Bottom Atomic Power Station.

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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 key potential contributors to core melt frequency. These analyses consisted of plant thermal hydraulic response and fission product transport if the progression of the accident sequence led to core degradation and melting. These analyses include the performance of the ECCS systems and the containment engineered safety systems, such as the suppression pool, containment inerting, 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, as 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 PBAPS 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 "FPRAT User's Manual".

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 Peach Bottom response to postulated severe accidents. This code includes models for the primary system and containment response, fission product release, and fission product transport. In addition, the thermal hydraulic conditions as well as the fission product behavior are modeled for the reactor building which surrounds the primary containment.

3.1 Plant Specific Information

Each of Peach Bottom Units 2 and 3 is a single cycle, forced circulation, 3293 MW(t) General Electric BWR-4 producing steam for direct use in the steam turbine. Each unit has a Mark I primary containment housed in a secondary containment (reactor building) Both units went into commercial operation in 1974.

3.1.1 Nuclear System

The reactor vessel contains the core and supporting structure, the steam separators and dryers, the jet pumps, the control rod guide tubes, distribution lines for the feedwater, core spray, and standby liquid control, the in-core instrumentation, and other components. The main connections to the vessel include the steam lines, the coolant recirculation lines, feedwater lines, control rod drive housings, and core standby cooling lines.

The reactor vessel is designed and fabricated in accordance with applicable codes for pressure of 1250 psig. The nominal operating pressure is 1020 psia in the steam space above the separators. The reactor core is cooled by demineralized water which enters the lower portion of the core and boils as it flows upward around the fuel rods. The steam leaving the core is dried by steam separators and dryers located in the upper portion of the reactor vessel. The steam is then directed to the turbine through the main steam lines. Each steam line is provided with two isolation valves in series, one on each side of the primary containment barrier.

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When a scram is initiated by the Reactor Protection System, the Control Rod Drive system (CRD) inserts the negative reactivity necessary to shutdown the reactor. Each control rod is controlled individually by a hydraulic control unit. When a scram signal is received, high pressure water from an accumulator for each rod forces each control rod rapidly into the core. There are 185 control rods which enter through the bottom of the reactor vessel.

A pressure relief system, consisting of relief and safety valves mounted on the main steam lines, prevents excessive pressure inside the nuclear system following either abnormal operational transients or accidents.

Although process lines which penetrate the primary containment and offer a potential release path for radioactive material are provided with redundant isolation capabilities, the main steam lines, because of their large size and large mass flow rates, are given special isolation consideration. Two automatic isolation valves, each powered by both air pressure and spring force, are provided in each main steam line.

The Reactor Core Isolation Cooling system (RCIC) provides makeup water to the reactor vessel whenever the vessel is isolated. RCIC uses a steam driven turbine pump unit and operates automatically, in time and with sufficient coolant flow, to maintain adequate reactor vessel water level.

The Residual Heat Removal system (RHR) is a system of pumps, heat exchangers, and piping that fulfills the following functions:

1. Removal of residual heat during and after plant shutdown.

2. Injection of water into the reactor vessel, following a LOCA, rapidly enough to reflood the core and prevent excessive fuel clad temperatures, independent of other core cooling systems.

3. Removal of heat from the primary containment following a LOCA to limit the increase in primary containment pressure. This is accomplished by cooling and recirculating the water inside the

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primary containment. The redundancy of the equipment provided for containment cooling is further extended by a separate part of the RHR system which sprays cooling water into the drywell.

A number of Core Standby Cooling (CSC) systems are provided to prevent excessive fuel clad temperatures in the event a breach in the nuclear system process barrier results in a loss of reactor coolant. The four CSC systems are:

1. High Pressure Coolant Injection system (HPCI)

2. Automatic Depressurization System (ADS)

3. Core Spray System (LPCS)

4. Low Pressure Coolant Injection (an operating mode of the RHR system) (LPCI)

Although not intended to provide rapid reactor shutdown, the standby liquid control (SLC) system provides a redundant, independent, and different way from the control rods to bring the reactor subcritical and to maintain it subcritical as the reactor cools. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The system is sized to counteract the positive reactivity effect from rated power to the cold shutdown condition.

The standby AC power supply consists of four diesel generator sets. The diesel generators are sized so that three diesels can supply all necessary power requirements for one unit under postulated design basis accident conditions plus necessary power requirements for the safe shutdown of the second unit. The diesel generators start and attain rated voltage and frequency within 10 seconds. The diesel generator system is arranged with four independent 4-kV buses for each unit, each bus being connected to one diesel generator. Each diesel generator starts automatically upon loss of off-site power or detection of a nuclear accident. The necessary engineered safeguard system loads are applied on a preset time sequence. Each generator operates independently without paralleling.

Two independent sets of 125/250-V batteries are provided for each reactor unit. The sets are not interconnected. In addition, a separate 250-V battery is provided for each main turbine generator emergency bearing oil pump. One battery charger is provided for each battery.

The 125/250-V DC system is designed to provide an adequate power source for supplying the engineered safeguard loads of one unit, and the shutdown loads of the second unit, with concurrent loss of off-site power and any single failure in the DC system.

3.1.2 Containment

The primary containment is a pressure suppression system and houses the reactor vessel, the reactor coolant recirculation systems, and other primary system piping. The primary containment system consists of a drywell, a pressure suppression chamber which stores a large volume of water, a connecting vent system between the drywell and the suppression pool, isolation valves, vacuum breakers, containment cooling systems, and other service equipment.

In the event of a primary system piping failure within the drywell, reactor water and steam would be released into the drywell atmosphere. The resulting increased drywell pressure would force a mixture of drywell atmosphere, steam, and water through the vents into the suppression pool, resulting in a pressure reduction in the drywell due to steam condensation.

Vacuum breakers are provided in the vent headers and located in the suppression chamber, to equalize the pressure between the drywell and the suppression chamber. A vacuum breaker system is also provided between the suppression chamber and secondary containment. Cooling systems are provided to remove heat from the drywell and from the water in the suppression chamber. Appropriate isolation valves are provided to ensure containment of radioactive materials.

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The vent system conducts flow from the drywell to the suppression chamber and distributes this flow uniformly in the suppression pool. The suppression pool condenses the steam portion of this flow and the suppression chamber contains the noncondensable gases and fission products. The suppression chamber-to-drywell vacuum breakers and the suppression chamber-to-secondary containment vacuum breaker system limit the pressure differential so as not to exceed the design limit of 2 psi. The suppression chamber is designed for the same leakage rate as the drywell.

The suppression pool also provides for steam condensation during the actuation of a safety relief valve and the subsequent blowdown through the discharge piping. The dynamic suppression pool loads resulting from a safety relief valve discharge are reduced by a sparger (T-quencher) on the discharge end of the safety relief valve piping. The sparger also provides for uniform and stable condensation of steam in the suppression pool.

The stiffened pressure suppression chamber is a steel pressure vessel in the shape of a torus. It is located below and encircles the drywell, with a centerline diameter of approximately 111 ft. and a cross-sectional diameter of 31 ft. It contains approximately 123,000 ft^3 of water and has a gas space volume of approximately 132,000 ft^3. The drywell vents are connected to a 4 ft. 9 inch diameter vent header, in the form of a torus, which is contained within the airspace of the suppression chamber. Projecting downward from the header are 96 downcomer pipes, nominally 24 inch in diameter and terminating 4 ft. below the design water level of the pool.

The vent system outside the torus consists of eight circular vent pipes, each having a diameter of 6 ft. 9 inches. These vent pipes are connected to the vent header located inside the torus. The vent pipes and vent header have the same temperature and pressure design requirements as the containment. Jet deflectors are provided in the drywell at the entrance of each vent pipe to prevent damage to the vent pipes from jet forces which might accompany a pipe break in the drywell.

Pressure suppression pool temperature and pool level are continuously indicated in the main control room.

The RHR system can be placed into operation in the suppression pool cooling mode to limit the temperature of the water in the suppression pool. In this mode of operation, the RHR system pumps take suction from the suppression pool and deliver the water through the RHR system heat exchangers, where cooling takes place by transferring heat to the service water. The fluid is then discharged back to the suppression pool.

Another portion of the RHR system is provided to spray water into the primary containment as a means of reducing containment pressure following a LOCA. This capability is in excess of the required energy removal capability and can be placed into service at the discretion of the operator.

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) [3.1] 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 the MAAP. A complete Peach Bottom parameter file is given in Appendix A.1.

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, HPCI, RCIC, LPCI and LPCS as well as the standby liquid control system (SLCS), which 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

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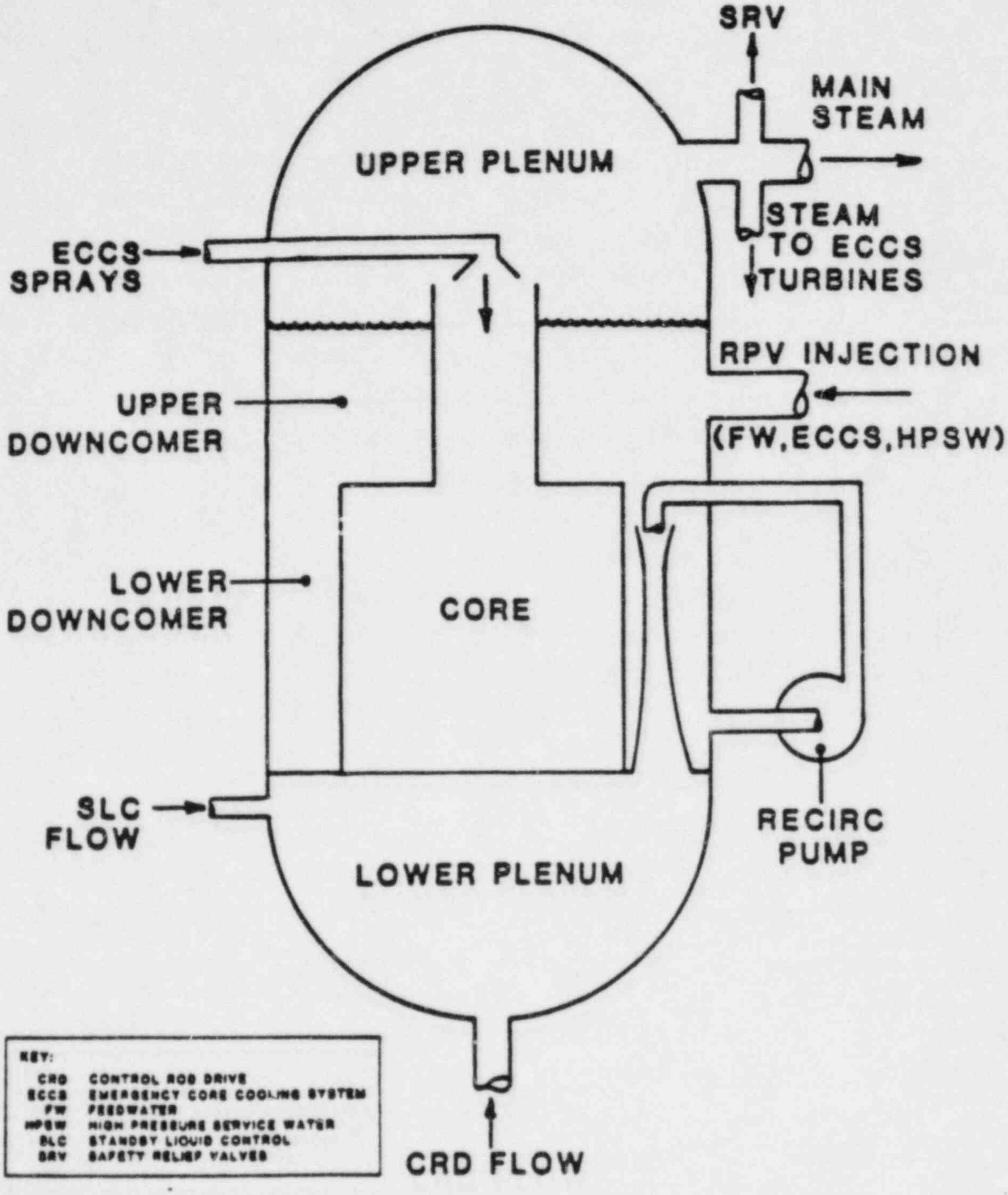

Fig. 3.1 Primary system model.

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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 which 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, (5) can be recovered before vessel failure, and predicts the time 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 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 of the fuel and cladding, including steam cooling and the oxidation of the Zircaloy clad and fuel channel cans as determined by the appropriate rate laws and oxygen starvation. The model permits incorporation of CRD flow to evaluate the potential of specific sequences, such as TW, being terminated with limited core damage.

The Mark I (Peach Bottom) containment nodalization scheme as shown in Fig. 3.2 separates the containment into: the pedestal, the drywell, and the wetwell 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 water throughout the containment due to drainage, vaporization, condensation and mass addition to assess the potential for cooling core debris should the vessel have failed. Separate water and corium temperatures are calculated for each containment compartment.

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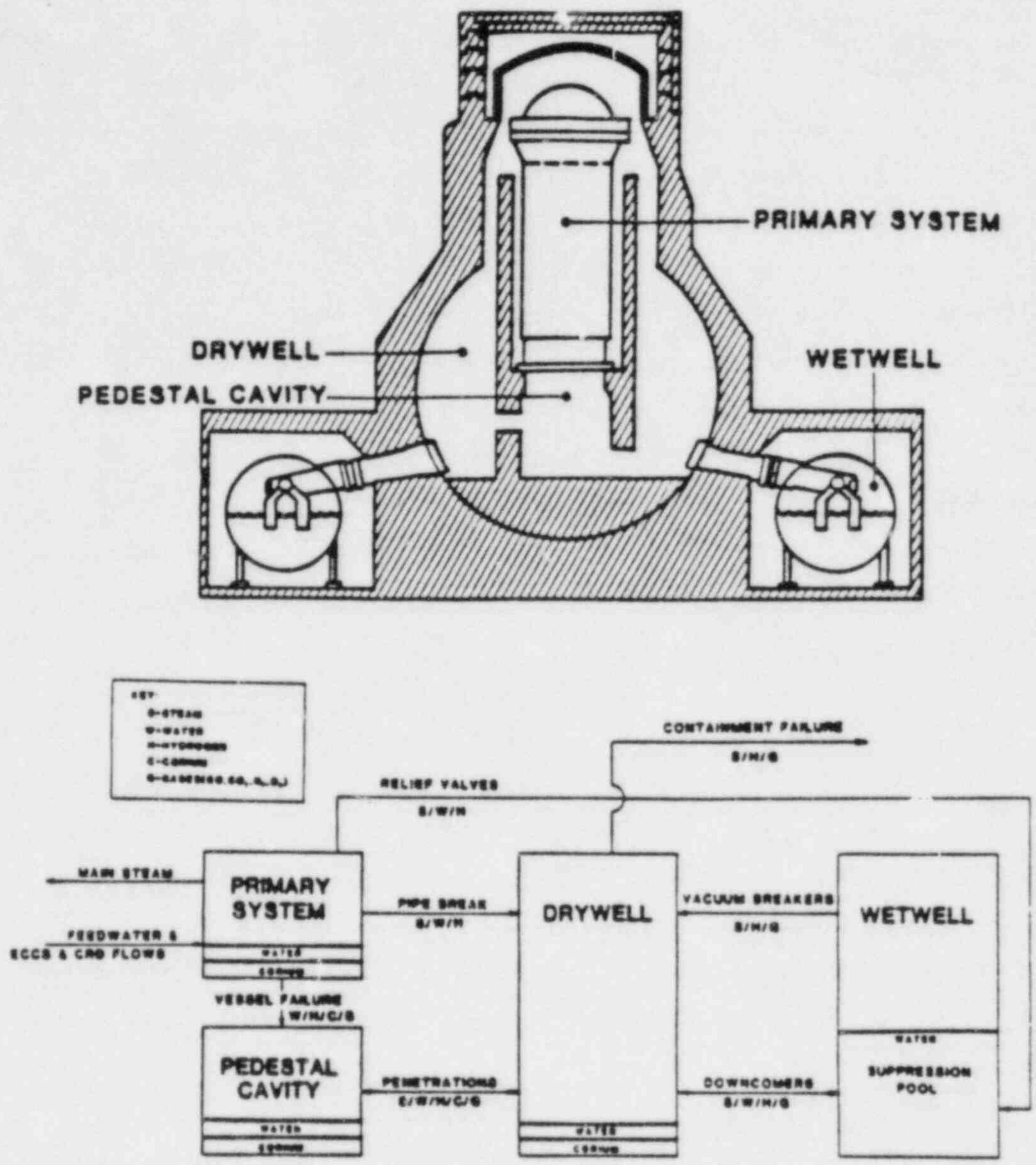

Fig. 3.2 Mark I nodalization.

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3.2.2 Peach Bottom 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 coolant injection (HPCI), 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 containment to model their response to postulated core damage and recovery scenarios.

3.2.3 Fission Product Release and Transport

The rate of release of fission products from the fuel matrix was calculated with the FPRAT module in MAAP. The FPRAT code was developed as part of the IDCOR program and is described in the report for Subtask 15.1B [3.2]. FPRAT was integrated into the MAAP coding structure such that the fission product release and transport from the core is evaluated at each time step.

The release of fission products due to corium-concrete thermal attack and ablation was calculated as described in Section 3.2.7. Transport of fission products through the primary system and containment was calculated with the CIRC module in MAAP and includes models for the fission product source terms, primary system compartment temperatures, primary system heat losses, gas flow due to forced and natural circulation, and steam condensation for steam and fission products.

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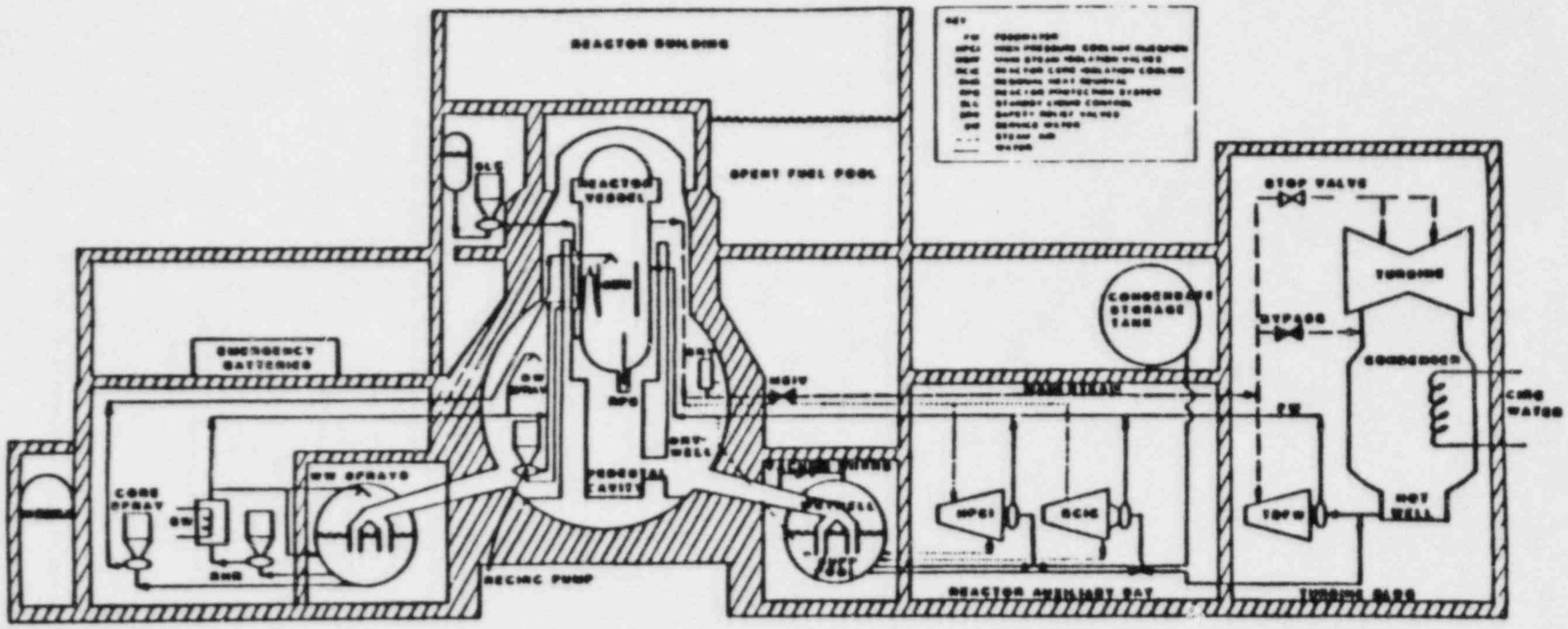

Fig. 3.3 BWR Mark I system features modeled in MAAP.

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Estimates of thermal-hydraulic characteristics of the flow of the containment effluent through the reactor building (secondary containment) were developed as described in Section 3.3.

3.2.4 Fission Product Release from Fuel

The initial fission product inventories were obtained from Ref. [3.3] and are given in Table 3.1. Fission product release rates depend on fuel temperature history during heatup, and on the flow through the core. 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 are 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 for 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.

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

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Table 3.1

INITIAL INVENTORIES OF FISSION PRODUCTS
AND STRUCTURAL MATERIALS RELEASED AS AEROSOLS

Fission Products	Initial Inventory (kg)
Kr	25.7
Xe	387
Cs	207
I	16.6
Te	34.9
Sr	62.7
Ru	172
La	98.3
Mo	237
Sn	1050
Mn	432

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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 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.

3.2.5 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, fission products could 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 BWR-CIRC module models natural circulation flows within the primary system. This includes descriptions for fission product heat generation, material vaporization, condensation and deposition. Also, this 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.

Since natural circulation flows are driven by the gas density differences between various regions, and since the volatile fission products are dense vapors, the gaseous flows must have a detailed accounting of the gas

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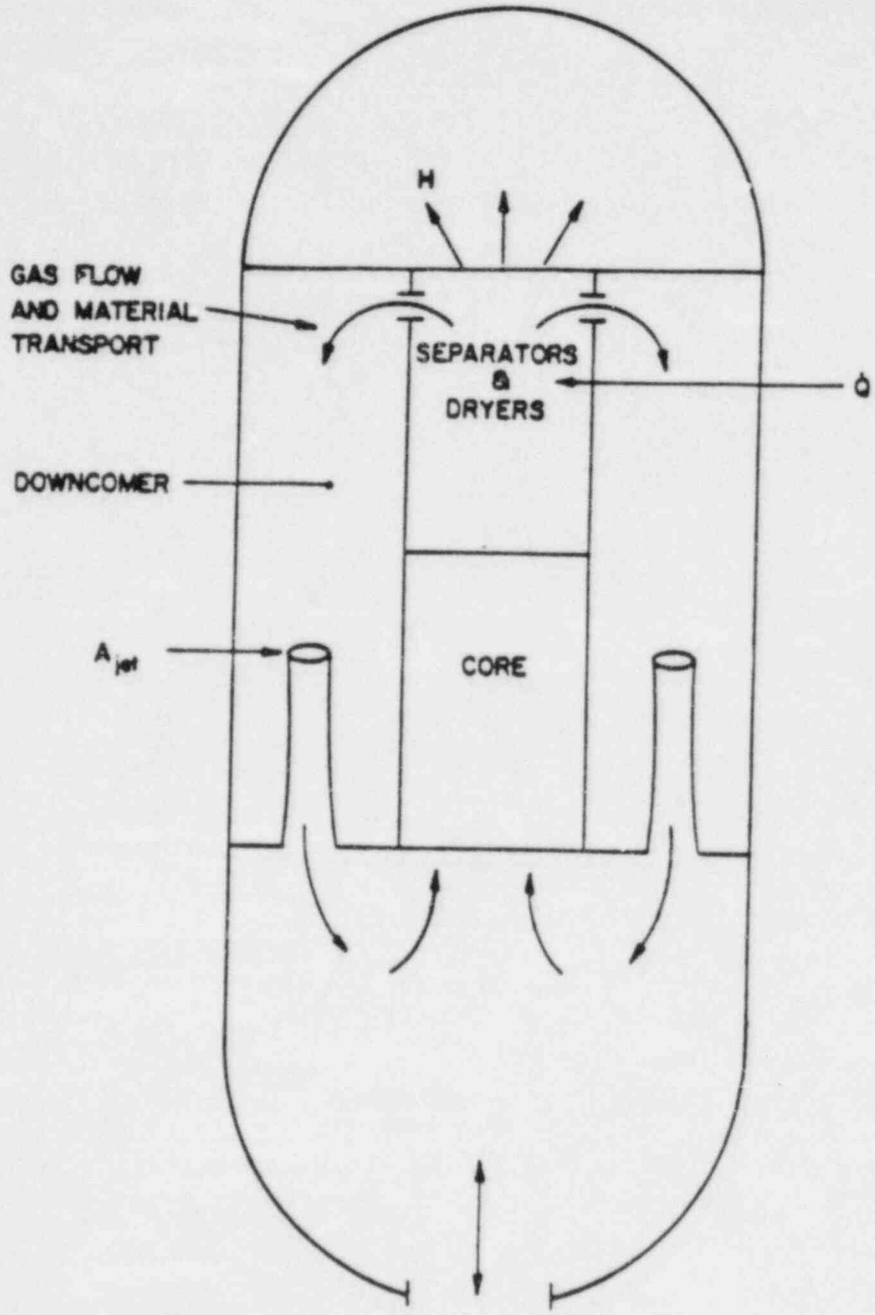

Fig. 3.4 BWR natural circulation model.

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mixture properties in the various nodes. In addition, with the reflective insulation used on the Peach Bottom 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 released 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.

3.2.6 Aerosol Deposition

IDCOR Task 11.3 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 [3.4]. 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 also 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 are illustrated in Fig. 3.5 are for the primary system and containment and in Fig. 3.6 for the reactor building and the SGTS.

The key assumptions in the aerosol modeling are:

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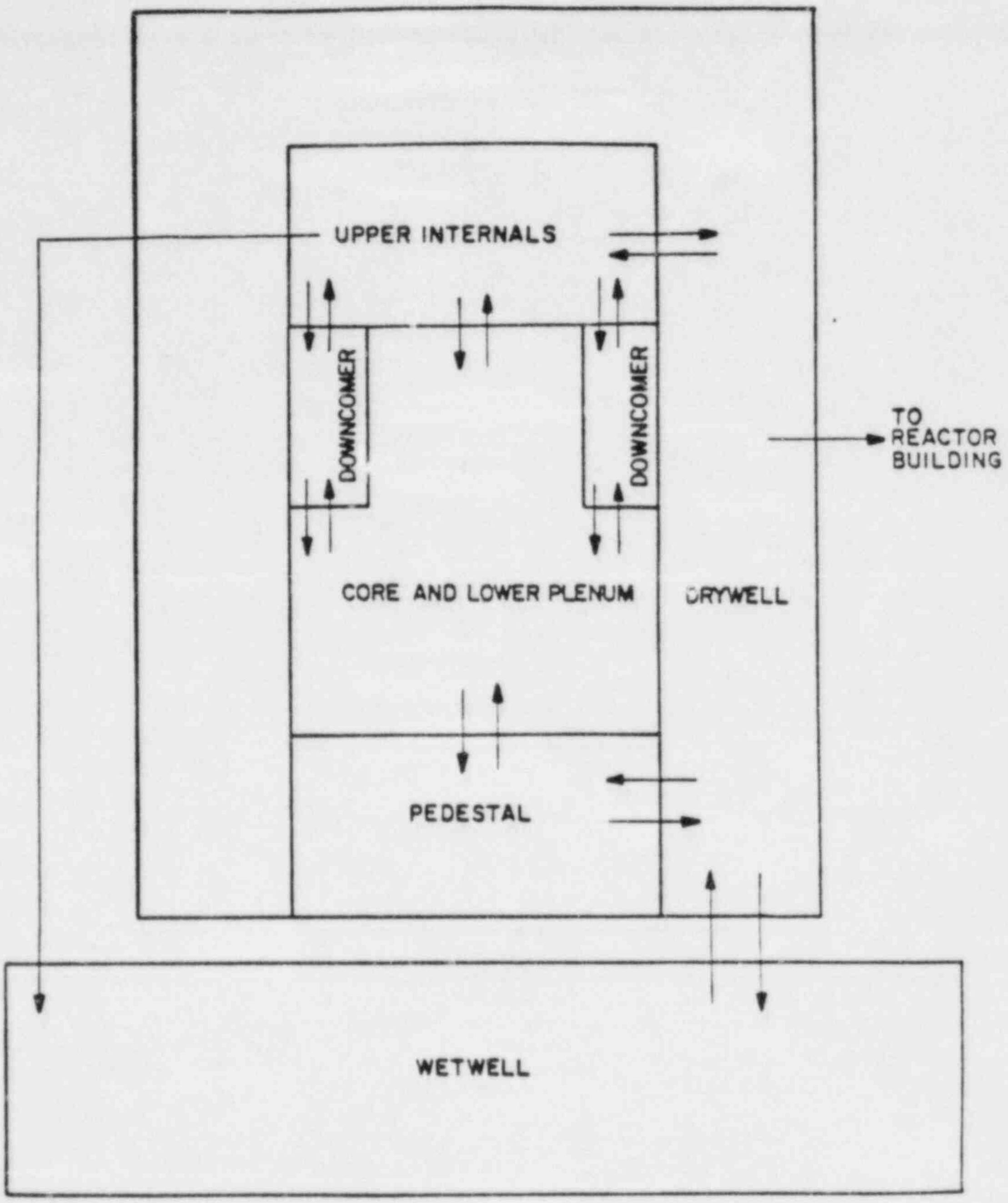

Fig. 3.5 Fission product transport paths for the primary system and containment.

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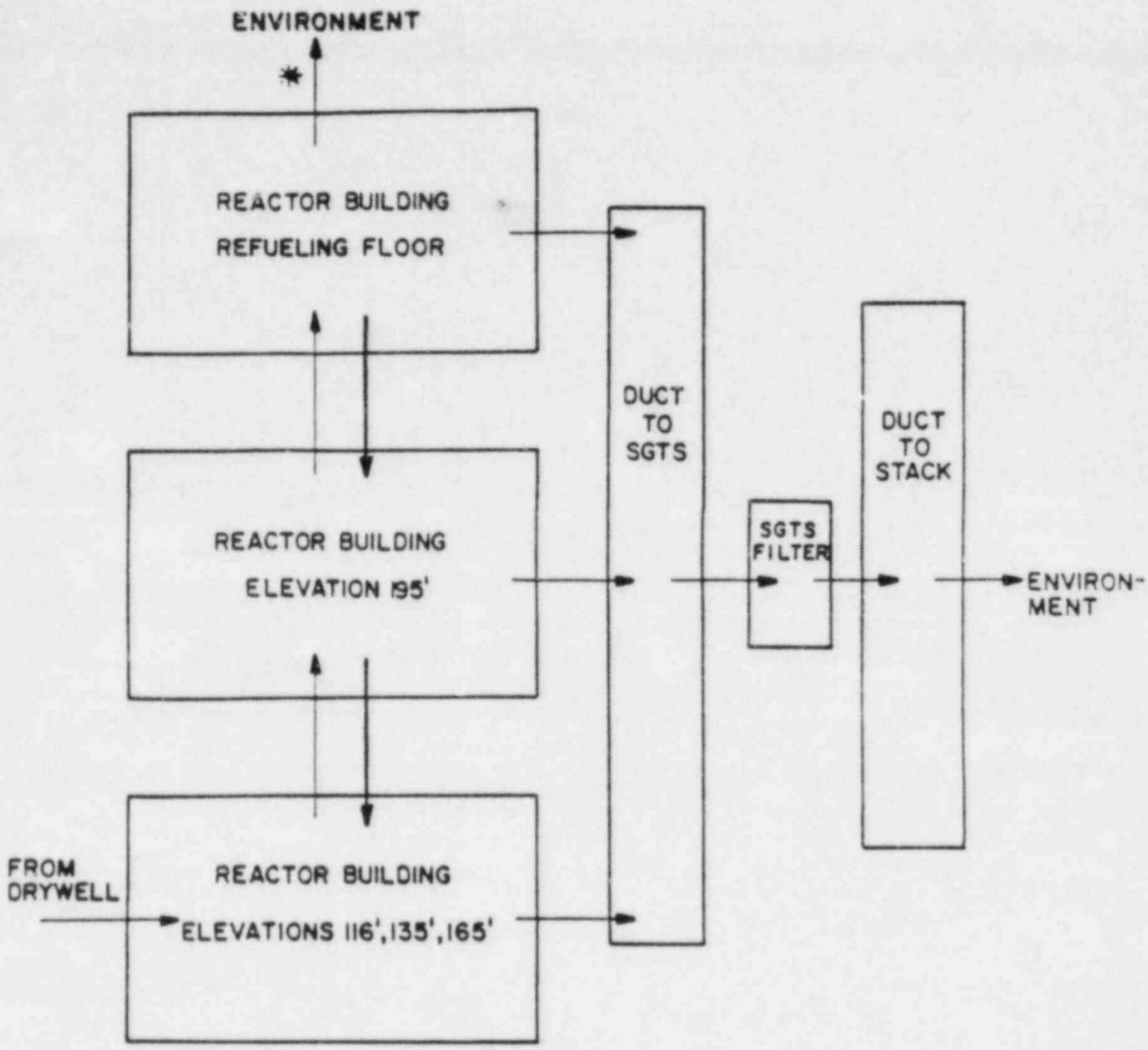

*FLOW PATH FOR TQVW SEQUENCE, ALSO PORTION OF FLOW GREATER THAN SGTS CAPACITY FOR THE OTHER SEQUENCES

Fig. 3.6 Nodalization scheme for the reactor building.

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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 release through the spargers and 600 for releases through the downcomers.

3. Compartments representing the release pathway are: three regions in the reactor vessel, pedestal, drywell, wetwell, reactor building, standby gas treatment system (SGTS) ducts, and SGTS charcoal filter (physical removal mechanisms only). Figures 3.5 and 3.6 depict the release pathways and compartments for the analyses. Specific release paths and the flow rates are dependent on the thermal-hydraulic conditions in the reactor building as well as the flow capacity of SGTS and percent steam composition of the carrier gas as determined by the thermal hydraulic models in MAAP (see Section 3.3).

4. Steam carrying fission products out of the containment and along the release pathway would condense in the cooler reactor building. Steam condensation rates, volumetric gas flows through the reactor building and temperatures are calculated as described in Section 3.3.

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

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

3.2.7 Fission Product and Aerosol Release from Core-Concrete Attack

The release of aerosols due to core-concrete attack was not included in the Peach Bottom analysis. This omission leads to an underprediction of the overall fission product removal in the primary and secondary containments.

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3.3 Analysis of Reactor Building Thermal-Hydraulic Conditions

3.3.1 Reactor Building and Standby Gas Treatment System (SGTS)

Each of Peach Bottom Units 2 and 3 primary containments is housed in a multilevel reactor building. Under accident conditions, the reactor building atmosphere is isolated from the normal ventilation system and exhausted through the Standby Gas Treatment System (SGTS) HEPA and charcoal filters at a rate which maintains the building pressure negative relative to the environment. The reactor building and SGTS comprise the secondary containment system at Peach Bottom.

As shown in Fig. 3.7, the reactor building is divided into five major levels with gaseous flow communication between them through open hatches. An equipment transfer shaft from the 135' level up to the refueling floor is the major pathway for communication between the various volumes of the reactor building. The lowest elevation (~ 92' to 133') contains the torus room and torus, and the RHR and core spray pump rooms located in the four corners of the building separated from the torus room by concrete walls and water tight doors. The next elevations (135' to 163') contain the main steam pipe tunnel, components of the CRD hydraulic system, neutron monitoring system and other instrumentation partially separated by several partition and shield walls. Elevation 165' to 193' contains auxiliary pumps, heat exchangers and instrumentation separated by many partitions and shield walls creating substantial interior surface area. Much of the space in elevation 195' to 232' is occupied by the spent fuel pool and steam separator and dryer storage pool. This level also contains the standby liquid control system and reactor building fan room. Therefore, much of the volume is closed off by major walls and available surface area is less than that available in the lower elevations. The top elevation (234' to roof at 296') is essentially a wide open area comprising the refueling floor. Most of the exterior wall area is insulated corrugated sheet metal on this elevation as compared to concrete on all lower elevations.

The SGTS takes suction from multiple intakes located on all major elevations in the reactor building. Tests have indicated that reactor

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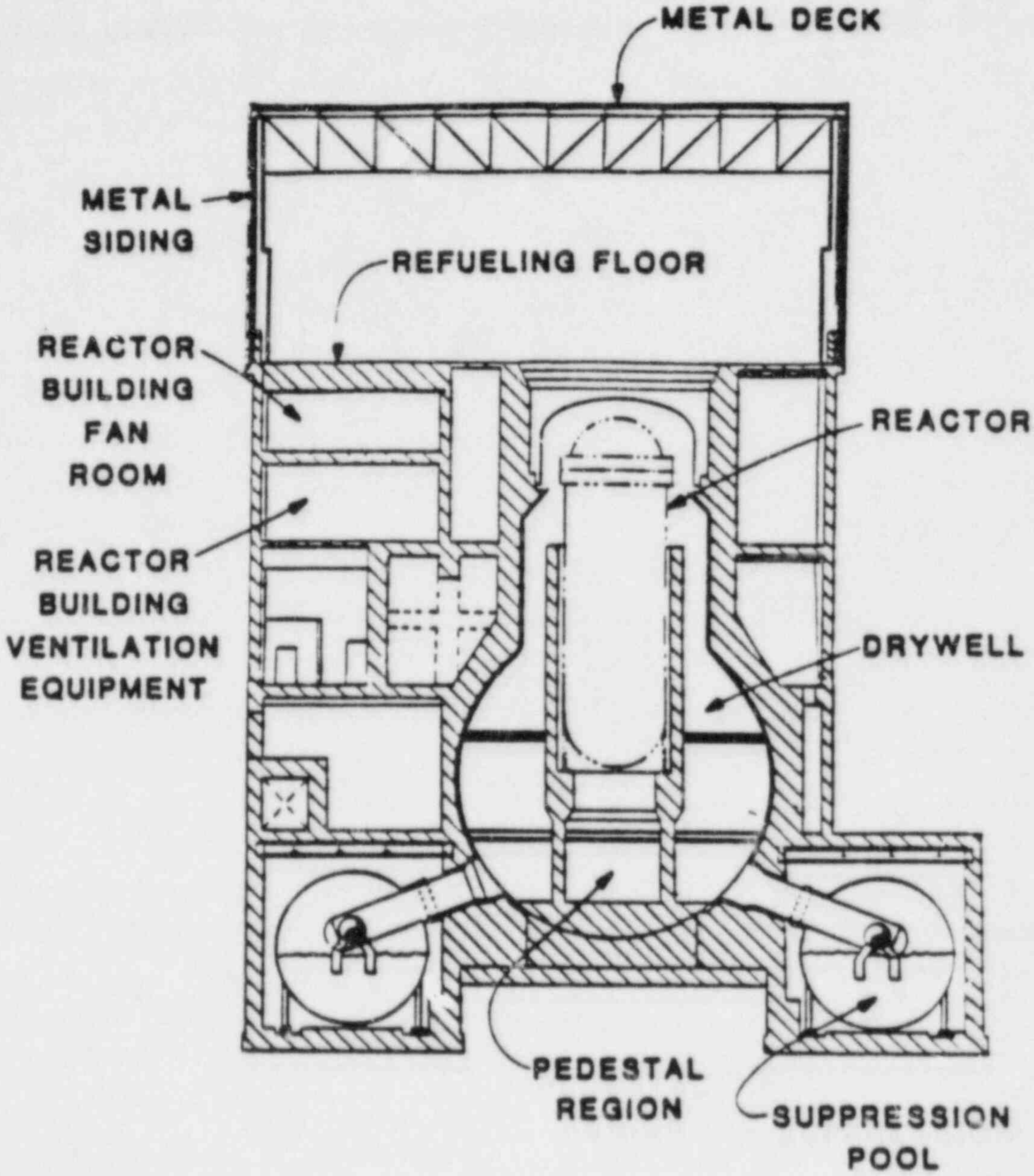

Fig. 3.7 Peach Bottom secondary containment.

building inleakage at Unit 2 is approximately 5500 cfm when the building is isolated and SGTS is operating. System design capacity is 10,500 cfm. Another design feature is the fusible links which close the fire dampers on the SGTS if the temperature in a given region of the building exceeds 74°C (165°F). This temperature limitation can control the SGTS transport path for many accident sequences.

3.3.2 Modeling Approach

A separate computer code was constructed from MAAP subroutines to model the reactor building, which can be divided into many nodes, to represent the major regions in the building. As noted earlier, the equipment transfer shaft provides a path for natural circulation between the major volumes. Estimates of the compartment temperature differences and the resulting natural circulation flows under accident conditions show the circulation flows between volumes to be large compared to the through flows. As a result, the building can be represented as a single volume. This provides for some conservatism in the analysis, since this somewhat overestimates the aerosol concentration in the upper region of the building and thus overestimates the release to the environment. In addition to the natural circulation flows between compartments, the circulation flows within a compartment due to temperature differences between the gas and the compartment walls can also be important. These effects are also included in the analysis. The building is assumed to be pressure equilibrated throughout the accident. As a result of this equilibration, flow is driven through the reactor building as determined by the source coming from the primary containment following wetwell or containment failure, and the imposed flows resulting from the SGTS.

The SGTS, which provides a suction flow at each elevation,is normally fed by inleakage from the outside, will then flow passing through the respective volume and into the ducts and filters in the system. This flow is partitioned between the various elevations and is represented in the computer model by a specific suction flow at each elevation. The inleakage at each elevation is determined by the strength of the source at that elevation, including flow from other compartments. For example, following containment failure, the region at the equator of the drywell has a substantial source

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from the drywell which can be greater than the suction flow for that elevation. As a result, the inleakage from the environment to this volume would be reduced to zero with any excess flow going to the higher elevations. If the source flow is less than the required inleakage from the environment, then the inleakage is set to be the difference between that required by the SGTS suction from the volume and the source flow. The refueling floor has a direct connection to the environment representing the leakage through the sheet metal siding or the opening of blowout panels.

The physical processes occurring in each volume, including heat losses to the structural surfaces, thermal profile within the structures and steam condensation are treated in the same manner as the primary containment compartments in the MAAP code. These processes as discussed in detail in the MAAP User's Manual Volume 2 under the subroutine titles of PTCAL and HTWALL. Since the reactor building volumes are coupled by the equipment shaft, small open hatches, etc., water accumulation on each elevation is assumed to drain to the lowest part of the reactor building and is neglected in the remainder of the calculation. The relative rates of single phase and two-phase energy transfer determine the response of each reactor building region and subsequently determine the flow between the three compartments. It should be noted that with significant condensation, excess flow can be required from the environment back into the building in addition to the normal inleakage associated with the SGTS.

With the models for flow between compartments and condensation within individual compartments and the source term from the drywell following containment failure, the resulting aerosol agglomeration and removal can be assessed. This is also evaluated using the aerosol deposition model in MAAP discussed in Section 3.2.

3.3.3 Model Inputs

As discussed in Section 3.3.2 and illustrated in Section 3.4.3, the reactor building is characterized by three major volumes which are intimately coupled. The first volume (or compartment) represents the lower three major elevations of the reactor building for those sequences in which SGTS is

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assumed to be operating. The second compartment represents the volume above elevation 195', and the third compartment represents the volume above the refueling floor, elevation 234'. For sequences in which the SGTS would be unavailable, the reactor building is analyzed as a single node due to the coupling between compartments by natural circulation in the equipment shaft. The parameters and values used to characterize each of these compartments are listed in Table 3.2.

The SGTS exhaust rates from each compartment used in the model are also presented in Table 3.2. These rates are based on a total SGTS flow of 10,500 cfm assuming that the initial pressure spike following containment failure causes increased building leakage area, (i.e. blowout panels) resulting in maximum SGTS flow as the system tries to maintain a negative pressure in the reactor building. The flows are proportioned for each compartment in accordance with plant data for the actual flows from the elevations. In accordance with the design, if a compartment temperature exceeds the 74°C limit on the fusible links for the fire dampers, the SGTS flows in that compartment are zero thereafter.

3.3.4 Influence on Fission Product Release

The reactor building completely surrounds the primary containment for the Mark I configurations, and as a result, would receive the fission products released following containment failure. Since this building is in a direct path for the release, it is an important part of the fission product retention for this reactor design. In particular, it has a substantial influence on retaining the fission products within the plant and limiting the subsequent release to the environment. The large volume represented by the reactor building provides a substantial residence time for materials released from the primary containment and significant deposition occurs due to vapor condensation, gravitational settling, and steam condensation. Flow through the building is particularly important for sequences in which there is no SGTS flow, since the release to the environment is determined by this flow.

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Table 3.2

REACTOR BUILDING MODEL INPUTS

Compartment	Volume (m^3)	Surface Area (m^2)	SGTS Exhaust* (m^3/sec)
1	36,500	15,800	2.83
1 (Without SGTS Flow)	9,940	10,200	-
2	4,600	2,500	0.47
3	27,450	4300 (Steel)	1.42

*SGTS exhaust is zero for the station blackout sequence. Also, the SGTS flow from each compartment is zero if the compartment temperature exceeds the 75°C limit for the fusable links on the fire dampers.

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Inclusion of the SGTS with the associated suction flows for each major volume increases the flow through the reactor building volumes, thereby decreasing somewhat the material deposited in these volumes but provides for deposition within the SGTS filter system if it has not been overburdened by moisture. For sequences in which the SGTS is available, the principal release to the environment is determined by the flow through the filters and through the stack to the environment.

The influence on each individual sequence considered is discussed in Section 6. However, these all illustrate that the reactor building has a substantial influence on retention of fission products for the Mark I containment design.

3.4 References

3.1 MAAP - Modular Accident Analysis Program, User's Manual Volume II, August, 1983.

3.2 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.

3.3 J. A. Gieseke, et al., "Radionuclide Release Under Specific LWR Accident Conditions," Draft Version of BMI-2104, Battelle Columbus Laboratories Report, July, 1983.

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

3.5 IDCOR Technical Report, "MAAP Models for Aerosol Deposition," to be published.

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4.0 PLANT RESPONSE TO SEVERE ACCIDENTS

Four base accident sequences were analyzed for Peach Bottom using MAAP to determine plant response and temperature and pressure challenges to containment. These sequences, described below, in general are based on the sequences identified in Subtask 3.2.

Transient initiated sequences requiring a reactor shutdown and subsequent decay heat removal have been identified in WASH-1400 [4.1] as potential dominant contributors to the core melt frequency. These types of sequences may have a broad spectrum of possible outcomes due to the wide variety of possible system performance characteristics and operator actions. For the Task 23.1 MAAP evaluation one specific set of boundary conditions and assumptions has been postulated for each sequence.

The base sequences are:

1. TW - Transient followed by loss of containment heat removal.

2. TC - Transient followed by failure of the reactor to scram and standby liquid control (without operator action to reduce power level).

3. S_1E - Medium break loss of coolant accident with failure of emergency core cooling injection.

4. TQVW - Loss of offsite and onsite AC power.

The sequences analyzed in this section are low probability core damage events and include no, or minimal, recognition of operator actions that would significantly delay the progression toward core melt or mitigate consequences. 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

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operators regulate low pressure injection to maintain water level at the high level trip.

Consequently, the results presented here do not represent what would be expected to occur for the defined equipment failures and are extremely improbable. A more probable plant response to the specified equipment failures is evaluated in Section 5. This later section includes in the sequence definition some of the 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 plant parameters utilized to characterize Peach Bottom in these analyses are listed in Appendix A.

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 for the Browns Ferry Mark I design contained in the IDCOR Task 10.1 report [4.2], Containment Structure Capability of Light Water Nuclear Power Plants, which concludes that "it is felt that the Browns Ferry results are a sufficient representation of the containment capability" of Peach Bottom. The ultimate pressure capability was calculated to be 132 psia with the defined failure condition (twice the elastic strain) occurring at the "knuckle" between the cylindrical and spherical parts of the drywell. (It should be noted that a detailed assessment of penetration behavior under high strain conditions was not part of the analysis.) Given the similarity between the Peach Bottom and Browns Ferry designs,

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this value is assumed to represent Peach Bottom. In cases where high temperatures in containment were reached before ultimate pressure, the containment was assumed to fail when the containment atmosphere temperature reached 1200°F. The Peach Bottom containment is made from high strength carbon steel. The properties of this material will limit its ability to carry load at 1200°F. At this temperature the material strength is reduced to approximately 30-40% of its nominal value and will exhibit a significant creep rate under load [4.3]. Also, penetrations and/or penetration seals could have failed at these temperatures.

A containment break size of 0.1 ft^2 (0.2 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 the concept of yield leading to rupture resulting in diminishing yield as the containment depressurizes.

4.1 Plant Response to the TW Sequence

4.1.1 Sequence Description

This sequence is assumed to be initiated by MSIV closure isolating the reactor from the power conversion system. High pressure injection (HPCI and RCIC) are initially available until high suppression pool temperatures cause loss of these systems. Low pressure injection (LPCI and LPCS) are available as long as NPSH requirements are met*. Control rod drive (CRD) flow remains on until the available inventory of water in the condensate storage tank (CST) is depleted. No operator actions to either prolong injection or utilize alternate means of injection are assumed to occur.

4.1.2 Primary System and Containment Response

The timing of the key events for this sequence is summarized in Table 4.1. Plots of key parameters are presented in Figs. 4.1 through 4.5.

*In MAAP, the available NPSH was calculated but the requirement was set to zero. The resulting time dependent behavior was then reviewed to determine if cavitation had occurred and if it would have been sufficient to fail the pumps.

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Table 4.1

PEACH BOTTOM - TW

EVENT SUMMARY

Time	Event
0	Transient (MSIV closure)
4 sec	Reactor scrammed
4.5 min	HPCI, RCIC on
8.0 hr	High SP temperature failure assumed for HPCI (200°F)
10 hr	RCIC lost
14 hr	CRD flow ceases
15 hr	ADS on, LPCI and LPCS injecting
25 hr	ADS valves close
32 hr	Containment failure (overpressurization); LPCI and LPCS lost
34 hr	Top of core uncovered
37 hr	ADS valves open
39 hr	Start of core melt
40 hr	Vessel failure

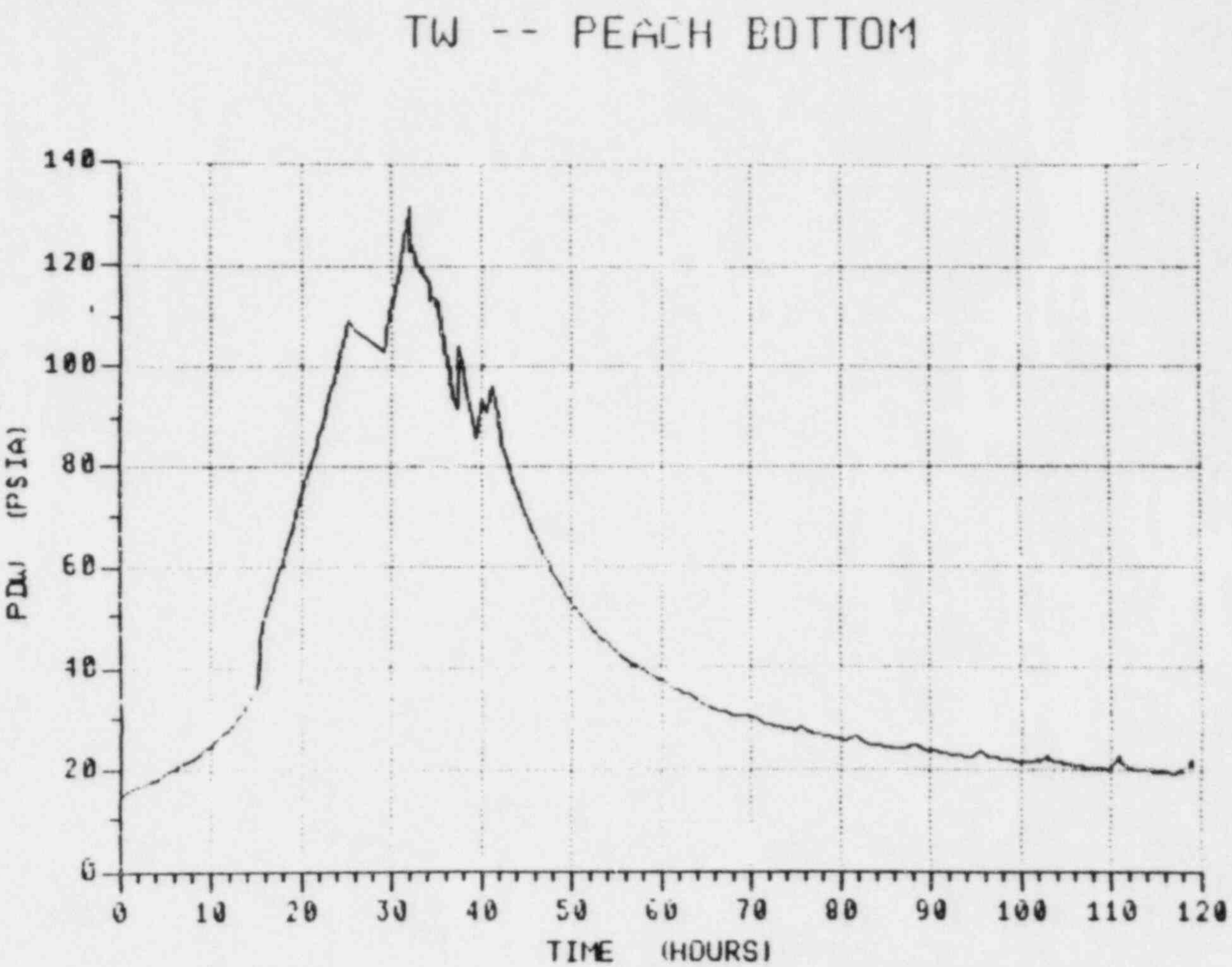

Fig. 4.1 Pressure in the drywell.

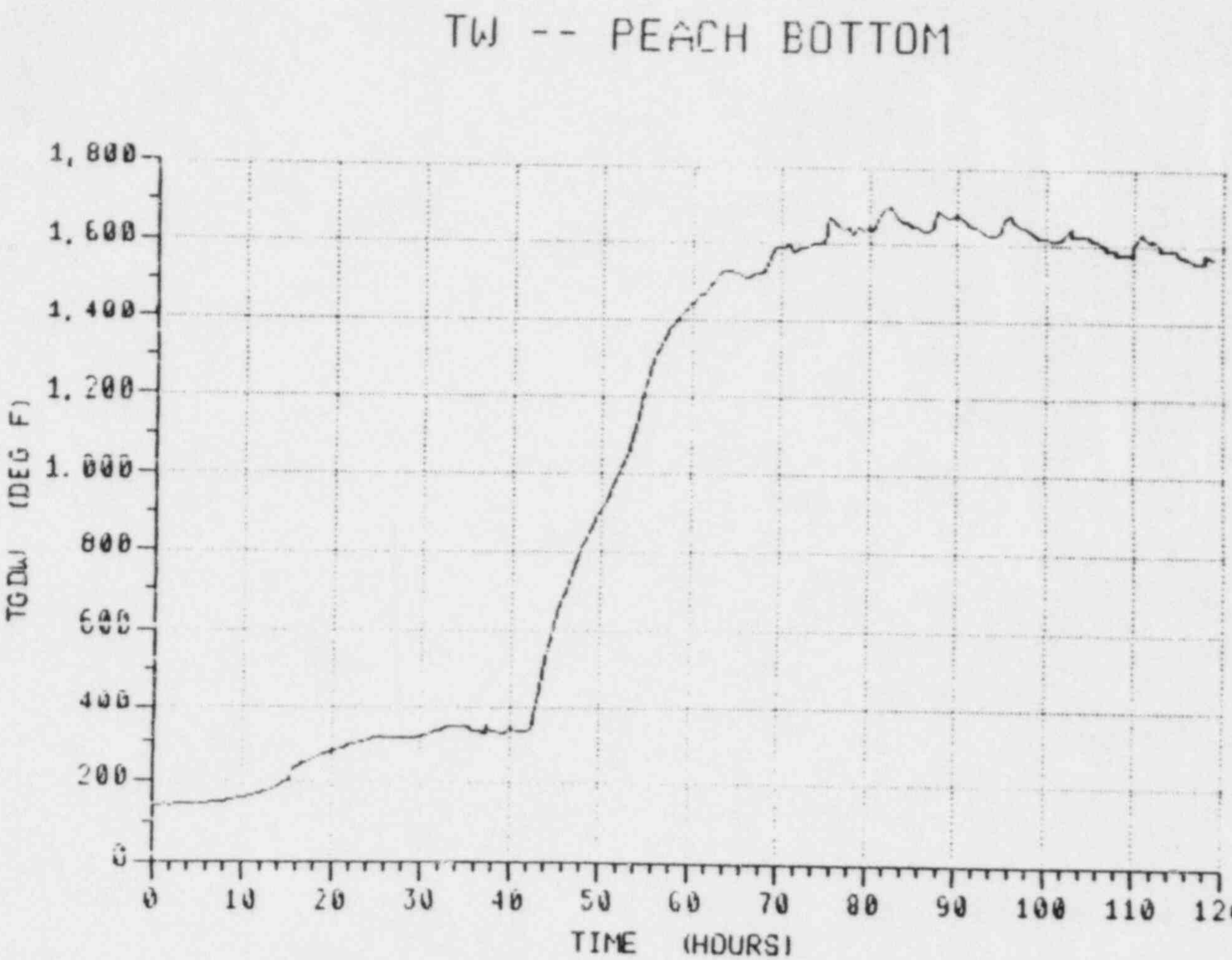

Fig. 4.2 Temperature of gas in the drywell.

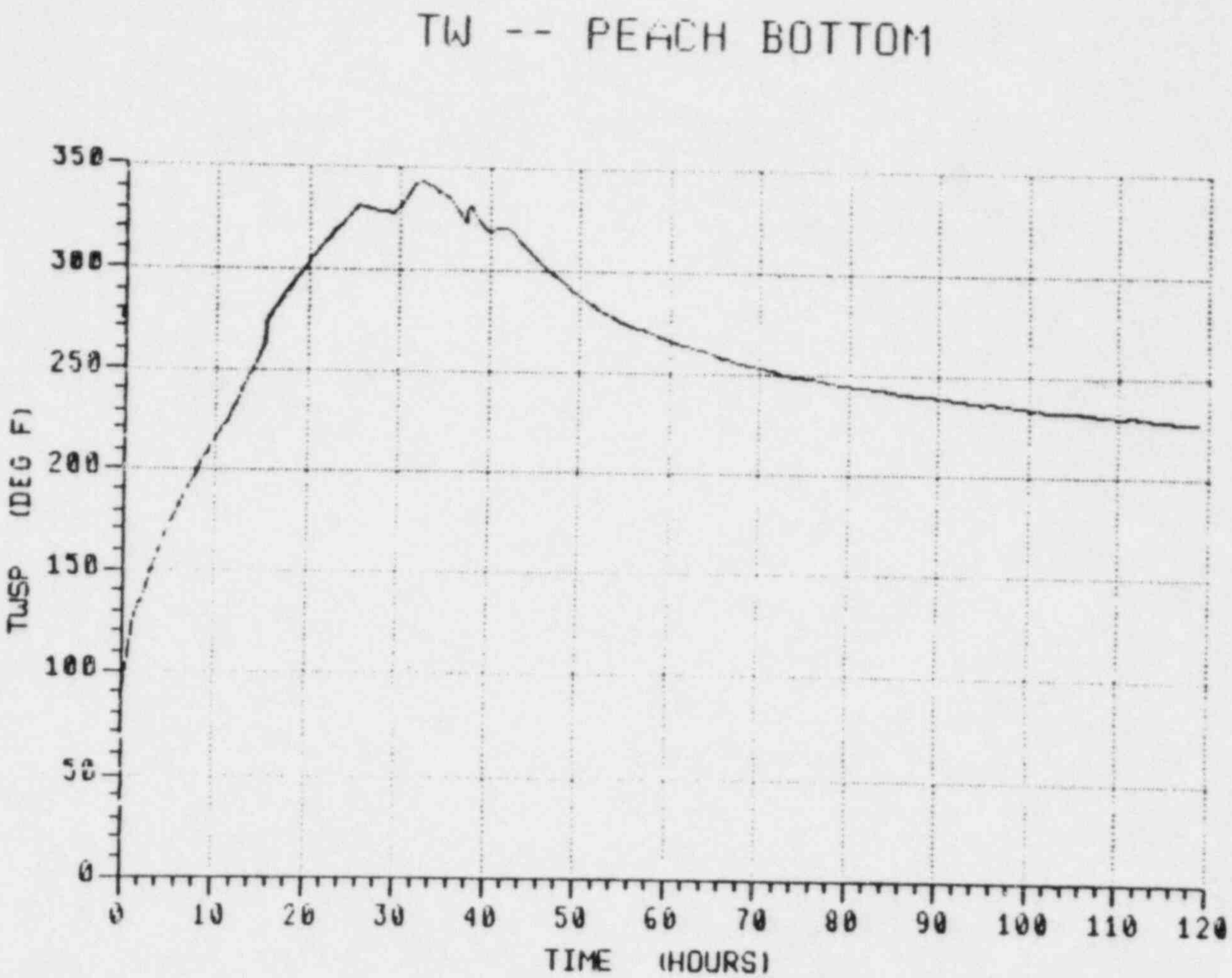

Fig. 4.3 Temperature of the suppression pool.

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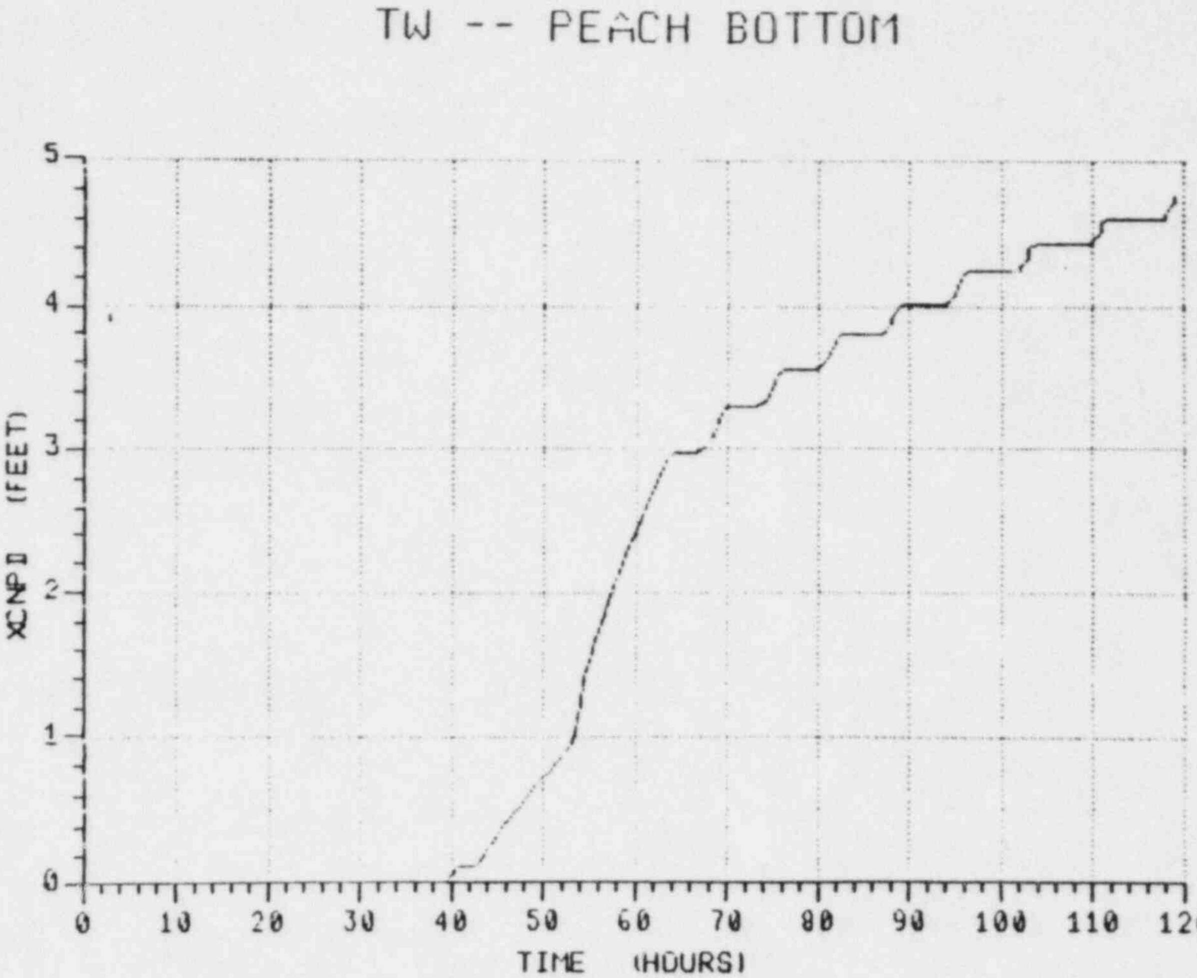

Fig. 4.4 Concrete ablation depth in the pedestal.

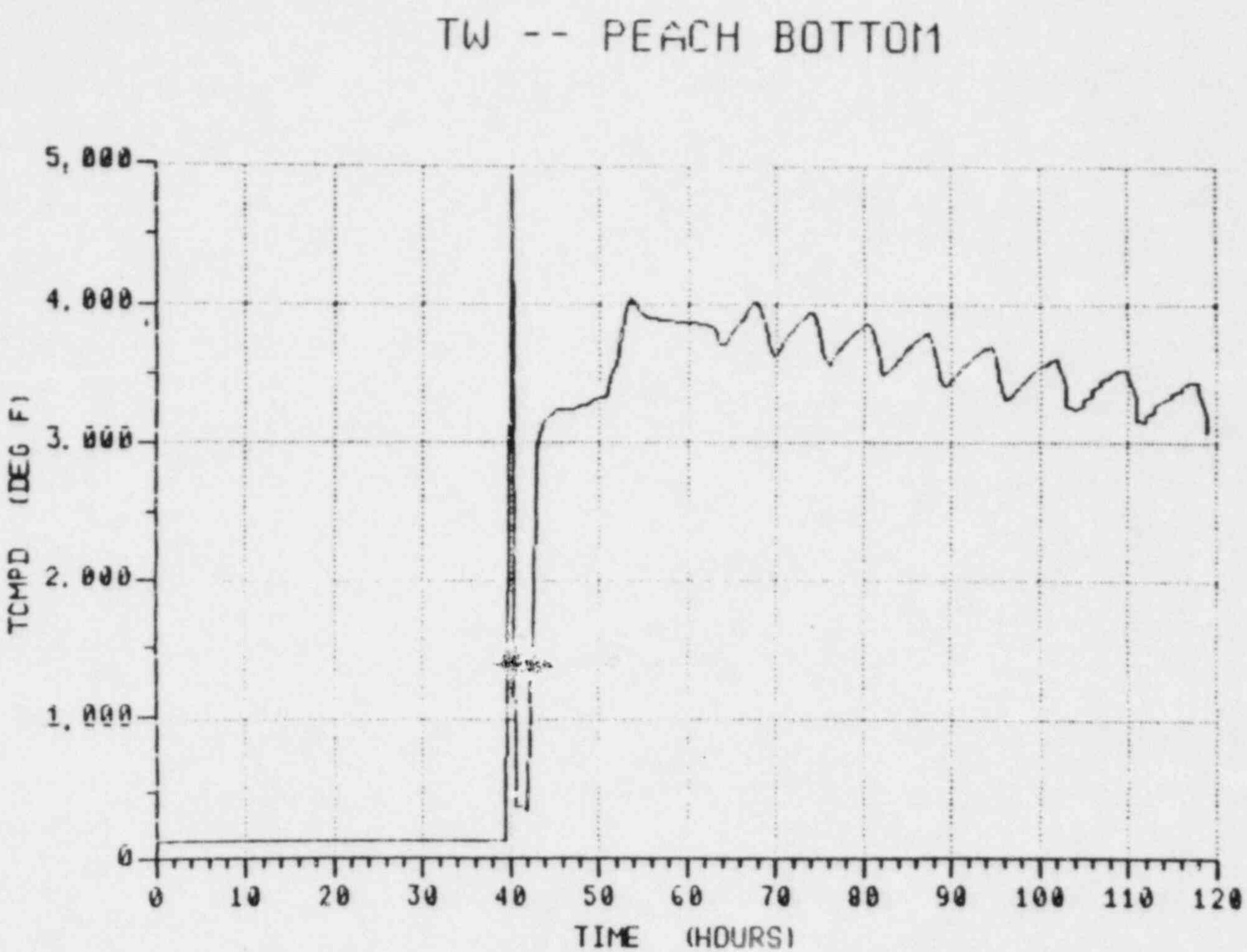

Fig. 4.5 Average corium temperature in the pedestal.

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This sequence is characterized by heatup of primary containment since adequate containment heat removal is unavailable. This results in containment failure due to overpressurization followed by core melting and vessel failure which occur after the coolant injection systems are lost. Operator actions to utilize alternate heat removal paths or means of injection which draw from sources other than the suppression pool were neglected.

The sequence is assumed to be initiated by main steam isolation valve (MSIV) closure on all four main steam lines isolating the reactor from the power conversion system. The initial reactor power level is assumed to be at 100%. This results in a reactor scram signal followed by successful reactor scram within four seconds. The reactor stored energy and decay power are transmitted to the suppression pool through the safety relief valve (SRV) lines. This results in a continuous heatup of the suppression pool because it is assumed that the pool cooling mode of the RHR system is unavailable. Reactor water level decreases due to boil-off which cannot be made up by the control rod drive (CRD) flow rate (111-177 gpm) due to the high reactor decay power at this time. High pressure injection systems (HPCI, RCIC) successfully come on in about 4.5 minutes and maintain required reactor water inventory.

HPCI suction is automatically transferred from the condensate storage tank (CST) to the suppression pool on pool high water level signal at 7.2 hours. RCIC suction remains from the CST until a low CST level signal is received at 10 hours when it is automatically transferred to the suppression pool. At 8 hours the suppression pool reaches 200°F resulting in the assumed loss of the HPCI pump due to bearing degradation. RCIC injection is assumed to be lost for the same reason when its suction is transferred to the suppression pool at 10 hours. At this time the post-scram CRD flow rate is sufficient to keep the core covered until the water source in the condensate storage tank is depleted at 14 hours assuming a CST inventory of 156,000 gallons. This inventory is conservative in light of the discussion below.

After CRD flow ceases, reactor water level boils down actuating the automatic depressurization system (ADS) at 15 hours. This permits low pressure systems (LPCI and LPCS) to maintain reactor water inventory.

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The CRD pumps are normally aligned to take suction from the main condenser hotwell via the reject line. Should this suction source be unavailable for whatever reason, the CRD pumps will be provided suction through its connection with the condensate storage tank with no operator action required. The volume of each CST reserved for ECCS use is 135,000 gallons. However, this volume is not restricted from CRD pump use. An average CST inventory is estimated to be about 156,000 gallons. Although there are no specific plant procedures or operating limits governing the alignment of these various tanks they are arranged such that they are easily cross-connected. For example, the two CSTs are frequently intertied such that their water levels "float" together. This mode of operation effectively doubles the condensate inventory available to the CRD pumps without operator actions. In addition, simple operator actions can be taken to interconnect the inventory of other various tanks increasing capacity to approximately 750,000 gallons.

As the containment pressurizes, the differential pressure between the ADS-SRV actuating gas and containment atmosphere decreases. When the drywell pressure reaches 110 psia, the differential pressure falls below the 5 psid required to hold the valves open, and the SRVs close. The operators are assumed to take no action to increase pneumatic supply pressure. This stops steam flow from the primary system to the suppression pool until the reactor vessel repressurizes and lifts the SRVs on high vessel pressure (approximately 1100 psia). Therefore, the containment pressurization is essentially halted during this period of reactor vessel repressurization. LPCI and LPCS injection ceases as the primary system pressure rises above their injection capability.

After the vessel is repressurized and the SRVs open and permit steam to flow to the suppression pool, the continuing heatup of the suppression pool results in pressurization of the containment until the assumed failure pressure of 132 psia is reached at 32 hours. Containment failure is assumed to fail the low pressure injection systems. This could potentially result from mechanical failures (piping movement and rupture) induced by containment failure, from electrical failures due to a steam environment in the reactor building, or possibly due to insufficient NPSH and large scale cavitation in the low pressure pumps. Knowledge of the actual failure mechanism is not

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required for this analysis but may be necessary for the assessment of core damage probabilities and public health risk.

As the containment depressurizes, the ADS-SRVs reopen when the actuating gas-drywell atmosphere differential pressure reaches 25 psid. This occurs when the drywell pressure decreases to 90 psia at 37 hours.

With the assumption of no further injection after loss of low pressure injection, the gradual boil down of reactor water inventory results in the top of the core being uncovered at about 34 hours. As the water level continues to boil down further uncovering the core, melting in the upper region of the core begins at about 39 hours. At approximately 40 hours when 20% of the core inventory is molten and has collected on the core plate, it is assumed to fail. This is equivalent to 153 fuel assemblies (764 assemblies in the core) achieving a molten state. Since the assemblies are supported from below by the CRD guide tubes, melting of the material and the subsequent flow into the bypass region will begin to load the core support plate which is only designed for transverse loads. Accumulation of this molten mass of UO_2 and the associated Zircaloy is assumed to fail this structure and allow the molten debris to flow into the lower plenum. The influence of this assumption on the overall effects is discussed in the uncertainty analysis report for Subtask 23.4.

The reactor pressure vessel fails within a few minutes after the core plate fails due to rapid melting of the instrument and CRD tubes that penetrate the bottom vessel head. Vessel failure and the subsequent generation of steam as the core debris mixes with water on the base mat results in a secondary pressure rise in containment from 90 psia to 96 psia. Heatup of the drywell occurs with temperatures reaching 1500°F at approximately 62 hours.

The slow boil down of reactor water results in a relatively long period during which the fuel cladding is at high temperature in a steam environment resulting in oxidation. During the period from 36 to 40 hours into the sequence, approximately 430 lbm of hydrogen are generated.

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This analysis demonstrates that plant operators have a significant amount of time (nearly two full days) to take action to prevent fuel melting and preserve containment integrity. A number of alternative means of arresting this sequence exist which were not included in this analysis but which are explicitly called for in the Peach Bottom plant specific procedures. Section 5 discusses some of these alternatives and their impact on the course of the accident. Appendix B includes several plots showing results for this sequence.

4.2 Plant Response to the TC Sequence (Without Operator Action to Reduce Power Level)

4.2.1 Sequence Description

This sequence is assumed to be initiated by MSIV closure followed by failure of the reactor protection system to scram the reactor as well as failure to initiate standby liquid control (SLC). Successful recirculation pump trip occurs and both high pressure (HPCI and RCIC) and low pressure (LPCI and LPCS) injection systems are available until either high suppression pool temperatures or insufficient NPSH cause loss of these systems*. CRD flow remains on until the inventory in the CST is depleted. The condensate pump is assumed to be unavailable. No operator actions are assumed to either control power level by reducing reactor water level or to utilize alternate means of injection. The operator is assumed to follow the written procedures for venting the wetwell to protect against containment overpressurization. For this sequence, the steam release would exceed the capacity of the SGTS system and would be discharged to the reactor building. Temperatures within the building would exceed that required to fail the fusible links on the fire dampers of the SGTS, thereby isolating the system. In this sequence, it is assumed that pumping capacity is lost after wetwell venting is initiated. The low pressure ECCS pumps have been tested in a steam environment and have performed satisfactorily. Therefore, it should be noted this is a conservative assumption regarding pump performance and that the core would not be damaged if the pumps continued to operate.

*NPSH was treated in a similar manner as discussed for the TW sequence.

4.2.2 Primary System and Containment Response

The timing of the key events for this sequence is summarized in Table 4.2. Plots of key parameters are presented in Figs. 4.6 through 4.12. This sequence is characterized by rapid heatup of the suppression pool resulting in loss of the high pressure injection systems, wetwell venting and loss of low pressure injection systems. This is followed by core melting and vessel failure.

As in TW, this sequence is assumed to be initiated by MSIV closure isolating the reactor from the power conversion system. However, it is assumed that the reactor fails to scram and that subsequent initiation of standby liquid control is not attempted or is unsuccessful. It is also assumed that condensate flow is unavailable. Successful recirculation pump trip followed by initiation of HPCI and RCIC at approximately 1-1/2 minutes results in an estimated reactor power level of 18 percent of normal. This is based on the power level required to boil off reactor water at a rate equal to the total injection rate at this time. If the power level were greater than this, the reactor water level would boil down resulting in a power level reduction until this balance was achieved. This power level was confirmed to be in the correct range with a neutronics model, RETRAN, and assumes no operator action is taken to throttle injection to reduce power level.

Steam flow through the SRV lines results in suppression pool heatup. At approximately 26 minutes, the suppression pool reaches 200°F resulting in the assumed degradation of the HPCI pump bearings causing loss of this system.

RCIC injection is insufficient to maintain water level while the core is at 18% power. Therefore, the level boils down resulting in low pressure system (LPCI, LPCS) initiating signals at 36 minutes, ADS actuation at 38 minutes and effective low pressure injection at 40 minutes. Before low pressure injection can be established, reactor water level drops to the top of the core, but no fuel overheating occurs. Reactor power level is linearly reduced to six percent of normal [4.4] as the water level decreases from a level 20 ft above the core inlet down to the top of the core which is reached at 37 minutes. After this time the power level continues to be a function of

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Table 4.2

PEACH BOTTOM - TC

EVENT SUMMARY

Time	Event
0	Transient (MSIV closure)
3 sec	Failure to scram
1.7 min	HPCI, RCIC on
26.4 min	HPCI assumed lost (SP at 200°F)
38 min	ADS on
40 min	LPCI, LPCS on (reduced flow)
54 min	RCIC lost
1.3 hr	ADS valves close
1.3 hr	Top of core uncovered
1.3 hr	Open wetwell vent
1.3 hr	LPCI, LPCS assumed lost
1.5 hr	ADS valves reopen
3 hr	Start of core melting
3.9 hr	Vessel failure
6.9 hr	CRD flow ceases

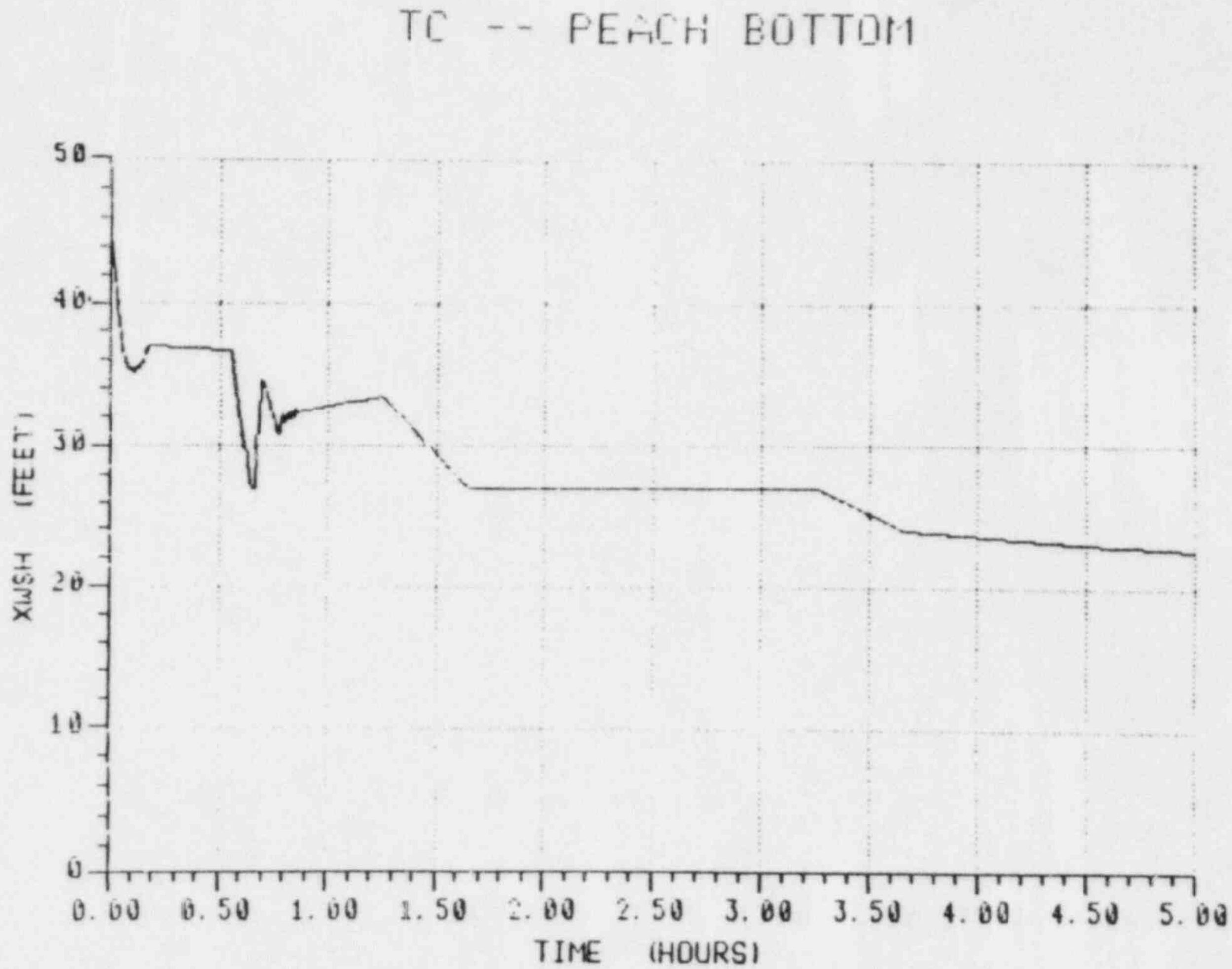

Fig. 4.6 Reactor pressure vessel water level.

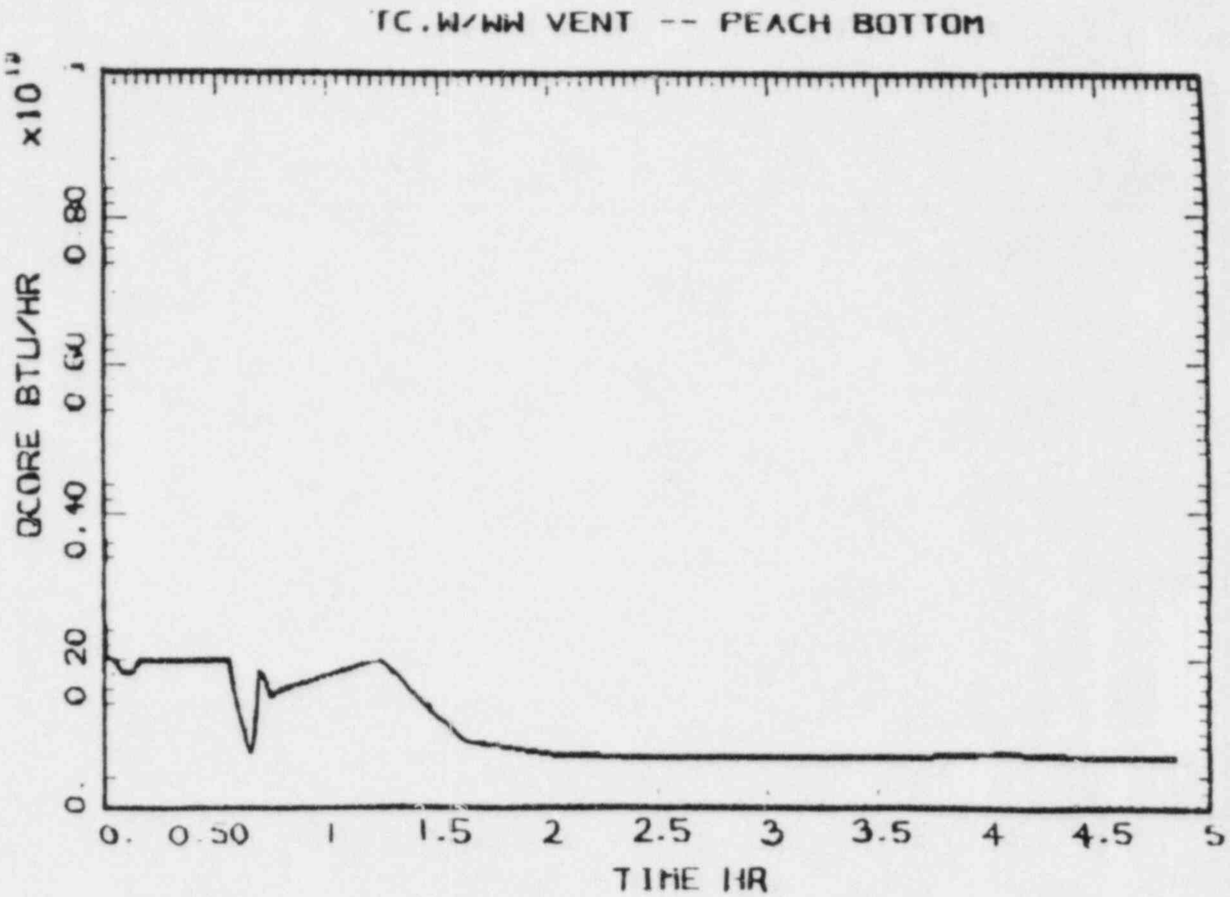

Fig. 4.7 Average core power.

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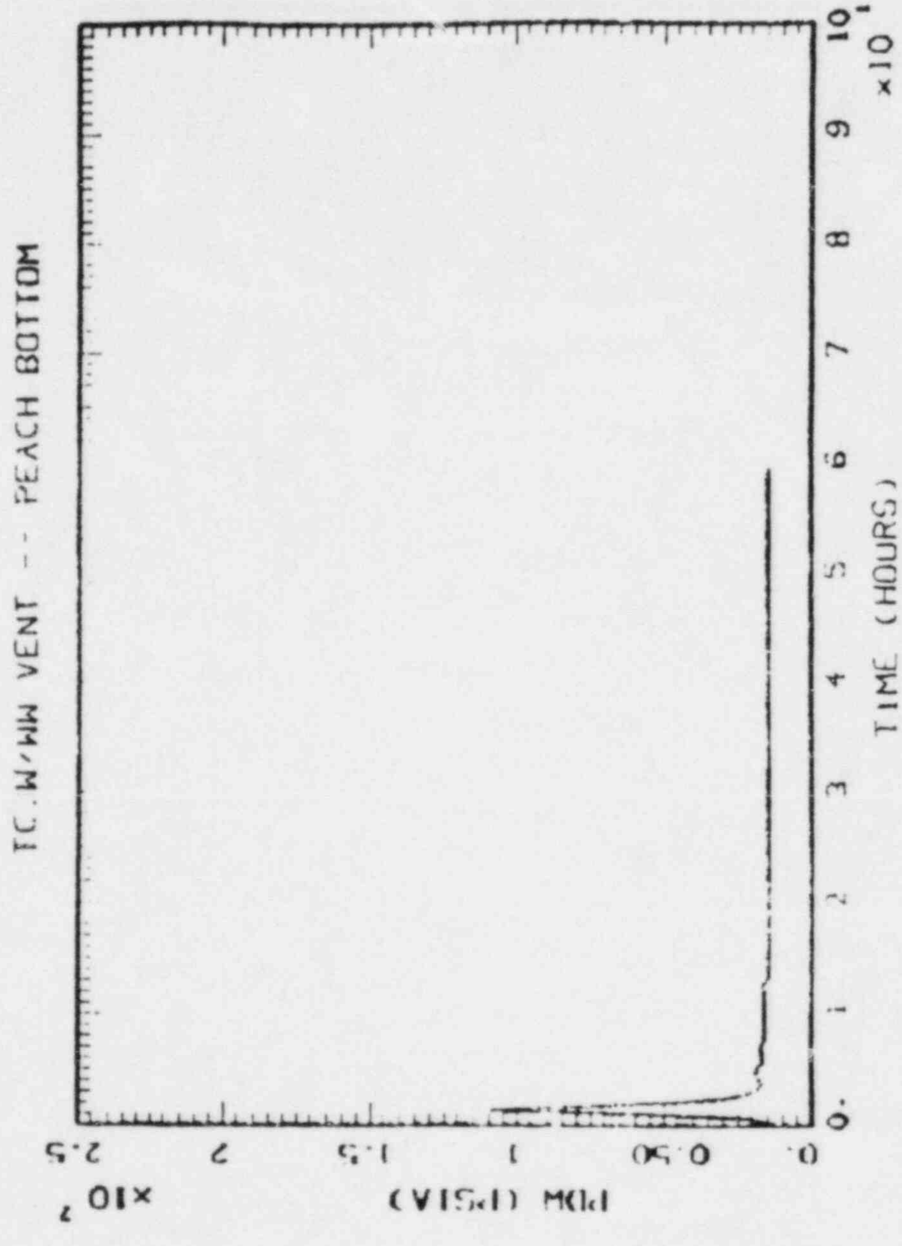

Fig. 4.8 Pressure in the drywell.

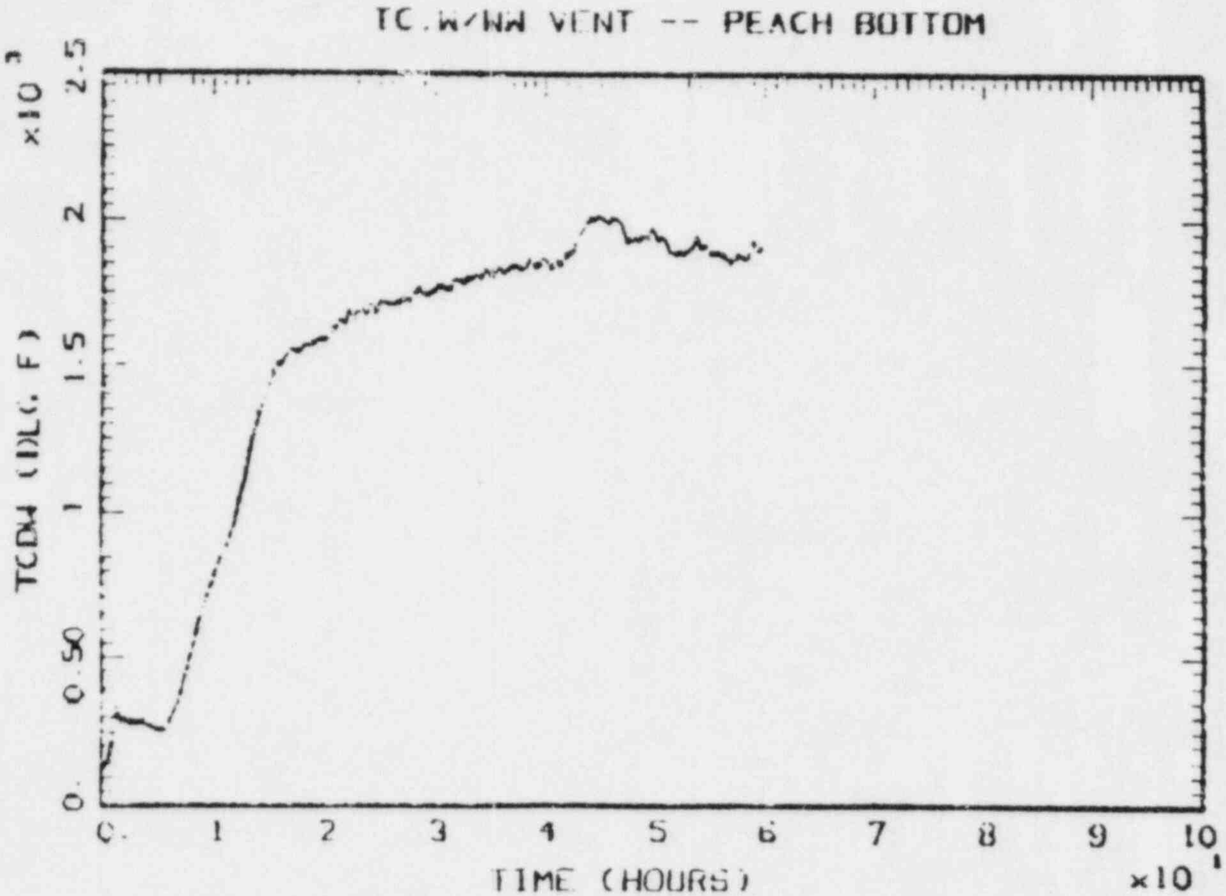

Fig. 4.9 Temperature of gas in the drywell.

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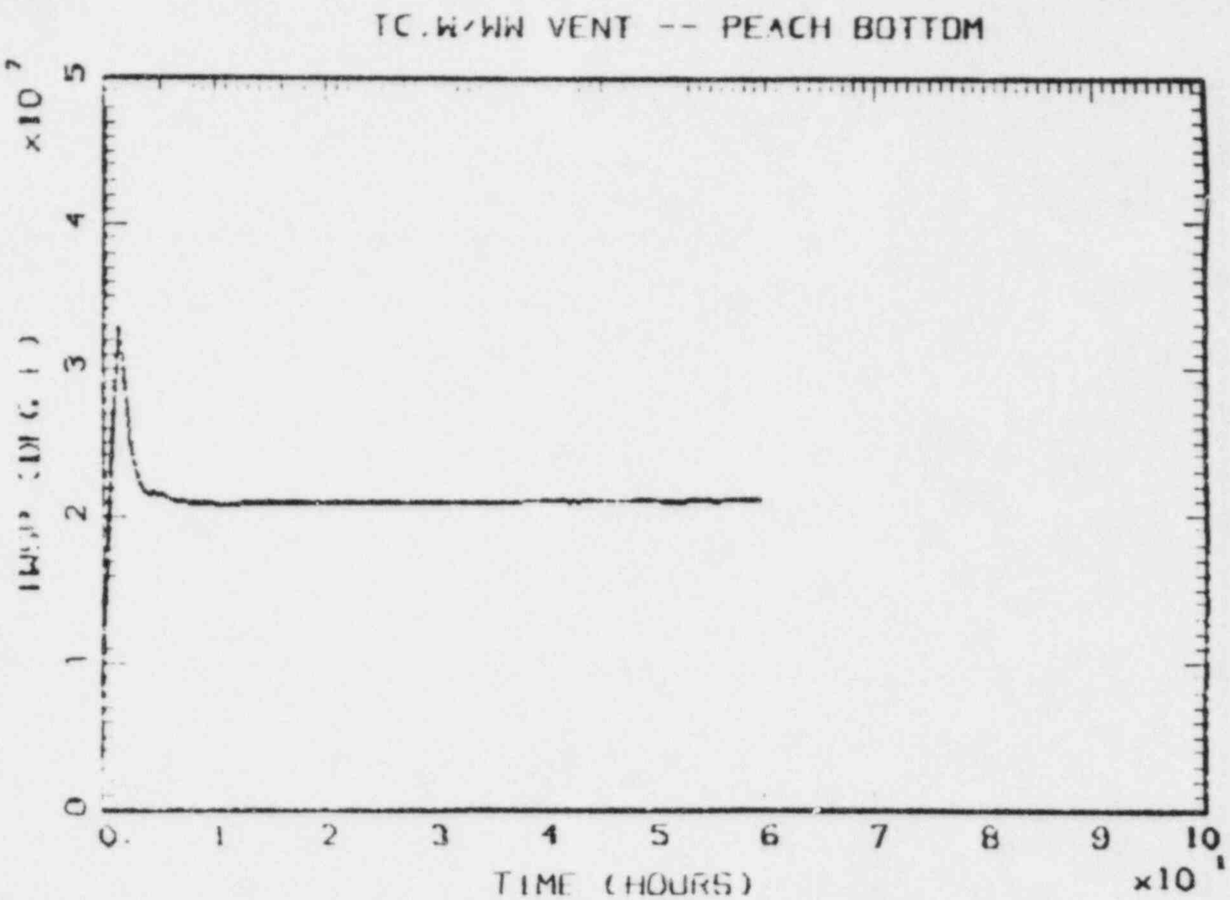

Fig. 4.10 Temperature of the suppression pool.

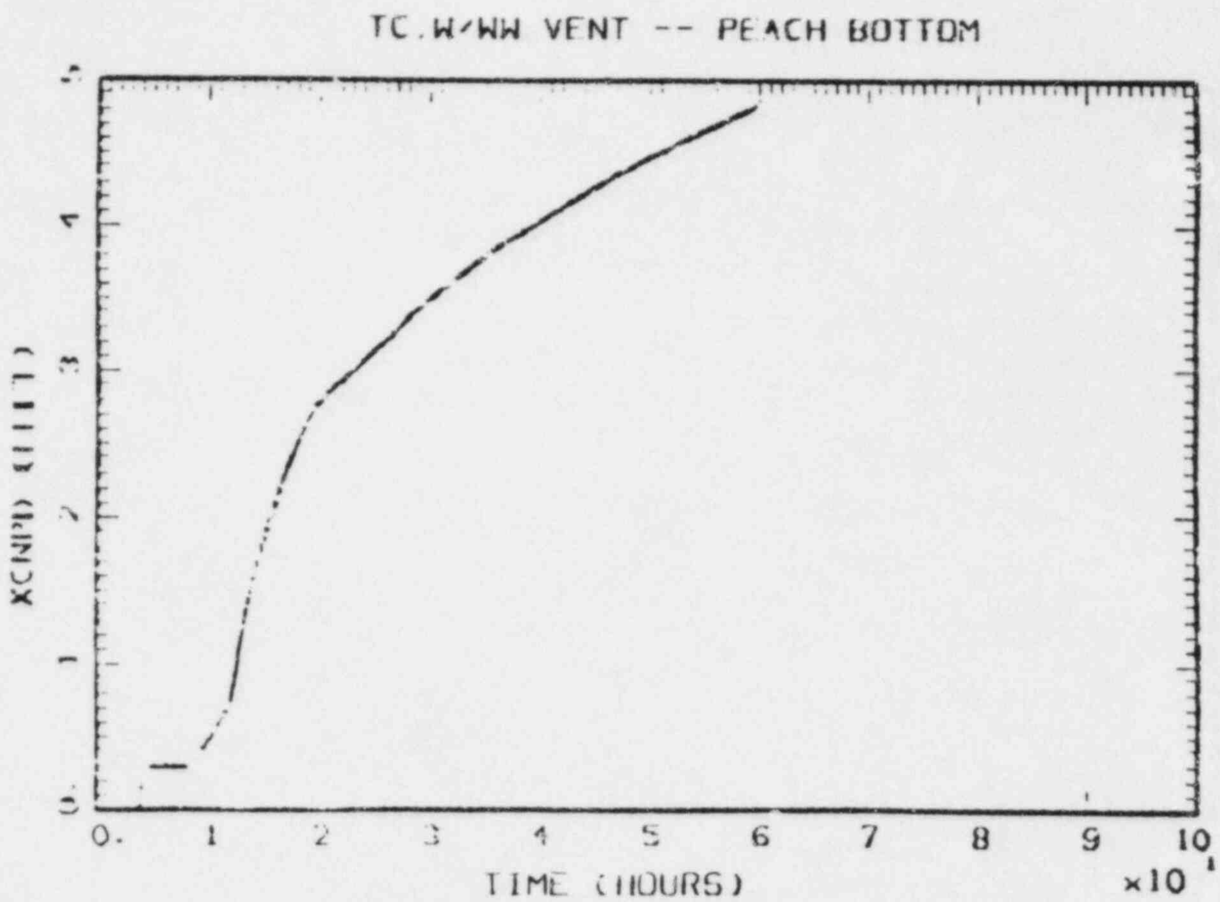

Fig. 4.11 Concrete ablation depth in the pedestal.

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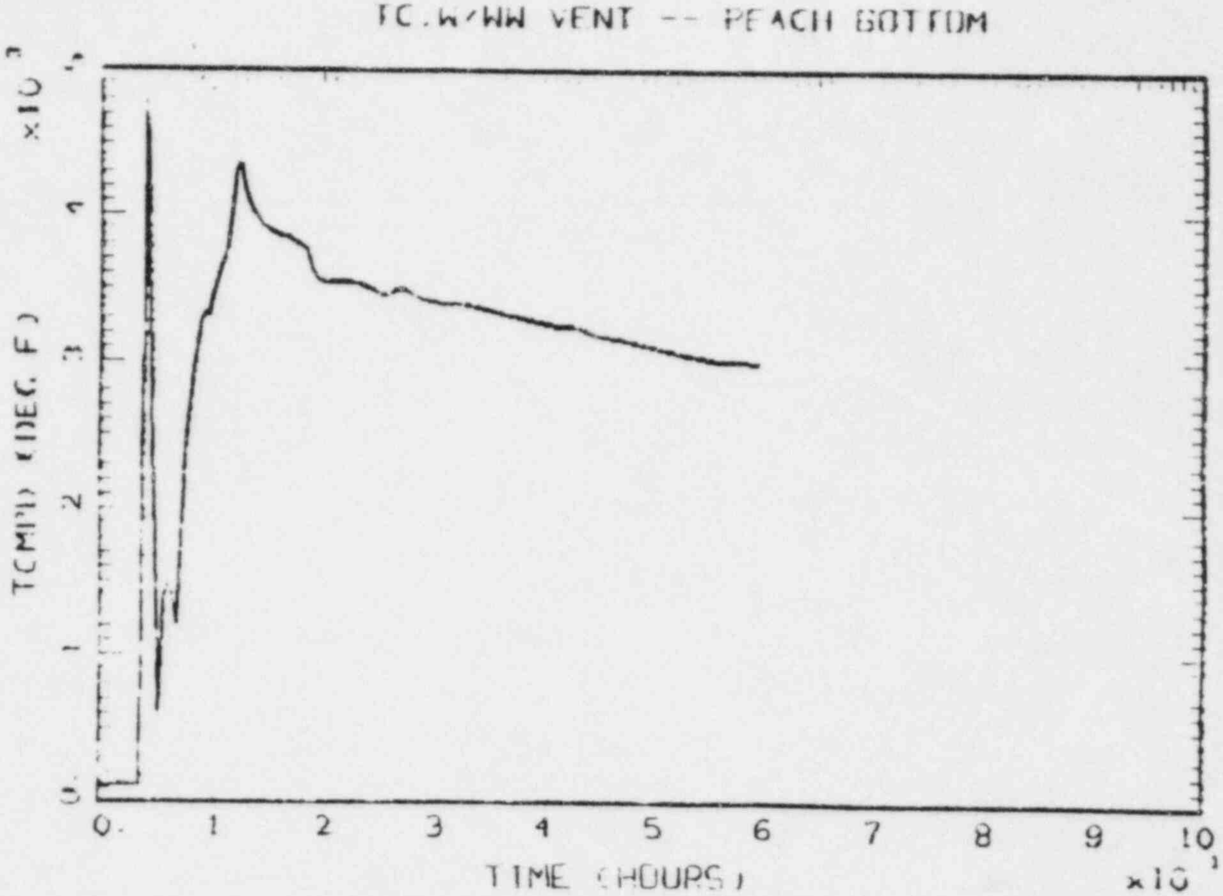

Fig. 4.12 Average corium temperature in the pedestal.

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reactor water level and is balanced by the primary system pressure, the resulting injection rates of LPCI and LPCS and the relief capacity through the ADS valves. This results in the water level hovering near the top of the core.

RCIC suction remains from the CST until 54 minutes when it is automatically transferred to the suppression pool and assumed lost due to bearing degradation. However, low pressure injection is sufficient to maintain reactor water level near the top of the core.

As the containment pressure rises due to suppression pool heatup, the SRVs previously actuated by ADS close when the drywell pressure reaches 110 psia for the reasons discussed under the TW sequence. This occurs at 1.3 hours, causing rapid repressurization of the reactor vessel and loss of injection by the low pressure systems (LPCI, LPCS). The rapid vessel repressurization and lifting of the SRVs on high reactor pressure result in continued containment pressurization until the wetwell is vented through a vent area of 0.18 m^2 (1.98 ft^2) at a pressure of 115 psia which is reached at 1.8 hours. The vent size corresponds to the opening of a 2 in., 6 in. and 18 in. vent lines from the wetwell. This depressurization and steam flow is assumed to cause loss of the low pressure injection systems as a result of the same possible mechanisms discussed for the TW sequence. As the containment depressurizes, the ADS valves reopen when the pressure decreases to 90 psia (1.5 hours), as discussed for the TW sequence.

With only CRD flow remaining, reactor water level boils down resulting in the start of core melting at about 3 hours. When 20% of the core has melted core plate failure is assumed resulting in vessel failure at 3.9 hours. Approximately 300 lbm of hydrogen are generated from cladding oxidation.

Following reactor vessel failure, the core debris disperses over the pedestal and drywell floors. Drywell heatup begins at about 6 hours reaching a temperature of 1500°F at approximately 13 hours. This occurs as a result of radiative and convective heat transfer from the core debris which lies on the pedestal and drywell floors in a relatively thin flat geometry with large surface area. Appendix B includes other plots for this sequence.

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The influence of natural circulation within the primary system is illustrated by Figs. B.15 and B.16 which show the structure temperatures and decay power associated with the various primary system nodes. As illustrated by these figures, the volatile fission products are initially released into the upper plenum and deposited there as a result of both vapor condensation and gravitational sedimentation. Over an extended time interval, the upper plenum structure temperature increases and circulation is set up between the upper plenum, the downcomer and the core. As the temperature continues to increase, material is transported from the upper plenum into the downcomer, where it is again retained and remains in this locale until all of the water is vaporized in the downcomer. Following vaporization of the water, the material deposited in the downcomer heats the structural mass, vaporizes and is transported throughout the primary system and eventually into the containment. It is this release into the containment that is eventually transported into the reactor building with a small fraction being released to the environment.

The analysis of this sequence, which assumes no early operator action to reduce power level, indicates that operators have approximately 1/2 hour after the core is uncovered to recover the core and prevent fuel melting. The analysis also indicates that if fuel melting and vessel failure did occur, operators would have approximately 22 additional hours to prevent containment heatup above 1500°F and mitigate those releases resulting from revaporization in the drywell through the use of such systems as HPSW for flooding or containment sprays, condensate pumps, and CRD flow.

4.3 Plant Response to the S_1E Sequence

4.3.1 Sequence Description

This sequence is assumed to be initiated by a 0.1 ft^2 break in the main steam line inside containment (drywell). High pressure injection (HPCI and RCIC) and low pressure injection (LPCI and LPCS) are assumed to be unavailable. Injection from the condensate pump is also assumed to be unavailable, but CRD flow is available until the inventory in the CST is depleted. It is assumed that suppression pool cooling is manually initiated at 10

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minutes into the sequence. No actions by the operator to establish alternate means of injection to the core are assumed.

4.3.2 Primary System and Containment Response

The timing of the key events for this sequence is summarized in Table 4.3. Plots of key parameters are presented in Figs. 4.13 through 4.17. In general this sequence is characterized by loss of makeup to the core resulting in fuel melting and vessel failure. However, suppression pool cooling is available preventing the containment from overpressurizing on steam. Containment failure occurs due to an overtemperature condition in the drywell before sufficient noncondensable gas generation has occurred to overpressurize the containment.

This sequence is initiated by a 0.1 ft^2 break in the primary system at the elevation of the main steam lines. This causes rapid containment pressurization to above the 2 psig set point for reactor scram, resulting in a successful scram within 7 seconds of initiation of the break. As the primary system is rapidly depressurized through the break, a low reactor pressure signal for MSIV closure is received, and closure occurs by 84 seconds isolating the reactor from the power conversion system and shutting down feedwater. It is assumed that the condensate pumps fail to inject through the feedwater pumps. As reactor water level boils down the high pressure (HPCI and RCIC) and low pressure (LPCI and LPCS) systems are assumed to fail to inject water into the core. Post-scram CRD flow (177 gpm) is sufficient to keep the core covered until automatic depressurization (ADS) is actuated at 1.1 hours due to high drywell pressure and low reactor water level, further depressurizing the primary system. There is a low pressure pump permissive signal for ADS because the residual heat removal (RHR) systems in suppression pool cooling mode were manually initiated at 10 minutes. This depressurization causes reactor water level to drop to near the bottom of the core at 1.5 hours. As reactor water level boils down, the fuel overheats and oxidation of cladding occurs. The core is heated up leading to fuel melting at 2.5 hours and vessel failure at 3.4 hours. Approximately 240 lbm of hydrogen are generated between 1.5 and 2 hours.

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Table 4.3

PEACH BOTTOM - S_1E

EVENT SUMMARY

Time	Event
0	Break in steam line (0.1 ft^2)
6.8 sec	Reactor scrammed
84 sec	MSIVs closed, feedwater tripped
10 min	Suppression pool cooling on
1.05 hr	Automatic depressurization on (ADS)
1.13 hr	Top of core uncovered
2.5 hr	Start of core melt
3.4 hr	Vessel failure
15 hr	CRD flow ceases
23 hr	Containment failure (overtemperature)

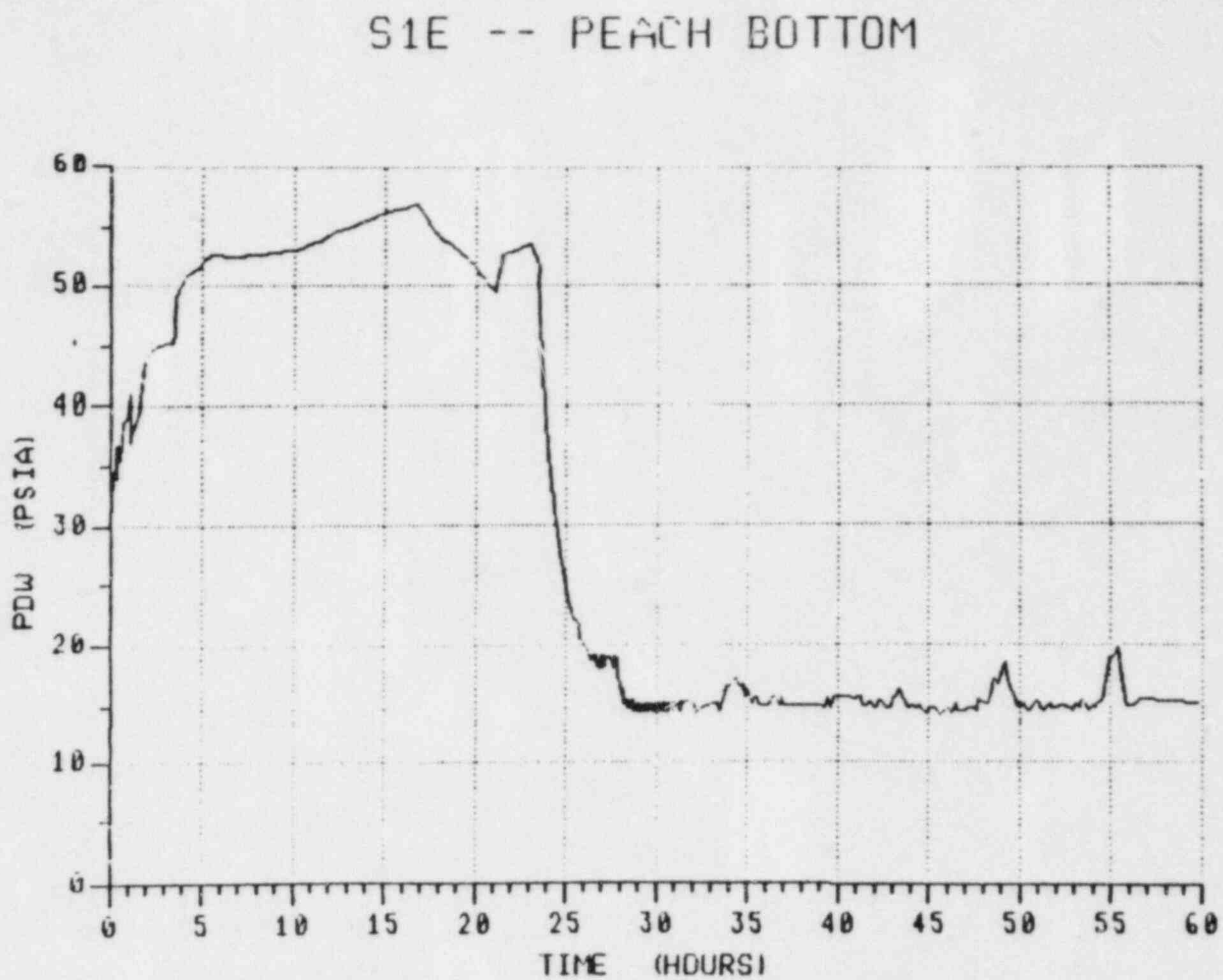

Fig. 4.13 Pressure in the drywell.

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S1E -- PEACH BOTTOM

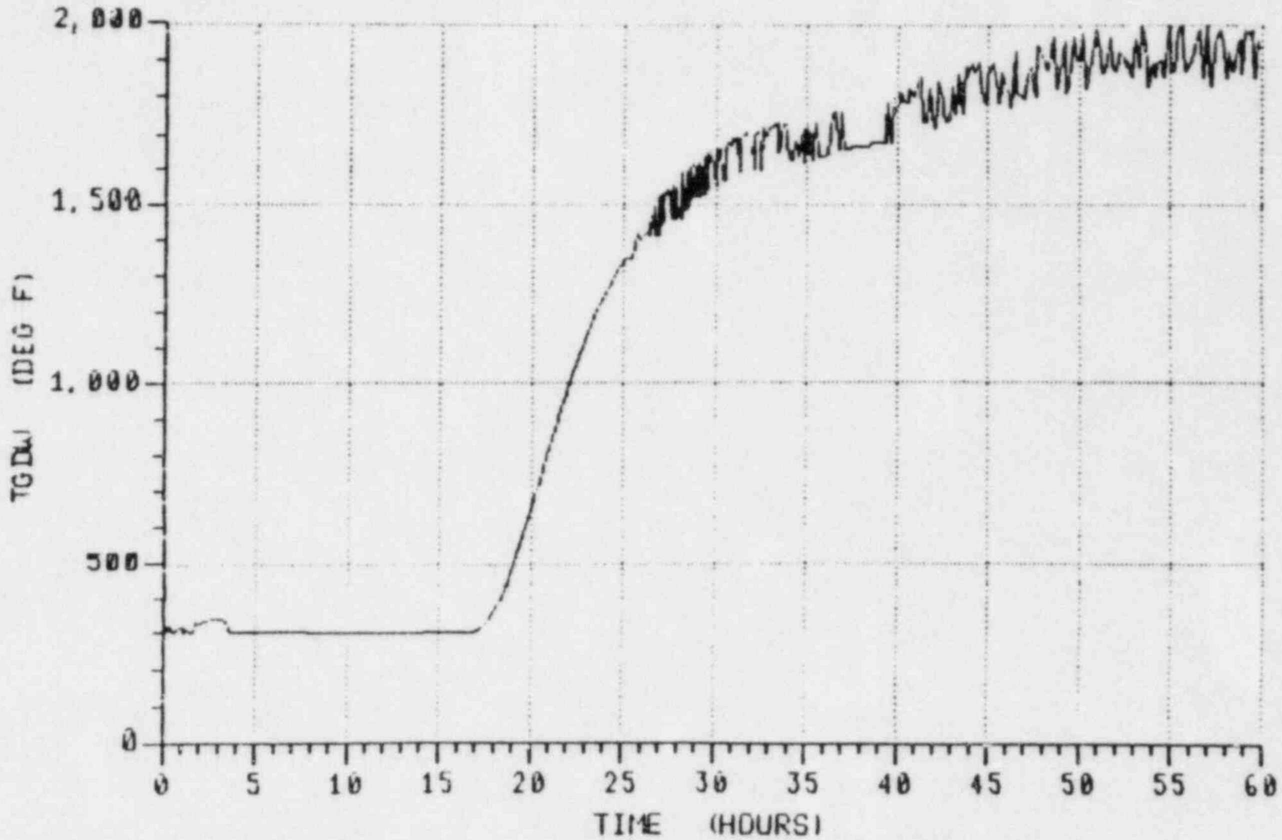

Fig. 4.14 Temperature of gas in the drywell.

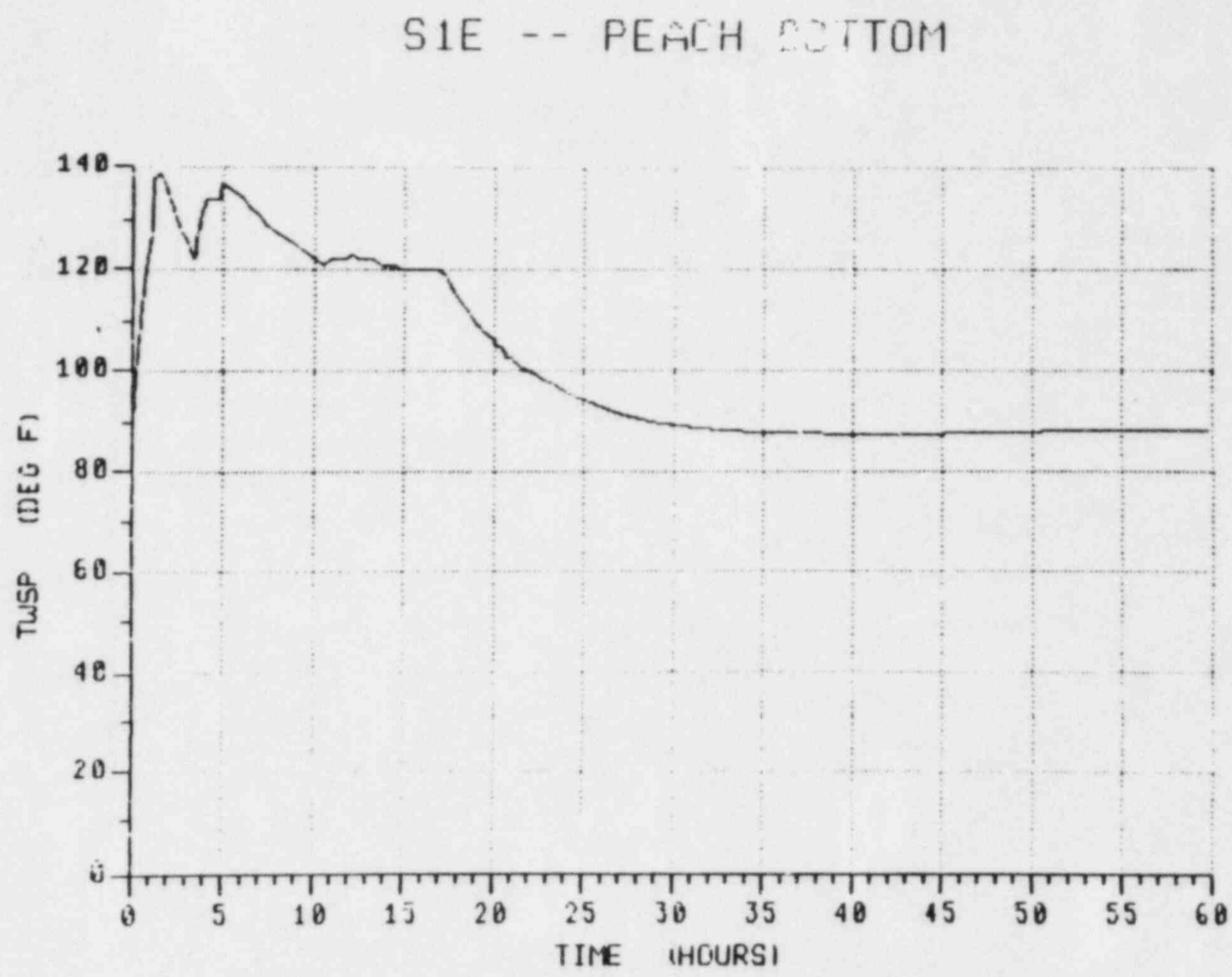

Fig. 4.15 Temperature of the suppression pool.

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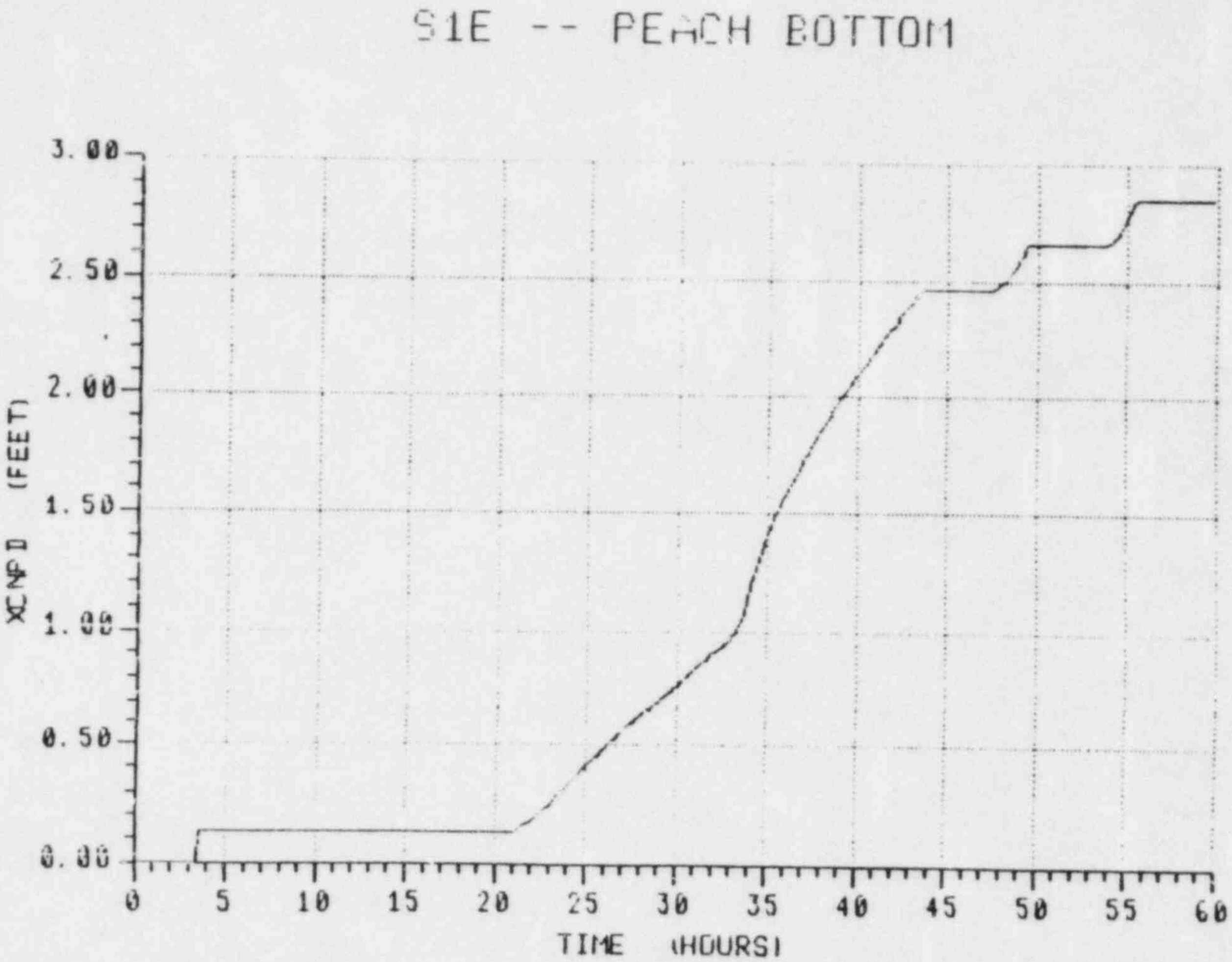

Fig. 4.16 Concrete ablation depth in the pedestal.

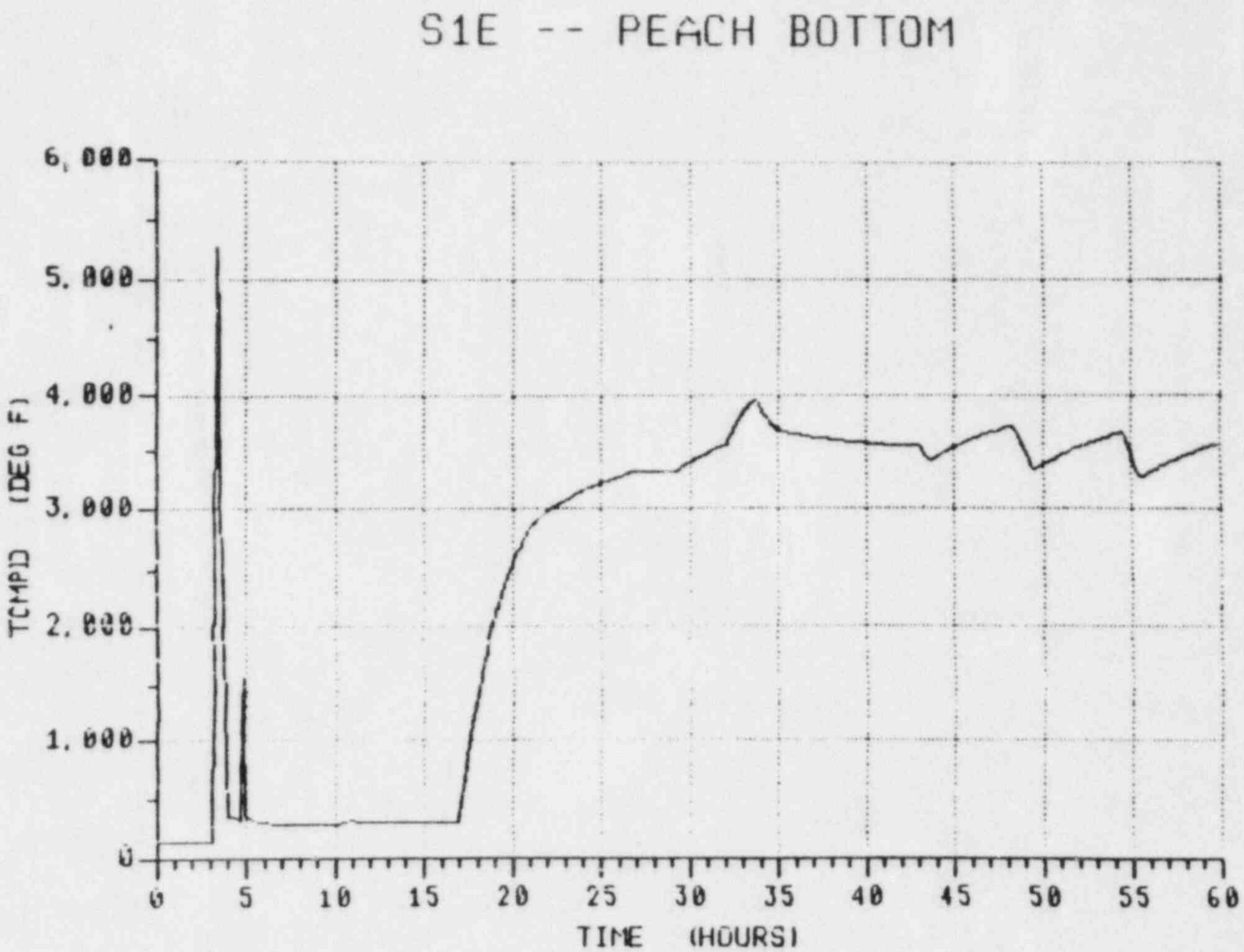

Fig. 4.17 Average corium temperature in the pedestal.

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The pressure in the drywell rises initially to about 40 psia as a result of the steam line break. The steam is quenched in the suppression pool and the pressure is about 45 psia at the time of vessel failure. Noncondensable gas generation from cladding oxidation and initial ablation of concrete in the pedestal and drywell results in a further pressure increase to about 55 psia. Heatup of the drywell atmosphere and structure from radiation and convective heat transfer from the core debris commences, after the loss of CRD flow, at about 17 hours.

When the temperature reaches 1200°F at 23 hours the containment is assumed to fail with a 0.1 ft^2 break. The containment pressure is approximately 55 psia when the failure occurs, dropping to atmospheric pressure at about 28 hours. Appendix B includes several plots for this sequence.

The analysis of this sequence demonstrates that operators have approximately 2 hours to establish alternate injection to the reactor prior to fuel melting. If vessel failure did occur, operators would have an additional 20 hours to take action to preserve containment integrity and mitigate releases.

4.4 Plant Response to the TQVW Sequence

4.4.1 Sequence Description

This sequence is assumed to be initiated by loss of all off-site and on-site AC power (station blackout). This results in reactor scram and loss of the power conversion system. It is assumed that DC power is available for a period of 6 hours permitting control of the steam driven HPCI and RCIC systems to maintain injection to the core for this duration. After 6 hours, no further injection is assumed available and no operator action to utilize alternate sources are assumed to occur.

4.4.2 Primary System and Containment Response

The key events for this sequence are summarized in Table 4.4. Plots of key parameters are presented in Figs. 4.18 through 4.22. In general, this

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Table 4.4

PEACH BOTTOM - TQVW

EVENT SUMMARY

Time	Event
0	Loss of off-site and on-site AC power
4 sec	Reactor scrammed
5 min	High pressure injection on (HPCI, RCIC)
6 hr	HPCI, RCIC off (loss of DC power)
8.4 hr	Top of core uncovered
11.4 hr	Start of core melt
12.4 hr	Vessel failure
18 hr	Containment failure (overtemperature)

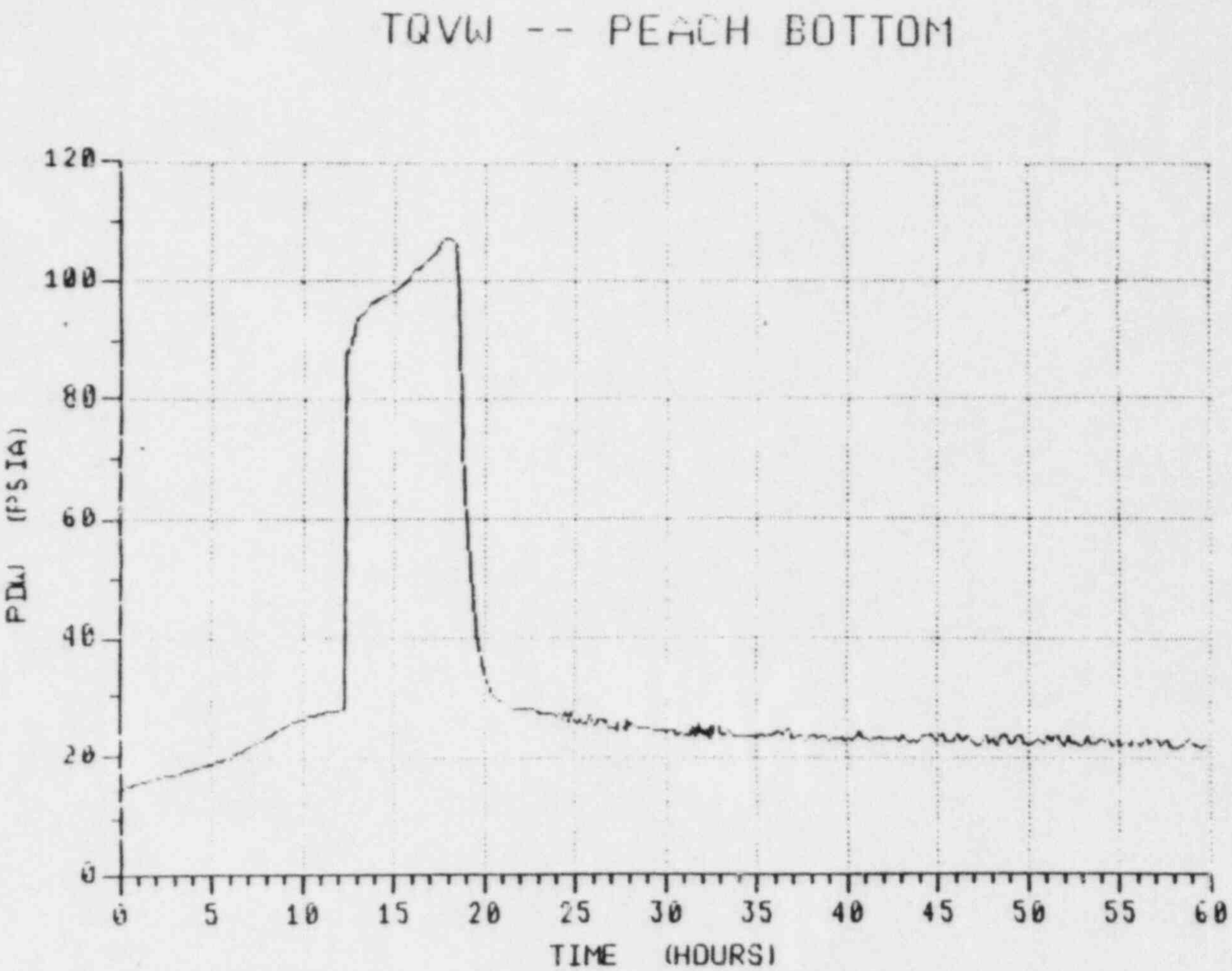

Fig. 4.18 Pressure in the drywell.

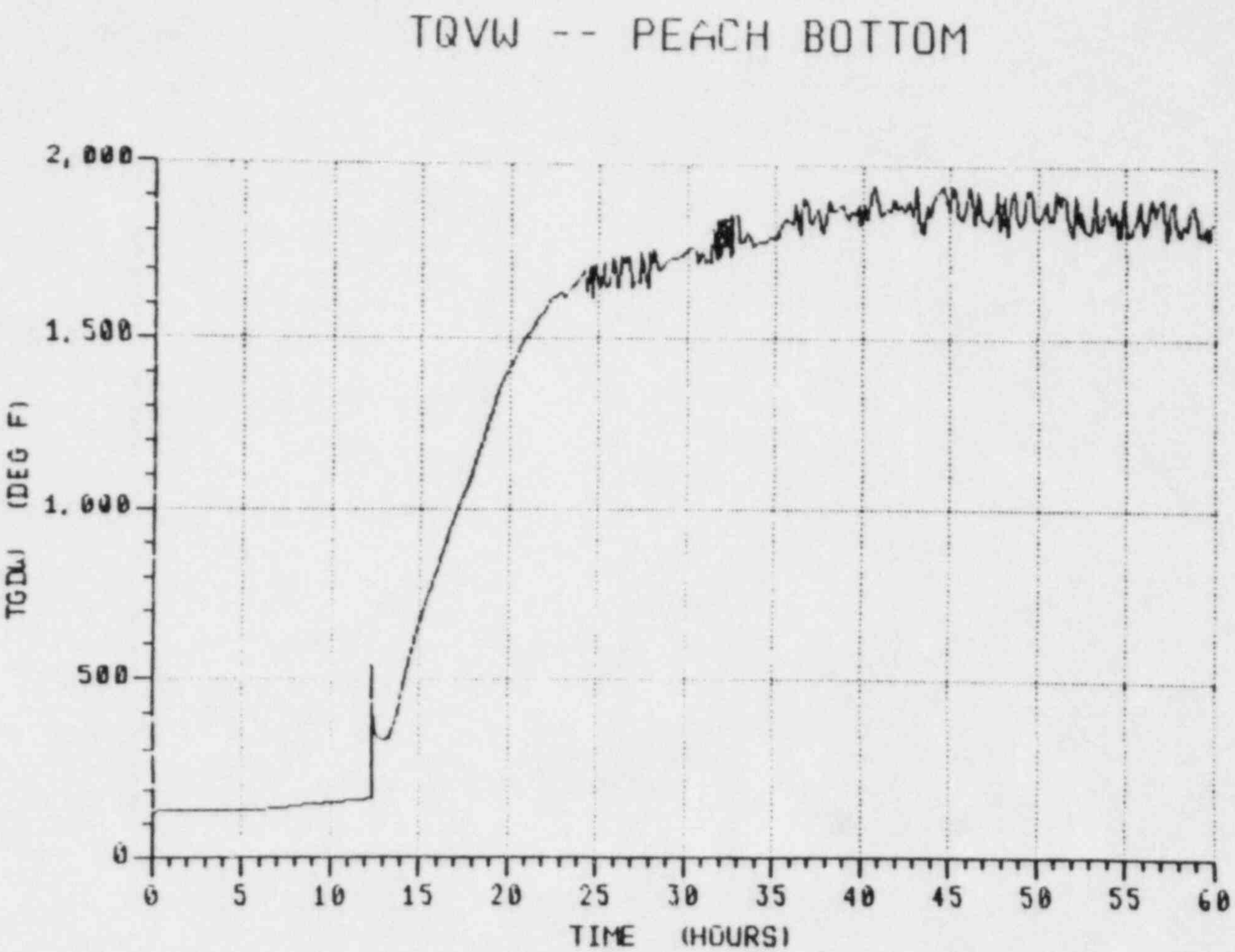

Fig. 4.19 Temperature of gas in the drywell.

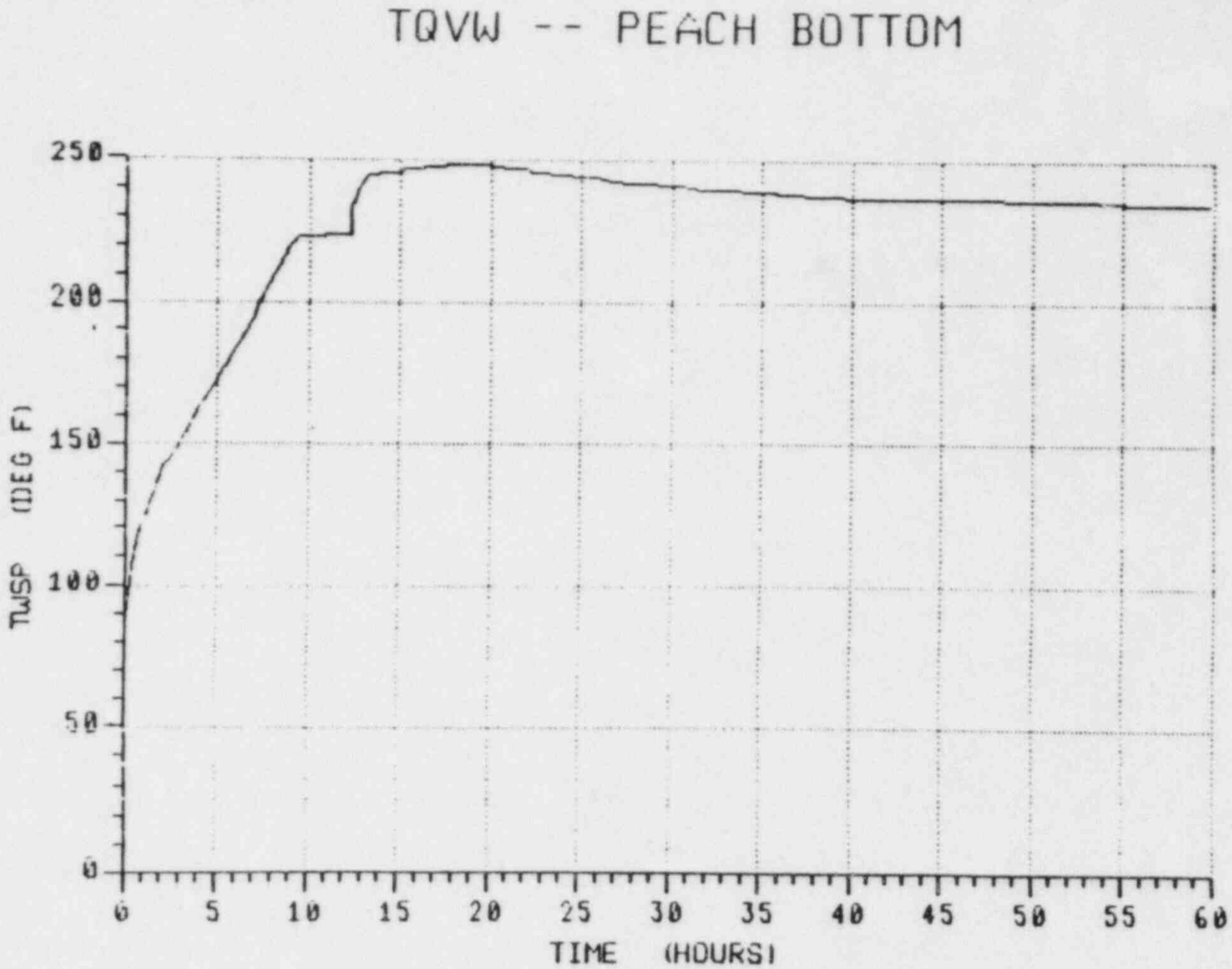

Fig. 4.20 Temperature of the suppression pool.

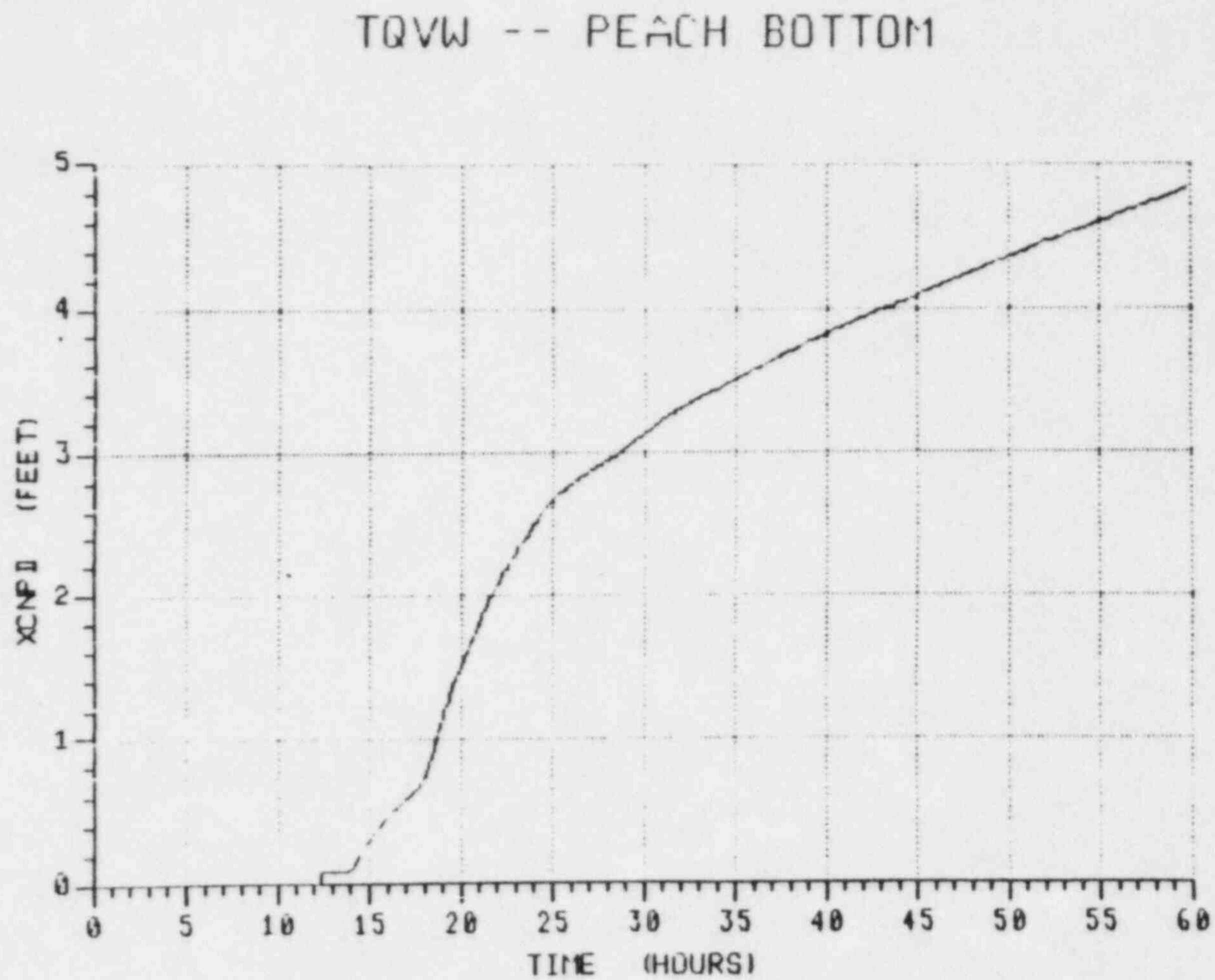

Fig. 4.21 Concrete ablation depth in the pedestal.

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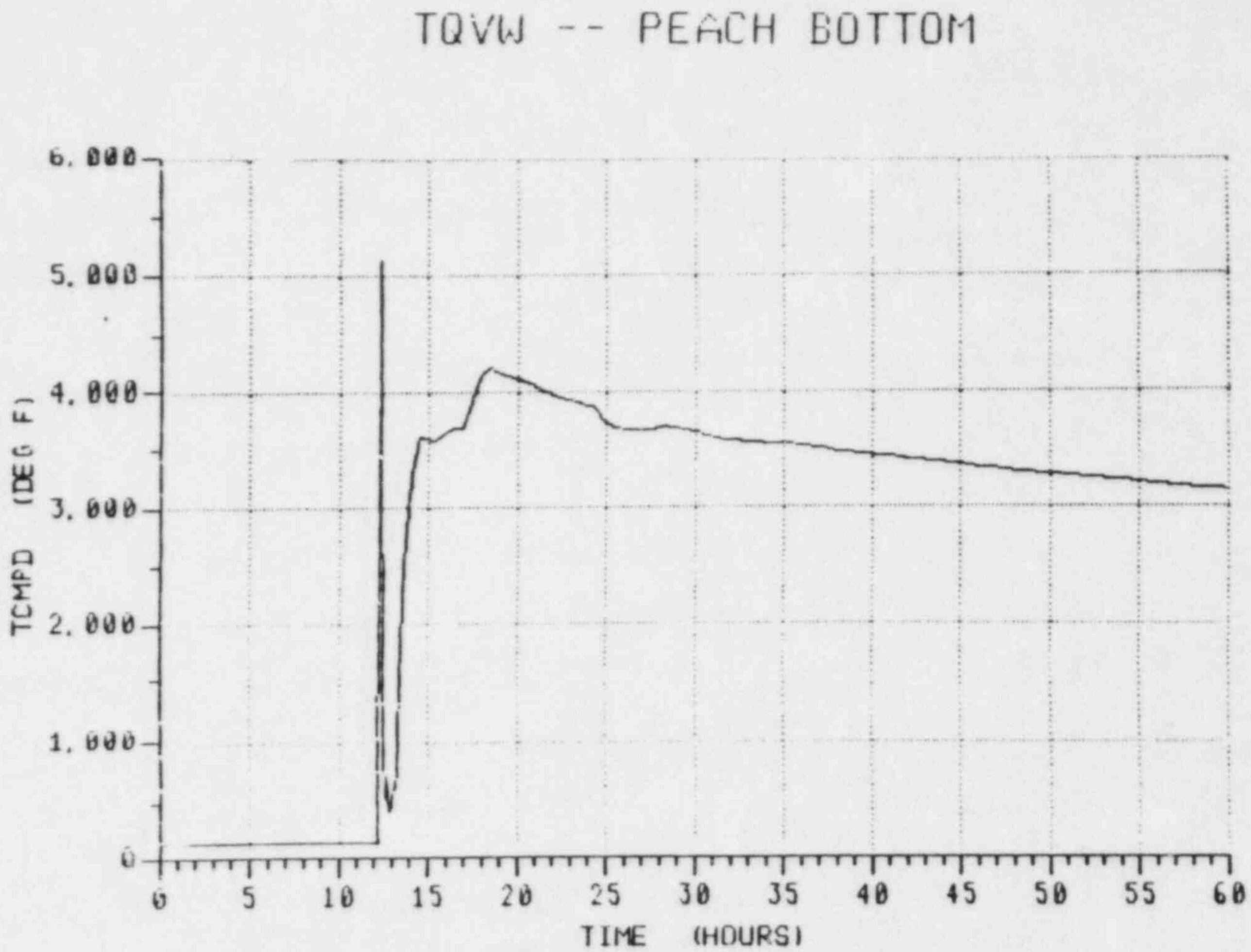

Fig. 4.22 Average corium temperature in the pedestal.

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sequence leads to core melt and vessel failure due to lack of coolant injection followed by containment failure approximately 6 hours later.

Loss of all AC power results in an immediate reactor scram signal followed by successful reactor scram within 4 seconds. The power conversion system is lost and the stored energy and decay power are transmitted to the suppression pool through the SRV lines resulting in suppression pool heatup. The only coolant injection assumed available to the reactor is through the HPCI and RCIC systems, because these pumps are steam turbine driven. All other injection pumps require AC power.

HPCI and RCIC maintain reactor water inventory until DC power required for control of these systems is lost. This is assumed to occur at 6 hours into the accident. It is assumed that heatup of the rooms containing the HPCI and RCIC systems does not cause system failure because of the reasons discussed below. After HPCI and RCIC are lost, the reactor water level boils down uncovering the top of the core at about 8.4 hours. During the boil down, approximately 500 lbm of hydrogen are generated due to cladding oxidation. The core begins to melt at approximately 11.4 hours, and the melting progresses until core plate and vessel failure occur at approximately 12.4 hours. Prior to vessel failure the primary system is at a pressure of 1100 psia, due to the assumption that the operator does not act to open SRVs and ADS actuation does not occur due to the lack of a low pressure pump permissive signal. The pressure in the primary containment rises sharply at vessel failure from 30 psia to 90 psia due to flashing of residual water in the vessel and the generation of noncondensable gases from initial concrete ablation and additional cladding oxidation. After the residual water from the vessel is vaporized there is no water available to quench the core debris. Thus concrete ablation is initiated, but at a slower rate, generating additional gases which continue to pressurize the containment.

The core debris during this time is dispersed over the pedestal and drywell floors in a geometry that results in substantial thermal radiation to the drywell atmosphere and structure. There is a significant temperature rise in the drywell commencing after 13 hours. The assumed failure temperature of 1200°F is reached at approximately 18 hours when the containment pressure is at 105 psia. Following the failure, the containment pre .ure decreases to about 25 psia at 21 hours. Appendix B includes several plots for this sequence.

The indicated progression of this sequence is not likely to occur if one considers possible operator actions due to the following reasons:

- Explicit plant procedures exist for the conservation of DC capacity during a loss of AC power. Such actions would extend the availability of DC power considerably beyond the 6 hours assumed in this analysis.

- For loss of AC power events, plant procedures require that the HPCI and RCIC systems be placed in a manual mode of operation such that any instrumentation failures which could result from elevated room temperatures will not adversely effect s'stem operation. In addition, it is expected that opening of the ECCS compartment doors would provide sufficient room cooling to prevent equipment failure.

- Even if HPCI and RCIC were lost, additional means of vessel makeup are available which do not rely on plant AC or DC power, (i.e. fire trucks or diesel driven pumps through HPSW/RHR). Plant emergency procedures call for the use of this type of equipment under appropriate conditions.

- The conservative analysis of this sequence described in this section indicates that the operators have over 10 hours to restore power or establish an alternative means of injection prior to fuel melting.

- If fuel melting and vessel failure did occur, the operator would have an additional 6 hours to take action to maintain containment integrity to mitigate releases.

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4.5 References

4.1 Reactor Safety Study, WASH-1400, NUREG/75-0114, 1975.

4.2 IDCOR Technical Report on Task 10.1, "Containment Structural Capability of Light Water Nuclear Power Plants," July, 1983.

4.3 "An Evaluation of the Elevated Temperature Tensile Creep Properties of Wrought Carbon Steel," ASTM DS11 and ASTM DS11 Supplement 1.

4.4 L. Chu, "Power Suppression and Boron Remixing Mechanism for General Electric Boiling Water Reactor Emergency Procedures Guidelines," NEDC 22166, August 1983.

5.0 PLANT RESPONSE WITH RECOVERY ACTIONS

-- 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 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

6.3.1 TW Fission Product Release

As previously described the TW sequence is very extended. Essentially a day is available to take corrective action to prevent containment failure. Also, failure of the ECCS systems following containment failure is by definition, not the result of a mechanistic process. With the assumption of no corrective action and loss of all injection results in containment failure prior to melt-through of the reactor pressure vessel. Table 6.1 shows the CsI distribution at vessel failure. Due to the relatively low flows to the suppression pool and the large settling area in the upper plenum 93% of the CsI remains in the primary system. After vessel failure the core debris begins to heat the drywell. Fission products deposited within the primary system heat the structures, vaporize and move within the primary system to colder surfaces. This material transport is illustrated in Figs. B.3 and B.4 of Appendix B. Eventually the entire primary system achieves sufficient temperatures to begin transporting the fission products out into the containment. At this time ($\sim$ 60 hrs) the fission products begin to be discharged into the reactor building. The ultimate CsI distribution at 120 hrs into the accident is shown in Table 6.1 and the fraction release of all fission products at this time is shown in Table 6.2.

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Table 6.1

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

	At Vessel Failure			
	TW	TC	S_1E	TQVW
RPV	.93	0.81	.47	.997
Drywell	0	0	.30	0
Suppression Pool	.07	0.19	.23	.003
Secondary Containment	3×10^{-5}	4×10^{-4}	0	0
Environment	2×10^{-5}	0	0	0

	At Containment Failure			
	TW	TC	S_1E	TQVW
RPV	1.0	0.45	0.42	.80
Drywell	0	0.04	.03	.17
Suppression Pool	0	0.51	.55	.03
Secondary Containment	0	5×10^{-4}	0	0
Environment	0	0	0	0

	Ultimate Distribution			
	TW	TC	S_1E	TQVW
RPV	.01	0.01	0	0
Drywell	0	0	0	0
Suppression Pool	.07	0.51	.57	.034
Secondary Containment	.79	0.45	.42	.92
Environment	.13	0.03	.01	.05

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Table 6.2

TW FISSION PRODUCT RELEASE

Assumptions
Containment Failure Location - Drywell, El 165'
Containment Failure Size - .1 ft^2

Fission Product Group	Release Fraction to Environment
Cs, I	0.13
Te, Sb	0.13
Sr, Ba	9E-5
Ru, Mo	4E-4
Time of Release: 34 hr.	
Duration of Release: 80 hr.	

6.3.2 TC Fission Product Release

The MAAP analysis of this sequence shows an initial deposition of volatile fission products in the reactor vessel and a subsequent redistribution among the different vessel regions after vessel failure as these fission products revaporize due to decay heat.

As indicated in Table 6.1, prior to vessel failure, most of the inventory of volatile fission products is retained in the vessel upper plenum, but significant quantities are transferred to the suppression pool at vessel failure and for several hours thereafter.

The drywell temperature is maintained at a moderate level by the CRD water, which flows into the pedestal and cools the debris by vaporization. After the CST is depleted, the drywell temperature increases to levels which could threaten the integrity of electrical penetrations and allow a bypass of the suppression pool. As the drywell heats up after vessel failure due to the core debris on the floor, heat is transferred to the reactor vessel which ultimately comes into thermal equilibrium with the drywell. As the vessel heats up, all of the volatile fission products retained in the vessel are revaporized and are convected out of the vessel at a low flow rate. These fission products are released from the vessel and transported into the drywell where some are deposited and others are transported to the suppression pool. The drywell is assumed to fail at a temperature of 920 K (1200°F), which allows the airborne fission products to bypass the suppression pool. As a result of the elevated drywell temperature the material is transported mostly as vapor and little deposition occurs in the drywell. At this time a significant amount of concrete aerosols are being released due to core-concrete attack in the pedestal region. The volatile fission products condense and form aerosols as they flow into the reactor building along with the inert aerosols. Most of these materials are removed due to gravitational settling and condensation within the building. Consequently a relatively small fraction of the volatile fission products are released to the environment as indicated in Table 6.3.

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Table 6.3

TC FISSION PRODUCT RELEASE

Assumptions
Containment Failure Location - Drywell, El 165'
Containment Failure Size - .2 ft^2

Fission Product Group	Release Fraction to Environment
Cs, I	0.034
Te, Sb	0.066
Sr, Ba	5E-5
Ru, Mo	2E-4
Time of Release: 13 hr.	
Duration of Release: 50 hr.	

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6.3.3 S_1E Fission Product Release

This sequence was analyzed to determine the time dependent distribution of volatile fission products within the vessel, the rate of release from the vessel to the drywell and, after containment failure, the release to the reactor building and subsequently to the environment. It can be seen that drywell heatup, which occurs from the core debris on the floor, influences the long term heatup of the entire reactor vessel.

Drywell heatup results in the revaporization of the volatile fission products which have been retained in the drywell. Most of this material is convected from the drywell to the reactor building within five hours after containment failure. Revaporization in the reactor vessel is also occurring, but due to low flows from the vessel to the drywell most of the release from the vessel is not complete until about 20 hours after containment failure. This material is passed through the drywell to the reactor building. As indicated in Table 6.1, none of the volatile fission products are ultimately retained in the reactor vessel or drywell. Aerosols initially released from the containment would be sucked into the SGTS system and retained in the filter. After about 80 kg were were accumulated in the filters, the reduction in flow area would cause the SGTS fans to trip on low flow. With the SGTS system shutdown, the building through flow is determined by the flow from the primary containment. This results in a long residence time in the reactor building.

Gravitational settling of fission product aerosols in the reactor building results in substantial retention in the building. The volatile fission products released to the environment are given in Table 6.4.

6.3.4 TQVW Fission Product Release

Since SGTS is unavailable, the path to the environment is through the reactor building with direct leakage to the atmosphere. The reactor building flow rates are governed solely by the containment break flow and the ensuing thermal hydraulic conditions in the reactor building. Therefore,

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Table 6.4

S_1E FISSION PRODUCT RELEASE

Assumptions
Containment Failure Location - Drywell, El 135'
Containment Failure Size - 0.1 ft^2

Fission Product Group	Release Fraction to Environment
Cs, I	0.01
Te, Sb	0.01
Sr, Ba	2E-5
Ru, Mo	6E-5
Time of Release: 23 hr.	
Duration of Release: 30 hr.	

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there is no forced convection or fission product removal resulting from SGTS operation.

As in the other sequences, drywell heatup contributes to the reactor vessel heatup. As indicated in Table 6.1 most of the volatile fission products are in the reactor vessel and the suppression pool at the time of containment failure. The inventories of cesium, iodine and tellurium are somewhat less than those calculated for other sequences since the primary system does not depressurize until the vessel fails. As the drywell and reactor vessel heatup, revaporization of the volatile fission products in the vessel occurs and they are convected out of the vessel with a low flow rate. Consequently, most of these fission products are out of the vessel by about 25 hours after containment failure. These volatile fission products pass through the drywell as vapors and into the comparatively cool reactor building where they condense to form aerosols and result in substantial gravitational settling. The effectiveness of this removal mechanism is enhanced by the low temperature and the long residence times in the reactor building because of the absence of forced convection from SGTS operation. Consequently, only 0.05% of the Cs and I is released to the environment 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.

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Table 6.5

TQVW FISSION PRODUCT RELEASE

Assumptions
Containment Failure Location - Drywell, El 135'
Containment Failure Size - 0.1 ft^2

Fission Product Group	Release Fraction to Environment
Cs, I	0.05
Te, Sb	0.04
Sr, Ba	5E-5
Ru, Mo	2E-4

Time of Release: 18 hr.

Duration of Release: 30 hr.

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6.7 GESARII, 238 Nuc. Island, Appendix 15-D, Severe Accidents - Rev. 2 - General Electric Company, 1982.

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7.0 SUMMARY OF RESULTS

Four severe accident sequences were analyzed for Peach Bottom. These sequences were identified in Task 3.2 as dominant sequences that could potentially lead to core melting. The analyses assumed the accidents proceeded with minimal operator intervention in order to determine the timing and magnitude of the major phenomenological events. The results of MAAP produce time estimates for core melting, vessel failure, and containment failure as well as estimates of fission product release. The results of the analyses of the four sequences are shown in Tables 7.1 and 7.2. As seen from Table 7.1 all sequences led to core melting. Assuming the pressure and temperature failure criteria utilized in these analyses, two sequences resulted in containment failure due to pressure. Containment failure for the other two sequences resulted due to high temperature which occurred prior to the overpressurization criteria. Table 7.2 compares the results of the analyses with WASH-1400. The release fractions of fission products are considerably less than those reported in WASH-1400 due to the more realistic modeling of fission product behavior as well as the transport paths from the containment to the environment. In addition the results indicate that the release of fission products to the environment occurs from several hours to over a day after initiation of the accident.

The base sequences were reanalyzed with MAAP to demonstrate the effectiveness of selected operator actions in mitigating the consequences of these severe accidents. The examples presented in Section 5 demonstrate that proper operator actions are extremely beneficial. There are several alternatives available to operators with present systems and procedures at various stages of the accident sequences to bring the plant to a safe stable state. The assumption that these actions are not taken in the base sequences is unrealistic and makes these cases very low probability events.

Section 5 describes the capabilities that exist at Peach Bottom for venting primary containment. As indicated, venting capacity from the wetwell and drywell is extensive. The gases released from the small lines through the Standby Gas Treatment System and all lines connected to the wetwell are effectively filtered or scrubbed prior to release. The effectiveness of

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Table 7.1

SUMMARY OF MAAP RESULTS FOR BASE SEQUENCES*

Event	TW	TC	S_1E	TQVW
ECCS Start (hrs)	0	0	NA	0
ECCS Stop (hrs)	32	1.8	NA	6
Core Uncovered (hrs)	34	1.3	1.1	8.4
Cladding Temp. at 2000°F (hrs)	36	2.5	1.6	9.8
Fuel Melting Begins (hrs)	39	3.0	2.5	11.4
Vessel Failure (hrs)	40	3.9	3.4	12.4
Fuel Melting Complete (hrs)	75	22	30	35
Containment Failure (hrs)	32	NA	23	18
Drywell Temp. at 600°F (hrs)	45	9	19	14
Drywell Temp. at 1500°F (hrs)	62	14	27	21
Max. Containment Pressure (psia)	132	115	55	105
In-Vessel Zirc Oxidation (lbm H2)	430	300	250	500
Containment Failure Mode (Based on Assumed Criteria)	Pres	NA	Temp	Temp

*These sequences assume minimal operator intervention.

Table 7.2

SUMMARY OF FISSION PRODUCT RELEASE FRACTIONS[(a)]

	Sequence				WASH-1400	
F.P. Group	TW	TC	S_1E	TQVW	BWR2[(b)]	BWR[(c)]
Cesium, Iodine	0.13	0.034	0.01	0.05	0.50, 0.90	0.10
Tellurium	0.13	0.066	0.01	0.04	0.30	0.30
Strontium	9×10^{-5}	5×10^{-5}	2×10^{-5}	5×10^{-5}	0.10	0.01
Ruthenium	4×10^{-4}	2×10^{-4}	6×10^{-5}	2×10^{-4}	0.03	0.02

(a) Fraction of core inventory released to the environment.

(b) Containment failure prior to vessel failure; can be compared with (TW, TC).

(c) Failure to scram or remove decay heat; can be compared with (TC, S_1E, TQVW).

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venting in reducing pressure is also demonstrated in Section 5 for the TW analyses for which it was assumed operators vented from the wetwell in accordance with existing emergency procedures.

It is apparent that a large margin exists in suppression pool venting capacity. Thus, mitigating features such as additional containment vent filters (FVCs) are of considerably diminished incremental value, in the unlikely event that venting would be required.

Review of the base sequences, as well as those with operator interventions, indicates that through realistic assessment of phenomenology, releases are reduced and delayed and safe stable states are achievable.

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8.0 CONCLUSIONS

Task 23 analyses for Peach Bottom have demonstrated several key items relevant to nuclear power plant severe accident analysis for BWRs with Mark I containment designs similar to Peach Bottom.

1. The viability of the Modular Accident Analysis Program (MAAP) in analyzing challenges to containment resulting from degraded core accidents has been demonstrated. This provides an independently developed alternative to the models available prior to the IDCOR Program.

2. The use of MAAP to more realistically determine the release of fission products to the environment following a set of selected low probability, degraded core nuclear power accident sequences indicates that, in general, radionuclide releases would be smaller fractions than those previously estimated in WASH-1400 for similar accident sequences. In addition these releases would occur much later in time.

3. Based on the sequences analyzed, it is clear that reasonable actions by trained operators using existing systems and procedures could effectively mitigate the accident consequences by bringing the plant to a safe stable state. Additionally, fission product releases could be substantially reduced from those calculated in the base cases through the use of existing primary containment venting capabilities and procedures.

4. The containment floor (pedestal) concrete ablation depths at the time of containment failure illustrated in the graphs of Section 4 indicate that base mat penetration is not a likely mode of Mark I containment failure for severe nuclear power accidents.

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5. Heatup of containment from radiative and convective heat transfer from core debris on the containment floor may result in a reduction of the ultimate pressure capability of the containment for some sequences.

6. The reactor building (secondary containment) is extremely effective in retaining aerosol and condensed volatile fission products released from primary containment.

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APPENDIX A

Peach Bottom Parameter File

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**MARK I BWR PLANT PARAMETER VALUES-- TYPICAL OF PEACH BOTTOM          !
**SI UNITS (M-KG-SEC-DEGK)                                             !
**  7-22-83                                                            !
**                                                                     !
*PRIMARY SYSTEM                                                        PS
01   8.862D0    AFLCOR FLOW AREA OF REACTOR CORE                       PS
** ALSH = VOLUME WATER IN LOWER SHROUD BELOW TOP OF ACTIVE FUEL DIVIDED
** BY (ZTOAF-ZBJET)
02   11.8D0     ALSH   FLOW AREA IN LOWER SHROUD                       PS
03   2.D0       AFLBYP CORE BYPASS FLOW AREA                           PS
** AUSH = VOLUME OF WATER IN UPPER DOWNCOMER ABOVE TOAF DIVIDED BY
** THE NORMAL WATER HEIGHT ABOVE TOAF
04   20.0D0     AUSH   FLOW AREA IN UPPER SHROUD                       PS
05   1.35D5     HCRD   SPECIFIC ENTHALPY OF CRD INLET                  PS
06   8.171D5    HFW    SPECIFIC ENTHALPY OF FEEDWATER                  PS
07   1.586D5    MU2COR TOTAL MASS OF UO2 IN CORE                       PS
08   7.64D2     NASS   NUMBER OF FUEL ASSEMBLIES IN REACTOR CORE       PS
09   6.4D1      NPINS  NUMBER OF FUEL RODS IN A FUEL ASSEMBLY          PS
10   1.85D2     NCRD   NUMBER OF CRD TUBES                             PS
11   5.0D0      NQFPS  SESNIBLE ENERGY STORED IN FUEL (FULL POWER SECONDS)PS
12   3.0D0      TDMSIV DELAY TIME FOR MSIV CLOSURE                     PS
13   3.5D0      TDSCRM DELAY TIME FOR FULL SCRAM                       PS
14   3.0D7      TIRRAD TOTAL EFFECTIVE IRRADIATION TIME FOR CORE       PS
** ALL PUMP CURVES ASSOCIATE THE FIRST FLOW ENTRY WITH THE FIRST PRESSURE
** ENTRY
15   7.D-3      WVCRDI  CRD FLOW RATE  PUMP CURVE FOR CRD VOL FLOW     PS
16   1.12D-2    WVCRDI  CRD FLOW RATE      PRESSURE PRIM SYS. VS WVCRDI PS
17   1.12D-2    WVCRDI  CRD FLOW RATE                                  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   2275.D0    WFWMAX MAXIMUM FEEDWATER FLOW RATE (RUN OUT)           PS
32   4.42D2     WBPMAX MAXIMUM TURBINE BYPASS FLOW RATE                PS
33   1.4D-1     NXCORE EXIT CORE QUALITY                               PS
34   5.26D0     XDCORE REACTOR CORE DIAMETER TO INNER SHROUD WALL      PS
35   22.23D0    XHRV   INTERIOR HEIGHT OF REACTOR VESSEL               PS
36   3.188D0    XRRV   INTERIOR RADIUS OF REACTOR VESSEL               PS
37   48.11D0    ZBJET  ELEVATION AT BOTTOM OF JET PUMPS                PS
38   46.29D0    ZBRDT  ELEVATION AT BOTTOM OF CRD TUBES                PS
39   58.19D0    ZBSEP  ELEVATION AT BOTTOM OF STEAM SEPARATORS         PS
40   45.19D0    ZBV    ELEVATION AT BOTTOM OF REACTOR VESSEL           PS
41   50.65D0    ZCPL   ELEVATION AT CORE PLATE                         PS
42   53.40D0    ZTJET  ELEVATION AT TOP OF JET PUMPS                   PS
43   .65D0      AJET   TOTAL JET PUMP AREA                             PS
44   54.43D0    ZTOAF  ELEVATION AT TOP OF ACTIVE FUEL                 PS
45   60.35D0    ZTSEP  ELEVATION AT TOP OF STEAM SEPARATORS            PS
46   59.49D0    ZWNORM ELEVATION AT NORMAL SHROUD WATER LEVEL          PS
47   62.00D0    ZLOCA  ELEVATION AT BREAK                              PS
48   .0093D0    ALOCA  AREA OF BREAK                                   PS
49   60.00D0    ZWL8   ELEVATION AT LEVEL 8 TRIP                       PS
50   0.0D0      NOT USED
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51   58.87D0     ZSCRAM LEVEL 2 TRIP                                          PS
52   7.342D6     PSCRAM HIGH PRESSURE SCRAM SETPOINT                          PS
53   .20D0       FQATWS ATWS CONSTANT POWER ASSUMPTION                        PS
54   3.6D3       TDSLC  TIME FOR SCRAM WITH SLC                               PS
** RR PUMP COASTDOWN CURVE
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   12.D0       TIRR(7)                                                      PS
62   14.D0       TIRR(8)                                                      PS
63   1.D0        FWRR(1)                                                      PS
64   .77D0       FWRR(2)                                                      PS
65   .62D0       FWRR(3)                                                      PS
66   .53D0       FWRR(4)                                                      PS
67   .45D0       FWRR(5)                                                      PS
68   .34D0       FWRR(6)                                                      PS
69   .34D0       FWRR(7)                                                      PS
70   0.D0        FWRR(8)                                                      PS
71   1.46D5      HSLC   INLET ENTHALPY OF SLC                                 PS
72   0.D0        PSLC(1) PRESSURE POINTS FOR SLC FLOW CURVE                   PS
73   1.D7        PSLC(2)                                                      PS
74   1.D7        PSLC(3)                                                      PS
75   1.D7        PSLC(4)                                                      PS
76   1.D7        PSLC(5)                                                      PS
77   1.D7        PSLC(6)                                                      PS
78   1.D7        PSLC(7)                                                      PS
79   1.D7        PSLC(8)                                                      PS
80   1.72D-3     WVSLC(1) SLC FLOW RATE AT PSLC(1)     -- M3/S                PS
81   1.72D-3     WVSLC(2)                                                     PS
82   1.72D-3     WVSLC(3)                                                     PS
83   1.72D-3     WVSLC(4)                                                     PS
84   1.72D-3     WVSLC(5)                                                     PS
85   1.72D-3     WVSLC(6)                                                     PS
86   1.72D-3     WVSLC(7)                                                     PS
87   1.72D-3     WVSLC(8)                                                     PS
88   0.D0        TDRPT  DELAY TIME FOR RECIRC PUMP TRIP                       PS
89   54.79D0     ZLMSIV LOW WATER LEVEL FOR MSIV CLOSURE                      PS
90   57.64D0     ZLRPT  LOW WATER LEVEL FOR RPT                               PS
91   7.886D6     PHRPT  HIGH VESSEL PRESSURE FOR RPT                          PS
92   1.1513D5    PDWSCM HIGH DRYWELL PRESSURE SCRAM SIGNAL                    PS
93   .0299D0     FENRCH NORMAL FUEL ENRICHMENT                                PS
94   20000.D0    EXPO   AVERAGE EXPOSURE 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   6.96D-1     FQFR1  FISSION POWER FRACTION OF U235 AND PU241              PS
98   2.238D-1    FQFR2  FISSION POWER FRACTION OF PU239                       PS
99   8.D-2       FQFR3  FISSION POWER FRACTION OF U238                        PS
100  .3048D0     XPCRDT PITCH OF CRD TUBES                                    PS
101  .2755D0     XDCRDT OUTER DIAMETER OF CRD TUBES                           PS
102  55.D0       NINST  NUMBER OF INSTRUMENT TUBES                            PS
103  .00419D0    XTHCRD THICKNESS OF CRD TUBE WALL                            PS
104  .0508D0     XDINST OUTER DIAMETER OF INSTRUMENT TUBE                     PS
105  .075D0      XDRIVE LOWER CRD DRIVE OUTER DIAMETER                        PS
106  1.0051D-3   VWCRD  SPECIFIC VOMUME OF CRD WATER                          PS
107  1.0051D-3   VWSLC  SPECIFIC VOLUME OF SLC WATER                          PS
108  1.05D5      MEQPS  MASS OF UPPER PLENUM HEAT SINK                        PS
109  1.016D3     AEQPS  AREA OF UPPER PLENUM HEAT SINK                        PS
110  .21437D0    XTRV   THICKNESS OF LOWER VESSEL HEAD                        PS
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111  0.D0       TIFWCD TIME SINCE MSIV CLOSURE SIGNAL VS. FEEDWATER          PS
112  0.D0              COASTDOWN MASS FLOW RATE                              PS
113  0.D0              8 TIME POINTS,8 FLOW RATES                            PS
114  0.D0                                                                    PS
115  0.D0                                                                    PS
116  0.D0                                                                    PS
117  0.D0                                                                    PS
118  0.D0                                                                    PS
119  0.D0        WFWCD                                                       PS
120  0.D0                                                                    PS
121  0.D0                                                                    PS
122  0.D0                                                                    PS
123  0.D0                                                                    PS
124  0.D0                                                                    PS
125  0.D0                                                                    PS
126  0.D0                                                                    PS
127  5.96D6      PLMSIV LOW RPV PRESSURE FOR MSIV CLOSURE                    PS
128  62.35D0     ZMSL   ELAVATION AT CENTER LINE OF THE MAIN STEAM LINE      PS
**                                                                           HE
*CIRC
01   0.D0        ACSHS(1) CORE + LOWER PLENUM
**                        CARBON STEEL-HEAT SINK HEAT TRANSFER AREA
02   140.D0      ACSHS(2) UPPER PLENUM
03   0.D0        ACSHS(3) DOWNCOMER
04   0.D0        ACSHS(4)
05   0.D0        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.D0        MCS(4)
10   0.D0        MCS(5)
11   0.D0        MHS(1)   CORE + LOWER PLENUM HEAT SINK MASS
12   100.D3      MHS(2)   UPPER PLENUM
13   0.D0        MHS(3)   DOWNCOMER
14   0.D0        MHS(4)
15   0.D0        MHS(5)
16   0.D0        ACSX(1)  CORE + LOWER PLENUM CARBON STEEL TO DRYWELL
**                        HEAT TRANSFER ARE
17   0.D0        ACSX(2)  UPPER PLENUM
18   240.D0      ACSX(3)  DOWNCOMER
19   0.D0        ACSX(4)
20   0.D0        ACSX(5)
21   0.D0        AHSX(1)  CORE + LOWER PLENUM HEAT SINK TO DRYWELL
**                        HEAT TRANSFER AREA
22   140.D0      AHSX(2)  UPPER PLENUM
23   0.D0        AHSX(3)  DOWNCOMER
24   0.D0        AHSX(4)
25   0.D0        AHSX(5)
26   100.D0      AGCS(1)  CORE + LOWER PLENUM GAS TO CARBON STEEL
**                        HEAT TRANSFER AREA
27   5.D3        AGCS(2)  UPPER PLENUM
28   240.D0      AGCS(3)  DOWNCOMER
29   0.D0        AGCS(4)
30   0.D0        AGCS(5)
31   0.D0        AGHS(1)  CORE + LOWER PLENUM GAS TO HEAT SINK
**                        HEAT TRANSFER AREA
32   140.D0      AGHS(2)  UPPER PLENUM
33   0.D0        AGHS(3)  DOWNCOMER
34   0.D0        AGHS(4)
35   0.D0        AGHS(5)
36   8.0D0       XL(1)    CORE + LOWER PLENUM LENGTH
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37   5.D0       XL(2)    UPPER PLENUM LENGTH
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
*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   3.5D4      MZRCAN TOTAL MASS OF ZR IN ASSEMBLY CANS                 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 CENTER BOTTOM, 1,10 IS CENTER TOP, 2,1 IS SECOND RADIAL
** REGION AT THE BOTTOM, ETC.
07   1.061D0    FPEAK(1,1) PEAKING FACTOR FOR NODE (1,1)                 HE
08   .898D0     FPEAK(2,1) PEAKING FACTOR FOR NODE (2,1)                 HE
09   .326D0     FPEAK(3,1) PEAKING FACTOR FOR NODE (3,1)                 HE
15   1.438D0    FPEAK(1,2) PEAKING FACTOR FOR NODE (1,2)                 HE
16   1.217D0    FPEAK(2,2) PEAKING FACTOR FOR NODE (2,2)                 HE
17   .442D0     FPEAK(3,2) PEAKING FACTOR FOR NODE (3,2)                 HE
23   1.553D0    FPEAK(1,3) PEAKING FACTOR FOR NODE (1,3)                 HE
24   1.315D0    FPEAK(2,3) PEAKING FACTOR FOR NODE (2,3)                 HE
25   .477D0     FPEAK(3,3) PEAKING FACTOR FOR NODE (3,3)                 HE
31   1.592D0    FPEAK(1,4) PEAKING FACTOR FOR NODE (1,4)                 HE
32   1.348D0    FPEAK(2,4) PEAKING FACTOR FOR NODE (2,4)                 HE
33   .489D0     FPEAK(3,4) PEAKING FACTOR FOR NODE (3,4)                 HE
39   1.605D0    FPEAK(1,5) PEAKING FACTOR FOR NODE (1,5)                 HE
40   1.358D0    FPEAK(2,5) PEAKING FACTOR FOR NODE (2,5)                 HE
41   .493D0     FPEAK(3,5) PEAKING FACTOR FOR NODE (3,5)                 HE
47   1.631D0    FPEAK(1,6) PEAKING FACTOR FOR NODE (1,6)                 HE
48   1.381D0    FPEAK(2,6) PEAKING FACTOR FOR NODE (2,6)                 HE
49   .501D0     FPEAK(3,6) PEAKING FACTOR FOR NODE (3,6)                 HE
55   1.567D0    FPEAK(1,7) PEAKING FACTOR FOR NODE (1,7)                 HE
56   1.326D0    FPEAK(2,7) PEAKING FACTOR FOR NODE (2,7)                 HE
57   .482D0     FPEAK(3,7) PEAKING FACTOR FOR NODE (3,7)                 HE
63   1.541D0    FPEAK(1,8) PEAKING FACTOR FOR NODE (1,8)                 HE
64   1.304D0    FPEAK(2,8) PEAKING FACTOR FOR NODE (2,8)                 HE
65   .473D0     FPEAK(3,8) PEAKING FACTOR FOR NODE (3,8)                 HE
71   1.465D0    FPEAK(1,9) PEAKING FACTOR FOR NODE (1,9)                 HE
72   .986D0     FPEAK(2,9) PEAKING FACTOR FOR NODE (2,9)                 HE
73   .358D0     FPEAK(3,9) PEAKING FACTOR FOR NODE (3,9)                 HE
79   .777D0     FPEAK(1,10)PEAKING FACTOR FOR NODE (1,10)                HE
80   .658D0     FPEAK(2,10)PEAKING FACTOR FOR NODE (2,10)                HE
81   .239D0     FPEAK(3,10)PEAKING FACTOR FOR NODE (3,10)                HE
87   0.33D0     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   2.D0       NLPCI1 NUMBER OF LPCI PUMPS IN LOOP 1                    ES
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02   2.0D0       NLPCI2 NUMBER OF LPCI PUMPS IN LOOP 2                       ES
03   0.0D0       NLPCI3 NUMBER OF LPCI PUMPS IN LOOP 3                       ES
04   4.0D0       NLPCSP NUMBER OF LPCS PUMPS                                 ES
05   0.0D0       NOT USED
06   100.D0      VMNCST MIN. WATER VOLUME IN CONDENSATE STORAGE TANK         ES
07   1.0051D-3   VWCST  SPECIFIC VOLUME OF CST WATER                         ES
** PUMP CURVES ARE DEFINED SO THE FIRST PRESSURE ENTRY CORRESPONDS WITH
** THE FIRST VOLUMETRIC FLOW ENTRY
08   8.124D6     PHPCI(1)      PUMP CURVES FOR ECCS                          ES
09   8.D6        PHPCI(2)      PPS-PDW VS VOLUMETRIC FLOW                    ES
10   6.8D6       PHPCI(3)      HPCI                                          ES
11   5.1D6       PHPCI(4)                                                    ES
12   3.06D6      PHPCI(5)                                                    ES
13   1.7D6       PHPCI(6)                                                    ES
14   1.134D6     PHPCI(7)                                                    ES
15   6.17D5      PHPCI(8)                                                    ES
16   .31567D0    WVHPCI(1)                                                   ES
17   .31567D0    WVHPCI(2)                                                   ES
18   .31567D0    WVHPCI(3)                                                   ES
19   .31567D0    WVHPCI(4)                                                   ES
20   .31567D0    WVHPCI(5)                                                   ES
21   .31567D0    WVHPCI(6)                                                   ES
22   .31567D0    WVHPCI(7)                                                   ES
23   0.0D0       WVHPCI(8)                                                   ES
24   2.172D6     PLPCI(1)          LPCI                                      ES
25   2.123D6     PLPCI(2)                                                    ES
26   1.916D6     PLPCI(3)                                                    ES
27   1.66D6      PLPCI(4)                                                    ES
28   1.4788D6    PLPCI(5)                                                    ES
29   1.065D6     PLPCI(6)                                                    ES
30   7.894D5     PLPCI(7)                                                    ES
31   1.01342D5   PLPCI(8)                                                    ES
32   0.D0        WVLPCI(1)                                                   ES
33   6.313D-2    WVLPCI(2)                                                   ES
34   1.894D-1    WVLPCI(3)                                                   ES
35   2.525D-1    WVLPCI(4)                                                   ES
36   3.47D-1     WVLPCI(5)                                                   ES
37   4.48D-1     WVLPCI(6)                                                   ES
38   5.05D-1     WVLPCI(7)                                                   ES
39   6.31D-1     WVLPCI(8)                                                   ES
40   2.099D6     PLPCS(1)             LPCS                                   ES
41   2.03D6      PLPCS(2)                                                    ES
42   1.961D6     PLPCS(3)                                                    ES
43   1.892D6     PLPCS(4)                                                    ES
44   1.824D6     PLPCS(5)                                                    ES
45   1.479D6     PLPCS(6)                                                    ES
46   8.928D5     PLPCS(7)                                                    ES
47   1.01342D5   PLPCS(8)                                                    ES
48   0.D0        WVLPCS(1)                                                   ES
49   .041D0      WVLPCS(2)                                                   ES
50   .063D0      WVLPCS(3)                                                   ES
51   .078D0      WVLPCS(4)                                                   ES
52   .0915D0     WVLPCS(5)                                                   ES
53   .1355D0     WVLPCS(6)                                                   ES
54   .1895D0     WVLPCS(7)                                                   ES
55   .246D0      WVLPCS(8)                                                   ES
72   7.0D6       PRCIC(1)          RCIC                                      ES
73   6.0D6       PRCIC(2)                                                    ES
74   5.0D6       PRCIC(3)                                                    ES
75   4.0D6       PRCIC(4)                                                    ES
76   3.0D6       PRCIC(5)                                                    ES
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77   2.0D6       PRCIC(6)                                                        ES
78   1.134D6     PRCIC(7)                                                        ES
79   6.17D5      PRCIC(8)                                                        ES
80   .03788D0    WVRCIC(1)                                                       ES
81   .03788D0    WVRCIC(2)                                                       ES
82   .03788D0    WVRCIC(3)                                                       ES
83   .03788D0    WVRCIC(4)                                                       ES
84   .03788D0    WVRCIC(5)                                                       ES
85   .03788D0    WVRCIC(6)                                                       ES
86   .03788D0    WVRCIC(7)                                                       ES
87   0.0D0       WVRCIC(8)                                                       ES
88   57.66D0     ZLHPCI LOW WATER INITIATION FOR HPCI                            ES
89   1.1513D5    PSHPCI HIGH DRYWELL PRESSURE SET POINT FOR HPCI                 ES
90   25.D0       TDHPCI TIME DELAY FOR HPCI                                      ES
91   6.205D5     PHHPCI MINIMUM VESSEL PRESSURE FOR HPCI TURBINE                 ES
92   50.D0       ZLHPCS LOW WATER INITIATION FOR HPCS                            ES
93   1.1513D5    PSHPCS HIGH DRYWELL PRESSURE SET POINT FOR HPCS                 ES
94   0.D0        TDHPCS TIME DELAY FOR HPCS                                      ES
95   54.79D0     ZLLPCI LOW WATER INITIATION FOR LPCI                            ES
96   1.1513D5    PSLPCI HIGH DRYWELL PRESSURE SET POINT FOR LPCI                 ES
97   24.D0       TDLPCI TIME DELAY FOR LPCI                                      ES
98   3.2057D6    PLLPCI LOW VESSEL PRESSURE PERMISSIVE FOR LPCI LOCA SIGNALES
99   54.79D0     ZLLPCS LOW WATER INITIATION FOR LPCS                            ES
100  1.1513D5    PSLPCS HIGH DRYWELL PRESSURE SET POINT FOR LPCS                 ES
101  12.D0       TDLPCS TIME DELAY FOR LPCS                                      ES
102  3.2057D6    PLLPCS LOW VESSEL PRESSURE PERMISSIVE FOR LPCS LOCA SIGNALES
103  57.66D0     ZLRCIC LOW WATER INITIATION FOR RCIC                            ES
104  1.1513D5    PSRCIC HIGH DRYWELL PRESSURE SET POINT FOR RCIC                 ES
105  30.D0       TDRCIC TIME DELAY FOR RCIC                                      ES
106  6.205D5     PHRCIC MINIMUM VESSEL PRESSURE FOR RCIC TURBINE                 ES
107  1.35.4D5    HCST   ENTHALPY OF CST                                          ES
108  290.D0      WSWHX  SERVICE WATER FLOW RATE (KG/S) PER RHR HTX               ES
** THE SAFETY RELIEF VALVES MUST BE ENTERED IN ORDER OF INCREASING
** PRESSURE ACTUATION SET POINTS
**   I.E.         GROUP #1 PSRV1=7.7D6
**                GROUP #2 PSRV2=7.75D6........ETC
109  .8605D-2   ASRV1   FLOW AREA OF RELIEF VALVE TYPE #1                       ES
110  .8605D-2   ASRV2   FLOW AREA OF RELIEF VALVE TYPE #2                       ES
111  .8607D-2   ASRV3   FLOW AREA OF RELIEF VALVE TYPE #3                       ES
112  .8597D-2   ASRV4   FLOW AREA OF RELIEF VALVE TYPE #4                       ES
** IF THE AREA FOR GROUP #5 IS INPUT AS A NEGATIVE NUMBER THEN THESE
** VALVES WILL DISCHARGE DIRECTLY INTO THE DRYWELL, IF THE AREA IS
** POSITIVE THEN THESE VALVES DISCHARGE INTO THE SUPPRESSION POOL
113  -.8659D-2 ASRV5   FLOW AREA OF SAFETY 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  6.0D0      NSRV3   NUMBER OF TYPE #3 RELIEF VALVES                         ES
117  3.0D0      NSRV4   NUMBER OF TYPE #4 RELIEF VALVES                         ES
118  2.D0       NSRV5   NUMBER OF TYPE #5 RELIEF VALVES                         ES
119  0.D0       NADS1   NUMBER OF ADS VALVES IN GROUP 1                         ES
120  0.D0       NADS2   NUMBER OF ADS VALVES IN GROUP 2                         ES
121  2.D0       NADS3   NUMBER OF ADS VALVES IN GROUP 3                         ES
122  3.D0       NADS4   NUMBER OF ADS VALVES IN GROUP 4                         ES
123  7.7179D6   PSRV1   PRESSURE SETPOINT FOR #1 RELIEF VALVE                   ES
124  7.7179D6   PSRV2   PRESSURE SETPOINT FOR #2 RELIEF VALVE                   ES
125  7.7730D6   PSRV3   PRESSURE SETPOINT FOR #3 RELIEF VALVE                   ES
126  7.8557D6   PSRV4   PRESSURE SETPOINT FOR #4 RELIEF VALVE                   ES
127  8.6003D6   PSRV5   PRESSURE SETPOINT FOR #5 RELIEF VALVE                   ES
128  54.79D0    ZLADS   LOW WATER LEVEL FOR ADS INITIATION                      ES
129  115.13D3   PSADS   HIGH DRYWELL PRESSURE SET POINT FOR ADS                 ES
130  105.D0     TDADS   TIME DELAY FOR ADS ACTUATION                            ES
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** LPCS,LPCI HAVE NPSH REQUIRMENTS, RCIC AND HPCI TRIP ON HIGH SUPP POOL
** TEMPERATURE
131  366.33D0   TCHPCI INLET TEMP LIMIT FOR HPCI                              ES
132  27.88D0    ZCLHPS PUMP CENTER LINE ELAVATION FOR HPCS                    ES
133  27.88D0    ZCLLPI PUMP CENTER LINE ELAVATION FOR LPCI                    ES
134  27.88D0    ZCLLPS PUMP CENTER LINE ELAVATION FOR LPCS                    ES
135  366.33D0   TCRCIC INLET TEMP LIMIT FOR RCIC                              ES
136  300.D0     TWSW   SERVICE WATER TEMP (RHR HEAT EXCHANGERS,TCOLD)         ES
137  1.D0       TDDG1  HPCS LOAD DELAY TIME FOR DIESEL                        ES
138  11.D0      TDDG2  LPCI LOAD DELAY TIME FOR DIESEL                        ES
139  11.D0      TDDG3  LPCS LOAD DELAY TIME FOR DIESEL                        ES
140  1.D-3      XDDROP SPRAY DROPLET DIAMETER                                 ES
141  3.66D0     XHSPWW SPRAY FALL HEIGHT IN WETWELL                           ES
142  14.02D0    XHSPDW SPRAY FALL HEIGHT IN DRYWELL                           ES
** THE FOLLOWING PUMP CURVES CAN BE USED TO DEFINE ANY INJECTION SYSTEM
143  1.35D5     HWHPSW ENTHALPY OF HIGH PRES SERVICE WATER (MARK I )          ES
144  1.D-3      VWHPSW SPEC VOL OF HIGH PRES SERVICE WATER (MARK I )          ES
** PUMP CURVES ARE WRITTEN SO THAT THE FIRST PRESSURE POINT CORRESPONDS
** TO THE FIRST FLOW RATE FOR THAT PUMP
145  1.D10      PHPSW(1) PPS VS. VOLUMETRIC FLOW FOR HPSW CORE INJECTION      ES
146  1.D10      PHPSW(2)                        (MARK I CORE INJECTION)       ES
147  1.D10      PHPSW(3)                                                      ES
148  1.D10      PHPSW(4)                                                      ES
149  1.D10      PHPSW(5)                                                      ES
150  1.D10      PHPSW(6)                                                      ES
151  1.D10      PHPSW(7)                                                      ES
152  1.D10      PHPSW(8)                                                      ES
153  3.79D-2    WVHPSW(1)                                                     ES
154  3.79D-2    WVHPSW(2)                                                     ES
155  3.79D-2    WVHPSW(3)                                                     ES
156  3.79D-2    WVHPSW(4)                                                     ES
157  3.79D-2    WVHPSW(5)                                                     ES
158  3.79D-2    WVHPSW(6)                                                     ES
159  3.79D-2    WVHPSW(7)                                                     ES
160  3.79D-2    WVHPSW(8)                                                     ES
161  1.D10      PDWSPR DRYWELL PRES SET PT FOR MARK III CONTAINMNT SPRAYS     ES
162  1.D10      PWWSPR WETWELL PRES SET PT FOR MARK III CONTAINMNT SPRAYS     ES
163  0.D0       TDSPR  TIME DELAY FOR MARK III CONTAINMENT SPRAYS             ES
164  2.413D5    PDSRV1 DEAD BAND FOR SRV#1                                    ES
165  2.413D5    PDSRV2 DEAD BAND FOR SRV#2                                    ES
166  2.413D5    PDSRV3 DEAD BAND FOR SRV#3                                    ES
167  2.413D5    PDSRV4 DEAD BAND FOR SRV#4                                    ES
168  2.413D5    PDSRV5 DEAD BAND FOR SRV#5                                    ES
169  7.48D6     PTURHP(1)  PPS-PWW VS. STEAM FLOW TO HPCI TURBINE             ES
170  7.928D5    PTURHP(2)                                                     ES
171  7.928D5    PTURHP(3)                                                     ES
172  7.928D5    PTURHP(4)                                                     ES
173  7.928D5    PTURHP(5)                                                     ES
174  7.928D5    PTURHP(6)                                                     ES
175  7.928D5    PTURHP(7)                                                     ES
176  7.928D5    PTURHP(8)                                                     ES
177  23.D0      WSTHPI(1)                                                     ES
178  12.D0      WSTHPI(2)                                                     ES
179  12.D0      WSTHPI(3)                                                     ES
180  12.D0      WSTHPI(4)                                                     ES
181  12.D0      WSTHPI(5)                                                     ES
182  12.D0      WSTHPI(6)                                                     ES
183  12.D0      WSTHPI(7)                                                     ES
184  12.D0      WSTHPI(8)                                                     ES
185  7.7D6      PTURRI(1)  PPS-PWW VS. STEAM FLOW TO RCIC TURBINE             ES
186  1.013D6    PTURRI(2)                                                     ES
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187  1.013D6     PTURRI(3)                                                        ES
188  1.013D6     PTURRI(4)                                                        ES
189  1.013D6     PTURRI(5)                                                        ES
190  1.013D6     PTURRI(6)                                                        ES
191  1.013D6     PTURRI(7)                                                        ES
192  1.013D6     PTURRI(8)                                                        ES
193  3.5D0       WSTRCI(1)                                                        ES
194  1.0D0       WSTRCI(2)                                                        ES
195  1.0D0       WSTRCI(3)                                                        ES
196  1.0D0       WSTRCI(4)                                                        ES
197  1.0D0       WSTRCI(5)                                                        ES
198  1.0D0       WSTRCI(6)                                                        ES
199  1.0D0       WSTRCI(7)                                                        ES
200  1.0D0       WSTRCI(8)                                                        ES
201  1.1355D6    PHTURH HIGH HPCI TURBINE EXHAUST PRESSURE                        ES
202  3.77D5      PHTURR HIGH RCIC TURBINE EXHAUST PRESSURE                        ES
203  9.0794D5    PCFAIL CONTAINMENT FAILURE PRESSURE                              ES
204  3.202D6     PHLPCI HIGH PRESSURE TRIP FOR LPCI                               ES
205  3.202D6     PHLPCS HIGH PRESSURE TRIP FOR LPCS                               ES
206  33.64D0     ZHISP  HIGH SUPP. POOL LEVEL FOR HP/RCIC SUCTION SWITCH          ES
207  0.D0        ZLSPR  LOW WATER LEVEL FOR AUTO WETWELL SPRAYS (M-III)           ES
** IF THE DETAILED HEAT EXCHANGER IS NOT USED ONLY SUPPLY THE
** NTU VALUE AND NUMBER OF HTXS
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
219  .654D0      NTUHX1 NTU FOR RHR HTX #1                                        ES
220  .654D0      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  1.D0        FHX    TYPE OF RHR HTX(1=STRAIGHT TUBE,2=U TUBE)                 ES
224  21.6D3      TDBATT BATTERY OPERATION TIME FOR STATION BLACK-OUT              ES
** THE NPSH POINTS CORRESPOND WITH THE ABOVE FLOW RATE FOR THAT PUMP
233  0.D0        ZHDLPI NPSH CURVE FOR LPCI VS. FLOW (ABOVE)                      ES
234  0.D0               (METERS)                                                  ES
235  0.D0                                                                         ES
236  7.808D0                                                                      ES
237  7.9768D0                                                                     ES
238  8.213D0                                                                      ES
239  8.375D0                                                                      ES
240  8.796D0                                                                      ES
241  0.D0        ZHDLPS NPSH CURVE FOR LPCS VS. FLOW (ABOVE)                      ES
242  7.969D0                                                                      ES
243  8.024D0                                                                      ES
244  8.076D0                                                                      ES
245  8.134D0                                                                      ES
246  8.381D0                                                                      ES
247  9.116D0                                                                      ES
248  10.64D0                                                                      ES
249  27.88D0     CENTER LINE ELEVATION FOR RCIC PUMP                              ES
250  27.88D0     CENTER LINE ELEVATION FOR HPCI PUMP                              ES
251  .0093D0     ACVENT AREA OF CONTAINMENT VENT                                  ES
252  0.D0        ZCFAIL ELEVATION OF CONTAINMENT VENT IN WETWELL (MII ONLY)ES
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233  [illegible]3.8DO  ZSRVD  AVERAGE ELEVATION OF SRV DISCHARGE IN SUPP POOL    ES
254  0.DO      TGDWHX(1)  COOLING CURVE FOR DRYWELL COOLERS
255  0.DO      TGDWHX(2)  TEMP IN DRYWELL VS. HEAT LOSS RATE (J/S)
256  0.DO      TGDWHX(3)
257  0.DO      TGDWHX(4)
258  0.DO      TGDWHX(5)
259  0.DO      TGDWHX(6)
260  0.DO      TGDWHX(7)
261  0.DO      TGDWHX(8)
262  0.DO      QGDWHX(1)  HEAT LOSS RATE FOR DRYWELL COOLERS (J/S)
263  0.DO      QGDWHX(2)
264  0.DO      QGDWHX(3)
265  0.DO      QGDWHX(4)
266  0.DO      QGDWHX(5)
267  0.DO      QGDWHX(6)
268  0.DO      QGDWHX(7)
269  0.DO      QGDWHX(8)
**                                                                             DW
**                                                                             DW
*DRYWELL                                                                       DW
01    .5DO       RELHDW RELATIVE HUMIDITY IN DRYWELL                           DW
02    4841.DO    VOLDW  FREE VOLUME OF DRYWELL                                 DW
03    36.55DO    ZDWF   ELEVATION AT DRYWELL FLOOR                             DW
04    84.DO      ADWF   AREA OF DRYWELL FLOOR                                  DW
05    37.24DO    ZWDWWW ELEVATION OF DRYWELL-WETWELL WALL                      DW
06    0.DO       NIGDW  NUMBER OF IGNITERS IN THE DRYWELL                      DW
07    0.DO       XIGDW  AVERAGE DISTANCE FROM IGNITERS TO CEILING              DW
08    0.DO       ACHDW  FLOOR BURN AREA                                        DW
**                                                                             WW
**                                                                             WW
*WETWELL                                                                       WW
01    28.65DO    ZWWF   ELEVATION AT WETWELL FLOOR                             WW
02    .169DO     AVB    FLOW AREA THROUGH VACUUM BREAKERS                      WW
03    12.0DO     NVB    NUMBER OF VACUUM BREAKERS                              WW
04    3.447D3    PSETVB PRESSURE SETPOINT FOR VACUUM BREAKERS                  WW
05    2.757D3    PDVB   DEAD BAND FOR VACUUM BREAKERS                          WW
06    7419DO     VOLWW  FREE VOLUME OF WETWELL (MARK II AND MARK III ONLY)     WW
07    1.DO       RELHWW RELATIVE HUMIDITY IN WETWELL                           WW
08    0.DO       NIGWW  NUMBER OF IGNITERS IN THE WETWELL                      WW
09    0.DO       XIGWW  AVERAGE DISTANCE FROM IGNITERS TO CEILING              WW
10    0.DO       ACHWW  FLOOR BURN AREA                                        WW
11    0.DO       AWWF   AREA OF WETWELL FLOOR (MARK II)                        WW
**                                                                             PD
**                                                                             PD
*PEDESTAL                                                                      PD
01    2.917D1    APDF   AREA OF PEDESTAL FLOOR                                 PD
02    4.5DO      APDVT  AREA OF PEDESTAL-DRYWELL OPENING                       PD
03    2.40D2     VOLPD  VOLUME OF PEDESTAL                                     PD
04    36.55DO    ZWPDDW ELEVATION OF WALL BETWEEN PED AND DRYWELL              PD
05    36.35DO    ZPDF   ELEVATION AT PEDESTAL FLOOR                            PD
06    .5DO       RELHPD RELATIVE HUMIDITY IN PEDESTAL                          PD
07    0.DO       NIGPD  NUMBER OF IGNITERS IN THE PEDESTAL                     PD
08    0.DO       XIGPD  AVERAGE DISTANCE FROM IGNITERS TO CEILING              PD
09    0.DO       ACHPD  FLOOR BURN AREA                                        PD
10    0.DO       AWPDVT WIDTH OF PEDESTAL DOOR (MARK II ONLY)                  PD
11    0.DO       ADCPD  AREA OF PEDESTAL DOWNCOMERS
12    0.DO       NDCPD  NUMBER OF DOWNCOMERS IN PEDESTAL
13    2.DO       XHPDDW DISTANCE BETWEEN UPPER AND LOWER VENTS FOR
**                      PED-DRYWELL NATURAL CIRCULATION
**                                                                             TO
**                                                                             TO
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*TORUS AND MARK II WETWELL                                                         TO
01    4.72D0     XRTOR  MINOR RADIUS OF TORUS (MI ONLY)                             TO
02    106.7D0    XLTOR  CIRCUMFERENCE OF TORUS (MI ONLY)                            TO
03    .292D0     ADC    AREA OF DOWNCOMER (MI AND MII ONLY)                         TO
04    96.D0      NDC    NUMBER OF DOWNCOMERS (MI AND MII ONLY)                      TO
** THE NEXT PARAMETER HAS BEEN REPLACES WITH MEQWW AND MEQWWS
05    848.43D0   VSSTOR VOLUME OF STRUCTURE IN TORUS (MI ONLY)                      TO
06    32.D0      ZBDC   ELEVATION AT BOTTOM OF DOWNCOMER (MI AND MII ONLY)          TO
07    34.74D0    ZTDC   ELEVATION AT TOP OF DOWNCOMER (MI AND MII ONLY)             TO
08    28.8D0     ZBTOR  ELEVATION AT BOTTOM OF TORUS (MI ONLY)                      TO
09    0.0168D0   XTOR   THICKNESS OF TORUS SHELL (MI ONLY)                          TO
10    5072.D0    ATR    AREA OF TORUS ROOM WALL (MI ONLY)                           TO
11    101342.D0  PTR    PRESSURE IN TORUS ROOM (MI ONLY)                            TO
12    1.5D4      VOLTR  VOLUME OF TORUS ROOM (MI ONLY)                              TO
13    34.747D0   ZVBTOR CENTER LINE ELEVATION OF VACUUM BRKRS(MI AND MII)           TO
14    .015D0     XTHDC  THICKNESS OF DOWNCOMER PIPE (MII ONLY)                      TO
**
**
*INITIAL CONDITIONS                                                                IN
01    3.293D9    QPOWER CORE POWER                                                  IN
02    7.033D6    PPSO   INITIAL PRESSURE IN PRIMARY SYSTEM                          IN
03    1.041D5    PPDO   INITIAL PRESSURE IN PEDESTAL                                IN
04    1.041D5    PDWO   INITIAL PRESSURE IN DRYWELL                                 IN
05    1.041D5    PWWO   INITIAL PRESSURE IN WETWELL                                 IN
06    33.22D0    ZSPDWO INIT.ELEV. OF WATER LEVEL IN DW SIDE OF SUPP.POOL           IN
**                      (FOR MI AND MII THIS IS ELEVATION IN DOWNCOMER)             IN
07    33.22D0    ZSPWWO INIT.ELEV. OF WATER LEVEL IN WW SIDE OF SUPP.POOL           IN
08    3.3D2      TPDO   INITIAL TEMPERATURE IN PEDESTAL                             IN
09    3.3D2      TDWO   INITIAL TEMPERATURE IN DRYWELL                              IN
10    3.05D2     TWWO   INITIAL TEMPERATURE IN WETWELL                              IN
11    3.05D2     TWSPO  INITIAL TEMPERATURE OF SUPPRESSION POOL WATER               IN
12    59.49D0    ZWSHO  INITIAL ELEVATION OF WATER IN THE SHROUD                    IN
13    0.D0       MWCBO  MASS OF WATER IN UPPER POOL (MARKIII ONLY)                  IN
14    591.D0     VCSTO  VOLUME OF WATER IN CONDENSATE STORAGE TANK                  IN
15    305.D0     TAMB   AMBIENT TEMPERATURE                                         IN
16    1.0134D5   PAMB   AMBIENT PRESSURE                                            IN
**
**
*HTSINKS                                                                           HS
01    189.D0     AHS1   AREA OF WALL #1 PEDESTAL-DRYWELL WALL                       HS
02    1507.D0    AHS2   AREA OF WALL #2 DRYWELL WALL                                HS
03    0.D0       AHS3   AREA OF WALL #3 DRYWELL FLOOR                               HS
04    5073.D0    AHS4   AREA OF WALL #4 TORUS ROOM WALL (MI ONLY)                   HS
09    1.3D0      KHS1   THERMAL CONDUCTIVITY OF WALL #1                             HS
10    1.3D0      KHS2   THERMAL CONDUCTIVITY OF WALL #2                             HS
11    1.3D0      KHS3   THERMAL CONDUCTIVITY OF WALL #3                             HS
12    1.3D0      KHS4   THERMAL CONDUCTIVITY OF WALL #4                             HS
17    1.32D0     XHS1   THICKNESS OF WALL #1                                        HS
18    1.83D0     XHS2   THICKNESS OF WALL #2                                        HS
19    1.D0       XHS3   THICKNESS OF WALL #3                                        HS
20    1.07D0     XHS4   THICKNESS OF WALL #4                                        HS
25    0.D0       XLHSI1  INNER LINER THICKNESS FOR WALL #1                          HS
26    2.713D-2   XLHSI2  INNER LINER THICKNESS FOR WALL #2                          HS
27    0.D0       XLHSI3  INNER LINER THICKNESS FOR WALL #3                          HS
28    0.D0       XLHSI4  INNER LINER THICKNESS FOR WALL #4                          HS
33    0.D0       XLHSO1  OUTER LINER THICKNESS FOR WALL #1                          HS
34    0.D0       XLHSO2  OUTER LINER THICKNESS FOR WALL #2                          HS
35    0.D0       XLHSO3  OUTER LINER THICKNESS FOR WALL #3                          HS
36    0.D0       XLHSO4  OUTER LINER THICKNESS FOR WALL #4                          HS
41    2300.D0    DHS1    DENSITY OF WALL #1                                         HS
42    2300.D0    DHS2    DENSITY OF WALL #2                                         HS
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43   2300.D0     DHS3     DENSITY OF WALL #3                                      HS
44   2300.D0     DHS4     DENSITY OF WALL #4                                      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
57   0.D0        MEQPD    MASS OF EQUIPMENT IN PEDESTAL                           HS
58   1.9D5       MEQDW    MASS OF EQUIPMENT IN DRYWELL                            HS
59   5.5682D4    MEQWW    MASS OF EQUIPMENT IN WETWELL                            HS
62   0.D0        AEQPD    AREA OF EQUIPMENT IN PEDESTAL                           HS
63   1.5D3       AEQDW    AREA OF EQUIPMENT IN DRYWELL                            HS
64   20.D0       AEQWW    AREA OF EQUIPMENT IN WETWELL                            HS
67   50.D0       HTOUTW   HEAT TRANSFER COEFF. AT OUTER WALL                      HS
68   0.D0        RGAPI1   INNER LINER TO WALL GAP RESISTANCE #1                   HS
69   1.D0        RGAPI2   INNER LINER TO WALL GAP RESISTANCE #2                   HS
70   0.D0        RGAPI3   INNER LINER TO WALL GAP RESISTANCE #3                   HS
71   0.D0        RGAPI4   INNER LINER TO WALL GAP RESISTANCE #4                   HS
72   0.D0        RGAPI5   INNER LINER TO WALL GAP RESISTANCE #5                   HS
73   0.D0        RGAPI6   INNER LINER TO WALL GAP RESISTANCE #6                   HS
74   0.D0        RGAPI7   INNER LINER TO WALL GAP RESISTANCE #7                   HS
75   0.D0        RGAPI8   INNER LINER TO WALL GAP RESISTANCE #8                   HS
76   0.D0        RGAPO1   OUTER LINER TO WALL GAP RESISTANCE #1                   HS
77   0.D0        RGAPO2   OUTER LINER TO WALL GAP RESISTANCE #2                   HS
78   0.D0        RGAPO3   OUTER LINER TO WALL GAP RESISTANCE #3                   HS
79   0.D0        RGAPO4   OUTER LINER TO WALL GAP RESISTANCE #4                   HS
80   0.D0        RGAPO5   OUTER LINER TO WALL GAP RESISTANCE #5                   HS
81   0.D0        RGAPO6   OUTER LINER TO WALL GAP RESISTANCE #6                   HS
82   0.D0        RGAPO7   OUTER LINER TO WALL GAP RESISTANCE #7                   HS
83   0.D0        RGAPO8   OUTER LINER TO WALL GAP RESISTANCE #8                   HS
84   5.3232D4    MEQWWS   MASS EQUIP. WETWELL (SUBMERGED)                         HS
85   1.D2        AEQWWS   AREA EQUIP. WETWELL (SUBMERGED)                         HS
86   36.D0       AHSPS1   AREA OF RPV IN GAS SPACE OF VESSEL                      HS
87   44.D0       AHSPS2   AREA OF RPV IN DOWNCOMER REGION                         HS
88   16.D0       AHSPS3   AREA OF RPV IN LOWER PLENUM                             HS
89   2.642D5     MRPV1    RPV WALL MASS IN UPPER DOME REGION                      HS
90   2.0213D5    MRPV2    RPV WALL MASS IN DOWNCOMER REGION                       HS
91   1.5176D5    MRPV3    RPV WALL MASS IN LOWER PLENUM REGION                    HS
**                                                                            HS
**                                                                            HS
*MODEL PARAMETERS FOR BWR                                                     HS
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       XCMREF   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
17   983.D0      TCFLAM   CRITICAL FLAME TEMPERATURE                              MO
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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   .90D0      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                                                        MO
01   56.D0      MOLWCN MOLECULAR WEIGHT OF CONCRETE                         CN
02   1743.D0    TCNMP  CONCRETE MELTING TEMPERATURE                         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 FOR CONCRETE MELTING                     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   412.7      MFP(1) MASS OF FISSION PRODUCT GROUP 1 -NOBLES              FI
07   223.6      MFP(2) MASS OF FISSION PRODUCT GROUP 2 -CS+I                FI
08   34.9       MFP(3) MASS OF FISSION PRODUCT GROUP 3 -TE                  FI
09   167.7      MFP(4) MASS OF FISSION PRODUCT GROUP 4 -SR                  FI
10   409.0      MFP(5) MASS OF FISSION PRODUCT GROUP 5 -RU                  FI
11   1050.0     MSMO(1) MASS OF SN IN CORE REGION                           FI
12   432.0      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    FDFSP(1) SRV DECON. FACTOR FOR FP GROUP 1                   FI
24   1000.D0    FDFSP(2) SRV DECON. FACTOR FOR FP GROUP 2                   FI
25   1000.0D0   FDFSP(3) SRV DECON. FACTOR FOR FP GROUP 3                   FI
26   1000.0D0   FDFSP(4) SRV DECON. FACTOR FOR FP GROUP 4                   FI
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27   1000.0D0  FDFSP(5) SRV DECON. FACTOR FOR FP GROUP 5                    FI
**                                                                          FI
**                                                                          FI
*CONTROL CARDS                                                              CC
01   1         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 NUMBER (TOTAL OF 4 FILES)             CC
06   600       IPTSMX MAXIMUM NUMBER OF PLOTTED POINTS                      CC
07   6         IPTSPK MAXIMUM NUMBER OF PLOT POINTS TRACED FOR FULL         CC
**                    SCALE SPIKE                                           CC
08   150       IPTSAV NUMBER OF POINTS SAVED FOR VARIABLE 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   0         IRET   WRITE RETAIN PLOT FILE                                CC
15   49        IFPPLT RETAIN PLOT FILE UNIT NUMBER                          CC
36   0         IFPRAT 1=NUREG-0772 FP RELEASES ; 0=CUBICCIOTTI              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    4.D-2    FMCHMX MAXIMUM MASS CHANGE FRACTION FOR INTEGRATION          TD
04    5.D-2    FUCHMX MAXIMUM ENERGY CHANGE FRACTION FOR INTEGRATION        TD
05    1.D10    MAXMST MAXIMUM MASS OF STEAM CHANGE PER TIME STEP IN PS      TD
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APPENDIX B

Supplemental Plots for the Base Accident Sequences

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SUPPLEMENTAL PLOTS FOR SEQUENCE TW

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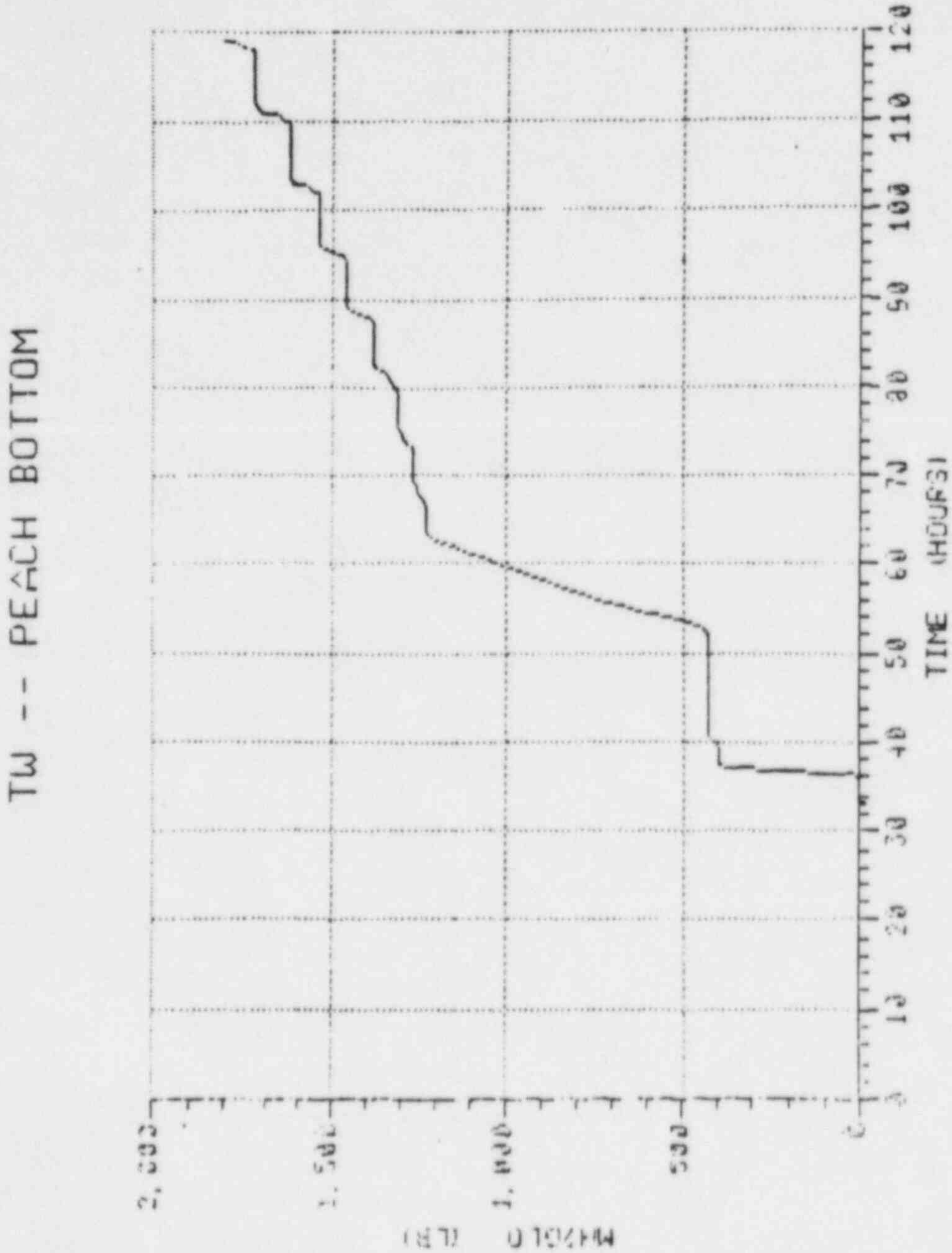

Fig. B.1 Total H_2 generated.

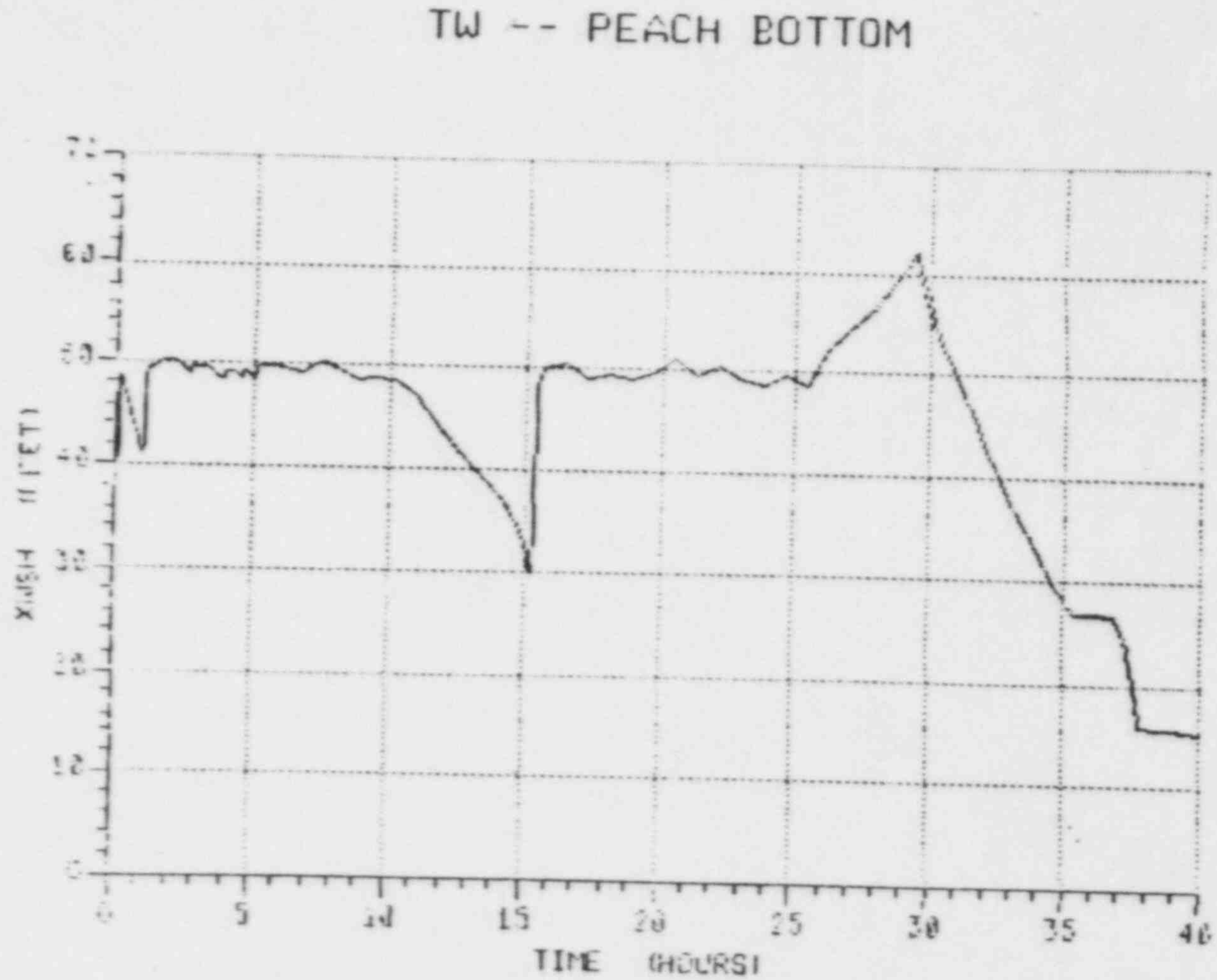

Fig. B.2 Reactor vessel water level.

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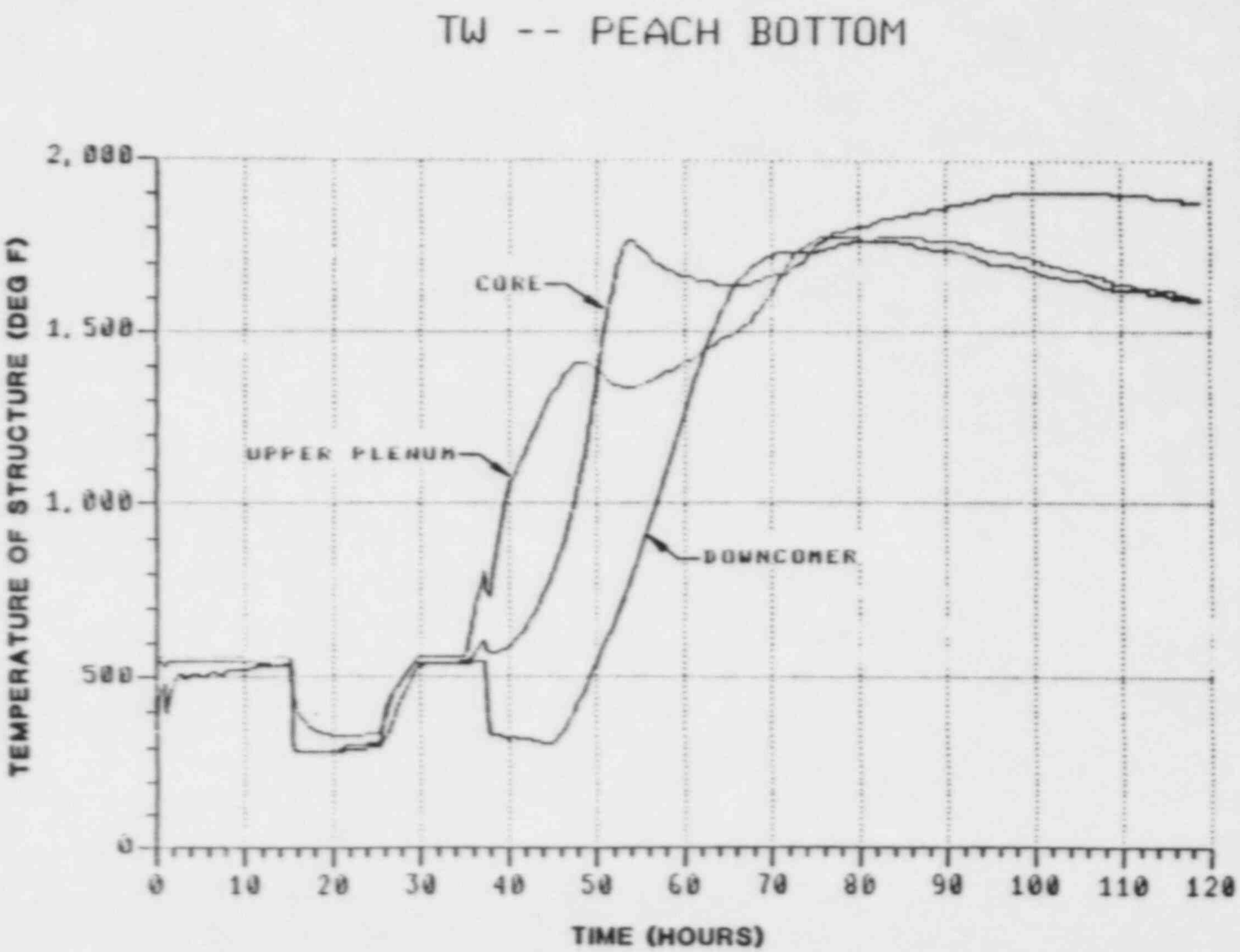

Fig. B.3 Temperature of structure, °F.

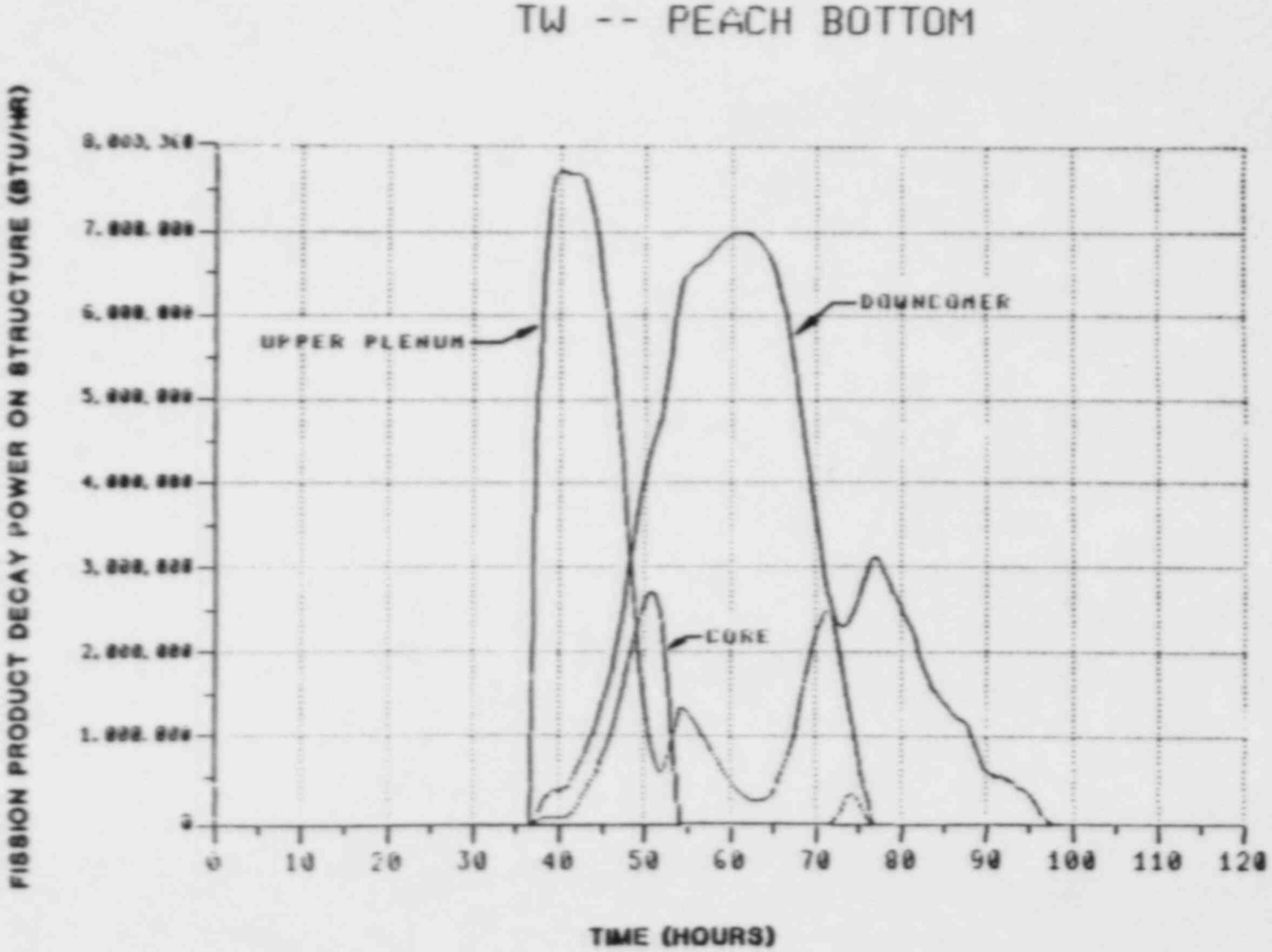

Fig. B.4 Fissian product decay power on structure, Btu/hr.

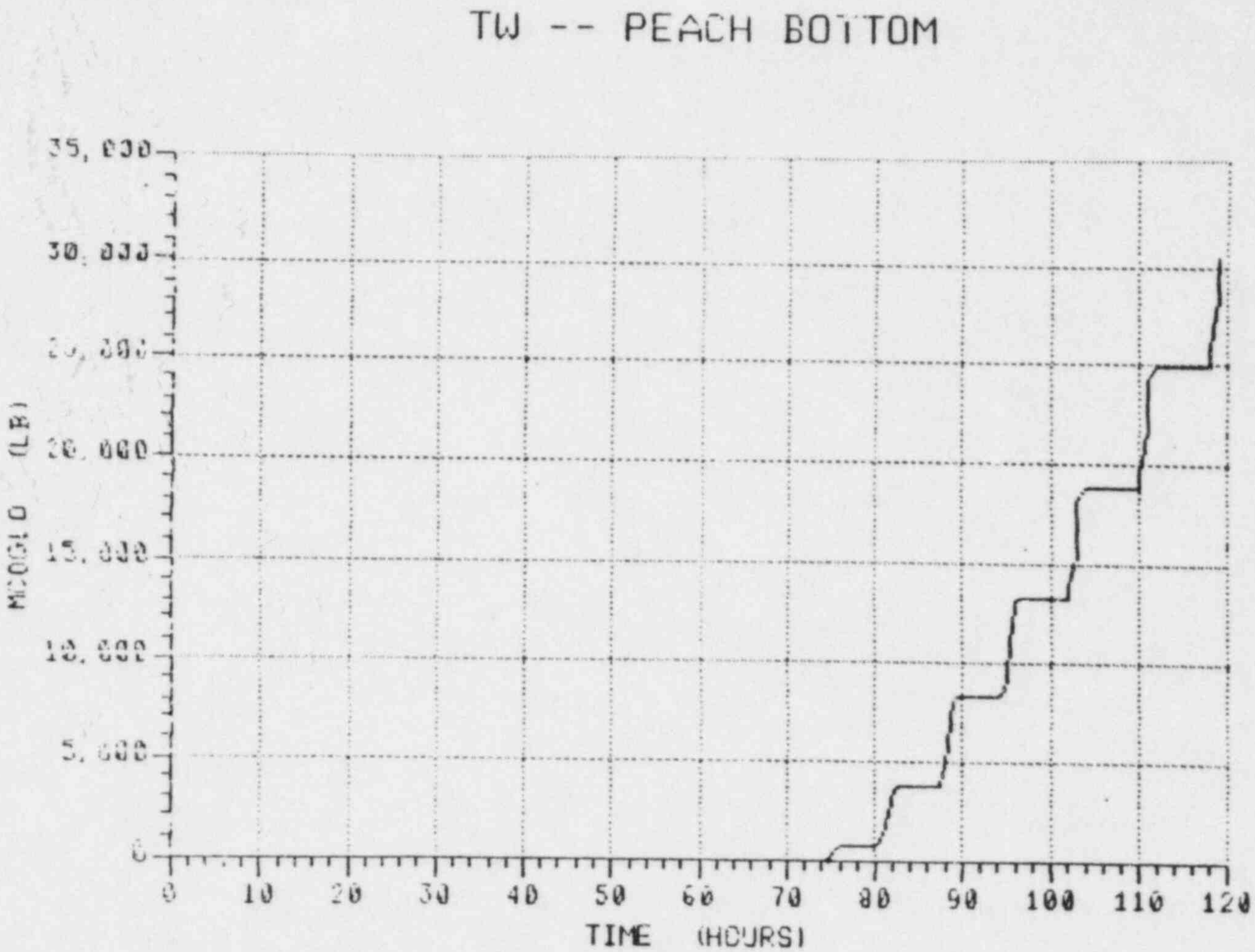

Fig. B.5 Total CO generated.

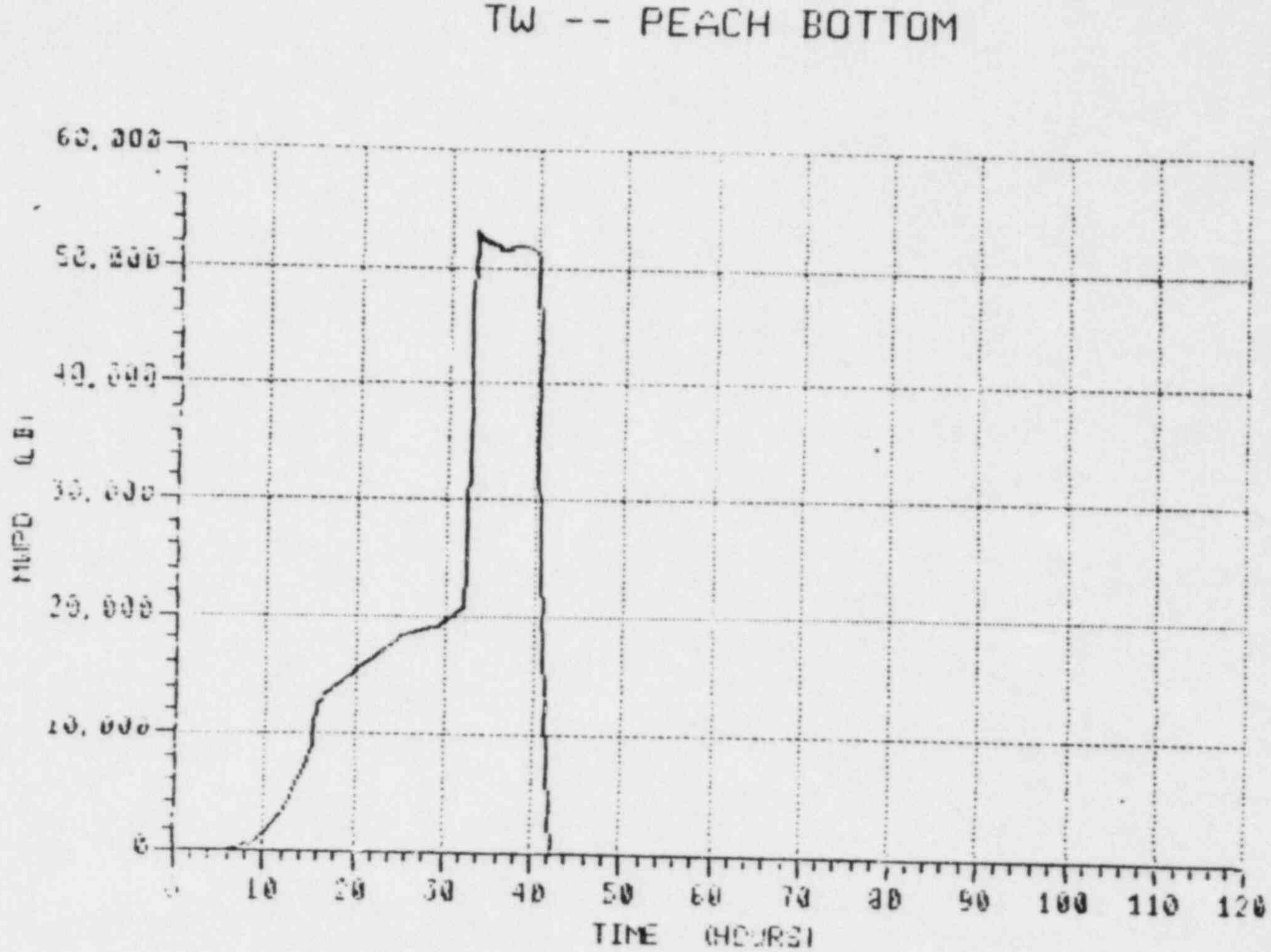

Fig. B.6 Mass of water in the pedestal.

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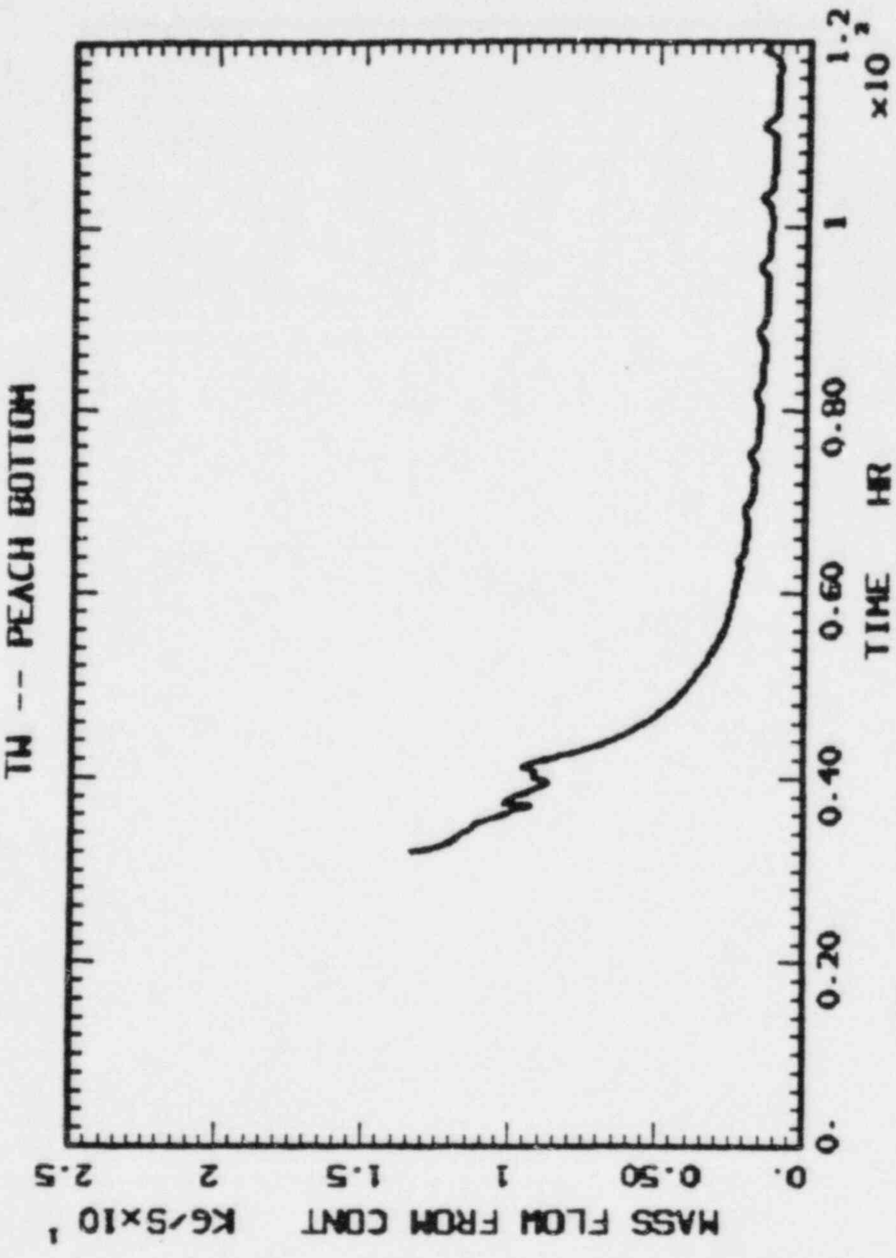

Fig. B.7 Mass flow out of containment.

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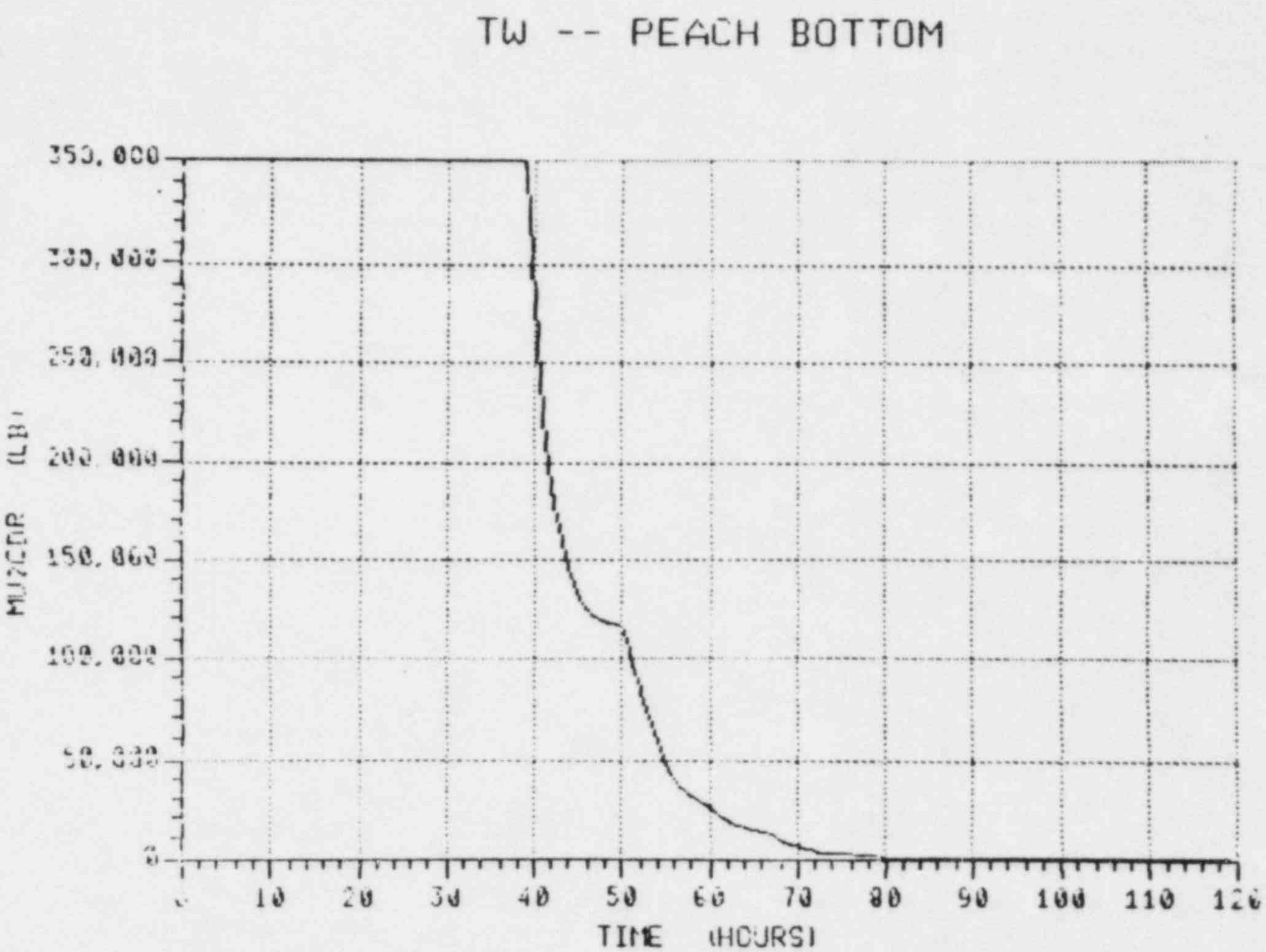

Fig. B.8 Mass of UO_2 in core region.

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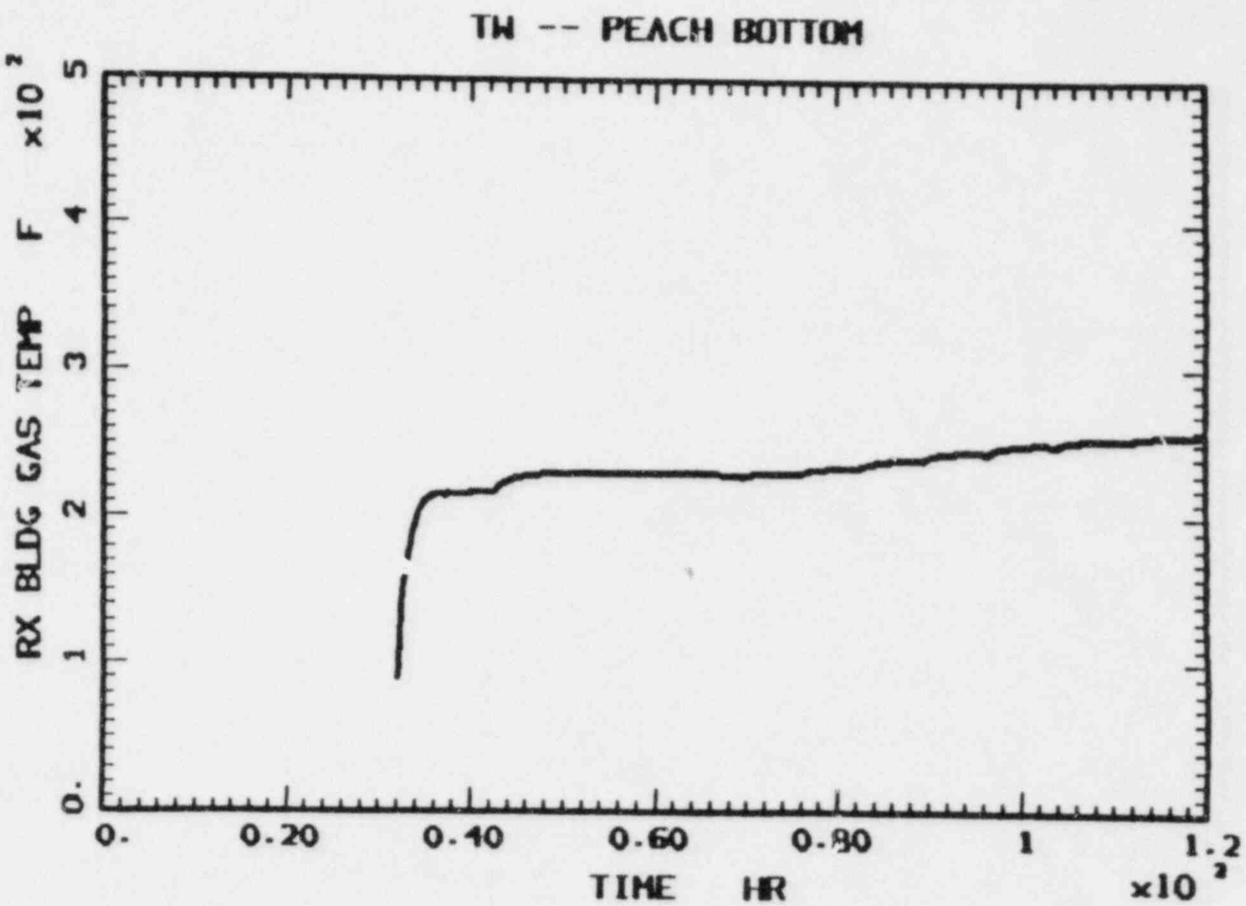

Fig. B.9 Gas temperature in reactor building, °F.

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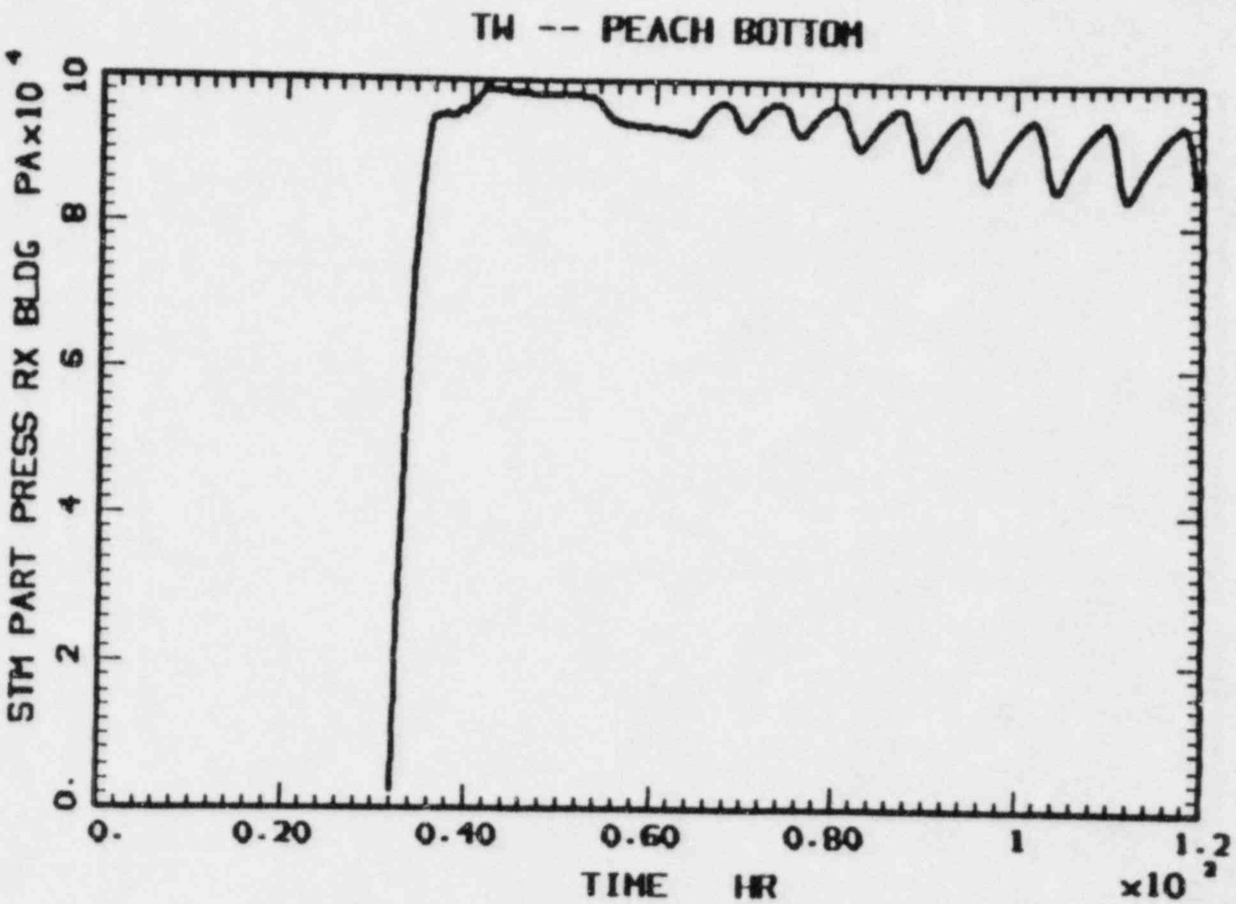

Fig. B.10 Steam pressure in reactor building, Pa.

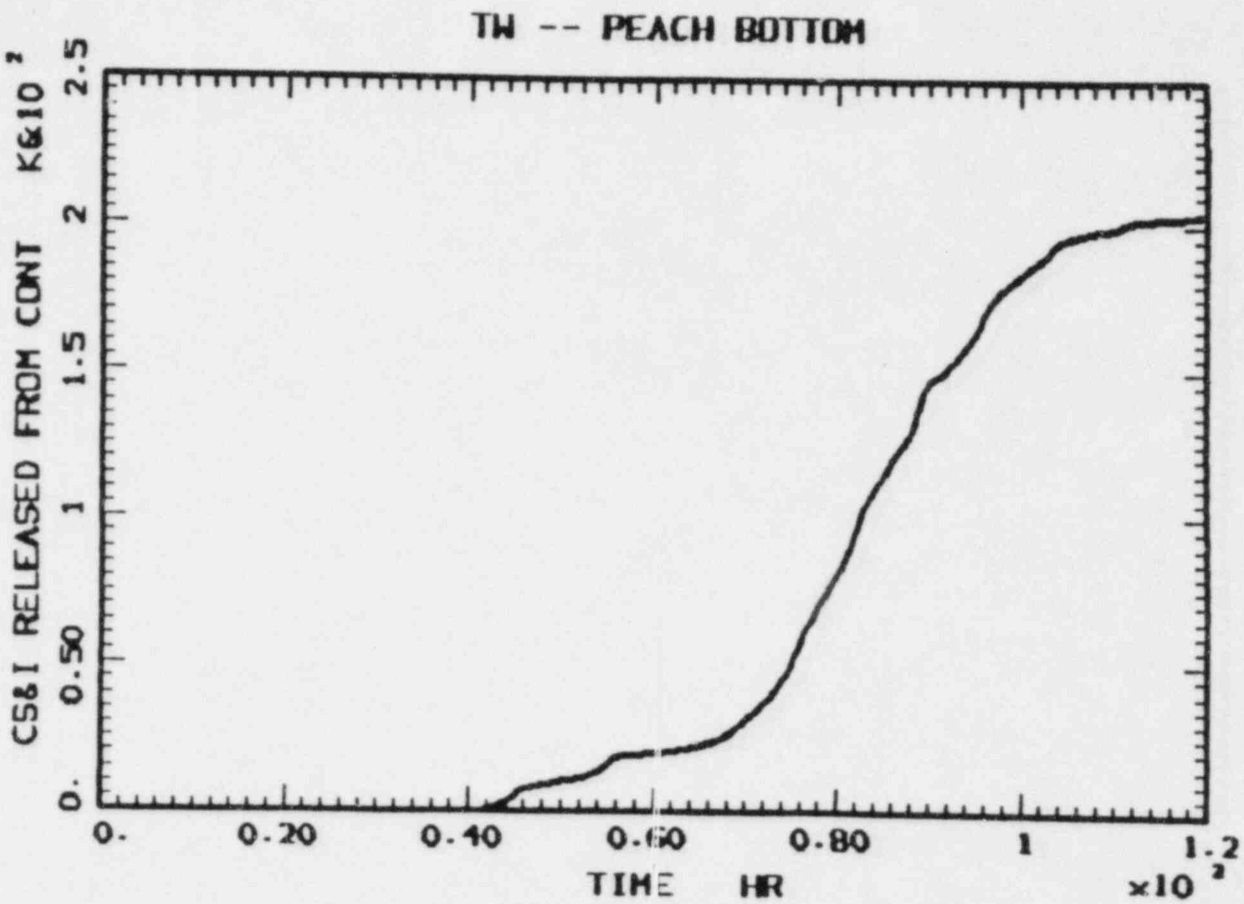

Fig. B.11 Cesium and iodine released from containment, kg.

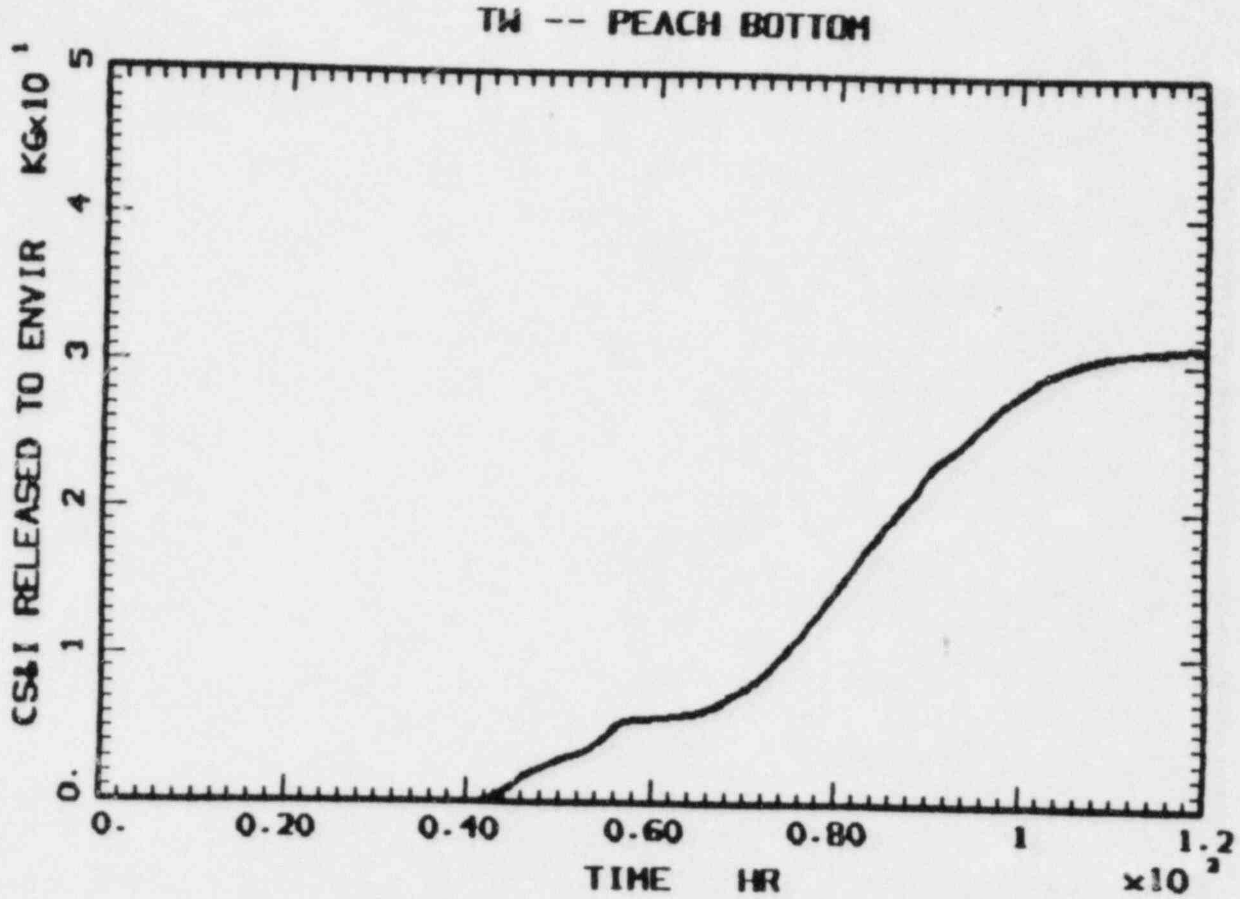

Fig. B.12 Cesium and iodine released to environment, kg.

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SUPPLEMENTAL PLOTS FOR SEQUENCE TC

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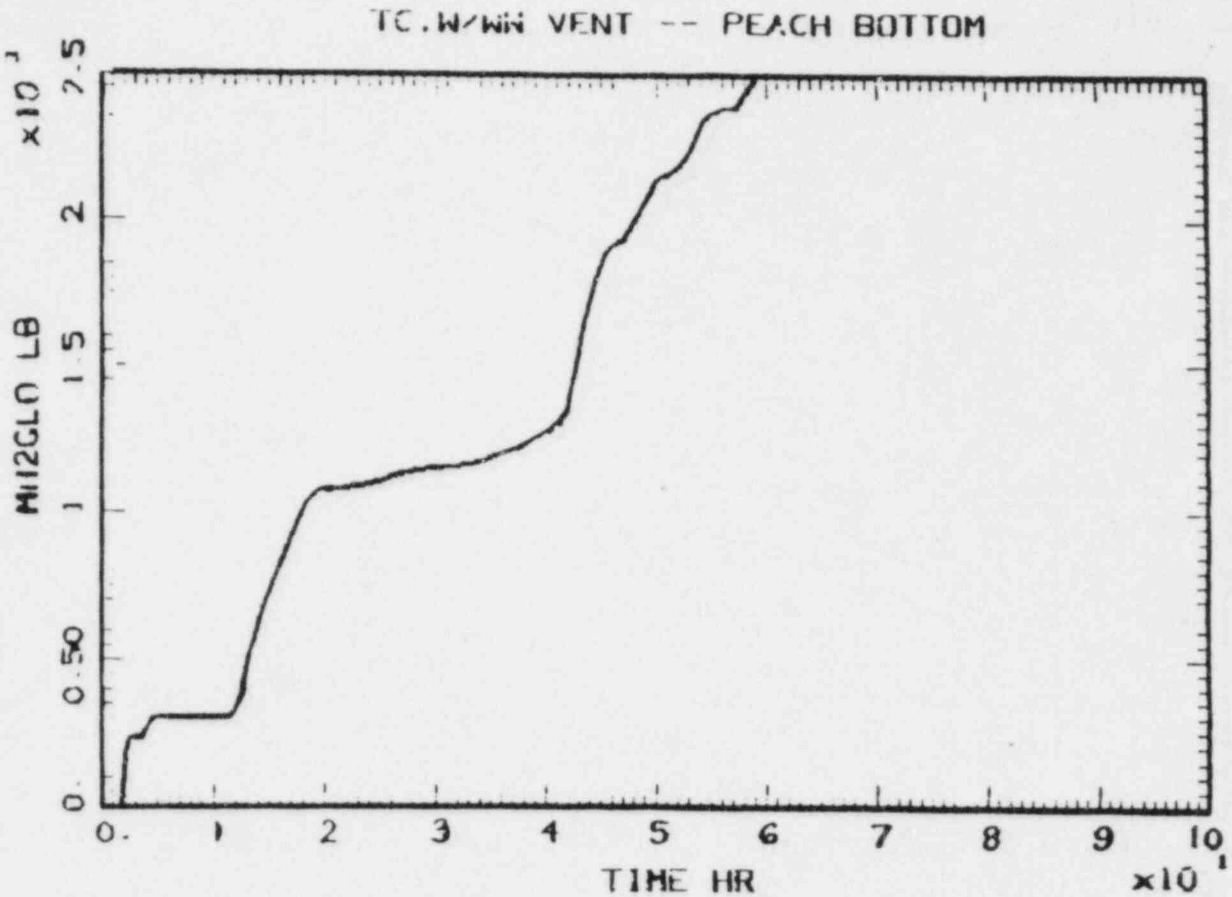

Fig. B.13 Total H_2 generated.

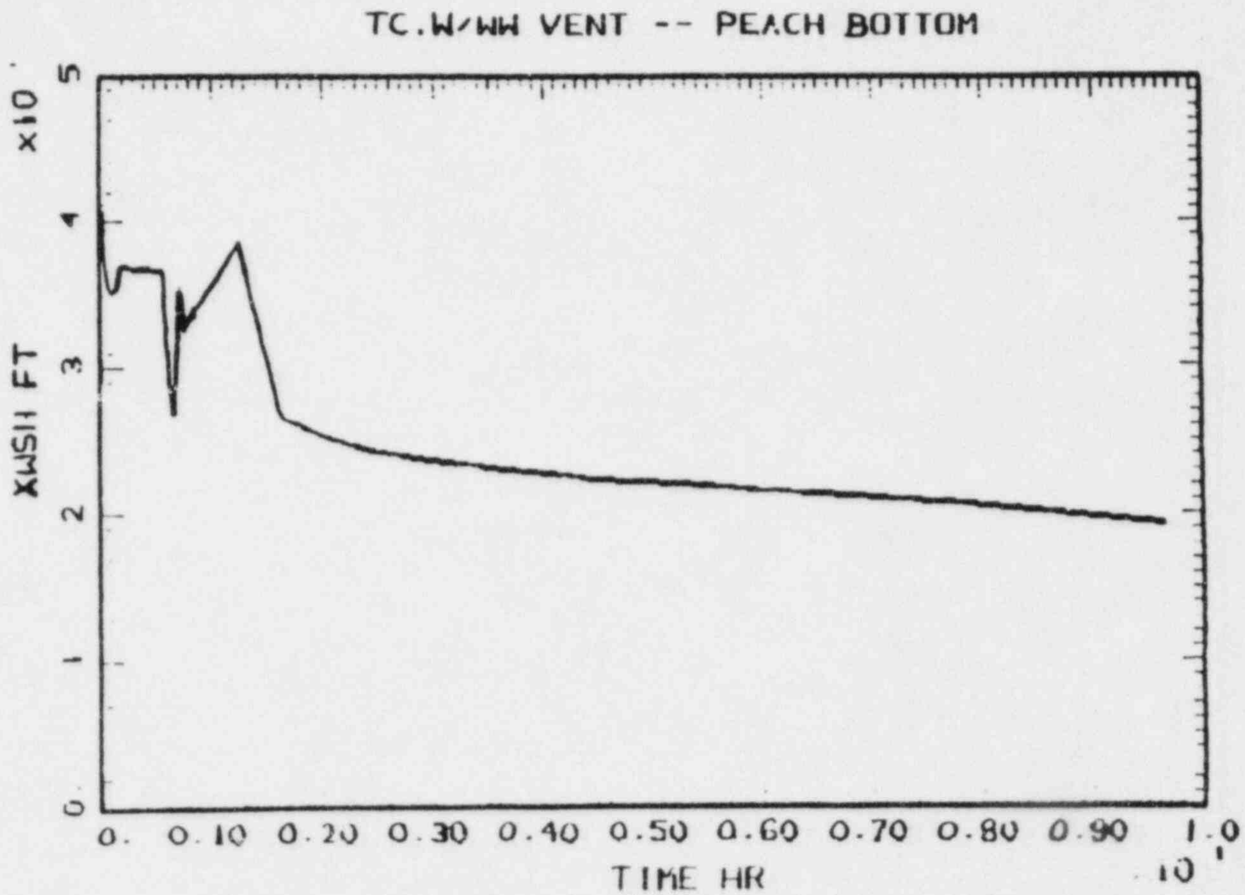

Fig. B.14 Reactor vessel water level.

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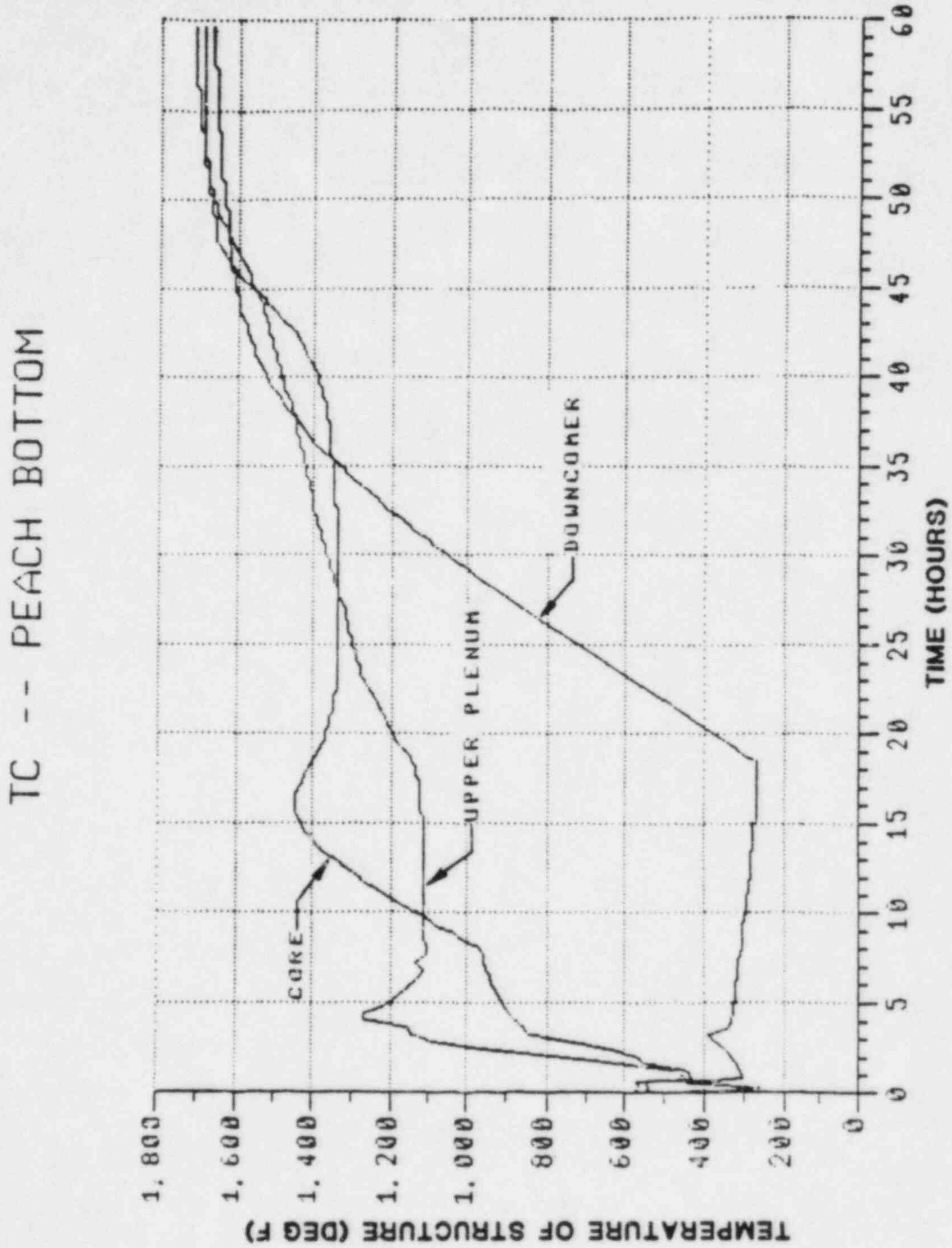

Fig. B.15 Temperature of structure, °F.

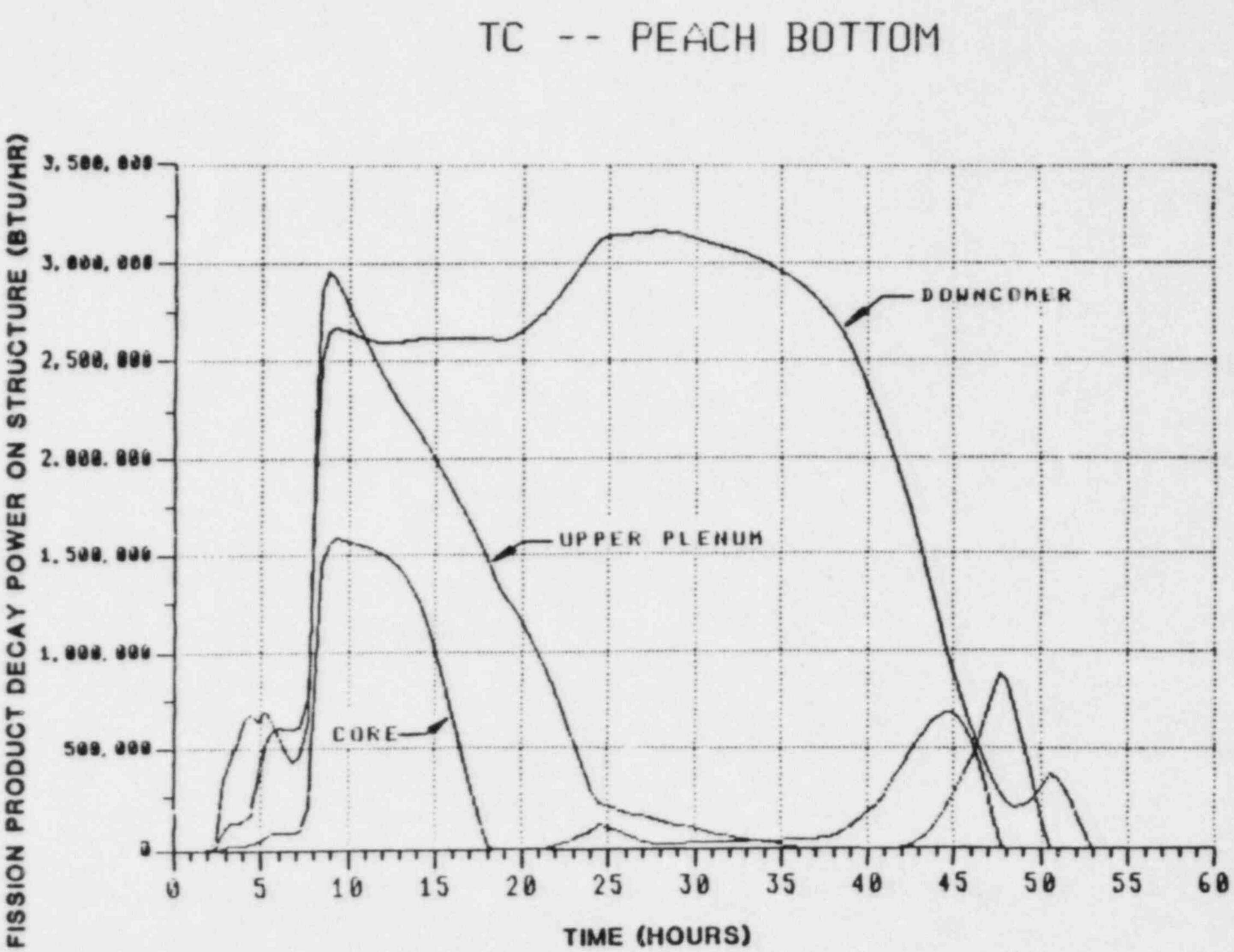

Fig. B.16 Fission product decay power on structure, Btu/hr.

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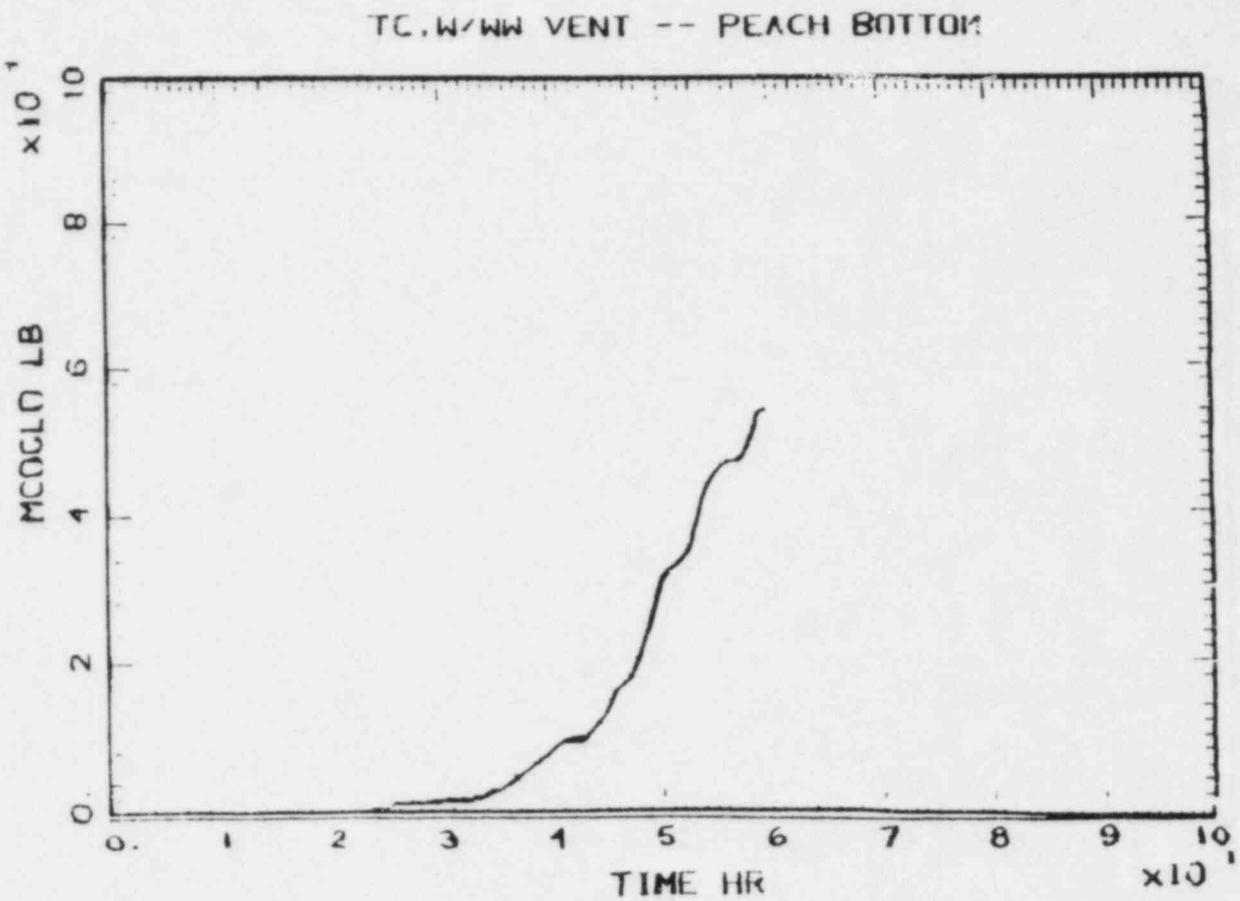

Fig. B.17 Total CO generated.

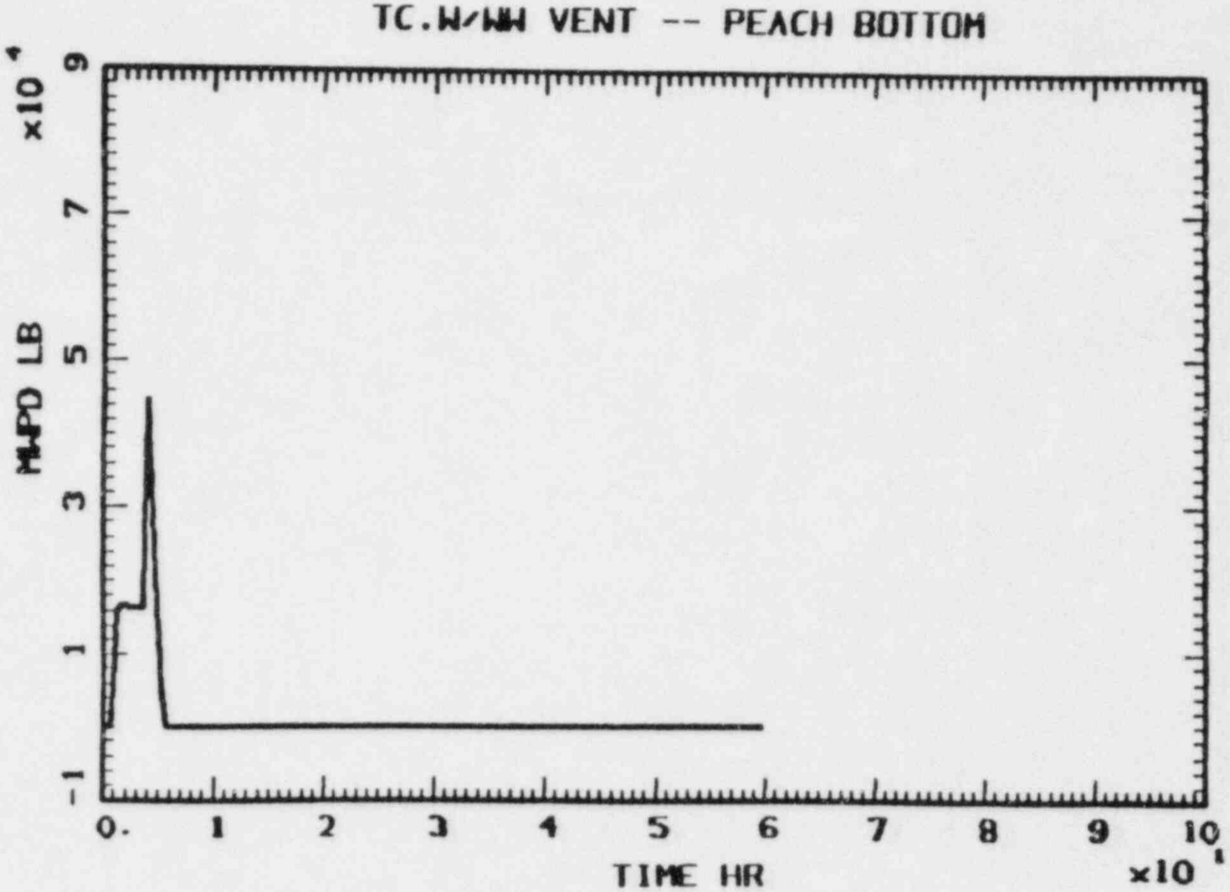

Fig. B.18 Mass of water in the pedestal.

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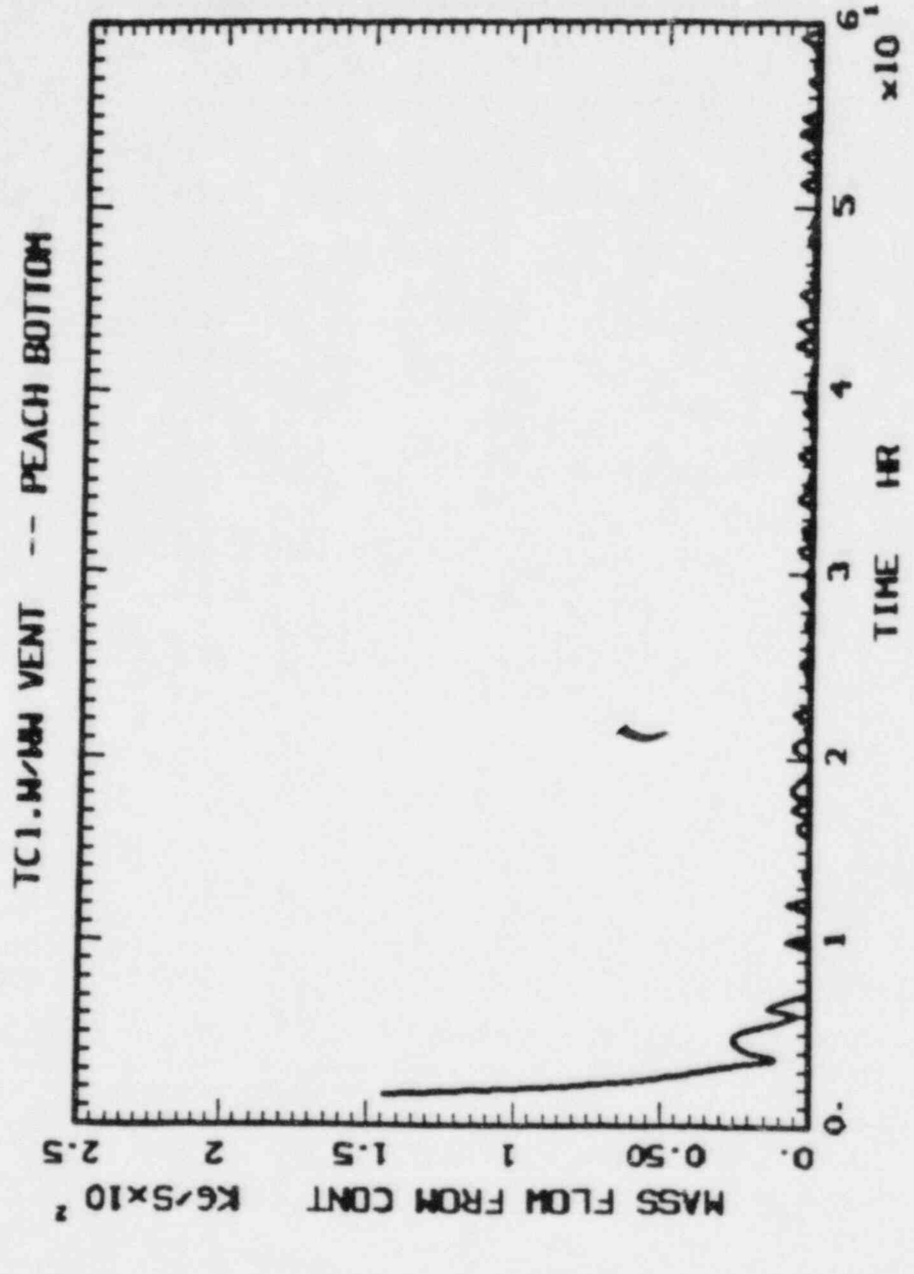

Fig. B.19 Mass flow out of containment.

DRAFT

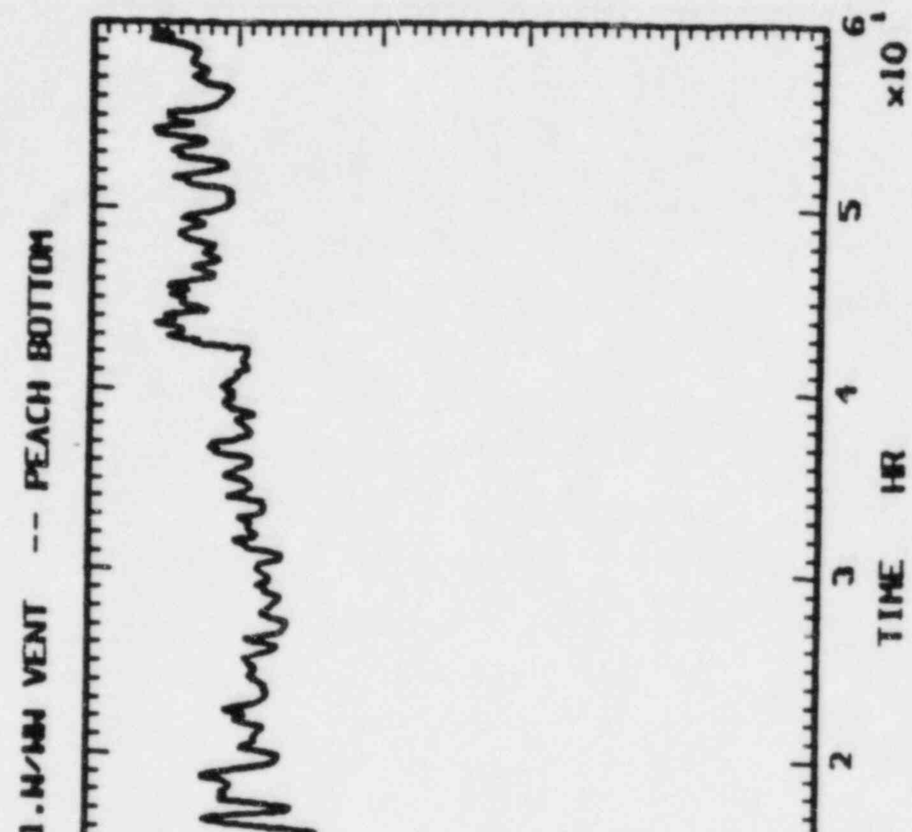

Fig. B.20 Reactor building gas temperature, °F.

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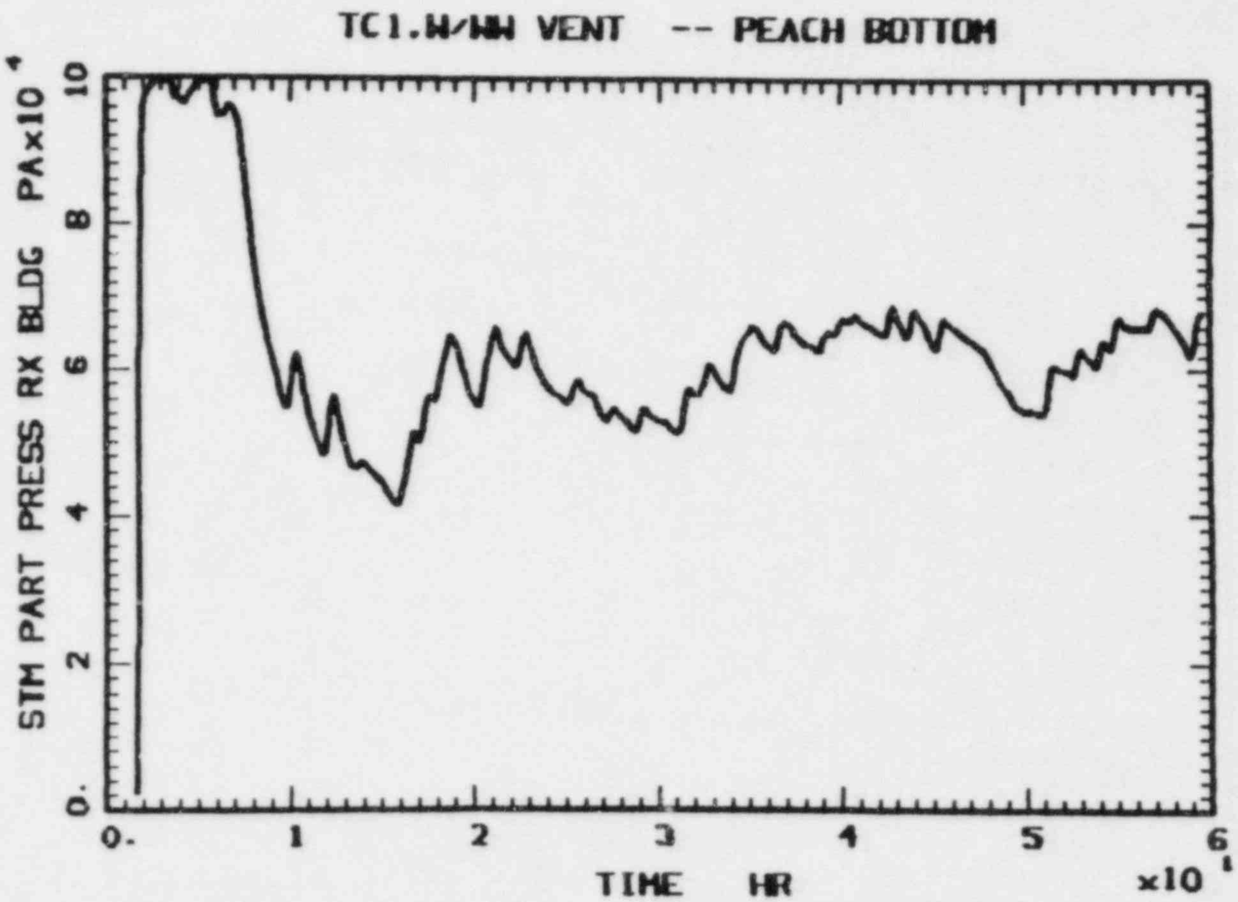

Fig. B.21 Reactor building steam partial pressure.

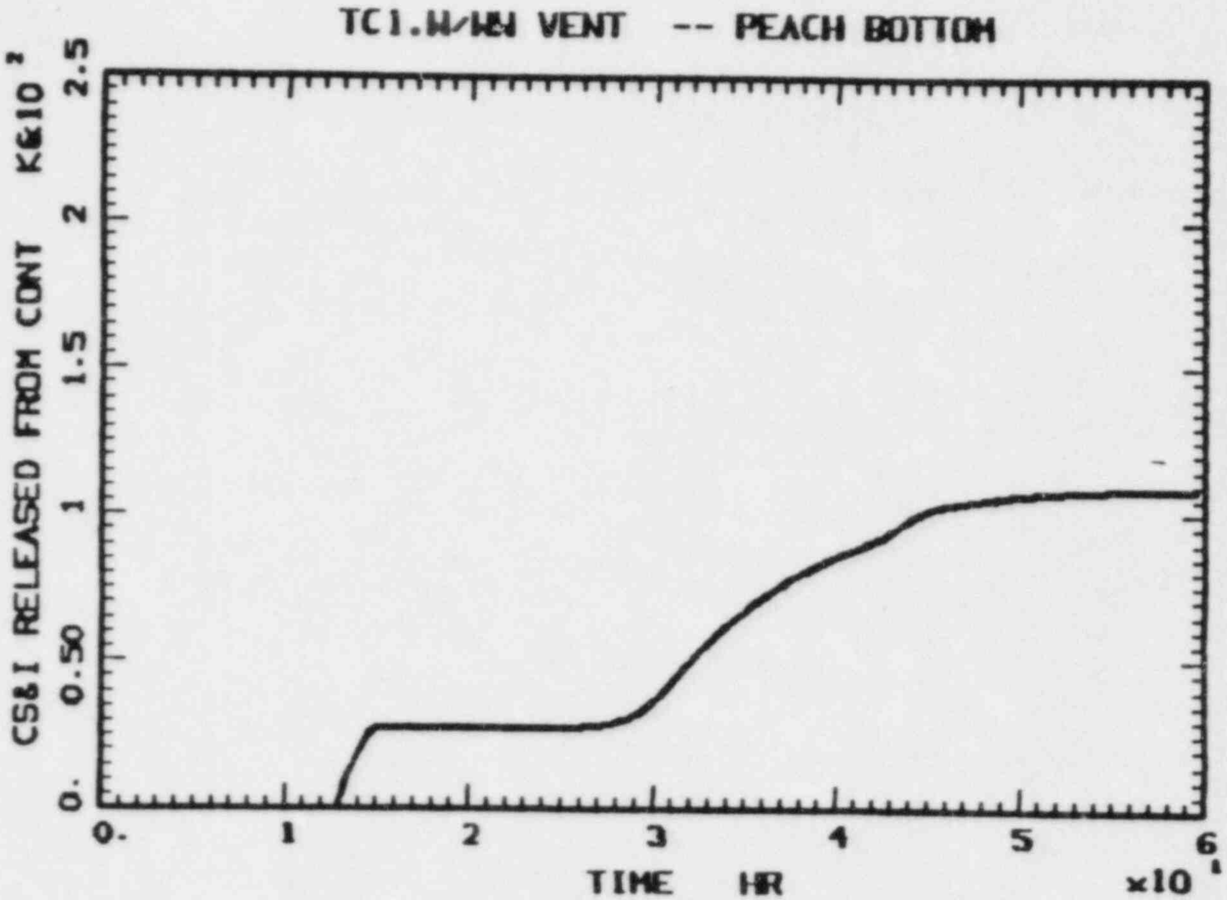

Fig. B.22 Cesium and iodine released from containment, kg.

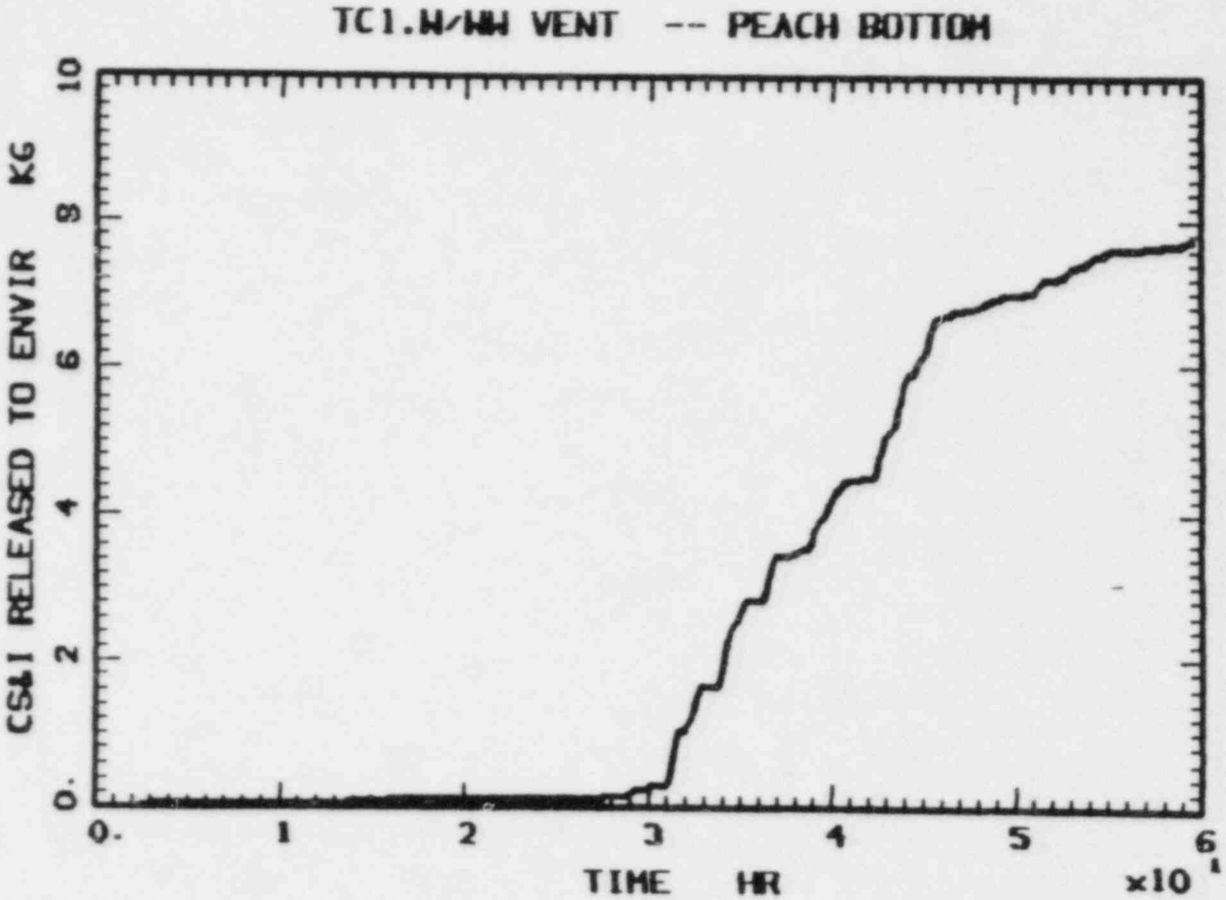

Fig. B.23 Mass of cesium and iodine released to environment.

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SUPPLEMENTAL PLOTS FOR SEQUENCE S_1E

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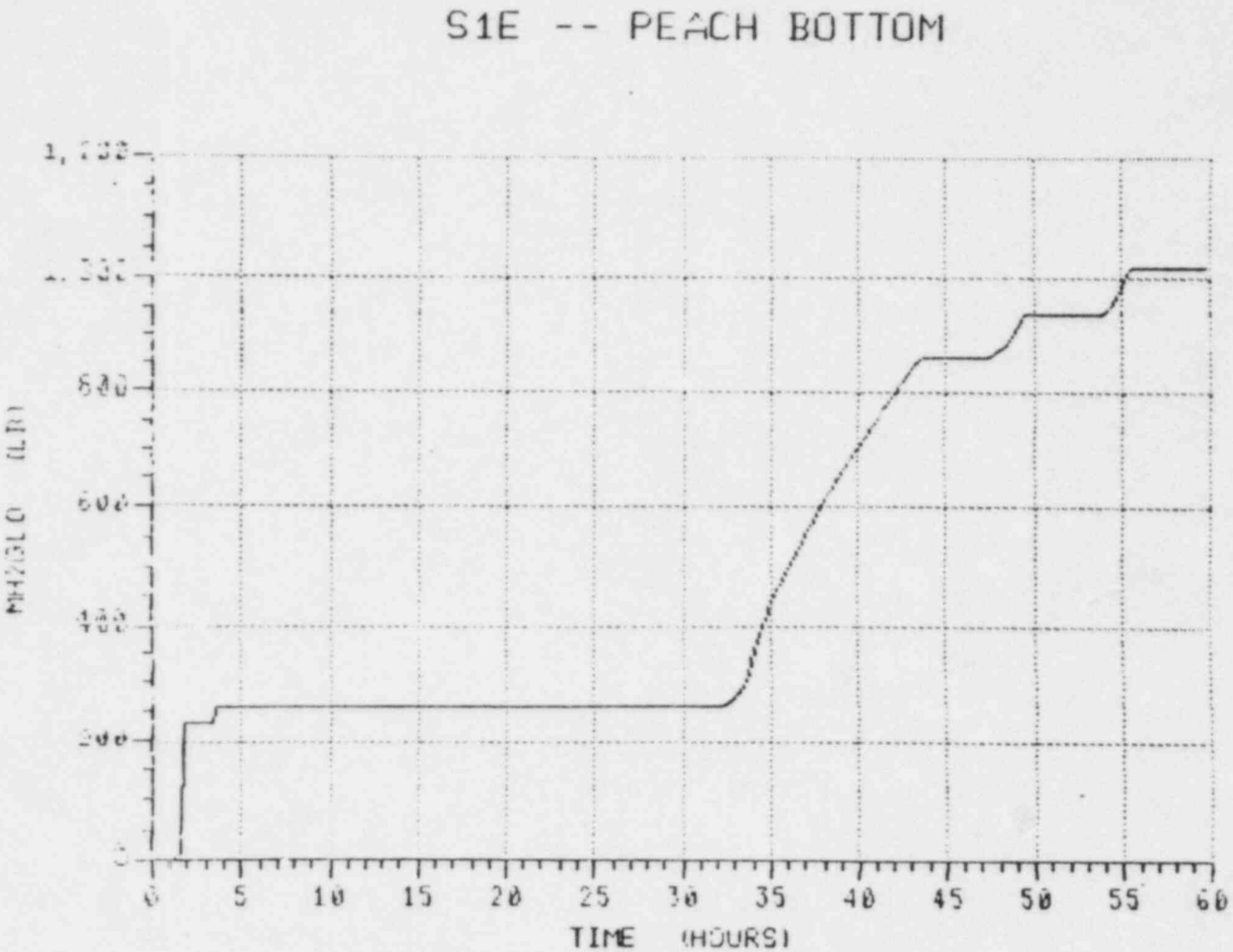

Fig. B.24 Total H_2 generated.

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S1E -- PEACH BOTTOM

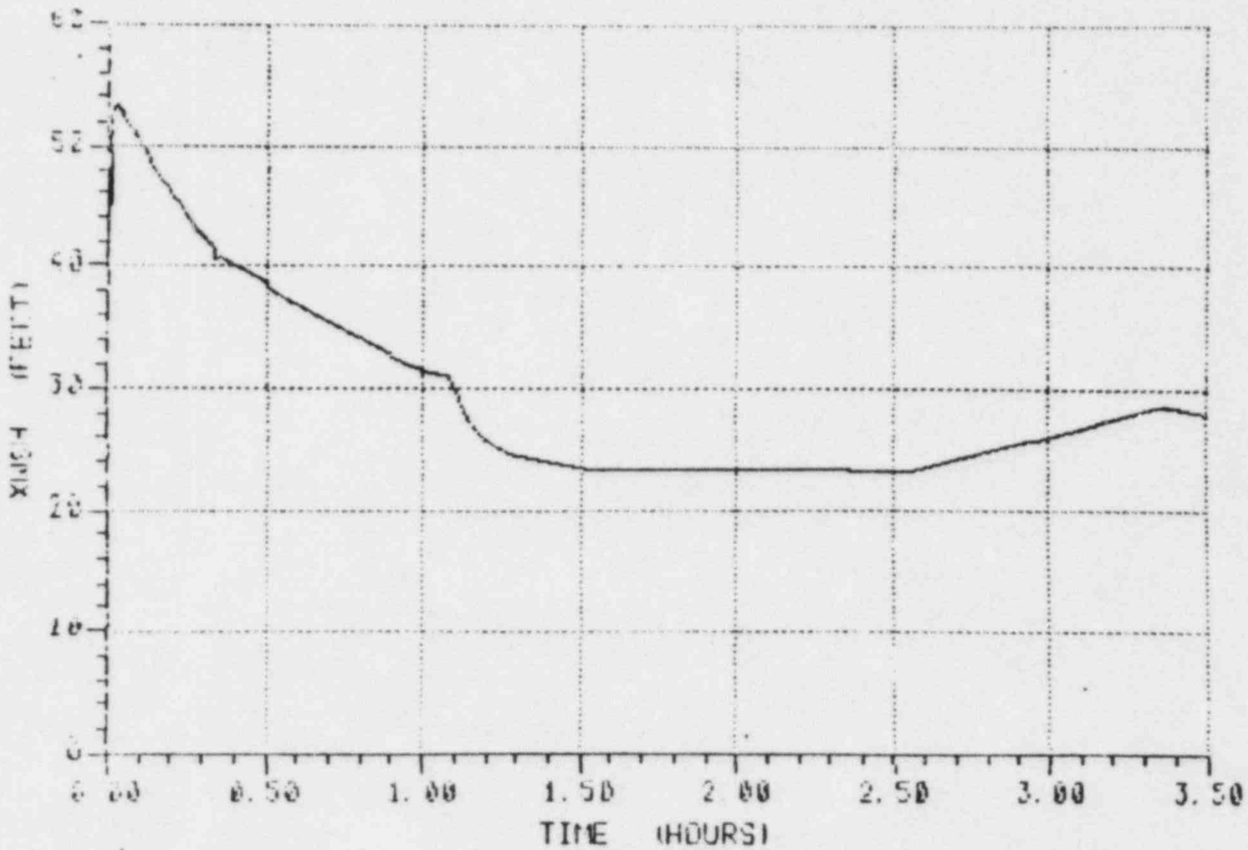

Fig. B.25 Reactor vessel water level.

DRAFT

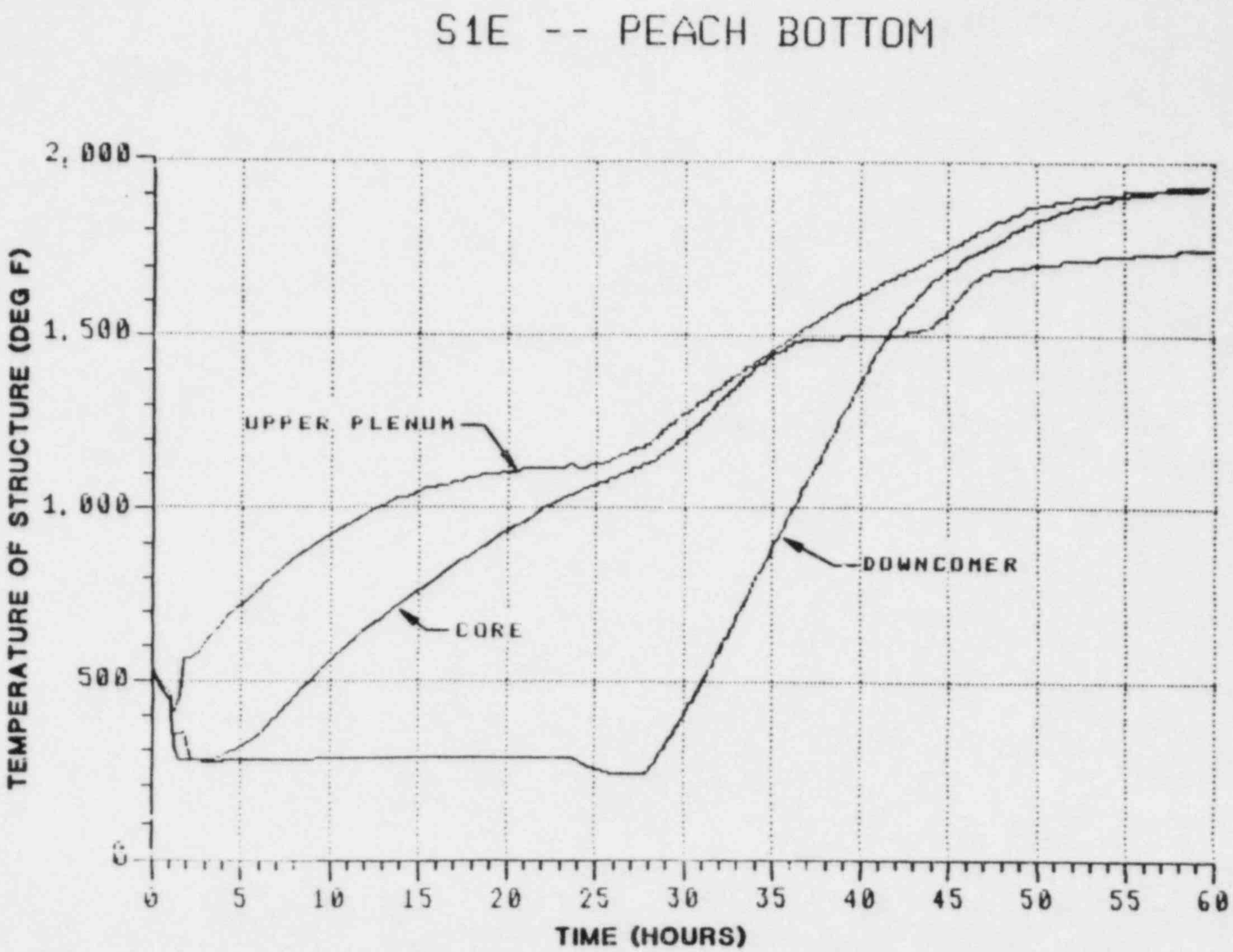

Fig. B.26 Temperature of structure, °F.

DRAFT

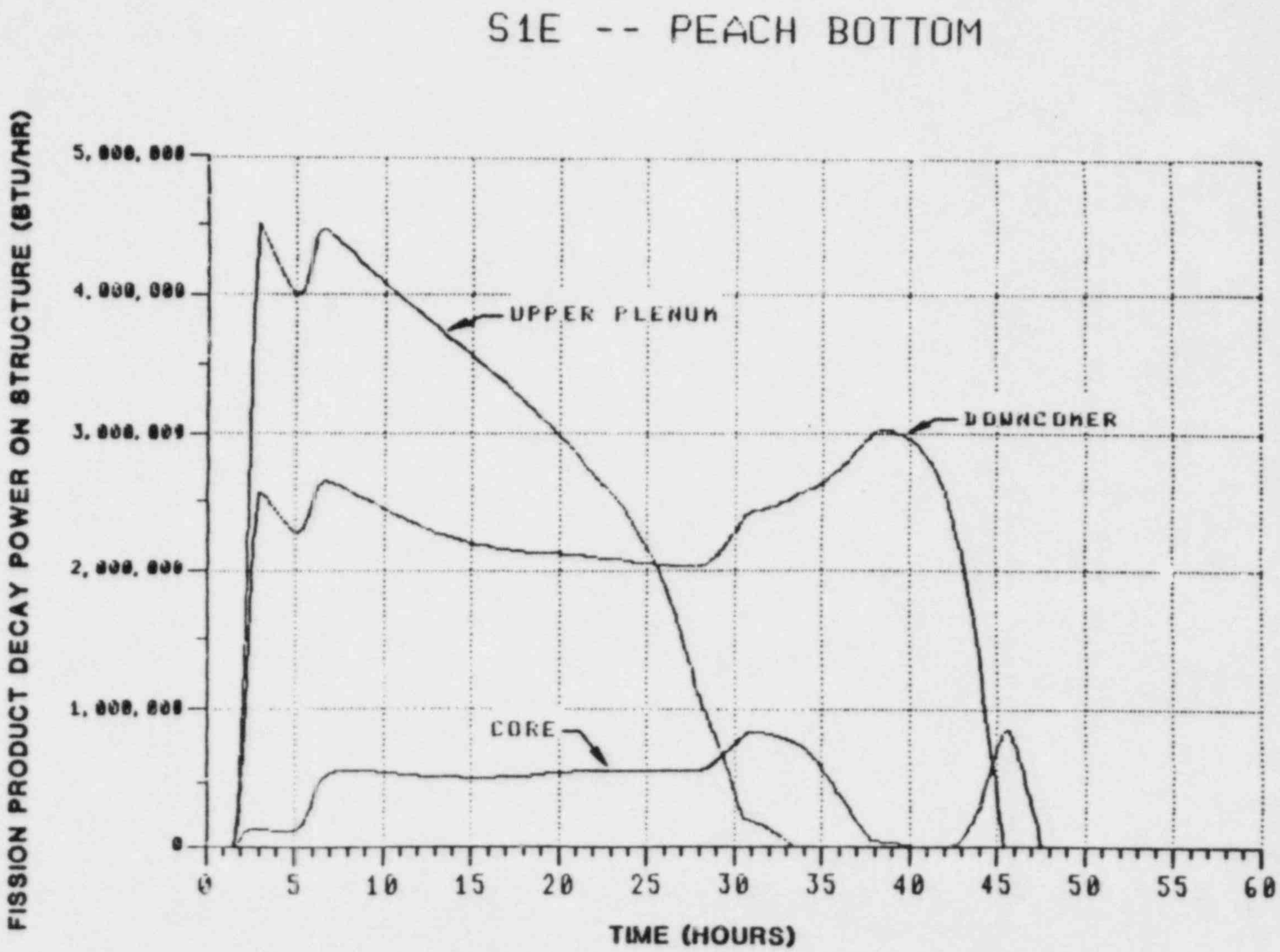

Fig. B.27 Fission product decay power on structure, Btu/hr.

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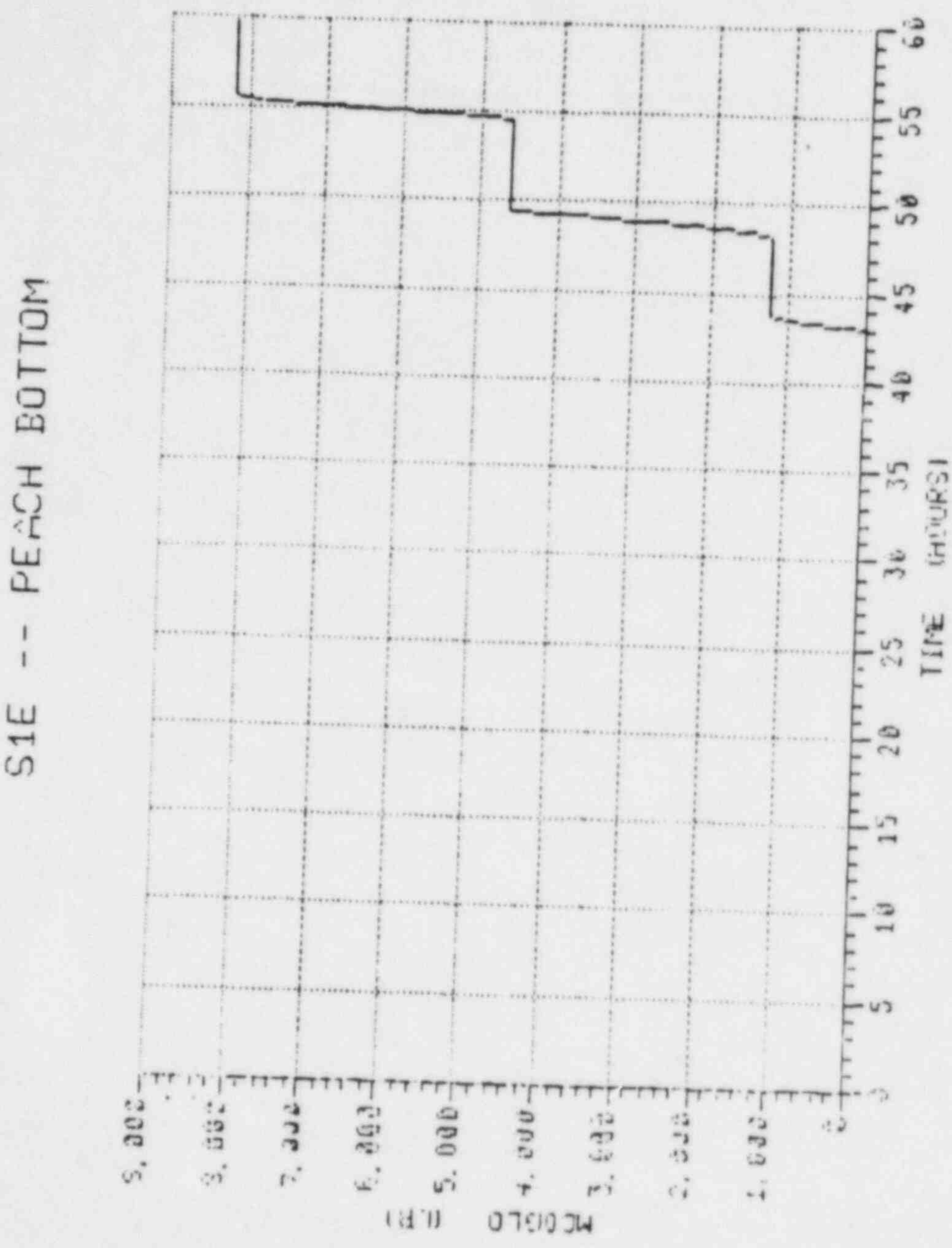

Fig. B.28 Total CO generated.

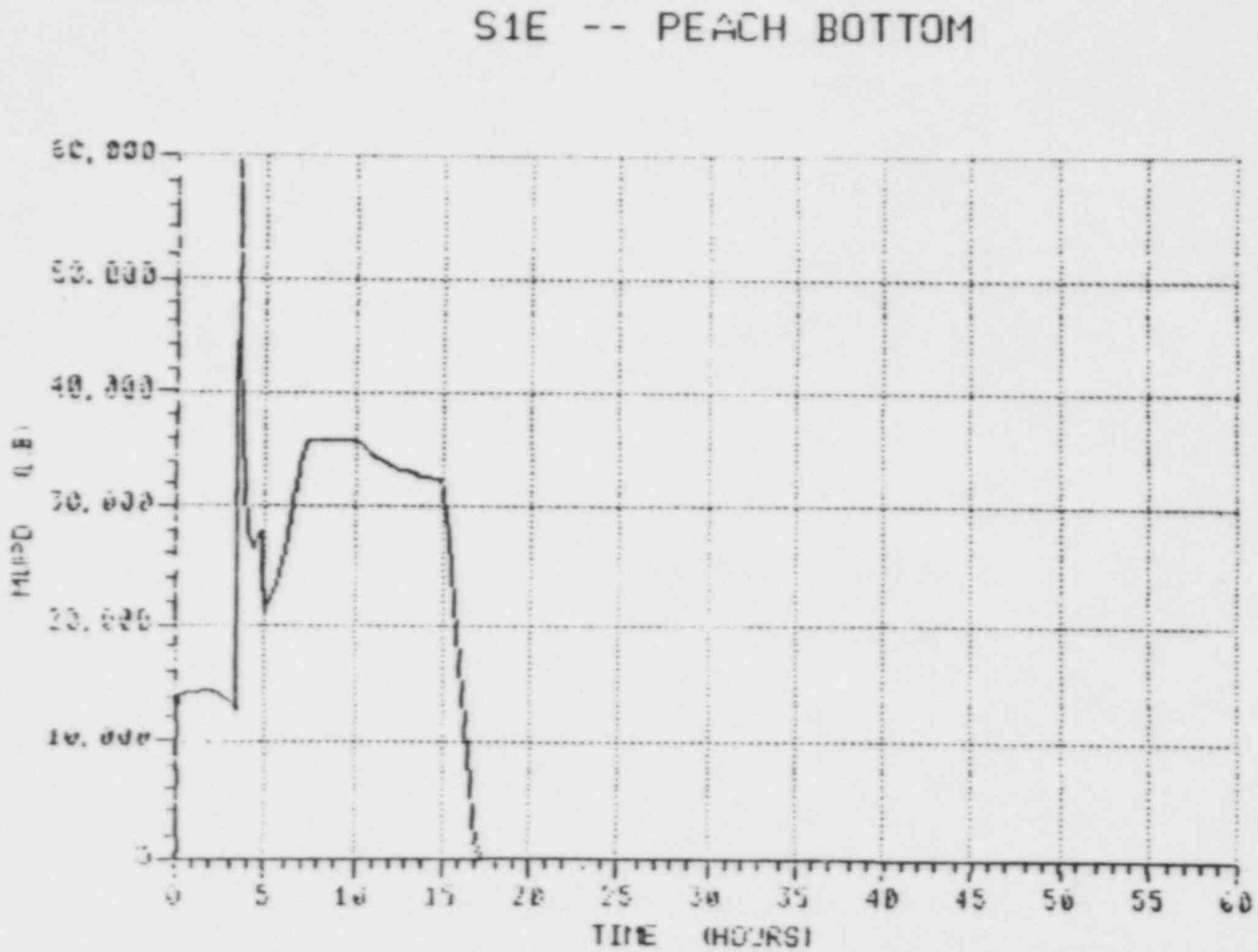

Fig. B.29 Mass of water in the pedestal.

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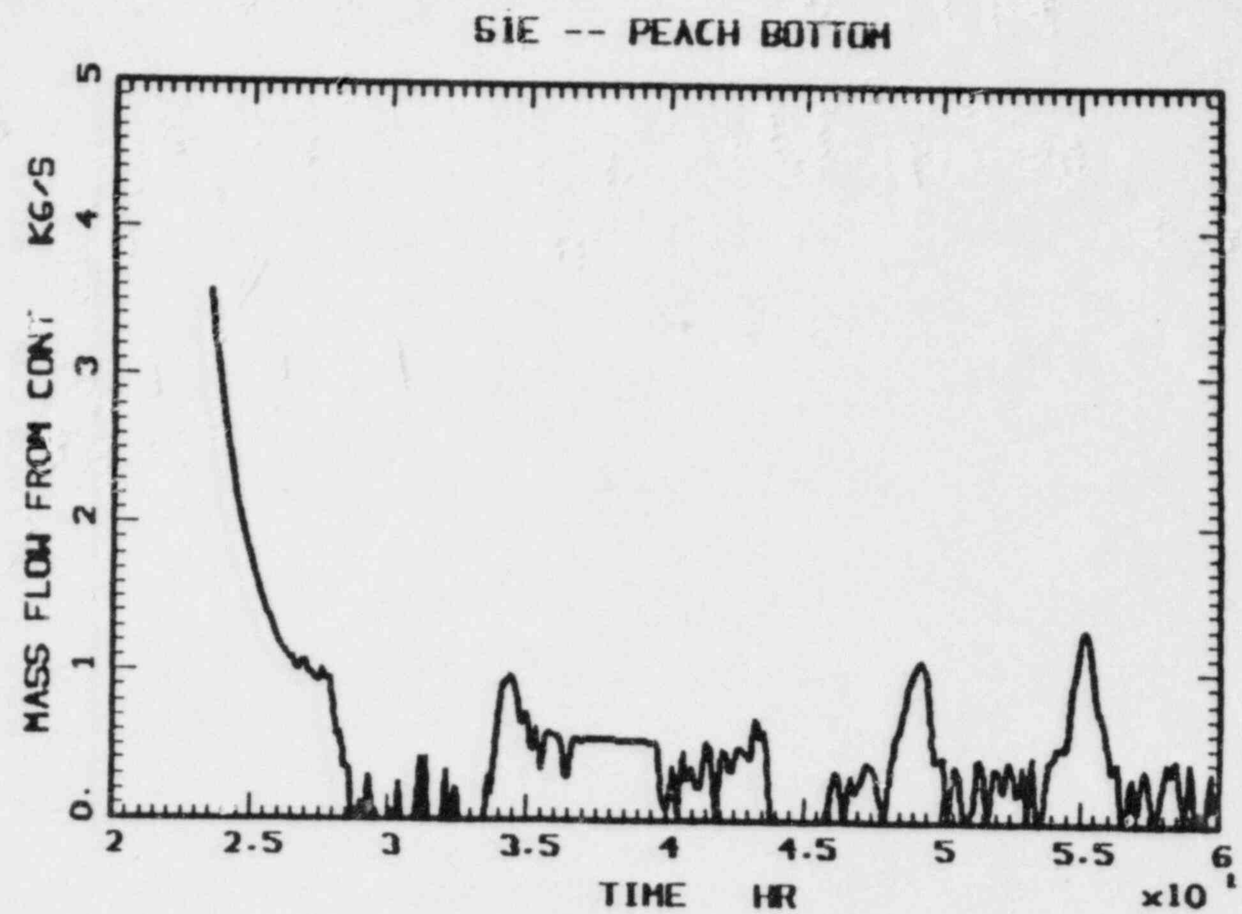

Fig. B.30 Mass flow out of containment.

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S1E -- PEACH BOTTOM

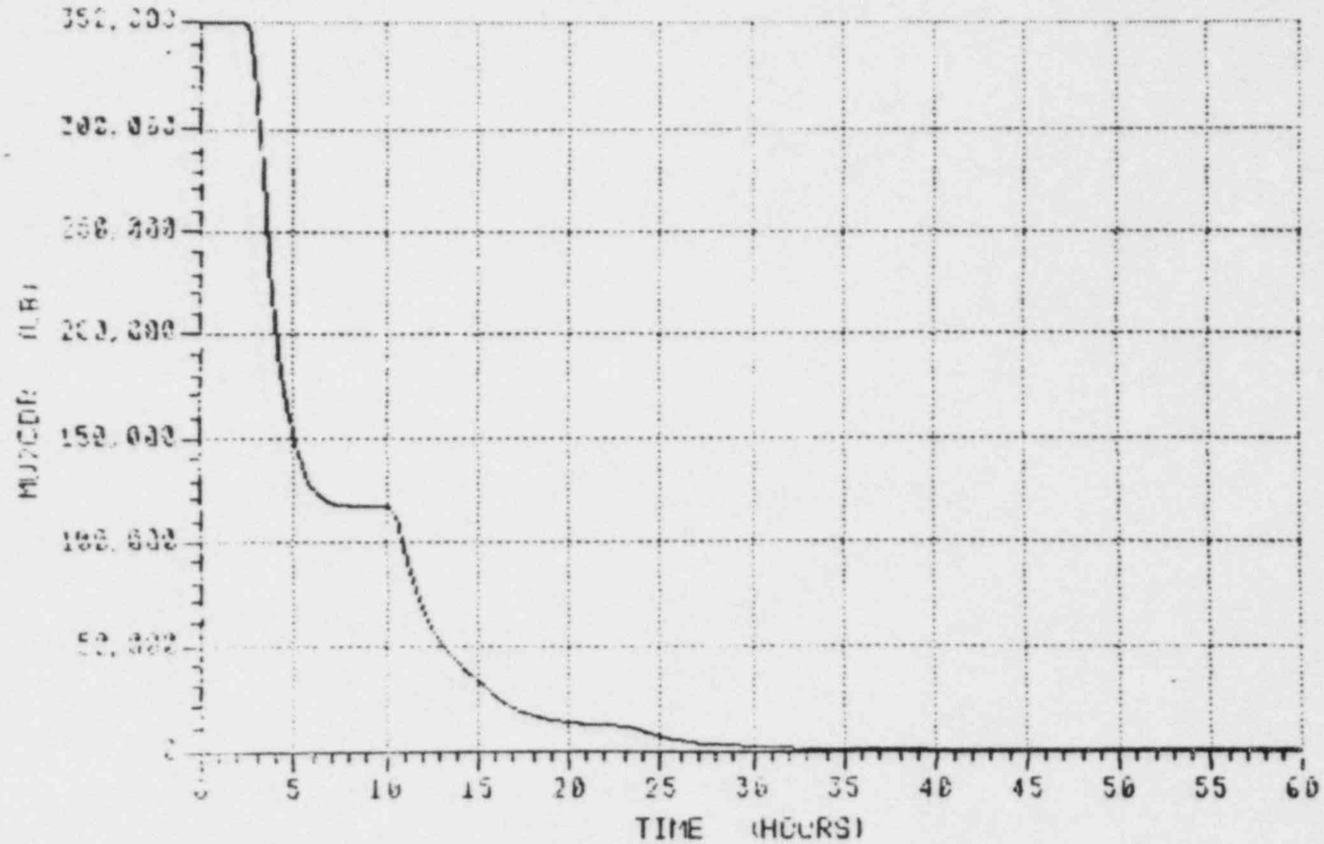

Fig. B.31 Mass of UO_2 in core region.

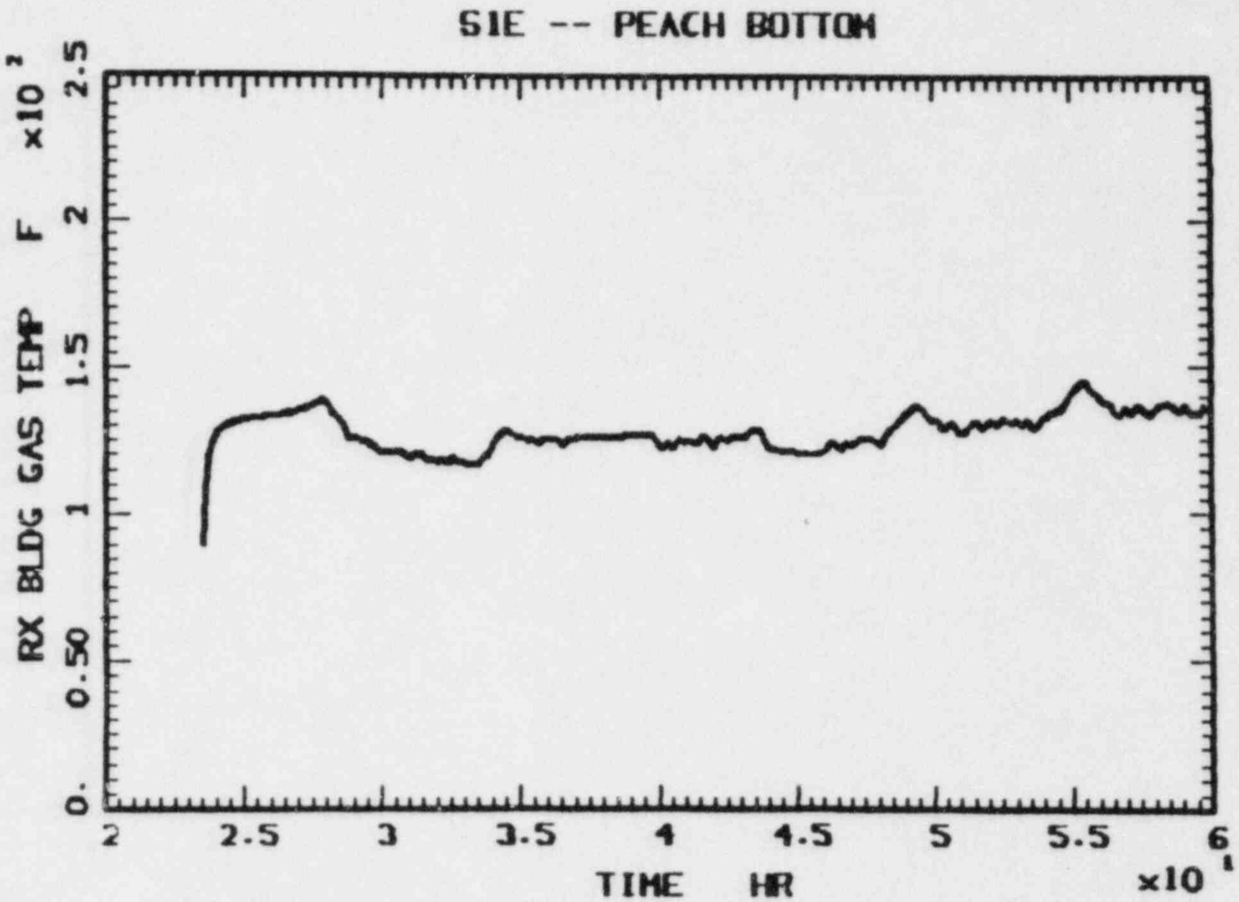

Fig. B.32 Gas temperature in reactor building, °F.

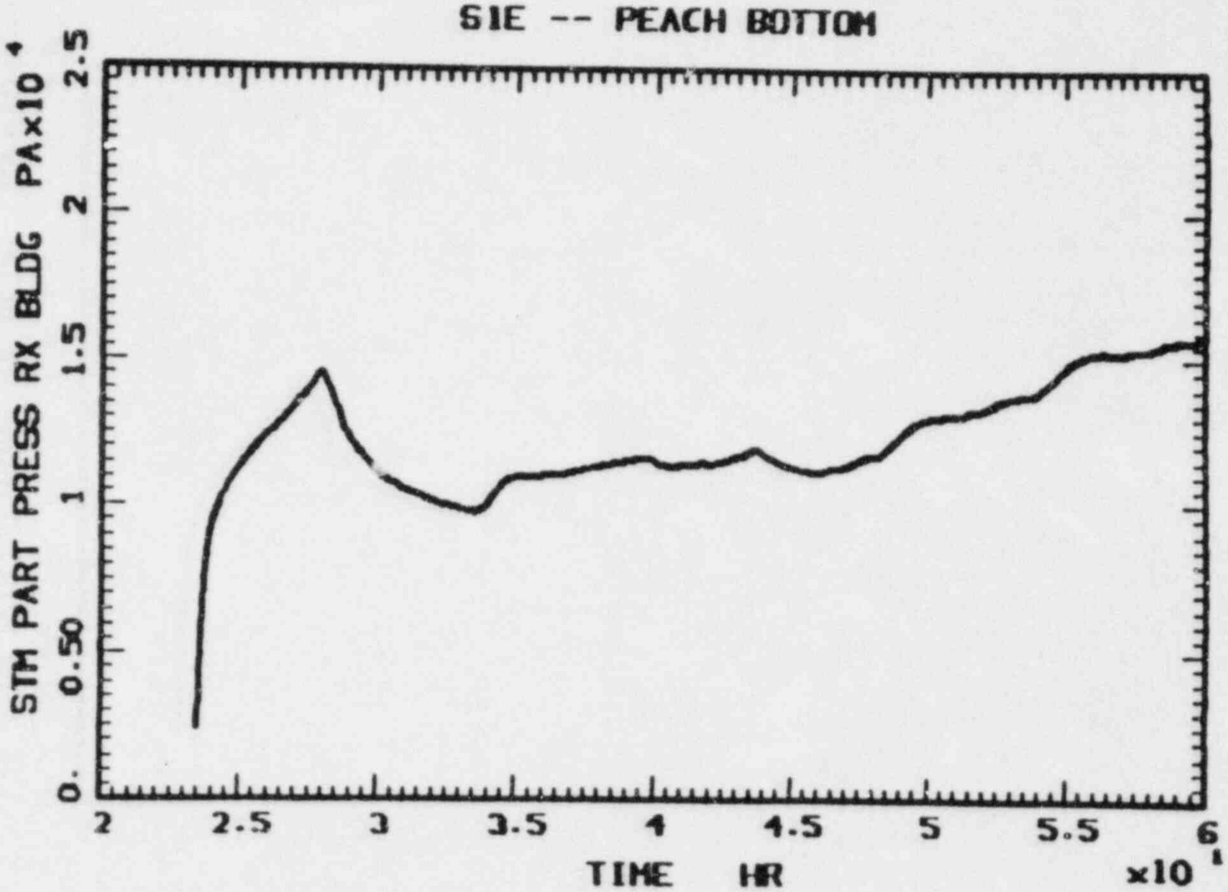

Fig. B.33 Steam pressure in reactor building, Pa.

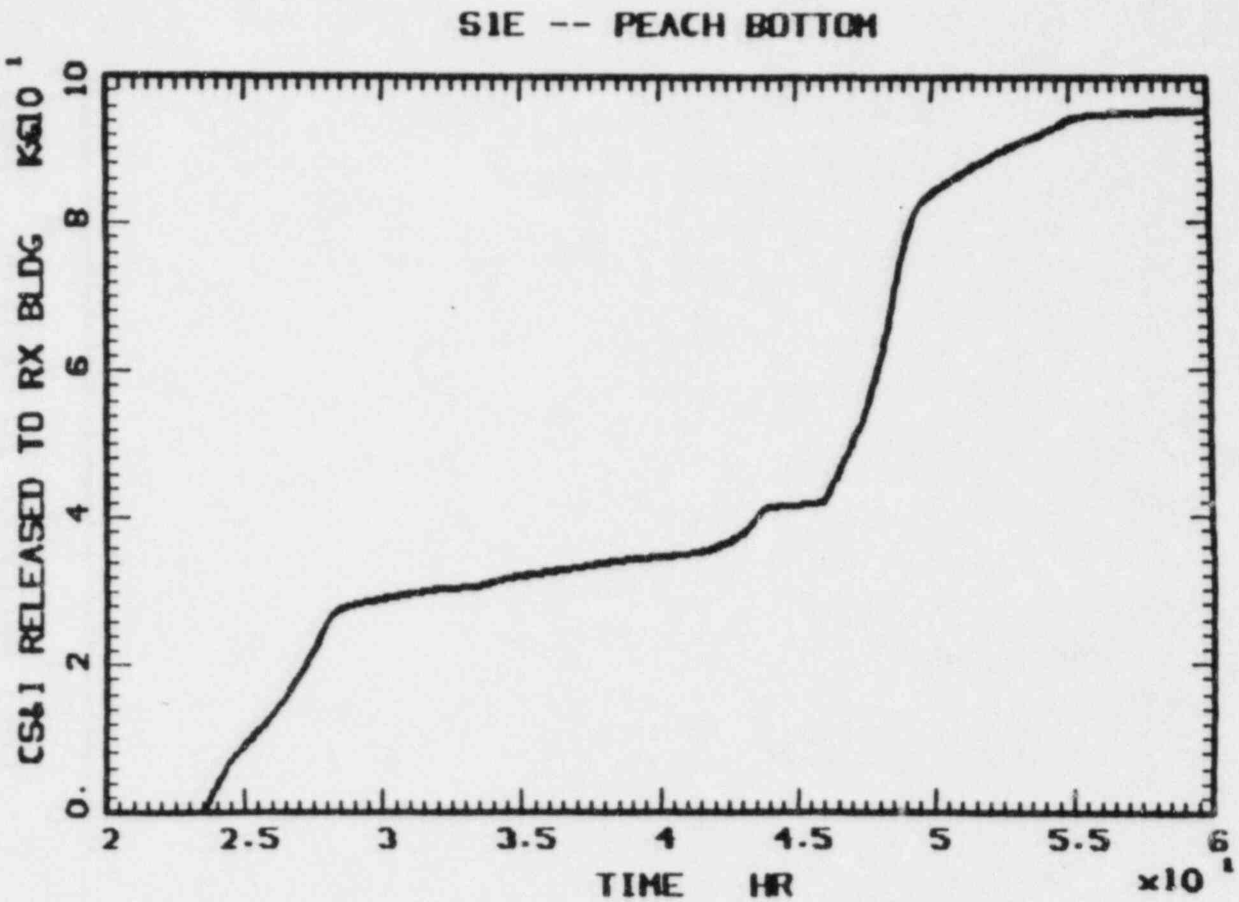

Fig. B.34 Cesium and iodine released from containment, kg.

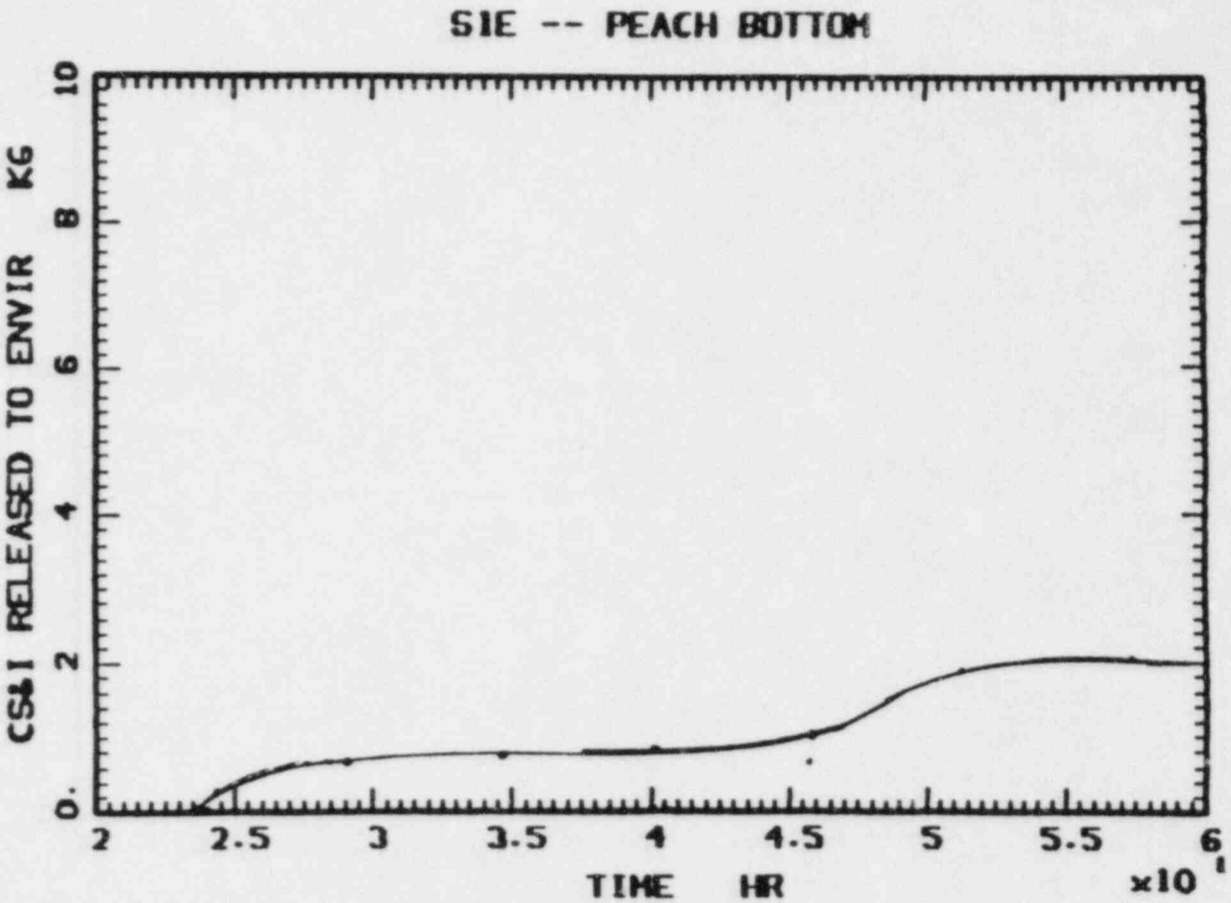

Fig. B.35 Cesium and iodine released to environment, kg.

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SUPPLEMENTAL PLOTS FOR SEQUENCE TQVW

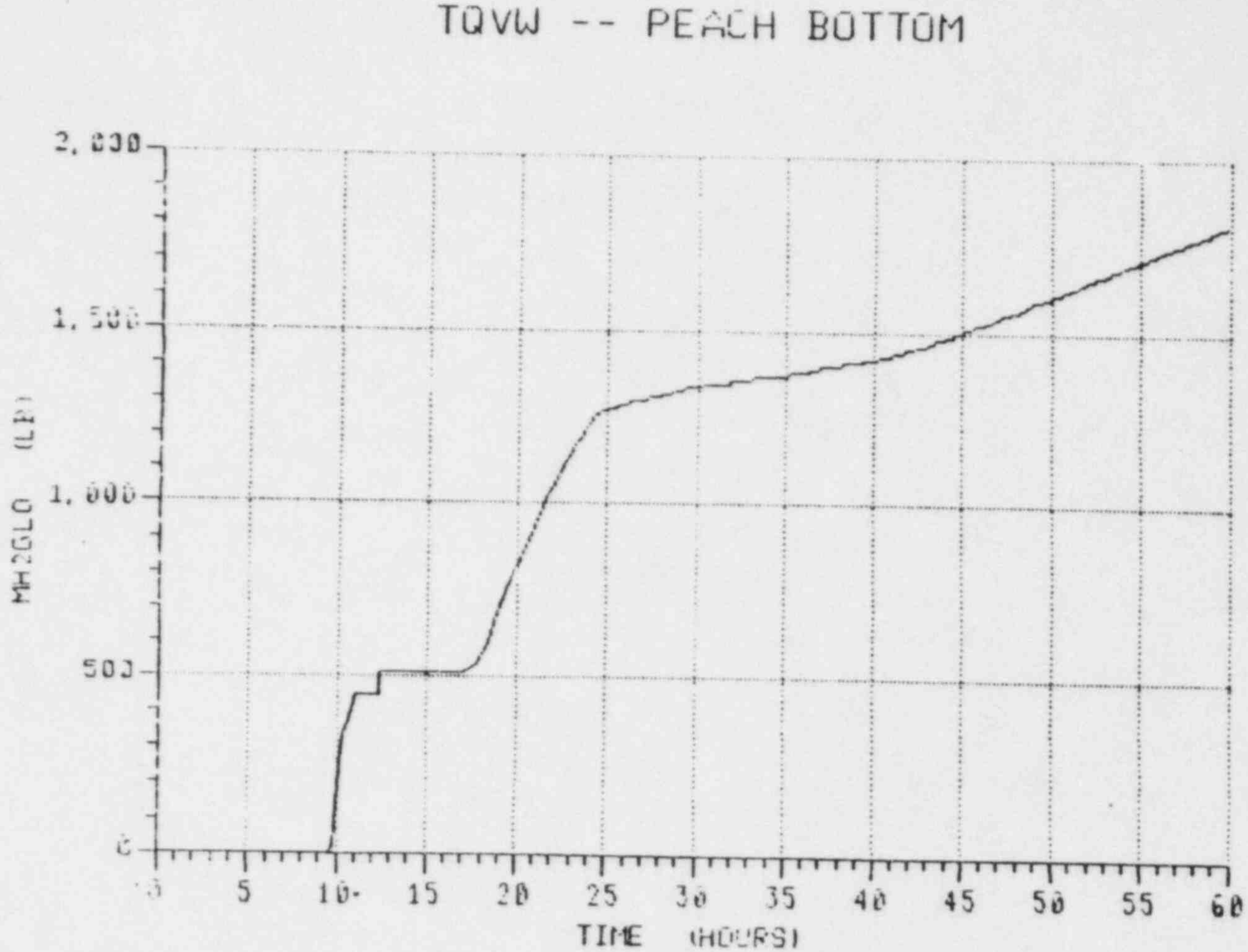

Fig. B.36 Total H_2 generated.

TQVW -- PEACH BOTTOM

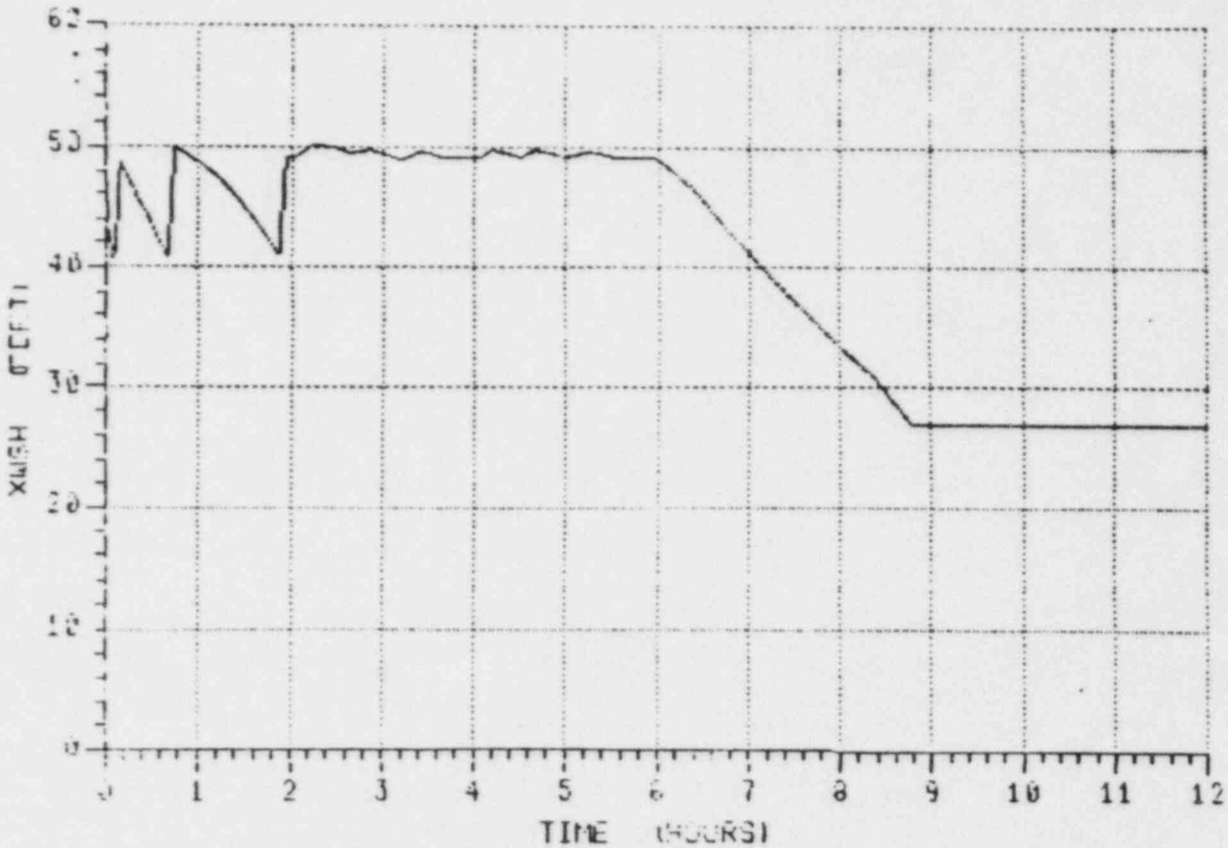

Fig. B.37 Reactor vessel water level.

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TQVW -- PEACH BOTTOM

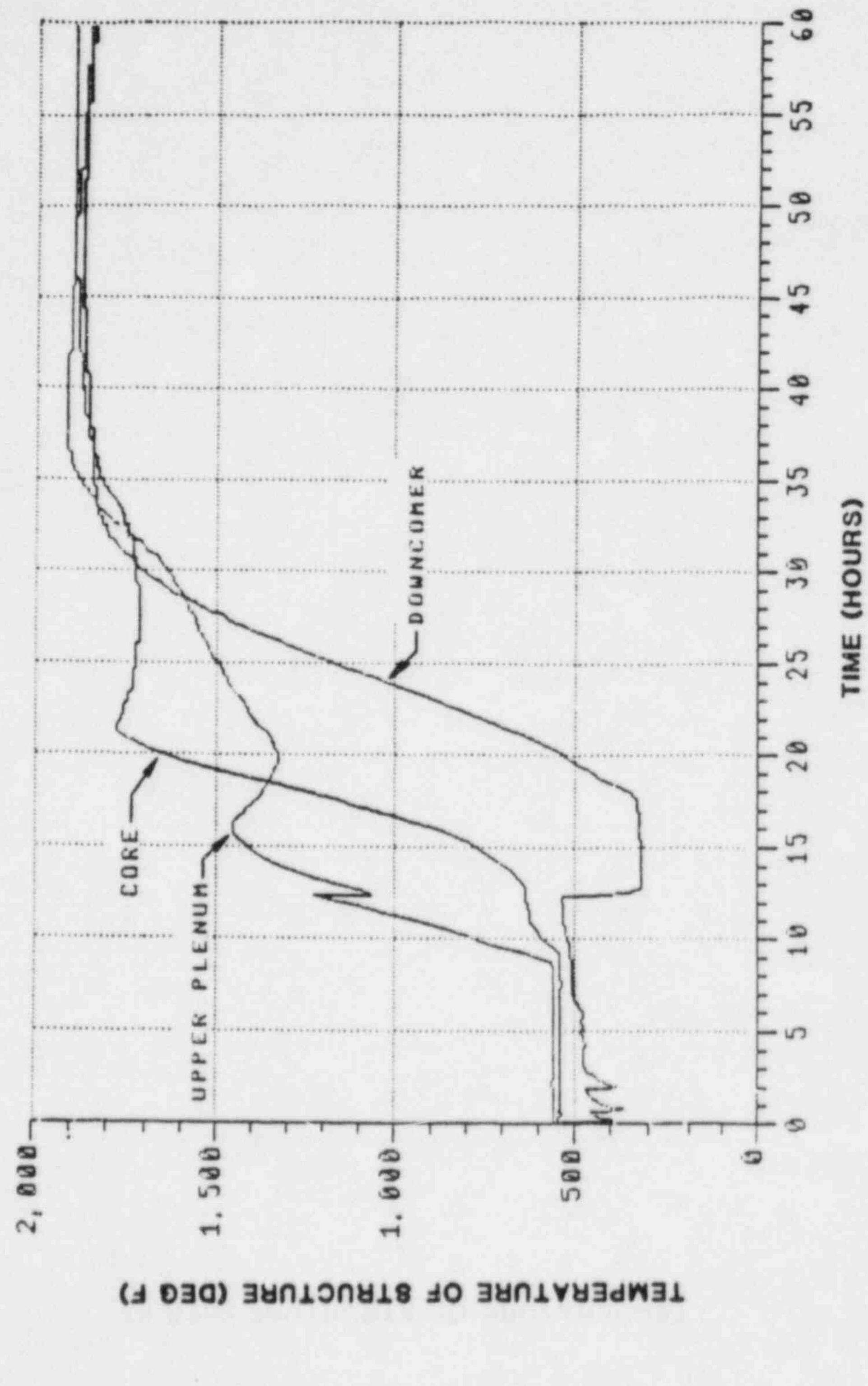

Fig. B.38 Temperature of structure, °F.

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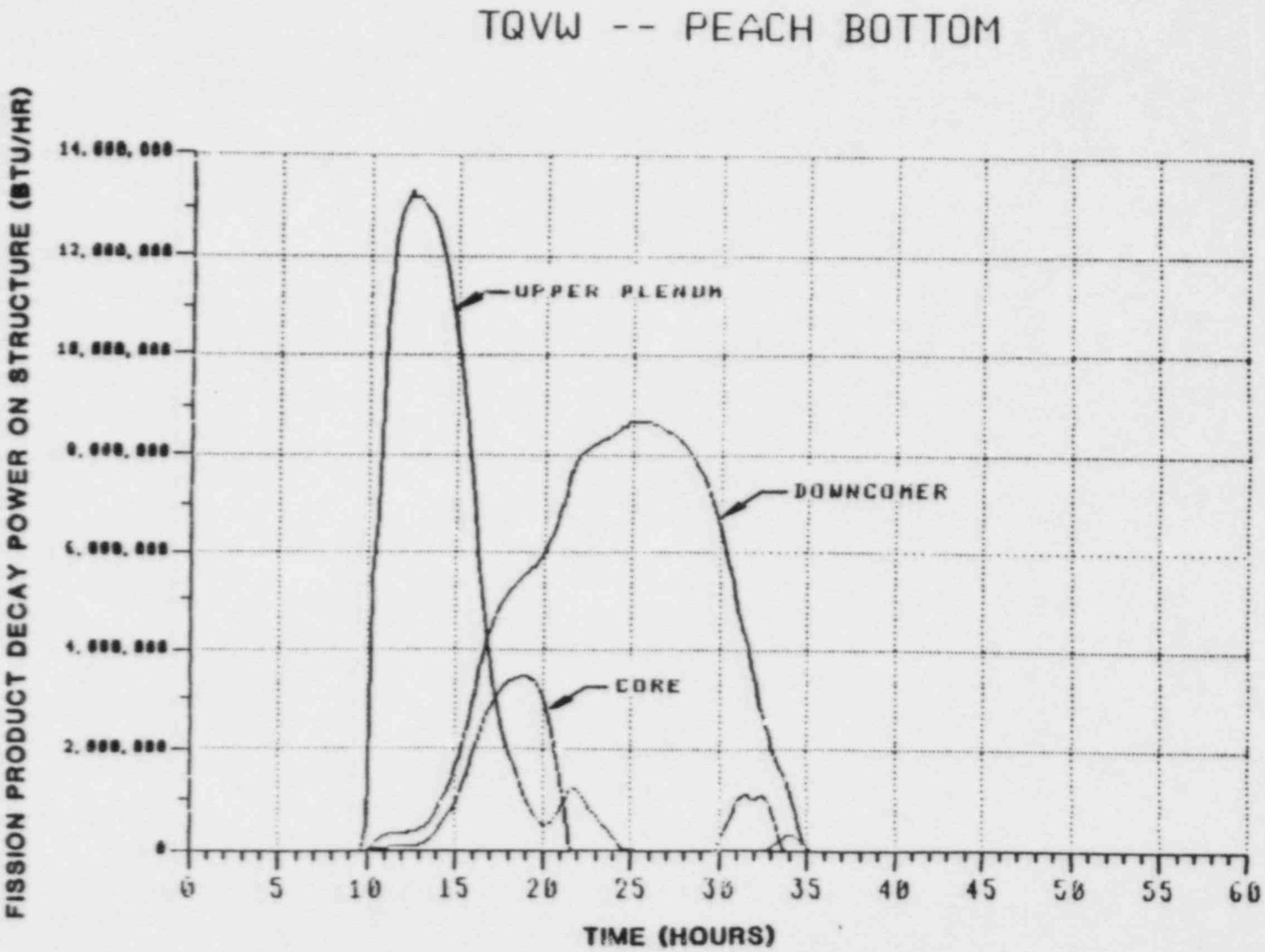

Fig. B.39 Fission product decay power on structure, Btu/hr.

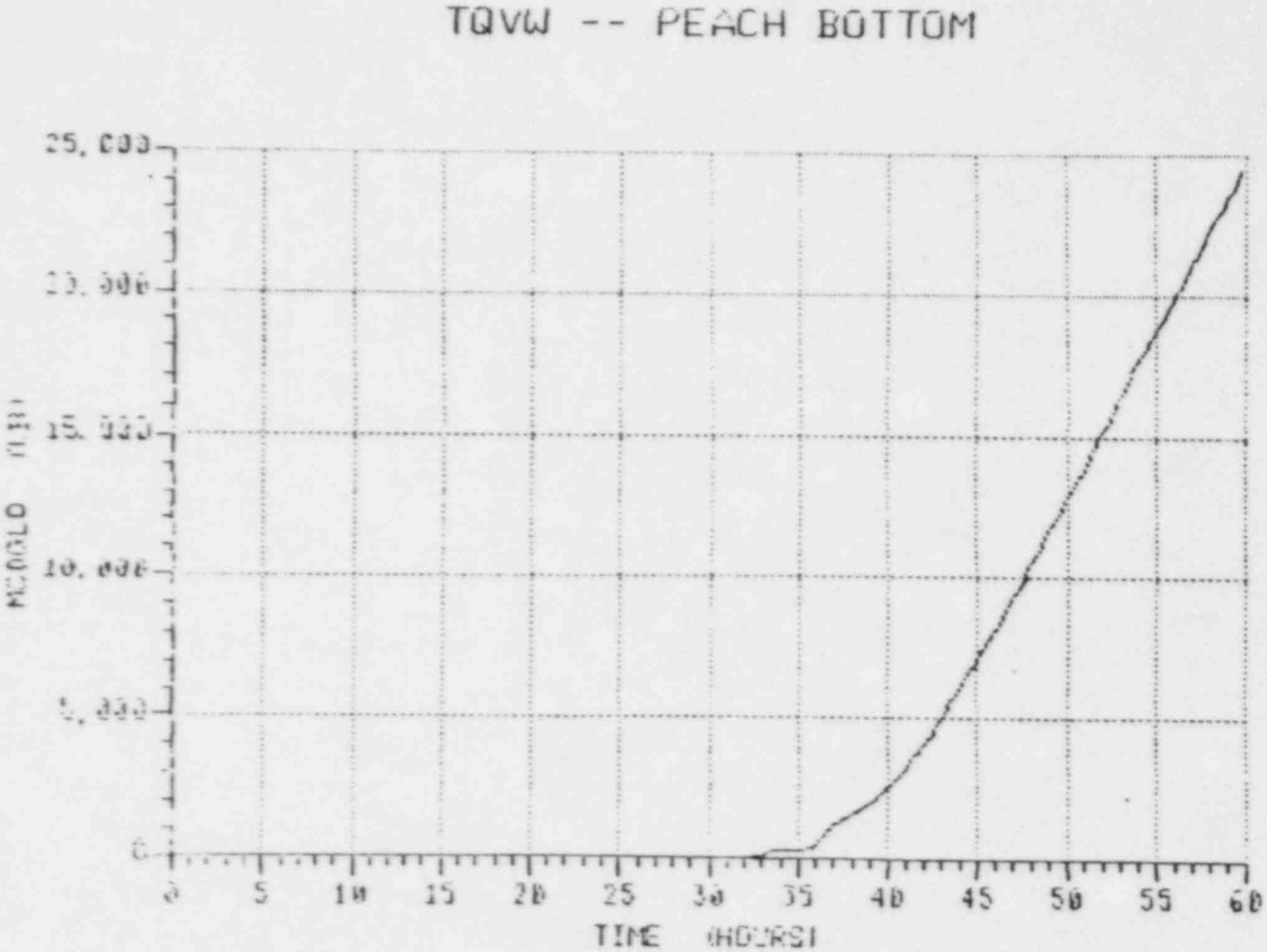

Fig. B.40 Total CO generated.

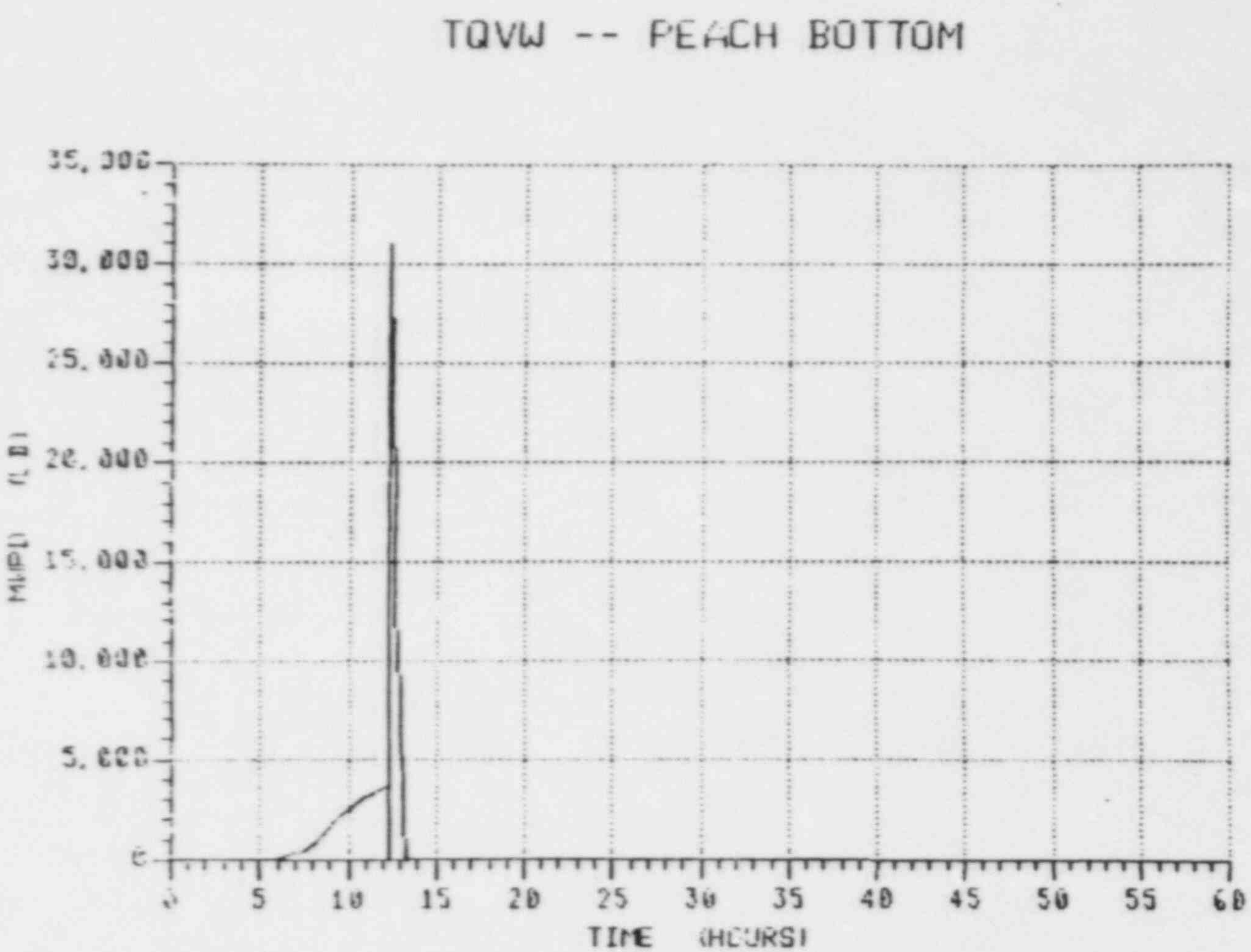

Fig. B.41 Mass of water in the pedestal.

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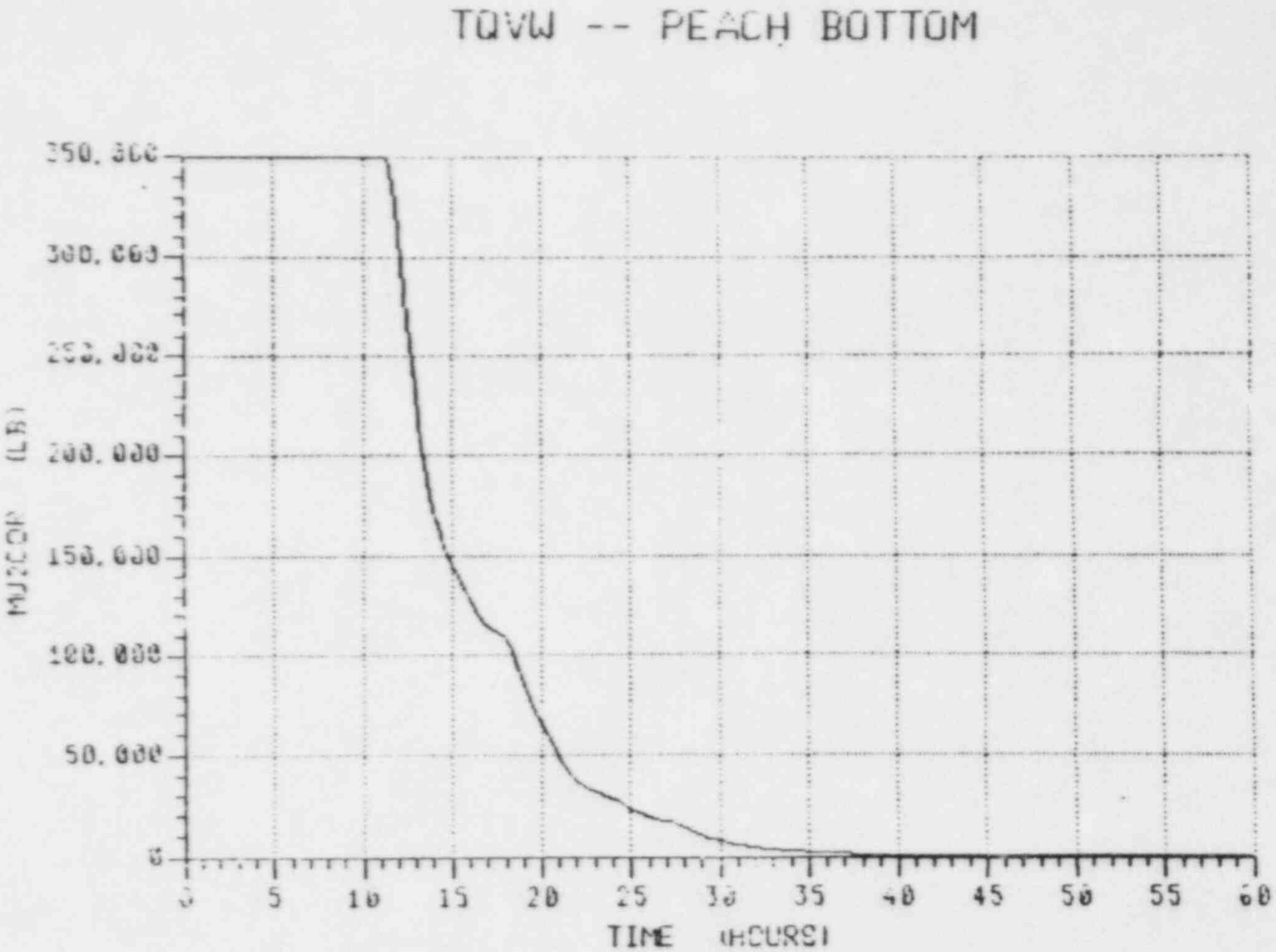

Fig. B.42 Mass of UO_2 in core region.

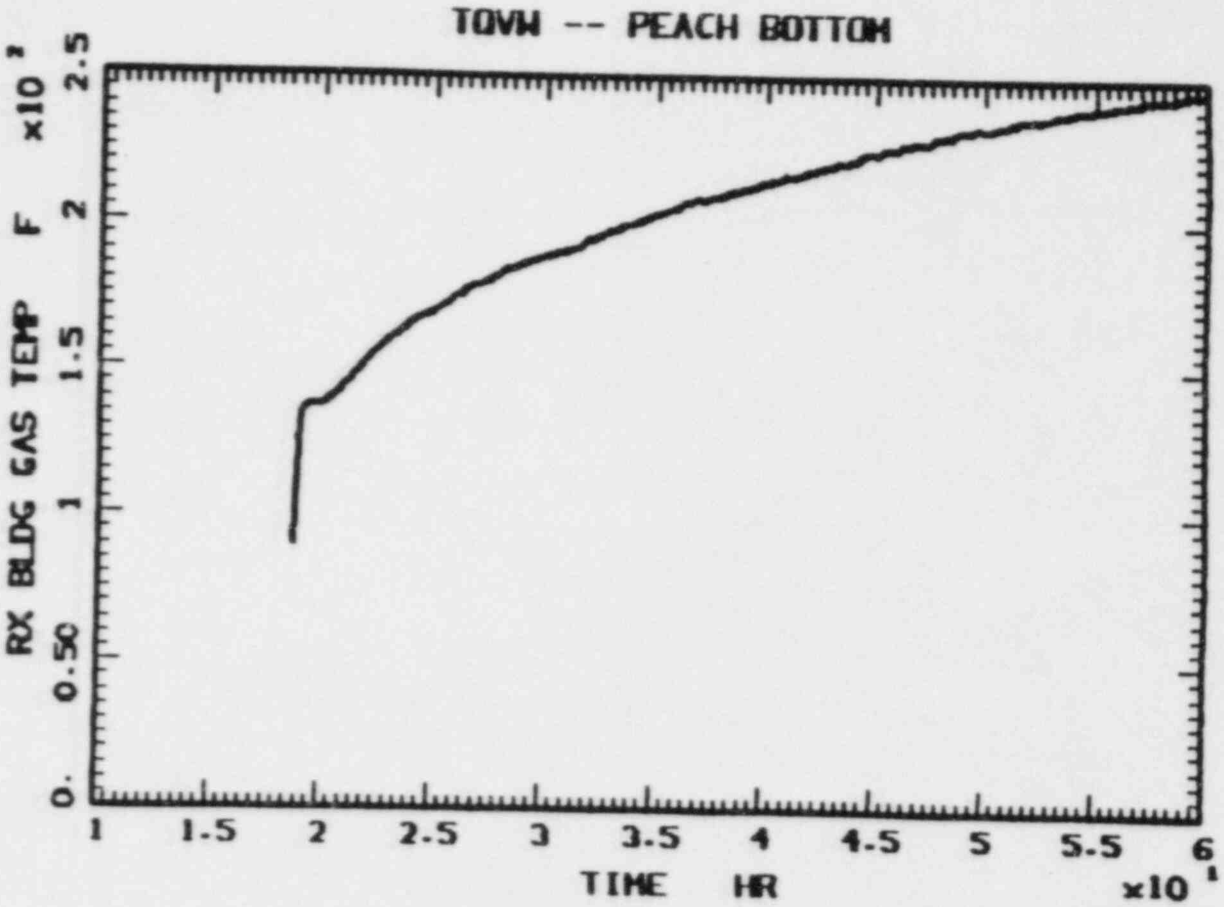

Fig. B.43 Gas temperature in reactor building, °F.

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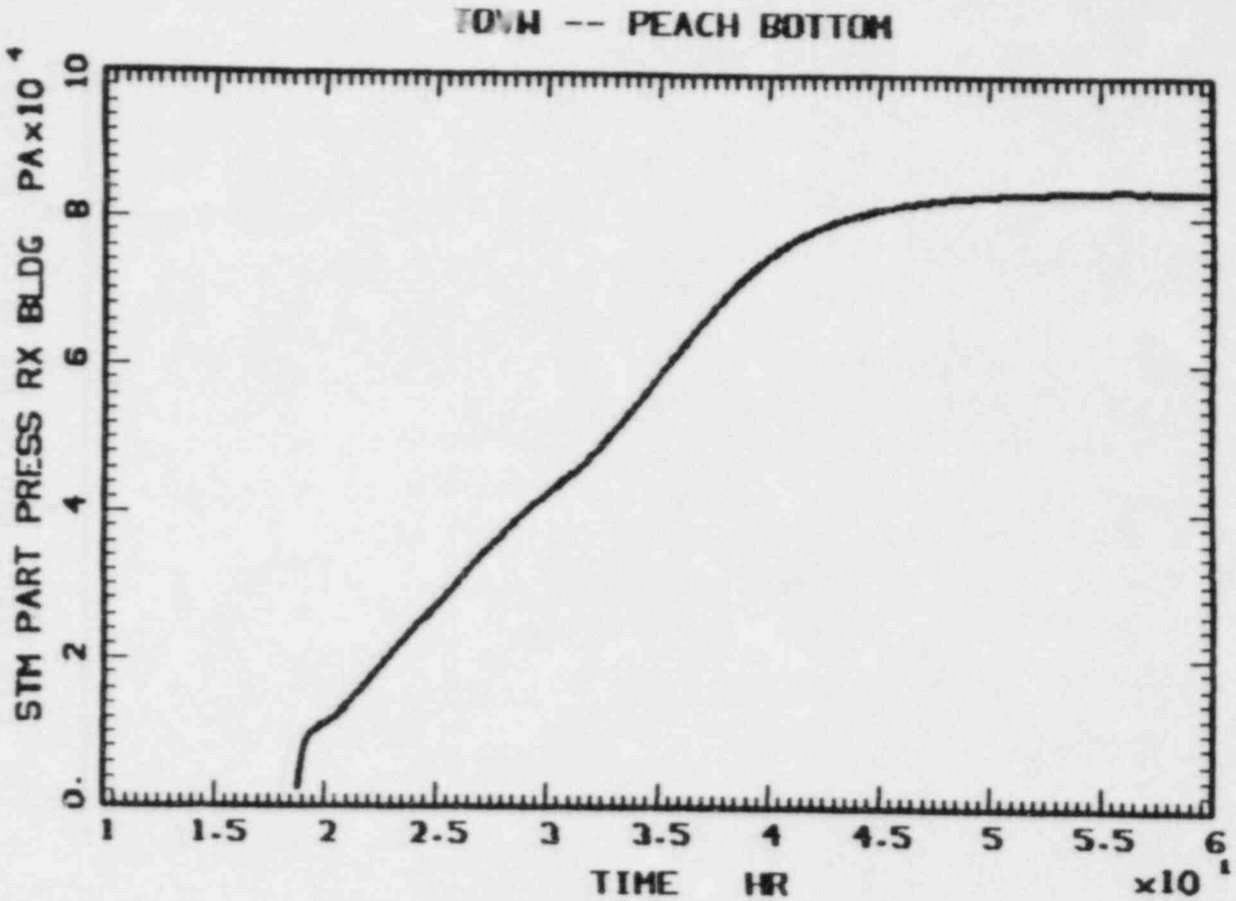

Fig. B.44 Steam pressure in reactor building, Pa.

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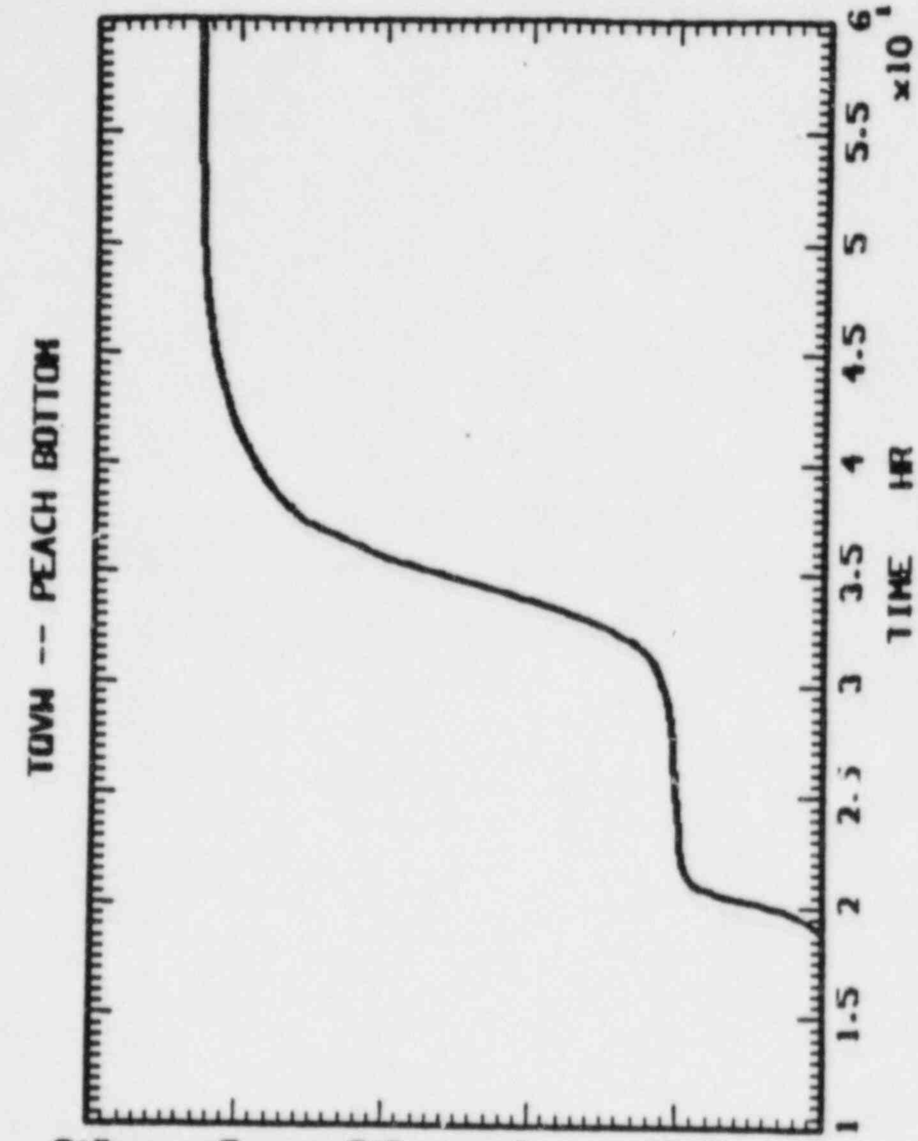

Fig. B.45 Cesium and iodine released from containment, kg.

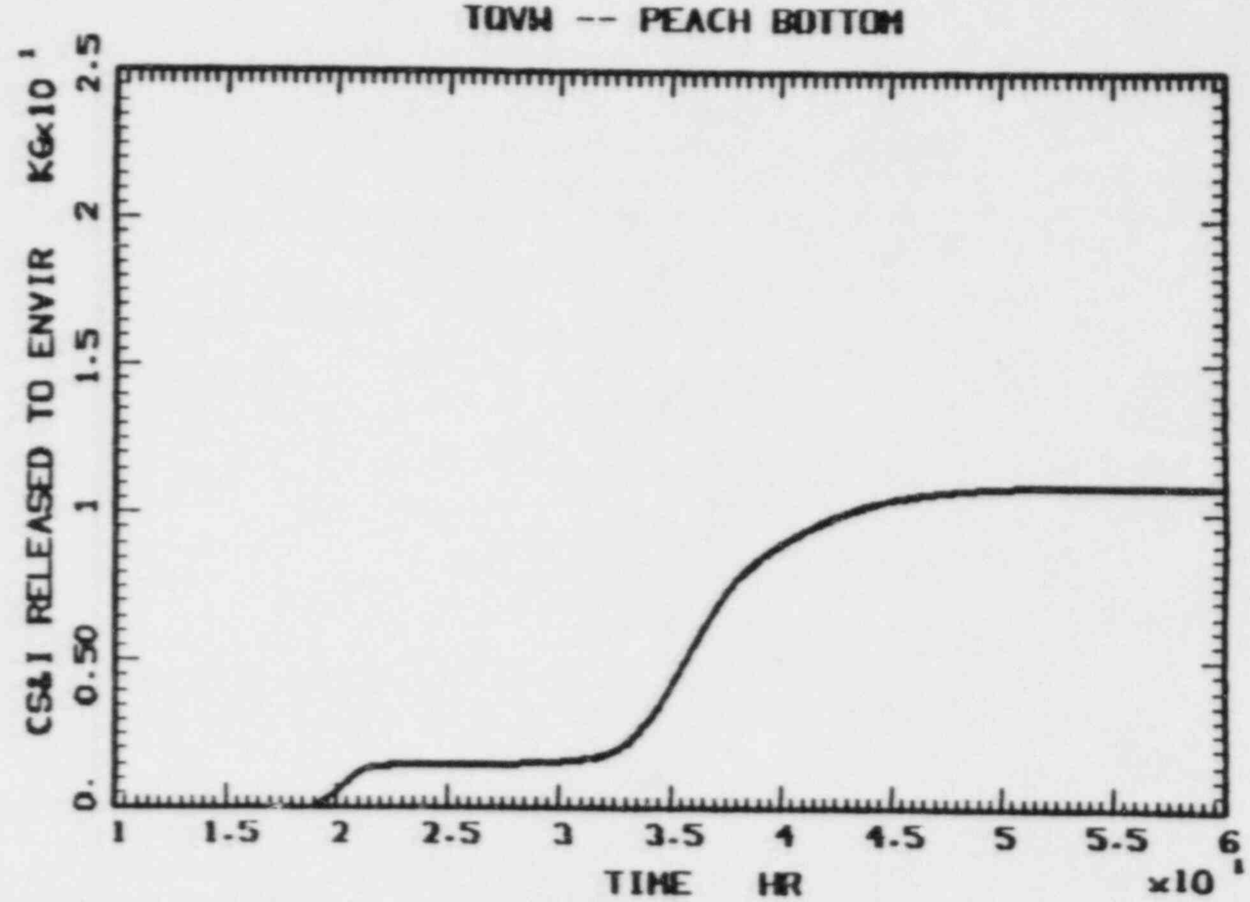

Fig. B.46 Cesium and iodine released to environment, kg.