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MEMORANDUM FOR: Document Control Desk
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FROM: Mark Cunningham, Chief
Probabilistic Risk Analysis Branch
Division of Safety Issue Resolution
Office of Nuclear Regulatory Research

SUBJECT: PLACEMENT OF RESEARCH REPORTS IN THE PUBLIC DOCUMENT ROOM

Enclosed are draft research reports on the risk of accidents initiated during low power and shutdown operations:

- T.D. Brown et al., "Evaluation of Potential Severe Accident During Low Power and Shutdown Operations at Grand Gulf, Unit 1: Evaluation of Severe Accident Risks for Plant Operational State 5 During a Refueling Outage—Main Report and Appendices," NUREG/CR-6143, Vol. 6, Part 1, Sandia National Laboratories, SAND93-2440, draft, June 1994.

- L.K. Kmetyk and T.D. Brown, "Evaluation of Potential Severe Accident During Low Power and Shutdown Operations at Grand Gulf, Unit 1: Evaluation of Severe Accident Risks for Plant Operational State 5 During a Refueling Outage—Supporting MELCOR Calculations," NUREG/CR-6143, Vol. 6, Part 2, Sandia National Laboratories, SAND93-2440, draft, June 1994.

- J.Jo et al., "Evaluation of Potential Severe Accident During Low Power and Shutdown Operations at Surry Unit-1: Evaluation of Severe Accident Risks During Mid-loop Operations—Main Report ," NUREG/CR-6144, Vol. 6, Part 1, Brookhaven National Laboratory, BNL-NUREG-52399, June 1994.

- J.Jo et al., "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry Unit-1: Evaluation of Severe Accident Risks During Mid-loop Operations—Appendices ," NUREG/CR-6144, Vol. 6, Part 2, Brookhaven National Laboratory, BNL-NUREG-52399, draft, June 1994.

These reports provide background information on the soon-to-be published proposed rule on lower power and shutdown operations. Since they are not

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quite ready for publication as NUREG/CR reports, please deposit these draft reports in the Public Document Room. If you have questions, then please contact Christopher Ryder at (301) 415-6102.

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Office of Nuclear Regulatory Research

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NUREG/CR-6143
SAND93-2440
Vol. 6, Part 1

Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1

Evaluation of Severe Accident Risks for Plant Operational State 5 During a Refueling Outage

Main Report and Appendices

Draft Completed: June 1994
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Abstract

Traditionally, probabilistic risk assessments (PRA) of severe accidents in nuclear power plants have considered initiating events potentially occurring only during full power operation. Recent studies and operation experience have, however, implied that accidents during low power and shutdown could be significant contributors to risk. In response to this concern, in 1989 the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied. The program consists of two parallel projects being performed by Brookhaven National Laboratory (Surry) and Sandia National Laboratories (Grand Gulf). The program objectives include assessing the risks of severe accidents initiated during plant operational states other than full power operation and comparing the estimated risks with the risk associated with accidents initiated during full power operation as assessed in NUREG-1150. The scope of the program includes that of a Level-3 probabilistic risk assessment (PRA).

The subject of this report is the PRA of the Grand Gulf Nuclear Station, Unit 1. The Grand Gulf plant utilizes a 3833 MWt BWR-6 boiling water reactor housed in a Mark III containment. The Grand Gulf plant is located near Port Gibson, Mississippi. The regime of shutdown analyzed in this study was plant operational state (POS) 5 during a refueling outage which is approximately Cold Shutdown as defined by Grand Gulf Technical Specifications. The entire PRA of POS 5 is documented in a multi-volume NUREG/CR report (i.e., NUREG/CR-6143). The internal events analysis is documented in Volume 2. Internal fire and internal flood analyses are documented in Volumes 3 and 4, respectively. A separate study on seismic analysis, documented in Volume 5, was performed for the NRC by Future Resources Associates, Inc. The Level-2/3 study for traditional internal events is documented in Volume 6, and a summary of the results for all analyses is documented in Volume 1.

The analysis documented in this volume of the report is the Level 2/3 analysis of the traditional internal events. Plant damage states, which define the configuration of the plant and its systems at the onset of core damage for the accidents scenarios developed in the Level 1 analysis, were used to define the interface between the Level 1 and Level 2/3 analyses. In the Level 2/3 analysis, the possible progressions of the accident following the onset of core damage were delineated and the amount of radioactive material released to the environment was estimated. Based on the amount of radioactive material released to the environment, health effects to the general public were estimated. In addition to the offsite consequences, a scoping analysis of the potential doses and dose rates within the site were also estimated. The final product of the analysis was the integration of the accident frequencies with the consequences of the accidents to form an expression for aggregate risk.

The risk associated with Grand Gulf as it operates in POS 5 during a refueling outage was shown to be comparable with the risk associate with full power operation. In NUREG-1150 the risk from full power operation of Grand Gulf was shown to be quite low. While the risk associated with POS 5 is low, there are very few features of the plant that are available to attenuate a release should one occur. The most likely accidents in POS 5 have an open containment, the suppression pool is bypassed, the containment sprays are not available, and the vessel fails releasing the core debris into the containment. The low values for risk given the high conditional releases are, in part, due to the extremely low core damage frequency and the sparse population around the plant.

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Acronyms

ADHR	Auxiliary Decay Heat Removal
ADS	Automatic Depressurization System
APB	Accident Progression Bin
APET	Accident Progression Event Tree
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CCI	Core-Concrete Interaction
CD	Core Damage
CDS	Condensate
CNMT	Containment
CRD	Control Rod Drive
CS	Containment Spray
CVS	Containment Venting System
DCH	Direct Containment Heating
ECCS	Emergency Core Cooling Systems
EOP	Emergency Operating Procedures
EPS	Emergency Power System
ES	End State
ESF	Engineered Safety Feature
FCI	Fuel-Coolant Interaction
HEP	Human Error Probability
HIS	Hydrogen Igniter System
HPCS	High Pressure Core Spray
HRA	Human Reliability Analysis
LHS	Latin Hypercube Sampling
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LP&S	Low Power and Shutdown
MSIV	Main Steam Isolation Valve
NRC	Nuclear Regulatory Commission
OC	Operating Condition
PDS	Plant Damage State
POS	Plant Operating State
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat removal
RPV	Reactor Pressure Vessel
SBO	Station Blackout
SDC	Shutdown Cooling System(s)
SGTS	Standby Gas Treatment System
SP	Suppression Pool
SPC	Suppression Pool Cooling
SPMU	Suppression Pool Makeup
SRV	Safety Relief Valve
SSW	Standby Service Water Crosstie
STG	Source Term Group
VB	Vessel Breach

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Foreword

(NUREG/CR-6143 and 6144) Low Power and Shutdown Probabilistic Risk Assessment Program

Traditionally, probabilistic risk assessments (PRA) of severe accidents in nuclear power plants have considered initiating events potentially occurring only during full power operation. Some previous screening analyses that were performed for other modes of operation suggested that risks during those modes were small relative to full power operation. However, more recent studies and operational experience have implied that accidents during low power and shutdown could be significant contributors to risk.

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects performed by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL), with the seismic analysis performed by Future Resources Associates. Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied.

The objectives of the program are to assess the risks of severe accidents due to internal events, internal fires, internal floods, and seismic events initiated during plant operational states other than full power operation and to compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150. The scope of the program includes that of a level-3 PRA.

The results of the program are documented in two reports, NUREG/CR-6143 and 6144. The reports are organized as follows:

For Grand Gulf:

NUREG/CR-6143 - Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1

Volume 1: Summary of Results
Volume 2: Analysis of Core Damage Frequency from Internal Events for Plant Operational State 5 During a Refueling Outage
- Part 1: Main Report
- Part 2: Internal Events Appendices A to H
- Part 3: Internal Events Appendices I and J
- Part 4: Internal Events Appendices K to M

Volume 3: Analysis of Core Damage Frequency from Internal Fire Events for Plant Operational State 5 During a Refueling Outage
Volume 4: Analysis of Core Damage Frequency from Internal Flooding Events for Plant Operational State 5 During a Refueling Outage
Volume 5: Analysis of Core Damage Frequency from Seismic Events for Plant Operational State 5 During a Refueling Outage
Volume 6: Evaluation of Severe Accident Risks for Plant Operational State 5 During a Refueling Outage
- Part 1: Main Report
- Part 2: Supporting MELCOR Calculations

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Foreword (Continued)

For Surry:

NUREG/CR-6144 - Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry Unit-1

- Volume 1: Summary of Results
- Volume 2: Analysis of Core Damage Frequency from Internal Events During Mid-loop Operations
  - Part 1: Main Report
  - Part 2: Internal Events Appendices A to D
  - Part 3: Internal Events Appendix E
  - Part 4: Internal Events Appendices F to H
  - Part 5: Internal Events Appendix I
- Volume 3: Analysis of Core Damage Frequency from Internal Fires During Mid-loop Operations
- Volume 4: Analysis of Core Damage Frequency from Internal Floods During Mid-loop Operations
- Volume 5: Analysis of Core Damage Frequency from Seismic Events During Mid-loop Operations
- Volume 6: Evaluation of Severe Accident Risks During Mid-loop Operations
  - Part 1: Main Report
  - Part 2: Appendices

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1 Executive Summary

Traditionally, probabilistic risk assessments (PRAs) of severe accidents in nuclear power plants have only considered accidents initiated during full power operation. Some previous screening analyses that have been performed for other than full-power modes of operation suggest that risks during those modes of operation were small relative to those occurring during full power operation. However, recent studies and operational experiences indicate that low power and shutdown accident risk may be significant. Although the power of the reactor core is much less in off power conditions than at full power, the technical specifications allow for more equipment to be inoperable in off power conditions (e.g., in certain conditions the containment can be open).

In response to the concerns over risk during low power and shutdown conditions, the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research (NRC RES) has undertaken a two phase project to analyze the frequencies, consequences, and risk of accidents occurring during modes of operation other than full power.

Phase 1 of the project was completed in September of 1991 [Whitehead, et al., 1991]. This phase involved a coarse screening of potential accidents that could occur at a Boiling Water Reactor (BWR) while operating a other than full power and was adopted as a means of obtaining, in a relatively short time, some estimate of the potential for accidents during low power and shutdown conditions. The BWR examined was the Grand Gulf Nuclear Power Station, a single unit 1250 MWe (net) BWR 6 power plant with a Mark III containment, located near Port Gibson, Mississippi.

The coarse screening analysis indicated that risk during these conditions cannot be shown to be insignificant by a conservative screening analysis. Hence, the NRC decided to have follow-on detailed analyses performed. Consequently, the NRC decided to first perform a detailed analysis consisting of a Level 3 PRA on one of the off power conditions. Based on trends indicated in the results of the coarse screening analysis, plant operational state (POS) 5 (consisting mainly of Cold Shutdown Operating Condition) was selected for detailed analysis. (NOTE: Plant operational states are artificial subdivision of the time plants spend in low power and shutdown conditions. This concept was developed during Phase 1 of this project to all ow the analyst to better represent the plant as it transitions form power operation to non power operation.) The report presents the results of the detailed analysis of the Grand Gulf facility in POS 5 during a refueling outage.

A companion project for the Surry Pressurized Water Reactor (PWR) is being conducted by Brookhaven National Laboratory (BNL).

This volume of the report, Volume 6, presents the Level 2/3 portion of the Level 3 PRA that was performed on POS 5.

1.1 Objectives

The primary objective of this study was to perform an analysis of potential accidents that could occur at Grand Gulf while the plant is in POS 5. Additional specific objectives for this study include:

- For POS 5, perform a preliminary characterization of the accident progressions following core damage and estimate the consequences that result from these accidents.
- Determine quantitatively the risk and estimate the uncertainty in risk for the risk significant mode of operation.
- Compare the risk associated with POS 5 to the safety goals and to the risk associated with full power operation.
- Provide an assessment of the potential for a radioactive release to cause onsite consequences.

1.2 Approach

The risk associated with POS 5 was determined in the Level 2 and 3 portions of the PRA using a simplified form of the NUREG-1150 methodology [U.S. NRC, 1990]. The Level 2/3 portion of the PRA is concerned with the progression of postulated accidents following the onset of severe core damage and the estimation of the consequences that result from the release of any radioactive material and, as such, consists of the following constituent analyses: plant damage state (PDS), accident progression, source term, consequence, and risk integration. A brief summary of the approach used in each of the constituent analyses is provided below.

Plant Damage State Analysis: PDSs were developed to define the interface between the accident frequency analysis (Level 1) and the accident progression analysis (Level 2). Core damage accidents that have similar plant and system configurations at the onset of core damage are grouped together; each group is called a plant damage state.

Accident Progression Analysis: Based on the configuration of the plant defined by the PDSs, event tree techniques were used to delineate the accident progressions following the onset of core damage. The accident progressions define the status of the containment and other features of the plant that are used to mitigate the accident during the various phases of the accident and also identify phenomena that may impact the release of radioactive material. The accident progression event tree (APET) developed in this study is similar in concept to the APETs developed in NUREG-1150, however, it is not as detailed. Compared to the NUREG-1150 APETs, the POS 5 APET included fewer questions (i.e., top events), issues were addressed in less detail, and formal expert judgement procedures were not used to quantify the APET.

Source Term Analysis: Source terms, which characterize the type and amount of radioactive material releases from the plant, were estimated for groups of accident progression using the parametric approach developed in NUREG-1150 [Jow, et al., 1993]. The parametric expression was quantified, to the extent possible, using information from the NUREG-1150 full power analysis of Grand Gulf [Harper, et al., 1992]. The source terms were then combined into a manageable number of source term groups using partitioning algorithm first developed in NUREG-1150 study [Iman, et al., 1990] and then modified in the full power study of the LaSalle plant [Brown, et al., 1992].

Consequences: Offsite consequences were estimated for each source term group using the MACCS code. The emergency response assumption used in this study are the same as those used in the NUREG-1150 analysis [Brown, et al., 1990]. In addition to offsite consequences, this study also included a scoping analysis of onsite consequences.

Risk Integration: The risk results reported in this study are estimates of aggregate risk which is the sum over all accidents scenarios of the product of the accident frequency with its consequence.

A limited uncertainty analysis, which included variables from the PDS, accident progression, and source term analyses, was also performed. In contrast to NUREG-1150, expert opinion techniques were not used in this study to quantify the accident progression and source term models.

To analyze the potential accident that can occur during POS 5 it was necessary to divide POS 5 into three distinct time regimes (also called time windows): (1) entry into POS 5 to 24 hours after shutdown, 24 hours after shutdown to entry into POS 6 (i.e., POS 6 begins approximately 94 hours after shutdown), and POS 5 after core alterations (i.e., this last time regime starts 40 approximately days after shutdown and lasts for approximately 10. 4 days). For each time window the appropriate core power and radionuclide inventory was used to estimate the timing of the accident and its potential consequences.

1.3 Results

1.3.1 Core Damage Frequency

For discussion purposes, the core damage scenarios identified in the Level 1 analysis can be combined into the following three PDS groups (12 PDSs were actually evaluated in the accident progression analysis): LOCAs, Station Blackouts (SBOs), and Other Transients. The mean core damage frequencies and the mean fractional contributions to the core damage frequency for these three groups are provided in Table 1-1. The LOCA PDS group is the dominant contributor to the core damage frequency followed the by SBO PDS group and the Other Transients PDS group.

Table 1-1
Core Damage Frequencies for LOCA, SBO, and Other Transient PDS Groups

PDS Group	Mean Core Damage Frequency (1/yr)	Mean Fractional Contribution
LOCA	1.1E-06	0.51
SBO	7.4E-07	0.33
Other Transients	2.4E-07	0.17
Total	2.1E-06	

1.3.2 Accident Progression

A simplified representation of the APET that addresses the major aspects of the accident is shown in Figure 1-1. The actual APET included 59 top events or questions). Figure 1-1 combines the result from all of the accidents and is conditional on the occurrence of core damage; the values displayed in these figures are mean conditional probabilities. From the simplified tree presented in Figure 1-1, it can be seen that the most likely accidents in POS 5 have an open containment, the suppression pool is bypassed, and the vessel fails. For the cases where the vessel fails, there is a significant probability that the core debris will either be quenched in a flooded cavity or the interactions between the core debris and the concrete structures beneath the vessel (referred to as CCI) will occur in a flooded cavity. For the former, the releases associated with CCI are prevented. In latter case, the releases are scrubbed by the water in the flooded cavity. If the containment is closed prior to core damage, it is predicted to either fail or to be vented after core damage since containment heat removal is not available in these accidents; venting the containment late in the accident is the most likely scenario. For the accidents identified in POS 5, the containment sprays were never available during core damage.

1.3.3 Aggregate Risk

Table 1-2 presents the offsite risk results for the following six measures: early fatalities, total latent cancer fatalities, population dose within 50 miles of the site, population dose within 1000 miles of the site, individual early fatality risk within 1 mile of the site, and individual latent cancer risk within 10 miles of the site. The core damage frequency for POS 5 is also provided in Table 1-2. The factors that lead to low offsite risk values include the following:

- The core damage frequency calculated for the Grand Gulf plant is extremely low. Thus, while a significant release may occur, the frequency of the release is sufficiently small that the resulting risk is also small.

- The population density around the Grand Gulf plant is also low. Although many factors influence the magnitude of the consequences, in general, for a given release, the smaller the population, the smaller the number of fatalities. Of the four Mark III plants in the United States, Grand Gulf has the fewest number of people living within 50 miles of the plant according the 1990 census data.

- Although in many of the accidents the containment equipment hatch is open, the suppression pool is bypassed, and the containment sprays are unavailable, the releases pass through the auxiliary building before escaping into the environment. Because of its large volume and surface area, the auxiliary building provides a location for the radionuclides to be attenuated by deposition and thereby reduce the source term to the environment. Without

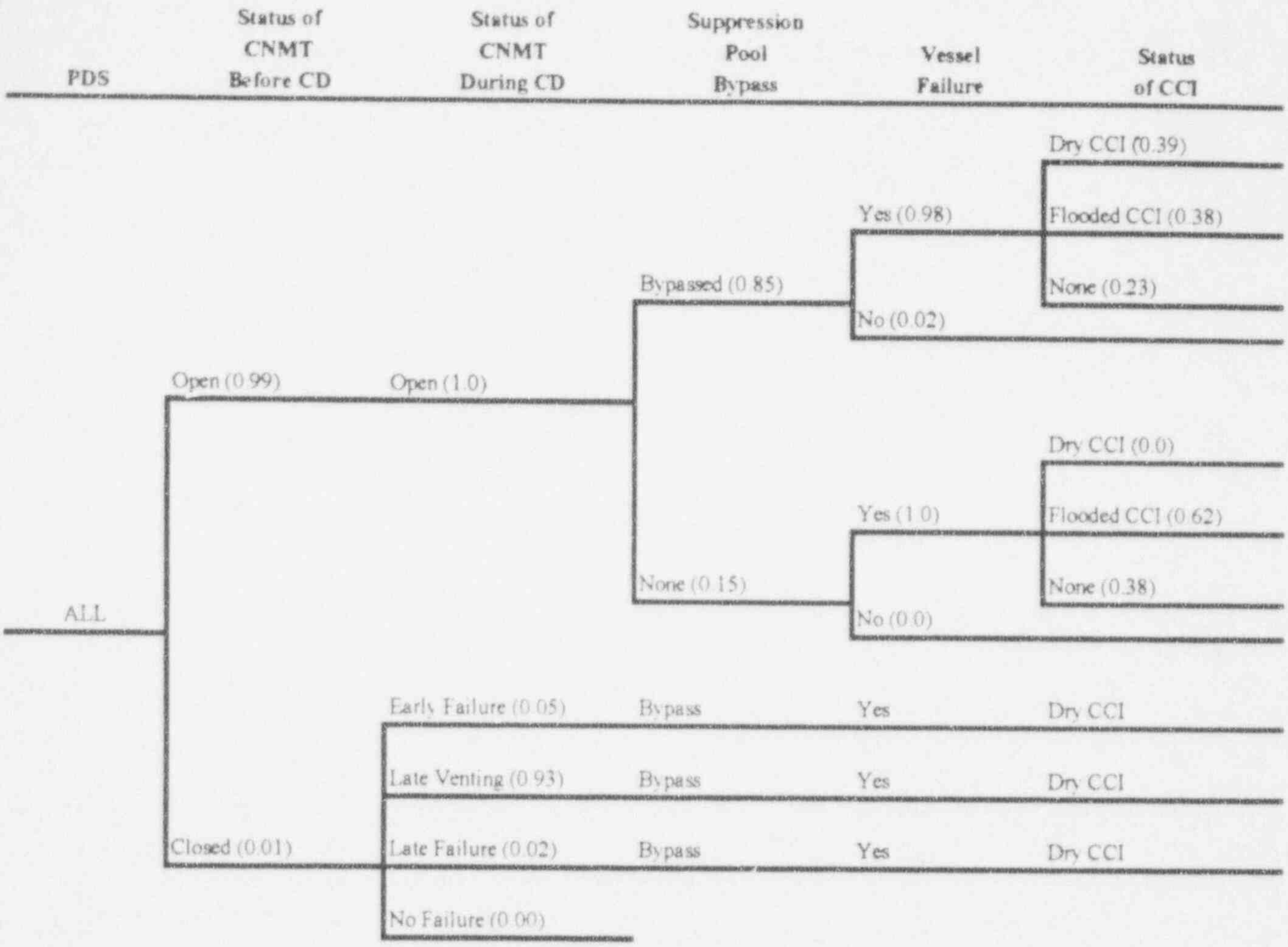

Figure 1-1
Simplified Representation of POS 5 Accident Progressions

the auxiliary building, considerably more radioactive material would be released to the environment.

- The accidents delineated for these shutdown conditions progress more slowly and, therefore, there is generally more time for the public to respond to the accident and evacuate before they are exposed to the release. This is primarily important for the early health effects consequence measures which are more strongly affected by the time available to evacuate.

- Radioactive decay has reduced the radioactive potential of these shutdown accidents relative to the inventory that is present at shutdown. This factor is primarily important for early health effects which are more strongly affected by the shorter lived radionuclides. This affect is much less noticeable for latent health effects which are more strongly affected by the long lived isotopes.

To place the risk from POS 5 into context, it was compared to the risk from power operation as estimated in NUREG-1150 [Brown, et al., 1990] and was also compared to the NRC quantitative safety goals. From Figure 1-2, it can be seen that the risk from POS 5 is comparable to the risk from power operation. While the mean risk from POS 5 is

Table 1-2
Distributions for Aggregated Risk for POS 5
(All values are per year)
(Population doses are in person-rem)

Consequence Measure	Descriptive Statistics				
	5th PCT	50th PCT	95th PCT	MEAN	STD Dev.
Core Damage Frequency	4.1E-07	1.4E-06	5.6E-06	2.1E-06	2.7E-06
Early Fatality Risk	3.7E-11	2.8E-09	3.9E-08	1.4E-08	5.4E-08
Total Latent Cancer Risk	4.3E-04	1.9E-03	1.2E-02	3.8E-03	7.7E-03
Population Dose within 50 miles of the plant	1.3E-01	5.3E-01	3.1E+00	9.9E-01	1.9E+00
Population Dose within 1000 miles of the plant	9.9E-01	4.4E+00	2.8E+01	8.7E+00	1.8E+01
Individual Early Fatality Risk-- 0 to 1 mile	4.2E-13	2.7E-11	3.0E-10	9.6E-11	3.4E-10
Individual Latent Cancer Risk-- 0 to 10 miles	2.5E-10	9.4E-10	4.9E-09	1.6E-09	2.4E-09

greater than the mean risk from full power operation, there is considerable overlap between the distribution suggesting that any difference that exists is small. In Figures 1-3, the risk from POS 5 is also shown to be well below the safety goals. While the safety goals do not necessarily apply to selected modes of operation (i.e., ideally they should be compared to the plants total risk), a comparison of POS 5 risk to the safety goals does provide an indication, in an absolute sense, of the risk associated with this mode of operation.

The mean fractional contributions to risk from the LOCA, SBO, and Other Transient PDS groups are provided in Table 1-3. The SBO PDS group is the dominant contributor to the early fatality risks (total and individual). The SBO PDS group's large contribution to early fatality risk can be attributed to its relatively high contribution to the core damage frequency coupled with the fact that the containment equipment hatch is off in these accidents, the suppression pool is bypassed, and the auxiliary building fails early in the accidents. Combined, these factors cause the SBOs to have relatively high risk values. The LOCA PDS group, however, is not a dominant contributor to early fatality risk even though it is a dominant contributor to the core damage frequency. This stems primarily from the fact that the accidents that are the dominant contributor to the LOCA core damage frequency occur in POS 5 after core alterations (i.e., many weeks after shutdown) by which point the amount short-lived radionuclides that are important to early health have been significantly reduced by radioactive decay.

Table 1-3
Mean Fractional Contribution to Aggregate Risk for the LOCA, Station Blackout (SBO), and Other Transients Plant Damage State Groups

PDS Groups	Core Damage Frequency	Early Fatalities	Total Latent Cancers	Population Dose (<50 miles)	Population Dose (<1000 miles)	Individual Early Fatalities (0-1miles)	Individual Latent Cancers (0-10 miles)
LOCA	0.51	0.16	0.42	0.43	0.41	0.17	0.51
SBO	0.33	0.73	0.45	0.42	0.45	0.70	0.35
Other	0.17	0.12	0.13	0.15	0.14	0.12	0.14

Figure 1-2
Comparison of POS 5 Risk with Full Power Risk

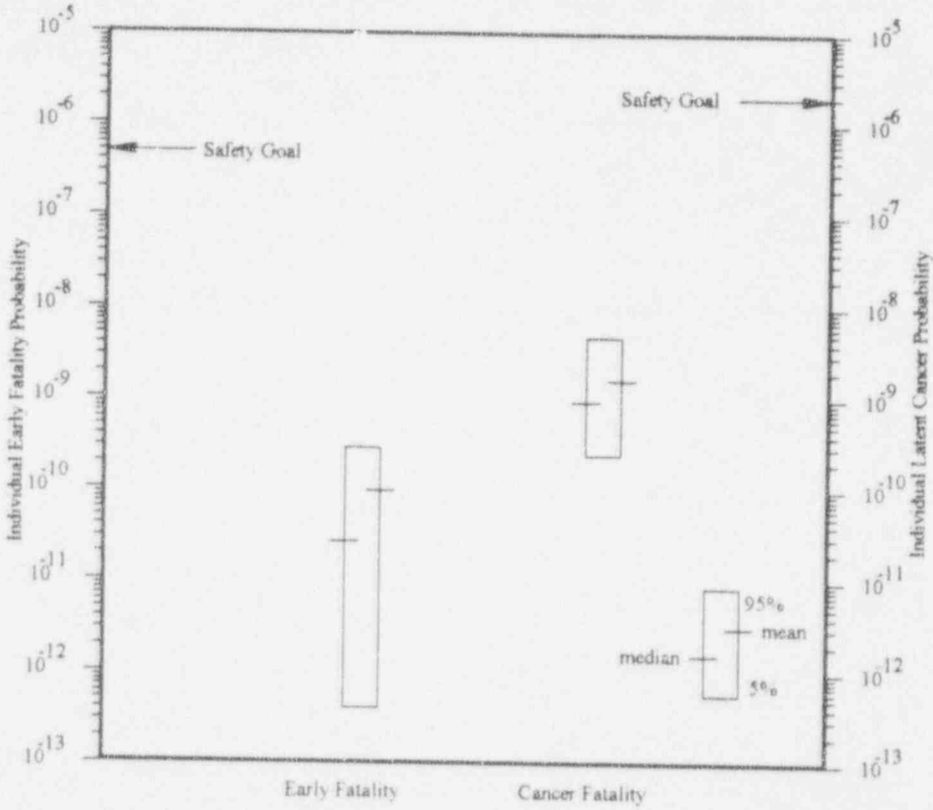

Figure 1-3
Comparison of Individual Risks with the Safety Goals

For latent cancer health effects, the LOCA and SBO PDS groups are the dominant contributors to risk. Since the radionuclides that are important to the latent health effects are the long lived radionuclides, these risk measures are not particularly sensitive to when the accident occurs relative to shutdown. Latent cancers primarily depend on the total amount of radioactive material released and not on when it was released (i.e., early in the accident versus late in the accident). Since latent cancers are not strongly dependent on the timing characteristics of the accident (i.e., start of release or release duration), the latent cancer risk will depend on the likelihood of the accident and on the total amount of radioactive material released. In all of the accidents delineated in this study, the containment is either open at the start of the accident or fails during the accident and in most of the accidents the core damage process is not arrested in the vessel. Although the timing of the accident may vary, all of the accidents have the potential to release a significant amount of radioactive material to the environment. Hence, the mean fractional contribution to latent cancer risk tends to be roughly proportional to the contribution to the core damage frequency. The fraction contributions from the LOCA and Other Transients tend to be less than there fractional contribution to the core damage frequency because for these PDSs portions of the release are scrubbed by either the suppression pool or by water in the reactor cavity. The fractional contribution from the SBO PDS group tends to be greater than the fractional contribution to the core damage frequency because for these accidents the containment is open at the start of the accident, the auxiliary building fails early in the accident, vessel always fails, CCI always occurs and none of the releases are scrubbed by water. Therefore, the releases associated with the SBO tend to be large relative to the other accidents analyzed in this study.

1.4 Issues

The study presented in this volume is for a single POS (namely POS 5) and, as such, assesses the risk associated with this POS. This study does not, however, attempt to assess the risk with the entire LP&S regime of operation. While the Level 1 screening study and other qualitative insights suggest that POS 5 is the risk dominant mode of shutdown, no detailed study has been performed on the other POSs to confirm this belief.

It is important to realize that reducing the risk in one POS, for example by changing when equipment is available and unavailable, can shift the risk to another POS. Since this study only addresses the risk associated with one POS, the affect of this change on overall risk (i.e., risk across all the POSs) cannot currently be quantitatively assessed.

Since only a single plant was analyzed, these results cannot be considered generic and applicable to a population of plants. The plant and system models used in this study are based on the Grand Gulf plant as it operates in a selected mode of operation. Thus, while some insights may be applicable to other plants, in general, the results from this study should not be arbitrarily applied to other plants or conditions. The model used to develop the progression of the accidents after the onset of core damage is, in part, based on the Grand Gulf Emergency Operating Procedures and other procedures and practices at the plant. Changes in these procedures and practices can certainly affect the progression of the accident and the ultimate risk of the POS. Similarly, since the offsite consequences are sensitive to the site characteristics and surrounding region (e.g., weather, population, land usage), for a given release of radioactive material, the consequences can be expected to vary from one site to the next.

1.5 Conclusions

The following conclusion can be drawn from this study:

- Without many plant features available to mitigate a release, the potential exists for a large release of radioactive material should core damage occur. For the most likely accidents the containment is open, the suppression pool is bypassed, and the containment sprays are not available. The auxiliary building is one of the few plant features that is available to attenuate a release.

- In the event that the containment is closed prior to the onset of core damage, it is always predicted to fail since containment heat removal was not available in the analyzed accidents.

- The risk associated with the operation of the plant during POS 5 is comparable to full power risk and the individual risks are well below the safety goals. The low values for risk given the high conditional releases described above are, in part, due to the extremely low core damage frequency and the sparse population around the plant.

- Although only a simplified scoping study of the onsite consequences was performed, the possible onsite consequences of an accident during shutdown could be significant, particularly since in many of the accidents the containment remains open allowing for an early release of radioactive material.

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1.6 References

[Whitehead, et al., 1991] D. W. Whitehead et al., "BWR Low Power and Shutdown Accident Frequencies Project, Phase 1 - Coarse Screening Analysis", Vol. 1, Draft Letter Report, Sandia National Laboratories and Science and Engineering Associates, Inc. November 23, 1991 update, Copy available in the NRC Public Document Room.

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2 Introduction

2.1 Background

The safety of commercial nuclear plants during power operation has been previously assessed in many probabilistic safety assessment studies. The U.S. Nuclear Regulatory Commission (NRC) has been an active participant in these studies including the landmark Reactor Safety Study [USNRC, 1975], the five plant studies performed as part of the NUREG-1150 study [USNRC, 1990] and the LaSalle plant analysis performed under RMIEP/PRUEP programs [Payne, 1992 and Brown, et al., 1992]. Furthermore, all licenses are required to perform an individual plant examination which also assesses the safety of the plant during full power operation.

Recent events at several nuclear power generating stations, recent safety studies, and operational experience, however, have all highlighted the need to assess the safety of plants during low power and shutdown modes of operation. In contrast to full power operation, there is very little information on the safety of plants during low power and shutdown modes of operation. In the past, the assumption has been that power operation is the risk dominant mode of operation because the decay energy is greatest at the time of shutdown and then decays as a function of time. Thus, the rationale was that during shutdown modes of operation the decay heat would be sufficiently low that there would be plenty of time to respond to any abnormal event that may threaten the core cooling function. Furthermore, given the unlikely event that a release did occur, radioactive decay would lessen the radiological potential of the release. This argument's Achilles' heel is that the technical specifications allow for more equipment to be inoperable in off power conditions. Thus, while there may be more time to respond to an accident during shutdown, many of the systems that are relied on to mitigate an accident during power operation may not be available during shutdown.

To gain a better understanding of the risk significance of low power and shutdown modes of operation, the Office of Nuclear Regulatory Research at the NRC has undertaken a two phase program to analyze the frequencies, consequences, and risk of accidents occurring during modes of operation other than full power. To investigate the likelihood of severe core damage accidents during off power conditions, probabilistic risk assessments (PRAs) were performed for two nuclear plants: Unit 1 of the Grand Gulf Nuclear Station which is a BWR-6 Mark III boiling water reactor (BWR) and Unit 1 of the Surry Power Station which is three loop, subatmospheric, pressurized water reactor (PWR). This report discusses the analysis that was performed on the Boiling Water Reactor.

Phase 1 of the project was completed in September of 1991 [Whitehead, et al., 1991]. This phase involved a coarse screening of potential accidents that could occur at a BWR while operating a other than full power and was adopted as a means of obtaining, in a relatively short time, some estimate of the potential for accidents during low power and shutdown conditions. The coarse screening analysis indicated that risk during these conditions cannot be shown to be insignificant by a conservative screening analysis. Hence, the NRC decided to have follow-on detailed analyses performed. Consequently, the NRC decided to first perform a detailed analysis consisting of a Level 3 PRA on one of the off power conditions. Based on trends indicated in the results of the coarse screening analysis, plant operational state (POS) 5 (consisting mainly of Cold Shutdown Operating Condition) was selected for detailed analysis. (NOTE: Plant operational states are artificial subdivision of the time plants spend in low power and shutdown conditions. This concept was developed during Phase 1 of this project to all ow the analyst to better represent the plant as it transitions form power operation to non power operation.) The report presents the results of the detailed analysis of the Grand Gulf facility in POS 5 during a refueling outage.

There are several reasons to perform a Level 3 PRA and produce estimates of risk as opposed to only estimating the likelihood of core damage. The configuration of the plant during shutdown can be very different from the configuration during full power which can lead to drastically different accident progressions and releases of radioactive material. Hence, it is not sufficient to determine the frequency of core damage during shutdown and then infer the risk based on PRA results from full power operation. If the core damage frequency associated with shutdown is less than the full power core damage frequency, for example, it is not valid to infer that the risk will also be less since the consequence could be higher during shutdown. Also, to be able to assess the safety of shutdown modes of operation it will be necessary to (1) determine the relative importance of shutdown risk as compared to the risk of full power operation, and (2) compare the shutdown risk with the NRC's safety goals; the safety goals are expressed in terms of risk. The risks

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associated with the operation of these two plants at full power were characterized in the NUREG-1150 study [USNRC, 1990].

The analysis of the BWR was conducted at Sandia National Laboratories while the analysis of the PWR was performed at Brookhaven National Laboratory. The LP&S PWR analysis is reported in NUREG/CR-6144 [Chu, et al., 199?] and will not be discussed any further in this report. This multi-volume report presents and discusses the results of the BWR analysis. Volume 1 is a summary of the BWR study. Volumes 2-5 present the accident frequency analysis (i.e., Level 1) performed under FIN L1923. Volume 6 presents the Level 2/3 analyses performed under FIN L1679. Part 1 of Volume 6 presents and discusses the accident progression, radionuclide release and transport, consequence, and risk analyses. Part 2 of Volume 6 presents the deterministic code calculations, performed with the MELCOR code [Summers, et al., 1991], that were used to support the development and quantification of the PRA models.

2.2 Study Objectives

The primary objective of this study was to perform an analysis of potential accidents that could occur at Grand Gulf while the plant is in POS 5 during a refueling outage. Additional specific objectives for this study include:

- For POS 5, perform a preliminary characterization of the accident progressions following core damage and estimate the consequences that result from these accidents.
- Determine quantitatively the risk and estimate the uncertainty in risk for the risk significant mode of operation.
- Compare the risk associated with POS 5 to the safety goals and to the risk associated with full power operation.
- Provide an assessment of the potential for a radioactive release to cause onsite consequences.

This study will address the following types of questions:

- What are the characteristics of accident progressions for the selected regime of shutdown? Are there any significant differences between these progressions and progressions typical of full power accidents? What are the dominant phenomena?
- What is the risk associated with this regime of operation and how does this compare to the safety goals and to full power operation? What are the risk significant configurations? Can anything be done to reduce this risk?
- What is the potential for releases from these accidents to cause onsite consequences? Is the plant in a particularly vulnerable configuration (i.e., containment open, large numbers of people in the vicinity of the plant)?

2.3 Scope of Study and Major Assumptions

2.3.1 Study Scope

The study reported in this volume is the Level 2/3 portion of a Level 3 PRA that was performed to investigate the risk associated a selected regime of shutdown. The Level 2/3 portion of the PRA is concerned with the progression of postulated accidents following the onset of severe core damage and the estimation of the consequences that result from the release of any radioactive material and, as such, consists of the following constituent analysis: plant damage state (PDS), accident progression, source term, consequence, and risk integration. The Level 2/3 portion of this PRA utilized an *abridged* form of the NUREG-1150 methodology in that simplified models were used to perform the accident progression analysis and only a limited uncertainty analysis was performed. This analysis is *focused* in that the models developed and used in this analysis (e.g., accident progression event tree and parametric source term model) are specific to accidents represented by the PDSs and no effort has been expended to make the models general in the sense that they would be applicable to any type of postulated accident or other plants. The advantage of the focused approach is that resources are expended on the accidents that are of concern, the disadvantage to this approach is that if a different type of accident is postulated after the models are constructed, the models must be modified. Other point that define the scope of the analysis include:

- The consequence analysis includes the traditional offsite assessment (i.e., similar to the offsite consequence assessment performed in the NUREG-1150 plant studies) and will also include a scoping assessment of onsite consequences.

- The simplified uncertainty analysis will only include *issues* thought to be the most important to risk and will only include issues from the accident progression and source term analyses. (The uncertainty in the core damage frequency will be propagated through the analysis via the uncertainty in the plant damage states). For many of these issues, distributions developed during the NUREG-1150 project will be used. If an applicable distribution from an existing study does not exist, the project staff will develop the appropriate distribution. Formal expert elicitation techniques (i.e., the use of panels of experts from a variety of organizations and companies) **will not** be used to select issues nor be used to develop distributions.

- Only one plant was analyzed in this study. The plant selected for this study is Unit 1 of the Grand Gulf Nuclear Station.

- While the low power and shutdown modes of operation encompass many plant and system configurations, only the cold shutdown regime of operation during a refueling outage was investigated in detail in this study. This regime of shutdown is referred to as plant operationg state (POS) 5. This mode of operation was selected because of its importance in the Level 1 coarse screening analysis [Whitehead, et al., 1991] and a qualitative assessment of its importance relative to the other modes of operation. For more discussion on the rationale used to select POS 5 for this study, see Section 3 Volume 2, Part 1.

- Only accidents initiated by traditional internal events were analyzed (i.e., accident initiated by internal fire, internal flood are not included in this study).

Abridged Analysis of POS 6: An abridged PRA of POS 6 (i.e., refueling mode of operation prior to fuel movement) was performed in the spring of 1992. POS 6 was selected because the containment and vessel are both open during this mode of operation; this plant configuration was of particular interest to the NRC. The scope of the abridged PRA of POS 6 was considerably narrower than the abridged analysis of POS 5. The POS 6 analysis relied on very simplified plant damage state, accident progression and uncertainty analyses. Compared to the POS 5 analysis, the POS 6 accident progression analysis considered fewer issues and addressed them in less detail. Since the PDSs were based on results from the Level 1 coarse screening study [Whitehead, et al., 1991], which only grouped core damage sequences into categories and did not provide frequency estimates, the POS 6 results were conditional on the occurrence of the PDS and, therefore, estimates of risk were not calculated. While the abridged analysis of POS 6 was very limited in scope and detail, it did, however, provide insights into the impact that the configuration of the plant has on the progression of the accident and the resulting consequences. For the sake of completeness, this study is presented in Appendix F. It is important to remember that the scope, level of detail, methods and assumptions used in the analysis of POS 6 are very different from those used in the analysis of POS 5.

2.3.2 Major Assumptions

During the course of the study it was necessary to make a number of assumptions to keep the analysis manageable. Many of these assumptions, if changed, could have a significant impact on the results. While many assumptions are listed in the individual analysis chapters, a list of the more significant assumptions is presented below. The major assumptions used in the Level 1 analysis are listed in Volume 2 of this report.

- During POS 5 the plant is in the cold shutdown mode of operation with the vessel head attached. The variability of the plant configuration and system availability during POS 5 can be adequately captured by dividing the cold shutdown mode of operation into 3 segments or "time windows"

- Each time window has a characteristic decay heat load and radionuclide inventory.

- Accidents assigned to a particular time window are assumed to be initiated at the start of the time window.

- Core damage is defined in the Level 1 analysis as the start of fuel heatup. For the sake of consistency this same definition was used in the Level 2/3 analysis.

- The mission time for the Level 1 analysis was 24 hours. Thus, sequences for which core damage did not occur within 24 hours from the start of the accident were dropped from the analysis.

- DC power from the station batteries is required to restore offsite power to the plant in the event that offsite power is lost prior to or during the accident. Therefore, if the station batteries deplete prior to the restoration of offsite power, it is assumed that offsite power cannot be restored during the accident.

- At the start of the accident the reactor pressure vessel head vent is open. This vent can be closed prior to core damage if ac power is available.

- The containment and drywell are both open at the start of the accident. The containment can only be closed if offsite ac power is available; containment closure must be completed prior to the onset of core damage. The drywell is assumed to remain open throughout the accident.

- If the containment fails, it is assumed to fail above the auxiliary building roof, thereby allowing radioactive releases to bypass the auxiliary building and enter the environment directly. The enclosure building that surrounds the portion of the containment that is above the auxiliary building roof is estimated to offer essentially no attenuation to the release. This assumption is consistent with the assumption used in the Grand Gulf plant analysis performed as part of the NUREG-1150 study [Brown, et al., 1990].

- If the containment is closed prior to the onset of core damage and then subsequently vented after the onset of core damage, it is assumed that the vent stays open throughout the accident. While the emergency operating procedures (EOPs) direct the operators to close the containment once its pressure drops below a certain pressure, without containment heat removal, the containment will have to be vented again later in the accident when the pressure again increases above the vent pressure. There was no attempt in this analysis to model the opening and closing of the containment vent. Furthermore, since the availability of the containment purge system during POS 5 is not being modelled (and it is not required by the technical specification), it is assumed that the containment will be vented directly to the environment and will not pass through the containment purge system with its associated filters and charcoal beds.

- If the containment equipment hatch and/or personnel locks are open, the airborne radioactive material will enter the auxiliary building prior to being released to the environment. The auxiliary building is assumed to fail on a 5 psi overpressure. Neither the standby gas treatment system nor the ventilation system are modelled. Thus, it is assumed that no engineered features are available to attenuate the release in the auxiliary building and the only attenuation that the release will experience in the building is that due to natural processes (e.g., natural deposition).

- The Grand Gulf Emergency Operating Procedures are applicable after the onset of core damage and the operators will continue to follow them.

- No operator actions were modelled that would require the operators to enter the containment or auxiliary building following the onset of core damage.

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- Recovery of coolant injection after the onset of core damage is only considered if (1) injection systems were not available prior to core damage but become available following core damage, or (2) conditions occur following the onset of core damage that would cause the operators to use a system that was previously available but not used prior to core damage.

2.4 Strengths and Limitations

As with any study, this study has it strengths and limitations. In order to place this study in its proper context and to use the results appropriately, it is necessary to be aware of these attributes of the study. The strengths are listed first, followed by the study's limitations.

The strengths of this study include:

- The analysis is a Level 3 PRA that accounts for the progression of the accident following the onset of core damage, the release of radioactive material from the core and its transport through the primary system and the containment, and the transport of the radioactive material in the environment and the resulting health effects. The PRA techniques allow the many possible types of accident to be delineated and systematically evaluated.

- Risk results are calculated.

- The analysis includes a detailed coupling with the accident frequency analysis via the plant damage state analysis.

- An estimate of the uncertainty in risk that results from the uncertainties associated with input parameters to the accident progression analysis and source term analysis is calculated and displayed.

- The study includes a limited assessment of onsite consequences. Doses and dose rates within the site boundary that result from a radioactive release are estimated.

- A Human Reliability Analysis (HRA) was performed to quantify the human error probabilities (HEPs) associated with key operator actions during the progression of the accident following the onset of core damage.

- MELCOR calculations were used to support the development and quantification of accident frequency, accident progression, and source term analyses. Wherever possible, consistent calculations were used to quantify the Level 1 and Level 2/3 models.

The limitation of this study include:

- Since only a single plant was analyzed, these results cannot be considered generic and applicable to a population of plants. The plant and system models used in this study are based on the Grand Gulf plant as it operates in a selected mode of operation. Thus, while some insights may be applicable to other plants, in general, the results from this study should not be arbitrarily applied to other plants or conditions. The model used to develop the progression of the accidents after the onset of core damage is, in part, based on the Grand Gulf Emergency Operating Procedures and other procedures and practices at the plant. Changes in these procedures and practices can certainly affect the progression of the accident and the ultimate risk of the POS. Similarly, since the offsite consequences are sensitive to the site characteristics and surrounding region (e.g., weather, population, land usage), for a given release of radioactive material, the consequences can be expected to vary from one site to the next.

- Only a single POS was analyzed and, therefore, the risk of the entire low power and shutdown regime of operation was not assessed.

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- While it is believed that progression of the accident following core damage is adequately model for the intended purposes, the accident progression model is not as detailed as the models developed in the NUREG-1150 study or the PRUEP study.

- The uncertainty in the values for input parameters to the consequence models was not characterized. The only uncertainty included in the consequence assessment was the stochastic variability in the weather at the time of the accident; this uncertainty was only accounted for in the offsite consequence assessment.

- Formal use of expert elicitation techniques were not used to quantify the uncertainty in input parameters. Distributions for important parameters were either developed by the project staff or obtained from existing PRAs (i.e., primarily NUREG-1150).

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3 Methodology

3.1 Background

The probabilistic risk assessment (PRA) performed in this study utilizes methods developed and applied in the NUREG-1150 study [USNRC, 1990] and the PRUEP study [Brown, et al., 1992]. A primary objective of the NUREG-1150 study was to provide a current assessment of the risks of five nuclear power plants of different designs and to provide quantitative estimates of the risk uncertainties. To achieve this objective, an analytical PRA framework was developed that allowed accid· · s to be modelled at a level of detail that was consistent with the state of knowledge and that also allowed uncertaintı· s in important physical and chemical phenomena to be characterized. Areas where significant advancements in the state-of-the-art were made included the consistent and comprehensive treatment of uncertainty in all areas of the PRA, the development of detailed accident progression and source term models, and the development and application of expert judgement techniques to assess the likelihood and nature of rare and complex phenomena associated with severe core damage accidents. NUREG-1150 was peer reviewed and when it was published in 1990 it represented the state-of-the-art in Level 3 PRA. While improvements have been made to the PRA models and additional data on severe accidents have been acquired since NUREG-1150 was published, the general methodology was judged to represent the state-of-the-art in Level 3 PRAs. Therefore, its framework was used, to the extent possible, in this study.

The objectives of the NUREG-1150 study lead to an analysis approach based on the following ideas:

o general and relatively fast-running models for the individual analysis components,

o well defined interfaces between the individual analysis components,

o use of Monte Carlo techniques in conjunction with an efficient sampling procedure to propagate uncertainties,

o use of expert panels to develop distributions for important phenomenological issues (as explained later, this aspect of the NUREG-1150 approach was not used in this study),

o automation of the overall analysis.

An overview of the analysis approach is presented in Section 3.2. A simplified version to the NUREG-1150 approach was applied to study the risk of POS 5. The significant differences between the NUREG-1150 methods and the methods employed in this study are discussed where appropriate.

3.2 Overview of Analysis Approach

Since the NUREG-1150 methods are described in detail elsewhere [Ericson, et al., 1990], [Gorham, et al., 1994], this section will only provide a brief overview of these methods. Much of this discussion has been extracted from Volume 1 of NUREG/CR-4551 and a related journal article [Breeding, et al., 1992].

The NUREG-1150 plant studies are fully integrated probabilistic risk assessments which can be characterized as consisting of four analysis components, a risk integration component and an uncertainty analysis component. The first component is the accident frequency analysis which determines the likelihood and nature of accidents that result in a loss of cooling to the fuel and that subsequently lead to fuel damage. This state of the core is referred to as core damage. The second component is the accident progression analysis which determines the progression of the accident following the onset of fuel damage and addresses the response of engineered barriers, such as the reactor pressure vessel and the containment, to loads that occur during the accident. The third component is the radioactive material release and transport analysis (this analysis is also called the source term analysis) which determines the amount of radioactive material released during the accident and its subsequent transport and deposition in the engineered environment (e.g., containment). The fourth component is the consequence analysis which determines the transport of radioactive material outside the plant and estimates the health effects and costs associated with the release of this radioactive material. The fifth component is risk integration which assembles the results of the preceding analysis components into an overall expression of risk. The sixth and final component is the uncertainty analysis which estimates the uncertainty in the risk

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results due to uncertainty in the characterization of important physical and chemical phenomena. Measures of uncertainty in risk are obtained by repeating the calculations just indicated many times with different values for important parameters selected randomly by a form of Monte Carlo sampling. This provides a distribution of risk estimates that is a measure of the uncertainty in risk.

The representations of risk used in this report are defined in the next section. Following the definition of risk is a description of the individual analysis components and a description of the treatment of uncertainty in the PRA.

3.2.1 Representation of Risk

Two representations of risk are used in this report. The first definition is the order set of triples [Kaplan and Garrick, 1981] of the form:

$$R = \{ (s_i, f_i, o_i), i=1,...nS\} \qquad (3.1)$$

where

s_i	=	a scenario (i.e., accident) that leads to an outcome (i.e., consequence) of interest,
f_i	=	frequency (units: yr^{-1}) for scenario i,
o_i	=	outcome associated with scenario i,

and

nS	=	number of scenarios under consideration.

The objective of a probabilistic risk assessment is the determination of the triples that constitute the set R. Combined, the accident frequency and accident progression analysis define the accident scenarios, s_i, and the frequency of each scenario, f_i. Similarly, the source term analysis and the consequence analysis combine to determine the outcome or consequence of the accident scenario, o_i. When complicated technical systems, such as nuclear power plants, are analyzed the number of scenarios can be quite large (i.e., thousands) at which point it is neither practical to present the risk results in this form nor reasonable to expect one to draw conclusions or make decisions based on this representation. In such cases it is convenient to collapse the set of triples into a value for aggregate risk. Aggregate risk is defined as

$$r = \sum_{i=1}^{nS} f_i o_i \qquad (3.2)$$

Although aggregate risk is appealing because it is a summary measure of the order set of triples, it is important to recognize that information is lost when the order set of triples is converted into an expression for aggregate risk. To assess the importance of various contributors to aggregate risk, it will often be necessary to examine the constituents of the risk triple. Thus, both representations of risk are employed in a PRA.

The term risk is often used loosely and may refer to any number of possible products of the PRA, for example, the core damage frequency or the probability of containment failure. To avoid ambiguity, in this volume, the term risk will refer to aggregate risk (i.e., defined in equation 3.2) where the outcome of interest, o_i, represents the offsite consequences that result from the release of radioactive material from the plant. Since there are various measures of offsite consequences (e.g., early fatalities, total latent cancer fatalities, population dose within 50 miles of the plant, etc.), there will be an equal number of risk measures. The various consequence measures used in this study are discussed in Section 3.2.2.4.

3.2.2 Description of Analysis Components

3.2.2.1 Accident Frequency Analysis

The accident frequency analysis uses event tree and fault tree techniques to identify the combinations of events that can lead to core damage and to estimate their frequencies of occurrence. On a system level, these combinations of events are denoted "sequences." On an individual fault level (e.g., failures of specific pumps and valves), these combinations of events are called "cut sets." The cut sets of interest are those which contain no more faults than those required to cause core damage. These cut sets are denoted "minimal cut sets." The cut sets are identified by means of fault trees, and the minimal cut sets are sorted into accident sequences by means of event trees. The frequency of an accident sequence is obtained by combining the frequency of the initiating event with the sum of the probabilities of all the minimal cut sets in the sequence. The accident frequency analysis that was performed as part of this program is discussed in detail in Volumes 2-5 of this report.

Typically, there are too many sequences to individually analyze in the subsequent analyses. Furthermore, within the resolution of the PRA, many of these sequences will result in similar progressions following core damage and, therefore, it would be redundant to individually analyze each sequence. Instead, accident sequences that provide a similar set of initial and boundary conditions for the subsequent accident progression analysis are grouped into a plant damage state (PDS). The PDSs form the interface between the accident frequency analysis and the accident progression analysis. The number of PDSs that are defined for a given analysis depends on the diversity of accident sequences and the resolution desired in the subsequent analyses. In some cases the definition of the PDSs does not correspond exactly to the accident sequence definitions so that it may be necessary to plac[illegible] ie minimal cut sets from a sequence in one PDS and the remaining minimal cut sets in another PDS. The frequ[illegible] of a PDS is the sum of the frequencies of the minimal cut sets that it contains.

3.2.2.2 Accident Progression Analysis

The purpose of the accident progression analysis is to represent the progression of the accident from the point of core damage until the completion of the release of radioactive material from the containment. This analysis models the response of the radioactive barriers (e.g., reactor pressure vessel and containment) to the stresses placed upon them during the various phases of the accident. The accident progression analysis uses an event tree called an accident progression event tree or APET to determine the possible ways in which an accident might evolve from each PDS. Each different progression is represented by a different path through the APET. The definition of each PDS provides enough information to define the initial conditions for the APET. Past observations, experimental data, and mechanistic code calculations are used in the development of the model for the accident progression that is embodied in the APET. These same sources of information were utilized in determining the probabilities at the branch points in the APET.

The APET developed for POS 5 is similar in concept and structure to the APETs developed in NUREG-1150 study, however, it is not as detailed. As compared to the NUREG-1150 APETs, the abridged APET included fewer questions (i.e., top events), issues were addressed in less detail (e.g., hydrogen combustion phenomena), and formal expert judgement procedures were not used to quantify the APET. Some of this simplification was possible because the configuration of the plant during POS 5 precluded or minimized the need to address certain issues (e.g., once the drywell equipment hatch has been removed it is no longer necessary to determine the structural response of the drywell to loads that occur during the accident). In other cases, it was necessary to make simplifying assumptions in order to keep the study manageable. While the analysis of POS 5 did not use formal expert judgement techniques to quantify issues (e.g., branch point probabilities) in the accident progression analysis, it did make extensive use of distribution developed in NUREG-1150. Many of these NUREG-1150 distributions were generated using formal expert judgement techniques.

In the NUREG-1150 plant studies, the APET was evaluated for each PDS individually and, therefore, the results from these individual analysis were conditional on the occurrence of the PDS that was evaluated. The analysis of POS 5 adopted the approach used in the PRUEP study [Brown, et al., 1992]; a single APET was developed for all of the various PDS and it was only evaluated once. In this case the APET analysis is conditional on the occurrence of core damage, not any particular PDS. This approach was taken because issues that affected many PDS could be easily treated in a consistent manner, the truncation of individual accident progression paths could be treated in a consistent manner across all PDSs, and the management of output files is simplified.

The APET was quantified using information from: (1) the Level 1 analysis (see Volumes 2-5 of this report), (2) a human reliability analysis performed for this study to determine human error probabilities associated with operator actions during core damage; (3) MELCOR calculations performed specifically for this study (see Volume 6, Part 2 of this report), and data from the NUREG-1150 Grand Gulf plant study [Brown, et al., 1990]. For those events that were judged by the project staff to be important to risk and for which there was a large amount of uncertainty as to the value to assign to the branch probability, a distribution of probabilities were assigned to the branch. No rigorous analytical process was used to select these events. Rather, the events were selected based on prior experience and results from existing PRAs. For the remaining events that were either judged to be less important or for which the branch probability was not believed to uncertain, a single value was used.

The interface between the accident progression and source term analysis is defined through accident progression bins (APB). Due to the large number of questions in a typical APET and the fact that many of these questions have more than two branches, there are far too many paths through the tree to permit each path to be considered in the subsequent source term and consequence analyses. Furthermore, many of the progressions (i.e., event paths) developed with the APET will lead to similar source terms (i.e., within the resolution of the analysis). Therefore, to avoid performing redundant calculations, similar accident progression paths are collected into groups called APBs. Each APB defines a set of unique initial and boundary conditions for the source term analysis.

The codes EVNTRE [Griesmeyer, et al., 1989] and PSTEVNT [Higgins, 1989] were used to evaluate the APET and process the results. The products of the accident progression analysis are the accident progression bin definitions, their associated probabilities conditional on core damage, and an expression of the uncertainty in these probabilities.

3.2.2.3 Source Term Analysis

The source term analysis models the release and transport of radioactive material from the fuel and core debris to the environment; it is performed so that the radiological severity of the accident can be assessed. The product of this analysis is a collection of parameters, referred to as the source term, that characterizes the type and amount of radioactive material released from the containment, the start and duration of the release, and the location of the release. In this context, containment is generalized to include the region in which engineered barriers are available to attenuate a release before it enters the environment (e.g., auxiliary building that surrounds the containment building), and the environment is the region beyond the containment. The inputs to the source term analysis are the accident progression bins (APBs) defined in the accident progression analysis. The APBs describe the configuration of the plant, the status of systems that can be used to mitigate the release, and the occurrence of phenomena that can impact the source term. A source term is calculated for each APB. Although the source term analysis follows the accident progression analysis, the two are actually intimately coupled in that the release and transport of the radioactive material occurs during the accident progression and is affected by phenomena that occur during this phase of the accident.

Source terms for the various APBs were estimated using the parametric expressions developed in the NUREG-1150 study. These parametric expressions are implemented in a set of codes that are collectively known as XSOR [Jow, et al., 1993], [Cybulskis, et al., 1989]. These codes are similar in approach, however, a different code is developed for each plant analysis in order to reflect the features unique to each plant. The parametric code GGSOR [Jow, et al., 1993], [Brown, et al., 1990] was modified to reflect the different plant configuration and the different possible release paths during shutdown. In this approach a simple parametric model, which is based on results from mechanistic code calculations and other sources of information, is created and used to calculate a source term for each APB. Parametric codes use a combination of parameters, which represent the physics of the accident at a very general level, to estimate the release to the environment. For example, a parameter can be used to represent the fraction of a particular radionuclide in the fuel that is released to the vessel. Another parameter is then used to represent the fraction of the radionuclide that is in the vessel that is released to the containment. The parametric approach is not meant to be a substitute for detailed calculations. Rather, it is a framework for integrating the results of these codes together with experimental results and expert judgement.

The parametric approach was selected for the following reasons:

- The code is relatively fast running and, therefore, can be used to estimate source terms for each APB.

- The approach is amenable to uncertainty analysis.
- The NUREG-1150 XSOR code for Grand Gulf already exists, thus, the only modifications that would have to be made are those required to take into account the plant configuration during shutdown.
- To a large extent, the XSOR code data base already exists (some modifications/additions were made to the data base to account for unique phenomena and the plant configuration during shutdown).

While this approach also has several disadvantages, which include very little physics explicitly included in the model and the timing of the release is characterized in a very coarse manner, its advantages make it a good choice when the uncertainty in the source term is to be characterized.

The XSOR model was quantified using the following sources of information:

- Wherever appropriate, distribution developed during the NUREG-1150 study were used [Harper, et al., 1992]. MELCOR calculations were used to identify areas where full power accidents are similar enough to shutdown accidents that the use of full power data was reasonable.
- MELCOR calculations were used to determine timing information (e.g., release times and release durations) and the energy associated with the release.

The uncertainties in important input parameters was characterized and propagated through the XSOR model to develop an expression for the uncertainty in the resulting source term. While no new source term issues were quantified with formal expert judgement techniques, an internal "Source Term Advisory Group" was formed to provide guidance on the use of existing methods and data and to review the source term issues being treated and identify any new issues that may be important to shutdown accidents. This guidance was directed at an earlier study (see Appendix F) and was factored into this analysis.

Since the parametric approach results in a source term for every accident progression bin, it is impractical to estimate consequences for each source term individually. Instead, the source terms must be collapsed into a manageable number of groups. Source terms with a similar potential to cause early and chronic health effects are collected into a source term group (STG); a single source term is then defined (e.g., the mean of the group of source terms) for each STG. The STGs were created using the PARTITION algorithm that was first developed during the NUREG-1150 study [Iman, et al., 1990] and then modified in the PRUEP program [Brown, et al., 1992]. Consequences for the STG are then estimated using this representative source term. The source term groups form the interface between the source term analysis and the consequence analysis.

The product of this analysis is the mean source term associated with each source term group.

3.2.2.4 Consequence Analysis

While the source term analysis is the analysis of the release and transport of radioactive material from the fuel and core debris to the environment, the consequence analysis is the analysis of the transport of this material through the environment, the health effects, and the costs that result from the release of this radioactive material. Consequences that would accompany a core damage accident are typically divided into two categories: offsite consequences and onsite consequences. The offsite consequence analysis predicts the health effects to the public and economic impacts that are associated with the dispersal of radioactive materials into the environment beyond the site boundary. The onsite consequence analysis is confined to the region within the site boundary. As such, onsite consequences include health effects to personnel working at the plant at the time of the accident and the cost of replacement power, capital loss, and cleanup of the reactor facility. The consequence assessment for most commercial reactor PRAs is focused on the assessment of offsite consequences because the regulations promulgated and enforced by the NRC emphasize protection of the public. Thus, most of the Level 3 PRAs that are performed, including the NUREG-1150 PRAs, include only an assessment of offsite consequences. Because of this emphasis on offsite consequences, methods to perform offsite consequence assessments are available, have been subjected to considerable review, and have been applied in many

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PRAs This is not so with onsite consequence assessments The consequence analysis for this study includes both a traditional offsite consequence analysis and a limited scope onsite consequence analysis

Offsite Consequences

The offsite consequences to the general public were estimated using the MACCS code [Chanin, et al., 1990], [Jow, et al., 1990], [Rollstin, et al., 1990]. This code models the transport and dispersion of plumes of radioactive material released from the plant. As the plumes travel through the atmosphere, material is deposited on the ground. Various pathways through which the general population can be exposed are considered. Emergency response and protective action guides are also considered as means to mitigate the extent of the public exposure. For this study, the input to MACCS, aside from the source terms and the core inventory, were the same as what was used in the NUREG-1150 Grand Gulf study [Brown, et al., 1990], [Sprung, et al., 1990]. The same offsite consequence measures calculated and reported in NUREG-1150, which are listed in Table 3-1, were also calculated in this study.

Onsite Consequences

In this limited onsite analysis, health effects are not calculated. Instead, doses and dose rates are estimated for a range of distances out to the site boundary using building wake effect correlations (i.e., referred to as parking lot doses). For comparative purposes, two sets of correlations were used. The first correlation was developed by Ramsdell [Ramsdell, 1990] whereas the second set uses a combination of models developed by Wilson [Wilson, 1984] and the NRC [USNRC, 1982]. The calculations were performed for two sets of weather scenarios: one that is stable and another that is unstable. For simplicity, the directional dependence of the weather is ignored. The dose rate is based only on the immersion exposure path whereas the dose is based on inhalation exposure path as well as the immersion exposure path. The total dose is a 50 year committed dose. Two exposure times are assumed: the passage of the entire plume and a 15 minute exposure. The products of this analysis include, for each source term group, the dose rate for each plume segment, the total dose received by a receptor that is exposed to the entire plume, and the total dose received by a receptor that is only exposed to the first 15 minutes of the plume.

3.2.2.5 Risk Integration

Risk to the general public was calculated using the aggregate risk definition. In this calculation, the core damage frequency, the conditional probabilities of the source term groups, and the offsite consequences associated with each source term group are combined. The fractional contribution to aggregate risk from selected PDSs and APBs was also determined. Because of the scoping nature of the onsite consequence analysis, an analogous calculation was not performed to estimate onsite risk. The products of this analysis include the following risk measures: early fatality risk, total latent cancer risk, population dose within 50 miles, population dose within 1000 miles, individual early fatality risk within 1 mile, and the individual latent cancer risk within 10 miles. The uncertainty in each of these measures was also characterized.

3.2.3 Treatment of Uncertainty

An important and distinguishing feature of the NUREG-1150 plant studies was its consistent and comprehensive treatment of the uncertainties in the PRA and its quantitative estimates of the uncertainties in aggregate risk. The types of uncertainties addressed in the PRA and the propagation of these uncertainties through the constituent analysis are discussed in the next two subsections.

3.2.3.1 Types of Uncertainty

In the NUREG-1150 studies, two kinds of uncertainty were considered: stochastic uncertainty and state-of-knowledge uncertainty. Stochasitc uncertainty is the characterization of the intrinsic variability associated with a system or process within the resolution of our ability to understand the system. Phenomena may not be inherently stochastic, but can be considered stochastic within the resolution of a particular analysis and/or within our ability to understand nature. State-

Table 3-1
Offsite Consequence Measures Calculated in NUREG-1150

Consequence Measure	Description
Early fatalities	Number of fatalities occurring within 1 year of the accident due to early exposure (i.e., exposure incurred within seven days of the accident).
Total latent cancer fatalities	Number of latent cancer fatalities due to both early and chronic exposure (i.e., chronic exposure is that incurred more than seven days after the accident).
Population dose within 50 miles	Population dose, expressed in effective dose equivalents for whole body exposure (person-rem), due to early and chronic exposure pathways within 50 miles of the reactor. Due to the nature of the chronic pathways models, the actual exposure due to food and water consumption may take place beyond 50 miles (e.g., food and water originating within 50 miles of the plant may be consumed by people located beyond 50 miles).
Population dose within entire region	Population dose, expressed in effective dose equivalents for whole body exposure (person-rem), due to early and chronic exposure pathways within the surrounding region.
Individual early fatality risk within one mile	Probability of dying within one year for an individual within one mile of the site exclusion boundary (i.e., ef/pop, where ef is the number of early fatalities within one mile of the exclusion boundary, and pop is the population within one mile of the exclusion boundary).
Individual latent cancer fatality risk within 10 miles	Probability of dying from cancer for an individual within ten miles of the plant (i.e., cf/pop, where cf is the number of cancer fatalities due to direct exposure in the resident population within ten miles of the plant, and pop is the population size within ten miles of the plant). The calculation does not include ingestion but does include integrated groundshine and inhalation exposure.

of-knowledge uncertainty results from a lack of complete information about systems, phenomena, and processes.

Both types of uncertainties exist throughout the PRA. In the accident frequency and accident progression analyses, stochastic uncertainty is expressed in fault trees and event trees. The trees account for alternative outcomes that are expected to vary from one accident to the next in a random manner. State-of-knowledge uncertainty is represented by the values assigned to the inputs (e.g., branch probabilities) to the trees. Similar to NUREG-1150, the source term analysis performed for this study included the state-of-knowledge uncertainty associated with the uncertainty in the values for the parameters in the parametric code. State-of-knowledge uncertainties were not addressed in the consequence analysis. Only the stochastic uncertainty due to weather variability was explicitly represented. The

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calculation of aggregate risk combines the results from all the individual accident scenarios and, as such, eliminates the display of stochastic uncertainty. A distribution of aggregate risk values is a representation of the state-of-knowledge uncertainty that arises from the uncertainty in the input values to the PRA models.

In the NUREG-1150 studies, these two types of uncertainty were deliberately separated such that the effects of each could be ascertained. Since this study uses simplified models and issues are treated at a general level, it is not always possible to make a clear distinction between stochastic processes and state-of-knowledge processes.

3.2.3.2 Propagation of Uncertainties

To evaluate the state-of-knowledge uncertainties, the parameters that represent the events and phenomena thought to be the most important to risk were represented by distributions rather than fixed values or point estimates. Only issues that were thought to be the most important to the uncertainty in risk were included in this study. The issues selected for such treatment were chosen by the project staff. No rigorous analytical process was used to select these issues. Rather, the issues were selected based on prior experience, results from existing PRAs, uncertainty in the issue, anticipated contribution to uncertainty in risk, and interest within the reactor safety community. For many of these issues, distributions used in the NUREG-1150 study were applied to this study. If an applicable distribution from an existing study did not exist, the project staff developed the appropriate distribution. The use of formal expert elicitation techniques (i.e., the use of panels of experts from a variety of organizations and companies) **was not** used in this study.

The propagation of these uncertainties in the PRA was accomplished by using a modified form of Monte Carlo simulation known as Latin hypercube sampling (LHS). In simple terms, the uncertainty was addressed by performing the accident frequency, accident progression, and source term analysis many times (e.g., 100) with different sets of input values selected with LHS [Iman, et al., 1984], a value for aggregate risk was calculated for each set of inputs and the collection of risk values resulted in a distribution of risk. The LHS form of Monte Carlo simulation was selected because:

- It creates a mapping from analysis input to analysis results,
- It allows consideration of essentially any variable that can be supplied to a model as input or generated as an output,
- It will operate in the presence of large uncertainties and discontinuities,
- It is possible to incorporate correlations between variables, and
- it is easy to implement.

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4 Plant Description

Section 4.1 provides a general description of the Grand Gulf primary system, containment, and important systems that can be used to mitigate an accident. Section 4.2 describes the configuration of the plant as modelled in the Level 2/3 analysis.

4.1 General Description

The Grand Gulf Nuclear Station, Unit 1 utilizes a Mark III containment design to house a BWR/6 boiling water reactor (BWR). The Grand Gulf Nuclear Station is operated by Entergy Operations Inc.. Unit 1 was constructed by Bechtel Corporation and began commercial operation in July 1985. The plant is located on the east bank of the Mississippi river in southwestern Mississippi, about 6 miles northwest of Port Gibson, Mississippi. The nearest large city is Jackson, Mississippi approximately 55 miles to the northeast of the plant.

Because of their importance to the progression of an accident following the onset of core damage, the subsections that follow will discuss in greater detail the following features of the plant:

- Primary system,
- the containment structure,
- the drywell structure and suppression pool,
- the reactor pedestal cavity,
- emergency Power System,
- the hydrogen ignition system,
- the containment heat removal systems,
- the coolant injection systems, and
- secondary containment.

Much of the discussion provided in the following subsections has been extracted from the Grand Gulf UFSAR [Grand Gulf UFSAR] and from Volume 6 of NUREG/CR-4550 [Drouin, et al., 1989].

4.1.1 Primary System

The nuclear reactor of Grand Gulf Unit 1 is a 3833 MWt BWR-6 single-cycle forced circulation boiling water reactor (BWR) designed and supplied by General Electric Company. In the Mark III design the reactor pressure vessel (RPV) is founded on the reactor pedestal located in the drywell. The RPV contains the core, the jet pumps, the steam separators, and the steam dryers. The vessel has an internal diameter of 20'-11" and an internal height of 73'. It is fabricated of low alloy steel and is clad internally with stainless steel (except for the top head, nozzles, and nozzle weld zones which are unclad). The reactor vessel has a design pressure and temperature of 1250 psig and 575 °F, respectively. The nominal pressure and temperature in the steam dome is 1040 psia and 549 °F. The reactor is cooled by water that 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 the steam separators and dryers located in the upper portion of the reactor vessel. The steam is then directed to the turbine through four main steam lines. Each steam line is provided with two isolation valves in series (i.e., main steam line isolation valves, MSIVs); one on each side of the containment barrier. Following reactor isolation, the steam in the vessel is directed to the suppression pool via a series of tailpipes. A safety relief valve (SRV) forms the boundary between the main steam line and the tailpipe. To help disperse the steam in the pool, the tailpipe is fitted with a quencher which is located near the bottom of the suppression pool. Following closure of the MSIVs, 20 SRVs and associated piping are available for pressure relief. Eight of these valves are connected to the automatic depressurization system (ADS) which is designed to rapidly depressurize the primary system to a pressure at which the low pressure injection systems can provide coolant to the core. The SRVs are located in the drywell and drywell pressures of approximately 100 psi will prevent opening the valves.

The reactor core is arranged as an upright circular cylinder composed of essentially two components: fuel assemblies and control rods. The core contains 800 fuel assemblies. The fuel assembly consists of a Zircaloy-4 fuel channel and the fuel rods (the number of fuel rods and water rods can vary depending on the fuel design). The fuel channel provides a fixed flow path for the boiling coolant, serves as a guiding surface for the control rods, and protects the fuel during handling operations. A fuel rod consists of slightly enriched UO_2 pellets sealed in a Zircaloy-2 cladding tube. The reactivity of the core is controlled by cruciform control rods dispersed throughout the lattice of fuel assemblies. The control rods, which consist of B_4C in stainless steel tubes surrounded by a stainless steel sheath, enter the core from the bottom and are positioned by individual control rod drives. The core has an equivalent diameter of approximately 16 feet and an active fuel height of 12.5 feet.

The RPV includes a two inch vent line. One end of the vent line is attached to the top of the vessel head, the other end of the line discharges into the sump located in the reactor cavity directly below the vessel. While this line is closed and is not used during normal operation, it is opened during cold shutdown. Even though the vent line is small, the status of this line (i.e., opened or closed) can impact the time available to respond to a core damage accident and if core damage does occur it can also impact the magnitude of the release of radioactive material to the environment.

4.1.2 Containment Structure

The Grand Gulf plant has a Mark III containment. The general arrangement of the containment is displayed in Figure 4-1. The containment is a cylindrical reinforced concrete structure with a steel liner and a hemispherical dome. The containment encloses both the drywell and the suppression pool. During normal operation, the drywell and containment communicate through passive vents in the suppression pool. In addition to the passive vents, there are vacuum breakers in between the containment and the drywell that allow the containment atmosphere to be vented into the drywell if the drywell pressure should drop below the containment pressure. An important feature of the Mark III containment is its large free volume (1.4×10^6 ft^3) which allows it to have a low design pressure (15 psig). The internal design temperature is 185° F. The assessed mean failure pressure of the containment is 55 psig [Harper, ***]. Because of its large volume, the Grand Gulf containment is not inerted. Hydrogen control is accomplished via the hydrogen ignition system (HIS). The HIS is designed to deliberately burn the hydrogen at low concentrations so the accompanying containment pressurization is negligible. The ultimate heat sink is comprised of mechanical draft cooling tower structures.

Personnel can enter the containment through 3 penetrations: the equipment hatch, the upper personnel lock and the lower personnel lock. The equipment hatch is a 19 ft diameter, steel pressure seating hatch. The center line of the equipment hatch penetration is located at an elevation of 172'-3". The hatch is attached from inside the containment via 20 bolts. The hatch uses two compression seals (gasket concept) around its periphery to maintain tightness along the mating surfaces. The hatch is stored inside the containment in a storage bin above the opening. Offsite ac power is required to move and position the hatch. Each personnel airlock consists of a cylindrical steel shell with steel bulkheads at each end and two steel doors in the bulkheads which open toward the reactor. Sealing of each door is accomplished by two, continuous inflatable seals which surround the door edge. When the door is closed the seals inflate outwardly from the door and impinge against a smooth stainless steel sealing surface. The normal operating pressure of the airlock inflatable seals is 60 psig. The airlock doors are 6'-8" high by 3'-6" wide. The center line of the upper lock is 212'-8". The center line elevation of the lower lock is 124'-8" which is approximately 13 feet above the nominal suppression pool level.

In the event that the containment pressure cannot be maintained below the primary containment pressure limit, the containment vent system (CVS) can be used to reduce the containment pressure. The vent path is a 20-inch diameter purge exhaust line which is part of the containment ventilation and filtration system. This line includes four air-operated dampers which are normally closed. The CVS discharges to the roof of the auxiliary building. Containment venting requires instrument air for opening the air-operated dampers. The dampers also require power from Divisions 1 and 2 of emergency ac power for operation of the solenoids. The emergency operating procedures require containment venting when the containment pressure exceeds 20 psig.

4.1.3 Drywell Structure and Suppression Pool

In the Mark III design, the drywell and suppression pool are completely surrounded by the containment structure. The drywell structure is a cylindrical reinforced concrete structure with a flat roof and a steel drywell head. The drywell

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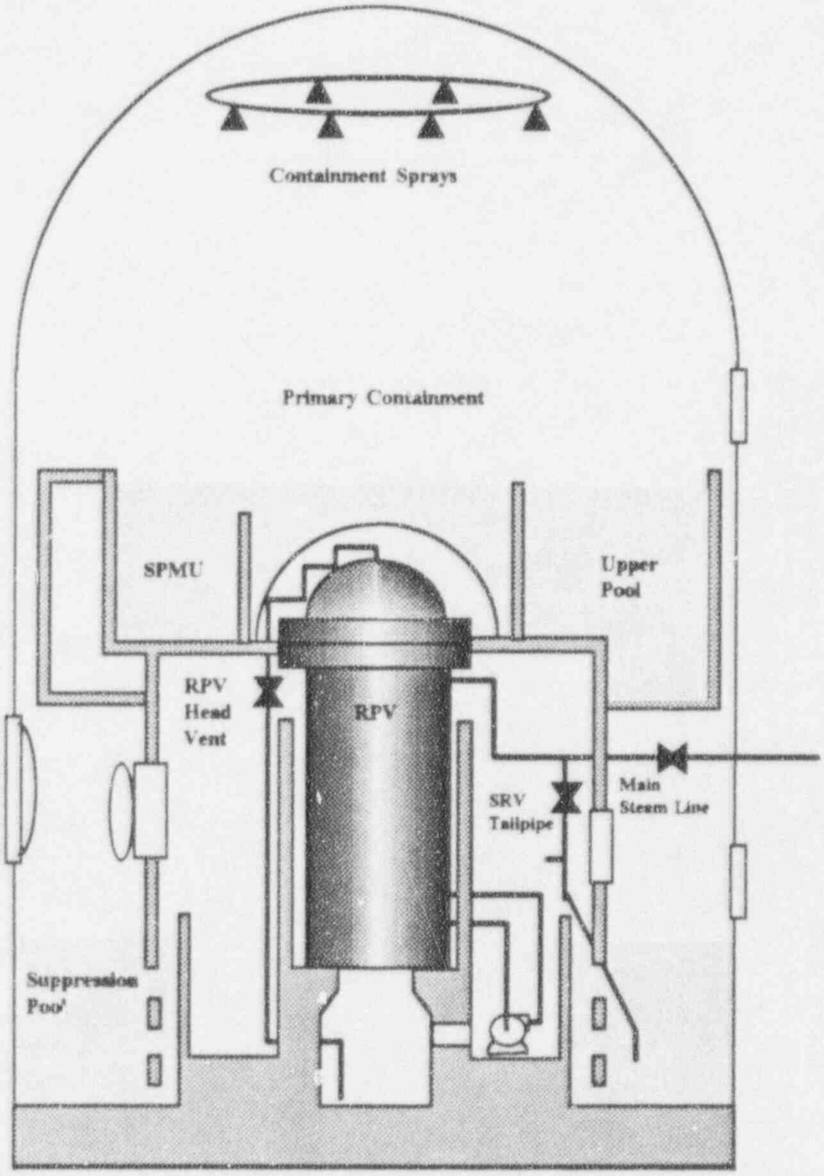

Figure 4-1
Schematic of Grand Gulf Containment

contains the reactor vessel, the SRV valves, the control rod drive (CRD) housings and the recirculation pumps. The drywell has a free volume of $2.7x10^5$ ft^3, a design pressure of 30 psid and an internal design temperature of 330° F. The assessed mean failure pressure of the drywell structure is 85 psid [Harper, ***].

The drywell volume communicates with the containment volume through the vapor suppression pool. The suppression pool serves as a heat sink during accident conditions. Passive horizontal vents in the drywell wall allow steam and noncondensibles released in the drywell to pass into the suppression pool where the steam is condensed and the noncondensibles are released into the containment atmosphere. The suppression pool has two regions. The first region is located in the containment (i.e., wetwell) and is bounded on one side by the containment wall and on the other side by the drywell wall. The second region is in the drywell and is bounded on the one side by the drywell wall and on the other side by the weir wall. The passive horizontal vents in the drywell wall connect the two regions of the pool. There are a total of 135 vents (three rows of vents and each row has 45 vents); each vent has a nominal diameter of 2.33 feet. The suppression pool has a nominal volume of 136,000 ft^3.

In the event that the drywell pressure drops below the containment pressure, there are vacuum breakers in the drywell wall that will open and allow the pressure in the two volumes to equilibrate. These vacuum breakers are powered by emergency ac power.

Personnel can access the drywell through two penetrations: the drywell equipment hatch and the drywell personnel lock. The drywell equipment hatch is approximately 10 feet in diameter and its center line is located at an elevation of 122'-4". The drywell personnel lock is similar in design to the containment personnel locks. The center line of the drywell personnel lock is located at an elevation of 120'.

4.1.4 Reactor Pedestal Cavity

The reactor pedestal cavity is located directly below the RPV. The upper section of the cavity is formed by the 5.75 ft thick pedestal wall and the lower section of the cavity is recessed into the drywell floor. The pedestal cavity is essentially a right cylinder with a diameter of 21.17 ft and a depth of approximately 28 ft. The upper section of the cavity contains CRD housings. The major pedestal penetrations are the CRD piping penetrations at the top of the pedestal and the CRD removal opening which is a 3 ft by 7 ft doorway located 9.5 ft above the cavity floor. It is estimated that the cavity can contain all of the core debris released at the time of vessel failure. Thus, direct attack of the drywell wall by core debris is not an issue at Grand Gulf as it is for the Mark I containments.

When the drywell is flooded to the top of the weir wall, a water depth of 22.8 ft can be established in the cavity. Water can enter the cavity from either the vessel following failure of the bottom head of the RPV or from the drywell. Water can enter the drywell during a LOCA or from overflow from the suppression pool. There are two paths by which water in the drywell can enter the reactor cavity. The first pathway is through the drywell floor drains. There are four 4-inch drains in the drywell floor that connect to the equipment drain sump in the pedestal. The second pathway is through a door in the pedestal located 3.33 feet above the drywell floor. The potential for large amounts of water to be in the cavity has two major implications. First, a large amount of water in the cavity has the potential to cool the core debris that is released from the vessel and thereby prevent the erosion of concrete by the core debris. The water will also retain a portion of the radionuclides that are released from the core debris in the event that it is not quenched. Second, water in the cavity when the core debris breaches the vessel creates the possibility of large fuel-coolant interactions (FCIs).

4.1.5 Emergency Power System

The emergency power system (EPS) consists of the ac and dc power divisions required by the Engineered Safety Features (ESF) to safely shutdown the plant. Both ac and dc are divided into three separate divisions. Divisions 1 and 2 are for the majority of the ESF while Division 3 is dedicated to the high pressure core spray system and its required support systems. The ac divisions normally receive power from one of three offsite sources through the ESF transformers. In addition to the normal supply, each ESF 4.16 kV bus has a standby diesel generator which is available to supply bus loads upon a loss of normal ac power.

4.1.6 Hydrogen Ignition System

The Grand Gulf containment utilizes a hydrogen ignition system (HIS) to control the accumulation of hydrogen during accident conditions. In the core region there is an abundant supply of zirconium (i.e., fuel cladding, channel boxes) which, at the elevated temperatures typical of core damage accidents, readily reacts with steam to produce hydrogen. The function of the HIS is to prevent the buildup of large quantities of hydrogen inside the containment during accident conditions. This is accomplished by igniting, via a spark, small amounts of hydrogen before large amounts accumulate. The HIS consists of 90 General Motors ac division glow plugs (Model 7G), 45 powered by each ac power division. The HIS is manually actuated. Igniters are located throughout the containment and drywell volumes. The Grand Gulf Emergency Procedures indicate that the HIS is not to be used after hydrogen levels exceed 9%.

4.1.7 Containment Heat Removal Systems

Suppression pool cooling (SPC) and the containment spray system (CS) are two modes of the residual heat removal (RHR) system. The RHR system is a two train system with motor-operated valves and pumps. Both trains have two heat exchangers in series downstream from the pump. The function of SPC is to remove decay heat from the suppression pool during accident conditions. The SPC system takes suction from the suppression pool, cools the water by passing the water through heat exchangers (with service water on the shell side), and returns the water to the suppression pool. The SPC system is manually initiated and controlled. The function of the CS system is to suppress the pressure in the containment during accidents. This is accomplished by taking suppression pool water, passing it through a heat exchanger and distributing the water as fine droplets into the containment atmosphere via a series of spray headers in the containment dome. There are no spray headers in the drywell. Both the SPC and the CS modes of RHR require ac power.

4.1.8 Coolant Injection Systems

In a BWR there are many systems that can be used to supply coolant to the core. Systems that can be used when the reactor pressure is high include the high pressure core spray system (HPCS) and the reactor core isolation cooling system (RCIC). The control rod drive system (CRD) can be used as a backup source of high pressure injection. Systems that are used when reactor pressure is low include the low pressure core spray system (LPCS) and the low pressure coolant injection system (LPCI). Additional systems that can be aligned and used as alternate sources of low pressure injection include the service water cross-tie system (SSW cross-tie), the condensate system, and the firewater system.

The function of the HPCS system is to provide coolant to the reactor vessel during accidents in which the pressure in the vessel is high. The HPCS system consists of a single train with motor-operated valves and a motor driven pump which are powered by Division 3 emergency power. The pump is capable of delivering 550 gpm against a reactor pressure of 1177 psig and full flow of 7115 gpm against a reactor pressure of 200 psig. Suction is taken from either the condensate storage tank or the suppression pool.

The RCIC system consists of a single train with motor-operated valves and a turbine-driven pump. The RCIC pump can deliver 825 gpm at any reactor pressure greater than 200 psig. Suction is taken from either the condensate storage tank or the suppression pool. The coolant is supplied to the core via the feedwater line. Steam from the vessel is used to drive the turbine. The technical specification do not require the RCIC system to be available during cold shutdown and, therefore, was not modelled in this analysis.

The CRD hydraulic system can be used as a backup source of high pressure injection. This system includes two pumps which together can achieve a flow rate of approximately 238 gpm with the reactor a 1103 psia. The CRD pumps take suction from the condenser hotwell makeup/reject line. CRD pump A requires Division 1 ac power; CRD pump B requires Division 2 ac power.

The function of the LPCS system is to provide coolant to the reactor vessel during accidents in which the vessel pressure is low. The LPCS system is a single train system consisting of motor-operated and manual valves and a motor-driven pump. The LPCS pump is rated at 7115 gpm with a discharge head of 319 psig. The source of water for the LPCS pump is the suppression pool. The LPCS system is powered by the Division 1 emergency power.

The function of the LPCI system is to provide coolant to the reactor vessel during accidents in which system pressure is low. The LPCI system is but one mode of the RHR system and, as such, shares components with other modes. The LPCI system is a three train system consisting of motor-operated valves and motor-driven pumps. Trains A and B each have two heat exchangers in series downstream of the pump. Train C is injection dedicated and has no heat exchangers. The LPCI pump suction source is the suppression pool. Train A is powered by Division 1 emergency power; Trains B and C are powered by Division 2 emergency power.

The SSW cross-tie system is used as a source of coolant makeup during accidents in which normal sources of emergency injection have failed. The SSW cross-tie system uses SSW pump B (motor-driven pump) to inject water into the reactor via the LPCI system Train B injection lines. SSW pump B is a motor-driven pump and takes suction from the cooling tower basins. Both the SSW Train B and the LPCI system Train B are powered by Division 2 emergency power. The system must be manually aligned and manually actuated.

The condensate system has three condensate pumps and three condensate booster pumps. The pumps are motor-driven and each pump is rated at 9170 gpm. The condensate system takes suction from the condensate storage tank and injects coolant into the vessel through the feedwater line. The condensate pumps are powered by non-safety buses.

The firewater system can be used as a backup source of low pressure injection. The firewater system is a three train system consisting of one motor-driven pump and two diesel-driven pumps. The pumps feed into a common header that supplies water to the fire hoses. The pumps take suction from two 300,000 gallon water storage tanks. The fire hoses are connected, via an adapter, to various test connections in the auxiliary building. These connections feed into various injection systems and water can then be injected through the systems' injection valve. The firewater system can supply approximately 320 GPM at a vessel pressure of 0 psig; the shut off head is approximately 92 psig. The operator is required to align the system and to start the pumps. The diesel-driven pumps do not require ac power from the emergency power system.

4.1.9 Secondary Containment

The Grand Gulf plant utilizes a secondary containment that completely encloses the primary containment. This secondary containment provides a method for controlling the unlikely release of radioactive materials from the primary containment. Two buildings form the secondary containment. The auxiliary building, which contains safety systems, fuel storage and shipping equipment and necessary auxiliary support systems, surrounds the lower portions of the containment. The enclosure building encloses the upper portion of the containment above the auxiliary building roof. The enclosure building provides a boundary for the standby gas treatment system, which maintains a negative pressure in the volume between the containment and enclosure building to ensure that leakage of radioactive materials from the containment is filtered prior to release to the environment in the unlikely event of a loss-of-coolant accident.

4.2 Definition of Plant Operating State (POS) 5

During full power operation the technical specification rigorously defines the configuration of the plant and its associated systems to ensure that essentially no equipment important for mitigating an accident is unavailable for an extended period of time. This includes both core cooling systems and containment systems. During shutdown, the technical specifications allow much more latitude in the availability of systems. Furthermore, the configuration of the plant changes during shutdown to allow for maintenance and refueling (e.g., systems are taken off line for maintenance and the reactor vessel is opened to replace the fuel). Because technical specifications are not as prescriptive during shutdown as they are during full power, and because of the need to perform maintenance on systems and alter the configuration of the plant to refuel, the configuration of the plant and the availability of accident mitigation systems varies drastically from one mode of operation to the next. To accommodate this variability, regimes of operation, or Plant Operating States (POS), were defined where the configuration of the plant and its associated systems could be defined such that a plant model could be developed and potential accidents identified.

4.2.1 Definition of Plant Operating States

A Plant Operating State (POS) is defined as a plant condition for which the status of the plant systems (operating, standby, unavailable) can be specified with sufficient accuracy to model subsequent accident events. A POS is not identical to a Mode (or Operating Condition) as defined in the technical specifications [Grand Gulf, Tech Specs]; however, POSs are defined based on Operating Conditions. Using the OCs as a starting point, seven POSs were defined. The relationship between the OCs and the POSs is provided in Table 4-1. A description of the process used to identify and characterize a POS is provided in Appendix A of Volume 2 of this report.

Table 4-1
Relationship Between Operating Conditions and Plant Operating States

Operating Condition	Plant Operating State
1. Power Operation 2. Startup	1. Vessel pressure from rated conditions to 500 psig and thermal power not greater than 15%. Core coolant can be at any temperature.
3. Hot Shutdown (core coolant temperature greater than 200 F)	2. Vessel pressure from rated conditions to 500 psig
	3. Vessel pressure from 500 psig to above 100 psig
	4. Vessel pressure less than 100 psig and RHR/SDC on
4. Cold Shutdown	5. Until vessel head is detensioned (include part of OC 5)
5. Refueling (vessel head detensioned or removed, temperature of core coolant is no greater than 140 F)	6. Head off and coolant level raised to the steam lines
	7. Head off, upper pool filled, and the refueling transfer tube open.

4.2.2 Characterization of POS 5

POS 5 is rigorously defined as Cold Shutdown (OC 4) and Refueling (OC 5) only to the point where the vessel head is off. POS 5 can be entered either coming down from power or going back up to power.

For the purposes of delineating accident scenarios and estimating consequences, it was necessary to divide POS 5 into time segments or "time windows". During a refueling outage the plant can be in POS 5 for an extended period of time; the event that initiates the accident can occur anytime during this time period. Since the decay heat load from the core decreases with time, the amount of time that is available to the operators to respond to an accident will depend on when the event that initiates the accident occurs during POS 5. Furthermore, the radionuclide inventory also changes with time and, therefore, the radiological potential of the accident will also change with time. Because of this dependency on time, the time the plant is in POS 5 is divided into segments or time windows; a unique decay heat level and radiological inventory is then assigned to each window. To keep the calculations manageable, only three time windows were defined. In POS 5 there are two natural time segments. The first segment corresponds to the time the plant is in POS 5 as it is coming down from power prior to refueling. The second segment correspond to the time the plant again enters POS 5 after refueling. In between these two POS 5 segments, the plant is in POS 6 and POS 7. Since on average, about 36 days elapses between the first and second POS 5 time segments, the decay heat and the radionuclide inventory for the first time segment will be significantly different from the second segment. The first segment was further subdivided to account for the availability of an alternate source of decay heat removal. The Alternate Decay Heat Removal System (ADHRS) can be used to remove decay heat from the core once the reactor has been shutdown for at least 24 hours. Thus, the first segment was divided to distinguish the time in POS 5 prior to 24 hours after shutdown from the time in POS 5 after 24 hours after shutdown.

The time after shutdown that the plant enters POS 5 and the time spent in POS 5 are based on Grand Gulf refueling outage data. While information was available for the first four refueling outages, only data from the second, third, and fourth refueling outages were used in this study. Because of the number of special test that were conducted during the first RFO, it was considered atypical and, therefore, data from this outage was excluded from the analysis. On average, the plant enters POS 5 14 hours after shutdown and remains in POS 5 for 80 hours before entering POS 6. On the way back up to power, the plant again enters POS 5 40 days after shutdown and remains in POS 5 for 10.4 days. Based on this information, the three time windows were defined as:

Time Window 1:	Starts 14 hours after shutdown and has a duration of 10 hours,
Time Window 2:	Starts 24 hours after shutdown and has a duration of 70 hours, and
Time Window 3:	Starts 40 days after shutdown and has a duration of 10.4 days.

Although the plant can enter POS 5 during a refueling outage as early as 7 hours after shutdown, 7 hours was not used as the start time for Window 1 because the average value of 14 hours was judged to be more representative of the time it takes the plant to enter POS 5. However, to account for the fact that the plant could enter POS 5 as soon as 7 hours after shutdown, the decay heat load used to represent Window 1 was the decay heat load 7 hours after shutdown. The decay heat used to represent Window 2 is the decay heat load 24 hours after shutdown. Similarly, the decay heat used to represent Window 3 is the decay heat load 40 days after shutdown. The three time windows are depicted graphically in Figure 4-2.

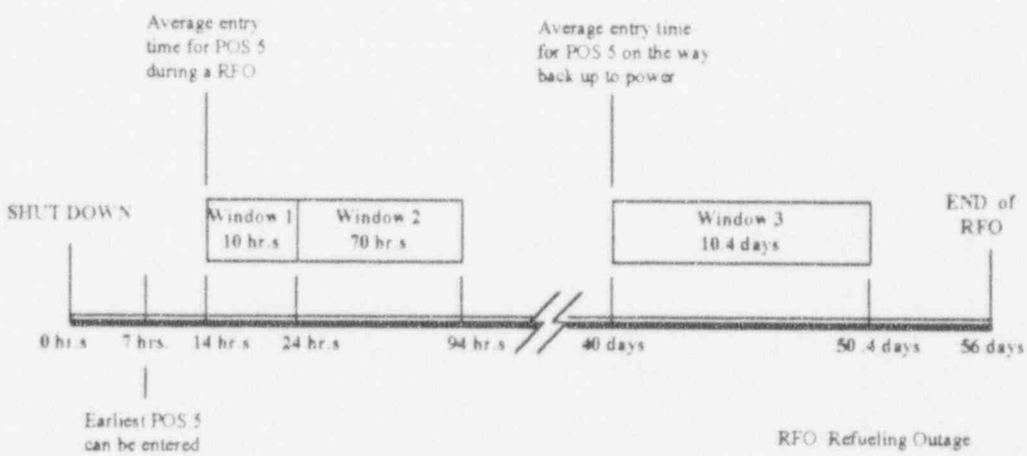

Figure 4-2
POS 5 Time Windows

The configuration of the plant during POS 5, as modelled in the Level 2/3 analysis, was determined from requirements imposed by the technical specifications, from plant procedures and practices during a refueling outage, critiques of refueling outages, and interviews with plant personnel. The technical specifications were used to define the minimum set of requirements. If a system was not required by the technical specifications to be operable, then the plant procedures and practices were reviewed. For example, the technical specifications do not require the HIS to be operable during POS 5; however, the practice at the plant is to keep at least one train operable. Thus, in this analysis, even though the

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technical specifications do not require the HIS to be available, it was assumed that a least one train was always available. Even though the configuration of the plant continues to change during POS 5, to model the plant in the Level 2/3 analysis, a plant configuration at the start of the accident was specified. For example, the containment equipment hatch is removed during this POS. Thus, when the POS is initially entered, the hatch is attached, which is one plant configuration, and then it is subsequently removed during the POS creating a second configuration. However, it was judged that the time spent in the first configuration was sufficiently small that only the second configuration needed to be analyzed. The configuration of the plant at the start of the accident, as modelled in the Level 2/3 analysis, is defined below.

Containment: The technical specifications do not require the integrity of the primary or the secondary containments to be maintained during POS 5. A review of the Grand Gulf refueling critiques indicated that the containment equipment hatch is typically removed shortly after entering POS 5. As modelled, the equipment hatch and both personnel locks are open when the accident is initiated. Given that the necessary support systems are available, the model allowed the containment to be closed prior to core damage and if closed the containment could be vented it necessary.

Drywell Integrity: The technical specifications do not require that the drywell integrity be maintained during POS 5. A review of the Grand Gulf refueling critiques indicated that the drywell personnel lock is open and equipment hatch is typically removed early in POS 5. Furthermore, during POS 5 a portion of the upper reactor pool is drained and the drywell head is removed. As modelled, either the drywell equipment hatch or the drywell personnel locks were open and remained open throughout the accident.

Reactor Pressure Vessel: In cold shutdown the reactor pressure vessel head is on. While the technical specifications do not require any SRVs to be available, Grand Gulf administrative procedures require at least two SRVs to be available. Therefore, in this analysis the modelled allowed two SRVs to be available. The temperature of the vessel water is required by the technical specifications to be less than 200° F. The water level can either be at the normal level or the natural circulation level. For the purposes of this analysis, it was assumed that at the start of the accident the reactor water was at the normal level and its temperature was 200° F. The RPV head vent was assumed to be open at the start of the accident. The status of the MSIVs (i.e., open or closed) is accident specific.

Suppression Pool: The suppression pool inventory depends on the accident. Three levels were considered: (1) Low water level (18 ft -4 1/2 in), (2) Drained level 12 ft 8 in, and (3) empty with 170,000 gal available to HPCS from the condensate storage tank.

Hydrogen Ignition System: The technical specifications do not require the HIS to be available during POS 5. However, since it is the practice at the plant to perform train based maintenance during a refueling outage, and half of the igniters are on Train A and the other half are on Train B, it was assumed in this analysis that at least one train of HIS will always be available (Note, however, the HIS will not operate without ac power).

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4.3 References

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[Harper, ***] F. T. Harper, "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters: Determination of Parameter Values Not Quantified Using Formal Expert Elicitation," NUREG/CR-4551, Vol. 2, Rev. 1, Part 6, Sandia National Laboratories, to be published.

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5 Plant Damage State Analysis

Plant damage states form the interface between the accident frequency analysis (i.e., Level 1 analysis) and the accident progression analysis (i.e., Level 2 analysis) and as such define the initial and boundary conditions for the Level 2 analysis. In the Level 1 analysis the sequence of events that will lead to core damage are identified. The minimum set of events that will result in core damage is called a cut set. In the plant damage state analysis, cut sets with similar characteristics that are important to the progression of the accident following core damage are grouped together, each group constituents a PDS. The Level 1 analysis is documented in Volume 2 of this report.

5.1 Development of Plant Damage States

A four step approach was used to develop the PDSs. Each step is discussed below.

- In the first step, general features of the accidents that will define the initial and boundary condition for the Level 2 analysis are identified. These general features define the configuration of the plant at the start of core damage and the status of systems than can be used to mitigate the accident.

- In the second step, specific systems and plant features are identified that address each of these general features. Each specific feature is call a *characteristic*; the possible configurations of each system or characteristic is call an *attribute*. More than one characteristic may be used to define a general feature. For example, the following four systems (i.e., characteristics) could be used to define the general feature that addresses the status of core cooling : HPCS, LPCI, SSW crosstie, and CDS. That is, HPCS is one of four characteristics that defines the status of core cooling. The possible configurations of the HPCS system, or attributes, during the accident are: (A) HPCS available but not being used, (B) HPCS not available and not recoverable, and (C) HPCS not available but recoverable with the recovery of offsite power. In this example, the HPCS characteristic has three attributes. The list of characteristics and their associated attributes define the possible plant/system configuration for a particular accident. This is displayed as a string of alphanumeric characters. The first position corresponds to the first characteristic, the second position corresponds to the second characteristic and so on. The alphanumeric character assigned to each position is the attribute for the appropriate characteristic.

- In the third step, the cut sets are reviewed and the appropriate attributes for each characteristic are assigned to each cut set. Since the list of characteristics generally describes the accident in less detail than the cut set, groups of cut sets will have the same string of letters. A unique string of letters is call an End State (ES). (ESs are similar to PDSs except that they define the accident in more detail than the PDS.) While the number of ESs can be significantly less than the number of cut sets, typically there are still too many ESs to analyze individually in the Level 2 analysis.

- In the fourth and final step, the many ESs are combined into a manageable number of PDSs. This step is possible because within the resolution of the Level 2 analysis many of the ESs will result in similar accident progressions and releases of radioactive material. To form the PDSs, ES characteristics are combined such that only the information that is needed to define the initial and boundary conditions for the Level 2 analysis are defined by the PDS. For example, the individual ESs indicate the availability of many different coolant injection systems (e.g., LPCI, SSW cross-tie, and CDS). However, if LPCI is recoverable and the vessel is at low pressure then the status of the other systems is not important for the model used in the Level 2 analysis. Thus, assuming the other characteristics of the ESs are the same, all those ESs with LPCI recoverable would be combined regardless of the status of SSW cross-tie and CDS. Through this process the majority of the ESs can be combined into a dozen or so PDSs, however, the actual number of PDSs developed will depend on the diversity of the accident sequences and the resolution desired for the Level 2 analysis.

The general features of the accident that were used in this study to develop the PDSs are: the status of electric power, the status of core cooling, the status of containment heat removal, the status of reactor pressure vessel integrity, the status of containment integrity, and accident timing characteristics. Each of these general accident features is discussed below.

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Status of Electric Power: There are systems and components that can influence the progression of the accident following core damage that were not modelled in the Level 1 analyses. Many of these systems depend primarily on electric power and, therefore, in many cases this feature of the accident can be used to determine the availability of these systems following the onset of core damage. For example, offsite ac power is required to close the containment. Similarly, emergency ac power is required to operate the hydrogen ignition system.

Status of Core Cooling: This feature is used to identify systems that can be used to restore core coolant during the core damage process. Restoration of core cooling offers the potential to arrest the core damage process and prevent vessel failure. Preventing vessel failure can substantially reduce the consequences of the accident.

Status of Containment Heat Removal: This feature addresses the status of systems that can be used to remove decay heat from the containment such as containment sprays and the suppression pool cooling systems. In cases where the containment is closed, the energy released to the containment atmosphere during core damage will pressurize the containment. These systems are used to attenuate this pressurization and thereby reduce the load on the containment structure. Containment heat removal is generally necessary to prevent containment failure. Containment sprays are also useful in that they remove aerosols from the containment atmosphere and thereby reduce any potential release of radioactive material. Since the suppression pool is an integral part of containment heat removal, this feature also addresses the status of the suppression pool at the time of core damage (i.e., amount of water in the pool and the temperature of the pool) and is used to identify situations where its performance may be impaired. The suppression pool is used as a heat sink for the reactor, supplies water to ECCS, and is an effective device for removing radioactive material released from the vessel.

Status of Reactor Pressure Vessel: This feature defines the integrity of the reactor pressure vessel and the pressure in the vessel at the time of core damage. The integrity of the vessel is important because it will determine the path by which steam and radioactive material will escape from the vessel. If the vessel integrity is maintained the releases will pass from the vessel to the suppression pool via the SRV tailpipes. As mentioned previously, the suppression pool is an effective device for mitigating the release. For a LOCA, the vessel releases will enter the drywell. For interfacing systems LOCA, the release will bypass the containment altogether and enter auxiliary building. If the vessel head vent is open a portion of the release will enter the drywell while the remaining portion will enter the suppression pool via the SRV tailpipes. When the vessel integrity is maintained, the pressure in the vessel will affect the timing of the accident, the amount of radioactive material released during core damage, and the pressure in the containment following vessel failure. The vessel pressure will also determine which systems can be used to provide makeup (i.e., high pressure systems or low pressure systems).

Status of Containment Integrity: This feature defines the integrity of the containment boundary at the time of core damage. The integrity of the containment boundary is one of the most important factors that will determine the severity of the accident. For severe core damage accidents in which the containment boundary remains intact, the offsite consequences are generally small. On the other hand, when the containment boundary is not maintained the consequences can be quite severe. Since in POS 5 the containment equipment hatch and personnel locks can be open, it is important to know the status of these penetrations at the time of core damage. This feature also addresses the status of the containment vent system which can be used to relieve pressure in the containment when containment heat removal systems are not available or are inadequate. Opening the containment vent, however, will allow radioactive material in the containment atmosphere to enter the environment.

Accident Timing Characteristics: This feature defines the time window that the plant is in when the initiating event occurs and the amount of time that elapses between the occurrence of the initiating event and the onset of core damage. The time window will directly affect the amount of decay heat and the radionuclide inventory that is present at the start of the accident. The time window combined with the amount of time that elapses between the start of the accident and the onset of core damage will determine the amount of decay heat that is available at the onset of core damage which will in turn affect the timing of key events following the onset of core damage (e.g., vessel failure and containment failure). The speed with which the accident proceeds can affect the amount of time that is available to restore core cooling and will also affect the relative timing between when the release of radioactive material occurs and when the public begins to evacuate. This last item can have a major impact of the magnitude of early health effects.

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The characteristics that are used to define the ESs are provided in Appendix A.1. Similarly, the characteristics used to define the PDSs and the rationale used to collapse the ES characteristics into PDS characteristics is provided in Appendix A.2.

5.2 Description of Plant Damage States

The Level 1 analysis generated 28 core damage sequences which contained a total of 38 cut sets. These cut sets were grouped into 22 ESs which were then collapsed into twelve PDSs. The sequences contained in each ES and the ESs contained in each PDS are presented Appendix A.1. A brief description of each of the twelve PDSs is provided below. The first number in the PDS name identifies the Time Window in which the accident occurs, the second number identifies the PDS number. For example, PDS1-3 is the third PDS in Time Window 1.

Plant Damage State PDS1-1:

The accidents in PDS1-1 are initiated by a LOCA (A or S1) while the plant is in Time Window 1. The break drains the vessel to the top of the jet pumps (i.e., 2/3 core height). The operators attempt to establish water solid operation (i.e., form a water solid loop between the RPV and the suppression pool) with low pressure ECCS. HPCS is unavailable due to maintenance or random hardware failures. In a LOCA the water will drain from the vessel via the break into the drywell. To form a water solid loop enough water must be pumped into the drywell (i.e., via the break) to flood the drywell up to the weir wall to form a connection between the drywell and suppression pool. To establish a connection between the drywell and the wetwell the operators must dump the SPMU into the suppression pool. In this scenario, however, the operators fail to dump the SPMU in sufficient time to prevent ECCS from failing on inadequate suction head. Once the ECCS pumps fail, they are assumed to be lost for the entire accident. Unable to perform a water solid operation, the operators attempt to flood the containment with the standby service water crosstie. To successfully flood the containment, the lower personnel lock must be closed. In this plant damage state, the operators did not close the lower personnel lock. Thus, the containment is flooded up to the lower personnel lock at which point the water then enters the auxiliary building. It was assumed that the flooding operation would flood the auxiliary building resulting in the loss of all core and containment cooling one hour after the start of the accident. Core damage is estimated to occur approximately one hour after the loss of core cooling (i.e., core damage is estimated to occur two hours after the initiating event).

At the onset of core damage, the vessel integrity is breached (i.e., the break), the vessel is at low pressure, the containment is open. While offsite power is available during the accident, both the core cooling and the containment cooling functions are lost for the entire accident.

Plant Damage State PDS1-2:

The accidents in PDS1-2 are initiated by a loss of offsite power (T1) followed by a failure of the Train B emergency diesel generator to either start or to run for sufficient time to prevent core damage. The initiating event occurs while the plant is in Time Window 1. The Train A emergency diesel generator is unavailable due to maintenance. HPCS is also unavailable due to either maintenance or random hardware failures. These events result in a station blackout (i.e., loss of all onsite and offsite ac power) resulting in a loss of all emergency core cooling. Furthermore, the station batteries deplete within 2 hours from the start of the accident. Without dc power, the SRVs cannot be opened to keep the vessel at low pressure. To complicate matters, the valves that isolate the low pressure piping and components on the SDC system from the high pressure piping associated with the RPV require ac power to change position. Thus, without ac power these valves remain open. Without core cooling the vessel inventory begins to boil and the resulting steam pressurizes the RPV. The pressurization of the RPV fails the low pressure components associated with the SDC system resulting in a break outside the containment. The break drains the vessel to the top of the jet pumps (i.e., 2/3 core height). Without a means to supply coolant to the core, the accident proceeds to core damage. It is estimated that core damage occurs 3.5 hours after the initiation of the accident.

At the time of core damage the vessel integrity has been breached and the primary system is at low pressure. It is assumed that the loss of dc power precludes the recovery of offsite power. Without electric power, the containment cannot be closed and the core and containment cooling systems cannot be restored.

Plant Damage State PDS1-3:

This PDSs is similar to PDS1-2 except that the station batteries continue to provide dc power for 12 hours. With dc power available, the operators are able to open two SRVs to keep the vessel depressurized. The operators then align the firewater system to provide coolant to the core. Once the station batteries fail 12 hours after the initiating event, the SRVs close, the vessel pressurizes and the firewater system is lost due to high vessel pressure. In this PDS, the operators manually isolate the low pressure components of the SDC system from the high pressure primary system. (Note: In this PDS, as apposed to PDS1-2, there is considerable amount of time between the loss of offsite power and the pressurization of the vessel which provides the operators sufficient time to manually isolate the SDC system.) Without core cooling the accident proceeds to core damage in approximately 12 hours.

At the time of core damage, the vessel is at system pressure with pressure relief being provided by the SRVs cycling at their setpoints. The reactor vessel head vent is also open. It is assumed that the loss of dc power precludes the recovery of offsite power. Without electric power, the containment cannot be closed and the core and containment cooling systems cannot be restored.

Plant Damage State PDS1-4:

This PDS is similar to PDS1-2 except that the station batteries continue to provide dc power for at least 3.5 hours. In this PDS the operators fail to open two SRVs and align the firewater system for core injection. Furthermore, there is insufficient time for the operators to manually isolate the SDC system from the primary system. Without core cooling the vessel inventory begins to boil and the resulting steam pressurizes the RPV. The pressurization of the RPV fails the low pressure components associated with the SDC system resulting in a break outside the containment. The break drains the vessel to the top of the jet pumps (i.e., 2/3 core height). Without a means to supply coolant to the core, the accident proceeds to core damage. It is estimated that core damage occurs 3.5 hours after the initiation of the accident.

At the time of core damage the vessel integrity has been breached, the primary system is at low pressure, and the containment equipment hatch is open. Since dc power is available it is possible to restore offsite power after the onset of core damage. Following the recovery of ac power, low pressure ECCS can be used to provide coolant to the core.

Plant Damage State PDS1-5:

The accidents in PDS1-5 are initiated by a valve misalignment that diverts vessel water to the suppression pool via the RHR system (H1) while the plant is Time Window 1. The diversion of water is automatically isolated when the vessel water level reaches Level 3. The operators recognize the diversion and attempt to restore core cooling using the water solid operation (i.e., form a water solid loop between the vessel and the suppression pool), however, the suppression pool is empty. Next, the operators attempt to flood the containment by injecting water into the vessel using SSW crosstie. Once the vessel is full, the water passes through the SRVs and enters the suppression pool. In this PDS, however, the operator fails to close the lower containment personnel lock. With the lower personnel lock open, the water being used to flood the containment will pass through the lock and flood the auxiliary building. Once the auxiliary building is flooded, it is assumed that all core and containment cooling systems are lost. At this point in the accident the water level in the vessel is at the main steamlines. Without core cooling the temperature of the core coolant will increase until it reaches the saturation temperature at which point it will begin to boil. The steam generated during the boiling process passes through the SRVs and is condensed in the suppression pool. Core damage is estimated to occur 7 hours after the initiating event.

At the onset of core damage, two SRVs are open, the reactor vessel head vent is closed, and the primary system is at low pressure. Even though offsite power is available, the containment is open and all core and containment cooling systems are lost for the entire accident.

Plant Damage State PDS2-1:

The accidents in PDS2-1 are initiated by a LOCA (A or S1). This PDS is the same as PDS1-1 except that the accident is initiated while the plant is in Time Window 2.

Plant Damage State PDS2-2:

The accidents in PDS2-2 are initiated by a loss of offsite power (T1). This PDS is the same as PDS1-2 except that the initiating event occurs while the plant is in Time Window 2.

Plant Damage State PDS2-3:

The accidents in PDS2-3 are initiated by a loss of offsite power (T1). This PDS is the same as PDS1-4 except that the initiating event occurs while the plant is in Time Window 2.

Plant Damage State PDS2-4:

The accidents in PDS2-4 are initiated by a diversion of vessel water to the suppression pool via the RHR system due to a misalignment of valves. This PDS is the same as PDS1-5 except that the initiating event occurs while the plant is in Time Window 2.

Plant Damage State PDS2-5:

The accidents in this PDS are initiated by a loss of all SSW (T5A) which leads to loss of both Train A and Train B ECCS. HPCS is unavailable due to maintenance of random hardware failures. The operators recognize the loss of SDC but are unable to establish water solid operation because all of the ECCS injection systems are unavailable. Without core cooling, the core coolant inventory temperature increases until it reaches saturation at which point the vessel coolant begins to boil and the vessel begins to pressurize. In this PDS, the operators do not open the SRVs to keep the vessel at low pressure and, therefore, the vessel will pressurize to system pressure and will be maintained at system pressure with the SRVs providing pressure relief at their pressure setpoints. The steam generated in the vessel is directed to the suppression pool, via the SRV tailpipes, where it is condensed. Core damage is estimated to occur 12 hours after the initiating event.

At the onset of core damage, the primary system is at system pressure and the vessel head vent is open. All core and containment cooling systems are lost for the entire accident. The containment can be either open or closed. In the case that the containment is closed prior to core damage, the containment vent system is available to relieve the pressure in the containment.

Plant Damage State PDS2-6:

The accidents in PDS2-6 are initiated by a valve misalignment that diverts vessel water to the suppression pool via the RHR system (H1) while the plant is Time Window 2. The diversion of water is automatically isolated when the vessel water level reaches Level 3. The operators recognize the diversion and attempt to restore core cooling using the water solid operation (i.e., form a water solid loop between the vessel and the suppression pool). The operators turn on the ECCS pumps and pump water from the suppression pool into the vessel. In this PDS, the MSIVs are open at the start of the accident and the operators fail to close them during the accident. Furthermore, the operators do not turn ECCS off once the vessel is full. Instead, the water fills the vessel and flows out through the steam lines to the turbine. This is allowed to continue until the ECCS suction strainers in the suppression pool are uncovered. At this point ECCS will fail. Furthermore, it is assumed that the resulting flood in the turbine building will fail any remaining core and containment cooling systems. With the water level now at the main steamlines, the coolant temperature will increase until it reaches saturation at which boil the water will begin to boil. The steam generated during the boiloff process will be transported to the condenser via the main steamlines. Without core cooling the accident will proceed to core damage in 6.75 hours.

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At the time of core damage the MSIVs are open which establishes a direct path from the RPV to the turbine building which is outside the containment. Because the MSIVs are open, the primary system is at low pressure. Offsite power is available and the containment can be either open or closed. All core and containment cooling systems are lost for the entire accident.

Plant Damage State PDS3-1

The accidents in PDS2-1 are initiated by a LOCA (A or S1). This PDS is similar to PDS1-1 except that the accident is initiated while the plant is in Time Window 3 and HPCS was initially available. In this PDS, the failure by the operators to dump SPMU results in a loss of all ECCS (both high pressure and low pressure).

5.3 Plant Damage State Results

The core damage frequencies and fractional contributions to the core damage frequency for the 12 PDSs are presented in Tables 5-1. Two fractional contribution measures were calculated: fractional contribution to the mean core damage frequency and the mean fractional contribution to the core damage frequency. In the first calculation, the mean of each PDS is divided by the total mean core damage frequency. This is the measure that was used to display the contribution of groups of initiating events in the Level 1 analysis reported in Volume 2 of this report. The second measure is calculated by dividing the PDS frequency for a single observation by the total core damage frequency for the same observation. This is repeated for all of the observations, the fractions are added together, and then the sum is divided by the total number of observations. This second measure is more representative of the fractional contribution of the PDS across the entire distribution and is the measure that is used in the remaining sections of the report.

The Level 1 analysis used an LHS sample size of 1000 whereas the Level 2/3 analysis used a sample size of 200. While a sample size of 1000 can be used in the Level 1 analysis, the large computational requirements of accident progression and consequence analyses precluded the use of such a large sample size in the Level 2/3 analysis. When selecting the LHS sample size for the Level 2/3 analyses two objectives had to be considered: (1) the sample size had to be sufficiently large such that the Level 2/3 PDS results were reasonably similar to the Level 1 results, and (2) the sample size had to be small enough that the calculations could be performed in a timely manner. A sample size of 200 satisfied these two objectives. The PDS frequencies from these two samples are compared in Appendix A.3.

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Table 5-1
Plant Damage State Results

Plant Damage States	Descriptive Statistics[1]: Core Damage Frequency (1/yr)				Fractional Contribution[2]	
	5%	50%	95%	Mean	FCM-CDF	MFC-CDF
PDS1-1	1.6E-09	1.4E-08	1.9E-07	4.1E-08	0.020	0.018
PDS1-2	1.4E-10	4.3E-09	1.3E-07	2.3E-08	0.011	0.015
PDS1-3	2.9E-09	1.7E-08	1.6E-07	4.4E-08	0.021	0.030
PDS1-4	6.0E-11	2.0E-09	3.5E-08	9.2E-09	0.004	0.006
PDS1-5	4.9E-10	6.9E-09	4.8E-08	1.4E-08	0.007	0.010
PDS2-1	1.3E-08	1.4E-07	1.5E-06	3.5E-07	0.168	0.153
PDS2-2	2.2E-08	1.5E-07	1.6E-06	5.5E-07	0.264	0.217
PDS2-3	2.7E-09	2.9E-08	4.5E-07	1.1E-07	0.053	0.059
PDS2-4	7.7E-09	8.8E-08	7.5E-07	2.0E-07	0.097	0.140
PDS2-5	8.6E-11	2.7E-09	5.3E-08	1.3E-08	0.006	0.010
PDS2-6	2.6E-11	1.1E-09	2.8E-08	7.4E-09	0.004	0.006
PDS3-1	6.3E-08	3.8E-07	2.4E-06	7.3E-07	0.347	0.338
Total	4.1E-07	1.4E-06	5.6E-06	2.1E-06		

Note 1: Statistics based on a sample size of 200

Note 2: FCM-CDF = Fractional contribution to mean core damage frequency

MFC-CDF = Mean fractional contribution to core damage frequency

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6 Accident Progression Analysis

In the accident progression analysis, event tree techniques are used to delineate the many possible paths (i.e., accident progressions) that the accident can follow after the onset of core damage. The event tree that is used to model this portion of the accident is called an Accident Progression Event Tree (APET). Many different paths are possible because there is uncertainty as to how equipment will operate, the actions the operators will perform, and the outcome of complex phenomena. Of primary concern is the identification of events that can affect the release of radioactive material from the core and the transport of this material through the engineered environment (e.g., primary system, containment, and auxiliary building) since this information will be used to estimate source terms in the subsequent analysis (see Section 7). The inputs to the APET are the PDSs described in Section 5. The products of the accident progression analysis are the delineations of the possible paths the accident may following after the onset of core damage and the probability of each path. Since a typical APET used to model severe accidents can delineate thousands or even hundreds of thousands of paths, it is not practical to estimate a source term for each path individually. Instead, groups of progressions, called Accident Progression Bins (APBs), are form that have similar characteristics that affect the formation of the source term. In the subsequent analysis, the amount of radioactive material released to the environment is estimated for each APB. In the following subsections the APET model will be described, the sources of information used to quantify the APET identified, the characteristics of the accident used to develop the APBs discussed, and summary of results provided.

6.1 Accident Progression Model

The APET developed for this analysis is similar in concept and structure to the APETs used in the NUREG-1150 study, however, it is not be as detailed. As compared to the NUREG-1150 APETs, the POS 5 APET includes fewer questions (i.e., top events), issues were addressed in less detail (e.g., hydrogen combustion phenomena), and formal expert judgement procedures were not used to quantify the APET. While there are substantially fewer questions included in the POS 5 APET, as compared to the NUREG-1150 APETs, the POS 5 APET included a sufficient number of questions so that important interactions between phenomenon/systems/operator actions were captured. Experience and insights gained from other PRAs and the abridged analysis of POS 6 were used to focus the development of the trees and thereby limit the number of questions. In particular, the following factors allowed the size of the tree to be reduced (relative to the APET used in NUREG-1150): (1) in many cases, several related issues were combined and addressed as a single issue, (2) the plant and system configuration during shutdown minimized and/or eliminated the need to address many of the issues that were considered in the full power PRA (e.g., once the drywell equipment hatch has been removed it is no longer necessary to assess the structural response of the drywell to loads that occur during the accident) and (3) where necessary, simplifying assumptions were used to limit the size of the analysis (e.g., the impact that the standby gas treatment system has on the release of radioactive material is not included in this analysis). The selection of appropriate top events was based on PRAs of full power operation, characteristics of the PDSs, results from relevant deterministic calculations that were generated using state-of-the-art severe accident codes such as MELCOR [Summers, et al., 1991], and on knowledge of how the plant operates based on plant procedures, discussions with plant personnel, and relevant technical descriptions (e.g., technical specifications, safety analysis report, and system descriptions).

6.1.1 Major Assumptions

The major assumptions that were made during the development and quantification of the APET are presented below. Addition assumptions are discussed in Appendix B.1 which provides a detailed discussion of the questions in the APET. The major assumptions include:

- During POS 5 the plant is in the cold shutdown mode of operation with the vessel head attached. Two SRVs are available to control the pressure in the vessel.

- Core damage is defined as the start of fuel heatup. The time to core damage and other timing characteristics of the accident were determined from MELCOR calculations performed for this study. MELCOR calculations were performed for various PDSs and are documented in Volume 6, Part 2 of this report.

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- The containment and drywell are both open at the start of the accident. The containment can only be closed if offsite ac power is available; containment closure must be completed prior to the onset of core damage. The drywell is assumed to remain open throughout the accident. Also, for those accidents in which the containment was unsuccessfully flooded (i.e., the lower personnel lock was inadvertently left open resulting in a flood in the auxiliary building) it was assumed in the Level 1 analysis that the lock remained open prior to core damage. Since no credit is given for closing the containment after the onset of core damage, it is assumed that the containment remains open for the duration of the accident.

- DC power from the station batteries is required to restore offsite power to the plant in the event that offsite power is lost prior to or during the accident. Therefore, if the station batteries deplete prior to the restoration of offsite power, it is assumed that offsite power cannot be restored during the accident.

- If the containment equipment hatch and/or personnel locks are open, the airborne radioactive material will enter the auxiliary building prior to being released to the environment. The auxiliary building can either fail from pressurization by steam and noncondensibles or from pressurization that accompanies a hydrogen burn. The auxiliary building is assumed to fail on a 5 psi overpressure. Prior to failure of the auxiliary building it is assumed that no radioactive material enters the environment. While the auxiliary building is not a leak tight structure, it is assumed that the radioactive material is released into the building slowly enough that a negligible amount of radiation escapes into the environment prior to failure of the building. Neither the standby gas treatment system nor the ventilation system are modelled. Thus, it is assumed that no engineered features are available to attenuate the release in the auxiliary building and the only attenuation that the release will experience in the building is that due to natural processes (e.g., natural deposition).

- If the containment is closed prior to core damage, containment heat removal must be available to prevent subsequent containment failure from long term overpressurization. This assumption is supported by MELCOR calculations that show that the containment will ultimately fail from overpressure if the decay heat is not removed from the containment. For the PDSs developed for this analysis, containment heat removal is never available and, therefore, the containment will never remain intact throughout the accident. If the containment is closed prior to core damage and does not fail from loads accompany hydrogen combustion or vessel failure and it not vented, it is assumed to fail late in the accident from the accumulation of steam and noncondensibles.

- If the containment is closed prior to the onset of core damage and then subsequently vented after the onset of core damage, it is assumed that the vent stays open throughout the accident. While the emergency operating procedures (EOPs) direct the operators to close the containment once its pressure drops below a certain pressure, without containment heat removal, the containment will have to be vented again later in the accident when the pressure again increases above the vent pressure. There was no attempt in this analysis to model the opening and closing of the containment vent. Furthermore, since the availability of the containment purge system during POS 5 is not being modelled (and it is not required by the technical specification), it is assumed that the containment will be vented directly to the environment and will not pass through the containment purge system with its associated filters and charcoal beds.

- If the containment fails, it is assumed to fail above the auxiliary building roof, thereby allowing radioactive releases to bypass the auxiliary building and enter the environment directly. The enclosure building that surrounds the portion of the containment that is above the auxiliary building roof is estimated to offer essentially no attenuation to the release. This assumption is consistent with the assumption used in the Grand Gulf plant analysis performed as part of the NUREG-1150 study [Brown, et al., 1990].

- Since the assessed containment failure pressure at the 99th percentile is only 97 psig and the drywell pressure required to prevent the SRVs from opening must exceed 100 psi, failure of the SRVs due to high containment pressure is not considered in this study.

- The Grand Gulf Emergency Operating Procedures are applicable after the onset of core damage and the operators will continue to follow them.

- No operator actions were modelled that would require the operators to enter the containment or auxiliary building following the onset of core damage.

- Recovery of coolant injection after the onset of core damage is only considered if (1) injection systems were not available prior to core damage but become available following core damage, or (2) conditions occur following the onset of core damage that would cause the operators to use a system that was previously available but not used prior to core damage.

- If the reactor head vent is open and the vessel is pressurized, it is assumed that all of the in-vessel releases escape though the head vent and bypass the suppression pool. This assumption is based on results from MELCOR calculations performed for this study. If the vessel is depressurized prior to core damage or the vessel is breached by a LOCA, the in-vessel release will either pass through the SRVs and enter the suppression pool or escape out the break, which ever the case may be.

- It is assumed that the core cannot be cooled and vessel failure cannot be prevented by flooding the lower portion of the containment which submerges in water the lower portion of the lower vessel head.

- Although the containment is flooded in the LOCA PDSs, it is assumed that the break occurs above the water and, therefore, any releases that occur before vessel failure will not be scrubbed by water.

6.1.2 Overview of the APET

The APET for POS 5 considers the progression of the accident from the onset of core damage, defined in this analysis as the start of fuel heatup as predicted by the MELCOR code, through the completion of the interactions between the core debris and the concrete structure below the vessel. These interaction that occur between the core debris released from the vessel and the concrete structures below the vessel are termed core-concrete interactions or simply CCI. To model these accidents, the APET addresses the 59 events or questions listed in Table 6-1. The first seventeen questions are used to define the characteristics of the PDSs which form the initial conditions for the analysis. Following the definition of the PDSs, the questions in the APET are divided into four general time regimes: (1) before core damage, (2) during the in-vessel phase of the core damage process, (3) from vessel failure to the start of significant CCI, and (4) from the start of significant CCI to the end of the accident. Events that are considered before core damage include events that are important to the accident progression but that were not included in the Level 1 analysis. These events include operator actions associated with containment closure and the initiation the of hydrogen ignition system (HIS). Questions in the second time regime are address the core degradation and relocation process in the vessel and the status of plant features that can be used to mitigate the release. Events that are included during the in-vessel phase of the accident include events that address the recovery of core coolant, events that address the status of the reactor vessel integrity and the pressure in the reactor vessel, and events that address the status of the containment and auxiliary building. To address the status of containment integrity events associated with hydrogen combustion, containment heat removal, and containment venting are also considered. Since many of the systems that can be used to mitigate the accident depend on ac power, the recovery of ac power during a station blackout is also addressed in this section of the APET. Events that are included in the vessel failure time regime include events that determine the likelihood that the core debris is cooled in the vessel resulting in termination of the accident with the core in a safe stable condition. For accidents in which the core debris is not cooled and the vessel fails, events are included that address the phenomena associated with vessel failure (e.g., vessel melt-though, high pressure melt ejection, and steam explosions) and the accompanying loads. The response of the containment or auxiliary building to these loads are also assessed in this section of the APET. Events that are included in the time regime after vessel failure include events associated with CCI, the long-term pressurization of the containment from the steam and noncondensibles generated during the CCI process, and the status of the

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Table 6-1
POS 5 Accident Progression Event Tree Questions

No.	APET Questions
	Plant Damage State Definitions
1	What is the Plant Damage State?
2	What is the status of electric power at core damage (PDS Char. 1)?
3	What is the status of dc power at core damage (PDS Char. 1)?
4	What is the status of high pressure injection at core damage (PDS Char. 2)?
5	What is the status of low pressure injection at core damage (PDS Char. 2)?
6	What is the status of containment sprays and SPC at core damage (PDS Char. 3)?
7	What is the suppression pool level at the onset of core damage (PDS Char. 4)?
8	What is the suppression pool temperature at the onset of core damage (PDS Char. 5)?
9	What is the status of the reactor head vent at the onset of core damage (PDS Char. 6)?
10	What is the status of the RPV integrity at the onset of core damage (PDS Char. 7)?
11	What is the status of the containment access penetrations at the onset of core damage (PDS Char. 8)?
12	What is the status of the containment vents system at the onset of core damage (PDS Char. 9)?
13	When does core damage occur (PDS Char. 10)?
14	While in POS 5, when does the initiating event occur (PDS Char. 11)?
15	What type of event initiates the accident?
16	What is the pressure in the RPV at the time of core damage?
17	How much water is in the reactor pedestal cavity at the time of core damage?
	Events that Occur Before Core Damage
18	Is the containment equipment hatch opened at the start of the accident?
19	Do the operators close the containment before core damage?
20	Does the auxiliary building fail before core damage?
21	What is the status of the drywell before core damage?
22	Do the operators turn on the HIS before core damage?
	Events that Occur During Core Damage
23	Do the station batteries depleted during core damage?
24	Is offsite power restored during core damage?
25	Is the RPV isolated during core damage?
26	Do the operators initiate containment sprays during core damage?
27	Do the operators depressurize the RPV during core damage?
28	What is the status of the SRV vacuum breakers during core damage?
29	Is core cooling restored during core damage?
30	What is the peak hydrogen concentration in the containment during CD?
31	What is the fraction of zirconium that is oxidized in the vessel during core damage?
32	Do the operators turn on the HIS during CD ?
33	Does an uncontrolled hydrogen combustion event occur during CD?

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Table 6-1 (continued)

No.	APET Questions
34	What is the pressure in the containment during CD (no uncontrolled burn)?
35	Does the containment fail from quasi-static loads during core damage?
36	Do the operators vent the containment during core damage?
37	What is the status of the containment during core damage?
38	What is the size of the containment opening during core damage?
39	Does the auxiliary building fail during core damage?
	Events that Occur Around the Time of Vessel Failure
40	Is there water in the RPV pedestal cavity just prior to VB?
41	Is the core damage process arrested in the vessel?
42	What fraction of the core debris would be mobil at VB?
43	Does a large in-vessel steam explosion occur?
44	Does an Alpha mode event occur?
45	Does a large in-vessel steam explosion fail the vessel?
46	What is the mode of VB?
47	Does high pressure melt ejection occur?
48	Does a large ex-vessel steam explosion accompany VB?
49	Does the containment fail from pressure loads accompanying VB?
50	What is the status of containment integrity just after VB?
51	What is the size of the containment opening just after VB?
52	Does the auxiliary building fail just after VB?
	Events that Occur After Vessel Failure
53	What is the status of dc power late in the accident?
54	Is ac power recovery late in the accident?
55	Is the core debris in the cavity coolable?
56	Do the operators vent the containment after VB?
57	Does the containment fail late in the accident?
58	What is the status of the containment late in the accident?
59	What is the size of the containment opening late in the accident?

containment's integrity late in the accident. Because many of the systems that can be used to control the pressure in the containment depend on electric power (e.g., containment sprays and containment venting) the status of electric power is also addressed during this phase of the accident. By the end of the tree, the extent of core damage (i.e., only in-vessel releases versus both in-vessel and ex-vessel releases), the release path, and the status of the systems that can be used to mitigate the release have been identified. These features of the accident are then used as a basis for estimating the magnitude of the release in the radioactive release and transport analysis.

Each question included in the APET is discussed in detail in Appendix B.1. Because of the number of questions included in the APET, it is not practical to represent the tree graphically. Instead, Boolean statements are used to represent the tree which is then evaluated with the EVNTRE code [Griesmeyer and Smith 1989]. The Boolean representation of the POS 5 APET is provided in Appendix B.2.

6.2 Quantification of Accident Progression Model

The quantification of the APET consists primarily of assigning probabilities to the branches in the APET. These branches represent the possible outcomes for the various events included in the tree (i.e., the occurrence of human actions, the system responses, the occurrence of phenomenological events). The probabilities in the POS-5 APET were quantified using information from the following sources:

- Level 1 Analysis: The frequencies for the PDSs were obtained from the Level 1 analysis described in Volume 2 of this report.

- HRA analysis: A Human Reliability Analysis (HRA) was performed to determine the human error probability for operator actions during the core damage process (e.g., containment closure and the recovery of core cooling and containment cooling functions). For the sake of consistency, wherever possible, the same HRA models and techniques used in the Level I analysis were also used in this study. The results from the HRA analysis are provided in Appendix B.3.

- MELCOR calculations: A series of MELCOR calculations were performed specifically for this study. Results from these calculations helped guide the development and quantification of the APET. Specifically, the MELCOR calculation were used to determine the timing of key events (e.g., onset of core damage, vessel failure, containment failure) and the pressure, temperature, and composition histories of the containment and auxiliary building. The MELCOR calculations are documented in Volume 6, Part 2 of this report.

- Data from the NUREG-1150 PRAs: Where appropriate, data used in the NUREG-1150 full power PRA of Grand Gulf was also used in this study (e.g., structural capacity of the containment to static loads).

For those events that were judged to be important to risk and for which there was a large amount of uncertainty as to the value to assigned to the branch probability, an uncertainty distribution was assigned to the probability. Twenty three variables in the APET were included in the uncertainty analysis. For the remaining events that were either judged to be less importance or for which the branch probability was not believed to be uncertain, a single value was used. The primary sources of information used to quantify the question in the APET listed in Appendix B.3.1. Also, if the question was included in the uncertainty analysis, B.3.1 identifies the distribution that was used to characterize the uncertainty and the variable name.

6.3 Accident Progression Bins

As each path through the APET is evaluated, the result of that evaluation is stored by assigning it to an Accident Progression Bin (APB). The APBs are the means by which information is passed from the accident progression analysis to the source term analysis (see Section 7) and as such the bin describes the evaluation in enough detail that a source term (the release of radioactive material) can be estimated for it. The binning scheme for the POS 5 analysis utilizes fourteen characteristics or quantities which define a certain feature of the accident progression (the definition of the APBs is analogous to the definition of PDSs). A bin is defined by specifying a letter for each of the 14 characteristics, where each letter for each characteristic has a certain meaning. For each characteristic, the possible states are termed attributes. The selection of the characteristics and attributes is based on the information that is needed in the radionuclide release and transport analysis to estimate the source term. The fourteen characteristics used in this analysis are identified and described in Table 6-2. The attributes for each characteristic are identified and described in Table 6-3.

6.4 Evaluation of Accident Progression Event Tree

The Grand Gulf POS 5 APET was evaluated using the EVNTRE code [Griesmeyer and Smith, 1989]; the results from EVNTRE were post processed with the PSTEVNT code [Higgins, 1989]. All of the plant damage states are evaluated in a single APET and EVNTRE run and, hence, the probabilities for the APBs generated in this analysis are conditional on the occurrence of core damage, not a particular PDS. A path in the APET was dropped from the analysis when its

Table 6-2
Accident Progression Bin Characteristics

Characteristic No.	Abbreviation	Description
1	PDS	Identifies the plant damage state
2	CNT-STATUS	Identifies the status of containment integrity during the various stages of the accident
3	AUX-STATUS	Identifies the status of auxiliary building integrity during the various stages of the accident
4	DW-STATUS	Identifies the status of drywell integrity at the start of the accident
5	RPV-ISO	Identifies the status of the reactor vessel integrity prior to core damage
6	RPV-VNT	Identifies the status of the reactor head vent before core damage
7	SRV-VBkr	Identifies the status of the SRV tailpipe vacuum breaker during core damage
8	RPV-VB	Identifies both the pressure in the reactor and the status of core coolant at the time of vessel failure
9	CNT-SPRAYS	Identifies the status of containment sprays during core damage
10	ZROXID-CD	Identifies the amount of zirconium oxidized during core damage
11	HPME-SE	Identifies the occurrence of high pressure melt ejection and steam explosion events
12	TYPE-CCI	Identifies the coolability of the core debris in the reactor pedestal cavity following vessel failure
13	IE-TIME	Identifies the time window during POS 5 in which the accident occurs
14	SP-TEMP	Identifies the temperature of the suppression pool at the onset of core damage

probability dropped below 1.0E-07. Thus, the APBs consist of groups of individual progressions with each progression having a conditional probability of at least 1.0E-07. The logic used to form the APBs is included in the APET logic model which is described in Appendix B.2. The uncertainties associated with important events that affect the accident progression analysis were propagated through the APET using a stratified form of simple random sampling call Latin Hypercube Sampling (LHS) (see Section 2 for more discussion). In this technique, the APET is evaluated many times using different sets of inputs for each evaluation. The entire set of inputs is called the sample whereas the set of inputs used for a single evaluation is called an observation. A sample that consisted of 200 observations was used in this analysis. The evaluation of the APET resulted in the generation of 242 unique APBs.

6.5 Results from Accident Progression Analysis

Since there are far too many APBs to present and discuss each one individually, only aspects of the accident progression that have a major affect on the source term and risk will be discussed in this subsection. Features of the accident that can have a major impact on the amount of radioactive material released to the environment include the containment's

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Table 6-3
Accident Progression Bin Definitions

Attribute	Mnemonic	Description
Characteristic 1: Plant Damage State (PDS)		
A - P	PDS1-1 - PDS3-1	Attributes A through P identify individually the 16 PDSs
Characteristic 2: Containment Status (CNT-STATUS)		
A	OCnt-BCD	The containment equipment hatch is open before core damage and remains open throughout the accident.
B	Vnt-CD	The containment is vented during core damage.
C	Cnt-Rpt-CD	The containment fails, via a rupture in the containment wall above the auxiliary building roof, during core damage.
D	Cnt-Lk-CD	The containment fails, via a leak in the containment wall above the auxiliary building roof, during core damage.
E	Cnt-Rpt-VB	The containment fails, via a rupture in the containment wall, from loads accompanying vessel failure.
F	Cnt-Lk-VB	The containment fails, via a leak in the containment wall, from loads accompanying vessel failure.
G	Vnt-Late	The containment is vented during the late time regime
H	Cnt-Rpt-Late	The containment fails, via a rupture in the containment wall, from loads that occur late in the accident.
I	Cnt-Lk-Late	The containment fails, via a leak in the containment wall, from loads that occur late in the accident.
J	Cnt-NF-Late	The containment is closed prior to core damage and its pressure boundary is maintained during the remaining portion of the accident.
Characteristic 3: Auxiliary building pressure integrity status - AUX-STATUS		
A	OAux-BCD	The containment is open during the accident and the auxiliary building fails from overpressurization prior to the onset of core damage.
B	OAux-CD	The containment is open during the accident and the auxiliary building fails from overpressurization during core damage.
C	OAux-VB	The containment is open during the accident and the auxiliary building fails from overpressurization after vessel failure.
D	nOAux	The containment is closed prior to the onset of core damage.

Table 6-3 (continued)

Attribute	Mnemonic	Description
Characteristic 4: Drywell pressure integrity status (DW-STATUS)		
A	Op-DW-BCD	The drywell equipment hatch and/or personnel lock are open at the start of the accident. This is the case that is assumed in this analysis.
B	Cls-DW-BCD	The drywell equipment hatch and personnel lock are closed prior to the onset of core damage.
Characteristic 5: Status of the reactor vessel pressure boundary (RPV-ISO)		
A	Iso-RPV-E	The reactor vessel is isolated and its pressure boundary is intact prior to core damage.
B	RPV-LOCA	The accident is initiated by a pipe break in the drywell (i.e., LOCA), thus, the reactor vessel pressure boundary is breached prior to core damage.
C	Iso-RPV-CD	The reactor vessel was not isolated prior to core damage (either open MSIVs or an unisolated interfacing systems LOCA) and the vessel is not isolated during core damage. It is assumed that any releases from the vessel, prior to vessel failure, will bypass the containment and will pass directly into the auxiliary building.
D	nIso-RPV-CD	The reactor vessel was not isolated prior to core damage (either open MSIVs or an unisolated interfacing systems LOCA), the vessel is, however, isolated during core damage.
Characteristic 6: Status of the reactor head vent prior to core damage (RPV-VNT)		
A	RPV-nVnt	The reactor head vent is closed during core damage.
B	RPV-OVnt	The reactor head vent is open throughout the accident.
Characteristic 7: Status of the SRV tailpipe vacuum breakers (SRV-VBkr)		
A	OSRV-VBkr	A vacuum breaker on the SRV tailpipe that is being used to relieve the pressure in the RPV sticks open either before or during core damage.
B	cSRV-VBkr	None of the vacuum breakers on the SRV tailpipes stick open prior to vessel failure.
Characteristic 8: Status of the reactor vessel just prior to vessel failure (RPV-VB)		
A	RPV-HiP-nInj	The vessel is pressurized (i.e., near system pressure) just prior to vessel failure and core coolant is not being injected into the vessel.
B	RPV-LoP-nInj	The pressure in the vessel is less than 200 psig just prior to vessel failure and core coolant is not being injected into the vessel.
C	RPV-HiP-Inj	The vessel is pressurized (i.e., near system pressure) just prior to vessel failure and core coolant is being injected into the vessel.
D	RPV-LoP-Inj	The pressure in the vessel is less than 200 psig just prior to vessel failure and core coolant is being injected into the vessel.

Table 6-3 (continued)

Attribute	Mnemonic	Description
Characteristic 8: Status of the reactor vessel just prior to vessel failure (RPV-VB) (continued)		
E	nVB-HiP	Core cooling is restored with the vessel at high pressure and the core damage process is arrested in the vessel and the accident is terminated with the core in a safe stable state (e.g., TMI).
F	nVB-LoP	Core cooling is restored with the vessel at low pressure and the core damage process is arrested in the vessel and the accident is terminated with the core in a safe stable state (e.g., TMI).
Characteristic 9: Status of containment sprays (CNT-SPRAYS)		
A	nCS-CD	Containment sprays are not used during the core damage process.
B	CS-CD	Containment sprays are used during the core damage process.
Characteristic 10: Fraction of zirconium oxidized in the vessel prior to vessel failure (ZROXID-CD)		
A	ZrOxid-Hi	The fraction of zirconium oxidized in the vessel prior to vessel failure is greater than 0.21.
B	ZrOxid-Lo	The fraction of zirconium oxidized in the vessel prior to vessel failure is less than 0.21.
Characteristic 11: Fraction of core participating in HPME or steam explosions (HPME-SE)		
A	HiHPME	Forty percent of the core participates in HPME
B	LoHPME	Ten percent of the core participates in HPME
C	HiEXSE	An HPME event does not occur, however, 40% percent of the core participates in an ex-vessel steam explosion.
D	LoEXSE	An HPME event does not occur, however, 10% of the core participates in an ex-vessel steam explosion.
E	nHPME-SE	Neither an HPME event nor an ex-vessel steam explosion occurs.
Characteristic 12: Status of the core debris in the reactor pedestal cavity (TYPE-CCI)		
A	DryCCI	Core-concrete interactions proceed in a dry cavity following vessel failure.
B	FldCCI	Core-concrete interactions proceed in a flooded cavity following vessel failure.
C	noCCI	The core debris in the cavity is quenched and core-concrete interactions are avoided.

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Table 6-3 (continued)

Attribute	Mnemonic	Description
Characteristic 8: Status of the reactor vessel just prior to vessel failure (RPV-VB) (continued)		
E	nVB-HiP	Core cooling is restored with the vessel at high pressure and the core damage process is arrested in the vessel and the accident is terminated with the core in a safe stable state (e.g., TMI).
F	nVB-LoP	Core cooling is restored with the vessel at low pressure and the core damage process is arrested in the vessel and the accident is terminated with the core in a safe stable state (e.g., TMI).
Characteristic 9: Status of containment sprays (CNT-SPRAYS)		
A	nCS-CD	Containment sprays are not used during the core damage process.
B	CS-CD	Containment sprays are used during the core damage process.
Characteristic 10: Fraction of zirconium oxidized in the vessel prior to vessel failure (ZROXID-CD)		
A	ZrOxid-Hi	The fraction of zirconium oxidized in the vessel prior to vessel failure is greater than 0.21.
B	ZrOxid-Lo	The fraction of zirconium oxidized in the vessel prior to vessel failure is less than 0.21.
Characteristic 11: Fraction of core participating in HPME or steam explosions (HPME-SE)		
A	HiHPME	Forty percent of the core participates in HPME.
B	LoHPME	Ten percent of the core participates in HPME.
C	HiEXSE	An HPME event does not occur, however, 40% percent of the core participates in an ex-vessel steam explosion.
D	LoEXSE	An HPME event does not occur, however, 10% of the core participates in an ex-vessel steam explosion.
E	nHPME-SE	Neither an HPME event nor an ex-vessel steam explosion occurs.
Characteristic 12: Status of the core debris in the reactor pedestal cavity (TYPE-CCI)		
A	DryCCI	Core-concrete interactions proceed in a dry cavity following vessel failure.
B	FldCCI	Core-concrete interactions proceed in a flooded cavity following vessel failure.
C	noCCI	The core debris in the cavity is quenched and core-concrete interactions are avoided.

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integrity during the accident, the recovery of core cooling and arrest of the core damage process in the vessel, the coolability of the core debris that is released from the vessel and the availability of the plant features that can be used to attenuate the release such as the suppression pool and the containment sprays. In none of these accidents are the containment sprays available. Simplified representations of the APET that address these aspects of the accident for all PDSs considered together, the LOCA PDS considered as a group, the Station Blackout (SBO) PDSs considered as a group, and the "Other" PDSs considered as a group are shown in Figures 6-1, 6-2, 6-3, and 6-4, respectively. The "Other" PDS group consists of those PDSs that are not LOCAs and are not SBO and includes the following PDSs: PDS1-5, PDS2-4, PDS2-5, and PDS2-6. Figure 6-1 is conditional on the occurrence of core damage whereas the other figures are conditional on the occurrence of the PDS under consideration. The values displayed in these figures are mean conditional probabilities.

From the simplified tree presented in Figure 6-1, it can be seen that the most likely accidents in POS 5 have an open containment, the suppression pool is bypassed, and the vessel fails. The uncertainty in the probability that the containment is closed prior to core damage and the uncertainty in the probability that core cooling is restored and vessel failure is prevented is displayed in Figure 6-5. The probabilities displayed in Figure 6-5 are conditional on the occurrence of core damage. For the cases where the vessel fails, there is a significant probability that the core debris will either be quenched in a flooded cavity or CCI will occur in a flooded cavity. For the former, the releases associated with CCI are prevented. In latter, the releases are scrubbed by the water in the flooded cavity. If the containment is closed prior to core, it is always predicted to either fail or to be vented after core damage since containment heat removal is not available in these accidents; venting the containment late in the accident is the most likely scenario. Given that the containment is closed prior to core damage the uncertainties in the probabilities that the containment fails early in the accident (i.e., early is defined as during core damage or at the time of vessel failure), that the containment is vented late in the accident, and that the containment fails late in the accident are displayed in Figure 6.6. From this figure it is clear to see that if the containment is closed, venting the containment late in the accident is the most likely event.

For the LOCA PDS group the containment is always open, the suppression pool is always bypassed, and core cooling never restored. Without core cooling the vessel is always predicted to fail. Since the containment was flooded prior to core damage in these PDSs, the core debris ejected from the vessel is released into a pool of water. The mean probability that the core debris is quenched and CCI is avoided is 0.38. For those cases where CCI does occur, the releases will always be scrubbed by the water in the flooded cavity. Thus, while the vessel is always predicted to fail, in half of these cases the core debris is quenched. The primary difference in the source term between cases with core damage arrest and those cases with no CCI, is that for the latter cases it is still possible for ex-vessel steam explosions or debris ejected at high pressure to contribute to the release of radioactive material (in the source term model used for this analysis there is also a "puff" release associated with vessel failure, however, this release is small compared to the sources) and, hence, the source term associated with the latter cases tends to be larger. Thus, for the LOCA accidents the release associated with the core damage process can be large since the suppression pool is bypassed and the containment sprays are unavailable. The ex-vessel releases, however, may be limited because there is a substantial probability that the core debris will be quenched and for those cases where CCI does occur the releases will be scrubbed by a pool of water. For the LOCA accidents, the primary features that can attenuate the release are the auxiliary building and the pool of water form when the containment was flooded.

For the SBO PDS group the containment is always open and the suppression pool is bypassed[1]. While there are accidents in this PDS group in which ac power can be recovered and core cooling restored, the mean probability that the core damage process is arrested and vessel failure prevented is only 0.04. For those accident that do involve vessel failure the core debris will be released into a dry cavity. Without a means to cool the core debris in the cavity, CCI is

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1 For station blackout accidents that involve a break in the SDC system which is subsequently isolated when ac power is recovered, it is conservatively assumed that all of the in-vessel releases escape out the break and bypass the suppression pool before the break is isolated. This assumption is made because it is not known when during core damage that ac power is restored and the vessel isolated.

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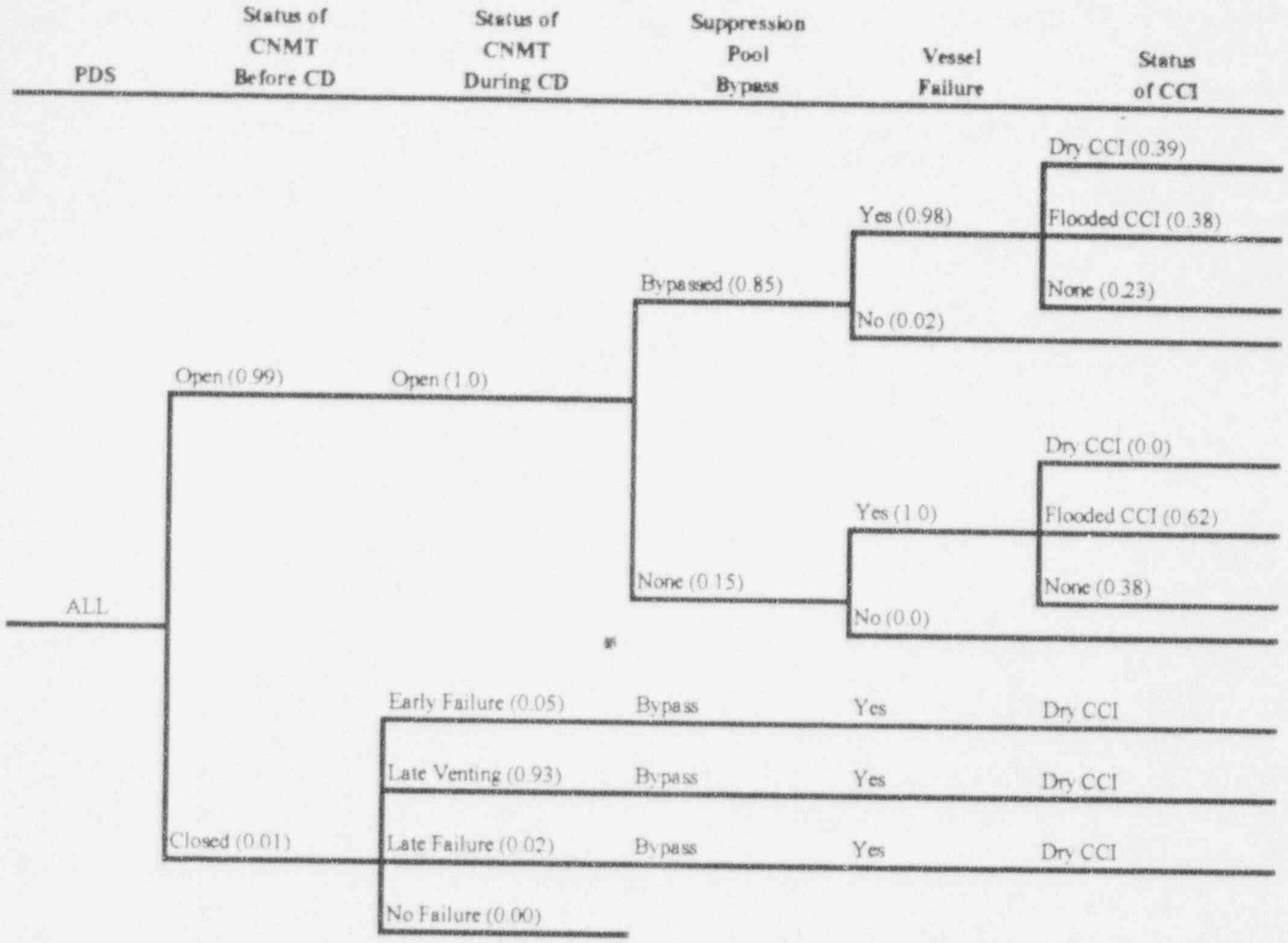

Figure 6-1
Simplified Representation of POS 5 Accident Progressions

certain to occur. Hence, for this PDS group the most likely scenario is that the vessel will fail and CCI will occur in a dry cavity. Because the in-vessel releases bypass the vessel and CCI occurs in a dry cavity, the releases associated with these accident can be large.

As can be seen from Figure 6-4, for the "Other" PDS group, many different accident progressions are possible. Of the three PDS groups discussed, this is the only group in which the containment can be closed prior to core damage. The mean probability that the containment is close, however, is only 0.09. If the containment is closed prior to core, it is always predicted to either fail or to be vented after core damage; venting the containment late in the accident is the most likely scenario. The progressions in which the containment is closed prior to core damage stem from PDS2-5. In this

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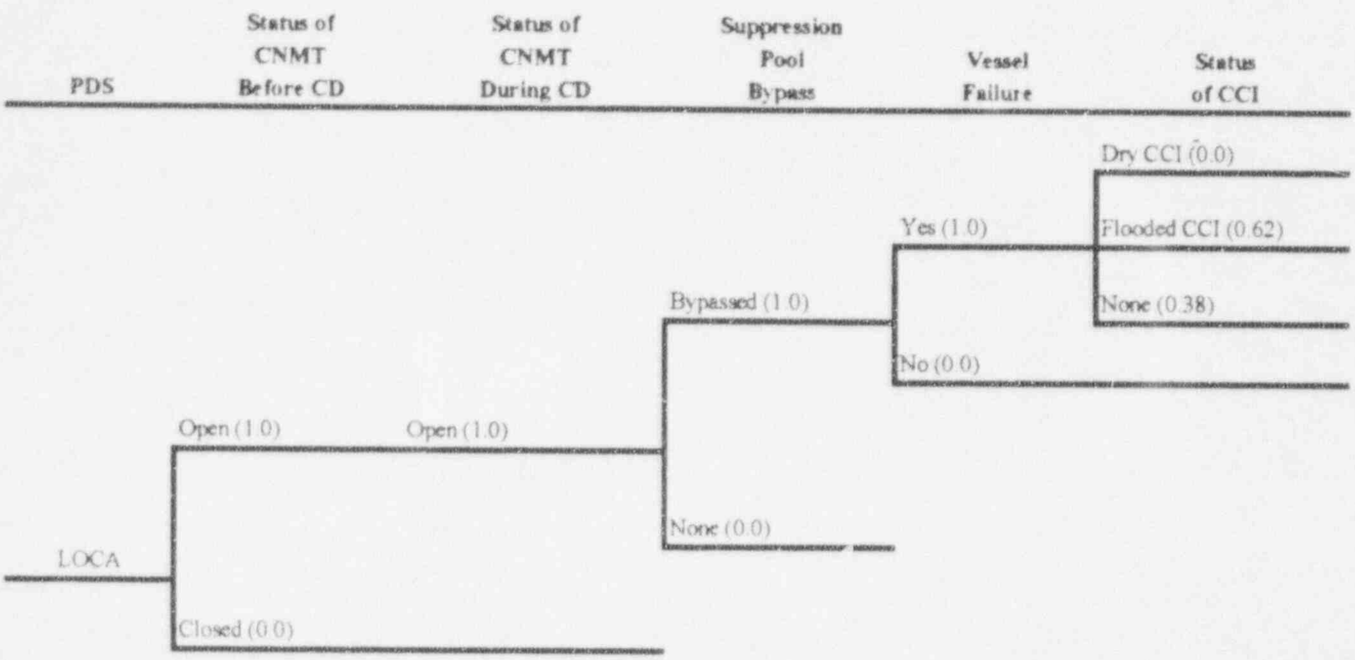

Figure 6-2
Simplified Representation of LOCA PDS Group Accident Progression

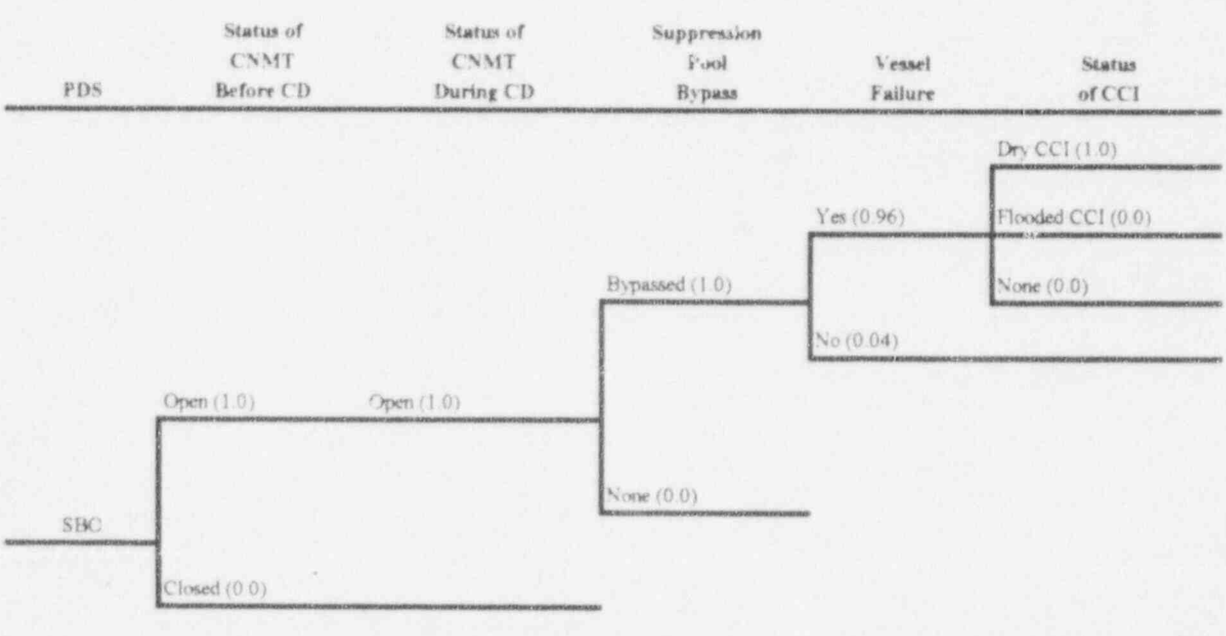

Figure 6-3
Simplified Representation of Station Blackout PDS Group Accident Progression

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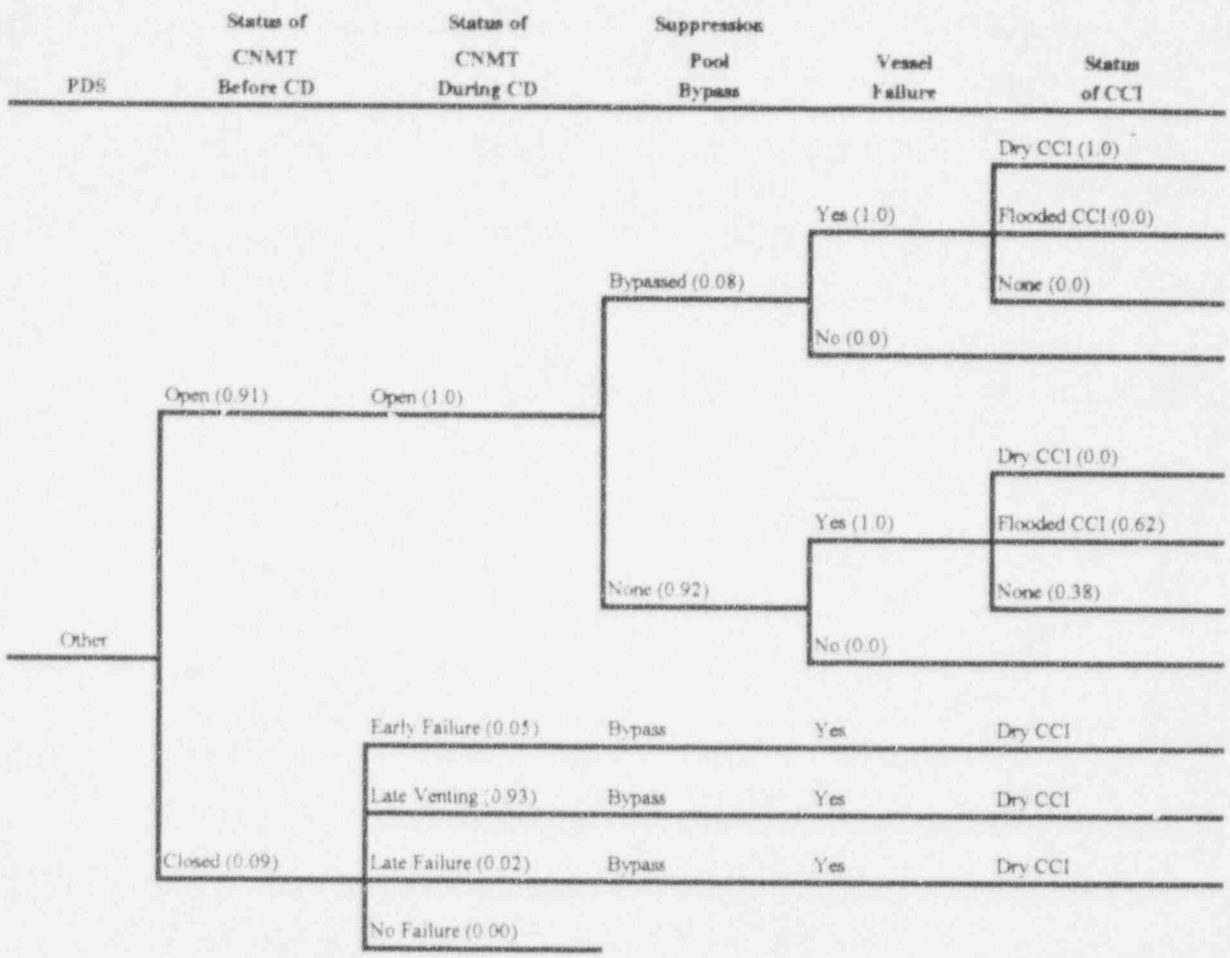

Figure 6-4
Simplified Representation of "Other" PDS Group Accident Progression

PDS, the suppression pool is always bypassed[2], core cooling is never restored, and CCI always occurs in a dry cavity. The accidents in which the containment remains open during the entire accident consist of cases where the suppression pool is bypass and cases where the radioactive material released during the core damage process passes through the suppression pool. For the cases where the suppression pool is bypassed the vessel is always predicted to fail and the resulting CCI occurs in a dry cavity. For these accidents, there is very little attenuation of the release. The accidents that allow the in-vessel releases to pass through the suppression pool also have a flooded cavity. In these cases the vessel is always predicted to fail, however, there is a substantial probability that the core debris released from the vessel will be quenched. If the debris is not quenched, the CCI will occur under a pool of water and any radioactive releases

[2] In PDS2-5 the reactor head vent is open and the vessel is pressurized at the onset of core damage. If the reactor remains at high pressure all of the releases will escape out the vent and bypass the suppression pool. If, however, the vessel is at low pressure the release will pass through the SRVs and enter the suppression pool. While it is likely that the operators will depressurize the vessel during core damage, it is not known when during core damage the operators will perform the action. Hence, it is conservatively assumed that the release escapes through the head vent while the vessel is pressurized.

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will be scrubbed by the water. For this, both the in-vessel and a portion of the ex-vessel release will be scrubbed by a pool of water and, therefore, the it is likely that the release will be less than the previous case.

The configuration of the plant in POS 5 is such that there are very few plant features that are available to attenuate the release once core damage occurs. Contrary to full power, in many of accidents that could occur during POS 5 the containment is open at the start of the accident, the suppression pool is bypassed, and the containment sprays are not available. Also in these POS 5 accidents, there are very few PDSs where it is possible to recover core cooling and arrest the core damage process in the vessel. Without these features to attenuate the release, a large release to the environment is possible.

6.6 References

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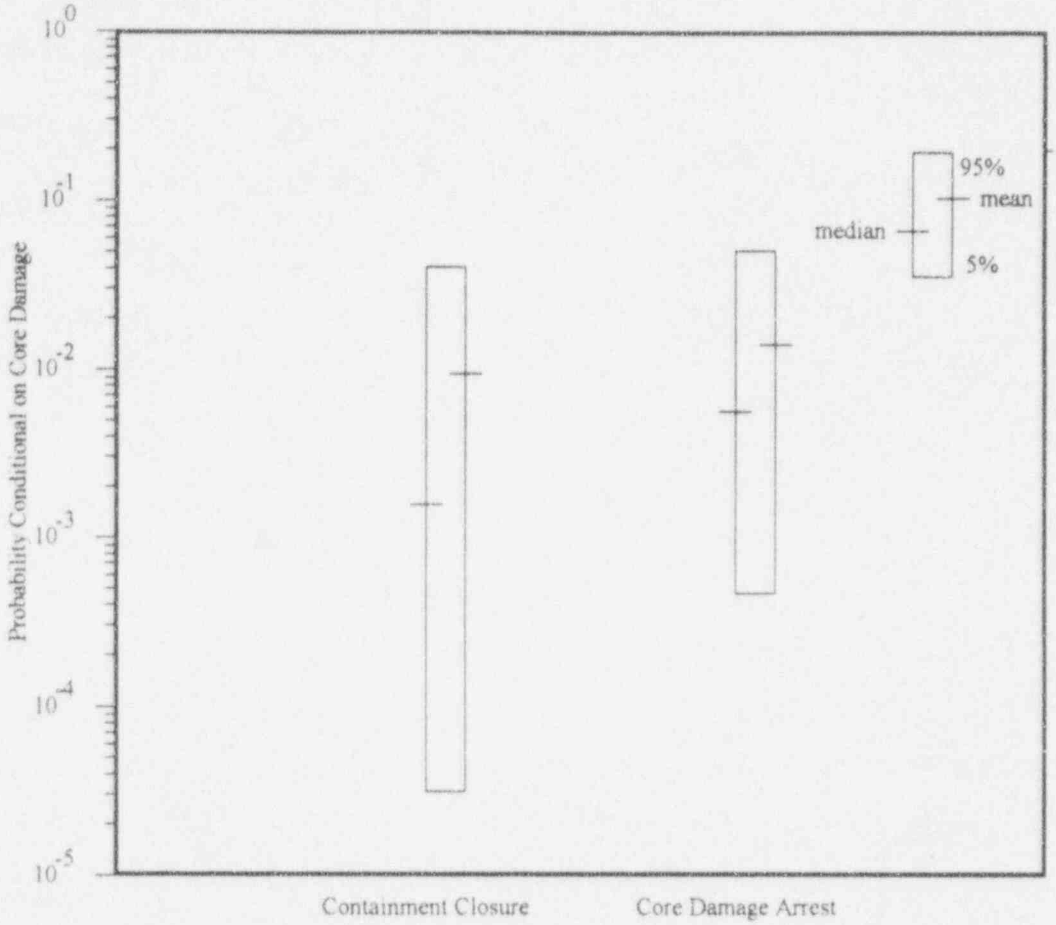

Figure 6-5
Probability of Containment Closure and Probability of Core Damage Arrest

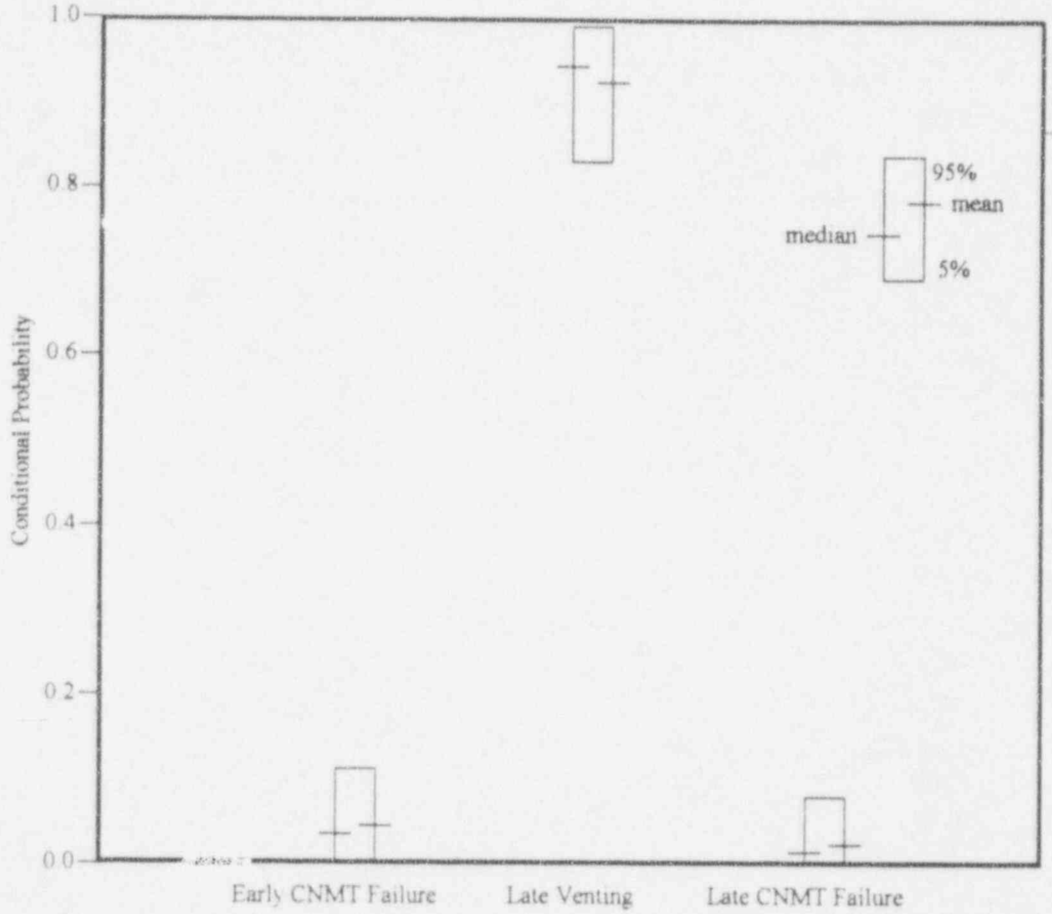

Figure 6-6
Probability of Containment Failure Given that is was Closed Before Core Damage

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7 Radionuclide Release and Transport Analysis

The Radionuclide Release and Transport Analysis, commonly referred to as the Source Term Analysis, addresses the release of radioactive material from the fuel and core debris and its subsequent transport and deposition in the primary system and containment. The inputs to the source term analysis are the accident progression bins (APBs) defined in the Accident Progression Analysis. The APBs describe the configuration of the plant, the status of systems that can be used to mitigate the release, and the occurrence of phenomena that can impact the source term. The product of the analysis is a collection of parameters, referred to as the source term, that characterizes the release of radioactive material from the containment[1] to the environment. The source term is then used in the consequence analysis to estimate health effects attributable to the release of radioactive material for each APB.

7.1 Definition of the Source Term

The source term, as defined in this analysis, consists of the following information: the amount and type of radioactive material released from the containment, timing characteristics of the release, the energy of the release, the elevation of the release, and the time at which a general emergency is declared and evacuation plans initiated (i.e., referred to as the warning time). The amount of material released is expressed as a fraction of the radionuclide[2] inventory present in the core at the start of the accident for the radionuclides considered in this analysis. While many different radionuclides would be released from the damaged fuel during an accident, health effects from only 60 radionuclides are considered in the consequence analysis. Furthermore, since in the source term analysis the release and transport of these radionuclides are of concern and not the health effects from the radionuclides, it is possible to form groups of radionuclides that are expected to have similar release and transport characteristics. The 60 radionuclides considered in the consequence analysis are combined into the nine release classes shown in Table 7-1. The definition for each release class is based on the definitions used in the NUREG-1150 study [Gorham, et al., 1993]. Similar to the NUREG-1150, the timing of the release is characterized by two release segments: the first or early release segment and the second or late release segment. For each release segment, the time when the release segment begins, relative to the start of the accident, the duration of the segment, release fractions for the nine release classes and the energy release rate are provided.

7.2 Description of the Source Term Model

The source term is estimated using a modified version of the parametric code GGSOR that was developed for the NUREG-1150 Grand Gulf plant study [Brown, et al., 1990]. Since the concept of the parametric approach for estimating source terms is discussed in Section 2.0 and the code itself is described in detail in the XSOR Users Manual [Jow, et al., 1993], it will not be described in detail in this report. Instead, the basic parametric equation will be presented and aspects of the model that were modified will be identified. The version of GGSOR that was used in this analysis, GGSOR-P5, is listed in Appendix C. An assessment of the ability of the XSOR codes to produce source terms for PRA purposes is provided in NUREG/CR-5346 [Cybulskis, et al., 1989].

GGSOR was modified to reflect the configuration of the plant during POS 5 and the types of accidents that are possible during this mode of operation. Specifically, GGSOR was modified to account for: (1) LOCAs in the containment, (2) interfacing system LOCAs in the auxiliary building, (3) the passage of releases through the auxiliary building, (4) the passage of releases through the reactor pressure vessel head vent, and (5) the timing characteristics of the accidents initiated while the plant is in POS 5.

GGSOR accounts for two releases from the containment. The first release occurs roughly at the time of containment failure. The second release begins after the first release has finished. When the containment is open prior to core

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1 In this context, containment is generalized to include the region in which engineered features are available to attenuate a release before it enters the environment (e.g., the auxiliary building that surrounds the containment building).

2 The terms radionuclide, isotope, and fission product are used interchangeably in this report.

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Table 7-1
Isotopes in Each Radionuclide Release Class

Release Class	Isotopes Included
1. Inert Gases	Kr-85, Kr-85M, Kr-87, Kr-88, Xe-133, Xe-135
2. Iodine	I-131, I-132, I-133, I-134, I-135
3. Cesium	Rb-86, Cs-134, Cs-136, Cs-137
4. Tellurium	Sb-127, Sb-129, Te-127, Te-127M, Te-129, Te-129M, Te-131M, Te-132
5. Strontium	Sr-89, Sr-90, Sr-91, Sr-92
6. Ruthenium	Co-58, Co-60, Mo-99, Tc-99M, Ru-103, Ru-105, Ru-106, Rh-105
7. Lanthanum	Y-90, Y-91, Y-92, Y-93, Zr-95, Zr-97, Nb-95, La-140, La-141, La-142, Pr-143, Nd-147, Am-241, Cm-242, Cm-244
8. Cerium	Ce-141, Ce-143, Ce-144, Np-239, Pu-238, Pu-239, Pu-240, Pu-241
9. Barium	Ba-139, Ba-140

damage or fails before vessel breach, the first release is due to fission products that escape from the fuel while the core is still in the RPV (i.e., in-vessel releases) and releases that occur at the time of vessel failure. For this case, the second release includes fission products that are released after vessel breach. Releases after vessel breach, referred to as the late releases, include fission products from core-concrete interactions (CCI), material revolatilized from the RPV after vessel breach and iodine released from the suppression pool (and in some cases the RPV cavity water). For situations where the containment fails many hours after vessel breach, both release segments consist of in-vessel releases, fission products released at vessel breach, and the late releases. The timing and duration of these releases depend primarily on the PDS and the time and mode of containment failure.

For radionuclide class i, the basic parametric equation for GGSOR has the following form:

$$
\begin{aligned}
\mathrm{ST}_i \quad & \qquad\qquad\qquad\qquad\qquad\qquad\qquad\qquad\qquad\qquad (7.1) \\
= {} & \mathrm{FCOR}_i*\mathrm{FVES}_i*[(1.0 - \mathrm{HVSPLT})*(\mathrm{RELF1} + \mathrm{RELF2} + \mathrm{RELF3}) + \mathrm{HVSPLT}/\mathrm{DFCAV}_i]*\mathrm{FCONV}_i/\mathrm{RBDF}_i \\
& + \mathrm{VBPUF}_i*(\mathrm{RELF4} + \mathrm{RELF5})*\mathrm{FCONC}_i/\mathrm{RBDF}_i \\
& + (1 - \mathrm{FCOR}_i - \mathrm{VBPUF}_i)*\mathrm{FLV}*\mathrm{EVSE}*\mathrm{FEVSE}_i*(\mathrm{RELF6} + \mathrm{RELF7})*\mathrm{FCONC}_i/\mathrm{RBDF}_i \\
& + (1 - \mathrm{FCOR}_i - \mathrm{VBPUF}_i)*\mathrm{FLV}*\mathrm{FHPE}*\mathrm{FDCH}_i*(\mathrm{RELF6} + \mathrm{RELF7})*\mathrm{FCONC}_i/\mathrm{RBDF}_i \\
& + (1 - \mathrm{FCOR}_i - \mathrm{VBPUF}_i)*\mathrm{FLV}*\mathrm{XCCI}*\mathrm{FCCI}_i*(\mathrm{RELF8} + \mathrm{RELF9})*\mathrm{FCONC}_i/\mathrm{RBDF}_i \\
& + \mathrm{FCOR}_i*(1 - \mathrm{FVES}_i)*\mathrm{FREVO}_i*(\mathrm{RELF10} + \mathrm{RELF11})*\mathrm{FCONC}_i/\mathrm{RBDF}_i \qquad (i=2,\ 3,\ \&\ 4\ \mathrm{ONLY}) \\
& + [\mathrm{FLTI1}*\mathrm{POOLI} + \mathrm{FLTI2}*\mathrm{CAVWI}*(\mathrm{RELF12} + \mathrm{RELF13})]*\mathrm{RELF14},
\end{aligned}
$$

where

$$
\begin{aligned}
\mathrm{RELF1} &= \mathrm{FTLP}*\mathrm{FPLBYE}/\mathrm{DFSPRV}_i, \\
\mathrm{RELF2} &= \mathrm{FTLP}*(1 - \mathrm{FPLBYE})/\mathrm{MAX}(\mathrm{DFCPA}_i,\mathrm{DFSPRV}_i), \\
\mathrm{RELF3} &= (1 - \mathrm{FTLP})/\mathrm{MAX}(\mathrm{DFVPA}_i,\mathrm{DFSPRV}_i),
\end{aligned}
$$

RELF4 = FPLBYP/DFSPRC$_i$,
RELF5 = (1 - FPLBYP)/MAX(DFCPA$_i$,DFSPRC$_i$),
RELF6 = FPLBYD/DFSPRC$_i$,
RELF7 = (1 - FPLBYD)/MAX(DFCPA$_i$,DFSPRC$_i$),
RELF8 = FPLBYC/MAX(DFCAV$_i$,DFSPRC$_i$),
RELF9 = (1 - FPLBYC)/MAX(DFCAV$_i$,DFCPA$_i$,DFSPRC$_i$),
RELF10 = FPLBYC/DFSPRC$_i$,
RELF11 = (1 - FPLBYC)/MAX(DFCPA$_i$,DFSPRC$_i$),
RELF12 = FPLBYC/DFCPA$_i$,
RELF13 = (1 - FPLBYC)/DFCPA$_i$,
RELF14 = FCONC(1) (if no containment failure, use FCONC for Noble gases),
= 1.0 (if containment failure),
XCCI = 1 - FHPE (if DCH occurs),
= 1 - EVSE (if an ex-vessel steam explosion occurs),
= 1.0 (if neither DCH nor ex-vessel steam explosion occurs).

The first summation term on the right side of Equation 7.1 represents the in-vessel releases. The second term describes the puff release at vessel breach. The third term represents the ex-vessel steam explosion release. The fourth term represents the DCH release and is mutually exclusive with the third term. The fifth term represents the CCI release. The sixth term is the revolatilization release from the reactor coolant system after vessel breach and is for I, Cs, and Te classes only. The last term represents the evolution of iodine from the suppression pool and reactor cavity water late in the accident. The definitions of the various parameters in Equation 7.1 are as follows:

CAVWI = fraction of initial iodine core inventory scrubbed by the cavity water during the CCI release,

DFSPRC$_i$ = scrubbing decontamination factor for sprays acting on species i released into containment after vessel breach,

DFSPRV$_i$ = scrubbing decontamination factor for sprays acting on species i released into containment from the vessel before vessel breach,

DFCAV$_i$ = scrubbing decontamination factor for aerosol species i released into cavity water during CCI release,

DFCPA$_i$ = scrubbing decontamination factor for aerosol species i flowing from drywell to the suppression pool,

DFVPA$_i$ = scrubbing decontamination factor for aerosol species i flowing from the vessel to the suppression pool,

FCCI$_i$ = fraction of material released from the melt during molten CCI,

FCONC$_i$ = fraction of species i released from containment for CCI and other releases after vessel breach, not including the effects of scrubbing by pools and sprays,

FCONV$_i$ = fraction of species i released from containment for material released into containment before vessel breach, not including the effects of scrubbing by pools and sprays,

FCOR$_i$ = fraction of initial inventory of species i released from the fuel prior to vessel failure,

FDCH$_i$ = fraction of species i in the portion of the core involved in direct containment heating that is released to the drywell at vessel breach,

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EVSE = fraction of core material leaving the vessel that participates in an ex-vessel steam explosion and, therefore, is not available to participate in CCI,

$FEVSE_i$ = fraction of species i in the portion of the core involved in an ex-vessel steam explosion that is released to the drywell at vessel breach,

FHPE = fraction of core material leaving the vessel that participates in direct containment heating and, therefore, is not available to participate in CCI,

FLV = fraction of the core material that leaves the vessel after the vessel breach,

$FREVO_i$ = fraction of species i that is deposited on the surfaces of the reactor vessel and structural materials that is revaporized and released in the drywell after vessel breach,

FPLBYC = fraction of CCI releases that bypass the suppression pool,

FPLBYD = fraction of DCH releases or ex-vessel steam explosion releases that bypass the suppression pool,

FPLBYE = fraction of in-vessel releases that bypass the suppression pool,

FPLBYP = fraction of puff releases at vessel breach that bypass the suppression pool,

FTLP = fraction of the in-vessel releases that are released into the drywell through stuck-open SRV tailpipe vacuum breakers,

$FVES_i$ = fraction of species i released from the fuel that is released from the vessel,

FLTI1 = fraction of iodine in the suppression pool that is volatilized and released after vessel breach,

FLTI2 = fraction of iodine in the cavity water that is volatilized and released after vessel breach,

HVSPLT = fraction of in-vessel release that passes through the reactor head vent,

POOLI = fraction of initial core inventory for iodine scrubbed by the pool,

$RBDF_i$ = decontamination factor for aerosol species i flowing from the containment to the auxiliary building,

ST_i = fraction of the initial core inventory of species i that is ultimately released to the environment,

$VBPUF_i$ = fraction of initial core inventory of species i that is released to the drywell as puff at the time of vessel breach,

XCCI = fraction of core material that leaves the vessel and participates in CCI.

In addition to the parameters that appear in Equation 7.1, there are a series of parameters that are used to define the timing of the release and other characteristics of the release that are important for the determination of the consequences. These additional parameters include:

TW = warning time which is taken to correspond to when a general emergency is declared

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and is when evacuation procedures are initiated (evacuation starts a short time later, however, due to the time it takes to implement evacuation),

T1	=	start of release segment 1,
D1	=	duration of release segment 1,
E1	=	energy release rate associated with release segment 1,
T2	=	start of release segment 2,
D2	=	duration of release segment 2,
E2	=	energy release rate associated with release segment 2.

A detailed discussion of this equation is presented in NUREG/CR-5360 [Jow, et al., 1993]. The FORTRAN listing of GGSOR-P5 is contained in Appendix C.

7.3 Quantification of Source Term Model

The parameters in Equation 7.1 were quantified using the same data that was used to quantify the NUREG-1150 version of GGSOR. A major difference between the full power accidents analyzed in NUREG-1150 and the shutdown accidents analyzed in this study is the amount of decay heat that is present during the accident. While the decay heat load will have a major impact on the timing of the release and the energy of the release, it was judged that, given the large uncertainties associated with parameters quantified in NUREG-1150, the use of these full power parameters for the c shutdown accidents would be acceptable[3]. This judgement was based on comparisons between MELCOR calculations for shutdown accidents with NUREG-1150 source term data and on discussions with the Source Term Advisor Group that was convened to review source information associated with off power accidents. Many of the source term parameters depended on conditions in the containment and/or core and were not necessarily tied to specific accident sequences (e.g., FCOR depends on the amount of zirconium oxidized in the core). In these cases, shutdown accidents with similar containment/core conditions were associated with the relevant parameter values. In other cases, however, the parameter values used in NUREG-1150 were tied to a specific accident sequence (e.g., short-term station blackout). In these cases, the rationale behind the quantification of the parameter was reviewed and the shutdown accident sequence that most closely match the relevant attributes of the full power accident were associated with that parameter value. When new parameters were added for this analysis, data was obtained from relevant NUREG-1150 cases or from relevant MELCOR calculations that were performed for this analysis. Since the source term parameters that are used to define the timing and energy of the release are accident specific, these parameters were quantified using information from MELCOR calculations performed for this study (the MELCOR calculations are documented in Volume 6, Part 2 of this report). The values assigned to the various parameters are listed in Appendix C.

Two of the parameters that appear in Equation 7.1 were not used in the NUREG-1150 Grand Gulf analysis: the fraction of radionuclides that passes through the reactor head vent (HVSPLT) and the decontamination factor for the reactor building (RBDF). During full power operation, the reactor vessel head vent is closed and, therefore, HVSPLT is not an issue. During shutdown, the reactor head vent is open at the start of the accident allowing for the possibility that some

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3 Here we are discussing the parameters that are used to determine the fraction of the inventory at the start of the accident that is released to the environment. By using the same parameters it is implied that for a similar accident, the release fractions for a full power accident and a shutdown accident will be the same. While the release fractions may be the same, the amount of radioactive material release to the environment will not be the same because of the differences in the radioactive inventories at the start of the accident.

of the in-vessel release will leave the vessel via the head vent. The value used for the fraction of the in-vessel release that passes through the head vent, HVSPLT, was obtained from MELCOR calculations performed for this study. In the NUREG-1150 Grand Gulf analysis, the containment never failed into the auxiliary building and, therefore, a decontamination factor (DF) for the auxiliary building (RBDF) was not required. In this analysis, however, the containment equipment hatch is often open allowing radioactive material in the containment atmosphere to pass through the auxiliary building before escaping to the environment. Since a DF for the Grand Gulf auxiliary building was not available from NUREG-1150, it was decided that the DF developed in NUREG-1150 for the Peach Bottom reactor building would be an acceptable surrogate for the Grand Gulf DF. For these shutdown accidents, the absence of large driving forces (i.e., containment blowdown into the auxiliary building) result in fairly low flow rates in the auxiliary building. Also, due to the low flow rates and the relatively cool temperatures in the auxiliary building, there is a considerable amount of condensation in the auxiliary building. Based on these conditions, Peach Bottom Reactor Building DF Case 4 (Drywell shell melts though to the reactor building; the suppression pool is saturated) was used to quantify the Grand Gulf auxiliary building DF parameter. An assessment of the impact that the standby gas treatment system or the auxiliary building ventilation system would have on the source term was beyond the scope of this analysis and, hence, these systems were not modelled in this analysis.

The sources of information that were used to quantify the parameters in Equation 7.1 are listed in Table 7-2; the parameters that were included in the uncertainty analysis are also identified in this table. Sixteen of the parameters in the Equation 7.1 were sampled in the uncertainty analysis; distributions for these parameters were based on information from the NUREG-1150 study. Ten of these sixteen parameters were generated by the NUREG-1150 Source Term Expert Panel. For each parameter that was assessed by the Source Term Expert Panel, the distribution for the parameter, the reasoning that led each expert to his conclusions, and the aggregation of the individual distributions are fully described in NUREG/CR-4551, Volume 2, Part 4 [Harper, et al., 1992]. The remaining parameters included in the uncertainty analysis were quantified by the NUREG-1150 project staff and are discussed in NUREG-5360 [Jow, et al., 1993].

Unless specifically identified, there is no correlation between any of the source term variables, but complete correlation within a variable. FCOR is not correlated with FVES, FCONV, or any other variable, but the values for the different cases for a given parameter and for the different radionuclide classes are completely correlated. That is, if the 0.05 quantile value is chosen for iodine for low zirconium oxidation, the 0.05 quantile value is also chosen for all the other radionuclide classes and for all values for high zirconium oxidation.

7.4 Partitioning of Source Terms

As discussed above, a source term is estimated for each APB. Furthermore, since the uncertainty analysis includes variables from the source term analysis, there will be many source term estimates for a single APB. In this analysis, approximately 35,000 source terms were generated. Because of the large computational requirements of the consequence analysis, consequences cannot be estimated for each individual source term. Instead, the individual source terms that are expected to result in similar early and chronic health effects are combined into source term groups (STGs); the PARTITION program [Brown, et al., 1992] is used to form the STGs. The source term for each STG is a frequency weigthed average of the source terms for each of the APBs that are contained in the STG. Since a different radionuclide inventory will be used for each time window in the consequence calculations, the source terms from each time window were partitioned separately to ensure that each STG only contained source terms from a single time window.

7.5 Results

Since the source term is a collection of 28 parameters that characterize the radioactive release to the environment and the health effects that result from the release are a complicated function of these parameters and additional parameters that affect the transport and deposition of the material in the environment, it is not convenient or particularly useful to infer the impact of the accident based on the source term--particularly when a consequence analysis, which will take these factors into account, is being performed. Furthermore, since source terms were generated for several hundred APB, it is not practical to present the source term for individual APBs. Instead, to document the product of the source term analysis, the source terms for the STGs are presented. The partitioning process resulted in the generation of 54 STGs.

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The source term associated with each STG is provided in Appendix C. Insights regarding the affect that different accident characteristics have on risk will be deferred to the insights and conclusion section of this report.

7.6 References

[Gorham, et al., 1993]	E. D. Gorham et al., "Evaluation of Severe Accident Risks: Methodology for the Containment Source Term, Consequence, and Risk Integration Analysis," NUREG/CR-4551, SAND86-1309, Vol. 1, Rev. 1, Sandia National Laboratories, December 1993.
[Brown, et al., 1990]	T. D. Brown et al., "Evaluation of Severe Accident Risks: Grand Gulf Unit 1," NUREG/CR-4551, SAND86-1309, Vol. 6, Rev. 1, Sandia National Laboratories, December 1990.
[Jow, et al., 1993]	H.-N. Jow et al., "XSOR Code User's Manual," NUREG/CR-5360, SAND89-0943, Sandia National Laboratories, November 1993.
[Cybulskis, et al., 1989]	P. Cybulskis et al., "Assessment of the XSOR Codes," NUREG/CR-5346, BMI-2171, Battelle Columbus Division, November 1989.
[Harper, et al., 1992]	F. T. Harper, "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters: Experts' Determination of Source Term Issues," NUREG/CR-4551, Vol. 2, Rev. 1, Part 4, Sandia National Laboratories, June 1992.
[Brown, et al., 1992]	T. D. Brown et al., "Integrated Risk Assessment for the LaSalle Unit 2 Nuclear Power Plant," NUREG/CR-5305, SAND90-2765, Vols. 1-2, Sandia National Laboratories, August 1992.

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Table 7-2
Sources of Information Used to Quantify the Source Term Parameters

Source Term Parameter	Included in Uncertainty Analysis	Quantification Source
CAVWI	No	Determined by various combination of other parameters
DFSPRC	Yes	NUREG-1150 Project Staff (DFSPRC and DFSPRV are completely correlated, i.e., same LHS variable used for both distributions)
DFSPRV	Yes	NUREG-1150 Project Staff (DFSPRC and DFSPRV are completely correlated, i.e., same LHS variable used for both distributions)
DFCAV	Yes	NUREG-1150 Project Staff
DFCPA	Yes	NUREG-1150 Project Staff (DFCPA and DFVPA are completely correlated, i.e., same LHS variable used for both distributions)
DFVPA	Yes	NUREG-1150 Project Staff (DFCPA and DFVPA are completely correlated, i.e., same LHS variable used for both distributions)
FCCI	Yes	NUREG-1150 Source Term Expert Panel Case 1: Coolable Debris Bed (FCCI set to 0.0) Case 2: Dry cavity with high level of zirconium oxidized in the core Case 3: Dry cavity with low level of zirconium oxidized in the core Case 4: Flooded cavity w/ high level of zirconium oxidized in the core Case 5: Flooded cavity with low level of zirconium oxidized in the core
FCONC	Yes	NUREG-1150 Source Term Expert Panel Case 1: No containment failure Case 2: Early containment leak, subcooled suppression pool Case 3: Early containment leak, saturated suppression pool Case 4: Early containment rupture (includes case where containment equipment hatch is open), subcooled suppression pool Case 5: Early containment rupture (includes case where containment equipment hatch is open), subcooled suppression pool Case 6: Late containment leak Case 7: Late containment rupture
FCONV	Yes	NUREG-1150 Source Term Expert Panel Same cases as FCONC with the following consideration: Case 8: LOCA in auxiliary building or open MSIVs, FCONV = 1.0
FCOR	Yes	NUREG-1150 Source Term Expert Panel Case 1: High level of zirconium oxidation in the core Case 2: Low level of zirconium oxidation in the core
FDCH	Yes	NUREG-1150 Source Term Expert Panel
EVSE	No	NUREG-1150 Project Staff Case 1: Large amount of core debris participates in steam explosion Case 2: Small amount of core debris participates in steam explosion
FEVSE	Yes	NUREG-1150 Project Staff

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Table 7-2 (continued)

Source Term Parameter	Included in Uncertainty Analysis	Quantification Source
FHPE	No	NUREG-1150 Project Staff Case 1: Large amount of core debris participates in HPME Case 2: Small amount of core debris participates in HPME
FREVO	Yes	NUREG-1150 Source Term Expert Panel Case 1: No vessel failure (FREVO=0.0) Case 2: Injection available after vessel failure Case 3: No injection available after vessel failure
FPLBYC	No	Set to 1.0; drywell personnel lock and/or equipment hatch open
FPLBYD	No	Set to 1.0; drywell personnel lock and/or equipment hatch open
FPLBYE	No	NUREG-1150 Project Staff
FPLBYP	No	Set to 1.0; drywell personnel lock and/or equipment hatch open
FTLP	No	NUREG-1150 Project Staff Case 1: Reactor vessel is at low pressure during core damage Case 2: Reactor vessel is at high pressure during core damage
FVES	Yes	NUREG-1150 Source Term Expert Panel Case 1: Reactor vessel at high pressure during core damage Case 2: Reactor vessel at low pressure during core damage
FLTI1	Yes	NUREG-1150 Source Term Expert Panel (FLTI1 and FLTI2 are completely correlated, i.e., same LHS variable used for both distributions) Case 1: Subcooled suppression pool Case 2: Saturated suppression pool
FLTI2	Yes	NUREG-1150 Source Term Expert Panel (FLTI1 and FLTI2 are completely correlated, i.e., same LHS variable used for both distributions) Case 1: Dry cavity Case 2: Flooded cavity Case 3: No core-concrete interactions in cavity
HVSPLT	No	MELCOR calculations for POS 5 Case 1: Head vent open Case 2: Head vent closed (HVSPLT = 0.0)
POOL1	No	Determined by various combination of other parameters
RBDF	Yes	NUREG-1150 Source Term Expert Panel (Distribution from NUREG-1150 Peach Bottom Analysis: drywell shell melt-through with saturated suppression pool)
VBPUF	Yes	NUREG-1150 Project Staff
XCCI	No	Determined by various combination of other parameters

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Table 7-2 (continued)

Source Term Parameter	Included in Uncertainty Analysis	Quantification Source
TW	No	MELCOR calculations for POS 5 Case 1: PDS1-1 Case 2: PDS1-2 and PDS1-4 Case 3: PDS1-3 Case 4: PDS1-5 Case 5: PDS2-1 Case 6: PDS2-2 and PDS2-3 Case 7: PDS2-4 Case 8: PDS2-5: Containment equipment hatch is open Case 9: PDS2-5: Containment fails during core damage Case 10: PDS2-5: Containment fails at vessel breach Case 11: PDS2-5: Containment vented or fails late in the accident Case 12: PDS2-6 Case 13: PDS3-1
T1	No	MELCOR calculations for POS 5 Same cases as TW except that Cases 1, 4, 5, 7, and 13 are divided to distinguish between cases where the auxiliary building fails during core damage from those that fail at the time of vessel failure
D1	No	MELCOR calculations for POS 5 Case 1: Containment rupture or venting Case 2: Auxiliary building failure at the time of vessel failure Case 3: Short duration Case 4: Medium duration Case 5: Long duration
E1	No	MELCOR calculations for POS 5 Case 1: Flooded CNMT & Aux. Bldg failure at VB (no H_2 burn) Case 2: Aux. Bldg fails during CD (H_2 burn) or CNMT fails via rupture Case 3: SBO and the Aux. Bldg fails at beginning of CD, or CNMT fails via a leak, or CNMT open before core damage Case 4: PDS2-6 (Aux. Bldg fails prior to core damage)
T2	No	Combination of previously defined parameters (T2=T1 + D1)
D2	No	MELCOR calculations for POS 5 Case 1: No vessel breach Case 2: All other cases
E2	No	MELCOR calculations for POS 5 Case 1: No vessel breach Case 2: CNMT not flooded Case 3: Flooded CNMT

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8 Consequence Analysis

As is typically done in PRAs of nuclear power plants, the consequences to the general public that result from a release of radioactive material were estimated. This study is unique in that onsite doses were also estimated, something that is not typically estimated in full power PRAs of nuclear power plants. Another important difference between this analysis and those previously performed for full power accidents is that the radionuclides in the fuel have had, in some cases, a significant amount of time to decay resulting in a different inventory than that present at shutdown. ORIGEN2 [Croff, et al., 1989] was used to calculate the inventories associated with the shutdown accidents analyzed in this study; a unique inventory was defined for each of the three time windows. The resulting inventories were reduced to include only the sixty radionuclides currently available in the MACCS code [Chanin, et al., 1990],[Jow, et al., 1990],[Rollstin, et al., 1990]. It is these sixty radionuclide inventories that were then used as the basis for both the onsite and offsite consequence calculations. These inventories, which do not include short-lived radionuclides, are appropriate for both the onsite and offsite analyses since the reactor has been in shutdown for at least seven hours at the beginning of the accident thus allowing decay of the short-lived radionuclides. The radionuclide inventories are provided in Appendix D.1. The following subsections provide an overview of the methodology and list the pertinent results for both the onsite and offsite consequences.

8.1 Onsite Consequences

Onsite consequences have seldom been considered in the analysis of severe accidents at nuclear power plants. During shutdown there will be hundreds of onsite personnel and, thus, onsite consequences could be large. For this reason a method for estimating the potential doses to onsite personnel had to be developed as part of this study. In this onsite consequence assessment, only the doses and dose rates are estimated for the surrounding region near the plant, referred to as the parking lot region; healths effects to the onsite population are not estimated in this study. Since many simplifying assumptions are used in this assessment, the calculations should be viewed as scoping in nature. The intent of these calculations is to provide some insight into the potential magnitude of the onsite doses and dose rates. The method and results are discussed in the following two sections.

8.1.1 Method

Doses and dose rates were estimated for a range of distances from the reactor. For comparative purposes, two different wake effect models were used to estimate the relative concentrations downwind of the reactor. The first model was developed by Ramsdell [Ramsdell, 1990]. The second model actually consists of two models: a model developed by Wilson [Wilson, 1984] which was used to estimate doses within 100 meters of the plant and a model used by the NRC [USNRC, 1982] which was used to estimate doses beyond 100 meters. The Ramsdell model was developed by using multiple linear regression to fit experimental results to a statistical model that included the following four variables: (1) wind speed, (2) distance, (3) building area and (4) stability. The result was that the exponent on each of these variables was determined by the experimental results. The Wilson and NRC models are based on Gaussian plume theory. The NRC model allows for plume meander during low-speed, stable atmospheric conditions for distances of less than 800 m. The Wilson models uses experimental data to fit the Gaussian model to the relative concentration for distances very close to the release point. An interesting difference between the two sets of models is that in the case of the Ramsdell model the relative concentration is somewhat proportional to the wind speed and the stability class whereas in the case of the Wilson and NRC models the relative concentration is predicted to be inversely proportional to the wind speed. Using the integrated air concentrations for each building wake effect model, the dose and dose rate were estimated for each partitioned source term group (see Section 7). The major simplifying assumptions used in this analysis were that radioactive decay was neglected during the exposure time, the directional dependence of the weather was ignored and a single radioactive release location and building area was assumed. Two different weather scenarios were used for each set of models. The first weather scenario assumes stable conditions (stability class F) with a wind speed of 1 meter/second. The second weather scenario assumes unstable conditions (stability class A) with a wind speed of 5 meters/second. The dose calculation considered exposure from both the immersion and inhalation pathways and is a 50 year committed dose. The dose rate only considered exposure from the immersion pathway. Doses were calculated assuming exposure to the entire release and also exposure to only the first 15 minutes of the release. A dose rate was calculated for each of the release segments defined in the source term analysis. The code that implements these models is listed in Appendix D.2.

8.1.2 Results

Table 8.2-1 contains the mean total dose, 15 minute dose, and dose rates for each segment of the release based on the Wilson/Reg. Guide building wake effect model for several distances from the containment and for the two weather conditions. Similar estimates of the mean doses and dose rates based on the Ramsdell model are shown in Table 8.2-2. Results from these calculations for each of the source term group are provided in Appendix D.3. Since the relative air concentration predicted by the Wilson/Reg. Guide model is inversely proportional to the wind speed, the dose and dose rates that correspond to a wind speed of 1 m/s are larger than the doses that correspond to a wind speed of 5 m/s. The opposite relationship holds for the Ramsdell model. For the weather condition with F stability and a wind speed of 1 m/s, the doses predicted by the Wilson/Reg. Guide model are considerably greater than the corresponding doses predicted by the Ramsdell model. For the weather condition with A stability and a wind speed of the 5 m/s, the doses predicted by the Wilson/Reg. Guide model are greater than the corresponding doses predicted by the Ramsdell model when the receptor is within 100 meters of the plant. Beyond 100 meters, the doses predicted by the Ramsdell model are the greatest. It should be pointed out that the dose rates reflect the duration of the release. The release durations for the accidents delineated in this study were considerable (i.e., many hours to 10s of hours). Thus, while the dose rates shown in Table 8.1-1 and 8.1-2 are considerable, they would have been even higher had the release occurred over a shorter time period.

As a comparison to the doses reported in Tables 8.1-1 and 8.1-2, the lethal dose in 50% of the population (LD50) is approximately 400 rem [Evans, et al., 1986]. This highlights the fact that although only a simplified scoping study of onsite consequences has been performed, the possible onsite consequences of an accident during shutdown could be significant.

Table 8.1-1
Mean Parking Lot Doses and Dose Rates Predicted by Wilson Model

Distance (m)	A Stability, Wind Speed = 5m/s				F Stability, Wind Speed = 1m/s			
	Dose Rate (rem/hr)		Dose (rem)		Dose Rate (rem/hr)		Dose (rem)	
	First Release	Second Release	15 min Exposure	Total	First Release	Second Release	15 min Exposure	Total
10	44,640	17,064	546,000	40,600,000	223,560	85,320	2,730,000	203,000,000
50	1,789	684	21,800	1,620,000	8,928	3,416	109,000	8,120,000
100	75	29	918	68,300	2,646	1,012	32,300	2,400,000
250	5	2	57	4,260	666	255	8,150	605,000
500	1	0.2	7	522	235	90	2,870	213,000

Table 8.1-2
Mean Parking Lot Doses and Dose Rates Predicted by Ramsdell Model

Distance (m)	A Stability, Wind Speed = 5m/s				F Stability, Wind Speed = 1m/s			
	Dose Rate (rem/hr)		Dose (rem)		Dose Rate (rem/hr)		Dose (rem)	
	First Release	Second Release	15 min Exposure	Total	First Release	Second Release	15 min Exposure	Total
10	461	176	5,630	418,000	338	129	4,130	307,000
50	75	29	915	67,900	55	21	669	49,700
100	34	13	418	31,000	25	10	306	22,700
250	12	5	148	11,000	9	3	109	8,060
500	6	2	68	5,030	4	2	50	3,690

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8.2 Offsite Consequences

Offsite consequences were calculated with version 1.5.11.1 of the MACCS code [Chanin, et al., 1993] for each of the source term groups defined in the partitioning process. This code has been in use for some time and will not be described in any detail. Although the variables thought to be the largest contributors to the uncertainty in risk were sampled from distributions in the accident frequency analysis, the accident progression analysis, and the source term analysis, there was no analogous treatment of uncertainties in the consequence analysis. Variability in the weather was fully accounted for, but the uncertainty in other parameters such as the dry deposition speed or the evacuation rate was not considered.

8.2.1 Description of the Offsite Consequence Analysis

MACCS tracks the dispersion of the radioactive material in the atmosphere from the power plant and computes its deposition on the ground. MACCS then calculates the effects of this radioactivity on the population and the environment. Doses and the ensuing health effects from 60 radionuclides are computed for the following pathways:

Table 8.2-1
Offsite Consequence Measures Calculated for POS 5

Consequence Measure	Description
Early fatalities	Number of fatalities occurring within 1 year of the accident due to early exposure (i.e., exposure incurred within seven days of the accident).
Total latent cancer fatalities	Number of latent cancer fatalities due to both early and chronic exposure (i.e., chronic exposure is that incurred more than seven days after the accident).
Population dose within 50 miles	Population dose, expressed in effective dose equivalents for whole body exposure (rem), due to early and chronic exposure pathways within 50 miles of the reactor. Due to the nature of the chronic pathways models, the actual exposure due to food and water consumption may take place beyond 50 miles (e.g., food and water originating within 50 miles of the plant may be consumed by people located beyond 50 miles).
Population dose within entire region	Population dose, expressed in effective dose equivalents for whole body exposure (rem), due to early and chronic exposure pathways within the surrounding region.
Individual early fatality risk within one mile	Probability of dying within one year for an individual within one mile of the site exclusion boundary (i.e., ef/pop, where ef is the number of early fatalities within one mile of the exclusion boundary, and pop is the population within one mile of the exclusion boundary).
Individual latent cancer fatality risk within 10 miles	Probability of dying from cancer for an individual within ten miles of the plant (i.e., cf/pop, where cf is the number of cancer fatalities due to direct exposure in the resident population within ten miles of the plant, and pop is the population size within ten miles of the plant). The calculation does not include ingestion but does include integrated groundshine and inhalation exposure.

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immersion or cloudshine, inhalation from the plume, groundshine, deposition on the skin, inhalation of resuspended ground contamination, ingestion of contaminated water and ingestion of contaminated food.

MACCS treats atmospheric dispersion by the use of multiple, straight-line Gaussian plumes. Each plume can have a different direction, duration, and initial radionuclide concentration. Cross-wind dispersion is treated by a multi-step function. Both dry and wet deposition are treated as independent processes. The weather variability is treated by means of a stratified sampling process.

For early exposure, the following pathways are considered: immersion or cloudshine, inhalation from the plume, groundshine, deposition on the skin, and inhalation of resuspended ground contamination. Skin deposition and inhalation of resuspended ground contamination have generally not been considered in previous consequence models. For the long-term exposure, MACCS considers the following four pathways: groundshine, inhalation of resuspended ground contamination, ingestion of contaminated water, and ingestion of contaminated food. The direct exposure pathways, groundshine and inhalation of resuspended ground contamination, produce doses in the population living in the area surrounding the plant. The indirect exposure pathways, ingestion of contaminated water and food, produce doses in those who ingest food or water emanating from the area around the accident site. The contamination of water bodies is estimated for the washoff of material deposited on the land as well as direct deposition. The food pathway model includes direct deposition onto crops and uptake from the soil.

Both short-term and long-term mitigative measures are modeled in MACCS. Short-term actions include evacuation, sheltering and emergency relocation out of the emergency planning zone. Long-term actions include later relocation and restrictions on land use and crop disposition. Relocation and land decontamination, interdiction, and condemnation are based on projected long-term doses from groundshine and inhalation of resuspended radioactivity. The disposal of agricultural products is based on the products' contamination levels and the removal of farmland from crop production is based on ground contamination criteria. The health effects models link the dose received by an organ to predicted morbidity or mortality. The models used in MACCS calculate both short-term and long- term effects for a number of organs.

The MACCS consequence model calculates a large number of different consequence measures. Results for the following six consequence measures are given in this report: early fatalities, total latent cancer fatalities, population dose within 50 miles, population dose for the entire region, early fatality risk within 1 mile, and latent cancer fatality risk within 10 miles. These consequence measures are described in Table 8.2-1. For the analyses performed for NUREG-1150, 99.5% of the population is assumed to evacuate, and the remaining 0.5% of the population does not evacuate and continues normal activity. This same assumption is used in this analysis.

8.2.2 MACCS Input for Grand Gulf

The input used in this study is identical to that used for Grand Gulf in the NUREG-1150 study with the exception of the core inventories (see Appendix D.1) and the source terms which resulted from GGSORP5. The emergency response assumptions were not changed for this analysis. Since the methods used to calculate the MACCS parameters and the parameter values developed using those methods are documented Volume 2, Part 7 of NUREG/CR-4451 [Sprung, et al., 1990], only a small portion of the MACCS input is presented here.

Table 8.2.2 lists the MACCS input parameters that have strong site dependencies and presents the values of these parameters used in the MACCS calculations for the Grand Gulf site. The evac ion delay period begins when general emergency conditions occur and ends when the general public starts to evacuate. . on-farm wealth includes personal, business, and public property, the farmland fractions do not add to one because not all farmland is under cultivation. In addition to the site-specific data presented in Table 8.2-2, the Grand Gulf MACCS calculations used one year of

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meteorological data from the Grand Gulf site and regional population data developed from the 1980 census tapes[1] Table 8.2-3 gives the population within certain distances of the plant as summarized from the MACCS demographic input Table 8.2-4 lists the shielding parameters used in this analysis

Table 8.2-2
Site Specific Input Data for Grand Gulf MACCS Calculations

Parameter	Value
Reactor Power Level (MWt)	3833
Containment Height (m)	32
Containment Width (m)	32
Exclusion Zone Distance (m)	696
Evacuation Delay (h)	1.25
Evacuation Speed (m/s)	3.7
Farmland Fractions by Crop Categories	
Pasture	0.7
Stored Forage	0.05
Grains	0.18
Green Leafy Vegetables	0.0005
Legumes and Seeds	0.13
Roots and Tubers	0.0008
Other Food Crops	0.004
Non-Farm Wealth ($/person)	53,000
Farm Wealth	
Value ($/hectare)	1824
Fraction in Improvements	0.30

[1] Since the NUREG-1150 full power analysis of Grand Gulf was based on the 1980 census data and an objective of this study is to compare the risk from full power with the risk from shutdown, this data was also used in this study instead of using the more recent 1990 census data

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Table 8.2-3
Population Surrounding Plant

Distance from Plant (km)	(miles)	Population
1.6	1.0	34
4.8	3.0	879
16.1	10.0	10,255
48.3	30.0	97,395
160.9	100.0	1,614,883
563.3	350.0	22,259,422
1609.3	1000.0	142,024,448

Table 8.2-4
Shielding Factors used for Grand Gulf MACCS Calculations

Radiation Pathway	Population Response		
	Evacuate	Normal Activity	Take Shelter
Cloudshine	1.0	0.75	0.70
Groundshine	0.5	0.33	0.25
Inhalation	1.0	0.41	0.33
Skin	1.0	0.41	0.33

8.2.3 Results from the Offsite Consequence Analysis

The mean (over weather variation) consequences for the source term groups are reported in Table 8.2-5. The 55th Source Term Group in Table 8.2-5 is in case any source terms fall into the special case of no release; the consequences for this case are 0.0. The remaining results given in this table are conditional on the occurrence of a release. That is, given that a release takes place, with release fractions and other characteristics as defined by one of the source term groups, then the consequences reported in this section are calculated. Table 8.2-5 contains no information about the frequency with which these consequences may be expected. Information about the frequencies of consequences of various magnitudes is contained in the risk results (Section 9). An early fatality consequence value less than 1.0 may be interpreted as the probability of obtaining one death. The population dose is the effective dose equivalent to the whole body for the population in the region indicated.

8.3 References

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Table 8.2-5
Mean Offsite Consequence Results

Source Term Group	Early Fatalities	Total Latent Cancers	Population Dose (rem) <50 miles	Population Dose (rem) <1000 miles	Individual Early Fatality Risk	Individual Total Latent Cancer Risk
GG-001	1.24E-02	3.11E+03	7.38E+05	7.26E+06	1.08E-04	7.20E-04
GG-002	9.65E-03	3.39E+03	7.98E+05	7.83E+06	9.30E-05	8.43E-04
GG-003	1.20E-02	3.57E+03	8.54E+05	8.28E+06	1.06E-04	6.72E-04
GG-004	1.08E-01	5.83E+03	1.30E+06	1.27E+07	4.79E-04	1.84E-03
GG-005	1.77E-02	4.78E+03	9.83E+05	1.11E+07	1.41E-04	9.00E-04
GG-006	1.79E-03	2.06E+03	5.71E+05	4.79E+06	2.24E-05	9.01E-04
GG-007	1.35E-03	1.85E+03	5.48E+05	4.43E+06	1.70E-05	7.22E-04
GG-008	3.14E-03	2.73E+03	6.55E+05	6.30E+06	3.80E-05	1.01E-03
GG-009	3.31E-02	4.38E+03	9.76E+05	1.00E+07	1.99E-04	8.46E-04
GG-010	2.41E-03	1.69E+03	4.66E+05	3.90E+06	2.96E-05	6.61E-04
GG-011	4.38E-03	1.73E+03	4.13E+05	3.90E+06	5.05E-05	8.26E-04
GG-012	2.98E-02	5.12E+03	1.21E+06	1.20E+07	1.92E-04	7.70E-04
GG-013	2.20E-02	4.16E+03	9.69E+05	9.39E+06	1.62E-04	9.47E-04
GG-014	8.20E-02	5.37E+03	1.12E+06	1.10E+07	3.14E-04	1.11E-03
GG-015	1.56E-03	1.54E+03	4.13E+05	3.56E+06	1.95E-05	7.69E-04
GG-016	2.33E-03	1.59E+03	4.34E+05	3.82E+06	2.87E-05	7.57E-04
GG-017	3.94E-02	6.10E+03	1.55E+06	1.41E+07	2.60E-04	7.64E-04
GG-018	9.40E-04	9.02E+02	2.98E+05	2.13E+06	1.18E-05	5.81E-04
GG-019	7.25E-03	1.78E+03	4.78E+05	4.06E+06	7.30E-05	5.95E-04
GG-020	6.65E-03	2.87E+03	6.15E+05	6.59E+06	6.75E-05	6.12E-04
GG-021	1.05E-05	4.41E+02	1.74E+05	1.01E+06	1.33E-07	4.63E-04
GG-022	1.64E-07	8.22E+02	2.80E+05	1.86E+06	2.08E-09	7.37E-04
GG-023	0.00E+00	4.99E+01	2.88E+04	1.19E+05	0.00E+00	1.26E-04
GG-024	7.35E-03	1.35E+03	4.70E+05	3.30E+06	7.10E-05	4.33E-04
GG-025	3.17E-03	1.72E+03	4.74E+05	4.03E+06	3.72E-05	7.11E-04
GG-026	0.00E+00	6.45E+01	4.19E+04	1.79E+05	0.00E+00	1.66E-04
GG-027	6.00E-03	1.43E+03	4.14E+05	3.23E+06	6.25E-05	6.83E-04
GG-028	4.12E-02	3.94E+03	9.53E+05	8.74E+06	2.25E-04	1.31E-03
GG-029	1.54E-02	3.16E+03	7.78E+05	7.37E+06	1.23E-04	7.88E-04
GG-030	2.76E-02	4.95E+03	9.89E+05	1.13E+07	1.80E-04	1.19E-03
GG-031	1.44E-03	2.00E+03	5.71E+05	4.71E+06	1.81E-05	8.21E-04
GG-032	6.92E-02	6.52E+03	1.61E+06	1.47E+07	3.27E-04	1.49E-03
GG-033	2.54E-03	2.89E+03	6.85E+05	6.67E+06	3.14E-05	9.95E-04
GG-034	1.08E-03	1.70E+03	4.29E+05	3.89E+06	1.36E-05	8.22E-04
GG-035	3.93E-02	4.86E+03	1.06E+06	1.06E+07	2.24E-04	1.02E-03
GG-036	3.88E-03	2.89E+03	6.13E+05	6.64E+06	4.42E-05	7.33E-04
GG-037	8.10E-03	1.97E+03	5.07E+05	4.48E+06	7.85E-05	7.88E-04

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Table 8.2-5 (continued)
Mean Offsite Consequence Results

Source Term Group	Early Fatalities	Total Latent Cancers	Population Dose (rem) <50 miles	Population Dose (rem) <1000 miles	Individual Early Fatality Risk	Individual Total Latent Cancer Risk
GG-038	8.05E-03	3.26E+03	8.21E+05	7.62E+06	7.95E-05	8.60E-04
GG-039	9.35E-04	2.31E+03	5.70E+05	5.28E+06	1.15E-05	6.76E-04
GG-040	8.75E-05	2.79E+03	6.39E+05	6.39E+06	1.11E-06	8.95E-04
GG-041	6.80E-04	1.44E+03	4.32E+05	3.38E+06	8.60E-06	8.85E-04
GG-042	3.72E-07	5.82E+02	2.29E+05	1.39E+06	4.72E-09	4.44E-04
GG-043	2.87E-03	2.16E+03	6.24E+05	5.11E+06	3.50E-05	8.24E-04
GG-044	2.97E-07	9.85E+02	3.20E+05	2.24E+06	3.76E-09	6.65E-04
GG-045	1.58E-03	1.22E+03	4.15E+05	2.93E+06	1.98E-05	6.37E-04
GG-046	0.00E+00	6.72E+01	4.92E+04	1.88E+05	0.00E+00	1.79E-04
GG-047	4.03E-03	1.37E+03	4.84E+05	3.28E+06	4.64E-05	6.17E-04
GG-048	4.96E-04	1.76E+03	4.79E+05	4.07E+06	6.25E-06	8.05E-04
GG-049	1.20E-02	4.06E+03	8.64E+05	8.82E+06	1.08E-04	1.02E-03
GG-050	3.61E-03	3.08E+03	6.47E+05	6.76E+06	4.27E-05	9.24E-04
GG-051	6.35E-04	2.21E+03	5.37E+05	5.04E+06	8.05E-06	8.31E-04
GG-052	0.00E+00	1.64E+03	3.94E+05	3.60E+06	0.00E+00	9.60E-04
GG-053	0.00E+00	4.46E+02	1.74E+05	9.75E+05	0.00E+00	5.62E-04
GG-054	0.00E+00	7.47E+01	5.77E+04	1.70E+05	0.00E+00	2.04E-04
GG-055	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

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9 Risk Integration

As discussed in Section 2, the risk calculation combines the results from the previous four constituent analyses. Results from the accident frequency and accident progression analyses determine the frequency of a release while results from the source term and consequence analyses are used to estimate the magnitude of the release and the resulting consequences to the offsite population. Aggregate risk is the sum, over all accidents, of the accident frequency multiplied by the consequence of the accident. In the following subsections, the aggregate risk results are presented and the contributors to risk are discussed.

9.1 Risk Results

The aggregate risk results for POS 5 are presented in Table 9-1. Table 9-1 provides statistics that describe the risk distributions for each of the consequence measures defined in Table 8.2-1. The descriptive statistics include the following measures: 5th percentile, median value, 95th percentile, mean value, and standard deviation.

Table 9-1
Distributions for Aggregated Risk for POS 5
(All values are per year)
(Population doses are in person-rem)

Consequence Measure	Descriptive Statistics				
	5th PCT	50th PCT	95th PCT	MEAN	STD Dev
Core Damage Frequency	4.1E-07	1.4E-06	5.6E-06	2.1E-06	2.7E-06
Early Fatality Risk	3.7E-11	2.8E-09	3.9E-08	1.4E-08	5.4E-08
Total Latent Cancer Risk	4.3E-04	1.9E-03	1.2E-02	3.8E-03	7.7E-03
Population Dose within 50 miles of the plant	1.3E-01	5.3E-01	3.1E+00	9.9E-01	1.9E+00
Population Dose within 1000 miles of the plant	9.9E-01	4.4E+00	2.8E+01	8.7E+00	1.8E+01
Individual Early Fatality Risk-- 0 to 1 mile	4.2E-13	2.7E-11	3.0E-10	9.6E-11	3.4E-10
Individual Latent Cancer Risk-- 0 to 10 miles	2.5E-10	9.4E-10	4.9E-09	1.6E-09	2.4E-09

To place the results in context, descriptive statistics for the core damage frequency are also presented in Table 9-1 and the last two consequence measures are compared to the NRC safety goals in Figure 9-1. While the safety goals do not necessarily apply to selected modes of operation (i.e., ideally they should be compared to the plants total risk), a comparison of POS 5 risk to the safety goals does provide an indication, in an absolute sense, of the risk associated with this mode of operation. The NRC established two quantitative safety goals in 1986. The first safety goal [Haskin and Camp] is defined as:

> "The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed." The average accident fatality in the U.S. is approximately 5×10^{-4} per individual per year, so the quantitative value for the first goal is 5×10^{-7} per individual per year. The "vicinity of a nuclear power plant" is defined to be the area within one mile of the plant site boundary.

The second safety goal is defined as:

> "The risk to the population near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one tenth of one percent of the sum of cancer fatality risks resulting from all other causes." The average U.S. cancer fatality rate is approximately 2×10^{-3} per

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year, so the quantitative value for the second goal is 2 x 10^{-6} per average individual per year. The population "near a nuclear power plant" is defined as the population within ten miles of the plant site.

While it is the mean value of the risk distribution that is used to compare to the safety goals, the entire distribution should be considered when making an assessment of the plant's safety. From Figure 9-1 is can be seen that the mean values for the individual risks calculated in this study are well below the safety goals and since the distributions are also well below the safety goals, small changes in these distributions will not affect this conclusion.

There are several factors that contribute to the low offsite risk values at Grand Gulf in spite of the fact that the containment equipment hatch is open in many of the accidents and the probability that the core damage process is arrested in the vessel is small. At this point it is important to remind the reader that aggregate risk is a function of both the frequency of the accident and the consequences that result from the accident and, therefore, both parameters must be considered. The factors that lead to low offsite risk values include the following:

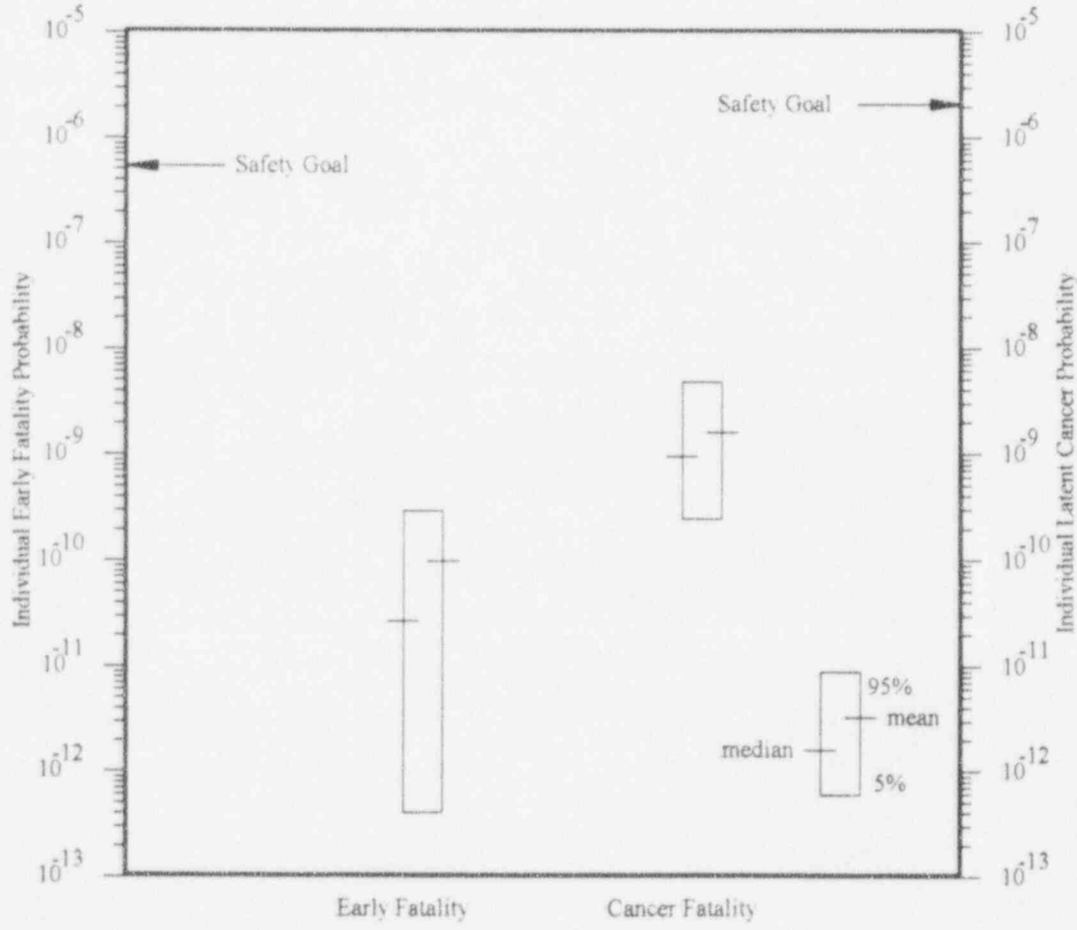

Figure 9-1
Comparison of Individual Risks with Safety Goals

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- The core damage frequency calculated for the Grand Gulf plant is low. Thus, while a significant release may occur, the frequency of the release is sufficiently small that the resulting risk is also small. Aspect of the core damage frequency are discussed in Volume 2 of this report.

- The population density around the Grand Gulf plant is also low. Although many factors influence the magnitude of the consequences, in general, for a given release, the smaller the population, the smaller the number of fatalities. Of the four Mark III plants in the United States, Grand Gulf has the fewest number of people living within 50 miles of the plant according the 1990 census data. The Mark III plant with the greatest number of people living within 50 miles of the site has population that is more than an order of magnitude greater than Grand Gulf 50 mile population. Therefore, for a different plant with similar accidents could have a substantially different risk.

- Although in many of the accidents the containment equipment hatch is open, the suppression pool is bypassed, and the containment sprays are unavailable, the releases pass through the auxiliary building before escaping into the environment. Because of its large volume and surface area, the auxiliary building provides a location for the radionuclides to be attenuated by deposition and thereby reduce the source term to the environment. Without the auxiliary building, considerably more radioactive material would be released to the environment.

- The accidents delineated for these shutdown conditions progress more slowly and, therefore, there is generally more time for the public to respond to the accident and evacuate before they are exposed to the release. This is primarily important for the early health effects consequence measures which are more strongly affected by the time available to evacuate.

- Radioactive decay has reduced the radioactive potential of these shutdown accidents relative to the inventory that is present at shutdown. This factor is primarily important for early health effects which are more strongly affected by the shorter lived radionuclides. This affect is much less noticeable for latent health effects which are more strongly affected by the long lived isotopes.

9.2 Contributors to Risk

This subsection provides the mean fractional contributions to risk, MFCR, for various groups of accidents. MFCR is defined as the ratio of the aggregate risk for a group of accidents (e.g., a PDS) to the total aggregate risk averaged over all observations. That is, the ratio is calculated for each observation and then an average value is determined by summing all of the ratios and then dividing by the number of observations. This measure is not equivalent to the simple ratio of the mean risk for a group of accidents to the total mean risk.

Table 9-2 provides the fractional contributions for the following three PDS groups: LOCAs, Station Blackouts, and Other Transients. From Table 9-2, it can be seen that the SBO PDS group is the dominant contributor to the early fatality risks

Table 9.2
Mean Fractional Contribution to Aggregate Risk for the LOCA, Station Blackout (SBO), and Other Transients Plant Damage State Groups

PDS Groups	Core Damage Frequency	Early Fatalities	Total Latent Cancers	Population Dose (<50 miles)	Population Dose (<1000 miles)	Individual Early Fatalities (0-1miles)	Individual Latent Cancers (0-10 miles)
LOCA	0.51	0.16	0.42	0.43	0.41	0.17	0.51
SBO	0.33	0.73	0.45	0.42	0.45	0.70	0.35
Other	0.17	0.12	0.13	0.15	0.14	0.12	0.14

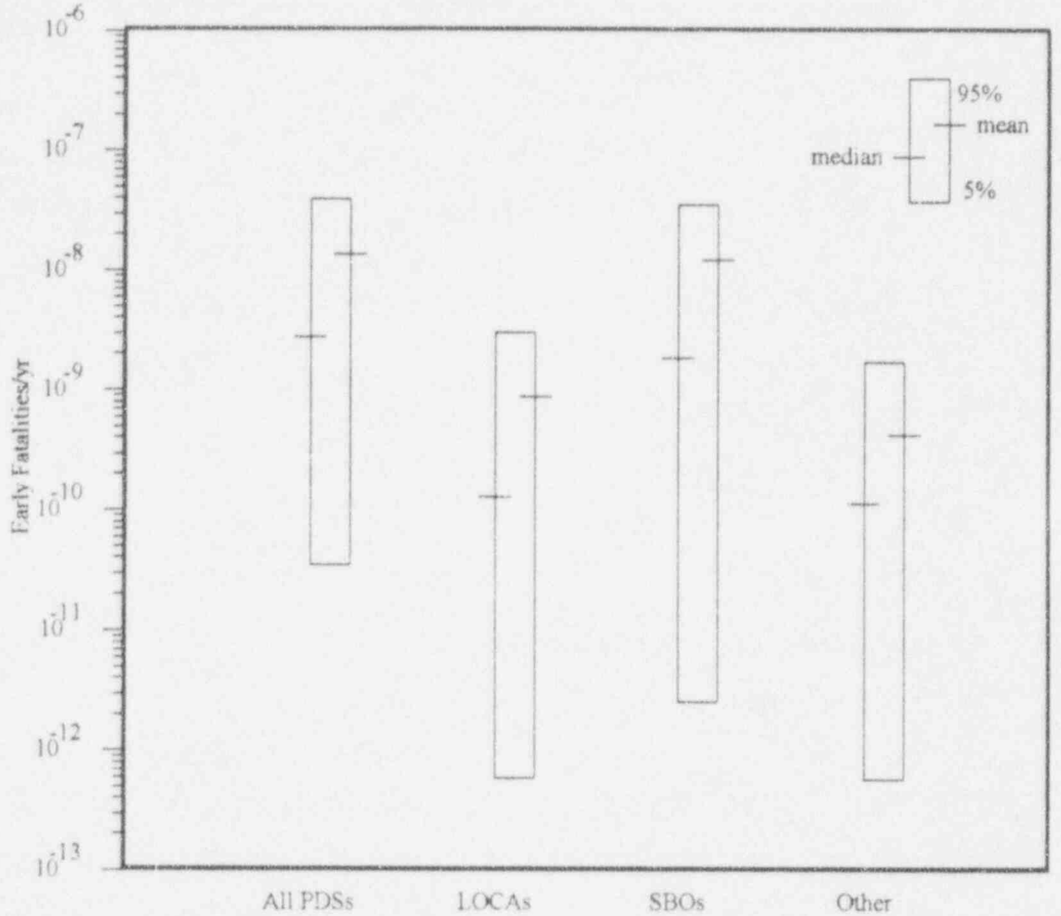

Figure 9-2
Early Fatality Risks for PDS Groups

(total and individual). The early fatality risk distributions for these PDS groups is provided in Figure 9.2. The SBO PDS groups large contribution to early fatality risk can be attributed to their relatively high contribution to the core damage frequency coupled with the fact that the containment equipment hatch is off in these accidents, the suppression pool is bypassed, and the auxiliary building fails early in the accidents. Combined, these factors cause the SBOs to have relatively high risk values. The LOCA PDS group, however, is not a dominant contributor to early fatality risk even though it is a dominant contributor to the core damage frequency. This stems primarily from the fact that the dominant contributor to the LOCA core damage frequency is PDS3-1 which is a negligible contributor to early fatality risk (see Table 9-5). To understand these results it is important to understand that the radionuclides that cause the early health effects tend to have short half lives whereas the radionuclides that cause latent health effects tend to have long half lives. Thus, for similar accidents, one would expect accidents in Time Window 3 to have the fewer early fatalities when compared similar accidents in the other time windows. Similarly, one would not expect there to be a large difference between the time windows for the number of latent cancers that result from an accident. To illustrate this point, Table 9-3 shows the mean number early fatalities and the mean number of total latent cancer fatalities for each of the three LOCA PDSs (note the conditional consequences are a frequency weighted average of all the accidents in the given LOCA PDS). From this table is can be seen that, as expected, there is a significant difference in the number of early fatalities between PDS1-1 and PDS3-1. The number of early fatalities for PDS1-1 and PDS2-1 are similar because the difference in the radionuclide inventories for these to PDS is small. Also, as expected, there is a similar number of latent cancers for each of the PDSs are similar.

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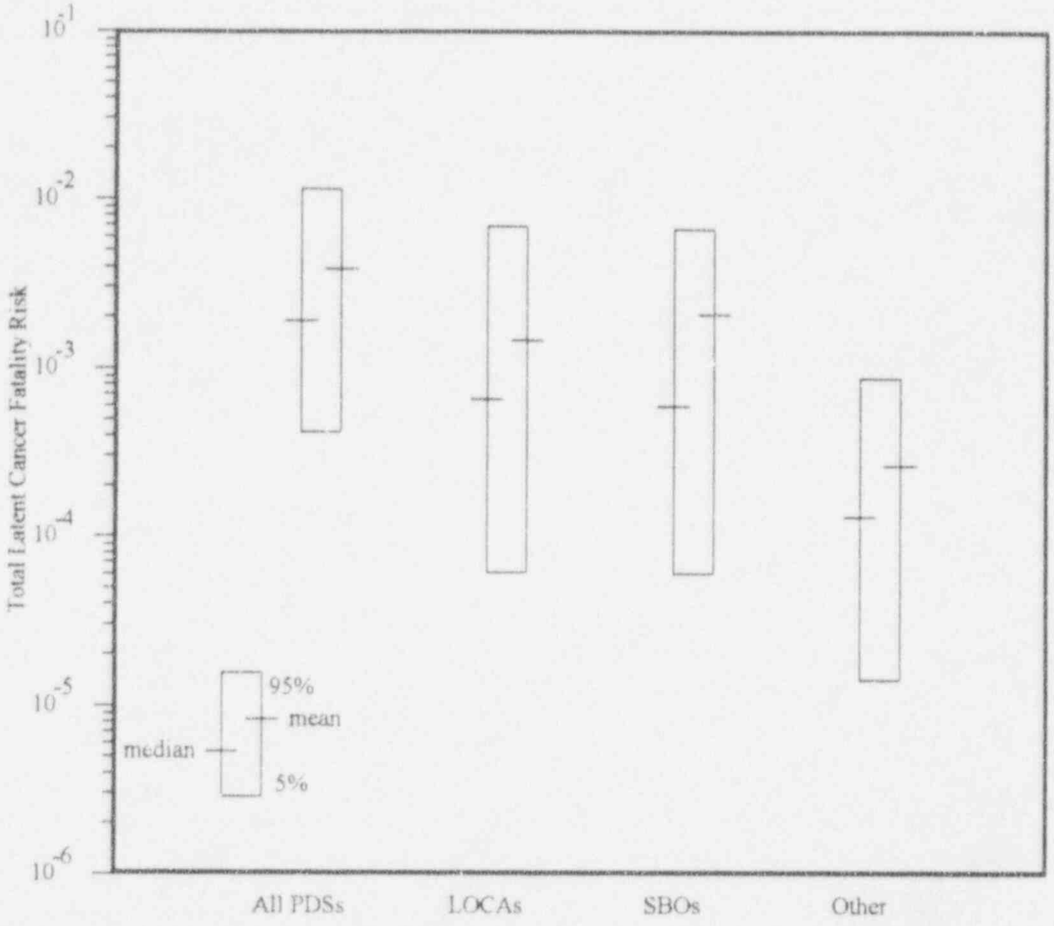

Figure 9-3
Total Latent Cancer Risk for PDS Groups

For latent cancer health effects, the LOCA and SBO PDS groups are the dominant contributors to risk. The latent cancer risk distributions for the three PDS groups are displayed in Figure 9-3. As explained above, since the radionuclides that are important to the latent health effects are the long lived radionuclides, these risk measures are not particularly sensitive to when the accident occurs relative to shutdown. Latent cancers primarily depend on the total amount of radioactive material released and not on when it was released (i.e., early in the accident versus late in the accident). Since latent cancers are not strongly dependent on the timing characteristics of the accident (i.e., start of release or release duration), the latent cancer risk will depend on the likelihood of the accident and on the total amount of radioactive material released. In all of the accidents delineated in this study, the containment is either open at the start of the accident or fails during the accident and in most of the accidents the core damage process is not arrested in the vessel. Thus, while the timing of the accident may vary, when the uncertainty in the source term is considered all of the accidents will result in roughly a similar release in radioactive material to the environment. Thus, as can be seen in Table 9-2, the mean fractional contribution to latent cancer risk tends to be roughly proportional to the contribution to the core damage frequency. The fraction contributions from the LOCA and Other Transients tend to be less than there fractional contribution to the core damage frequency because for these PDSs portions of the release are scrubbed by either the suppression pool or the pool formed by flooding the containment. The fraction contribution from the SBO PDS group tends to be greater than the fractional contribution to the core damage frequency because for these accidents the containment is open at the start of the accident, the auxiliary building fails early in the accident, vessel always fails, CCI always occurs and none of the releases are scrubbed by water. Therefore, the releases associated with the SBO tend

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to be large relative to the other accidents analyzed in this study.

Table 9-3
Mean Conditional Consequences for the LOCA PDSs
PDS1-1, PDS2-1, and PDS3-1

PDS	Early Fatalities	Total Latent Cancers
PDS1-1	3.8E-03	1600
PDS2-1	2.0E-03	1600
PDS3-1	2.2E-04	1100

The mean fractional contributions to aggregate risk for Time Windows 1, 2, and 3 are presented in Table 9-4. From Table 9-4 it can be seen that Time Window 2 is the dominant contributor to early health effects followed by Time Window 1. As explained above, Time Window 3 is a negligible contributor to the early health effects. For latent cancers, Time Window 2 is again the dominant contributor followed this time by Time Window 3 and then Time Window 1. As discussed above, for the latent health effects, the fractional contribution to risk is roughly proportional to the contribution to the core damage frequency.

Table 9-4
Mean Fractional Contribution to Aggregate Risk for Time Windows 1, 2, and 3

PDS Groups	Core Damage Frequency	Early Fatalities	Total Latent Cancers	Population Dose (<50 miles)	Population Dose (<1000 miles)	Individual Early Fatalities (0-1miles)	Individual Latent Cancers (0-10 miles)
TW1	0.08	0.22	0.09	0.09	0.09	0.22	0.07
TW2	0.58	0.78	0.67	0.66	0.67	0.77	0.58
TW3	0.34	0.00	0.24	0.24	0.23	0.01	0.34

For completeness, the mean fractional contributions to risk and to the core damage frequency for each of the 12 PDSs is tabulated in Table 9-5. The summary descriptive statistics for the aggregate risk distributions for each of the PDSs are provided in Appendix E.1.

9.3 References

[Haskin and Camp] F. E. Haskin and A. L. Camp, "Perspectives on Reactor Safety," NUREG/CR-6042, SAND93-0971, Sandia National Laboratories, March 1994.

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Table 9-5
Mean Fractional Contributions of Plant Damage States to Risk

PDS	Core Damage Frequency	Early Fatalities	Total Latent Cancers	Population Dose (<50 miles)	Population Dose (<1000 miles)	Individual Early Fatalities (0-1miles)	Individual Latent Cancers (0-10 miles)
PDS1-1	0.02	0.05	0.02	0.02	0.02	0.05	0.02
PDS1-2	0.01	0.05	0.02	0.02	0.02	0.05	0.02
PDS1-3	0.03	0.09	0.04	0.04	0.04	0.09	0.03
PDS1-4	0.01	0.02	0.01	0.01	0.01	0.02	0.01
PDS1-5	0.01	0.02	0.01	0.01	0.01	0.02	0.01
PDS2-1	0.15	0.11	0.16	0.17	0.16	0.12	0.15
PDS2-2	0.22	0.46	0.31	0.29	0.31	0.44	0.24
PDS2-3	0.06	0.12	0.08	0.07	0.08	0.11	0.06
PDS2-4	0.14	0.06	0.11	0.12	0.11	0.07	0.11
PDS2-5	0.01	0.04	0.01	0.01	0.02	0.03	0.01
PDS2-6	0.01	0.01	0.01	0.01	0.01	0.01	0.01
PDS3-1	0.34	0.00	0.24	0.24	0.23	0.01	0.34

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10 Comparison to Full Power Results

An objective of this analysis is to assess the risk significance POS 5. In the previous section the risk results were presented and the individual risk measures were shown to be well below the safety goals. To further this assessment of the risk significance of POS 5, in this section the POS 5 risk result will be compared to the risk of full power operation. The risk to the general public from the operation of the Grand Gulf nuclear plant during full power operation was analyzed in NUREG-1150 [USNRC, 1990],[Brown, et. al., 1990]. In that analysis it was shown that the full power risks were small relative to both the safety goals and to the other plants analyzed in the study. From this comparison the risk significance of POS 5 relative to full power can be determined which will help place the results from this study in context.

10.1 Comparison Between POS 5 and Full Power Risk

The POS 5 and full power distributions for early fatality risk, total latent cancer risk, population dose within 50 miles risk, and population dose within 1000 miles risk are displayed in Figures 10.1, 10.2, 10.3, and 10.4 respectively. To help place these risk distributions in context, the POS 5 and full power core damage frequency distributions are displayed in Figure 10.5. The mean values for the risk measures and the total core damage frequency from the two analyses are listed in Table 10-1. From these figures it can be seen that POS 5 risk and the full power risks are comparable as are the POS 5 and full power core damage frequencies. The mean, median, 5th percentile, and 95th percentile values from the POS 5 risk distributions are all greater than the corresponding values from the full power distributions with the greatest difference occurring between the risk distributions for the total latent cancer risk. However, from these figures it can be seen that any difference that exist between these distribution is small as is evident by the large overlap that exist between distributions for the same risk measure. For the core damage frequency, the difference that exist between POS 5 and full power is also small; the 95th percentile values from the POS 5 distribution are less than the corresponding values from the full power distribution while the median and 5th percentile values from the POS 5 distribution are greater than the full power distribution.

Table 10-1
Mean Core Damage Frequency and Mean Risks for POS 5 and Full Power

Analysis	Core Damage Frequency	Early Fatality Risk	Total Latent Cancer Risk	Population Dose (< 50 miles) Risk	Population Dose (<1000 miles) Risk
POS 5	2.1E-06	1.4E-08	3.8E-03	9.9E-01	8.7E+00
Full Power	4.1E-06	8.2E-09	9.5E-04	5.2E-01	5.8E+00

Since the core damage frequency for POS 5 is similar to the full power core damage frequency and since the site characteristics used to calculate offsite consequences (i.e., populations, land usage, weather conditions) are the same for both analyses, the primary difference between the two studies is the conditional probability of various magnitude releases. Even though decay has reduce the radionuclide inventory present during shutdown, the releases for POS 5 tend to be larger than the releases for full power because there are less features of the plant available to attenuate the release during shutdown. For many of these shutdown accidents the suppression pool is bypassed, the containment is open, and the containment sprays are unavailable. Since the latent health effects are affected by radionuclides with long half-lives, the affect of decay is not significant. For early health effect, however, which are affected by the radionuclides with short half-lives, the affect of decay is more important and the difference between full power accidents and shutdown accidents is not as significant. That is, for full power accidents there is a larger inventory of radionuclides that are important to early health effects, as compared to POS 5, however, because of the mitigative features of the plant that are available during power operation, not as much material escapes into the environment.

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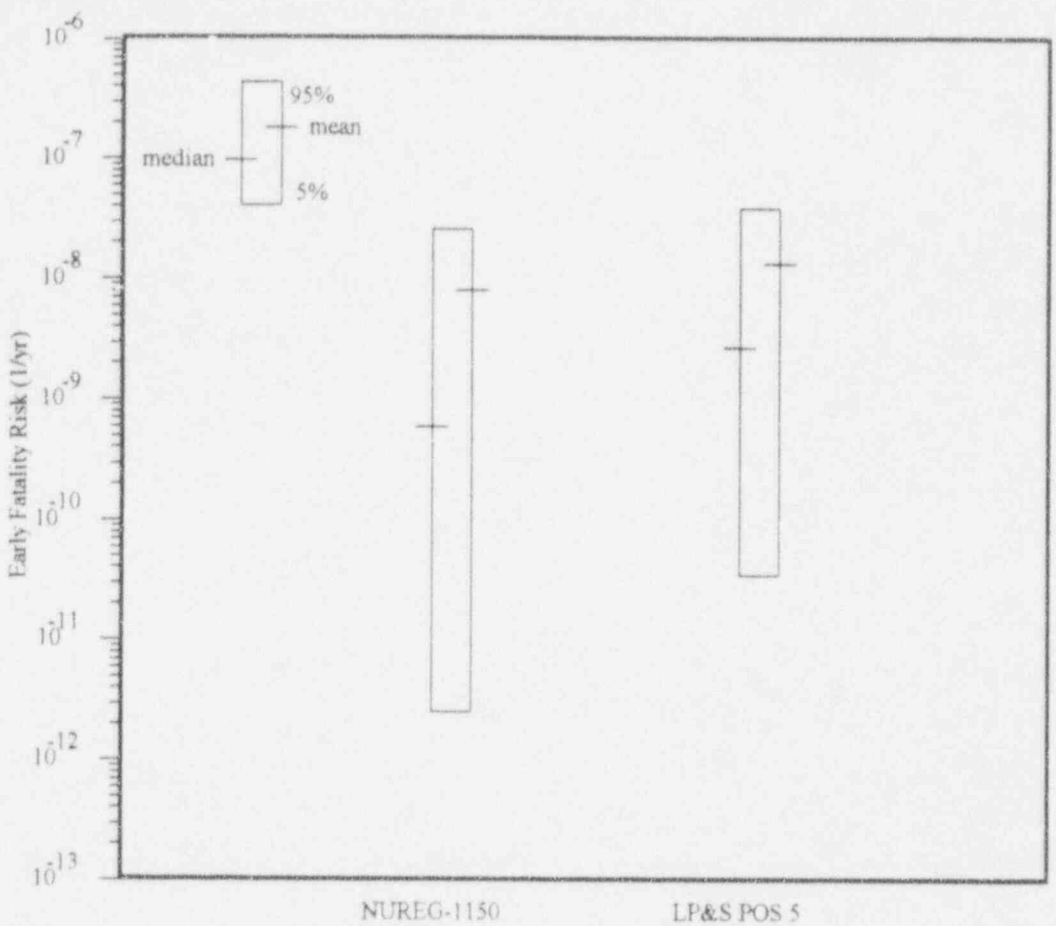

Figure 10-1
Comparison of Grand Gulf Early Fatality Risk

10.2 References

[USNRC, 1990] U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants," NUREG-1150, Vols. 1-3, December 1990-January 1991.

[Brown, et al., 1990] T. D. Brown et al., "Evaluation of Severe Accident Risks: Grand Gulf Unit 1," NUREG/CR-4551, SAND86-1309, Vol. 6, Rev. 1, Sandia National Laboratories, December 1990.

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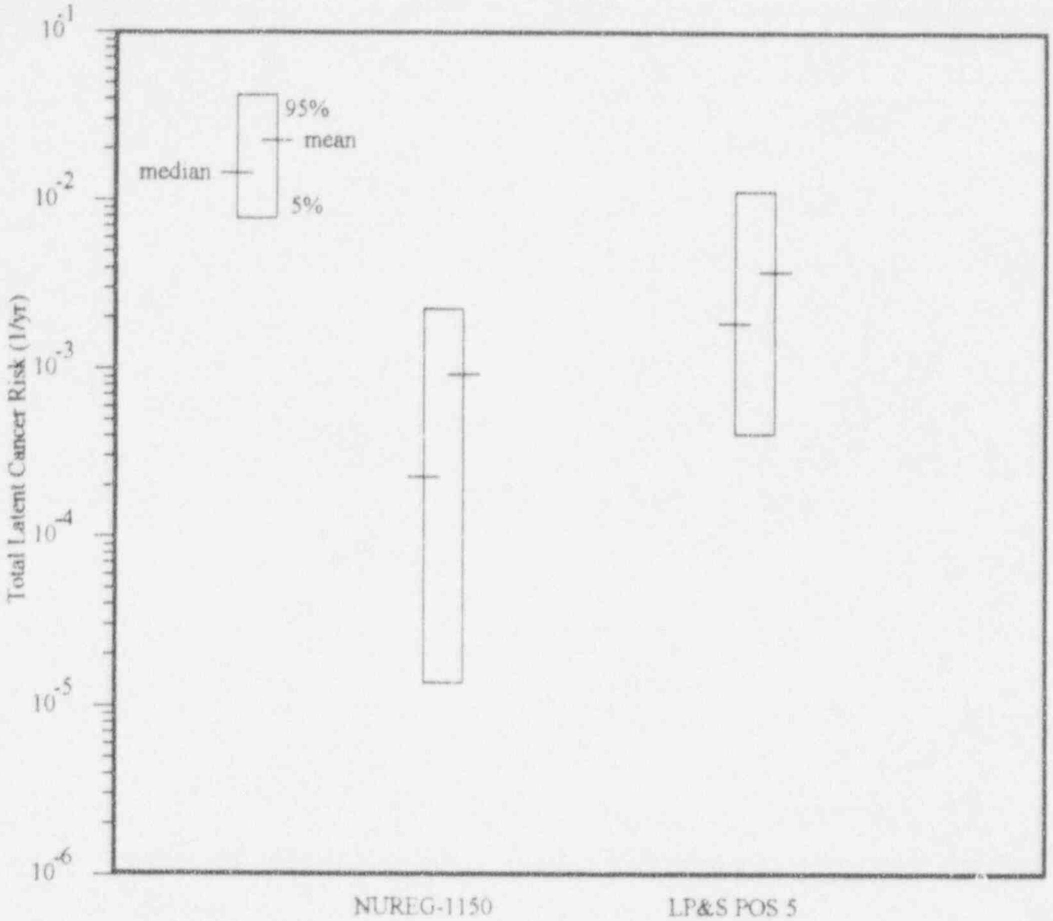

Figure 10-2
Comparison of Total Latent Cancer Risk

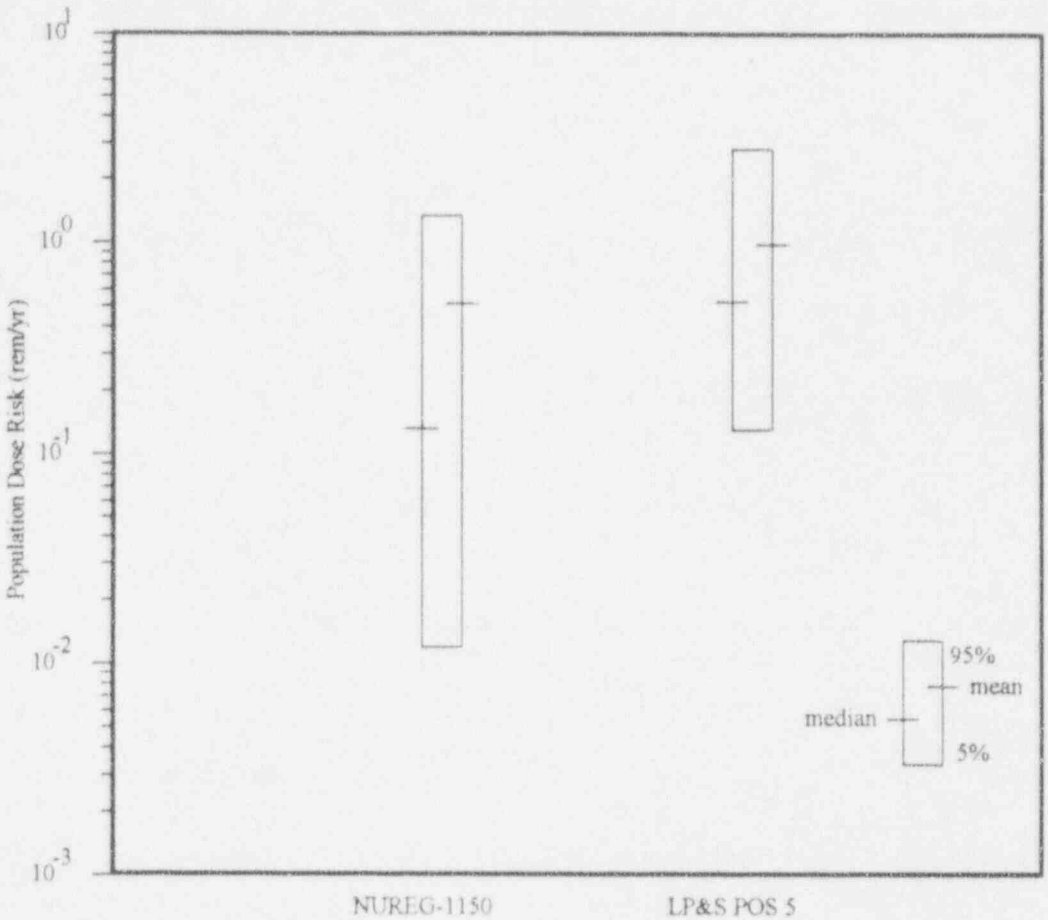

Figure 10-3
Comparison of Population Dose within 50 Miles of the Site Risk

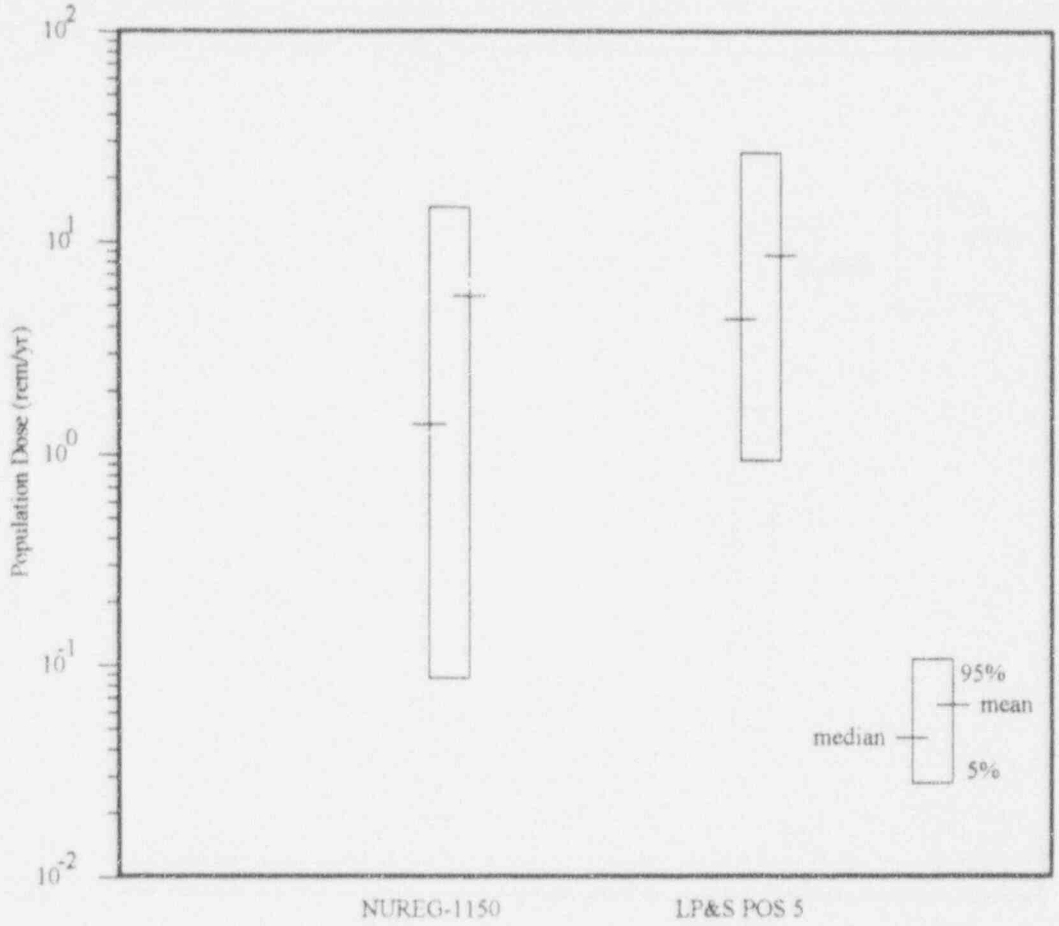

Figure 10-4
Comparison of Population Dose within 1000 miles of the Site Risk

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11 Open Issues

The study presented in this volume is for a single POS (namely POS 5) and, as such, assesses the risk associated with this POS. This study does not, however, attempt to assess the risk with the entire LP&S regime of operation. While the Level 1 screening study and other qualitative insights suggest that POS 5 is the risk dominant mode of shutdown, no detailed study has been performed on the other POSs to confirm this belief.

It is important to realize that reducing the risk in one POS, for example by changing when equipment is available and unavailable, can shift the risk to another POS. Since this study only addresses the risk associated with one POS, the affect of this change on overall risk (i.e., risk across all the POSs) cannot be quantitatively assessed.

For cases where the containment equipment hatch is open during the accident, the auxiliary building could play a major role in mitigating the release. The auxiliary building acts as a large holdup volume allowing time for natural processes to remove radionuclides from the building atmosphere before being released into the environment and is an important plant feature that mitigates the release to the environment. Although the auxiliary building was included in the MELCOR calculations performed for this study, no detailed analyses of the auxiliary building's structural capacity or its capability to retain radionuclides were performed. Instead, a number of assumptions were used in the modeling of this issue, any of which if changed, could have a significant impact on the results since the most likely accidents have the containment open and the radioactive releases pass through the auxiliary building. The assumptions include:

- While the auxiliary building is not a leak tight structure, it was assumed that none of the radioactive material escapes the building prior to it reaching its assumed failure pressure of 5 psig. This is only an issue for the accidents in which the operators flood the containment. In these accidents the auxiliary building pressurizes very slowly and is predicted to stay intact until vessel failure. In all of the other accident in which the containment is open, the building is predicted to fail prior to core damage.

- Neither the ventilation system nor the standby gas treatment system were modelled and, therefore, the affect (i.e., either beneficial or detrimental) that these systems would have on the building's ability to retain radioactive material was not considered.

- The decontamination factor used in this analysis could be approximated by Peach Bottom decontamination factor distribution developed for the NUREG-1150 analysis.

In addition to the issues discussed above, there is a more general issue that can affect the risk results generated in this study, namely, the use of NUREG-1150 data, which was developed for accidents initiated at power conditions, for these shutdown accidents. While timing information was obtained from the MELCOR calculations performed for this study, the NUREG-1150 data was used extensively to quantify the parameters in the parametric source term expression. Areas where there could be substantial differences between full power accidents and shutdown accident include the retention capabilities of the vessel during a LOCA and the containment when the containment equipment hatch is off.

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12 Summary

A Level 3 PRA was performed on the cold shutdown mode of operation up to the point where the reactor vessel head is detensioned for Unit 1 of the Grand Gulf Nuclear Station. In this analysis, the regime of shutdown analyzed is referred to as POS 5. The core damage frequency for POS 5 was determined in the Level 1 portion of the PRA and is documented in Volume 2 of this report. Risk for POS 5 was determined in the Level 2 and 3 portions of the PRA using a simplified form of the NUREG-1150 methodology. The Level 2/3 portion of the PRA is concerned with the progression of postulated accidents following the onset of severe core damage and the estimation of the consequences that result from the release of any radioactive material and, as such, consists of the following constituent analyses: plant damage state (PDS), accident progression, source term, consequence, and risk integration. Event tree techniques were used to delineate the accident progression following the onset of core damage and source terms were estimated using the parametric approach developed in NUREG-1150. The consequence analysis included the traditional offsite assessment (i.e., similar to the offsite consequence assessment performed in the NUREG-1150 plant studies) and also included a scoping assessment of onsite consequences. A limited uncertainty analysis, which included variables from the PDS, accident progression, and source term analyses, was also performed. In contrast to NUREG-1150, expert opinion techniques were not used in this study to quantify the accident progression and source term models.

For discussion purposes, the core damage scenarios identified in the Level 1 analysis can be combined into the following three PDS groups: LOCAs, Station Blackouts (SBOs), and Other Transients. The mean core damage frequencies and the mean fractional contributions to the core damage frequency for these three groups are provided in Table 12-1. The LOCA PDS group is the dominant contributor to the core damage frequency followed the by SBO PDS group and the Other Transients PDS group.

Based on the accident progression analysis performed for this study, the most likely accidents in POS 5 have an open containment, the suppression pool is bypassed, the containment sprays are not available, and the core damage process is not arrested in the vessel. For the cases where the vessel fails, there is a significant probability that the core debris will either be quenched in a flooded cavity or CCI will occur in a flooded cavity. For the former, the releases associated with CCI are prevented. In latter, the releases are scrubbed by the water in the flooded cavity. If the containment is closed prior to core damage, it is predicted to either fail or to be vented after core damage since containment heat removal is not available in these accidents; venting the containment late in the accident is the most likely scenario.

Table 12-2 presents the offsite risk results for the following six measures: early fatalities, total latent cancer fatalities, population dose within 50 miles of the site, population dose within 1000 miles of the site, individual early fatality risk within 1 mile of the site, and individual latent cancer risk within 10 miles of the site. The factors that lead to low offsite risk values include the following:

- The core damage frequency calculated for the Grand Gulf plant is extremely low. Thus, while a significant release may occur, the frequency of the release is sufficiently small that the resulting risk is also small.

Table 12-1
Core Damage Frequencies for LOCA, SBO, and
Other Transient PDS Groups

PDS Group	Mean Core Damage Frequency (1/yr)	Mean Fractional Contribution
LOCA	1.1E-06	0.51
SBO	7.4E-07	0.33
Other Transients	2.4E-07	0.17
Total	2.1E-06	

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Table 12-2
Distributions for Aggregated Risk for POS 5
(All values are per year)
(Population doses are in person-rem)

Consequence Measure	Descriptive Statistics				
	5th PCT	50th PCT	95th PCT	MEAN	STD Dev
Core Damage Frequency	4.1E-07	1.4E-06	5.6E-06	2.1E-06	2.7E-06
Early Fatality Risk	3.7E-11	2.8E-09	3.9E-08	1.4E-08	5.4E-08
Total Latent Cancer Risk	4.3E-04	1.9E-03	1.2E-02	3.8E-03	7.7E-03
Population Dose within 50 miles of the plant	1.3E-01	5.3E-01	3.1E+00	9.9E-01	1.9E+00
Population Dose within 1000 miles of the plant	9.9E-01	4.4E+00	2.8E+01	8.7E+00	1.8E+01
Individual Early Fatality Risk-- 0 to 1 mile	4.2E-13	2.7E-11	3.0E-10	9.6E-11	3.4E-10
Individual Latent Cancer Risk-- 0 to 10 miles	2.5E-10	9.4E-10	4.9E-09	1.6E-09	2.4E-09

- The population density around the Grand Gulf plant is also low. Although many factors influence the magnitude of the consequences, in general, for a given release, the smaller the population, the smaller the number of fatalities. Of the four Mark III plants in the United States, Grand Gulf has the fewest number of people living within 50 miles of the plant according the 1990 census data.

- Although in many of the accidents the containment equipment hatch is open, the suppression pool is bypassed, and the containment sprays are unavailable, the releases pass through the auxiliary building before escaping into the environment. Because of its large volume and surface area, the auxiliary building provides a location for the radionuclides to be attenuated by deposition and thereby reduce the source term to the environment. Without the auxiliary building, considerably more radioactive material would be released to the environment.

- The accidents delineated for these shutdown conditions progress more slowly and, therefore, there is generally more time for the public to respond to the accident and evacuate before they are exposed to the release. This is primarily important for the early health effects consequence measures which are more strongly affected by the time available to evacuate.

- Radioactive decay has reduced the radioactive potential of these shutdown accidents relative to the inventory that is present at shutdown. This factor is primarily important for early health effects which are more strongly affected by the shorter lived radionuclides. This affect is much less noticeable for latent health effects which are more strongly affected by the long lived isotopes.

To place the risk from POS 5 into context, it was compared to the risk from power operation as estimated in NUREG-1150 and was also compared to the NRC quantitative safety goals. The risk from POS 5 is comparable to the risk from power operation. While the mean risk from POS 5 is greater than the mean risk from full power operation, there is considerable overlap between the distribution suggesting that any difference that exists is small. The risk from POS 5 was also shown to be well below the safety goals. While the safety goals do not necessarily apply to selected modes of operation (i.e., ideally they should be compared to the plants total risk), a comparison of POS 5 risk to the safety goals does provide an indication, in an absolute sense, of the risk associated with this mode of operation.

The mean fractional contributions to risk from the LOCA, SBO, and Other Transient PDS groups are provided in Table 12-3. The SBO PDS group is the dominant contributor to the early fatality risks (total and individual). The SBO PDS

Table 12-3
Mean Fractional Contribution to Aggregate Risk for the LOCA, Station Blackout (SBO), and Other Transients Plant Damage State Groups

PDS Groups	Core Damage Frequency	Early Fatalities	Total Latent Cancers	Population Dose (<50 miles)	Population Dose (<1000 miles)	Individual Early Fatalities (0-1miles)	Individual Latent Cancers (0-10 miles)
LOCA	0.51	0.16	0.42	0.43	0.41	0.17	0.51
SBO	0.33	0.73	0.45	0.42	0.45	0.70	0.35
Other	0.17	0.12	0.13	0.15	0.14	0.12	0.14

group's large contribution to early fatality risk can be attributed to its relatively high contribution to the core damage frequency coupled with the fact that the containment equipment hatch is off in these accidents, the suppression pool is bypassed, and the auxiliary building fails early in the accidents. Combined, these factors cause the SBOs to have relatively high risk values. The LOCA PDS group, however, is not a dominant contributor to early fatality risk even though it is a dominant contributor to the core damage frequency. This stems primarily from the fact that the accidents that are the dominant contributor to the LOCA core damage frequency occur in POS 5 after core alterations (i.e., many weeks after shutdown) by which point the amount short-lived radionuclides that are important to early health have been significantly reduced by radioactive decay.

For latent cancer health effects, the LOCA and SBO PDS groups are the dominant contributors to risk. Since the radionuclides that are important to the latent health effects are the long lived radionuclides, these risk measures are not particularly sensitive to when the accident occurs relative to shutdown. Latent cancers primarily depend on the total amount of radioactive material released and not on when it was released (i.e., early in the accident versus late in the accident). Since latent cancers are not strongly dependent on the timing characteristics of the accident (i.e., start of release or release duration), the latent cancer risk will depend on the likelihood of the accident and on the total amount of radioactive material released. In all of the accidents delineated in this study, the containment is either open at the start of the accident or fails during the accident and in most of the accidents the core damage process is not arrested in the vessel. Although the timing of the accident may vary, all of the accidents have the potential to release a significant amount of radioactive material to the environment. Hence, the mean fractional contribution to latent cancer risk tends to be roughly proportional to the contribution to the core damage frequency. The fraction contributions from the LOCA and Other Transients tend to be less than there fractional contribution to the core damage frequency because for these PDSs portions of the release are scrubbed by either the suppression pool or by water in the reactor cavity. The fractional contribution from the SBO PDS group tends to be greater than the fractional contribution to the core damage frequency because for these accidents the containment is open at the start of the accident, the auxiliary building fails early in the accident, vessel always fails, CCI always occurs and none of the releases are scrubbed by water. Therefore, the releases associated with the SBO tend to be large relative to the other accidents analyzed in this study.

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Appendix A Supporting Information for the Plant Damage State Analysis

This appendix contains information that was used to support the PDS analysis. The approach used to develop PDS is described in Section 5.1 of the main report.

A.1 Development of End States

Sixteen characteristics were used to define the End States (ESs). These sixteen characteristics and their associated attributes are defined in Table A-1. In the Level 1 analysis, 28 sequences which contained a total of 38 cut sets were delineated. These cut sets were reviewed and the appropriate attribute for each characteristic was assigned to each cut set. The result was a 16 position alphanumeric string for each cut set. A unique string defines an ES. Through this process 22 ESs were defined. A list of these ESs is provided in Table A-2.

A.2 Development of Plant Damage States

Review of the ESs suggested that 11 characteristics would adequately define the PDSs. The characteristics and attributes for the PDSs are presented in Table A-3. ES characteristic 1 was dropped because there were too many initiating events to treat each one separately and in most cases the Level 2 analysis does not depend on the initiating event. This characteristic was included in the ES analysis for the sake of completeness. For cases where it is important to know the initiating event (e.g., LOCA or loss of offsite power), this information was included in a characteristic that addressed the event. For example, PDS Characteristic 1, Status of Electric Power, is used to identify accidents initiated by a loss of offsite power. Similarly, PDS Characteristic 7 is used to identify accidents initiated by a LOCA. Since in the Level 2 analysis large LOCAs (i.e., A) and intermediate size LOCAs (i.e., S1) are treated the same, this distinction does not have to be maintained in the PDS definition. ES Characteristics 3-6, Status of Core Cooling, were combined under one PDS characteristic. ES Characteristic 9, Status of Suppression Pool Makeup System, was incorporated into PDS Characteristics 3, Status of Containment Sprays and Suppression Pool Cooling, and Characteristic 4, Status of Suppression Pool Level. In going from ESs to PDSs, for a given characteristic the attributes were often changed to eliminate redundant or unnecessary information and to incorporate information from other characteristics that were eliminated.

In the development of PDSs from ESs a series of assumptions were made with regard to the station blackout ESs. These assumptions include:

- PDS1-2 consists of two sequences. In the first case the station batteries deplete within two hours of the initiating event resulting in the closure of the SRVs and the subsequent pressurization of the vessel. With the vessel pressurized, the firewater system cannot be used as an alternate source of injection. In this scenario, the vessel pressurizes to 440 psig and fails the shutdown cooling system resulting in an interfacing system LOCA. Core damage is estimated to occur 3.5 hours after the initiating event. In the second sequence the station batteries supply emergency dc power for at least 3.5 hours. In this scenario, two hours after the initiating event the operators open the SRVs and use the fire water system as an alternate source of coolant makeup. Injection is continued until the batteries deplete resulting in the closure of the SRVs and the pressurization of the vessel. The fire water system can not provide makeup once the vessel pressurizes. The batteries deplete sometime between 3.5 and 12 hours after the initiating event. In this analysis it is conservatively assume that the batteries fail 3.5 hours after the initiating event and, therefore, the firewater system injects water for only 1.5 hours. Following the loss of the firewater system, the accident progresses in a manner similar to the first sequence. Thus, the major difference between the two sequences is that core damage is delayed by approximately 1.5 hours in the second sequence. This delay of 1.5 hours is not sufficient to warrant a separate PDS and, therefore, the second sequence is conservatively model as though firewater was never used.

- In PDS1-3 it is assumed that the operators manually isolate the shutdown cooling system before the vessel begins to pressurize following the loss of core cooling and thereby prevent an interfacing systems LOCA in the SDC system. This action is possible in these accidents because there is a significant amount of time between the initiating event and the pressurization of the vessel (i.e., 12 hours). While this action was included in the

Level 1 model for Time Window 2, it was not included in the model for Time Window 1 because isolation of the SDC system in Time Window 1 does not prevent core damage. However, had this action been included in the Level 1 model, the cut sets that would have survived truncation would have the SDC system isolated. Thus, while the ES indicates there is a LOCA in the SDC system (Attribute F for Characteristic 14), the PDS indicates the vessel is at high pressure with pressure relief being provided by the SRVs.

- PDS2-2 consists of five similar sequences. In the first two sequences the station batteries deplete within 5.5 hours of the initiating event resulting in the closure of the SRVs and the subsequent pressurization of the vessel. With the vessel pressurized, the firewater system cannot be used as an alternate source of injection. In this scenario, the vessel pressurizes to 440 psig and fails the shutdown cooling system resulting in an interfacing system LOCA. Core damage is estimated to occur 5.5 hours after the initiating event. In the remaining three sequences the station batteries supply emergency dc power long enough for the operators to open the SRVs and align the fire water system to provide coolant to the core. The firewater system provides coolant makeup until the batteries deplete at which point the accident progress in a manner similar to the first two sequences except that core damage is delayed by several hours. Each of these last three sequences has a different battery depletion time (i.e., battery depletion times of 3, 5.5, and 12 hours). To keep the analysis manageable by reducing the number of PDS that needed to be analyzed, the last three sequences were combined with the first two sequences and were conservatively modelled as though the fire water system was not used.

The consolidation of ES characteristics and the simplifying assumptions regarding the use of the fire water system in the station blackout ESs resulted in the generation of twelve PDSs. A list of the ESs that are contained in each PDS is presented in Table A-2. The PDS definitions and their contribution to the point estimate core damage frequency are presented in Table A-4.

A.3 Comparison Between Level 1 and Level 2/3 PDS Frequencies

The Level 1 analysis documented in Volume 2 of this report used a LHS sample size of 1000 whereas the Level 2/3 analysis used a sample size of 200. While a sample size of 1000 can be used in the Level 1 analysis, the large computational requirements of accident progression and consequence analyses precluded the use of such a large sample size in the Level 2/3 analysis. When selecting the LHS sample size for the Level 2/3 analyses two objectives had to be considered: (1) the sample size had to be sufficiently large such that the Level 2/3 PDS results were reasonably similar to the Level 1 results, and (2) the sample size had to be small enough that the calculations could be performed in a timely manner. A sample size of 200 satisfied these two objectives. The PDS frequencies from these two samples are compared in Table A-5.

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Table A-1
End State Characteristics and Attributes

End State Characteristics and Attributes		
Charact.	Attribute	Description
1		**Initiating Event Type**
	A	T1-5
	B	A5
	C	A5HY
	D	E1B5H
	E	E1D5H
	F	E1T5H
	G	E1V5H
	H	E2B5H
	I	E2D5H
	J	E2T5H
	K	E2V5H
	L	H1-5H
	M	J2-5
	N	S1-5
	O	S1H-5
	P	S2-5
	Q	S2H-5
	R	T5A5H
	S	T5B5H
	T	T5C5H
	U	T5D5H
	V	TAB5H
	W	TDB5H
	X	TIOF5
	Y	TIA5H
	Z	TLM5H
	1	TRPT5

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End State Characteristics and Attributes		
Charact	Attribute	Description
		AVAILABILITY OF ELECTRICAL POWER
2		**Availability of Electrical Power**
	A	Offsite power (OSP) available, AC and DC B power available
	B	OSP not available - but recoverable
	C	OSP not available - not recoverable, delayed failure of core cooling
	D	OSP not available - not recoverable, prompt failure of core cooling
	E	OSP available - Emergency AC and DC power not available and not recoverable
		STATUS OF CORE COOLING
3		**Status of HPCS**
	A	HPCS available
	B	HPCS not available - not recoverable
	C	HPCS not available - but recoverable
4		**Status of LPCI**
	A	LPCI Train B available
	B	LPCI not available - not recoverable
	C	LPCI not available - but recoverable
	D	LPCI not available - but recoverable with recovery of OSP
5		**Status of Service Water Crosstie**
	A	SSW Crosstie available
	B	SSW Crosstie not available - not recoverable
	C	SSW Crosstie not available - but recoverable
	D	SSW Crosstie not available - but recoverable with recovery of OSP

End State Characteristics and Attributes		
Charact.	Attribute	Description
6		**Status of Condensate**
	A	CDS available
	B	CDS not available - not recoverable
	C	CDS not available - but recoverable
	D	CDS not available - but recoverable with recovery of OSP
		STATUS OF CONTAINMENT HEAT REMOVAL
7		**Status of Suppression Pool Level**
	A	Water at "Low Level" or "Drained Level"
	B	Suppression pool is empty
	C	Suppression pool level is at the ECCS suction strainers
8		**Status of Suppression Pool Temperature**
	A	Suppression Pool is sub-cooled
	B	Suppression Pool is saturated
	C	Suppression Pool temperature is not applicable
9		**Status of Suppression Pool Makeup**
	A	SPMU has been used
	B	SPMU is available but not used because it was not previously needed
	C	SPMU is available but not used because of operator error
	D	SPMU is not available but can be recovered with recovery of OSP
	E	SPMU is not available and cannot be recovered
10		**Status of Containment Sprays and Suppression Pool Cooling**
	A	CS/SPC available with heat exchangers
	B	CS/SPC not available - not recoverable
	C	CS/SPC not available - but recoverable with recovery of OSP

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End State Characteristics and Attributes		
Charact	Attribute	Description
		STATUS OF REACTOR VESSEL AND CONTAINMENT INTEGRITY
11		**Status of the Reactor Vessel Head Vent**
	A	Head vent is open during the accident
	B	Operators close the head vent prior to core damage
12		**Status of Containment Lower Personnel Lock**
	A	Containment lower personnel lock is open
	B	Containment status is unknown
13		**Status of Containment Vent System**
	A	CVS not required - but available
	B	CVS not required - not available
	C	CVS not required - not available - but recoverable with recovery of OSP
14		**Status of RPV Pressure**
	A	Vessel at high pressure - SRVs available but not used
	B	Vessel at high pressure - SRVs available but operator failed to use them
	C	Vessel at high pressure - SRVs not available
	D	Vessel at low pressure - SRVs are open by operator
	E	Vessel at low pressure - SRVs available and the vessel is open by LOCA
	F	Vessel at low pressure - SRVs available and the vessel is open by SDC break
	G	Vessel at low pressure - SRVs available and the vessel is open by open MSIVs

End State Characteristics and Attributes		
Charact.	Attribute	Description
		TIMING CHARACTERISTICS
15		**Time to Core Damage**
	A	Core damage occurs in 2 hour
	B	Core damage occurs in 2.35 hours
	C	Core damage occurs in 3 hours
	D	Core damage occurs in 3.5 hours
	E	Core damage occurs in 5.5 hours
	F	Core damage occurs in 6.75 hours
	G	Core damage occurs in 7 hours
	H	Core damage occurs in 7.35 hours
	I	Core damage occurs in 9.75 hours
	J	Core damage occurs in 12 hours
16		**Time Window**
	1	Time window 1: Ranges from 14 to 24 hours after shutdown
	2	Time window 2: Ranges from 24 to 94 hours after shutdown
	3	Time window 3: Ranges from 40 to 50.4 days after shutdown

Table A-2
End State and Plant Damage State Definitions

PDS	Seq. No.	No. Cut Sets	ES Characteristics																Pt. Est. CDF	Seq. Fract.	ES Fract.
			IE	PWR	HPCS	LPCI	SSW	CDS	SP-L	SP-T	SPMU	CS	HdVnt	Cnt	Vnt	RPV	TCD	Twin			
P1-1	1	1	B	A	B	B	B	B	A	A	C	B	A	A	B	E	A	1	2.05E-08	0.010	
P1-1	13	1	N	A	B	B	B	B	A	A	C	B	A	A	B	E	A	1	2.04E-08	0.010	0.019
P1-2	19	1	A	D	B	D	D	D	A	A	D	C	A	B	C	F	D	1	1.71E-08	0.008	
P1-2	21	1	A	D	B	D	D	D	A	A	D	C	A	B	C	F	D	1	1.43E-08	0.007	0.015
P1-3	18	2	A	C	B	D	D	D	A	B	D	C	A	B	C	F	J	1	6.71E-08	0.032	0.032
P1-4	20	1	A	B	B	D	D	D	A	A	D	C	A	B	C	F	D	1	1.15E-08	0.005	0.005
P1-5	9	1	L	A	B	B	B	B	A	A	E	B	B	A	B	D	H	1	1.62E-08	0.008	0.008
P2-1	2	1	B	A	B	B	B	B	A	A	C	B	A	A	B	E	B	2	5.79E-08	0.027	
P2-1	3	1	B	A	B	B	B	B	A	A	C	B	A	A	B	E	B	2	1.23E-07	0.058	
P2-1	14	1	N	A	B	B	B	B	A	A	C	B	A	A	B	E	B	2	5.76E-08	0.027	
P2-1	15	1	N	A	B	B	B	B	A	A	C	B	A	A	B	E	B	2	1.22E-07	0.058	0.170
P2-2	23	2	A	D	B	D	D	D	A	A	D	C	A	B	C	F	E	2	1.43E-07	0.067	
P2-2	25	2	A	D	B	D	D	D	A	A	D	C	A	B	C	F	E	2	8.54E-08	0.040	
P2-2	26	7	A	D	B	D	D	D	A	A	D	C	A	B	C	F	E	2	2.61E-07	0.123	
P2-2	27	1	A	D	B	D	D	D	A	A	D	C	A	B	C	F	C	2	1.39E-08	0.007	
P2-2	22	1	A	C	B	D	D	D	A	A	D	C	A	B	C	F	J	2	1.09E-08	0.005	0.242
P2-3	24	2	A	B	B	D	D	D	A	A	D	C	A	B	C	F	E	2	1.15E-07	0.054	0.054
P2-4	7	1	F	A	B	B	B	B	A	A	E	B	B	A	B	D	I	2	4.06E-08	0.019	
P2-4	8	1	J	A	B	B	B	B	A	A	E	B	B	A	B	D	I	2	5.10E-08	0.024	
P2-4	12	1	M	A	B	B	B	B	A	A	E	B	B	A	B	D	I	2	2.09E-08	0.010	
P2-4	10	1	L	A	B	B	B	B	A	A	E	B	B	A	B	D	I	2	1.08E-07	0.051	0.104
P2-5	28	1	R	A	B	B	B	C	A	A	B	B	A	B	A	A	J	2	1.39E-08	0.007	0.007
P2-6	11	1	L	A	B	B	A	B	C	A	A	B	B	B	A	G	F	2	1.31E-08	0.006	0.006
P3-1	4	1	B	A	B	B	B	B	A	A	C	B	A	A	B	E	G	3	2.57E-07	0.121	
P3-1	5	1	B	A	B	B	B	B	A	A	C	B	A	A	B	E	G	3	2.88E-08	0.014	
P3-1	6	1	C	A	B	B	B	B	A	A	C	B	A	A	B	E	G	3	2.03E-07	0.095	
P3-1	16	1	N	A	B	B	B	B	A	A	C	B	A	A	B	E	G	3	2.87E-08	0.014	
P3-1	17	1	O	A	B	B	B	B	A	A	C	B	A	A	B	E	G	3	2.00E-07	0.094	0.338
Totals	28	38																	2.12E-06	1.000	1.000

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Table A-3
Plant Damage State Characteristics and Attributes

Plant Damage State Characteristics and Attributes		
Charact.	Attribute	Description
		STATUS OF ELECTRICAL POWER
1		**Status of Electrical Power**
	A	Offsite power (OSP) available
	B	OSP not available - but recoverable
	C	OSP not available - not recoverable, delayed failure of core cooling
	D	OSP not available - not recoverable, prompt failure of core cooling
	E	OSP available - Emergency AC and DC power not available and not recoverable
		STATUS OF CORE COOLING
2		**Status of Core Coolant Injection**
	A	Core injection is not available and cannot be recovered
	B	LPCI and/or SSW crosstie are unavailable due to operator error
	C	LPCI and/or SSW crosstie are unavailable but recoverable with recovery of OSP
		STATUS OF CONTAINMENT HEAT REMOVAL
3		**Status of Containment Sprays and Suppression Pool Cooling**
	A	CS/SPC is not available and cannot be recovered
	B	CS/SPC is not available but can be recovered with recovery of OSP
	C	CS/SPC is available
4		**Status of Suppression Pool Level**
	A	Water at "Low Level" or "Drained Level"
	B	Suppression pool level is at the ECCS suction strainers
5		**Status of Suppression Pool Temperature**
	A	Suppression Pool is sub-cooled
	B	Suppression Pool is saturated

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Plant Damage State Characteristics and Attributes		
Charact.	Attribute	Description
		STATUS OF REACTOR VESSEL INTEGRITY
6		**Status of RPV Head Vent**
	A	Head vent is open during the accident
	B	Operators close the head vent prior to core damage
7		**Status of RPV Pressure and Integrity**
	A	Primary system is at system pressure
	B	Primary system is at low pressure (>400 psia)
	C	Primary system is at low pressure; RPV is breached by a LOCA inside containment
	D	Primary system is at low pressure; RPV is breached by a LOCA in SDC system
	E	Primary system is at low pressure; RPV is breached by open MSIVs
		STATUS OF CONTAINMENT INTEGRITY
8		**Status of Containment Lower Personnel Lock**
	A	Containment lower personnel lock is open
	B	Containment status is unknown
9		**Status of Containment Vent System**
	A	CVS is unavailable and cannot be recovered
	B	CVS is unavailable but can be recovered with recovery of OSP
	C	CVS is available but has not been used because is has not been needed

Plant Damage State Characteristics and Attributes		
Charact.	Attribute	Description
		TIMING CHARACTERISTICS
10		**Time to Core Damage**
	A	Core damage occurs in 2 hour
	B	Core damage occurs in 2.35 hours
	C	Core damage occurs in 3.5 hours
	D	Core damage occurs in 5.5 hours
	E	Core damage occurs in 6.75 hours
	F	Core damage occurs in 7 hours
	G	Core damage occurs in 9.75 hours
	H	Core damage occurs in 12 hours
11		**Time Windows**
	A	Time Window 1: Ranges from 14 to 24 hours after shutdown
	B	Time Window 2: Ranges from 24 to 94 hours after shutdown
	C	Time Window 3: Ranges from 40 to 50.4 days after shutdown

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Table A-4
Plant Damage State Definitions

Plant Damage States	Alpha-numeric Description	Fractional Contribution to PE CD Frequency
PDS1-1	A-A-AAA-AC-AA-A1	0.019
PDS1-2	D-C-BAA-AD-AB-C1	0.015
PDS1-3	C-C-BAB-AA-AB-H1	0.032
PDS1-4	B-C-BAA-AD-AB-C1	0.005
PDS1-5	A-A-AAA-BB-AA-F1	0.008
PDS2-1	A-A-AAA-AC-AA-B2	0.170
PDS2-2	D-C-BAA-AD-AB-D2	0.242
PDS2-3	B-C-BAA-AD-AB-D2	0.054
PDS2-4	A-A-AAA-BB-AA-G2	0.104
PDS2-5	A-B-AAA-AA-BC-H2	0.007
PDS2-6	A-B-ABA-BE-BC-E2	0.006
PDS3-1	A-A-AAA-AC-AA-F3	0.338
Total		1.00

Table A-5
Comparison Between Level 1 and Level 2 Plant Damage State Results

Plant Damage States	LHS Sample Size	Descriptive Statistics[1]: Core Damage Frequency (1/yr)				Fractional Contribution[2]	
		5%	50%	95%	Mean	FCM-CDF	MFC-CDF
PDS1-1	200	1.59E-09	1.41E-08	1.92E-07	4.13E-08	0.020	0.018
	1000	1.31E-09	1.51E-08	1.56E-07	4.05E-08	0.020	0.017
PDS1-2	200	1.44E-10	4.27E-09	1.29E-07	2.29E-08	0.011	0.015
	1000	2.15E-10	4.93E-09	1.11E-07	2.48E-08	0.012	0.016
PDS1-3	200	2.94E-09	1.73E-08	1.62E-07	4.39E-08	0.021	0.030
	1000	2.82E-09	1.75E-08	1.22E-07	3.59E-08	0.018	0.027
PDS1-4	200	5.98E-11	1.99E-09	3.49E-08	9.19E-09	0.004	0.006
	1000	9.33E-11	1.97E-09	3.79E-08	9.34E-09	0.005	0.007
PDS1-5	200	4.89E-10	6.85E-09	4.78E-08	1.36E-08	0.007	0.010
	1000	4.78E-10	6.03E-09	6.68E-08	1.59E-08	0.008	0.011
PDS2-1	200	1.28E-08	1.35E-07	1.45E-06	3.52E-07	0.168	0.153
	1000	1.13E-08	1.35E-07	1.41E-06	3.67E-07	0.184	0.154
PDS2-2	200	2.22E-08	1.45E-07	1.60E-06	5.53E-07	0.264	0.217
	1000	2.12E-08	1.72E-07	1.63E-06	4.47E-07	0.224	0.210
PDS2-3	200	2.74E-09	2.93E-08	4.46E-07	1.11E-07	0.053	0.059
	1000	3.05E-09	2.87E-08	3.33E-07	9.29E-08	0.046	0.055
PDS2-4	200	7.67E-09	8.80E-08	7.52E-07	2.03E-07	0.097	0.140
	1000	6.04E-09	9.93E-08	8.19E-07	2.14E-07	0.107	0.140
PDS2-5	200	8.55E-11	2.74E-09	5.32E-08	1.26E-08	0.006	0.010
	1000	1.06E-10	2.59E-09	6.01E-08	1.47E-08	0.007	0.010
PDS2-6	200	2.56E-11	1.11E-09	2.83E-08	7.43E-09	0.004	0.006
	1000	1.69E-11	1.07E-09	5.22E-08	1.17E-08	0.006	0.009
PDS3-1	200	6.25E-08	3.75E-07	2.40E-06	7.27E-07	0.347	0.338
	1000	6.50E-08	3.75E-07	2.29E-06	7.26E-07	0.363	0.343
Total	200	4.07E-07	1.37E-06	5.56E-06	2.10E-06		
	1000	4.13E-07	1.34E-06	5.38E-06	2.00E-06		

Note 1: A LHS sample size of 200 was used in the Level 2/3 and integrated analyses whereas a LHS sample size of 1000 was used in the Level 1 analysis

Note 2: FCM-CDF = Fractional contribution to mean core damage frequency

MFC-CDF = Mean fractional contribution to core damage frequency

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Appendix B Supporting Information for the Accident Progression Analysis

This appendix contains information that was used to support the Accident Progression Analysis. The many possible progressions of the accident following the onset of core damage are delineated and evaluated using event tree techniques. The event tree developed to model the Level 2 portion of the accident is called an Accident Progression Event Tree (APET). The APET is evaluated using the EVNTRE code. EVNTRE calculates the probability of each path through the APET and combines the paths into groups, specified by the user, call Accident Progression Bins (APBs). These APBs can then be further grouped, or rebinned, and sorted using the PSTEVNT code. Section B.1 provides a discussion of each of the questions in the Accident Progression Event Tree (APET); the APET that forms the input to the EVNTRE code is listed in Section B.2. The quantification of the APET is discussed in Section B.3.

B.1 Description of the Grand Gulf POS 5 APET

In the following subsection, the purpose for each question in the APET is discussed, the branches defined, and the sources of information used to quantify the branches presented. The probabilites assigned to each branch are also presented; for cases where a distribution of probabilities is used, the mean of the distribution is displayed. A single APET is used to evaluate all of the PDSs.

Question 1. What is the Plant Damage State?
Number of Branches: 12
Number of Cases: 1
Number of Cases Sampled: 1

The branches for this question are:

1. PDS1-1 PDS1-1: LOCA in Time Window 1
2. PDS1-2 PDS1-2: Station Blackout accident in Time Window 1
3. PDS1-3 PDS1-3: Station Blackout accident in Time Window 1
4. PDS1-4 PDS1-4: Station Blackout accident in Time Window 1
5. PDS1-5 PDS1-5: Flooded containment accident in Time Window 1
6. PDS2-1 PDS2-1: LOCA in Time Window 2
7. PDS2-2 PDS2-2: Station Blackout accident in Time Window 2
8. PDS2-3 PDS2-3: Station Blackout accident in Time Window 2
9. PDS2-4 PDS2-4: Flooded containment accident in Time Window 2
10. PDS2-5 PDS2-5: High pressure core damage accident in Time Window 2
11. PDS2-6 PDS2-6: Open MSIV accident in Time Window 2
12. PDS3-1 PDS3-1: LOCA in Time Window 3

This question defines the probability of each Plant Damage State (PDS) conditional on the occurrence of core damage. Twelve PDS were defined in this analysis. The branch probabilities are sampled; the probability distribution for each PDS is based on the frequency of each PDS obtained from the IRRAS code.

The quantification for this question is:

Branch 1:	PDS1-1	0.019
Branch 2:	PDS1-2	0.015
Branch 3:	PDS1-3	0.032
Branch 4:	PDS1-4	0.005
Branch 5:	PDS1-5	0.008
Branch 6:	PDS2-1	0.17
Branch 7:	PDS2-2	0.242
Branch 8:	PDS2-3	0.054
Branch 9:	PDS2-4	0.104
Branch 10:	PDS2-5	0.007

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Number of Cases Sampled: None

The branches for this question are:

1.	nLPInj	Low pressure injection is not available and cannot be recovered.
2.	nLPInj-op	Low pressure injection is not available because of operator errors
3.	rLPInj	Low pressure injection is not available but can be recovered with the recovery of OSP.
4.	aLPInj	Low pressure injection is available but not being used.

This question is used to define the PDSs. This question addresses PDS characteristic 2 which defines the availability of core coolant injection at the onset of core damage. For PDSs initiated by a loss of offsite power (PDS1-2, PDS1-3, PDS1-4, PDS2-2, and PDS2-3), low pressure injection is not available but can be recovered once offsite power is restored to the plant and, therefore, the probability of Branch 3 is set to 1.0. For PDS2-6, the service water cross-tie system was available, however the operators fail to use this system and therefore, the probability of Branch 2 is set to 1.0. For all of the other PDSs, the probability of Branch 1 is set to 1.0. There are no PDSs in this analysis in which low pressure injection is available at the onset of core damage and therefore Branch 4 is never used in this analysis.

Question 6. What is the status of containment sprays and suppression pool cooling at core damage (PDS Char. 3)?
Number of Branches: 4
Number of Cases: 3
Number of Cases Sampled: None

The branches for this question are:

1.	nCS	Containment sprays (CS) are not available and cannot be recovered.
2.	rCS	CS are not available but can be recovered with the recovery of OSP.
3.	alignCS	CS are not available not aligned to provide containment cooling.
4.	autoCS	CS is available and aligned to provide containment cooling.

This question is used to define the PDSs. This question addresses PDS characteristic 3 which defines the status of containment sprays (CS) at the onset of core damage. For the PDSs initiated by a loss of offsite power (PDS1-2, PDS1-3, PDS1-4, PDS2-2 and PDS2-3) CS are not available but can be recovered if offsite power is restore and, therefore, the probability of Branch 2 is set to 1.0. For all of the other PDSs the containment sprays are not available and cannot be recovered and, therefore, the probability of Branch 1 is set to 1.0.

Question 7. What is the suppression pool level at core damage (PDS Char. 4)?
Number of Branches: 2
Number of Cases: 2
Number of Cases Sampled: None

The branches for this question are:

1.	SPL-Lo	The suppression pool level is no lower than the low water level (LWL).
2.	SPL-Strain	The suppression pool has been drained to the ECCS suction strainers.

This question is used to define the PDSs. This question addresses PDS characteristic 4 which defines the level of the suppression pool level at the onset of core damage. For PDS2-6, the probability of Branch 2 is set to 1.0. For all of the other PDSs, the suppression pool is no lower than the LWL and, therefore, the probability of Branch 1 is set to 1.0.

Question 8. What is the suppression pool temperature at core damage (PDS Char. 5)?
Number of Branches: 2
Number of Cases: 3

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Branch 11: PDS2-6	0.006
Branch 12: PDS3-1	0.338

Question 2. What is the status of electric power at core damage (PDS Char. 1)?
Number of Branches: 3
Number of Cases: 3
Number of Cases Sampled: None

The branches for this question are:

1. aOSP Offsite and onsite electric power are available.
2. OSP-nDIVAC Offsite power is available, however, the onsite ac buses are failed.
3. nOSP Neither offsite nor onsite ac power is available.

This question is used to define the PDSs. This question addresses PDS characteristic 1 which defines the availability of electric power at the onset of core damage. For PDSs initiated by a loss of offsite power (PDS1-2, PDS1-3, PDS1-4, PDS2-2, and PDS2-3), both offsite and onsite power are unavailable and, therefore, the probability of Branch 3 is set to 1.0. For all of the other PDSs, the probability of Branch 1 is set to 1.0. Branch 2 is not used in this analysis but was include for the sake of completeness.

Question 3. What is the status of dc power at core damage (PDS Char. 1)?
Number of Branches: 2
Number of Cases: 2
Number of Cases Sampled: None

The branches for this question are:

1. nDC-BCD The station batteries have depleted and dc power cannot be recovered.
2. aDC-BCD DC power is available at the onset of core damage.

This question is used to define the PDSs. This question addresses PDS characteristic 1 which defines the availability of electric power at the onset of core damage. If ac power is not available and the station batteries deplete, it is assumed that neither ac nor dc power can be recovered. For PDSs PDS1-2, PDS1-3, and PDS2-2, the probability of Branch 1 is set to 1.0. For all of the other PDSs, the probability of Branch 2 is set to 1.0.

Question 4. What is the status of high pressure injection at core damage (PDS Char. 2)?
Number of Branches: 3
Number of Cases: 1
Number of Cases Sampled: None

The branches for this question are:

1. nHPInj High pressure injection is not available and cannot be recovered.
2. rHPInj High pressure injection is not available but can be recovered with the recovery of OSP.
3. HPInj High pressure injection is available but not being used.

This question is used to define the PDSs. This question addresses PDS characteristic 2 which defines the availability of core coolant injection at the onset of core damage. For all of the PDS defined in this analysis, the probability of Branch 1 is set to 1.0.

Question 5. What is the status of low pressure injection at core damage (PDS Char. 2)?
Number of Branches: 4
Number of Cases: 4

Number of Cases Sampled: None

The branches for this question are:

1. SPT-Sub The suppression pool is subcooled at the onset of core damage.
2. SPT-Sat The suppression pool is saturated at the onset of core damage.

This question is used to define the PDSs. This question addresses PDS characteristic 5 which defines the temperature of the suppression pool at the onset of core damage. For PDS1-3, the suppression pool is saturated at the time of core damage and, therefore, the probability of Branch 2 is set to 1.0. For all of the other PDSs, the suppression pool is subcooled and, therefore, the probability of Branch 1 is set to 1.0.

Question 9. What is the status of the reactor head vent at core damage (PDS Char. 6)?
Number of Branches: 2
Number of Cases: 2
Number of Cases Sampled: None

The branches for this question are:

1. RPV-nVnt The reactor head vent is not open during core damage.
2. RPV-OVnt The reactor head vent is open during core damage.

This question is used to define the PDSs. This question addresses PDS characteristic 6 which defines the status of the reactor head vent at the time of core damage. The reactor head vent is a two inch pipe that vents the reactor to the sump located in the reactor cavity directly below the vessel. While this line is closed and is not used during normal operation, it is open during cold shutdown and, therefore, will be open at the start of the accident. The plant's inadequate heat removal procedures, however, direct the operators to close motor operated valves on the head vent line in the event that core cooling cannot be maintained. The valves cannot be closed during a station blackout. Furthermore, if the operators have failed to follow procedures in a particular accident, it is assumed that the operators will also fail to close the head vent. The head vent is only closed in PDSs PDS1-5, PDS2-4, and PDS2-6 and, therefore, the probability of Branch 1 is set to 1.0. For all of the other PDSs, the head vent is open in which case the probability of Branch 2 is set to 1.0.

Question 10. What is the status of the reactor pressure vessel integrity at core damage (PDS Char. 7)?
Number of Branches: 5
Number of Cases: 6
Number of Cases Sampled: None

The branches for this question are:

1. RPV-HiP The reactor vessel has not been breached. The pressure in the vessel is at system pressure with pressure relief being provided by the SRVs at their setpoints.
2. RPV-LoP The reactor vessel has not been breach. At least two SRVs are open and the pressure in the vessel is low (<400 psia).
3. RPV-LOCA The reactor vessel has been breached by a LOCA located inside the containment. The vessel is at low pressure at the onset of core damage.
4. RPV-ILOCA The reactor vessel has been breached by a LOCA located outside the containment. The vessel is at low pressure at the onset of core damage.
5. RPV-oMSIV The MSIVs are open and, therefore, the vessel is at low pressure at the onset of core damage.

This question is used to define the PDSs. This question addresses PDS characteristic 7 which defines the status of the reactor integrity and the pressure of the reactor vessel at the onset of core damage. For PDSs initiated by a LOCA (PDS1-1, PDS2-1, PDS3-1), the probability for Branch 3 is set to 1.0. For PDSs initiated by a loss of offsite

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power followed by a break outside the containment in the shutdown cooling system (PDS1-2, PDS1-4, PDS2-2, and PDS2-3), the probability for Branch 4 is set to 1.0. For PDS2-6 in which the operators fail to close the MSIVs, the probability for Branch 5 is set to 1.0. Two SRVs are open in PDSs PDS1-5 and PDS2-4 and, therefore, the vessel is at low pressure and the probability for Branch 2 is set to 1.0. For the remaining PDSs, the reactor vessel is pressurized to system pressure and, therefore, for these PDSs the probability of Branch 1 is set 1.0.

Question 11. What is the status of the containment access penetrations at core damage (PDS Char. 8)?
Number of Branches: 2
Number of Cases: 2
Number of Cases Sampled: None

The branches for this question are:

1. o-LPersLk — The lower containment personnel lock is open at the onset of core damage.
2. LPersLk-Unk — The status of the lower personnel lock is unknown at the onset of core damage.

This question is used to define the PDSs. This question addresses PDS characteristic 8 which defines the status of the lower containment personnel lock. The lower personnel lock will be open if the operators flooded the containment and failed to close the lower personnel lock or if the accident is initiated by a loss of offsite power (the closure of the containment equipment hatch requires offsite power). There are only two PDSs for which the status of the containment penetrations is not known: PDS2-5 and PDS2-6. For these two PDSs, the probability for Branch 2 is set to 1.0. For all of the other PDSs, the lower containment personnel lock and/or the containment equipment hatch are open and, therefore, the probability of Branch 1 is set to 1.0. It should be pointed out, however, that even if the operators close the equipment hatch in PDS2-6, the MSIVs are still open and any releases will escape out into the turbine building.

Question 12. What is the status of the containment vent system at core damage (PDS Char. 9)?
Number of Branches: 3
Number of Cases: 4
Number of Cases Sampled: None

The branches for this question are:

1. nCVS — The containment vent system (CVS) is unavailable at the onset of core damage and cannot be recovered during the accident.
2. rCVS — The CVS is unavailable at the onset of core damage but can be recovered following the recovery of offsite power.
3. aCVS — The CVS is available at the onset of core damage.

This question is used to define the PDSs. This question addresses PDS characteristic 9 which defines the status of the containment vent system. To open the containment vent requires emergency ac power. For PDSs initiated by a loss of offsite power (PDS1-2, PDS1-3, PDS1-4, PDS2-2, and PDS2-3), the CVS is unavailable but recoverable once offsite power is restored to the plant. For these PDSs, the probability for Branch 2 is set to 1.0. For PDS2-5 and PDS2-6, the CVS is available and, therefore, the probability for Branch 3 is set to 1.0. For all of the other PDSs, the CVS is not available and cannot be recovered and, therefore, the probability of Branch 1 is set to 1.0.

Question 13. When does core damage occur (PDS Char. 10)?
Number of Branches: 8
Number of Cases: 9
Number of Cases Sampled: None

The branches for this question are:

1. TCD2 Core damage occurs approximately 2 hours after the initiation of the accident.
2. TCD2p3 Core damage occurs approximately 2.35 hours after the initiation of the accident.
3. TCD3 Core damage occurs approximately 3.5 hours after the initiation of the accident.
4. TCD5 Core damage occurs approximately 5.5 hours after the initiation of the accident.
5. TCD6 Core damage occurs approximately 6.75 hours after the initiation of the accident.
6. TCD7 Core damage occurs approximately 7 hours after the initiation of the accident.
7. TCD9 Core damage occurs approximately 9.75 hours after the initiation of the accident.
8. TCD12 Core damage occurs approximately 12 hours after the initiation of the accident.

This question is used to define the PDSs. This question addresses PDS characteristic 10 which defines the time at which core damage occurs. In this analysis, core damage is defined at the start of significant fuel heatup. The times to core damage were obtained from a series of MELCOR calculations that were performed specifically for this analysis. These calculations are documented in Part 2 of this Volume. For PDS1-1, the probability for Branch 1 is set to 1.0. For PDS2-1, the probability for Branch 2 is set to 1.0. For PDS1-2 and PDS1-4, the probability for Branch 3 is set to 1.0. For PDS2-2 and PDS2-3, the probability for Branch 4 is set to 1.0. For PDS2-6, the probability for Branch 5 is set to 1.0. For PDS1-5 and PDS3-1, the probability for Branch 6 is set to 1.0. For PDS2-4, the probability for Branch 7 is set to 1.0. For PDS1-3 and PDS2-5, the probability for Branch 8 is set to 1.0.

Question 14. While in POS 5, when does the initiating event occur (PDS Char. 11)?
Number of Branches: 3
Number of Cases: 5
Number of Cases Sampled: None

The branches for this question are:

1. IE-Win1 The initiating event occurs while the plant is in Time Window 1. Time Window 1 ranges from 14 hours after shutdown to 24 hours after shutdown.
2. IE-Win2 The initiating event occurs while the plant is in Time Window 2. Time Window 2 ranges from 24 hours after shutdown to 94 hours after shutdown.
3. IE-Win3 The initiating event occurs while the plant is in Time Window 3. Time Window 3 ranges from 40 days after shutdown to 50.4 days after shutdown.

This question is used to define the PDSs. This question addresses PDS characteristic 11 which defines the Time Window that the plant is in when the initiating event occurs. The Time Window defines the radionuclide inventory and the decay heat load at the start of the accident. Time Windows 1 and 2 occur during POS 5 prior to the plant entering Operating Condition 5 (Refueling). Time Window 3 occurs during POS 5 on the way back up to power after refueling. PDS1-1, PDS1-2, PDS1-3, PDS1-4, and PDS1-5 occur while the plant is in Time Window 1 and, therefore, for these PDSs the probability for Branch 1 is set to 1.0. PDS2-1, PDS2-2, PDS2-3, PDS2-4, PDS2-5 and PDS2-6 occur while the plant is in Time Window 2 and, therefore, for these PDSs the probability for Branch 2 is set to 1.0. PDS3-1 occurs while the plant is in Time Window 3 and, therefore, for this PDSs the probability for Branch 3 is set to 1.0.

Question 15. What type of event initiates the accident?
Number of Branches: 3
Number of Cases: 4
Number of Cases Sampled: None

The branches for this question are:

1. IE-LOCA The accident is initiated by a LOCA
2. IE-SBO The accident is initiated by a loss of offsite power followed by loss of emergency Division A and B power that results in a station blackout
3. IE-Other The accident is not initiated by a LOCA and is not initiated by a loss of offsite power.

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This is a summary question and is used to summarize information from previous questions. This question partitions the accidents into three classes: (1) those initiated by a LOCA, (2) those initiated by a loss of offsite power that result in a station blackout, and (3) those initiating by all other types of events.

Case 1: This case includes those accidents initiated by a LOCA (i.e., PDS1-1, PDS2-1, and PDS3-1). The quantification for this case is:

Branch 1:	IE-LOCA	1.0
Branch 2:	IE-SBO	0.0
Branch 2:	IE-Other	0.0

Case 2: This case includes those accidents initiated by a loss of offsite power while the plant is Time Window 1 followed by a loss of all emergency ac power resulting in a station blackout scenario. The quantification for this case is:

Branch 1:	IE-LOCA	0.0
Branch 2:	IE-SBO	1.0
Branch 2:	IE-Other	0.0

Case 3: This case includes those accidents initiated by a loss of offsite power while the plant is Time Window 2 followed by a loss of all emergency ac power resulting in a station blackout scenario. The quantification for this case is:

Branch 1:	IE-LOCA	0.0
Branch 2:	IE-SBO	1.0
Branch 2:	IE-Other	0.0

Case 4: This case includes all of the remaining accidents that are not captured by Cases 1-3. The quantification for this case is:

Branch 1:	IE-LOCA	0.0
Branch 2:	IE-SBO	0.0
Branch 3:	IE-Other	1.0

Question 16. What is the pressure in the RPV at the time of core damage?
Number of Branches: 2
Number of Cases: 2
Number of Cases Sampled: None

The branches for this question are:

1. RPV-HiP-BCD The reactor vessel is at system pressure (i.e., 1000 psia) at the onset of core damage
2. RPV-LoP-BCD The reactor vessel is at low pressure (i.e., < 400 psia) at the onset of core damage

This is a summary question and is used to summarize information from previous questions. This question partitions the accidents into two categories: (1) those accidents in which the vessel is pressurized and pressure relief is being provided by the SRVs cycling at their setpoints, (2) those accidents in which the vessel is at low pressure.

Case 1: This case includes those accidents identified as being at high pressure in Question 10. The quantification for this case is:

Branch 1: RPV-HiP-BCD 1.0

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Branch 2: RPV-LoP-BCD 0.0

Case 2: This case includes all of the rest of the accidents in which the reactor vessel is not at system pressure and, therefore, they are at low pressure (<400 psia). The reactor vessel will be at low pressure if: (1) the accident is initiated by a LOCA, (2) there is an unisolated LOCA outside the containment, (3) the MSIVs are open, or (4) the operators open two SRVs. The reactor head vent does not have the to capacity to keep the reactor vessel depressurized in the event that core cooling is unavailable. Thus, an open head vent will not by itself maintain the vessel at low pressure. The quantification for this case is:

Branch 1: RPV-HiP-BCD 0.0
Branch 2: RPV-LoP-BCD 1.0

Question 17. How much water is in the reactor pedestal cavity at the time of core damage?
Number of Branches: 2
Number of Cases: 3
Number of Cases Sampled: None

The branches for this question are:

1. Cav-Dry-BCD The reactor cavity is flooded at the onset of core damage
2. Cav-Fld-BCD The reactor cavity is essentially dry at the onset of core damage

This is a summary question and is used to summarize information from previous questions. This question partitions the accidents into two categories: (1) those accidents in which the reactor cavity is flooded with water at the onset of core damage and (2) those accidents in which the reactor cavity is essentially dry at the onset of core damage. The reactor cavity is located directly below the reactor and is partly recessed in the drywell floor. Water on the drywell floor will drain into the reactor cavity where it is collected in sumps. These sumps are then periodically drained. Water from several sources can enter the drywell: (1) normal equipment leakage, (2) LOCA, (3) overflow from the suppression pool (e.g., during containment flooding operations).

Case 1: This case includes those accidents initiated by a LOCA. In these accidents the operators flood the containment to a level that corresponds to the bottom of the lower containment personnel lock. In these accidents, the lower personnel lock is open and, therefore, the containment is only flooded to this level. The quantification for this case is:

Branch 1: Cav-Dry-BCD 0.0
Branch 2: Cav-Fld-BCD 1.0

Case 2: This case includes those accidents that were not initiated by a LOCA but in which the operators flood the containment to a level that corresponds to the bottom of the lower containment personnel lock in an attempt to prevent core damage. In these accidents, the operators fail to close the lower personnel lock and, therefore, the containment is only flooded to this level. The quantification for this case is:

Branch 1: Cav-Dry-BCD 0.0
Branch 2: Cav-Fld-BCD 1.0

Case 3: This case includes all of the remaining accidents which were not initiated by a LOCA and in which the operators did not flood the containment. In these accidents the reactor cavity was essentially dry. Normal leakage from the equipment would not flood the cavity during the time frame of interest. The quantification for this case is:

Branch 1: Cav-Dry-BCD 1.0
Branch 2: Cav-Fld-BCD 0.0

Question 18. Is the containment equipment hatch open at the start of the accident?
Number of Branches: 2
Number of Cases: 1
Number of Cases Sampled: None

The branches for this question are:

1.	nOCnt-S	The containment equipment hatch is not open at the start of the accident.
2.	OCnt-S	The containment equipment hatch is open at the start of the accident.

This question addresses status of the containment equipment at the start of the accident. The equipment hatch can be removed as soon as the plant enters POS 5. A review of the Grand Gulf refueling outage critiques indicated that the removal process typically begins soon after the plant enters POS 5; similarly, the closure process is typically performed near the end of POS 5. Thus, the amount of time that the plant is in POS 5 with the equipment hatch in place is relatively small compared to the total amount of time that the plant is in POS 5. Therefore, while there will be a small portion of time during Time Window 1 and Time Window 3 in which the equipment hatch is in place, for this analysis it is assumed that the equipment hatch is always off at the start of the accident. The quantification for this question is:

Branch 1:	nOCnt-S	0.0
Branch 2:	OCnt-S	1.0

Question 19. Do the operators close the containment before core damage?
Number of Branches: 2
Number of Cases: 5
Number of Cases Sampled: 1

The branches for this question are:

1.	nOCnt-BCD	The containment access penetrations are not open at the onset of core damage
2.	OCnt-BCD	The containment access penetrations are open at the onset of core damage

This question determines whether the operators successfully replace the containment equipment hatch and close the containment personnel locks prior to the onset of core damage. The information used to assess this issue was obtained from the Grand Gulf plant. The Grand Gulf plant utilizes a 19 ft diameter, steel pressure seating equipment hatch. The hatch is attached from inside the containment via 20 bolts. The hatch uses two compression seals (gasket concept) around its periphery to maintain tightness along the mating surfaces. The hatch is stored inside the containment in a storage bin above the opening. Offsite ac power is required to move and position the hatch. Each personnel airlock consists of a cylindrical steel shell with steel bulkheads at each end and two steel doors in the bulkheads which open toward the reactor. Sealing of each door is accomplished by two, continuous inflatable seals which surround the door edge. In this analysis, successful close of the access penetrations requires the following two conditions: (1) offsite power must be available to move the equipment hatch into place, (2) there must be sufficient time before the onset of core damage to position and attach the equipment hatch. In this analysis it is estimated that at least 5 hours is needed to perform the required action to close the containment. Because of the severe environment that will be present in the containment following the onset of core damage, no credit is given for closure of the containment after the onset of core damage. This issue was addressed in the Human Reliability Analysis (HRA) that was performed for the Level 2 analysis and is discussed further in Appendix B.3.

Case 1. This case includes those accidents in which the containment access penetrations were not open at the start of the accident and, therefore, there are no penetrations to close. The quantification for this case is:

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Branch 1:	nOCnt-BCD	1.0
Branch 2:	OCnt-BCD	0.0

Case 2: This case includes those accidents in which the operators failed to close the lower personnel lock and, therefore, were unable to successfully flood the containment and prevent core damage. Because of these prior operator errors and since the lower personnel lock is already open at the onset of core damage, it is also assumed that the equipment hatch is also open. The quantification for this case is:

Branch 1:	nOCnt-BCD	0.0
Branch 2:	OCnt-BCD	1.0

Case 3: This case includes those accidents initiated by a loss of offsite power. Without offsite power the containment cannot be closed. The quantification for this case is:

Branch 1:	nOCnt-BCD	0.0
Branch 2:	OCnt-BCD	1.0

Case 3: This case includes those accidents that were estimated to progress to core damage in less than 5 hours which, therefore, precluded successful closure of the containment. The quantification for this case is:

Branch 1:	nOCnt-BCD	0.0
Branch 2:	OCnt-BCD	1.0

Case 4: This case includes those accidents in which the containment is open at the start of the accident, offsite power is available, and core damage does not occur for at least 5 hours. Furthermore, the operators have not committed pervious errors that would preclude closure of the containment. In these accidents it is possible that the operators will close the containment. This case is sampled, the distribution for the probability that the operators fail to close the containment was developed in the HRA analysis and is discussed in Appendix B.3. The quantification (mean values) for this case is:

Branch 1:	nOCnt-BCD	0.898	
Branch 2:	OCnt-BCD	0.102	Lognormal distribution with a mean of 0.102 and an error factor of 5.

Question 20. Does the auxiliary building fail before core damage?
Number of Branches: 2
Number of Cases: 4
Number of Cases Sampled: None

The branches for this question are:

1. nOAux-BCD The auxiliary building fails before the onset of core damage.
2. OAux-BCD The auxiliary building does not fail before the onset of core damage.

This question determines the status of the auxiliary building integrity at the onset of core damage. The Grand Gulf plant utilizes a secondary containment that completely encloses the primary containment. The secondary containment consists of the auxiliary building and the enclosure building. The auxiliary building, which contains safety systems, fuel storage and shipping equipment and necessary auxiliary support systems, surrounds the lower portions of the containment. The containment personnel locks and the containment equipment hatch allow direct access from the containment to the auxiliary building. Thus, if the containment access penetrations are open, steam and radioactive releases from the containment will pass directly into the auxiliary building. It is assumed that the pressure retaining capability of the auxiliary building is 5 psig (assumed strength of doors for industrial buildings). The enclosure building encloses the upper portion of the containment above the auxiliary building roof and provides a boundary for the standby gas treatment system, which maintains a negative pressure in the volume between the containment and enclosure building to ensure that leakage of radioactive materials from the containment is filtered prior to release to the environment. The enclosure building has virtually no pressure retaining capability and is essentially isolated from the auxiliary building.

Case 1: This case includes those accidents in which there is an interfacing systems LOCA in the auxiliary building. In these accidents, the low pressure components of the shutdown cooling system (or ADHRS) are not isolated from the primary system. Following the loss of core cooling, the reactor vessel pressurizes resulting in a LOCA in the shutdown cooling system in the auxiliary building. The steam released from the vessel immediately following the LOCA pressurizes and fails the auxiliary building. The quantification for this case is:

Branch 1:	nOAux-BCD	0.0
Branch 2:	OAux-BCD	1.0

Case 2: This case includes those accidents in which the operators fail the close the MSIVs. Steam generated in the vessel following the loss of core cooling is transported to the condenser, via the main steam lines, and released into the turbine building. The steam released from the reactor vessel pressurizes and fails the turbine building. Since the turbine building and the auxiliary building are similar in that they are large volumes with little pressure retaining capability, releases into the turbine building are treated as though they were released into the auxiliary building and a separate analysis of the turbine building is not performed. The quantification for this case is:

Branch 1:	nOAux-BCD	0.0
Branch 2:	OAux-BCD	1.0

Case 3: This case includes those accidents in which the containment access penetrations were not open at the start of the accident, the MSIVs were not opened and there was not an interfacing systems LOCA in the auxiliary building. Thus, the auxiliary building is effectively isolated from the containment and does not pressurize. The quantification for this case is:

Branch 1:	nOAux-BCD	1.0
Branch 2:	OAux-BCD	0.0

Case 4: This case includes those accidents in which the containment access penetrations were open during core damage, however, the MSIVs are closed and an interfacing systems LOCA does not occur in the auxiliary building. The LOCAs that are included in this case occur when the reactor vessel is depressurized and, therefore, there is not a large release of steam that would threaten the auxiliary building (i.e., most of the reactor vessel inventory drains out of the vessel and is not available to be converted to steam). In the remaining accidents the steam that is generated in the vessel is released to the suppression pool where it is condensed. For the cases with an open reactor head vent, there is an insufficient amount of steam released via the vent to threaten the auxiliary building before the onset of core damage.

Branch 1:	nOAux-BCD	1.0
Branch 2:	OAux-BCD	0.0

Question 21. What is the status of the drywell before core damage?
Number of Branches: 2
Number of Cases: 1
Number of Cases Sampled: None

The branches for this question are:

1. Cls-DW-BCD Both the drywell personnel lock and the containment equipment hatch are closed before the

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onset of core damage.
2. Op-DW-BCD Either the drywell personnel lock and/or the drywell equipment hatch are open before the onset of core damage.

This question determines the status of the drywell integrity at the onset of core damage. The status of drywell integrity is an important feature of the accident because as long as the drywell's integrity is maintained, releases (i.e., steam and radioactive material) in the drywell are forced through the suppression pool. The suppression pool condenses the steam and thereby reduces the pressure load in the containment. Also, the suppression pool is an effective device for removing radioactive material from the drywell atmosphere. There are, however, no Technical Specification requirements for drywell integrity while the plant is in POS 5. A review of the Grand Gulf refueling outage critiques indicated that the drywell personnel lock is generally open during POS 5 and that the drywell equipment hatch is also removed during POS 5. The drywell head is also detached from the drywell during POS 5. Discussion with plant personnel indicated that during an accident the operators will be more concerned with closure of the containment instead of closure of the drywell. Since reattachment of the drywell equipment hatch and the drywell head are time consuming tasks and since closure of the drywell will not be the operators primary concern, in this analysis, it is assumed that the drywell will be open for the duration of the accident. The quantification for this question is:

Branch 1:	Cls-DW-BCD	0.0
Branch 2:	Op-DW-BCD	1.0

Question 22. Do the operators turn on the HIS before core damage?
Number of Branches: 2
Number of Cases: 2
Number of Cases Sampled: 1

The branches for this question are:

1. nHIS-BCD The operators do not turn on the Hydrogen Ignition System (HIS) before core damage.
2. HIS-BCD The operators do turn on the HIS before the onset of core damage.

This question determines the status of Hydrogen Ignition System (HIS) at the onset of core damage. The Grand Gulf containment utilizes a hydrogen ignition system (HIS) to control the accumulation of hydrogen during accident conditions. In the core region there is an abundant supply of zirconium (i.e., fuel cladding, channel boxes) which, at the elevated temperatures typical of core damage accidents, readily reacts with steam to produce hydrogen. The function of the HIS is to prevent the buildup of large quantities of hydrogen inside the containment during accident conditions. This is accomplished by igniting, via a spark, small amounts of hydrogen before it has had a chance to accumulate. The HIS consists of 90 General Motors ac division glow plugs (Model 7G), 45 powered by each ac power division. The HIS is manually actuated. Igniters are located throughout the containment and drywell volumes. The Grand Gulf Emergency Procedures direct the operators to enter the Hydrogen Control section of the Containment Control Procedure (GGNS EP-3) if the water level drops below the top of the active fuel (TAF) or if the water level in the reactor vessel cannot be determined. The Hydrogen Control Procedures direct the operators to activate the HIS if the drywell hydrogen concentration is less than 9% and the containment hydrogen concentration is below the Hydrogen Deflagration Overpressure Limit (HDOL). This issue was addressed in the Human Reliability Analysis (HRA) that was performed for the Level 2 analysis and is discussed further in Appendix B.3.

Case 1: This case includes those accidents initiated by a loss of offsite power that result in a station blackout. In station blackout scenarios where the reactor vessel water level has reach TAF, the operators would be unable to determine hydrogen concentrations and, therefore, could not determine whether or not the hydrogen concentration could be maintained below the HDOL. Given this situation, the emergency procedure guides the operators to "secure and prevent" operation of the igniters. Thus, it is assumed that the operators would not turn the HIS to the on position until they had recovered power and could determine the hydrogen levels (see Question 31). The quantification for this case is:

Branch 1:	nHIS-BCD	1.0
Branch 2:	HIS-BCD	0.0

Case 2: This case includes those accidents in which emergency ac power is available. The water level in the reactor vessel has dropped below TAF and very little if any hydrogen has been produced. In this situation, Hydrogen Control section of the Containment Control procedures would direct the operators to turn the HIS to the on position. This case is sampled; the distribution for the probability that the operators fail to turn on the HIS was developed in the HRA analysis and is discussed in Appendix B.3. The quantification (mean values) for this case is:

Branch 1:	nHIS-BCD	0.054	Lognormal distribution with a mean of 0.054 and an error factor of 5.
Branch 2:	HIS-BCD	0.946	

Question 23. Do the station batteries deplete during core damage?
Number of Branches: 3
Number of Cases: 5
Number of Cases Sampled: None

The branches for this question are:

1.	nDC-CD	DC power is not available during core damage.
2.	DC-ECD	DC power available during the early portion (i.e., first 1.5 hours) of core damage.
3.	DC-LCD	DC power is available during core damage.

This question determines the availability of dc power during core damage. The availability of dc power is important for the following two reasons: (1) dc power is required to keep the SRVs open and maintain the vessel at low pressure, and (2) dc power is required to restore offsite power to the plant (see Question 24). Restoration of offsite power is considered during two different portions of the core damage time regime: (1) during the first 1.5 hours of core damage and (2) from 1.5 hours after the onset of core damage to vessel failure (i.e., vessel failure is defined as the end of the core damage time regime). Since the restoration of offsite power depends of the availability of dc power, these two time regimes were also used to determine the availability of dc power. The failure probabilities for the time windows is based on a distribution that was developed for the NUREG-1150 Grand Gulf plant analysis that models the failure probability of the station batteries versus time for SBO sequences [Wheeler, et al., 1989]. The failure probabilities used in this analysis are conditional on dc power being available at the onset of core damage.

Case 1: This case includes those accidents in which offsite and onsite power are available. With ac power available, the battery chargers supply the necessary dc power and battery depletion is not an issue. For the PDSs analyzed in this study, there were no failures of the dc bus and, therefore, with ac power available, dc power is also assured. The quantification for this case is:

Branch 1:	nDC-CD	0.0
Branch 2:	DC-ECD	0.0
Branch 3:	DC-LCD	1.0

Case 2: This case includes those accidents in which dc power was not available before core damage and, therefore, is not available during core damage (no credit is given for recovery of the station batteries). The quantification for this case is:

Branch 1:	nDC-CD	1.0
Branch 2:	DC-ECD	0.0
Branch 3:	DC-LCD	0.0

Case 3: This case includes those accidents initiated by a loss of offsite power that result in a station blackout during Time Window 1. In these accidents, dc power is available at the onset of core damage. For this case the following times are used for core damage and vessel breach respectively, 3.5 hours and 14.4 hours. With these times, the first time period starts at 3.5 hours and ends at 5.5 hours; the second time period starts at 5.5 hours and ends at 14.4 hours. Therefore, the value for Branch 2 is probability that dc power is available at 5.5 hours given that it was available at 3.5 hours. Similarly, the value for Branch 3 is the probability that dc power is available between 5.5 hours and 14.4 hours given that it was available at 3.5 hours. The quantification for this case is:

Branch 1:	nDC-CD	0.011
Branch 2:	DC-ECD	0.242
Branch 3:	DC-LCD	0.747

Case 4: This case includes those accidents initiated by a loss of offsite power that result in a station blackout during Time Window 2. In these accidents, dc power is available at the onset of core damage. For this case the following times are used for core damage and vessel breach respectively, 5.5 hours and 12.6 hours. With these times, the first time period starts at 5.5 hours and ends at 7 hours, the second time period starts at 7 hours and ends at 12.6 hours. Therefore, the value for Branch 2 is probability that dc power is available at 7 hours given that it was available at 5.5 hours. Similarly, the value for Branch 3 is the probability that dc power is available between 7 hours and 12.6 hours given that it was available at 3.5 hours. The quantification for this case is:

Branch 1:	nDC-CD	0.015
Branch 2:	DC-ECD	0.103
Branch 3:	DC-LCD	0.882

Case 5: This case is not used.

Question 24. Is offsite power restored during core damage?
Number of Branches: 4
Number of Cases: 7
Number of Cases Sampled: 4

The branches for this question are:

1.	nAC-CD	Neither offsite nor onsite power is available during core damage.
2.	OSP-CD	Only offsite power is available during core damage (ac emergency bus failed). (Note, this situation does not occur in this analysis)
3.	EAC-CD	AC power available during the early portion (i.e., first 1.5 hours) of core damage.
4.	LAC-CD	AC power is available during all of the core damage process.

This question determines the availability of ac power during core damage. The availability of ac power is important because it will determine which systems can be used to mitigate the accident (e.g., core cooling systems, containment heat removal systems, Hydrogen Ignition System, Containment Vent System). Restoration of offsite power is considered during two different portions of the core damage time regime: (1) during the first 1.5 hours of core damage and (2) from 1.5 hours after the onset of core damage to vessel failure (i.e., vessel failure is defined as the end of the core damage time regime). The first tin e regime corresponds the time available to restore coolant the core and arrest the core damage process. The duration o; this first time regime is 1.5 hours which corresponds to the point in the core damage process where approximately 10% of the core is damaged. It was estimated that the restoration of coolant to the core with only 10% of the core damage will prevent any further damage. After this point in the core damage progression (i.e., the second time regime) there is a rapid escalation in the amount of core damage and, therefore, it is assumed that beyond this point the likelihood of core damage arrest is very small (see Question 41). It is important to also consider the second time regime because the availability of ac power is also important after the failure of the vessel (i.e., containment venting and containment heat removal).

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The probability of recovering offsite power during a given time period is determined by sampling from a set of distributions for power recovery [Iman and Hora, 1988] (also see Volume 2 of this report). These distributions reflect the type of electrical switchyard at Grand Gulf, as explained in NUREG-1032 [Baranowsky, 1985]. To get ac power to the safety systems, not only does ac power have to be restored to the site, but dc power must be available as well. DC power is required for circuit breaker control power; once the station batteries have been depleted, it is very difficult to get ac power back to the safety systems. Although the circuit breakers can be moved manually, this procedure is very complicated and slow. Thus, for the time frame considered in this analysis, it is assumed that once dc power is lost, ac power cannot be recovered. The generation of the power recovery curves used in this analysis is discussed in Appendix G of Volume 2, Part 2 of this report.

Case 1: This case includes those accidents in which offsite and onsite power are available at the start of the accident and, therefore, ac power is still available. The quantification for this case is:

Branch 1:	nAC-CD	0.0
Branch 2:	OSP-CD	0.0
Branch 3:	EAC-CD	1.0
Branch 4:	LAC-CD	0.0

Case 2: This case includes those accidents in which dc power is not available. The lack of dc power implies that both offsite an onsite ac power are unavailable. Without dc power, offsite power cannot be recovered. Furthermore, in this analysis, no credit is given for recovery of the emergency diesel generator during the core damage process. Therefore, in this case all ac power is unavailable. The quantification for this case is:

Branch 1:	nAC-CD	1.0
Branch 2:	OSP-CD	0.0
Branch 3:	EAC-CD	0.0
Branch 4:	LAC-CD	0.0

Case 3: This case includes those accidents initiated by a loss of offsite power that result in a station blackout during Time Window 1. DC power is only available during the early time period and, therefore, ac power can only be restored during this time period (i.e., in the late core damage time period dc power is not available and, therefore, ac power cannot be restore during the late time period). For this case the following times are used for core damage and vessel breach respectively, 3.5 hours and 14.4 hours. With these times, the early core damage time period starts at 3.5 hours and ends at 5.5 hours; the late core damage time period starts at 5.5 hours and ends at 14.4 hours. The value for Branch 3 is the probability of recovering ac power during the early core damage time period, given that offsite power was not available at the onset of core damage. The quantification for this case is:

Branch 1:	nAC-CD	0.69	
Branch 2:	OSP-CD	0.0	
Branch 3:	EAC-CD	0.31	Power recovery distribution.
Branch 4:	LAC-CD	0.0	

Case 4: This case includes those accidents initiated by a loss of offsite power that result in a station blackout during Time Window 1. DC power is available during the entire core damage process and, therefore, ac power can be restored during either the early or the late core damage time period. For this case the following times are used for core damage and vessel breach respectively, 3.5 hours and 14.4 hours. With these times, the early core damage time period starts at 3.5 hours and ends at 5.5 hours; the late core damage time period starts at 5.5 hours and ends at 14.4 hours. The value for Branch 3 is the probability of recovering ac power during the early core damage time period, given that offsite power was not available at the onset of core damage. The value for Branch 4 is the probability of recovering ac power during the late core damage time period, given that offsite power was not available at the onset of core damage. The quantification for this case is:

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Branch 1:	nAC-CD	0.16	
Branch 2:	OSP-CD	0.0	
Branch 3:	EAC-CD	0.31	Power recovery distribution.
Branch 4:	LAC-CD	0.53	Power recovery distribution.

Case 5: This case includes those accidents initiated by a loss of offsite power that result in a station blackout during Time Window 2. DC power is only available during the early time period and, therefore, ac power can only be restored during this time period (i.e., in the late core damage time period dc power is not available and, therefore, ac power cannot be restore during the late time period). For this case the following times are used for core damage and vessel breach respectively, 5.5 hours and 12.6 hours. With these times, the early core damage time period starts at 5.5 hours and ends at 7 hours; the late core damage time period starts at 7 hours and ends at 12.6 hours. The value for Branch 3 is the probability of recovering ac power during the early core damage time period, given that offsite power was not available at the onset of core damage. The quantification for this case is:

Branch 1:	nAC-CD	0.74	
Branch 2:	OSP-CD	0.0	
Branch 3:	EAC-CD	0.26	Power recovery distribution.
Branch 4:	LAC-CD	0.0	

Case 6: This case includes those accidents initiated by a loss of offsite power that result in a station blackout during Time Window 2. DC power is available during the entire core damage process and, therefore, ac power can be restored during either the early or the late core damage time period. For this case the following times are used for core damage and vessel breach respectively, 5.5 hours and 12.6 hours. With these times, the early core damage time period starts at 5.5 hours and ends at 7 hours; the late core damage time period starts at 7 hours and ends at 12.6 hours. The value for Branch 3 is the probability of recovering ac power during the early core damage time period, given that offsite power was not available at the onset of core damage. The value for Branch 4 is the probability of recovering ac power during the late core damage time period, given that offsite power was not available at the onset of core damage. The quantification for this case is:

Branch 1:	nAC-CD	0.11	
Branch 2:	OSP-CD	0.0	
Branch 3:	EAC-CD	0.26	Power recovery distribution.
Branch 4:	LAC-CD	0.43	Power recovery distribution.

Case 7: This case is not used.

Question 25. Is the RPV isolated during core damage?
Number of Branches: 2
Number of Cases: 5
Number of Cases Sampled. 1

The branches for this question are:

1. Op-RPV-CD — A breach in the reactor vessel integrity is not isolated during core damage.
2. Cls-RPV-CD — The reactor vessel integrity is maintained during core damage.

This question determines the status of the vessel integrity during core damage. The status of vessel integrity is important because it impacts: (1) the pressure in the vessel, (2) the release path of steam and radionuclides during core damage and (3) the possibility of arresting the core damage process. This question is primarily concerned with station blackout accidents that result in a break in the SDC system and whether the isolation valves (FOO8 and FOO9) in the SDC system isolate when offsite power is recovered. If the break is isolated and core coolant is

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restored to the vessel, it is possible that the core damage process will be arrested and vessel failure avoided. This question does not address the status of the SRVs or the reactor head vent; these issues are addressed in other questions.

Case 1: This case includes those accidents that are initiated by a LOCA. It is assumed that the LOCA cannot be isolated and, therefore, the RPV remains breached. The quantification for this case is:

Branch 1:	Op-RPV-CD	1.0
Branch 2:	Cls-RPV-CD	0.0

Case 2: This case includes those accidents that have an open MSIV before core damage. In these accidents the operators failed to recognize that an MSIV is open or were unable for some reason to close the MSIV. Since there was ample time to close the MSIV before cor mage and yet it wasn't closed, no credit is given for closing the MSIV during core damage. Thus, t emains breached. The quantification for this case is:

Branch 1:	Op-RPV-CD	1.0
Branch 2:	Cls-RPV-CD	0.0

Case 3: This case includes those accidents that were initiated by a loss of offsite power that resulted in a station blackout. In these accidents, the isolation valves on the SDC syster ere open when power was lost and since these are ac power valves, they remain open following the loss of . Following the loss of core cooling, the vessel pressurizes and ruptures the SDC system outside the co resulting in an interfacing systems LOCA. This case includes those accidents in which ac power is recovered during the early time regime of core damage. The valves should automatically close following the recovery of ac power. If the valves do not automatically close, the operators can close the valves. The issue being addressed is whether the valves were sufficiently damaged during the early phases of core damage that the resulting damage would preclude closure of the valve. Since ac power is recovered early in the core damage process it is expected that the environment that the valves would experience would not be severe and, therefore, it is expected that the valves would close. However, there is uncertainty regarding the performance of the valves (and the associated control logic) in this environment. Therefore, a maximum entropy distribution was u characterize the uncertainty in the probability that the valve will close. The quantification for thi is:

Branch 1:	Op-RPV-CD	0.1	
Branch 2:	Cls-RPV-CD	0.9	Maximum Entropy with an Lower Bound of 0.5, an Upper Bound of 1.0 and a Mean of 0.9,

Case 4: This case includes accidents that are similar to the accidents addressed by the previous case except in this case offsite power is not recovered. Without ac power the break cannot be isolated and, therefore, the vessel remains breached. This case also includes those accidents in which ac power is recovered late during the core damage process. In these accidents, compared to accidents in which ac power is restored early during core damage, the isolation valves will experience a more severe environment and, therefore, their performance is even less certain. Furthermore, the recovery of power can occur any time during the late time period of core damage. The closer in time to vessel failure that power is recovered, the less important the closure of the valves becomes since most of the in-vessel releases will have already escaped the vessel via the break. Following vessel failure, the isolation of the break is not particularly important because the containment is open in these accidents and the releases will escape into the auxiliary building in either case. The quantification for this case is:

Branch 1:	Op-RPV-CD	1.0
Branch 2:	Cls-RPV-CD	0.0

Case 5: This case includes those accidents in which the vessel was not breached prior to core damage. Nothing has happened to this point in the accident that would change that condition. The quantification for this case is:

Branch 1:	Op-RPV-CD	0.0
Branch 2:	Cls-RPV-CD	1.0

Question 26. Do the operators initiate the containment sprays during core damage?
Number of Branches: 2
Number of Cases: 7
Number of Cases Sampled: 0

The branches for this question are:

1. nCS-CD — The containment sprays are not used during core damage.
2. CS-CD — The containment sprays are used during core damage.

This question determines whether the containment sprays (CS) were used during core damage. The use of containment sprays is important because it: (1) can reduce the pressure in the containment, and (2) reduces the amount of airborne radioactive material that is present in the containment atmosphere. The Grand Gulf Emergency Procedures (GGNS EP-3) direct the operators to initiate containment sprays if the containment pressure is above 2.2 psig.

Case 1: This case includes those accidents in which the CS are not available. The quantification for this case is:

Branch 1:	nCS-CD	1.0
Branch 2:	CS-CD	0.0

Case 2: This case includes those accidents in which the CS are available or recoverable yet the containment is open and the CS system is not aligned to automatically start. In this analysis, it is assumed that if the containment is open and the containment sprays are not aligned to automatically start, the containment sprays will not be used because containment pressure control is not an issue. In this situation, it is assumed that the operators primary concern will be to restore core cooling and they will used any available systems to provide coolant to the core. The quantification for this case is:

Branch 1:	nCS-CD	1.0
Branch 2:	CS-CD	0.0

Case 3: This case is similar to the previous case except that the containment is closed, however, the MSIVs are open or there is an interfacing systems LOCA. Again, the pressure in the containment is not an issue in these accidents since all of the releases (both steam and radioactive material) during core damage will bypass the containment. The quantification for this case is:

Branch 1:	nCS-CD	1.0
Branch 2:	CS-CD	0.0

Case 4: This case includes those accidents in which LPCI is align to the SDC system yet was not used to provide makeup to the core. These previous operator errors preclude its used in the CS mode of operation. The quantification for this case is:

Branch 1:	nCS-CD	1.0
Branch 2:	CS-CD	0.0

Case 5: This case includes those accidents in which CSs are recoverable, however, emergency ac power (i.e., either station blackout or the ac buses have failed) is not available and, therefore, the CS system cannot be used.

The quantification for this case is:

Branch 1:	nCS-CD	1.0
Branch 2:	CS-CD	0.0

Case 6: This case includes those accidents in which CSs are recoverable and ac power is restored during core damage. In this case the containment is closed and the reactor vessel integrity is maintained. The Grand Gulf Emergency Procedures will direct the operators to initiate the containment sprays. It is likely the operators will follow the procedures and initiate CS; a small probability is assigned to failure of the containment sprays to account for failure of the system and/or failure of the operators to follow the procedures. The quantification for this case is:

Branch 1:	nCS-CD	0.01
Branch 2:	CS-CD	0.99

Case 7: This case includes those accidents in which CSs are available. In this case the containment is closed and the reactor vessel integrity is maintained. The Grand Gulf Emergency Procedures will direct the operators to initiate the containment sprays. It is likely the operators will follow the procedures and initiate CS; a small probability is assigned to the failure of the containment sprays to account for failure of the system and/or failure of the operators to follow the procedures. The quantification for this case is:

Branch 1:	nCS-CD	0.01
Branch 2:	CS-CD	0.99

Question 27. Do the operators depressurize the RPV during core damage?
Number of Branches: 2
Number of Cases: 5
Number of Cases Sampled: 2

The branches for this question are:

1. RPV-HiP-CD — The pressure in the reactor vessel is at system pressure (i.e., approx. 1000 psia) during core damage.
2. RPV-LoP-CD — The reactor vessel remains at low pressure (< 400 psia) during core damage.

This question determines the pressure in the reactor vessel during core damage. The pressure in the vessel is important because it: (1) determines which systems can be used to restore coolant to the core, (2) affects the amount of hydrogen produced during core damage, (3) affects the amount of radioactive material that is retained in the vessel, (4) affects the probability of in-vessel steam explosions, (5) affects the dispersal core debris during core damage which in turn affects the magnitude of the loads that accompany vessel breach and also affects the coolability of the ejected core debris (i.e., assuming water is available to cool the core debris).

Case 1: This case includes those accidents in which the reactor vessel is breached by either a LOCA, an unisolated interfacing systems LOCA, or an open MSIV. Since the reactor vessel is breached, it is at low pressure. The quantification for this case is:

Branch 1:	RPV-HiP-CD	0.0
Branch 2:	RPV-LoP-CD	1.0

Case 2: This case includes those accidents in which the reactor vessel integrity is maintained (i.e., no breach) and dc power is not available. Since the SRVs require dc power to remain open, without dc power the SRVs will close and the vessel will pressurize. An open reactor head vent will not prevent the vessel from pressurizing. The quantification for this case is:

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Branch 1:	RPV-HiP-CD	1.0
Branch 2:	RPV-LoP-CD	0.0

Case 3: This case includes those accidents that were initiated by a loss of offsite power that resulted in a station blackout followed by a break in the SDC systems outside the containment. In these accidents, ac power is restored during core damage and the break is isolated. With the break isolated, the vessel will pressurize. This case determines whether the operators open at least two SRVs such that low pressure injection systems, if available, can be used to cool the core. The quantification for this case is the same as Case 5. While this case is somewhat different from Case 5 in that an interfacing systems LOCA occurred and it was subsequently isolated, the vessel will still pressurize and conditions are still available that would lead the operators to depressurize the vessel. Thus, it is judged that within the resolution of this analysis, the use of Case 5 quantification for this case is appropriate. The quantification for this case is:

Branch 1:	RPV-HiP-CD	0.054	Bounded Lognormal distribution with a mean of 0.054 and an error factor of 5.
Branch 2:	RPV-LoP-CD	0.946	

Case 4: This case includes those accidents in which at least two SRVs were open before core damage. Since dc power is available in these accidents and there would be no reason for the operators to close the SRVs, the SRVs remain open during core damage. With the SRVs open the reactor vessel remains at low pressure. The quantification for this case is:

Branch 1:	RPV-HiP-CD	0.0
Branch 2:	RPV-LoP-CD	1.0

Case 5: This case includes those accidents in which the SRVs were not opened before core damage resulting in the reactor vessel being at high pressure at the onset of core damage. In some of these scenarios, the Level 1 model did not address the issue of vessel depressurization because depressurization of the vessel would not preclude core damage (e.g., if no injection systems were available). In any case, based on the conditions present in the plant during the core damage process, the Grand Gulf Emergency Procedures, would indicate that emergency depressurization of the reactor vessel is required. This case is sampled, the distribution for the probability that the operators fail to depressurize the vessel was developed in the HRA analysis and is discussed in Appendix B.3. The quantification (mean values) for this case is:

Branch 1:	RPV-HiP-CD	0.054	Bounded Lognormal distribution with a mean of 0.054 and an error factor of 5.
Branch 2:	RPV-LoP-CD	0.946	

Question 28. What is the status of the SRV vacuum breakers during core damage?
Number of Branches: 2
Number of Cases: 3
Number of Cases Sampled: 1

The branches for this question are:

1. Op-SRV-Bkr — A vacuum breaker on a SRV tailpipe sticks open and remains open during core damage.
2. Cls-SRV-Bkr — The vacuum breakers on the SRV tailpipes remain closed during core damage.

This question determines whether any of the vacuum breakers on the SRV tailpipes stick open and remain open during core damage. The status of the SRV tailpipe vacuum breakers is important because if a vacuum breaker is open, a portion of the release from the reactor vessel will enter the drywell and, thus, bypass the suppression pool. The suppression pool is an effective device for condensing steam and trapping radioactive material that is released from the vessel. Tailpipe vacuum breakers will open after the associated SRV discharges steam through the tailpipe into the suppression pool. When the steam in the tailpipe condenses on the pipe walls, a vacuum is formed in the tailpipe. The vacuum breaker is designed to relieve this vacuum and thereby prevent suppression pool water from being drawn up into the tailpipe. In this analysis, a stuck open tailpipe vacuum breaker is significant only if it is the vacuum breaker on the tailpipe for an SRV that is expected to be open after core damage occurs. Thus, the cases below consider which vacuum breakers are challenged by the sequence during the boil-down phase of the accident. This question reflects only significant vacuum breakers sticking open (i.e., one that will result in fission product releases bypassing the suppression pool).

Case 1: This case includes those accidents in which the reactor vessel was depressurized before core damage. In this situation the SRVs are not cycled repeatedly. Since the SRVs and their associated vacuum breakers are not cycled, it is very unlikely that a vacuum breaker will stick open. The quantification for this case is:

Branch 1:	Op-SRV-Bkr	0.00
Branch 2:	Cls-SRV-Bkr	1.00

Case 2: This case includes those accidents in which the reactor is at high pressure before core damage and the reactor vessel head vent is open. While the head vent will not prevent the vessel from pressurizing, it will significantly limit the number of cycles of the SRVs. Since the SRVs and their associated vacuum breakers are not cycled repeated, it is very unlikely that a vacuum breaker will stick open. The quantification for this case is:

Branch 1:	Op-SRV-Bkr	0.00
Branch 2:	Cls-SRV-Bkr	1.00

Case 3: This case includes those accidents in which the reactor is at high pressure before core damage and the operators closed the reactor vessel head vent. The SRVs are cycled to maintain the RPV at the SRV setpoints (i.e., near system pressure). Since the SRVs are repeatedly opened and closed, the SRV tailpipe vacuum breakers are also demanded to open and close a number of times. In this case the probability that the vacuum breakers stick open is not negligible. The probability that a vacuum breaker sticks open in based on the distribution used in the NUREG-1150 Grand Gulf plant analysis [Brown, et al., 1990]. The quantification for this case is:

Branch 1:	Op-SRV-Bkr	0.25	Uniform Distribution between 0.01 and 0.5
Branch 2:	Cls-SRV-Bkr	0.75	

Question 29. Is core cooling restored during core damage?
Number of Branches: 3
Number of Cases: 8
Number of Cases Sampled: 0

The branches for this question are:

1. E-CorCool — Core cooling is restored early during the core damage process.
2. L-CorCool — Core cooling is restored late during the core damage process.
3. n-CorCool — Core cooling is not restored during the core damage process.

This question determines whether core cooling is restored to the reactor vessel during the core damage process. When core cooling is restored is important because it affects (1) whether the core damage process can be arrested before the vessel fails, (2) the amount of hydrogen produced, and (3) the amount of core debris produced. The core damage time regime is divided into two segments or time regimes: (1) during the first 1.5 hours of core damage and (2) from 1.5 hours after the onset of core damage to vessel failure (i.e., vessel failure is defined as the end of the core damage time regime). The first time regime corresponds the time available to restore coolant the core and arrest the core damage process. The duration of this first time regime is 1.5 hours which corresponds to the point in the core damage process where approximately 10% of the core is damaged. It was estimated that the restoration of coolant to the core with only 10% of the core damage will prevent any further damage. After this point in the core damage progression (i.e., the second time regime) there is a rapid escalation in the amount of core damage and, therefore, it is assumed that beyond this point the likelihood of core damage arrest is very small (see Question 41).

Case 1: This case includes those accidents in which high pressure injection is available (i.e., HPCS). This case never occurs in this analysis, however, it is included for the sake of completeness. The quantification for this case is:

Branch 1:	E-CorCool	1.00
Branch 2:	L-CorCool	0.00
Branch 3:	n-CorCool	0.00

Case 2: This case includes those accidents in which high pressure injection is recoverable (i.e., HPCS) and ac power is restored in the early time segment of the core damage process. It is assumed that HPCS will automatically inject into the reactor vessel. This case never occurs in this analysis, however, it is included for the sake of completeness. The quantification for this case is:

Branch 1:	E-CorCool	1.00
Branch 2:	L-CorCool	0.00
Branch 3:	n-CorCool	0.00

Case 3: This case includes those accidents in which high pressure injection is recoverable (i.e., HPCS) and ac power is restored in the late time segment of the core damage process. It is assumed that HPCS will automatically inject into the reactor vessel. This case never occurs in this analysis, however, it is included for the sake of completeness. The quantification for this case is:

Branch 1:	E-CorCool	0.00
Branch 2:	L-CorCool	1.00
Branch 3:	n-CorCool	0.00

Case 4: This case includes those accidents in which the vessel is at high pressure during the core damage process and no high pressure injection systems are available or recoverable (i.e., HPCS). The quantification for this case is:

Branch 1:	E-CorCool	0.00
Branch 2:	L-CorCool	0.00
Branch 3:	n-CorCool	1.00

Case 5: This case includes those accidents in which low pressure injection systems are available, although they are not aligned to automatically inject into the reactor vessel, and the operators fail to use these systems to cool the core. Since credit was given for using these systems in the Level 1 analysis and nothing has happen during core damage that would make additional systems available, it is assumed that the operators are unable to provide coolant to the core. The quantification for this case is:

Branch 1:	E-CorCool	0.00
Branch 2:	L-CorCool	0.00
Branch 3:	n-CorCool	1.00

Case 6: This case includes those accidents in which at least one low pressure injection system is recoverable and ac power is restored in the early time segment of the core damage process. If the accident involves a LOCA in an interfacing system, the break has been isolated. It is assumed that the low pressure injection system will automatically inject into the reactor vessel following the recovery of ac power. The quantification

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for this case is:

Branch 1:	E-CorCool	1.00
Branch 2:	L-CorCool	0.00
Branch 3:	n-CorCool	0.00

Case 7: This case includes those accidents in which at least one low pressure injection system is recoverable and ac power is restored in the late time segment of the core damage process. If the accident involves a LOCA in an interfacing system, the break has been isolated. It is assumed that the low pressure injection system will automatically inject into the reactor vessel following the recovery of ac power. The quantification for this case is:

Branch 1:	E-CorCool	0.00
Branch 2:	L-CorCool	1.00
Branch 3:	n-CorCool	0.00

Case 8: This case includes those accidents in which either there are no injection systems available, ac power is not available, or the accident involves a LOCA in an interfacing system (i.e., SDC). In the latter case it is assumed that if injection was provided to the reactor vessel, all of the suppression pool water would be pumped out the break into the auxiliary building. The resulting flood in the auxiliary building would then fail all core cooling and containment cooling systems and the accident would proceed as though injection was never available. The quantification for this case is:

Branch 1:	E-CorCool	0.00
Branch 2:	L-CorCool	0.00
Branch 3:	n-CorCool	1.00

Question 30. What is the peak hydrogen concentration in the containment during core damage?
Number of Branches: 5
Number of Cases: 5
Number of Cases Sampled: 0

The branches for this question are:

1.	H2<4	The concentration of hydrogen in the containment is less than 4% by volume.
2.	H2<8	The concentration of H_2 in the containment is between 4% and 8% by volume.
3.	H2<12	The concentration of H_2 in the containment is between 8% and 12% by volume.
4.	H2<16	The concentration of H_2 in the containment is between 12% and 16% by volume.
5.	H2>16	The concentration of hydrogen in the containment is greater than 16% by volume.

This question determines the concentration of hydrogen in the containment during the core damage process. The concentration of hydrogen is important because it will determine the likelihood of a hydrogen combustion event and also the magnitude of the resulting load on the containment. This question is also used to determine the amount of zirconium that was oxidized in the vessel (see the next question). The ranges for the concentrations were selected to correspond to the ranges used for the loads distributions that were developed during the NUREG-1150 project [Harper, et al., 1991].

Case 1: This case includes those accidents in which core cooling is restored early during the core damage process. In this case, core cooling is restored before a significant fraction of the core is damaged which thereby limits the amount of hydrogen produced. The quantification for this case is:

Branch 1:	H2<4	1.00
Branch 2:	H2<8	0.00

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Branch 3:	H2<12	0.00
Branch 4:	H2<16	0.00
Branch 5:	H2>16	0.00

Case 2: This case includes those accidents in which the core damage process occurs with the vessel at high pressure and core cooling is either not restored or restored during the late time regime of the core damage process. The containment is closed and there are no open MSIVs and a LOCA in an interfacing system does not occur. The amount of hydrogen produced is based on the zirconium oxidation curves developed in the NUREG-1150 project [Harper, et al., 1990]. The concentrations were determined based on the assumption that the containment is closed and that the drywell and the containment volumes are well mixed. The quantification for this case is:

Branch 1:	H2<4	0.014
Branch 2:	H2<8	0.095
Branch 3:	H2<12	0.127
Branch 4:	H2<16	0.207
Branch 5:	H2>16	0.557

Case 3: This case includes those accidents in which the core damage process occurs with the vessel at low pressure and core cooling is either not restored or restored during the late time regime of the core damage process. The containment is closed and there are no open MSIVs and a LOCA in an interfacing system does not occur. The amount of hydrogen produced is based on the zirconium oxidation curves developed in the NUREG-1150 project [Harper, et al., 1990]. The concentrations were determined based on the assumption that the containment is closed and that the drywell and the containment volumes are well mixed. The quantification for this case is:

Branch 1:	H2<4	0.010
Branch 2:	H2<8	0.060
Branch 3:	H2<12	0.140
Branch 4:	H2<16	0.205
Branch 5:	H2>16	0.585

Case 4: This case includes those accidents in which the core damage process occurs with the vessel at high pressure and core cooling is either not restored or restored during the late time regime of the core damage process. The containment is effectively open to the auxiliary building (i.e., either the equipment hatch is open, the MSIVs are open, or there is a LOCA in the auxiliary building). (Note, the cases with the open MSIVs or the LOCA in the auxiliary building are not particularly important since in these cases the auxiliary building will fail early in the accident due to the steam release into the auxiliary building). The amount of hydrogen produced is based on the zirconium oxidation curves developed in the NUREG-1150 project [Harper, et al., 1990]. The concentrations were determined based on the assumption that the containment, drywell, and auxiliary building atmospheres are well mixed. Results from MELCOR calculations indicated that the atmosphere of the bottom floor of the auxiliary does not readily mix with the other volumes. Thus, bottom floor of the auxiliary building is not included in these calculations. The quantification for this case is:

Branch 1:	H2<4	0.066
Branch 2:	H2<8	0.213
Branch 3:	H2<12	0.303
Branch 4:	H2<16	0.243
Branch 5:	H2>16	0.175

Case 5: This case includes those accidents in which the core damage process occurs with the vessel at low pressure and core cooling is either not restored or restored during the late time regime of the core damage process. The containment is effectively open to the auxiliary building (i.e., either the equipment hatch is open,

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the MSIVs are open, or there is a LOCA in the auxiliary building). The amount of hydrogen produced is based on the zirconium oxidation curves developed in the NUREG-1150 project [Harper, et al., 1990]. The concentrations were determined based on the assumption that the containment, drywell, and auxiliary building atmospheres are well mixed. Results from MELCOR calculations indicated that the atmosphere of the bottom floor of the auxiliary does not readily mix with the other volumes. Thus, bottom floor of the auxiliary building is not included in these calculations. The quantification for this case is:

Branch 1:	H2<4	0.040
Branch 2:	H2<8	0.205
Branch 3:	H2<12	0.325
Branch 4:	H2<16	0.250
Branch 5:	H2>16	0.180

Question 31. What is fraction of zirconium that is oxidize in the vessel during core damage?
Number of Branches: 2
Number of Cases: 2
Number of Cases Sampled: 0

The branches for this question are:

1. ZrOxid<21 Less than 21 percent of the zirconium in the vessel is oxidized during core damage.
2. ZrOxid>21 More than 21 percent of the zirconium is the vessel is oxidized during core damage.

This question determines fraction of zirconium in the vessel that is oxidized during the core damage process. The fraction of zirconium oxidized is important because it will affect the amount of radioactive material released during the core damage process. The amount of zirconium oxidized during core damage will also affect the magnitude of the loads accompanying vessel failure since unoxidized zirconium can react with steam during vessel failure and enhance the resulting load. The ranges for the fraction of zirconium oxidized were selected to correspond to the ranges used for the loads distributions and the source term parameters distributions that were developed during the NUREG-1150 project [Harper, et al., 1991],[Harper, et al., 1992]. Since the hydrogen concentrations in the containment are based on a certain fraction of zirconium being oxidized, this information is used to determine the fraction of zirconium oxidized.

Case 1: This case includes two types of accidents. The first type consists of accidents in which the containment is closed and the hydrogen concentration is greater than 16 percent. A hydrogen concentration of 16% corresponds to slightly more than 21% zirconium oxidation. The second type consists of accidents in which the containment is open to the auxiliary building and the hydrogen concentration is greater than 8%. In this case, a hydrogen concentration of 9% corresponds to approximately 21% zirconium oxidation. The quantification for this case is:

Branch 1:	ZrOxid<21	0.00
Branch 2:	ZrOxid>21	1.00

Case 2: This case includes two types of accidents. The first type consists of accidents in which the containment is closed and the hydrogen concentration is less than 16 percent. A hydrogen concentration of 16% corresponds to slightly more than 21% zirconium oxidation. The second type consists of accidents in which the containment is open to the auxiliary building and the hydrogen concentration is less than 8%. In this case, a hydrogen concentration of 9% corresponds to approximately 21% zirconium oxidation. The quantification for this case is:

Branch 1:	ZrOxid<21	1.00
Branch 2:	ZrOxid>21	0.00

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Question 32. Do the operators turn on the HIS during CD?
Number of Branches: 2
Number of Cases: 4
Number of Cases Sampled: 1

The branches for this question are:

1. nHIS-CD The hydrogen igniters are on during core damage.
2. HIS-CD The hydrogen igniters are not on during core damage.

This question determines the status of the hydrogen ignition system (HIS) during the core damage process. The status of the HIS is important because if it is on and the containment is closed, the hydrogen generated during the core damage process will be burned with a minimal pressure load on the containment. This question is primarily concerned with the situation where the HIS could not be turned on prior to core damage because ac power was not available, however, ac power is recovered during core damage.

Case 1: This case includes those accidents in which the operators turned on the igniters before the onset of core damage. There is no reason for the operators to turn the HIS off and, therefore, the HIS remains on during core damage. The quantification for this case is:

Branch 1:	nHIS-CD	0.00
Branch 2:	HIS-CD	1.00

Case 2: This case includes those accidents in which ac power was available before core damage and there was sufficient time for the operators to recognize the need to turn on the HIS. In these accidents, however, the operators failed to turn the HIS on. If the operators failed to turn on the HIS before core damage, there is no reason to believe they will turn the HIS during core damage. The quantification for this case is:

Branch 1:	nHIS-CD	1.00
Branch 2:	HIS-CD	0.00

Case 3: This case includes those accidents in which ac power was not available before core damage but is recovered during core damage before the hydrogen concentration reaches 8%. In these accidents, the lack of ac power before core damage precluded the use of the HIS before core damage. This question assess the probability that the operators will fail to turn on the HIS following the recovery of ac power. This case is sampled; the distribution for the probability that the operators fail to turn on the HIS is the same distribution that was developed for Case 2 of Question 22. The rational for this distribution is discussed in Appendix B.3. The quantification for this case is:

Branch 1:	nHIS-CD	0.054	Lognormal distribution with a mean of 0.054 and an error factor of 5.
Branch 2:	HIS-CD	0.946	

Case 4: This case includes those accidents in which either ac power was not recovered during core damage or was recovered late during the core damage process and the hydrogen concentration was above 8%.

Branch 1:	nHIS-CD	1.00
Branch 2:	HIS-CD	0.00

Question 33. Does an uncontrolled hydrogen combustion event occur during CD?
Number of Branches: 3
Number of Cases: 6
Number of Cases Sampled: 2

The branches for this question are:

1.	nBrn-H2	The hydrogen does not burn.
2.	Brn-H2	A deflagration occurs in the containment during core damage.
3.	Brn-Dif	The hydrogen burns as a diffusion flame.

This question determines whether the hydrogen in the containment is ignited and if it is ignited does it burn as a large deflagration or as a relatively benign diffusion flame. The type of burn that occurs is important because is will affect magnitude of the loads on the containment.

Case 1: This case includes those accidents in which the HIS is on and the hydrogen is initially released into the containment (i.e., MSIVs are not open and there is not a LOCA in the auxiliary building). The HIS will burn the hydrogen as a diffusion flame. The quantification for this case is:

Branch 1:	nBrn-H2	0.00
Branch 2:	Brn-H2	0.00
Branch 3:	Brn-Dif	1.00

Case 2: This case includes those accidents in which the HIS is not on and the hydrogen concentration is less than 4%. With this concentration, the hydrogen will not burn as a deflagration. The quantification for this case is:

Branch 1:	nBrn-H2	1.00
Branch 2:	Brn-H2	0.00
Branch 3:	Brn-Dif	0.00

Case 3: This case includes those accidents in which ac power is available before core damage or was recovered early during the core damage process, however, the HIS was not turned on. Because ac power is available, there will be plenty of ignition sources; the same assumption is used in this analysis as was used in the NUREG-1150 Grand Gulf plant analysis that the hydrogen will burn. There is uncertainty, however, as to whether it will burn as a deflagration or as a diffusion flame. If there are ample ignition sources and the hydrogen burns before it has had a chance to accumulate, diffusion flames or small relatively benign burns are possible. If the hydrogen ignites after it has had a chance to accumulate, it is likely that it will burn as a deflagration. The case is sampled; the distribution is based on the distribution used in the NUREG-1150 Grand Gulf plant analysis [Brown, et al., 1990]. The quantification (mean values) for this case is:

Branch 1:	nBrn-H2	0.00	
Branch 2:	Brn-H2	0.25	
Branch 3:	Brn-Dif	0.75	UNIFORM distribution between 0.5 and 1.0

Case 4: This case includes those accidents which are initiated by a loss of offsite power that result in a station blackout. In these accidents ac power is recovered late during the core damage process. Furthermore, it is assumed that this hydrogen burns as a deflagration because: (1) since it is not known when ac power is recovered, it is assumed that it is recovered after the accumulation of a significant amount of hydrogen, and (2) since ac power is available there are plenty of ignition sources to ignite the hydrogen. The quantification for this case is:

Branch 1:	nBrn-H2	0.00
Branch 2:	Brn-H2	1.00
Branch 3:	Brn-Dif	0.00

Case 5: This case includes those accidents which are initiated by a loss of offsite power that result in a station blackout. In these accidents ac power is not recovered during the core damage process. Without identifiable

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ignition sources, there is uncertainty as to whether the hydrogen will ignite If it does ignite, it is conservatively assumed that ignition occurs after a significant amount of hydrogen has accumulated in the containment. This case is sampled; the distribution is based on information for the NUREG-1150 Grand Gulf plant analysis. While the NUREG-1150 analysis used a complicated distribution developed by a panel of experts, in this analysis, for the sake of simplicity, a uniform distribution between 0.0 and 0.75 is used to characterize the uncertainty in the likelihood that the hydrogen will ignite. The lower and upper bounds of the uniform distribution correspond to the upper and lower bounds on the expert distribution from NUREG-1150. The quantification (mean values) for this case is:

Branch 1:	nBrn-H2	0.50	
Branch 2:	Brn-H2	0.50	Uniform distribution between 0.0 and 0.75.
Branch 3:	Brn-Dif	0.00	

Case 6: This case is not used.

Question 34. What is the pressure in the containment during CD (no uncontrolled burn)?
Number of Branches: 2
Number of Cases: 3
Number of Cases Sampled: 0

The branches for this question are:

1.	P-Lo	The pressure is less than the vent pressure threshold (P< 20 psig)
2.	P-Vnt	The pressure in the containment exceeds the vent threshold.

This question determines whether the base pressure (i.e., does not include pressure spikes from hydrogen combustion) in the containment exceeds the vent threshold. The Grand Gulf Emergency Procedures (EP-3) direct the operators to vent the containment once the pressure exceeds 20 psig and cannot be maintained below 22 psig. During the accident, the containment will pressurize due to the accumulation of steam and non-condensibles (e.g., hydrogen). If the containment is open, it will only pressurize to the failure pressure of the auxiliary building which is well below the vent threshold.

Case 1: This case includes those accidents in which the containment is open to the auxiliary building (i.e., the containment equipment hatch has been removed). In this situation, the containment pressure will not exceed the auxiliary building failure pressure which is well below the vent threshold. The quantification for this case is:

Branch 1:	P-Lo	1.00
Branch 2:	P-Vnt	0.00

Case 2: This case includes those accidents in which the reactor vessel is open to the auxiliary building (i.e., open MSIVs or LOCA in the auxiliary building). In this situation, the containment will not pressurize significantly because the steam and non-condensibles released from the vessel bypass the containment and enter the auxiliary directly. The quantification for this case is:

Branch 1:	P-Lo	1.00
Branch 2:	P-Vnt	0.00

Case 3: This case includes those accidents in which the containment is closed and the releases from the reactor vessel either enter the containment directly (i.e., LOCA or stuck open head vent or SRV vacuum breaker) or enter the containment after first passing through the suppression pool. For the accidents analyzed in this study, all of the LOCAs occur with the vessel depressurized and the containment is open. For the cases with an open RPV head vent, MELCOR calculations indicated that there is no rapid or significant pressurization. Thus, there are no cases where the containment will pressurize to the vent threshold. The quantification for this case is:

Branch 1:	P-Lo	1.00
Branch 2:	P-Vnt	0.00

Question 35. Does the containment fail from quasi-static loads during core damage?
Number of Branches: 3
Number of Cases: 6
Number of Cases Sampled: 0

The branches for this question are:

1. nCF-CD — The containment does not failure from a hydrogen burn during core damage.
2. CF-Rpt-CD — The containment fails in the rupture mode (Nominal hole size is 1 ft^2) from a hydrogen burn during core damage.
3. CF-Lk-CD — The containment fails in the leak mode (Nominal hole size is 0.1 ft^2) from a hydrogen burn during core damage.

This question determines whether the containment fails from a hydrogen burn during core damage. The status of the containment integrity is important because failure of the containment will result in path for radionuclides to enter the environment. Similar to the NUREG-1150 Grand Gulf plant analysis [Brown, et al., 1990, two failure sizes are defined: a leak and a rupture. A leak is define as a failure that will not result in rapid depressurization of the containment, a nominal hole size of 0.1 ft^2 is assigned to this mode of failure. A rupture is define as a failure that will result in rapid depressurization of the containment, a nominal hole size of 1 ft^2 is assigned to this mode of failure. Based on previous structural analysis of the Grand Gulf containment, it was concluded that the most likely location for failure is the region near the junction of the dome and the cylindrical wall [Brown, et al., 1990]. A failure in this location will most likely result in a release to the enclosure building that surrounds the containment dome. Since the enclosure building has virtually no pressure retaining capability and is essentially isolated from the auxiliary building, it is assumed that following containment failure the release goes directly from the containment into the environment.

To determine if the containment fails from quasi-static loads that accompany the combustion of hydrogen, a distribution that characterizes the uncertainty in the load that results from the combustion of a specific concentration of hydrogen was convolved with the Grand Gulf containment failure pressure distribution. The result of this convolution is the probability that the containment fails given a specified concentration of hydrogen is ignited and burns. The hydrogen load distributions generated by the expert panels in the NUREG-1150 project [Harper, et al., 1991] were used in this study. Similarly, the containment failure pressure distribution developed for the NUREG-1150 Grand Gulf plant analysis [Harper, et al., 1994] was also used in this analysis.

Case 1: This case includes those accidents in which either the containment is open to the auxiliary building (i.e., the containment equipment hatch has been removed), the reactor vessel is open to the auxiliary building, the hydrogen does not burn, or if the hydrogen does burn it burns as a diffusion flame. In this situation, the containment does not fail from a hydrogen combustion event. The quantification for this case is:

Branch 1:	nCF-CD	1.00
Branch 2:	CF-Rpt-CD	0.00
Branch 3:	CF-Lk-CD	0.00

Case 2: This case includes those accidents in which the containment is closed and the concentration of hydrogen in the containment is less than 8%. While the hydrogen burns as a deflagration, the resulting load is not sufficient the fail the containment. In this situation, the containment does not fail from a hydrogen combustion event. The quantification for this case is:

Branch 1:	nCF-CD	1.00
Branch 2:	CF-Rpt-CD	0.00

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Branch 3:	CF-Lk-CD	0.00

Case 3: This case includes those accidents in which the containment is closed, the concentration of hydrogen in the containment is between 8% and 12%, and the hydrogen burns as a deflagration. The probability of containment failure was obtained from the convolution of the hydrogen load distribution with the containment failure pressure distribution. The quantification for this case is:

Branch 1:	nCF-CD	0.79
Branch 2:	CF-Rpt-CD	0.19
Branch 3:	CF-Lk-CD	0.02

Case 4: This case includes those accidents in which the containment is closed, the concentration of hydrogen in the containment is between 12% and 16%, and the hydrogen burns as a deflagration. The probability of containment failure was obtained from the convolution of the hydrogen load distribution with the containment failure pressure distribution. The quantification for this case is:

Branch 1:	nCF-CD	0.13
Branch 2:	CF-Rpt-CD	0.49
Branch 3:	CF-Lk-CD	0.38

Case 5. This case includes those accidents in which the containment is closed, the concentration of hydrogen in the containment is greater than 16%, and the hydrogen burns as a deflagration. The probability of containment failure was obtained from the convolution of the hydrogen load distribution with the containment failure pressure distribution. The quantification for this case is:

Branch 1:	nCF-CD	0.04
Branch 2:	CF-Rpt-CD	0.50
Branch 3:	CF-Lk-CD	0.46

Case 6. This case is not used.

Question 36. Do the operators vent the containment during core damage?
Number of Branches: 2
Number of Cases: 5
Number of Cases Sampled: 2

The branches for this question are:

1. nVnt-CD — The containment is not vented during core damage.
2. Vnt-CD — The operators vent the containment during core damage.

This question determines whether the operators vent the containment during core damage. The status of containment venting is important because opening the containment vent establishes a path from the containment to the environment that bypasses the auxiliary building which allows airborne radioactive material in the containment to escape directly to the environment. The size of the vent path is equivalent to a rupture in the containment. The Grand Gulf Emergency Procedures (EP-3) direct the operators to vent the containment if the containment pressure is greater than 20 psig and cannot be maintained below 22 psig. While the procedures allow the operators to close the vent once the pressure drops below 20 psig, the operators would have to open the vent again later in the accident since containment cooling is not available and the steam and non-condensibles generated during the accident would again cause the containment pressure to exceed 22 psig. This analysis makes no attempt to model the opening and the closing of the vent to maintain the pressure below 20 psig. Instead, it is assumed that once the vent is opened, it remains open for the duration of the accident.

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Case 1: This case includes those accidents in which either the containment vent system is not available or the containment pressure is below the vent threshold pressure. The quantification for this case is:

Branch 1:	nVnt-CD	1.00
Branch 2:	Vnt-CD	0.00

Case 2: This case includes those accidents in which the pressure in the containment is above the vent threshold. In these accidents, the reactor vessel was at high pressure before core damage and the operators followed the procedures and depressurized the reactor vessel during core damage. The containment vent system is available (i.e., either available from the start of the accident or was recoverable and ac power was recovered during core damage). Since the operators followed procedures and depressurized the reactor vessel, it is likely that they will follow procedures and vent the containment. This case is sampled; the distribution that characterizes the uncertainty in the probability that the operators will fail to vent the containment was developed in the HRA analysis and is discussed in Appendix B.2. The quantification (mean value) for this case is:

Branch 1:	nVnt-CD	0.031	Lognormal distribution with a mean of 0.031 and an error factor of 5.
Branch 2:	Vnt-CD	0.969	

Case 3: This case is the same as the previous case except that the operators failed to follow procedures and depressurize the reactor vessel. Since, the operators failed to follow procedures, it is assumed that they will not follow procedures and vent the containment. The quantification for this case is:

Branch 1:	nVnt-CD	1.0
Branch 2:	Vnt-CD	0.0

Case 4: This case includes those accidents in which the pressure in the containment is above the vent threshold. In these accidents, the reactor vessel was at low pressure before core damage and remains at low pressure during core damage. The containment vent system is available (i.e., either available from the start of the accident or was recoverable and ac power was recovered during core damage). In these accidents the operators have not failed to following procedures during core damage and, therefore, it is likely that they will follow procedures and vent the containment. This case is sampled; the distribution that characterizes the uncertainty in the probability that the operators will fail to vent the containment was developed in the HRA analysis and is discussed in Appendix B.2. The quantification (mean value) for this case is:

Branch 1:	nVnt-CD	0.031	Lognormal distribution with a mean of 0.031 and an error factor of 5.
Branch 2:	Vnt-CD	0.969	

Case 5: This case includes those accidents in which ac power was not available before core damage and the pressure in the containment rises above the vent threshold during core damage. In these accidents, the containment vent system was recoverable, however, ac power was not restored during core damage. Without ac power the containment vent cannot be opened. The quantification for this case is:

Branch 1:	nVnt-CD	1.0
Branch 2:	Vnt-CD	0.0

Question 37. What is the status of the containment during core damage?
Number of Branches: 2
Number of Cases: 2
Number of Cases Sampled: 0

The branches for this question are:

1. nOCnt-CD The containment is breached during core damage
2. OCnt-CD The containment remains intact during core damage.

This question summarizes the status of containment integrity at the end of core damage. It includes both situations where the containment was breached prior to core damage and situations where it was breached during core damage. The question does not address the size of the breach.

Case 1: This case includes those accidents in which either the containment access penetrations were open before core damage, the operators vented the containment, or the containment failed from a hydrogen combustion event during core damage. The quantification for this case is:

Branch 1:	nOCnt-CD	0.00
Branch 2:	OCnt-CD	1.00

Case 2: This case includes all other accidents in which the containment remains intact during core damage. The quantification for this case is:

Branch 1:	nOCnt-CD	1.00
Branch 2:	OCnt-CD	0.00

Question 38. What is the size of the containment opening during core damage?
Number of Branches: 3
Number of Cases: 4
Number of Cases Sampled: 0

The branches for this question are:

1. Cnt-Rpt-CD The containment opening is the size of a rupture (nominal size is 1 ft^2).
2. Cnt-Lk-CD The containment opening is the size of a leak (nominal size is 0.1 ft^2).
3. Cnt-NF-CD The containment remains intact during core damage.

This question summarizes the size of the containment opening during core damage.

Case 1: This case includes those accidents in which the containment remains intact during core damage. The quantification for this case is:

Branch 1:	Cnt-Rpt-CD	0.00
Branch 2:	Cnt-Lk-CD	0.00
Branch 3:	Cnt-NF-CD	1.00

Case 2: This case includes those accidents in which either the containment access penetration were open prior to core damage, the containment was vented, or a hydrogen combustion event ruptured the containment during core damage. The quantification for this case is:

Branch 1:	Cnt-Rpt-CD	1.00
Branch 2:	Cnt-Lk-CD	0.00
Branch 3:	Cnt-NF-CD	0.00

Case 3: This case includes those accidents in which a hydrogen combustion event caused a leak in the containment during core damage. The quantification for this case is:

Branch 1:	Cnt-Rpt-CD	0.00
Branch 2:	Cnt-Lk-CD	1.00

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Branch 3: Cnt-NF-CD 0.00

Case 4: This case is not used.

Question 39. Does the auxiliary building fail during core damage during core damage?
Number of Branches: 2
Number of Cases: 8
Number of Cases Sampled: 0

The branches for this question are:

1. nOAux-CD The auxiliary building remains intact during core damage.
2. OAux-CD The auxiliary building either fails prior to or during core damage.

This question determines the status of the auxiliary building integrity during core damage. The status of the auxiliary building integrity is important because, for cases with either the reactor vessel or the containment open to the auxiliary building, it defines when the radioactive material is released to the environment. If the containment is closed prior to core damage and the reactor vessel is not open to the auxiliary building, a subsequent containment failure will bypass the auxiliary building (i.e., containment fails above auxiliary building roof).

Case 1: This case includes those accidents in which the auxiliary building failed prior to core damage. The quantification for this case is:

Branch 1: nOAux-CD 0.00
Branch 2: OAux-CD 1.00

Case 2: This case includes those accidents in which the reactor vessel is open to the auxiliary building (i.e., open MSIVs or LOCA in auxiliary building) during core damage. Results for MELCOR indicate that the auxiliary building will failure prior to core damage and, therefore, it is open during core damage. The quantification for this case is:

Branch 1: nOAux-CD 0.00
Branch 2: OAux-CD 1.00

Case 3: This case includes those accidents in which the auxiliary building did not fail before core damage, the reactor is not open to the auxiliary building, and the containment was closed prior to the onset of core damage. In this situation, any releases of steam and/or non-condensibles will bypass the auxiliary building. The quantification for this case is:

Branch 1: nOAux-CD 1.00
Branch 2: OAux-CD 0.00

Case 4: This case includes those accidents in which the containment is open to the auxiliary building and there is a hydrogen deflagration in the containment. The pressure load from the containment will fail the relatively weak auxiliary building structure. The quantification for this case is:

Branch 1: nOAux-CD 0.00
Branch 2: OAux-CD 1.00

Case 5: This case includes those accidents in which the containment is open to the auxiliary building and the hydrogen generated during core damage burns as a diffusion flame. Based on sensitivity calculations using MELCOR, it is expected that the auxiliary building will fail. The quantification for this case is:

Branch 1: nOAux-CD 0.01
Branch 2: OAux-CD 0.99

Case 6: This case includes those accidents initiated by a LOCA. The containment is open to the auxiliary building and it is flooded with water. In this case, the hydrogen generated during core damage does not burn. Based on results from MELCOR, the pressure in auxiliary building is sufficiently low that failure of the auxiliary building is not expected before vessel failure. The quantification for this case is:

Branch 1: nOAux-CD 1.00
Branch 2: OAux-CD 0.00

Case 7: This case includes those accidents in which the reactor vessel is not breached prior to vessel failure, however, either the reactor head vent or a vacuum breaker on a SRV tailpipe sticks open. The containment is open and the auxiliary building was intact before the onset of core damage. The portion of the steam that leaves the reactor vessel via the open head vent or vacuum break will bypass the suppression pool. Based on results from MELCOR, the steam that bypasses the suppression pool is sufficient to pressurize the auxiliary building to its failure point. The quantification for this case is:

Branch 1: nOAux-CD 0.00
Branch 2: OAux-CD 1.00

Case 8: This case includes those accidents in which the reactor vessel is not breached prior to vessel failure and the reactor head vent remains closed. Also, there are no stuck open SRV tailpipe vacuum breakers. The containment is open and the auxiliary building was intact before the onset of core damage. The steam generated in the vessel is directed to the suppression pool where it is condensed. In this case, the hydrogen generated during core damage does not burn. The pressure in the auxiliary building remains below it failure pressure. The quantification for this case is:

Branch 1: nOAux-CD 1.00
Branch 2: OAux-CD 0.00

Question 40. Is there water in the RPV pedestal cavity just prior to VB?
Number of Branches: 2
Number of Cases: 2
Number of Cases Sampled: 0

The branches for this question are:

1. Cav-Dry — The cavity below the reactor vessel is essentially dry.
2. Cav-Fld — The reactor cavity is flooded with water.

This question determines the amount of water that is in the reactor cavity at the time of vessel failure. The amount of water in the cavity (i.e., dry or flooded) is important because it will determine if ex-vessel steam explosions are possible, it can impact the loads at vessel breach, and it will affect the coolability of debris release from the vessel following vessel failure. For this analysis, the only sources of water in the drywell before vessel failure are from either a LOCA or for cases where the operators deliberately try to flood the containment. The containment sprays are in the containment and, therefore, use of the sprays will not flood the cavity. Also, since in these accidents the drywell personnel lock and/or equipment hatch is open, significant pressure differences between the drywell and containment will not exist and, therefore, the possibility of pushing water over the weir wall is not a concern.

Case 1: This case includes those accidents in which the operators deliberately attempt to flood the containment (the LOCAs are included in these accidents). The cavity is flooded up to the bottom the of reactor vessel. The quantification for this case is:

Branch 1:	Cav-Dry	0.00
Branch 2:	Cav-Fld	1.00

Case 2: This case includes all the other accidents in which the operators do not flood the containment. The quantification for this case is:

Branch 1:	Cav-Dry	1.00
Branch 2:	Cav-Fld	0.00

Question 41. Is the core damage process arrested in the vessel?
Number of Branches: 2
Number of Cases: 4
Number of Cases Sampled: 1

The branches for this question are:

1	nCDArrest	The core damage process is arrested in the vessel.
2	CDArrest	The core damage process is not arrested.

This question determines whether the core damage process is arrested in the vessel. The state of the core in the vessel is important because if the core damage process is arrested, vessel failure is precluded and the release of radioactive material from the core debris is terminated. The heat generated by the core (and core debris) will fail the vessel if coolant is not supplied to the core. The probability of vessel failure depends on when during the core damage process the coolant is restore to the vessel. The in-vessel phase of the core damage process was divided into two time regimes. The first time regime corresponds to the time available to restore coolant to the core and arrest the core damage process. The duration of this first time regime is 1.5 hours which corresponds to the point in the core damage process where approximately 10% of the core is damaged. It is estimated the restoration of coolant to the core with only 10% of the core damage will prevent any further damage. After this point in the core damage progression (i.e., the second time regime) there is a rapid escalation in the amount of core damage and, therefore, it is assumed that beyond this point the likelihood of core damage arrest is very small.

Case 1: This case includes those accidents in which the vessel integrity is breached via a LOCA (either inside the containment or in an interfacing system) and the LOCA is not isolated during core damage. In these accidents, the use of injection will result in a flood in the auxiliary building (i.e., for LOCAs inside the containment the lower personnel lock is open) which will in turn fail any remaining injection systems. Thus, without coolant injection the core cannot be continuously cooled. The quantification for this case is:

Branch 1:	nCDArrest	1.00
Branch 2:	CDArrest	0.00

Case 2: This case includes those accidents in which core cooling is restored during the early time regime of the core damage process before a significant fraction of the core has been damage. In this case, most of the fuel is still in its original geometry (although the cladding may have failed) and, therefore, a significant debris bed has not formed. The restoration of core cooling in this situation will arrest the core damage process. The quantification for this case is:

Branch 1:	nCDArrest	0.00
Branch 2:	CDArrest	1.00

Case 3: This case includes those accidents in which core cooling is restored during the late time regime of the core damage process. Since in this case most of the fuel has been damaged by the time coolant is restored, it is unlikely that the core damage process will be arrested. The probability of vessel failure will depend on whether the core debris forms a coolable debris bed or instead forms a dense layer on the bottom head of the vessel.

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This case is sampled, the distribution used to characterize the uncertainty in the probability that the core debris is not coolable is a Maximum Entropy Distribution with a lower value, a mean value, and a upper value of 0.0, 0.01, and 0.5 respectively. The quantification for this case is:

Branch 1:	nCDArrest	0.99	
Branch 2:	CDArrest	0.01	Maximum Entropy Distribution: Lower value = 0.00, Mean = 0.01, Upper value = 0.5

Case 4: This case includes those accidents in which core cooling is not restored and, therefore, vessel failure is assured. The quantification for this case is:

Branch 1:	nCDArrest	1.00
Branch 2:	CDArrest	0.00

Question 42. What fraction of the core debris would be mobil at vessel breach?
Number of Branches: 2
Number of Cases: 2
Number of Cases Sampled: 0

The branches for this question are:

1. HiLiqVB — A large amount of core debris (nominally 40%) is mobile when VB occurs.
2. LoLiqVB — A small amount of core debris (nominally 10%) is mobile when VB occurs.

This question determines the amount of core debris that is mobile at the time of vessel breach (VB). The amount of core debris that is mobile is important because it affects: (1) the mode of vessel failure, (2) the magnitude of the loads that accompany vessel failure, and (3) the probability that the core debris released from the vessel is coolable for cases with water in the reactor cavity. Nominal values are used to characterize the amount of that would be mobile at VB. A nominal value of 10% represents low mobility, whereas a nominal value of 40% represents high mobility. The 10% value represents the range from 0% to 20% molten, and the 40% value represents any larger quantities. The nominal values used for this question and the probabilities associated with the various levels of mobility are based on the NUREG-1150 Grand Gulf plant analysis [Brown et al., 1990].

In NUREG-1150, it was felt that the amount of material molten at VB was tightly coupled to the mode of vessel failure. If the vessel fails early, then the mobility will be low. In BWRs, early vessel failure would be due to melt flowing through an instrument tube and failing the tube outside the vessel. If the melt were to freeze and plug the tube, then vessel failure would be delayed until a massive creep rupture occurs. Hence, the major uncertainty is whether the melt flowing in the instrument tube will freeze. If the vessel fails by a massive creep rupture and water is being injected into the vessel, it is likely that it will fail with a low mobility. On the other hand, if there is no water injection and a massive creep rupture occurs, it is uncertain as to how much core debris will be molten.

Case 1: This case includes those accidents in which injection is restored during core damage. The quantification for this case is:

Branch 1:	HiLiqVB	0.025
Branch 2:	CDArrest	0.975

Case 2: This case includes those accidents in which injection is not restored during core damage. The quantification for this case is:

Branch 1:	HiLiqVB	0.100
Branch 2:	CDArrest	0.900

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Question 43. Does a large in-vessel steam explosion occur?
Number of Branches: 2
Number of Cases: 3
Number of Cases Sampled: 2

The branches for this question are:

1.	nVStmExp	No large in-vessel steam explosions occur prior to VB.
2.	VStmExp	A large in-vessel steam explosion occurs prior to VB.

This question determines whether a large in-vessel steam explosion occurs prior to VB. The occurrence of a large in-vessel steam explosion is important because a steam explosion can fail the vessel. The quantification of the question is based on information from NUREG-1150 [Harper, et al., 1994]. In the NUREG-1150 analysis the likelihood of a steam explosion depended on the reactor vessel pressure.

Case 1: This case includes those accidents in which injection is restored early during the core damage process before a significant fraction of the core has been damaged. Since only a small fraction of the core is damage, there is not enough molten core debris to cause a large in-vessel steam explosion. The quantification for this case is:

Branch 1:	nVStmExp	1.00
Branch 2:	VStmExp	0.00

Case 2: This case includes those accidents in which the reactor vessel is depressurized and coolant injection is either restored late during the core damage process or is not restored. In either case, a significant fraction of the core will be damage which allows for the possibility of a large in-vessel steam explosion. The likelihood of a steam explosion depends on the pressure in the reactor vessel. Based on information used in NUREG-1150, steam explosions are more likely when the pressure is low than when the pressure is near system pressure (i.e., 1000 psia). Results from experimental programs and experience in the metal industry suggests that the occurrence of a steam explosion can be treated as a stochastic event. The likelihood of a steam explosion is, however, very uncertain. In this analysis, a maximum entropy distribution is used to characterize the uncertainty in the probability that a steam explosion occurs. The mean of the maximum entropy distribution corresponds to the value used in the NUREG-1150 analysis for the probability of a steam explosion when the vessel is at low pressure. The quantification for this case is:

Branch 1:	nVStmExp	0.14	
Branch 2:	VStmExp	0.86	Maximum Entropy: Lower Bound = 0.001, Mean = 0.86, Upper Bound = 1.0

Case 3: This case is the same as the previous case except that the pressure in the reactor vessel is at the system pressure (i.e., approximately 1000 psia). Based on information used in NUREG-1150, steam explosions are less likely when the pressure is high than when the pressure is low. Similar to the previous case, a maximum entropy distribution is used to characterize the uncertainty in the probability that a steam explosion occurs. The mean of the maximum entropy distribution corresponds to the value used in the NUREG-1150 analysis for the probability of a steam explosion when the vessel is at high pressure. The quantification for this case is:

Branch 1:	nVStmExp	0.90	
Branch 2:	VStmExp	0.10	Maximum Entropy: Lower Bound = 0.001, Mean = 0.10, Upper Bound = 1.0

Question 44. Does an Alpha Mode event occur?
Number of Branches: 2
Number of Cases: 3

Number of Cases Sampled: 2

The branches for this question are:

1.	nAlpha	An Alpha Mode event occurs.
2.	Alpha	An Alpha Mode event does not occur.

This question determines whether a Alpha Mode event occurs. An Alpha Mode event is an energetic fuel-coolant interaction (i.e., steam explosion) that fails the vessel and generates a missile that then fails the containment. The occurrence of an Alpha Mode event is important because it provides a path for radioactive material to escape both the reactor vessel and the containment. This event was postulated and analyzed during the Reactor Safety Study [USNRC, 1975]. The quantification for this question is based on distributions used in the NUREG-1150 study [Harper, et al., 1994]. In the NUREG-1150 analysis the likelihood of an Alpha Mode event depended on the pressure in the reactor vessel.

Case 1: This case includes those accidents in which a large in-vessel steam explosion occurs with the reactor vessel at low pressure. The quantification for this case is:

Branch 1:	nVStmExp	0.990	
Branch 2:	VStmExp	0.010	User Distribution with a Mean of 0.01

Case 2: This case includes those accidents in which a large in-vessel steam explosion occurs with the reactor vessel at system pressure. The probability of an Alpha Mode event at high pressure was estimated in NUREG-1150 to be an order of magnitude lower than the probability of an Alpha Mode event at low pressure. The quantification for this case is:

Branch 1:	nVStmExp	0.999	
Branch 2:	VStmExp	0.001	User Distribution with a Mean of 0.001

Question 45. Does a large in-vessel steam explosion fail the vessel?
Number of Branches: 5
Number of Cases: 3
Number of Cases Sampled: 0

The branches for this question are:

1.	SE-Alpha	An Alpha Mode fails the reactor vessel.
2.	SE-BtHd	An in-vessel steam explosion causes the bottom head of the reactor vessel to fail (Nominal failure size of 340 ft^2).
3.	SE-LgBrch	An in-vessel steam explosion causes a large breach (22 ft^2) in the reactor vessel.
4.	SE-SmBrch	An in-vessel steam explosion causes a small breach (1 ft^2) in the reactor vessel.
5.	SE-nFail	The reactor vessel does not fail from an in-vessel steam explosion

This question determines whether an in-vessel steam explosion fails the reactor vessel and if it does fail the reactor vessel, it determines the size of the failure. Failure of the reactor vessel is important because it establishes a path for radioactive material to escape the vessel and bypass the suppression pool and it releases the core debris into the reactor cavity below the reactor vessel. If the cavity is dry, the core debris will interact with the concrete cavity and will continue to release radioactive material. If the cavity is flooded, the potential exists for the core debris to be cooled in which case the release of radioactive material from the debris is terminated. The quantification of this question is based on the probabilities developed in the NUREG-1150 project [Harper, et al., 1994].

Case 1: This case includes those accidents in which an Alpha Mode event occurs. By definition, an Alpha Mode event results in a large failure to the reactor vessel. The quantification for this case is:

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Branch 1:	SE-Alpha	1.00
Branch 2:	SE-BtHd	0.00
Branch 3:	SE-LgBrch	0.00
Branch 4:	SE-SmBrch	0.00
Branch 5:	SE-nFail	0.00

Case 2: This case includes those accidents in which a large in-vessel steam explosion occurs. The quantification of this case is based on probabilities developed in the NUREG-1150 project. The quantification for this case is:

Branch 1:	SE-Alpha	0.00
Branch 2:	SE-BtHd	0.20
Branch 3:	SE-LgBrch	0.10
Branch 4:	SE-SmBrch	0.10
Branch 5:	SE-nFail	0.60

Case 3: This case includes those accidents in which a large in-vessel steam explosion does not occur and, therefore, the reactor vessel does not fail from a steam explosion. The quantification for this case is:

Branch 1:	SE-Alpha	0.00
Branch 2:	SE-BtHd	0.00
Branch 3:	SE-LgBrch	0.00
Branch 4:	SE-SmBrch	0.00
Branch 5:	SE-nFail	1.00

Question 46. What is the mode of vessel failure?
Number of Branches: 5
Number of Cases: 7
Number of Cases Sampled: 0

The branches for this question are:

1.	VB-Alpha	An Alpha Mode fails the reactor vessel.
2.	VB-BtHd	The bottom head of the reactor vessel to fail (Nominal failure size of 340 ft^2).
3.	VB-LgBrch	The mode of vessel failure is a large breach (22 ft^2).
4.	VB-SmBrch	The mode of vessel failure is a small breach (1 ft^2).
5.	nVB	The reactor vessel does not fail.

This question determines the mode of vessel failure. This question summarizes previous reactor vessel failure modes caused by large in-vessel steam explosions as well as considers reactor vessel failure modes caused by core debris attack. Failure of the reactor vessel is important because it establishes a path for radioactive material to escape the vessel and bypass the suppression pool and it releases the core debris into the reactor cavity below the reactor vessel. If the cavity is dry, the core debris will interact with the concrete cavity and will continue to release radioactive material. If the cavity is flooded, the potential exists for the core debris to be cooled in which case the release of radioactive material from the debris is terminated.

The vessel is predicted to fail if the core damage process is not arrested (See Question 41). The quantification of this question is based on the probabilities developed in the NUREG-1150 project [Harper, et al., 1994]. Given that the reactor vessel fails from core debris attack, the following modes of reactor vessel failure were considered:

1. Global thermally induced fracture/creep-rupture of the lower head;
2. Ejection of an in-core instrument guide tube or CRD, and
3. Flow of molten core materials through a guide tube, CRD, or drain line leading to their thermally

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induced rupture below the bottom head.

The most likely failure mode is flow-induced thermal failure of a guide tube or drain. Molten material can enter the tube and flow beyond the reactor vessel wall. Thermal weakening followed by rupture of the wall can occur if the melt gives up its latent and sensible heat to the guide tube or drain walls. It is uncertain whether the presence of water in these tubes will prevent tube failure. This uncertainty has been included in the probability assigned to this mode of failure. This failure mode will result in a small hole. It was assessed that there is a very small probability of multiple tube failures resulting in a large hole. It was also estimated that thermally induced binding between the guide tube and the vessel will prevent pressure ejection of the in-core instrument guide tube.

Case 1: This case summarizes previous failures caused by in-vessel steam explosions; it includes those accidents in which the reactor vessel fails as a result of an Alpha Mode event. The quantification for this case is:

Branch 1:	VB-Alpha	1.00
Branch 2:	VB-BtHd	0.00
Branch 3:	VB-LgBrch	0.00
Branch 4:	VB-SmBrch	0.00
Branch 5:	nVB	0.00

Case 2: This case summarizes previous failures caused by in-vessel steam explosions; it includes those accidents in which the reactor vessel bottom head fails. The quantification for this case is:

Branch 1:	VB-Alpha	0.00
Branch 2:	VB-BtHd	1.00
Branch 3:	VB-LgBrch	0.00
Branch 4:	VB-SmBrch	0.00
Branch 5:	nVB	0.00

Case 3: This case summarizes previous failures caused by in-vessel steam explosions; it includes those accidents in which an in-vessel steam explosion results in a large breach to the reactor vessel. The quantifi ation for this case is:

Branch 1:	VB-Alpha	0.00
Branch 2:	VB-BtHd	0.00
Branch 3:	VB-LgBrch	1.00
Branch 4:	VB-SmBrch	0.00
Branch 5:	nVB	0.00

Case 4: This case summarizes previous failures caused by in-vessel steam explosions; it includes those accidents in which an in-vessel steam explosion results in a small breach to the reactor vessel. The quantification for this case is:

Branch 1:	VB-Alpha	0.00
Branch 2:	VB-BtHd	0.00
Branch 3:	VB-LgBrch	0.00
Branch 4:	VB-SmBrch	1.00
Branch 5:	nVB	0.00

Case 5: This case includes those accidents in which the core damage process is arrested in the vessel and a large in-vessel steam explosion either does not occur or does not fail the vessel. Thus, in this situation the reactor vessel does not fail. The quantification for this case is:

Branch 1:	VB-Alpha	0.00
Branch 2:	VB-BtHd	0.00
Branch 3:	VB-LgBrch	0.00
Branch 4:	VB-SmBrch	0.00
Branch 5:	nVB	1.00

Case 6: This case includes those accidents in which the core damage process is not arrested in the vessel and a large in-vessel steam explosion either does not occur or does not fail the vessel. Thus, in this situation the core debris causes thermally induced reactor vessel failure. The probabilities assigned to the branches for this case are based on probabilities developed in the NUREG-1150 study [Harper, et al., 1994]. The quantification for this case is:

Branch 1:	VB-Alpha	0.000
Branch 2:	VB-BtHd	0.249
Branch 3:	VB-LgBrch	0.005
Branch 4:	VB-SmBrch	0.746
Branch 5:	nVB	0.000

Case 7: This case is not used.

Question 47. Does high pressure melt ejection accompany vessel failure?
Number of Branches: 2
Number of Cases: 2
Number of Cases Sampled: 0

The branches for this question are:

1. nHPME A high pressure melt ejection event does not accompany reactor vessel failure.
2. HPME A high pressure melt ejection event accompanies reactor vessel failure.

This question determines whether the core debris is released from the reactor vessel as a high pressure melt ejection (HPME). This question is referenced in the source term analysis to determine the source term associated with vessel breach. If HPME occurs, the fraction of radionuclides released during DCH, FDCH, is applied to the core debris ejected at vessel breach. If the reactor vessel is at high pressure when the vessel fails, the core debris will likely be ejected at a high velocity. Because of its high velocity, it is expected that the ejected material will undergo extensive fragmentation, and the result will be an HPME event. Thus, if the reactor vessel fails at high pressure, HPME is likely. The quantification of the issue is based on probabilities developed in the NUREG-1150 project [Brown, et al, 1990].

Case 1: This case includes those accidents in which either the reactor vessel does not fail or the reactor vessel is depressurized when it fails. Thus, in this situation a HMPE event does not occur. The quantification for this case is:

Branch 1:	nHPME	1.00
Branch 2:	HPME	0.00

Case 2: This case includes those accidents in which the reactor vessel is at high pressure when it fails. The probability that an HPME event occurs is based on the probability used in the NUREG-1150 analysis [Brown, et al., 1990]. The quantification for this case is:

Branch 1:	nHPME	0.20
Branch 2:	HPME	0.80

Question 48. Does a large ex-vessel steam explosion accompany vessel failure?
Number of Branches: 2
Number of Cases: 3
Number of Cases Sampled: 1

The branches for this question are:

1.	nExStmE	A large ex-vessel steam explosion accompanies reactor vessel failure.
2.	ExStmE	A large ex-vessel steam explosion does not occur.

This question determines whether a large ex-vessel steam explosion accompanies reactor vessel failure. This question is referenced in the source term analysis to determine the source term associated with vessel breach. This question is not, however, used to address the quasi-static loads associated with VB. The probability of containment failure at the time of reactor vessel failure (See Question 49) is based on the expected loads from all sources and does not attempt to distinguish between loads from individual events (e.g., steam explosions, direct containment heating, hydrogen combustion). Thus, while the loads from steam explosions are not explicitly represented as a separate source, they were considered when the loads associated with vessel failure where addressed. The quantification of the issue is based on probabilities developed in the NUREG-1150 project [Brown, et al., 1990]

The dropping of hot metal into water has been observed to cause energetic and violent reactions commonly known as Fuel-Coolant Interaction (FCIs) or steam explosions. They appear to be more likely when the water is considerably below its saturation temperature. At Sandia National Laboratories, steam explosions were observed in 86% of the tests in which hot metal was dropped into water [Harper, et al., 1994]. Some of these explosions were extremely energetic, others were not. For an ex-vessel steam explosion to occur in a severe accident reactor accident, there must be water in the reactor cavity prior to vessel failure or water must enter the cavity coincident with the debris, and the vessel must fail allowing the core debris to enter the reactor cavity.

Case 1: This case includes those accidents in which either the reactor vessel does not fail or an Alpha Mode event occurs. The affect that an Alpha Mode event has on the source term is addressed in the question that addresses HPME events (See Question 47). The quantification for this case is:

Branch 1:	nExStmE	1.00
Branch 2:	ExStmE	0.00

Case 2: This case includes those accidents in which core debris is released from the reactor vessel either into a flooded reactor cavity or into a dry cavity coincident with water from the reactor vessel that is supplied by an injection source. In this situation, steam explosions are possible. Results from experimental programs and experience in the metal industry suggests that the occurrence of a steam explosions can be treated as a stochastic event. The likelihood of a steam explosion is, however, very uncertain. In this analysis, a maximum entropy distribution is used to characterize the uncertainty in the probability that a steam explosion occurs. The mean of the maximum entropy distribution corresponds to the value used in the NUREG-1150 analysis for the probability of an ex-vessel steam explosion. The quantification for this case is:

Branch 1:	nExStmE	0.14	
Branch 2:	ExStmE	0.86	Maximum Entropy: Lower Bound = 0.001, Mean = 0.86, Upper Bound = 1.0

Case 3: This case includes those accidents in which the reactor cavity is dry and, therefore, an ex-vessel steam explosion is not possible. The quantification for this case is:

Branch 1:	nExStmE	1.00
Branch 2:	ExStmE	0.00

Question 49. Does the containment fail from pressure loads accompanying vessel failure?
Number of Branches: 3
Number of Cases: 11
Number of Cases Sampled: 0

The branches for this question are:

1.	nCF-VB	The containment does not fail from pressure loads accompanying vessel failure.
2.	CF-Rpt-VB	The containment fails in the rupture mode (Nominal hole size is 1 ft^2) from pressure loads accompanying vessel failure.
3.	CF-Lk-VB	The containment fails in the leak mode (Nominal hole size is 0.1 ft^2) from pressure loads accompanying vessel failure.

This question determines whether the containment fails from pressure loads accompanying vessel failure. The status of the containment integrity is important because failure of the containment will result in path for radionuclides to enter the environment. Similar to the NUREG-1150 Grand Gulf plant analysis [Brown, et al., 1990], two failure sizes are defined: a leak and a rupture. The containment failure issue is discussed in Question 35.

To determine if the containment fails from pressure loads that accompany vessel failure, a distribution that characterizes the uncertainty that in load that results from vessel failure was convolved with the Grand Gulf containment failure pressure distribution. The result of this convolution is the probability that the containment fails given a specified pressure load. The distributions for the pressure loads accompanying vessel failure generated by the expert panels in the NUREG-1150 project [Harper, et al., 1991] were used in this study. These distributions consider loads that result from steam explosions, direct containment heating, hydrogen combustion, and reactor vessel blowdown. The containment failure pressure distribution developed for the NUREG-1150 Grand Gulf plant analysis [Harper, et al., 1994] was also used in this analysis.

Case 1: This case includes those accidents in which the vessel does not fail and, therefore, there are no pressure loads to cause containment failure. The quantification for this case is:

Branch 1:	nCF-VB	1.00
Branch 2:	CF-Rpt-VB	0.00
Branch 3:	CF-Lk-VB	0.00

Case 2: This case includes those accidents in which the containment was already ruptured. This case also includes those accidents in which the containment fails as a result of an Alpha Mode event. In either case, the containment is already rupture and, therefore, further failure of the containment is not considered. The quantification for this case is:

Branch 1:	nCF-VB	0.00
Branch 2:	CF-Rpt-VB	1.00
Branch 3:	CF-Lk-VB	0.00

Case 3: This case includes those accidents in which the reactor vessel is pressurized when it fails; a large amount of molten core debris is released at the time of vessel failure. The probability of containment failure was obtained from the convolution of the pressure load distribution with the containment failure pressure distribution. The quantification for this case is:

Branch 1:	nCF-VB	0.22
Branch 2:	CF-Rpt-VB	0.35
Branch 3:	CF-Lk-VB	0.43

Case 4: This case includes those accidents in which the reactor vessel is pressurized when it fails; a small

amount of molten core debris is released at the time of vessel failure. The probability of containment failure was obtained from the convolution of the pressure load distribution with the containment failure pressure distribution. The quantification for this case is:

Branch 1:	nCF-VB	0.41
Branch 2:	CF-Rpt-VB	0.27
Branch 3:	CF-Lk-VB	0.32

Case 5: This case includes those accidents in which the reactor vessel is at low pressure when it fails; more than 21% of the zirconium in the core debris is oxidized and it is released from the vessel into a flooded reactor cavity. The probability of containment failure was obtained from the convolution of the pressure load distribution with the containment failure pressure distribution. The quantification for this case is:

Branch 1:	nCF-VB	0.90
Branch 2:	CF-Rpt-VB	0.04
Branch 3:	CF-Lk-VB	0.06

Case 6: This case includes those accidents in which the reactor vessel is at low pressure when it fails; less than 21% of the zirconium in the core debris is oxidized and it is released from the vessel into a flooded reactor cavity. The probability of containment failure was obtained from the convolution of the pressure load distribution with the containment failure pressure distribution. The quantification for this case is:

Branch 1:	nCF-VB	0.82
Branch 2:	CF-Rpt-VB	0.10
Branch 3:	CF-Lk-VB	0.08

Case 7: This case includes those accidents in which the hydrogen igniters have been on during core damage and, therefore, the hydrogen generated during core damage has been consumed. Since the vessel fails at low pressure into a dry cavity, it is unlikely that there will be a rapid production of hydrogen at the time of vessel failure. Hence, it is likely that the igniters will burn the hydrogen with minimal pressurization of the containment. A small probability is assigned to containment failure to account for the possibility of the rapid generation of hydrogen when the core debris and water are released at the time of vessel failure. The quantification for this case is:

Branch 1:	nCF-VB	0.99
Branch 2:	CF-Rpt-VB	0.005
Branch 3:	CF-Lk-VB	0.005

Case 8: This case includes those accidents in which the reactor vessel fails at low pressure and releases the core debris into a dry cavity. During core damage, hydrogen accumulated in the containment, however, it did not ignite because of a lack of ignition sources. The peak hydrogen concentration in the containment before vessel failure was between 8 and 12%. Since the vessel fails at low pressure into a dry cavity, it is unlikely that there will be a rapid production of hydrogen at the time of vessel failure, however, the hot debris released into the containment will provide ample ignition sources for the preexisting hydrogen. In this situation, ignition is certain. The containment failure probability used for this case is the same probability used for the corresponding case for hydrogen burns during core damage (See Question 35). The quantification for this case is:

Branch 1:	nCF-VB	0.79
Branch 2:	CF-Rpt-VB	0.19
Branch 3:	CF-Lk-VB	0.02

Case 9: This case includes those accidents in which the reactor vessel fails at low pressure and releases the core

debris into a dry cavity. During core damage, hydrogen accumulated in the containment, however, it did not ignite because of a lack of ignition sources. The peak hydrogen concentration in the containment before vessel failure was between 12 and 16%. Since the vessel fails at low pressure into a dry cavity, it is unlikely that there will be a rapid production of hydrogen at the time of vessel failure, however, the hot debris released into the containment will provide ample ignition sources for the preexisting hydrogen. In this situation, ignition is certain. The containment failure probability used for this case is the same probability used for the corresponding case for hydrogen burns during core damage (See Question 35). The quantification for this case is:

Branch 1:	nCF-VB	0.13
Branch 2:	CF-Rpt-VB	0.49
Branch 3:	CF-Lk-VB	0.38

Case 10: This case includes those accidents in which the reactor vessel fails at low pressure and releases the core debris into a dry cavity. During core damage, hydrogen accumulated in the containment, however, it did not ignite because of a lack of ignition sources. The peak hydrogen concentration in the containment before vessel failure was greater than 16%. Since the vessel fails at low pressure into a dry cavity, it is unlikely that there will be a rapid production of hydrogen at the time of vessel failure, however, the hot debris released into the containment will provide ample ignition sources for the preexisting hydrogen. In this situation, ignition is certain. The containment failure probability used for this case is the same probability used for the corresponding case for hydrogen burns during core damage (See Question 35). The quantification for this case is:

Branch 1:	nCF-VB	0.04
Branch 2:	CF-Rpt-VB	0.50
Branch 3:	CF-Lk-VB	0.46

Case 11: This case includes those accidents in which the reactor vessel fails at low pressure and releases the core debris into a dry cavity. In these accidents, either the hydrogen generated during core damage burned or the peak hydrogen concentration was less than 8%. In either case, there is an insufficient amount of hydrogen to cause containment failure. The quantification for this case is:

Branch 1:	nCF-VB	1.00
Branch 2:	CF-Rpt-VB	0.00
Branch 3:	CF-Lk-VB	0.00

Question 50. What is the status of containment integrity just after vessel failure?
Number of Branches: 2
Number of Cases: 2
Number of Cases Sampled: 0

The branches for this question are:

1.	nOCnt-VB	The containment is breached at the time of vessel failure.
2.	OCnt-VB	The containment is still intact just after vessel failure.

This question summarizes the status of containment integrity at the end of vessel failure. This question summarizes failures that occurred before vessel failure and also include failures that occurred at the time of vessel failure. This question does not address the size of the breach.

Case 1: This case includes those accidents in which either the containment was breached during or prior to core damage, the containment failed from an Alpha Mode event, or the containment failed from pressure loads that accompanied vessel breach. The quantification for this case is:

Branch 1:	nOCnt-VB	0.00
Branch 2:	OCnt-VB	1.00

Case 2: This case includes all other accidents in which the containment is still intact after vessel failure. The quantification for this case is:

Branch 1:	nOCnt-VB	1.00
Branch 2:	OCnt-VB	0.00

Question 51. What is the size of the containment opening just after vessel failure?
Number of Branches: 3
Number of Cases: 4
Number of Cases Sampled: 0

The branches for this question are:

1. Cnt-Rpt-VB The containment opening is the size of a rupture (nominal size is 1 ft^2).
2. Cnt-Lk-VB The containment opening is the size of a leak (nominal size is 0.1 ft^2).
3. Cnt-NF-VB The containment is still intact after vessel failure

This question summarizes the size of the containment opening at the end of the vessel breach time regime.

Case 1: This case includes those accidents in which the containment is still intact at the end of the vessel breach time regime. The quantification for this case is:

Branch 1:	Cnt-Rpt-VB	0.00
Branch 2:	Cnt-Lk-VB	0.00
Branch 3:	Cnt-NF-VB	1.00

Case 2: This case includes those accidents in which either the containment was rupture prior to or during core damage, the containment failed as a result of an Alpha Mode event, or pressure loads accompanying vessel failure caused a rupture in the containment. The quantification for this case is:

Branch 1:	Cnt-Rpt-VB	1.00
Branch 2:	Cnt-Lk-VB	0.00
Branch 3:	Cnt-NF-VB	0.00

Case 3: This case includes those accidents in which a hydrogen combustion event during core damage or pressure loads accompanying vessel breached caused a leak in the containment. The quantification for this case is:

Branch 1:	Cnt-Rpt-VB	0.00
Branch 2:	Cnt-Lk-VB	1.00
Branch 3:	Cnt-NF-VB	0.00

Case 4: This case is not used.

Question 52. Does the auxiliary building fail due to loads accompanying vessel failure?
Number of Branches: 2
Number of Cases: 3
Number of Cases Sampled: 0

The branches for this question are:

1. nOAux-VB The auxiliary building is intact at the end of the vessel breach time regime.
2. OAux-VB The auxiliary building has failed by the end of the vessel breach time regime.

This question determines the status of the auxiliary building integrity at the end of the vessel breach time regime. The status of the auxiliary building integrity is important because, for cases with either the reactor vessel or the containment open to the auxiliary building, it defines when the radioactive material is released to the environment. As discussed in Question 35, for accidents in which the containment is close prior to core damage and the reactor vessel is not open to the auxiliary building (i.e., via open MSIVs or LOCA in auxiliary building), any subsequent failure of the containment will occur above the auxiliary building roof and, therefore, all of the containment releases will bypass the auxiliary building.

Case 1: This case includes those accidents in which the auxiliary building failed prior to vessel failure. The quantification for this case is:

Branch 1:	nOAux-VB	0.00
Branch 2:	OAux-VB	1.00

Case 2: This case includes those accidents in which the auxiliary building did not fail prior to vessel failure, the reactor is not open to the auxiliary building, and the containment was closed prior to the onset of core damage. In this situation, any releases of steam and/or non-condensibles will bypass the auxiliary building. The quantification for this case is:

Branch 1:	nOAux-VB	1.00
Branch 2:	OAux-VB	0.00

Case 4: This case includes those accidents in which either the containment or the reactor vessel (i.e., via open MSIV or LOCA in auxiliary building) is open to the auxiliary building. Results from MELCOR indicated that in all the relevant scenarios, the auxiliary building will fail at the time of vessel failure from pressure loads that accompany vessel failure or will fail shortly after vessel failure from the accumulation of steam and non-condensible. The quantification for this case is:

Branch 1:	nOAux-VB	0.00
Branch 2:	OAux-VB	1.00

Question 53. What is the status of dc power late in the accident?
Number of Branches: 2
Number of Cases: 5
Number of Cases Sampled: None

The branches for this question are:

1. nDC-Late DC power is not available during the late time regime (i.e., after vessel failure).
2. DC-Late DC power is available during the late time regime (i.e., after vessel failure).

This question determines the availability of dc power during the late time regime. The availability of dc power is important since dc power is required to restore offsite power to the plant (see Question 24). Restoration of offsite power during the late time regime is considered from the time of vessel failure until two hours after vessel failure. Since the restoration of offsite power depends of the availability of dc power, the time regime used for restoration of offsite power was also used for dc power. The probability that dc power is not available during the late time regime is based on a distribution was developed for the NUREG-1150 Grand Gulf plant analysis that models the failure probability of the station batteries versus time for SBO sequences [Wheeler, et al., 1989]. The failure probabilities used in this question are conditional on dc power being available at the time of vessel failure.

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Case 1: This case includes those accidents in which ac power was available before vessel failure. With ac power available, the battery chargers supply the necessary dc power and battery depletion is not an issue. For the PDSs analyzed in this study, there were no failures of the dc bus and, therefore, with ac power available, dc power is also assured. The quantification for this case is:

Branch 1:	nDC-Late	0.0
Branch 2:	DC-Late	1.0

Case 2: This case includes those accidents in which dc power was not available at the time of vessel failure and, therefore, is not available after vessel failure (no credit is given for recovery of the station batteries). The quantification for this case is:

Branch 1:	nDC-Late	1.0
Branch 2:	DC-Late	0.0

Case 3: This case includes those accidents initiated by a loss of offsite power that result in a station blackout during Time Window 1. In these accidents, dc power is available at the time of vessel failure. For this case the following times are used for the start and end of the time period considered for restoration of ac power during the late time period: 14.4 hours and 16.4 hours. Therefore, the value for Branch 2 is probability that dc power is available at 16.4 hours given that it was available at 14.4 hours. The quantification for this case is:

Branch 1:	nDC-Late	0.18
Branch 2:	DC-Late	0.82

Case 4: This case includes those accidents initiated by a loss of offsite power that result in a station blackout during Time Window 2. In these accidents, dc power is available at the time of vessel failure. For this case the following times are used for the start and end of the time period considered for restoration of ac power during the late time period: 12.6 hours and 14.6 hours. Therefore, the value for Branch 2 is probability that dc power is available at 14.6 hours given that it was available at 12.6 hours. The quantification for this case is:

Branch 1:	nDC-Late	0.16
Branch 2:	DC-Late	0.84

Case 5: This case is not used.

Question 54. Is offsite power restored after vessel breach core damage?
Number of Branches: 2
Number of Cases: 5
Number of Cases Sampled: 2

The branches for this question are:

1.	nAC-Late	Neither offsite nor onsite power is available after vessel breach.
2.	OSP-Late	Only offsite power is available after vessel breach. (Note, this situation does not occur in this analysis)
3.	AC-Late	AC power is available after vessel breach.

This question determines the availability of ac power during the late time period after vessel failure. The availability of ac power is important because it will determine which systems can be used to mitigate the accident (e.g., containment heat removal systems and Containment Vent System). Restoration of offsite power during the late time regime is considered from the time of vessel failure until two hours after vessel failure. Two hours was selected because containment venting will typically be required within two hours of vessel breach and for situations where the containment is open, the containment sprays must be initiated shortly after vessel breach if they are to be effective as a means of trapping radioactive material released during the interactions between the core debris and the concrete cavity.

The probability of recovering offsite power during a given time period is determined by sampling from a set of distributions for power recovery [Iman and Hora, 1988] (also see Volume 2 of this report). These distributions reflect the type of electrical switchyard at Grand Gulf, as explained in NUREG-1032 [Baranowsky, 1985]. To get ac power to the safety systems, not only does ac power have to be restored to the site, but dc power must be available as well. DC power is required for circuit breaker control power; once the station batteries have been depleted, it is very difficult to get ac power back to the safety systems. Although the circuit breakers can be moved manually, this procedure is very complicated and slow. Thus, for the time frame considered in this analysis, it is assumed that once dc power is lost, ac power cannot be recovered. The generation of the power recovery curves used in this analysis is discussed in Appendix G of Volume 2, Part 2 of this report.

Case 1: This case includes those accidents in which offsite and onsite power are available at the start of the accident and, therefore, ac power is still available. The quantification for this case is:

Branch 1:	nAC-Late	0.0
Branch 2:	OSP-Late	0.0
Branch 3:	AC-Late	1.0

Case 2: This case includes those accidents in which dc power is not available. The lack of dc power implies that both offsite an onsite ac power are unavailable. Without dc power, offsite power cannot be recovered. Furthermore, in this analysis, no credit is given for recovery of the emergency diesel generator during the core damage process. Therefore, in this case all ac power is unavailable. The quantification for this case is:

Branch 1:	nAC-Late	1.0
Branch 2:	OSP-Late	0.0
Branch 3:	AC-Late	0.0

Case 3: This case includes those accidents initiated by a loss of offsite power that result in a station blackout during Time Window 1. DC power is available at the time of vessel failure. For this case the following times are used for the start and end of the time period considered for restoration of ac power during the late time period: 14.4 hours and 16.4 hours. Therefore, the value for Branch 3 is probability that ac power is recovered during the late time regime given that ac power was not available at the time of vessel failure. The quantification for this case is:

Branch 1:	nAC-Late	0.78	
Branch 2:	OSP-Late	0.0	
Branch 3:	EAC-Late	0.22	Power recovery distribution.

Case 4: This case includes those accidents initiated by a loss of offsite power that result in a station blackout during Time Window 2. DC power is available at the time of vessel failure. For this case the following times are used for the start and end of the time period considered for restoration of ac power during the late time period: 12.6 hours and 14.6 hours. Therefore, the value for Branch 3 is probability that ac power is recovered during the late time regime given that ac power was not available at the time of vessel failure. The quantification for this case is:

Branch 1:	nAC-Late	0.73	
Branch 2:	OSP-Late	0.0	
Branch 3:	EAC-Late	0.27	Power recovery distribution.

Case 5: This case is not used.

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Question 55. Is the core debris in the cavity coolable?
Number of Branches: 3
Number of Cases: 8
Number of Cases Sampled: 0

The branches for this question are:

1.	nCCI	The core debris is cooled in the cavity; there are no Core-Concrete Interactions (CCI).
2.	FldCCI	The core debris is not coolable, CCI occurs in a flooded cavity.
3.	DryCCI	The core debris is not coolable; CCI occurs in a dry cavity.

This question determines whether the core debris released from the reactor vessel is cooled by water in the cavity below the reactor vessel. If the debris is not coolable, this question also determines whether the core-concrete interactions take place in a dry or flooded cavity. The coolability of the core debris is important because it will affect the amount of radioactive material released from the core debris. If the core debris is coolable, it is assumed in the source term analysis that no radioactive material is released from the core debris in the cavity. If the core debris is not in a coolable configuration but the CCI occurs in a flooded cavity, some of the radioactive material released from the core debris will be trapped in the overlying pool of water.

The quantification of this question is based on probabilities developed in the NUREG-1150 study [Harper, et al., 1994]. The core debris bed will not be coolable if it is finely fragmented or if the debris reagglomerates after vessel failure. The coolability of the debris that is released at vessel breach as well as that of the debris slowly released following vessel breach is considered in this question. The core debris must be coolable in both cases if the CCI is not to occur. If any of the core debris released to the cavity, either at the time of vessel failure or after vessel failure, is not coolable, CCI will be initiated. Once CCI has been established, it is assumed that all of the material in the reactor cavity participates in CCI. Thus, the coolability of the debris released after vessel breach is important only if the debris released at vessel breach is coolable.

The likelihood that the debris released after vessel breach is coolable is the same for all the cases that have water in the cavity. The debris released after vessel breach was most likely solid at vessel breach. As the decay heat melts this remaining debris, it is released from the vessel. Thus, it is likely that this debris will be released with a low amount of superheat. It is expected that the debris bed that forms from this material will consist of large particles that may not be entirely molten. Assuming there is water in the cavity, it is likely that the debris bed will be coolable.

If the RPV fails at high pressure, most of the debris will be ejected from the cavity. Although this material will be finely fragmented, it will be coolable because it is spread throughout the drywell in a thin layer. In this case, the coolablility of the debris in the cavity is based on the material that is released after vessel breach.

Case 1: This case includes those accidents in which the vessel does not fail and, therefore, there is no CCI. The quantification for this case is:

Branch 1:	nCCI	1.0
Branch 2:	FldCCI	0.0
Branch 3:	DryCCI	0.0

Case 2: This case includes those accidents in which the vessel fails and the core debris is released into a dry cavity. Core cooling was not restored to the vessel during core damage. Furthermore, if the vessel was pressurized during core damage, none of the low pressure injection systems are aligned to automatically inject into the vessel when the vessel depressurizes at the time of vessel failure. Since there is no water in the cavity, CCI in a dry cavity is assured. The quantification for this case is:

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Branch 1:	nCCI	0.0
Branch 2:	FldCCI	0.0
Branch 3:	DryCCI	1.0

Case 3: This case includes those accidents in which the vessel, which is at system pressure, fails and the core debris is released into a dry cavity. Core coolant is being provided to the vessel and will enter the cavity coincident with the core debris. Since the vessel fails at high pressure, most of the debris released at vessel breach is ejected from the cavity. Thus, the coolability of the debris in the cavity is based on the material released after vessel failure. As explained above, it is likely that the debris released after vessel failure is coolable. If the debris bed is not cooled, CCI will occur in a flooded cavity. The quantification for this case is:

Branch 1:	nCCI	0.80
Branch 2:	FldCCI	0.20
Branch 3:	DryCCI	0.00

Case 4: This case includes those accidents in which the debris is released from the vessel at low pressure into a dry cavity. Core coolant is being provided to the vessel and will enter the cavity coincident with the core debris. Since the vessel fails at low pressure, most of the debris released at vessel breach will remain in the cavity. Even though water is released from the vessel at the time of vessel failure, the debris will contact essentially a dry floor, and CCI is likely to initiate. Once CCI is established, gases and steam flow upward through the debris and create a resistance to water that would penetrate and cool the debris. If the debris bed is not cooled, which is likely, CCI will occur in a flooded cavity. The quantification for this case is:

Branch 1:	nCCI	0.16
Branch 2:	FldCCI	0.84
Branch 3:	DryCCI	0.00

Case 5: This case includes those accidents in which the debris is released from the vessel at high pressure into a flooded cavity. Since the vessel fails at high pressure, most of the debris released at vessel breach is ejected from the cavity. Thus, the coolability of the debris in the cavity is based on the material released after vessel failure. As explained above, it is likely that the debris released after vessel failure is coolable. If the debris bed is not cooled, CCI will occur in a flooded cavity. The quantification for this case is:

Branch 1:	nCCI	0.80
Branch 2:	FldCCI	0.20
Branch 3:	DryCCI	0.00

Case 6: This case includes those accidents in which the debris is released from the vessel at low pressure into a flooded cavity. In this case, the debris also has a high amount of super heat. Since the vessel fails at low pressure, most of the debris released at vessel breach will remain in the cavity. Even though there is water in the cavity, it is likely that the core debris will agglomerate because of its high superheat. Thus, it is likely that the core debris released at the time of vessel failure will not be coolable. Thus, even though it is likely that the debris release after vessel breach is coolable, it is likely that CCI will be initiated by the debris released at the time of vessel breach. If the debris is not cooled, CCI will occur in a flooded cavity. The quantification for this case is:

Branch 1:	nCCI	0.16
Branch 2:	FldCCI	0.84
Branch 3:	DryCCI	0.00

Case 7: This case includes those accidents in which the debris is released from the vessel at low pressure into a flooded cavity. In this case, the debris has a small amount of super heat. Since the vessel fails at low pressure,

most of the debris released at vessel breach will remain in the cavity. Even though there is water in the cavity and the debris has a small amount of superheat, it is uncertain whether the debris released at vessel breach will be coolable. If the debris is not cooled, CCI will occur in a flooded cavity. The quantification for this case is:

Branch 1:	nCCI	0.40
Branch 2:	FldCCI	0.60
Branch 3:	DryCCI	0.00

Case 8: This case is not used.

Question 56. Do the operators vent the containment after vessel breach?
Number of Branches: 2
Number of Cases: 3
Number of Cases Sampled: 1

The branches for this question are:

1.	nVnt-Late	The containment is not vented after vessel breach.
2.	Vnt-Late	The operators vent the containment after vessel breach.

This question determines whether the operators vent the containment after vessel breach. The status of containment venting is important because opening the containment vent establishes a path from the containment to the environment that bypasses the auxiliary building which allows airborne radioactive material in the containment to escape directly to the environment. The size of the vent path is equivalent to a rupture in the containment. The Grand Gulf Emergency Procedures (EP-3) direct the operators to vent the containment if the containment pressure is greater than 20 psig and cannot be maintained below 22 psig. While the procedures allow the operators to close the vent once the pressure drops below 20 psig, the operators would have to open the vent again later in the accident since containment cooling is not available and the steam and non-condensibles generated during the accident would again cause the containment pressure to exceed 22 psig. This analysis makes no attempt to model the opening and the closing of the vent to maintain the pressure below 20 psig. Instead, it is assumed that once the vent is opened, it remains open for the duration of the accident.

Case 1: This case includes those accidents in which either the containment vent system is not available or the containment pressure is below the vent threshold pressure (i.e., containment already ruptured or reactor vessel open to auxiliary building). The quantification for this case is:

Branch 1:	nVnt-Late	1.00
Branch 2:	Vnt-Late	0.00

Case 2: This case includes those accidents in which the pressure in the containment is above the vent threshold. MELCOR calculations for the relevant scenarios indicate that the containment pressure will exceed the vent threshold at the time of vessel failure or shortly after vessel failure. The containment vent system is either available or was recoverable and ac power is recovered (i.e., either during core damage or after vessel breach). The Grand Gulf Emergency Procedures will direct the operators to vent the containment in this situation. This case is sampled; the distribution that characterizes the uncertainty in the probability that the operators will fail to vent the containment was developed in the HRA analysis and is discussed in Appendix B.3. The quantification (mean value) for this case is:

Branch 1:	nVnt-Late	0.031	Lognormal distribution with a mean of 0.031 and an error factor of 5.
Branch 2:	Vnt-Late	0.969	

Case 3: This case includes those accidents in which the pressure in the containment is above the vent threshold, however, ac power is not available. Without ac power the containment vent cannot be opened. The quantification for this case is:

Branch 1:	nVnt-Late	1.0
Branch 2:	Vnt-Late	0.0

Question 57. Does the containment fail late in the accident?
Number of Branches: 3
Number of Cases: 2
Number of Cases Sampled: 0

The branches for this question are:

1. nCF-Late — The containment does not fail late in the accident.
2. CF-Rpt-Late — The containment fails in the rupture mode (Nominal hole size is 1 ft^2) late in the accident.
3. CF-Lk-Late — The containment fails in the leak mode (Nominal hole size is 0.1 ft^2) late in the accident.

This question determines whether the containment fails late in the accident. The status of the containment integrity is important because failure of the containment will result in path for radionuclides to enter the environment. Similar to the NUREG-1150 Grand Gulf plant analysis [Brown, et al., 1990], two failure sizes are defined: a leak and a rupture. The containment failure issue is discussed in Question 35.

For the PDSs considered in this analysis, the accidents in which the containment could be closed prior to the onset of core damage also had the characteristic that containment cooling was not available and could not be recovered. Therefore, in all of the accidents addressed in this study, the containment will be breached at some point during the accident.

Case 1: This case includes those accidents in which the containment was ruptured either before vessel failure or at the time of vessel failure or the containment was vented after vessel failure. In any case, the containment boundary is already breached. The quantification for this case is:

Branch 1:	nCF-Late	1.00
Branch 2:	CF-Rpt-Late	0.00
Branch 3:	CF-Lk-Late	0.00

Case 2: This case includes those accidents in which the containment was intact at the end of the vessel breach time regime and was not vented during the late time regime. Without containment cooling, the containment will fail from the accumulation of steam and non-condensibles generated during the accident. The quantification for this case is:

Branch 1:	nCF-Late	0.00
Branch 2:	CF-Rpt-Late	0.50
Branch 3:	CF-Lk-Late	0.50

Question 58. What is the status of containment integrity late in the accident?
Number of Branches: 2
Number of Cases: 2
Number of Cases Sampled: 0

The branches for this question are:

1. nOCnt-Late — The containment does not remain intact throughout the accident.

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2. OCnt-Late The containment is still intact at the end of the accident.

This question summarizes the status of containment integrity at the end of the accident. This question summarizes failures that occurred before vessel failure, at the time of vessel failure, and after vessel failure. The question does not address the size of the breach.

Case 1: This case includes those accidents in which either the containment was breached during the accident. The quantification for this case is:

Branch 1:	nOCnt-Late	0.00
Branch 2:	OCnt-late	1.00

Case 2: This case includes all other accidents in which the containment is still intact at the end of the accident. The quantification for this case is:

Branch 1:	nOCnt-Late	1.00
Branch 2:	OCnt-Late	0.00

Question 59. What is the size of the containment opening at the end of the accident?
Number of Branches: 3
Number of Cases: 4
Number of Cases Sampled: 0

The branches for this question are:

1. Cnt-Rpt-Late The containment opening is the size of a rupture (nominal size is 1 ft^2).
2. Cnt-Lk-Late The containment opening is the size of a leak (nominal size is 0.1 ft^2).
3. Cnt-NF-Late The containment is still intact after vessel failure.

This question summarizes the size of the containment opening at the end of the accident.

Case 1: This case includes those accidents in which the containment is still intact at the end of the accident. The quantification for this case is:

Branch 1:	Cnt-Rpt-Late	0.00
Branch 2:	Cnt-Lk-Late	0.00
Branch 3:	Cnt-NF-Late	1.00

Case 2: This case includes those accidents in which either the containment was rupture prior to or during vessel failure, vented after vessel failure, or was ruptured late in the accident from the accumulation of steam and non-condensibles. The quantification for this case is:

Branch 1:	Cnt-Rpt-Late	1.00
Branch 2:	Cnt-Lk-Late	0.00
Branch 3:	Cnt-NF-Late	0.00

Case 3: This case includes those accidents in which loads caused a leak containment at some point during the accident. The quantification for this case is:

Branch 1:	Cnt-Rpt-VB	0.00
Branch 2:	Cnt-Lk-VB	1.00
Branch 3:	Cnt-NF-VB	0.00

Case 4: This case is not used.

References for Appendix B.1

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[Brown, et al., 1990] T. D. Brown et al., "Evaluation of Severe Accident Risks: Grand Gulf Unit 1," NUREG/CR-4551, SAND86-1309, Vol. 6, Rev. 1, Sandia National Laboratories, December 1990.

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[USNRC, 1975] U. S. Nuclear Regulatory Commission, "Reactor Safety Study - An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

B.2 Listing of POS 5 APET

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A listing of the APET is provided in this subsection. The APET in this format is provided as input to the EVNTRE code which is used to evaluate the event tree. The logic that is used to form the accident progression bins is provided near the end of the APET listing and is identified by the keyword "Binning".

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APET for POS 5, Ver. 2
One-Eval 1.0
$========================PDS Definitions===

What is the Plant Damage State?

	PDS1-1 (SFP1-1)	PDS1-2 (SFP1-2)	PDS1-3 (SFP1-3)	PDS1-4 (SFP1-4)	PDS1-5 (SFP1-5)	PDS2-1 (SFP2-1)	PDS2-2 (SFP2-2)	PDS2-3 (SFP2-3)	PDS2-4 (SFP2-4)	PDS2-5 (SFP2-5)	PDS2-6 (SFP2-6)	PDS3-1 (Excess)
$	0.019	0.015	0.032	0.005	0.008	0.17	0.242	0.054	0.104	0.007	0.006	0.338

What is the status of electric power at core damage (PDS Char. 1)?

	aOSP	OSP-nDIVAC	nOSP
PDS1-2 + PDS1-3 + PDS1-4	0.0	0.0	1.0
PDS2-2 + PDS2-3	0.0	0.0	1.0
Otherwise	1.0	0.0	0.0

What is the status of dc power at core damage (PDS Char. 1)?

	nDC-BCD	aDC-BCD
PDS1-2 + PDS1-3 + PDS2-2	1.0	0.0
Otherwise	0.0	1.0

What is the status of high pressure injection at core damage (PDS Char. 2)?

	nHPInj	rHPInj	HPInj
	1.0	0.0	0.0

What is the status of low pressure injection at core damage (PDS Char. 2)?

	nLPInj	nLPInj-op	rLPInj	aLPInj
PDS1-2 + PDS1-3 + PDS1-4	0.0	0.0	1.0	0.0
PDS2-2 + PDS2-3	0.0	0.0	1.0	0.0
PDS2-6	0.0	1.0	0.0	0.0
Otherwise	1.0	0.0	0.0	0.0

What is the status of containment sprays and SPC at core damage (PDS Char. 3)?

	nCS	rCS	alignCS	autoCS
PDS1-2 + PDS1-3 + PDS1-4	0.0	1.0	0.0	0.0
PDS2-2 + PDS2-3	0.0	1.0	0.0	0.0
Otherwise	1.0	0.0	0.0	0.0

What is the suppression pool level at the onset of core damage (PDS Char. 4)?

	SPL-Lo	SPL-Strain
PDS2-6	0.0	1.0

Otherwise	1.0	0.0

What is the suppression pool temperature at the onset of core damage (PDS Char. 5)?

	SPT-Sub	SPT-Sat
PDS1-3	0.0	1.0
Otherwise	1.0	0.0

What is the status of the reactor head vent at the onset of core damage (PDS Char. 6)?

	RPV-nVnt	RPV-OVnt
PDS1-5 + PDS2-4 + PDS2-6	1.0	0.0
Otherwise	0.0	1.0

What is the status of the RPV integrity at the onset of core damage (PDS Char. 7)?

	RPV-HiP	RPV-LoP	RPV-LOCA	RPV-ILOCA	RPV-oMSIV
PDS1-1 + PDS2-1 + PDS3-1	0.0	0.0	1.0	0.0	0.0
PDS1-2 + PDS1-4	0.0	0.0	0.0	1.0	0.0
PDS2-2 + PDS2-3	0.0	0.0	0.0	1.0	0.0
PDS2-6	0.0	0.0	0.0	0.0	1.0
PDS1-5 + PDS2-4	0.0	1.0	0.0	0.0	0.0
Otherwise	1.0	0.0	0.0	0.0	0.0

What is the status of the containment access penetrations at the onset of core damage (PDS Char. 8)?

	o-LPersLk	LPersLk-Unk
PDS2-5 + PDS2-6	0.0	1.0
Otherwise	1.0	0.0

What is the status of the containment vents system at the onset of core damage (PDS Char. 9)?

	nCVS	rCVS	aCVS
PDS1-2 + PDS1-3 + PDS1-4	0.0	1.0	0.0
PDS2-2 + PDS2-3	0.0	1.0	0.0
PDS2-5 + PDS2-6	0.0	0.0	1.0
Otherwise	1.0	0.0	0.0

When does core damage occur (PDS Char. 10)?

	TCD2	TCD2p3	TCD3	TCD5	TCD6	TCD7	TCD9	TCD12
PDS1-1	1.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS2-1	0.0	1.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS1-2 + PDS1-4	0.0	0.0	1.0	0.0	0.0	0.0	0.0	0.0
PDS2-2 + PDS2-3	0.0	0.0	0.0	1.0	0.0	0.0	0.0	0.0
PDS2-6	0.0	0.0	0.0	0.0	1.0	0.0	0.0	0.0
PDS1-5 + PDS3-1	0.0	0.0	0.0	0.0	0.0	1.0	0.0	0.0
PDS2-4	0.0	0.0	0.0	0.0	0.0	0.0	1.0	0.0

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PDS1-3 + PDS2-5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.0	
Otherwise	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.0	$NA

While in POS 5, when does the initiating event occur (PDS Char. 11)?

		IE-Win1	IE-Win2	IE-Win3	
PDS1-1 + PDS1-2 + PDS1-3 + PDS1-4 + PDS1-5	1.0	0.0	0.0		
PDS2-1 + PDS2-2 + PDS2-3 + PDS2-4		0.0	1.0	0.0	
PDS2-5 + PDS2-6		0.0	1.0	0.0	
PDS3-1		0.0	0.0	1.0	
Otherwise		1.0	0.0	0.0	$This case should not be used (NA).

$================================ End of PDS Definitions ===========================

$

$================================Summary Questions ===========================

What type of event initiates the accident? $ This question summarizes the type of IE.

	IE-LOCA	IE-SBO	IE-Other	
PDS1-1 + PDS2-1 + PDS3-1	1.0	0.0	0.0	$ (1): IE is a LOCA.
PDS1-2 + PDS1-3 + PDS1-4	0.0	1.0	0.0	$ (2): IE is a SBO (TW-1)
PDS2-2 + PDS2-3	0.0	1.0	0.0	$ (3): IE is a SBO (TW-2)
Otherwise	0.0	0.0	1.0	$ (4): All other IEs.

What is the pressure in the RPV at the time of core damage? $ This question summarizes the status of the RPV pressure

	RPV-HiP-BCD	RPV-LoP-BCD	
RPV-HiP	1.0	0.0	$ (1): RPV is pressurized BCD (i.e., SRVs closed).
Otherwise	0.0	1.0	$ (2): RPV is depressurized by either a breach or open SRVs.

How much water is in the reactor pedestal cavity at the time of core damage?

	Cav-Dry-BCD	Cav-Fld-BCD	
IE-LOCA	0.00	1.00	$ (1): Containment flooded in LOCAs
PDS1-5 + PDS2-4	0.00	1.00	$ (2): CNMT flooded in these PDSs
Otherwise	1.00	0.00	$ (3): Cavity esscentially dry.

$================================Events before CD ===========================

Is the containment equipment hatch opened at the start of the accident? $ This question addresses the status of the equipment hatch before CD.

nOCnt-S	OCnt-S	
		$ In this analysis it is assumed that the containment equipment hatch
0.0	1.0	$ is always open at the start of the accident.

Do the operators close the containment before core damage?

	nOCnt-BCD	OCnt-BCD	
nOCnt-S	1.0	0.0	$ (1): The equip. hatch was closed at the start of the accident are remains closed.

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o-LPersLk	0.0	1.0	$ (2): Low personnel lock was open at CD by definition of the PDS
IE-SBO	0.0	1.0	$ (3): The hatch was open prior to IE - SBO precludes closure
TCD2 + TCD2p3 + TCD3	0.0	1.0	$ (4): CD occurs within 3 hours of the IE - not enough time to close containment
Otherwise	(Excess)	(HEP5-CNT)	$ (5): CD occurs > 5 hours of IE - Cnt closure possible.

Data: HEP5-CNT 0.102 BOUNDED LOGNORMAL 0.102 5 0.001 0.999

Does the auxiliary building fail before core damage?

	nOAux-BCD	OAux-BCD	
RPV-ILOCA 0.0	1.0	$ (1): ILOCA in Aux Bldg, steam from LOCA fails Aux Bldg	
RPV-oMSIV 0.0	1.0	$ (2): RPV Level at steam lines; boiloff of coolant will fail Aux Bldg	
nOCnt-S	1.0	0.0	$ (3): The CNMT is closed at the start of the accident,thus, no early Aux Bldg failure
Otherwise	1.0	0.0	$ (4): Cnt is open BCD and SP subcooled

What is the status of the drywell before core damage? $ In this analysis it is assumed that the drywell remains open for the duration

	Cls-DW-BCD	Op-DW-BCD	
			$ of the accident.
	0.0	1.0	

Do the operators turn on the HIS before core damage?

	nHIS-BCD	HIS-BCD	$ This question determines whether the HIS was actuated prior to CD
IE-SBO	1.0	0.00	$ (1): SBO, HIS will not operate without Div 1 or 2 power. It is assumed
			$ the operators will not actuate a system that won't work.
Otherwise	(HEP-HIS-nSBO)	(Excess)	$ (2): AC power available. EOP call for the HIS to be turned on

Data: HEP-HIS-nSBO 0.054 LOGNORMAL 0.054 5

$==============================Events During Core Damage=========================

Do the station batteries depleted during core damage?

	nDC-CD	DC-ECD	DC-LCD	
/IE-SBO	0.000	0.000	1.000	$ (1): Not a SBO, thus dc power available
nDC-BCD	1.000	0.000	0.000	$ (2): SBO & dc power failed before CD
IE-Win1	0.011	0.242	0.747	$ (3): SBO during TW-1 (dc avail. @ 3.5 hr)
IE-Win2	0.015	0.103	0.882	$ (4): SBO during TW-2 (dc avail. @ 5.5 hr)
Otherwise	0.000	1.000	0.000	$ (5): This case shouldn't be used.

Is offsite power restored during core damage?

	nAC-CD	OSP-CD	EAC-CD	LAC-CD	
/IE-SBO	0.0	0.0	1.0	0.0	$ (1): Not a SBO, OSP available
nDC-CD	1.0	0.0	0.0	0.0	$ (2): No dc power, thus, no ac
IE-Win1 & DC-ECD	(Excess)	0.0	(AC-ECD-TW1)	0.0	$ (3): SBO in TW-1, dc early, OSP recoverable
IE-Win1	(Excess)	0.0	(AC-ECD-TW1)	(AC-LCD-TW1)	$ (4): SBO in TW-1, dc available all of CD, OSP recvr

```
IE-Win2 & DC-ECD        (Excess)    0.0     (AC-ECD-TW2)  0.0           $ (5): SBO in TW-2, dc early, OSP recoverbable
IE-Win2                 (Excess)    0.0     (AC-ECD-TW2)  (AC-LCD-TW2)  $ (6): SBO in TW-2, dc available all of CD, OSP recvr
Otherwise               1.0         0.0     0.0           0.0           $ (7): This case shouldn't be used.

Data: AC-ECD-TW1 0.5 File C:\LPS\POS5\CALC\LHS200\APET\ACRECBE.LSP   $ Distributions from PWRRECBE.FOR, uses Power recovery curves
Data: AC-LCD-TW1 0.5 File C:\LPS\POS5\CALC\LHS200\APET\ACRECBE.LSP   $ generated by MODELFOR
Data: AC-ECD-TW2 0.5 File C:\LPS\POS5\CALC\LHS200\APET\ACRECBE.LSP
Data: AC-LCD-TW2 0.5 File C:\LPS\POS5\CALC\LHS200\APET\ACRECBE.L..P
```

Is the RPV isolated during core damage?

```
                                            Op-RPV-CD     Cls-RPV-CD
IE-LOCA                                     1.0           0.0           $ (1): IE was a LOCA
RPV-oMSIV                        1.0        0.0           $ (2): MSIVs are open - no credit for closing during CD
RPV-ILOCA & IE-SBO & EAC-CD                 (Excess)      (ISO-SDC)     $ (3): Failure of SDC F008 and F009 to close due to severe invironment
RPV-ILOCA                        1.0        0.0           $ (4): SDC F008 and F009 open - no ac power
Otherwise                                   0.0           1.0           $ (5): RPV integrity maintained.

Data: ISO-SDC 0.9 MAXIMUM ENTROPY 0.5 0.9 1.0
```

Do the operators initiate containment sprays during core damage?

```
                                            nCS-CD        CS-CD
nCS                                         1.0           0.0           $ (1) CS are not available and not recoverable
OCnt-BCD & /autoCS                          1.0           0.0           $ (2) Containment is open, Cnt pressure control is not an issue
Op-RPV-CD & ( RPV-oMSIV + RPV-ILOCA) #                    $ (3) RPV releases bypass containment and CS will not autostart.
& /autoCS                                   1.0           0.0           $       Thus, no need to use sprays (i.e., cnt pres control not important)
alignCS                                     1.0           0.0           $ (3) LPCI aligned in SDC-previous errors preclude credit for realignment
rCS & ( nAC-CD + OSP-CD)         1.0        0.0           $ (4) CS recoverable, however, no ac power
rCS                                         0.01          0.99          $ (5) CS recoverable & ac power is recovered-EP will require CS
Otherwise                                   0.01          0.99          $ (6) CS available-Cnt Press. Control EP will require CS
```

Do the operators depressurize the RPV during core damage?

```
                                 RPV-HiP-CD        RPV-LoP-CD
Op-RPV-CD  0.0                   1.0               $ (1): RPV open via a breach (i.e., LOCA, MSIVs or I-LOCA)
nDC-CD                           1.0               0.0            $ (2): no dc power, thus, can't open SRVs
RPV-ILOCA  (HEP-PRPV-HiP)  (Excess)                $ (3): HRA Quantification - Do the operators open the SRVs following
                                                                  $ the recovery subsequent isolation of the SDC F008 and F009 valves
RPV-LoP-BCD                      0.0               1.0            $ (4): RPV at low pressure before CD and still at low pressure
Otherwise                        (HEP-PRPV-HiP)    (Excess)       $ (5): RPV at high pressure. EP call for depressurization.


Data: HEP-PRPV-HiP 0.054 BOUNDED LOGNORMAL 0.054 5 0.001 0.999
```

What is the status of the SRV vacuum breakers during core damage?

	Op-SRV-Bkr	Cls-SRV-Bkr	
RPV-LoP-BCD	0.0	1.00	$ (1): SRVs open, very few SRV cycles, if any, thus, unlikely to fail
RPV-OVnt	0.0	1.00	$ (2): The open vent will limit the number of SRV cycles, thus, failure unlikely
Otherwise	(SRV-VBkr)	(Excess)	$ (3): RPV vent closed and RPV is pressurized - NUREG-1150 Quantification

Data: SRV-VBkr 0.25 UNIFORM 0.01 0.5 $ Based on NUREG-1150 Grand Gulf plant analysis (NUREG/CR-4551, Vol 6, Part 2)

Is core cooling restored during core damage?

	E-CorCool	L-CorCool	n-CorCool	
HPInj	1.0	0.0	0.0	$ (1): This case will never be used.
rHPInj & EAC-CD	1.0	0.0	0.0	$ (2): This case will never be used.
rHPInj & LAC-CD	0.0	1.0	0.0	$ (2): This case will never be used.
RPV-HiP-CD 0.0	0.0	1.0	$ (3): RPV at high pressure and no high pressure makeup	
nLPInj-op	0.0	0.0	1.0	$ (4): Low pressure makeup is unavaileble due to operator error.
rLPInj & EAC-CD & (Cls-RPV-CD + RPV-oMSIV + RPV-LOCA) #			$ (5): LPInj recoverable early, ac power avail. & not unisolated ILOCA	
	1.0	0.0	0.0	$ Recvry of LPInj w/ an unisol. ILOCA would drain SP & fld Aux Bldg
rLPInj & LAC-CD & (Cls-RPV-CD + RPV-oMSIV + RPV-LOCA) #			$ (6): LPInj recoverable late, ac power avail. & not unisolated ILOCA	
	0.0	1.0	0.0	$ Recvry of LPInj w/ an unisol. ILOCA would drain SP & fld Aux Bldg
Otherwise	0.0	0.0	1.0	$ (7): Either no injection, no ac power, or RPV not isolated.

What is the peak hydrogen concentration in the containment during CD?

	H2<4	H2<8	H2<12	H2<16	H2>16	
E-CorCool	1.000	0.000	0.000	0.000	0.000	$ (1): Injection before significant H2 production
RPV-HiP-CD & nOCnt-BCD & /RPV-oMSIV & /RPV-ILOCA #						
	0.014	0.095	0.127	0.207	0.557	$ (2): RPV at HiP and CNMT Closed,
nOCNT-BCD & /RPV-oMSIV & /RPV-ILOCA #						
	0.010	0.060	0.140	0.205	0.585	$ (3): RPV at LoP and CNMT Closed,
RPV-HiP	0.066	0.213	0.303	0.243	0.175	$ (4): RPV at HiP, CNMT Open to Aux. Bldg
Otherwise	0.040	0.205	0.325	0.250	0.180	$ (5): RPV at LoP, CNMT Open to Aux. Bldg

What is the fraction of zirconium that is oxidized in the vessel during core damage?

	ZrOxid<21	ZrOxid>21	
H2>16 + (OCnt-BCD + RPV-oMSIV + RPV-ILOCA) & /H2<4 & /H2<8 #			$(1) CNMT closed and H2 concentration greater than 16% or CNMT open
	0.00	1.00	$ and H2 concentration is greater than 8%
Otherwise	1.00	0.00	$ (2) CNMT closed and H2 < 16% or CNMT open and H2< 8%

Do the operators turn on the HIS during CD ?

	nHIS-CD	HIS-CD	
			$ This question is primarially concerned with whether the operators turn HIS on following
			$ the recovery of ac power but before the H_2 exceeds the safe zone.
HIS-BCD	0.0	1.0	$ (1): HIS prior to core damage, thus, still on
nHIS-BCD & / IE-SBO	1.0	0.0	$ (2): Not a SBO, operators fail to turn on HIS, thus, still off.
EAC-CD + LAC-CD & (H2<4 + H2<8) #			$ (3): SBO, ac recovered early during CD or ac recovered late but

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	(HEP-HIS-nSBO)	(Excess)		$ H2 concentration is less than 8%.
Otherwise	1.0	0.0		$ (4): Either ac power recovered late during CD or wasn't recovered.

Does an uncontrolled hydrogen combustion event occur during CD?

	nBrn-H2	Brn-H2	Brn-Dif	
HIS-CD & /RPV-ILOCA & /RPV-oMSIV	0.0	0.0	1.0	$ (1): Igniter are on with release of H2 to CNMT
H2<4	1.0	0.0	0.0	$ (2): The H2 concentration is < 4%, will not burn $ as a deflagration
/ IE-SBO + IE-SBO & EAC-CD 0.0	(Excess)	(Brn-nSBO)		$ (3): ACpower available (plenty of ignition sources)
IE-SBO & / nAC-CD	0.0	1.0	0.0	$ (4): SBO with late recovery of AC, ignition assured, $ however, concentration not known, assume high
IE-SBO	(Excess)	(Brn-SBO)	0.0	$ (5): SBO, ac not recovered, few ignition sources
Otherwise	1.0	0.0	0.0	$ This case shouldn't be used.

Data: Brn-nSBO 0.75 UNIFORM 0.5 1.0 $ Based on NUREG/CR-4551, Vol 6 (Appendix A)
Data: Brn-SBO 0.5 UNIFORM 0.0 0.75 $ Min & Max values correspond to ignition freq. in NURE/CR-4551, Vol. 6, (Appendix A)

What is the pressure in the containment during CD (no uncontrolled burn)? $ This question addresses the pressure in the containment during

	P-Lo	P-Vnt	
			$ core damage.
OCnt-BCD	1.0	0.0	$ (1): Containment is open, thus, pressure is low
Op-RPV-CD & / RPV-LOCA	1.0	0.0	$ (2): RPV is open to containment or turbine building
Otherwise	1.0	0.0	$ (3): Cnt closed, MELCOR calc. indicate P< 20 psig.

Does the containment fail from quasi-static loads during core damage?

	nCF-CD	CF-Rpt-CD	CF-Lk-CD	
OCnt-BCD + Brn-Dif + nBrn-H2 # + RPV-oMSIV	1.0	0.00	0.00	$ (1) Either Cnt open or Open MSIV, H2 burned with $ HIS, or no deflag.
H2<4 +H2<8 1.0	0.00	0.00		$ (2) Peak H2 concen. is < 8%
H2<12	0.79	0.19	0.02	$ (3) Peak H2 concen. between 8 & 12%
H2<16	0.13	0.49	0.38	$ (4) Peak H2 concen. between 12 & 16 %
H2>16	0.04	0.50	0.46	$ (5) Peak H2 concen. is > 16%
Otherwise	1.0	0.00	0.00	$ (6) This case shouldn't be used.

Do the operators vent the containment during core damage?

	nVnt-CD	Vnt-CD	
nCVS + P-Lo	1.0	0.0	$ (1): Either CVS not available, or CNMT pressure below vent threshold
RPV-HiP & nOCnt-BCD & nCS-CD & RPV-LoP-CD # & (aCVS + rCVS & (EAC-CD + LAC-CD))	(HEP-EVnt)	(Excess)	$ (2): Cnt Pres > 20 psig and CVS availabile and operators previously $ followed CPC EP (i.e., depressurized the RPV)
RPV-HiP & nOCnt-BCD & nCS-CD & RPV-HiP-CD # & (aCVS + rCVS & (EAC-CD +LAC-CD))	1.0	0.00	$ (3): Cnt Pres > 20 psig and CVS availabile and operators previously $ failed to followed CPC EP (i.e., depressurized the RPV)
aCVS + rCVS & (EAC-CD + LAC-CD)	(HEP-EVnt)	(Excess)	$ (4): Cnt Pres > 20 psig and CVS is available, EP requires venting
Otherwise	1.0	0.0	$ (5): Cnt Pres > 20 psig, however CVS not available.

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Data: HEP-EVnt 0.031 LOGNORMAL 0.031 5

What is the status of the containment during core damage?

	nOCnt-CD	OCnt-CD	
OCnt-BCD + Vnt-CD + /nCF-CD	0.0	1.0	$ (1): CNMT open before CD, vented or fails during CD
Otherwise	1.0	0.0	$ (2): CNMT intact.

What is the size of the containment opening during core damage?

	Cnt-Rpt-CD	Cnt-Lk-CD	Cnt-NF-CD	
nOCnt-CD	0.0	0.0	1.0	$ CNMT intact.
OCnt-BCD + CF-Rpt-CD + Vnt-CD	1.0	0.0	0.0	$ CNMT hatches open, vented or failed by ruptured
CF-Lk-CD	0.0	1.0	0.0	$ CNMT failure mode is a leak
Otherwise	0.0	0.0	1.0	$ This case shouldn't be used.

Does the auxiliary building fail during core damage?

	nOAux-CD	OAux-CD	
OAux-BCD	0.0	1.0	$ (1): Aux. Bldg open BCD
(RPV-ILOCA + RPV-oMSIV) & Op-RPV-CD	0.0	1.0	$ (2): RPV open to aux. bldg during CD
nOCnt-BCD	1.0	0.0	$ (3): Cnt closed BCD; any Cnt failure will be above Aux. Bldg
Brn-H2	0.0	1.0	$ (4): H_2 in the Cnt or Aux Bldg without HIS, possible severe burn
Brn-Dif	0.01	0.99	$ (5): H_2 burns via the HIS, thus, less severe burn
RPV-LOCA	1.0	0.0	$ (6): LOCA with Open Cnt, MELCOR shows no Early Aux. Bldg failure
RPV-OVnt + Op-SRV-Bkr	0.0	1.0	$ (7): RPV head vnt or SRV vacuum breaker open (Based on MELCOR)
Otherwise	1.0	0.0	$ (8): Intact RPV, steam condensed in suppression pool

$==============Events Accompanying Vessel Breach===

Is there water in the RPV pedestal cavity just prior to VB? $ This question addressed the amount of water in the pedestal cavity below the RPV

	Cav-Dry	Cav-Fld	$ during CD.
Cav-Fld-BCD 0.0	1.0	$ (1): Cavity flooded before CD, thus, still flooded	
Otherwise	1.0	0.0	$ (2): No other souces of water to cavity, thus, dry

Is the core damage process arrested in the vessel? $ This question addresses the coolability of the core debris in the vessel

	nCDArrest	CDArrest	
(RPV-ILOCA + RPV-LOCA) # & Op-RPV-CD	1.0	0.0	$ (1): I-LOCA not isolated, thus, insufficient core coverage
E-CorCool	0.01	0.99	$ (2): Coolant restored early in the accident => likely that debris is coolable
L-CorCool	(Excess)	(CDArst-L)	$ (3): Coolant restored late in the accident => likely debris is not coolable
Otherwise	1.0	0.0	$ (4): Coolant is not restored => debris is not cooled.

Data: CDArst-L 0.01 MAXIMUM ENTROPY 0.00 0.01 0.5

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What fraction of the core debris would be mobil at VB?

	HiLiqVB	LoLiqVB	
			$ Quantification is based on NUREG-1150.
E-CorCool + L-CorCool	0.025	0.975	$ (1): Coolant restored during core damage
Otherwise	0.100	0.900	$ (2): Coolant either not restored or restored late in the accident

Does a large in-vessel steam explosion occur? $ This question addresses the likelihood that a large steam explosion occurs in the vessel

	nVStmExp	VStmExp	
			$ Quantification: Mean value is based on NUREG-1150.
E-CorCool	1.0	0.0	$ (1): Coolant restored early => not enough core debris to result in a large steam explosion
RPV-LoP-CD (Excess)	(StmExp-LoP)	$ (2): RPV is at low pressure and coolant is restored late or not at all.	
Otherwise	(Excess)	(StmExp-HiP)	$ (3): RPV is at system pressure and coolant is restored late or not at all.

Data: StmExp-LoP 0.86 MAXIMUM ENTROPY 0.001 0.86 1.0
Data: StmExp-HiP 0.10 MAXIMUM ENTROPY 0.001 0.10 1.0

Does an Alpha mode event occur? $ This question addresses the likelihood that a large stm expls. fails the RPV and Cnt.

	nAlpha	Alpha	
			$ Quantification is based on NUREG-1150.
VStmExp & RPV-LoP	(Excess)	(A-LoP)	$ (1): Steam explosion occurs when the vessel is at low pressure
VStmExp	(Excess)	(A-HiP)	$ (2): Steam explosion occurs when the vessel is at system pressure
Otherwise	1.0	0.0	$ (3): There is not in-vessel steam explosion.

Data: A-LoP 0.01 = Alpha-Dist / StmExp-LoP $ The distribution for Alpha mode includes the prob. of a steam explosion, thus, to
Data: A-HiP 0.001 = Alpha-Dist / 10.0 / StmExp-HiP $ make Alpha conditional on a Stm Expl, the Alpha distrib. is divided by the Stm
$ Stm Expl distrib. Note: Prob. of Alpha mode at high P is 0.1 the prob. at low P.

Data: Correlate Alpha-Dist StmExp-LoP 0.999
Data: Correlate Alpha-Dist StmExp-HiP 0.999

Does a large in-vessel steam explosion fail the vessel?

	SE-Alpha	SE-BtHd	SE-LgBrch	SE-SmBrch	SE-nFail	$ Quant. based on adjusted NUREG-1150 values
Alpha	1.0	0.0	0.0	0.0	0.0	$ (1): Alpha mode occurs
VStmExp	0.0	0.20	0.10	0.10	0.60	$ (2): In-vessel Stm.Exp. occurs: Used M.Berman values
Otherwise	0.0	0.0	0.0	0.0	1.0	$ (3): No Stm. Exp.

What is the mode of VB?

	VB-Alpha	VB-BtHd	VB-LgBrch	VB-SmBrch	nVB	
SE-Alpha	1.0	0.0	0.0	0.0	0.0	$ (1): Alpha mode event
SE-BtHd	0.0	1.0	0.0	0.0	0.0	$ (2) StmExp causes bottom head failure
SE-LgBrch	0.0	0.0	1.0	0.0	0.0	$ (3) StmExp causes large breach
SE-SmBrch	0.0	0.0	0.0	1.0	0.0	$ (4) StmExp causes small breach
CDArrest	0.0	0.0	0.0	0.0	1.0	$ (5) Core damae arrested invessel
HiLiqVB + LoLiqVB	0.0	0.249	0.005	0.746	0.0	$ (6) No StmExp, No core cooling
Otherwise	0.0	0.0	0.0	0.0	1.0	$ This case shouldn't be used

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Does high pressure melt ejection occur?

	nHPME	HPME	
			$ This event is only used in the source term analysis
nVB + RPV-LoP-CD	1.0	0.0	$(1): Either no VB, or RPV depressurized
Otherwise	0.20	0.80	$(2): RPV fails at high pressure

Does a large ex-vessel steam explosion accompany VB?

	nExStmE	ExStmE	
nVB + Alpha 1.0	0.0		$ (1): Either no vessel failure or alpha mode event, thus, no ex-vessel steam explosion
Cav-Fld + E-CorCool + L-CorCool #			$ (2): Vessel fails into a flooded cavity or water enters cavity coincident with debris
	(Excess)	(ExStmE)	$
Otherwise	1.0	0.0	$ (3): Dry cavity. Thus, no ex-vessel steam explosion.

Data: ExStmE 0.86 MAXIMUM ENTROPY 0.001 0.86 1.0 $ Mean based on value used in NUREG-1150

Does the containment fail from pressure loads accompanying VB?

		nCF-VB	CF-Rpt-VB	CF-Lk-VB	
nVB		1.0	0.0	0.00	$ (1): No VB, thus, no loads to cause CNMT failure
Cnt-Rpt-CD + Alpha		1.0	0.0	0.00	$ (2): CNMT already fail or is open
RPV-HiP-CD & HiLiqVB		0.22	0.35	0.43	$ (3) VB @ HiP with large amount of core debris ejected
RPV-HiP-CD	0.41	0.27	0.32	$ (4) VB @ HiP with small amount of core debris ejected	
Cav-Fld & ZrOxid>21		0.90	0.04	0.06	$ (5): VB @ LoP with flooded cavity and a large fraction
					$ of Zr oxidized
Cav-Fld		0.82	0.10	0.08	$ (6) Same as above except small fract of inves Zr oxid.
HIS-CD		0.99	0.005	0.005	$ (7): Slow release of H2 with igniters on.
nBrn-H2 & H2<12		0.79	0.190	0.02	$ (8): 8 to 12 % H2 accumuled BCD; no burn BCD
nBrn-H2 & H2<16		0.13	0.49	0.38	$ (9): 12 to 16 % H2 accumuled BCD; no burn BCD
nBrn-H2 & H2>16		0.04	0.50	0.46	$ (10): >16% H2 accumuled BCD; no burn BCD
Otherwise		1.0	0.0	0.0	$ (11): <8% H2 accumuled BCD; no burn BCD

What is the status of containment integrity just after VB? $ Note: This does not include ILOCAs

		nOCnt-VB	OCnt-VB	
OCnt-CD + Alpha + /nCF-VB	0.0	1.0	$ (1): CNMT hatches open or fails during CD or at VB	
Otherwise		1.0	0.0	$ (2): CNMT is intact

What is the size of the containment opening just after VB? $ Note: This does not include ILOCAs

	Cnt-Rpt-VB	Cnt-Lk-VB	Cnt-NF-VB	
nOCnt-VB	0.00	0.00	1.00	$ (1): The CNMT is intact
Cnt-Rpt-CD + Alpha + CF-Rpt-VB	1.00	0.00	0.00	$ (2): CNMT rupture during CD or VB or Alpha failure
Cnt-Lk-CD + CF-Lk-VB	0.00	1.00	0.00	$ (3): CNMT leak @ CD (and no rupt @ VB) or leak at VB
Otherwise	0.00	0.00	1.00	$ (4): No CNMT failure

Does the auxiliary building fail just after VB?

	nOAux-VB	OAux-VB	
OAux-CD	0.0	1.0	$ (1): Aux Bldg has already failed
nOCnt-BCD & /RPV-ILOCA & /RPV-oMSIV	1.0	0.0	$ (2):CNMT was closed and no ILOCA or no open MSIV
Otherwise	0.0	1.0	$ (3): MELCOR calc. show Aux. Bldg will fail shortly after VB

$==========================Event Associated with Ex-Vessel Phase of the Accident==========================

What is the status of dc power late in the accident?

	nDC-Late	DC-Late	
/nAC-CD	0.00	1.00	$ (1): ac power avail, thus, dc available
nDC-CD	1.00	0.00	$ (2): dc power not available before CD,thus, not avail. late
IE-Win1	0.18	0.82	$ (3): no ac or dc during CD for TW-1
IE-Win2	0.16	0.84	$ (4): no ac or dc during CD for TW-2
Otherwise	1.00	0.00	$ (5): This case shouldn't be used

Is ac power recovery late in the accident?

	nAC-Late	OSP-Late	AC-Late	
/nAC-CD	0.0	0.0	1.0	$ (1): ac power available during CD
nDC-Late	1.0	0.0	0.0	$ (2): no dc late, no ac
IE-Win1	(Excess)	0.0	(AC-LAT-TW1)	$ (3): SBO TW-1: no ac during CD, dc avail. late
IE-Win2	(Excess)	0.0	(AC-LAT-TW2)	$ (4): SBO TW-2: no ac during CD, dc avail. late
Otherwise	0.0	0.0	1.0	$ (5): This case shouldn't be used.

Data: AC-LAT-TW1 0.5 File C:\LPS\POS5\CALC\LHS200\APET\ACRECBE.LSP
Data: AC-LAT-TW2 0.5 File C:\LPS\POS5\CALC\LHS200\APET\ACRECBE.LSP

Is the core debris in the cavity coolable?

	nCCI	FldCCI	DryCCI	
nVB	1.00	0.00	0.00	$ (1): Vessel doesn't fail
Cav-Dry & n-CorCool & /aLPInj	0.00	0.00	1.00	$ (2): LP inject. will not start at VB
Cav-Dry & RPV-HiP-CD	0.80	0.20	0.00	$ (3): Cav. Dry and vessel fails at high pressure
Cav-Dry	0.16	0.84	0.00	$ (4): Cav. Dry and vessel fails at low pressure
Cav-Fld & RPV-HiP-CD	0.80	0.20	0.00	$ (5): Cav. Flooded and vessel fails at high pressure
Cav-Fld & HiLiqVB	0.16	0.84	0.00	$ (6): Cav. Fld, vessel fails at LP with lots of debris
Cav-Fld	0.40	0.60	0.00	$ (7): Cav. Fld, vessel fails at LP with little debris
Otherwise	1.00	0.00	0.00	$ This case shouldn't be used.

Do the operators vent the containment after VB?

	nVnt-Late	Vnt-Late	
nCVS + Cnt-Rpt-VB + Op-RPV-CD & #			$ (1): Either CVS not available, the Cnt is open, or the RPV
(RPV-ILOCA + RPV-oMSIV) 1.0	0.0	$ is open to the Cnt or turbine bulding.	
aCVS + rCVS & AC-Late	(HEP-EVnt)	(Excess)	$ (2): Cnt Pres > 20 psig and CVS is available, EP requires venting

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Otherwise	1.0	0.0	$ (3): Cnt Pres > 20 psig, however CVS not available.	

Does the containment fail late in the accident?

	nCF-Late	CF-Rpt-Late	CF-Lk-Late	
OCnt-VB + Vnt-Late	1.0	0.0	0.00	$ (1): Containment already breached.
Otherwise	0.0	0.5	0.50	$ (2): CNMT fails from late overpressurisation by steam
				$ (3): Mean of failure distribution

What is the status of the containment late in the accident?

	nOCnt-Late	OCnt-Late	
OCnt-VB + Vnt-Late + /nCF-Late	0.0	1.0	$ (1): CNMT already open or failed
Otherwise	1.0	0.0	$ (2): CNMT intact

What is the size of the containment opening late in the accident?

	Cnt-Rpt-Late	Cnt-Lk-Late	Cnt-NF-Late	
nOCnt-Late	0.0	0.0	1.0	$ (1): CNMT intact
Cnt-Rpt-VB + Vnt-Late + CF-Rpt-Late	1.0	0.0	0.0	$ (2): CNMT rupture or open
Cnt-Lk-VB + CF-Lk-Late	0.0	1.0	0.0	$ (3): CNMT fail as a leak (no ruptures)
Otherwise	0.0	0.0	1.0	$ This case shouldn't be used.

$=====================Binner==

Binning: POS5-Sample
PDS
1 PDS1 PDS1-1
2 PDS2 PDS1-2
3 PDS3 PDS1-3
4 PDS4 PDS1-4
5 PDS5 PDS1-5
6 PDS6 PDS2-1
7 PDS7 PDS2-2
8 PDS8 PDS2-3
9 PDS9 PDS2-4
10 PDS10 PDS2-5
11 PDS11 PDS2-6
12 PDS12 PDS3-1
CNT-STATUS
1 OCnt-BCD OCnt-BCD
2 Vnt-CD Vnt-CD
3 Cnt-Rpt-CD Cnt-Rpt-CD
4 Cnt-Lk-CD Cnt-Lk-CD
5 Cnt-Rpt-VB Cnt-Rpt-VB
6 Cnt-Lk-VB Cnt-Lk-VB

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7 Vnt-Late	Vnt-Late
8 Cnt-Rpt-Late	Cnt-Rpt-Late
9 Cnt-Lk-Late	Cnt-Lk-Late
10 Cnt-NF-Late	Cnt-NF-Late
AUX-STATUS	
1 OAux-BCD	OAux-BCD & (OCnt-BCD + RPV-oMSIV + RPV-ILOCA) $ Only want to consider aux. bldg open if releases go through
2 OAux-CD	OAux-CD & (OCnt-BCD + RPV-oMSIV + RPV-ILOCA) $ aux. bldg.
3 OAux-VB	OAux-VB & (OCnt-BCD + RPV-oMSIV + RPV-ILOCA)
4 nOAux	nOAux-VB + nOCnt-BCD & /RPV-oMSIV & /RPV-ILOCA
DW-STATUS	
1 Op-DW-BCD	Op-DW-BCD
2 Cls-DW-BCD	Cls-DW-BCD
RPV-ISO	
1 Iso-RPV-E	RPV-HiP + RPV-LoP
2 RPV-LOCA	RPV-LOCA
3 Iso-RPV-CD	Cls-RPV-CD
4 nIso-RPV-CD	Op-RPV-CD
RPV-VNT	
1 RPV-nVnt	RPV-nVnt
2 RPV-OVnt	RPV-OVnt
SRV-VBrk	
1 OSRV-Bkr	Op-SRV-Bkr
2 cSRV-Bkr	Cls-SRV-Bkr
RPV-VB	
1 RPV-HiP-nInj	RPV-HiP-CD & n-CorCool & /nVB
2 RPV-LoP-nInj	RPV-LoP-CD & n-CorCool & /nVB
3 RPV-HiP-Inj	RPV-HiP-CD & /nVB
4 RPV-LoP-Inj	RPV-LoP-CD & /nVB
5 nVB-HiP	RPV-HiP-CD & nVB
6 nVB-LoP	RPV-LoP-CD & nVB
CNT-SPRAYS	
1 nCS-CD	nCS-CD
2 CS-CD	CS-CD
ZrOXID-CD	
1 ZrOxid-Hi	ZrOxid>21
2 ZrOxid-Lo	ZrOxid<21
HPME-SE	
1 HiHPME	HPME & HiLiqVB
2 LoHPME	HPME
3 HiEXSE	ExStmE & HiLiqVB
4 LoEXSE	ExStmE
5 nHPME-SE	nHPME & nExStmE

```
TYPE-CCI
1 DryCCI                      DryCCI
2 FldCCI                      FldCCI
3 noCCI                       nCCI
IE-TIME
1 IE-Win1                     IE-Win1
2 IE-Win2                     IE-Win2
3 IE-Win3                     IE-Win3
SP-TEMP
1 SP-SupCld  SPT-Sub
2 SP-Sat                      SPT-Sat

$===========Sorting & Re-Binning Commands==========================================================

Sorting:
    CNT-STATUS

Rebinning:

$===========Distributions==========================================================================

Dataset:
    RAN-REC-LOSP 250 UNIFORM 1.0 500.999
    SFP1-1 0.092 File C:\LPS\POS5\CALC\LHS200\APET\GGP5PDS.LHS
    SFP1-2 0.042 File C:\LPS\POS5\CALC\LHS200\APET\GGP5PDS.LHS
    SFP1-3 0.158 File C:\LPS\POS5\CALC\LHS200\APET\GGP5PDS.LHS
    SFP1-4 0.065 File C:\LPS\POS5\CALC\LHS200\APET\GGP5PDS.LHS
    SFP1-5 0.012 File C:\LPS\POS5\CALC\LHS200\APET\GGP5PDS.LHS
    SFP2-1 0.014 File C:\LPS\POS5\CALC\LHS200\APET\GGP5PDS.LHS
    SFP2-2 0.214 File C:\LPS\POS5\CALC\LHS200\APET\GGP5PDS.LHS
    SFP2-3 0.020 File C:\LPS\POS5\CALC\LHS200\APET\GGP5PDS.LHS
    SFP2-4 0.043 File C:\LPS\POS5\CALC\LHS200\APET\GGP5PDS.LHS
    SFP2-5 0.091 File C:\LPS\POS5\CALC\LHS200\APET\GGP5PDS.LHS
    SFP2-6 0.021 File C:\LPS\POS5\CALC\LHS200\APET\GGP5PDS.LHS
    SFP3-1 0.102 File C:\LPS\POS5\CALC\LHS200\APET\GGP5PDS.LHS
    Alpha-Dist 0.001 CONTINUOUS LOGARITHMIC 14 #
                9.99E-8        0.0000 #
                1.00E-7        0.1818 #
                1.00E-6        0.2696 #
                1.01E-5        0.3552 #
                1.00E-4        0.5611 #
                1.00E-3        0.7162 #
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              5.03E-3        0.8200 #
              1.00E-2        0.9111 #
              3.71E-2        0.9670 #
              5.42E-2        0.9760 #
              1.01E-1        0.9844 #
              2.81E-1        0.9920 #
              5.70E-1        0.9960 #
              9.97E-1        1.0000
STVAR1    0.5 UNIFORM 0.0 1.0
STVAR2    0.5 UNIFORM 0.0 1.0
STVAR3    0.5 UNIFORM 0.0 1.0
STVAR4    0.5 UNIFORM 0.0 1.0
STVAR5    0.5 UNIFORM 0.0 1.0
STVAR6    0.5 UNIFORM 0.0 1.0
STVAR7    0.5 UNIFORM 0.0 1.0
STVAR8    0.5 UNIFORM 0.0 1.0
STVAR9    0.5 UNIFORM 0.0 1.0
STVAR10   0.5 UNIFORM 0.0 1.0
STVAR11   0.5 UNIFORM 0.0 1.0
STVAR12   0.5 UNIFORM 0.0 1.0
STVAR13   0.5 UNIFORM 0.0 1.0
STVAR14   0.5 UNIFORM 0.0 1.0
```

B.3 Quantification of the POS 5 APET

This appendix provides supporting information for the quantification of Accident Progression Analysis. Appendix B.3.1 lists the sources of information that were used to quantify the APET. Appendix B.3.2 provides the rationale that was used to determine the human error probabilities that were used in the Human Reliability Analysis.

B.3.1 Sources of Information Used to Quantify the APET

Table B.3.1-1 lists each question in the APET and indicates whether or not it was included in the uncertainty analysis (i.e., sampled) and lists the primary sources of information that were used to quantify the question. Also, if the question was included in the uncertainty analysis, Table B.3.1-1 identifies the distribution that was used to characterize the uncertainty and the variable name. A more detailed discussion of the information sources used to quantify each question is provided in the discussion of each question in Appendix B.1.

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Table B.3.1-1
APET Quantification

APET Question	Sampled	Distribution Type	Variable Name	Quantification Source
What is the Plant Damage State?	Yes	Computed from IRRAS	SFP1-1....SFP3-1	Based on IRRAS Output
What is the status of electric power at core damage (CD) (PDS Char. 1)?	No	None		PDS Description
What is the status of dc power at CD (PDS Char. 1)?	No	None		PDS Description
What is the status of high pressure injection at CD (PDS Char. 2)?	No	None		PDS Description
What is the status of low pressure injection at CD (PDS Char. 2)?	No	None		PDS Description
What is the status of containment sprays and SPC at CD (PDS Char. 3)?	No	None		PDS Description
What is the suppression pool level at the onset of CD (PDS Char. 4)?	No	None		PDS Description
What is the suppression pool temperature at the onset of CD (PDS Char. 5)?	No	None		PDS Description
What is the status of the reactor head vent at the onset of CD (PDS Char. 6)?	No	None		PDS Description
What is the status of the RPV integrity at the onset of CD (PDS Char. 7)?	No	None		PDS Description
What is the status of the CNMT access penetrations at CD (PDS Char. 8)?	No	None		PDS Description
What is the status of the CNMT vent system at the onset of CD (PDS Char. 9)?	No	None		PDS Description
When does CD occur (PDS Char. 10)?	No	None		PDS Description
While in POS 5, when does the initiating event occur (PDS Char. 11)?	No	None		PDS Description
What type of event initiates the accident?	No	None		Summary
What is the pressure in the RPV at the time of CD?	No	None		Summary
How much water is in the reactor pedestal cavity at the time of CD?	No	None		Summary
Is the containment equipment hatch opened at the start of the accident?	No	None		Initial Conditions
Do the operators close the containment before CD?	Yes	BOUNDED LOGNORMAL Mean = 0.102, EF = 5	HEP5-CNT	HRA Analysis [Appendix B.3]
Does the auxiliary building fail before CD?	No	None		MELCOR Calculations [NUREG/CR-6143, Vol. 6, Part 2]
What is the status of the drywell before CD?	No	None		Initial Conditions

Table B.3.1-1 (continued)
APET Quantification

APET Question	Sampled	Distribution Type	Variable Name	Quantification Source
Do the operators turn on the HIS before CD?	Yes	LOGNORMAL: MN= 0.054, EF= 5	HEP-HIS-nSBO	HRA Analysis [Appendix B.3]
Do the station batteries depleted during CD?	No	None		Station Battery Failure Curve [Wheeler, et al., 1989]
Is offsite power restored during CD?	Yes	LOSP Non-recovery curves	AC-ECD-TW1 AC-LCD-TW1 AC-ECD-TW2 AC-LCD-TW2	See Volume 2 of this report for Curves. Probabilities calculated using ACRECBE.FOR
Is the RPV isolated during CD?	Yes	MAXIMUM ENTROPY Lower End: 0.5 Mean: 0.9 Upper End: 1.0	ISO-SDC	Project Staff
Do the operators initiate containment sprays during CD?	No	None		PDS
Do the operators depressurize the RPV during CD?	Yes	BOUNDED LOGNORMAL Mean = 0.54, EF = 5	HEP-PRPV-HiP	HRA Analysis [Appendix B.3]
What is the status of the SRV vacuum breakers during CD?	Yes	UNIFORM Min. = 0.01 Max. =0.5	SRV-VBkr	NUREG-1150 [Brown, et al., 1990]
Is core cooling restored during CD?	No	None		
What is the peak hydrogen concentration in the containment during CD?	No	None		NUREG-1150 [Harper, et al., 1991]
What is the fraction of zirconium that is oxidized in the vessel during CD?	No	None		Summary
Do the operators turn on the HIS during CD ?	Yes	LOGNORMAL Mn= 0.054, EF= 5	HEP-HIS-CD	HRA Analysis [Appendix B.3]
Does an uncontrolled hydrogen combustion event occur during CD?	Yes	UNIFORM: Min. = 0.5, Max. = 1.0 UNIFORM: Min. = 0.0, Max. 0.75	Brn-nSBO Brn-SBO	Based on NUREG-1150 [Brown, et al., 1990]

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Table B.3.1-1 (continued)
APET Quantification

APET Question	Sampled	Distribution Type	Variable Name	Quantification Source
What is the pressure in the containment during CD (no uncontrolled burn)?	No	None		Project Staff
Does the containment fail from quasi-static loads during CD?	No	None		Based on NUREG-1150 [Brown, et al., 1990]
Do the operators vent the containment during CD?	Yes	LOGNORMAL Mn= 0.031, EF= 5	HEP-EVnt	HRA Analysis [Appendix B.3]
What is the status of the containment during CD?	No	None		Summary
What is the size of the containment opening during CD?	No	None		Summary
Does the auxiliary building fail during CD?	No	None		Project Staff
Is there water in the RPV pedestal cavity just prior to VB?	No	None		Summary
Is the core damage process arrested in the vessel?	Yes	MAXIMUM ENTROPY Lower Bound = 0.0 Mean = 0.01 Upper Bound = 0.5	CDArst-L	Project Staff
What fraction of the core debris would be mobil at VB?	No	None		NUREG-1150 [Brown, et al., 1990]
Does a large in-vessel steam explosion occur?	Yes	MAXIMUM ENTROPY Lower Bound = 0.001 Mean = 0.86 Upper Bound = 1.0 MAXIMUM ENTROPY Lower Bound = 0.001 Mean = 0.10 Upper Bound = 1.0	StmExp-LoP StmExp-HiP	Project Staff: Mean value based on NUREG-1150 [Brown, et al., 1990]
Does an Alpha mode event occur?	Yes	Aggregated Distribution Aggregated Distribution	A-LoP A-HiP	NUREG-1150 NUREG-1150
Does a large in-vessel steam explosion fail the vessel?	No	None		NUREG-1150
What is the mode of VB?	No	None		NUREG-1150
Does high pressure melt ejection occur?	No	None		NUREG-1150

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Table B.3.1-1 (concluded)
APET Quantification

APET Question	Sampled	Distribution Type	Variable Name	Quantification Source
Does a large ex-vessel steam explosion accompany VB?	Yes	MAXIMUM ENTROPY Lower Bound = 0.001 Mean = 0.86 Upper Bound = 1.0	ExStmE	Project Staff: Mean values based on NUREG-1150 [Brown, et al., 1990]
Does the containment fail from pressure loads accompanying VB?	No	None		Based on NUREG-1150 [Brown, et al., 1990]
What is the status of containment integrity just after VB?	No	None		Summary
What is the size of the containment opening just after VB?	No	None		Summary
Does the auxiliary building fail just after VB?	No	None		Project Staff
What is the status of dc power late in the accident?	No	None		Station Battery Failure Curve [Wheeler, et al., 1989]
Is ac power recovery late in the accident?	Yes	LOSP Non-recovery Curves	AC-LAT-TW1 AC-LAT-TW2	See Volume 2 of this report for Curves. Probabilities calculated with ACRECBE.FOR
Is the core debris in the cavity coolable?	No	None		NUREG-1150 [Brown, et al., 1990]
Do the operators vent the containment after VB?	Yes	Same distribution that was used for venting during core damage		
Does the containment fail late in the accident?	No	None		Project Staff
What is the status of the containment late in the accident?	No	None		Summary
What is the size of the containment opening late in the accident?	No	None		Summary

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B.3.2 Level 2 Human Reliability Analysis

This appendix describes the Human Reliability Analysis (HRA) that was performed for the Level 2 analysis. The approach used in this analysis is the same approach that was used in the Level 1 analysis. The general methodology used for conducting the HRA and determining the Human Error Probabilities (HEPs) for the identified human actions was the Accident Sequence Evaluation Program Human Reliability Analysis Procedure (ASEP HRAP) [Swain, 1987]. The details of the HRA methodology used in this study are described in Chapter 10 of Volume 2 of this report.

The following four HEPs were developed for the Accident Progression Analysis:

- HEP1: Closure of the containment before the onset of core damage,
- HEP2: Initiation of the Hydrogen Igniter System,
- HEP3: Initiation of containment venting, and
- HEP4: Depressurization of the reactor vessel during core damage.

A simple listing of each of the Level 2 human actions its mean HEP, and its associated error factor is presented in Table B.3.2-1. The calculation and supporting rationale for each of the individual HEPs using the ASEP HRAP procedure is presented in Tables B.3.2-2 through B.3.2-41. According to the ASEP HRAP, the HEPs obtained with the ASEP HRAP procedure are assumed to be median values from a lognormal distribution. The median values were converted to means for use in the analysis using the following formula:

$$\text{Mean} = \text{Median} * \exp\{[\ln(\text{error factor})]^2 / 5.412\}.$$

Table B.3.2-1
Summary of HEPs for Level 2 Analysis

HEP	Distribution	Mean Value	Error Factor
HEP1	Lognormal	0.102	5
HEP2	Lognormal	0.054	5
HEP3	Lognormal	0.031	5
HEP4	Lognormal	0.054	5

Reference

[Swain, 1987] A. D. Swain, "Accident Sequence Evaluation Program Human Reliability Analysis Procedure," NUREG/CR-4772, February 1987.

Table B.3.2-2
HEP 1 Calculation

Human Action Event (1)	Containment Closure
Event Description (2)	Containment Closure is the operator action to diagnose the need and accomplish the actions leading to containment closure.
Event Context (3)	An initiating event has occurred and vessel temperature is increasing due to inadequate decay heat removal. If the operators initiate containment closure early enough, the negative impacts of accident scenarios resulting in core damage can be lessened.
Applicable Procedures (4)	No specific procedures indicate when containment should be closed. However, the GGNS Shutdown Protection Plan used during refueling outages specifies that containment not be open during Operating Condition 3. If vessel water temperature exceeds 200 degrees F with the vessel head on, Operating Condition 3 is entered. GGNS does have procedures describing the process of containment closure.

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Table B.3.2-3
HEP 1: Sequence Timing and Indications

Event/Occurrence (of most interest) (1)	Time (T_o) Operator Alerted (2)	Annunciator/Indication (3)	Comments/ Source of Information (4)
Operators need to initiate containment closure early in accident scenarios in order to lessen the impact of core damage.	O	Vessel water temperature has increased to 200 degrees F. due to inadequate decay heat removal. With the vessel head on and vessel temperature at 200 degrees F, Operating Condition (OC) 4 is left and OC 3 is entered. In OC 3, containment is supposed to be closed per the GGNS Shutdown Protection Plan. Interviews with operators indicated that they would begin to close containment when vessel temperature reached 200 degrees. Interviews also indicated that an inability to establish a normal means of shutdown cooling within 1 hour of an accident would also be a cue to initiate containment closure.	The assumption that the cue for containment closure occurs at time "0" is appropriate because the T-H Calculations for the time available to close containment before core damage etc., assumed vessel temperature was initially at 200 degrees F. While vessel temperature could, in fact, be <= 140 degrees when the initiating event occurs, the time available for the crew to close containment would functionally be the same because their is no reason to assume that they would initiate containment closure prior to vessel temperature reaching 200 degrees F.

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Table B.3.2-4
HEP 1: Potential Operator Action

Description of Event (1)	Number of Abnormal Events (2)	Activities (Tasks) Required to Perform Action and Procedures (3)	Comments/ Source of Information (4)
For any number of possible reasons, a condition of inadequate decay heat removal exists and normal means of SDC are not available. Vessel temperature is approaching boiling (> 200 degrees F) and the operators could decide to close containment to avoid problems later if they are unable to restore adequate cooling.	One	1. Close containment equipment hatch, including using crane to move 19' high hatch into place and torquing down nuts to seal containment. 2. Close airlock at 119'level, including removal plywood frame, cables, hoses, etc. 3. Close airlock at 208' level. Significantly less removal of material required at this airlock than at 119'.	Descriptions of activities and times required to accomplish the necessary actions were obtained through telephone discussions with plant personnel and through responses to written questions submitted to GGNS.

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Table B.3.2-5
HEP 1: Time Available to Diagnose and Perform the Task

Action (1)	Time by Which Operator Must Act (T_{cd}) (2)	Time at Which Operator is Alerted that Symptom has Occurred (T_o) (3)	Maximum Time Available to Perform the Identified Operator Activities (T_m) (4)	Comments/ Source of Information (5)
Diagnose the need and carry-out the actions required to close primary containment.	5 hours	0	5 hours	MELCOR calculations indicate, that for cases where the containment can be closed, that TAF would not be reached for 8 to 10 hours. However, with the head vent open, the temperature in containment (in the very worst and unlikely case) could reach 123 degrees in 5 hours and up to 145 degrees in 8 hours. It was judged that temperatures greater than 123 degrees might restrict the crews ability to work in containment. Thus, it was conservatively assumed that they would only have 5 hours. It should noted, however, that the containment cooling system would have to fail and the initial containment temperature be at 100 degrees at the start of the accident (both of which are unlikely) for containment to reach 123 degrees in 5 hours.

Table B.3.2-6
HEP 1: Operator Action Performance Time

Activities (1)	Location (2)	Travel Time (T_t) (3)	Performance Time (T_p) (4)	Total Action Time (T_a) (5)	Comments/ Source of Information (6)
1. Close containment equipment hatch, including using crane to move 19' high hatch into place and torquing down nuts to seal containment.	Inside wall of primary containment	--	Estimated travel and performance time for closing the equipment hatch was 4 hours.	4 hours	Specific estimates regarding performance time were obtained from GGNS personnel. Per the GGNS Shutdown Protection Plan, different groups of individuals are assigned responsibility for closing the hatch, and the two airlocks during an outage. Thus, closing of the airlocks could be done in parallel with the closing of the hatch. GGNS personnel claim that the hatch could be in place with the nuts on in about 2 hours. An additional hour would be required to torque down the nuts to ensure an air tight seal. This estimate assumes no problems occur, e.g., problems with the crane. Thus, to provide some accounting for potential delays, it was assumed 4 hours would be necessary to seal containment. This estimate also assumes a crew of four would be available to close the hatch.
2. Close airlock at 119'level, including removal plywood frame, cables, hoses, etc.	119' level in containment	--	Estimated travel and performance time for closing the airlock was 1 hour.	0 minutes. Assumed to be done in parallel with hatch closure	GGNS estimated that the plywood frame (which is built to help reduce contamination) could be dismantled, the quick release hoses detached and thrown-out, the electrical wires disconnected, and the 119' level airlock shut within a hour "easily" (it takes about 10 min. to close the airlocks themselves. The inner door alone will seal containment.
3. Close airlock at 208' level. Significantly less removal of material required at this airlock than at 119'.	208' level in containment	--	Estimated travel and performance time for closing the airlock was 30 minutes	0 minutes Assumed to be done in parallel with hatch closure 4 hours Total	With less material to be removed, GGNS estimated it would take 30 minutes to close the 208' level air lock. Regardless, both airlocks could be closed in the time required to close the hatch.

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Table B.3.2-7
HEP 1: Diagnosis Time for Operator Action

Action (1)	Maximum Time Available (T_m) (2)	Total Action Time (T_a) (3)	Time Available to Diagnosis (T_d) (4)	Comments/ Source of Information (5)
Diagnose the need to initiate containment closure	5 hours	4 hours	60 minutes	

Table B.3.2-8
HEP 1: Diagnosis Analysis

Action (1)	Failure to Diagnose (2)	Skill-Based (3)	Adjusted/ Final HEP (4)	Comments/ Source of Information (5)
Diagnose the need to initiate containment closure.	Per ASEP HRAP Table 8-3, the median value from ASEP Figure 8-1 for 60 minutes diagnosis time was assigned.		Median = 1.0E-4 EF = 30 Mean = 8.5E-4	Discussions with plant personnel indicated that the operators would be concerned about an open containment and would initiate closure if vessel temperature reached 200 degrees F. However, they also acknowledged that they would not want to unnecessarily close containment. One operator indicated that even if 200 degrees was not reached, if they were unable restore normal shutdown cooling in an hour, then they would initiate containment closure.

Table B.3.2-9
HEP 1: Post-Diagnosis Action Type Identification
per Step 10, Table 8-1 of ASEP HRAP

Action (1)	Safety Systems Failed (2)	EOPs, Training, Use EOPs Well Designed EOPs (3)	Individual Operator Must Perform Concurrent Tasks (4)	Dynamic or Step-by-Step (5)	Comments/ Source of Information (6)
Close containment equipment hatch and both airlocks.	N/A - Operators may or may not have entered EOPs at this point. Regardless, closing containment is carried-out by individuals other than operators and is clearly a step x step task.	Discussions with plant personnel indicated that they were knowledgeable about the need for the actions and the requirements.	No	Step-by-Step	

Table B.3.2-10
HEP 1: Post-Diagnosis Stress-Level Identification per Step 10, Table 8-1 of ASEP HRAP

Action (1)	T_m <2h After IE (2)	Recirc. Phase in Large LOCA (3)	More Than Two Safety Systems Fail (4)	Operator Familiar W/Sequence (5)	Stress Level (6)	Comments/ Source of Information (7)
1. Close equipment hatch	N/A[1]	N/A	Maybe	Maybe	Extremely High	With the crew being instructed to close containment in an emergency (accident) situation in which containment temperature is likely to be increasing, extremely high stress must be assumed.
2. Close airlocks at 119' and 208'level	N/A[1]	N/A	Maybe	Maybe	Moderately High	Since closing the airlocks can be done essentially in parallel with closing the hatch, the time available to accomplish these tasks is much greater than that for closing the hatch. In addition, the work for closing the airlocks could be done before containment temperature has increased much and, in any case, the work is mainly done in the door area. Thus, only moderately high stress was assumed for these actions.

[1] At least moderately high stress was assumed for all events.

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Table B.3.2-11
HEP 1: Total HEP

Action (1)	Original Operator HEP (HEP_{op}) (2)	Independent Check/Correction HEP (HEP_{cl}) (3)	Total HEP (4)	EF (5)	Comments/ Source of Information (6)
1. Diagnose need to close primary containment	Median = 1.0E-4 EF = 30 Mean = 8.5E-4		Med. Mean 1.0E-4 8.5E-4	(30)	
2. Close containment equipment hatch, including using crane to move 19' high hatch into place and torquing down nuts to seal containment.	Median = 0.05 Mean = 0.081	Does not really make sense to give credit for a second check on closing the equipment hatch. There would very likely be insufficient time to get it closed if the crew failed to start the task when directed. Similarly, assuming some sort of unnoticed error in sealing the hatch was possible (it is difficult to think of one), the time available would most likely be insufficient.	Med. Mean 0.05 0.081	(5)	
3. Close airlock at 119'level, including removal of plywood frame, cables, hoses, etc.	Med. = 0.02 Mean = 0.032	Credit for a second check was given in this instance. Given the number of people around during shutdown, the fact that a specific team is responsible, and the ample time available, a failure to start this task could be recovered. Second check values for action 2 were: Median = 0.2 Mean = 0.323	0.004 0.01	(5)	Second check HEPs are multiplied by the original HEP for each action.
4. Close airlock at 208' level. Significantly less removal of material required at this airlock than at 119'.	Med. = 0.02 Mean = 0.032	Yes, rationale the same as for airlock at 119' level.	<u>0.004 0.01</u> 0.058 0.102 Total Median HEP =0.058 Total Mean HEP = 0.102	<u>(5)</u> (5)	The error factor associated with the dominant HEPs was assigned.

Table B.3.2-12
HEP 2 Calculation

Human Action Event (1)	Initiation of Hydrogen Igniter System
Event Description (2)	Initiation of the Hydrogen Igniter System (HIS) is the operator action to diagnose the need and accomplish the actions leading to the hydrogen igniters being turned-on per procedure.
Event Context (3)	An initiating event has occurred and the reactor vessel level has dropped to Top of Active Fuel (TAF). Vessel level reaching TAF (- 167 in.) is an entry condition for the Hydrogen Control section of GGNS EP-3 (Containment Control). If the operators initiate the HIS before hydrogen concentration levels reach 9%, potential explosions related to hydrogen release in the drywell and primary containment may be prevented. It is assumed that the containment H_2 concentration is in the safe zone of the Hydrogen Deflagration Overpressure Limit and presumably can be maintained there upon entering EP-3. (Note. In Station Blackout Scenarios where level has reached TAF, the operators would be unable to determine Hydrogen concentration levels and therefore could not determine whether or not H_2 concentration could be maintained in the safe zone of the Hydrogen Deflagration Overpressure Limit (HDOL). Given this situation, the EP guides the operators to "secure and prevent operation" of the igniters. Thus, it is assumed that the operators would not turn the hydrogen igniters to the on position until they had recovered power and could determine the H_2 levels.)
Applicable Procedures (4)	Hydrogen Control Section of Emergency Procedure 3 (EP-3, Containment Control, GGNS, Rev. 21)

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Table B.3.2-13
HEP 2: Sequence Timing and Indications

Event/Occurrence (of most interest) (1)	Time (T_o) Operator Alerted (2)	Annunciator/Indication (3)	Comments/ Source of Information (4)
Operators need to initiate the HIS to prevent the build-up of large pockets of hydrogen and the associated potential explosions which could damage containment.	O	The primary indicator is vessel level reaching TAF. Numerous alarms will have sounded prior to or in conjunction with vessel level reaching TAF. Thus, the operators will be tracking a significant drop in water level. Another indicator would be containment or drywell H_2 concentration above 0.5%, which also is an entry condition for the hydrogen control section of EP-3 . In all Initiating Event scenarios except for a LOCA, vessel level dropping to TAF would occur over a several hour period (3 to 13 hours depending on the event). For the LOCA scenario, TAF would be reached more or less immediately, but core damage would not be reached for approximately an hour after TAF is reached. At least 1 hour would be available between TAF and core damage for all relevant scenarios.	MELCOR calculations determined both the time before TAF would be reached for the various initiators and the time between TAF and Core Damage. It should be noted that hydrogen build-up after core damage would not reach a level that would mandate <u>not</u> using the hydrogen igniters for at least another 30 to 60 minutes, depending on the accident scenario.

Table B.3.2-14
HEP 2: Potential Operator Action

Description of Event (1)	Number of Abnormal Events (2)	Activities (Tasks) Required to Perform Action and Procedures (3)	Comments/ Source of Information (4)
For any number of possible reasons, vessel level has dropped to TAF.	One abnormal event is assumed per Table 8-1, Step 9.b.1. While failures of several systems may occur in the relevant scenarios, the entry condition for the EP-3 is clear and essentially only one diagnosis is required (level below -167"). The procedure should guide the operators to make the appropriate response.	Operate the igniters from the control room.	

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Table B.3.2-15
HEP 2: Time Available to Diagnose and Perform the Task

Action (1)	Time by Which Operator Must Act (T_{cd}) (2)	Time at Which Operator is Alerted that Symptom has Occurred (T_o) (3)	Maximum Time Available to Perform the Identified Operator Activities (T_m) (4)	Comments/ Source of Information (5)
Diagnose the need and carry-out the actions required to initiate the hydrogen ignition system.	1 hour	0	1 hour	Results from MELCOR calculations indicated that core damage would not occur for at least an hour after reaching TAF. H_2 would not be released until core damage occurred. Thus, at least 1 hour would be available to diagnose the need and initiate the hydrogen igniters. In fact, as noted above, hydrogen build-up after core damage would not reach a level that would mandate not using the hydrogen igniters for at least another 30 to 60 minutes, depending on the accident scenario.

Table B.3.2-16
HEP 2: Operator Action Performance Time

Activities (1)	Location (2)	Travel Time (T_L) (3)	Performance Time (T_P) (4)	Total Action Time (T_A) (5)	Comments/ Source of Information (6)
Operate Hydrogen Igniter System	Control Room	--	1 minute	1 minute	Per ASEP Table 8-1, Step 5b, a 1 min. travel and manipulation time was assumed for actions in the control room.

Table B.3.2-17
HEP 2: Diagnosis Time for Operator Action

Action (1)	Maximum Time Available (T_m) (2)	Total Action Time (T_a) (3)	Time Available to Diagnosis (T_d) (4)	Comments/ Source of Information (5)
Diagnose the need to initiate Hydrogen Ignition System	1 hour	1 minute	Approx. 59 minutes	

Table B.3.2-18
HEP 2: Diagnosis Analysis

Action (1)	Failure to Diagnose (2)	Skill-Based (3)	Adjusted/ Final HEP (4)	Comments/ Source of Information (5)
Diagnose the need to initiate the Hydrogen Igniter System.	Per ASEP HRAP Table 8-3, the median value from ASEP Figure 8-1 for approx. 59 minutes diagnosis time was assigned.		Median = 1.0E-4 EF = 30 Mean = 8.5E-4	

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Table B.3.2-19
HEP 2: Post-Diagnosis Action Type Identification
per Step 10, Table 8-1 of ASEP HRAP

Action (1)	Safety Systems Failed (2)	EOPs, Training, Use EOPs, Well Designed EOPs (3)	Individual Operator Must Perform Concurrent Tasks (4)	Dynamic or Step-by-Step (5)	Comments/ Source of Information (6)
Operate Hydrogen Ignition System	N/A - No additional safety systems are assumed to fail at this point.	Although the lower bound value from the diagnosis model was not used, the EOPs are clear and the operators need only initiate the system from the control room. Thus, the actions were not assumed to be dynamic.	No	Step-by-Step	

Table B.3.2-20
HEP 2: Post-Diagnosis Stress-Level Identification per Step 10, Table 8-1 of ASEP HRAP

Action (1)	T_w <2h After IE (2)	Recirc. Phase in Large LOCA (3)	More Than Two Safety Systems Fail (4)	Operator Familiar W/Sequence (5)	Stress Level (6)	Comments/ Source of Information (7)
Initiate Hydrogen Ignition System	N/A[1]	Maybe	Maybe	Maybe	Extremely High	With vessel level dropping to TAF in scenarios where safety systems have failed and possibly in the context of a LOCA, extremely high stress was assumed.

[1] At least moderately high stress was assumed for all events.

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Table B.3.2-21
HEP 2: Total HEP

Action (1)	Original Operator HEP (HEP_{op}) (2)	Independent Check/Correction HEP (HEP_{ct}) (3)	Total HEP (4)	EF (5)	Comments/ Source of Information (6)
1. Diagnose need to initiate hydrogen igniter system.	Median = 1.0E-4 EF = 30 Mean = 8.5E-4		Med. Mean 1.0E-4 8.5E-4		
2. Initiate hydrogen igniter system.	Med. = 0.05 Mean = 0.081	Credit for a second check and third was given in this instance. Given the initial time available for the task, the fact that additional time would be available even after CD was reached (at least 30 minutes and in most cases an hour), the importance of the action, the clarity of the EP, and the simplicity of the action, an initial failure to operate the system would have some probability of being recovered. Second and third check HEPs for the action were: Median = 0.5 Mean = 0.81	0.013 0.053 0.013 0.054	(5)	The second check HEP is multiplied by the original HEP for a given action. The resulting product is multiplied by the HEP for the third check.
Total HEP and Error Factor			Total Median HEP =0.013 Total Mean HEP = 0.054	(5)	The error factor associated with the dominant HEP was assigned.

Table B.3.2-22
HEP 3 Calculation

Human Action Event (1)	Initiation of Containment Venting
Event Description (2)	Initiation of containment venting is the operator action to diagnose the need and accomplish the actions leading to the venting of containment. The goal is to reduce extreme pressure build-up in containment and avoid a potential loss of primary containment during an accident. An emergency procedure (EP-3) clearly addresses the situation.
Event Context (3)	An initiating event has occurred and for a number of possible reasons has resulted in core damage and eventually in vessel breach. The relevant indicators for venting are that drywell pressure is greater than 1.23 PSIG and containment pressure has reached 20 PSIG. Drywell pressure above 1.23 PSIG is the entry condition for the containment pressure control section of GGNS EP-3 (Containment Control). Many hours have elapsed since the initiating event (approximately 16 hours). At this point in any of the relevant scenarios, the plant conditions would not be changing rapidly and an Emergency Response Team would be in place. Containment venting is clearly indicated in EP-3 and given that the relevant parameters have been reached, the Emergency Director would have to override EP-3 to prevent venting. The present analysis assumed that the Emergency Director would not have any basis for overriding EP-3 and that the crew would be attempting to follow the emergency procedures to prevent any damage to primary containment.
Applicable Procedures (4)	Containment Pressure Control Section of Emergency Procedure 3 (EP-3, Containment Control, GGNS, Rev. 21) and Attachment 13 of the EPs (05-S-01-EP-2, Attachment 13, Rev. 19).

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Table B.3.2-23
HEP 3: Sequence Timing and Indications

Event/Occurrence (of most interest) (1)	Time (T_o) Operator Alerted (2)	Annunciator/Indication (3)	Comments/ Source of Information (4)
Operators need to initiate containment venting to avoid overpressurization and possible loss of primary containment.	O	The primary indicators are drywell pressure greater than 1.23 PSIG and containment pressure at or above 20 PSIG. Core damage and a breach of the vessel has occurred. Numerous alarms will have sounded prior to reaching this point and drywell and containment pressure will be paramount to the crews in their attempts minimize the impact of the accident.	MELCOR calculations determined the time at which the relevant parameters would be reached and the time available for the operators to respond to the cues in order to prevent a loss of primary containment. In regards to any hesitancy on the part of the operators to vent containment after core damage, GGNS indicated that at this point in time EP-3 would be followed. GGNS did acknowledge that the issue is being discussed in the ongoing Severe Accident work.

Table B.3.2-24
HEP 3: Potential Operator Action

Description of Event (1)	Number of Abnormal Events (2)	Activities (Tasks) Required to Perform Action and Procedures (3)	Comments/ Source of Information (4)
For any of several possible reasons, core damage has occurred and has been followed by vessel breach. The parameters for entry into the venting procedure have been met. Substantial time is available for diagnosing the need and performing the relevant actions	One abnormal event is assumed per Table 8-1, Step 9. While failures of several systems <u>may</u> occur in the relevant scenarios, the entry condition for the relevant leg of EP-3 is clear. At least two hours are available for making the diagnosis and the procedure should guide the operators to make the appropriate response.	Operators will be required to override four containment vent path isolation interlocks per EP Attachment 13 and open six valves from the control room. Jumpering through the interlocks will require use of Jumper Kit No. 13. Two relays in the main control room and two in the upper control room will require "jumpering."	Per Attachment 13, Jumper Kit No. 13 is stored in the control room emergency locker. Vent procedure is clear. Relays were not inspected for adequacy of labeling.

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Table B.3.2-25
HEP 3: Time Available to Diagnose and Perform the Task

Action (1)	Time by Which Operator Must Act (T_{cd}) (2)	Time at Which Operator is Alerted that Symptom has Occurred (T_o) (3)	Maximum Time Available to Perform the Identified Operator Activities (T_m) (4)	Comments/ Source of Information (5)
Diagnose the need and carry-out the actions required to vent containment.	3 hours	0	3 hours Results from MELCOR calculations indicated that the indicators for containment venting would occur more or less in conjunction with the occurrence of a vessel breach related to core damage. Regarding the time available for the operators to respond, the emergency procedure directs that the operators vent containment when pressure reaches 20 PSIG. (continued in next column).	(continued from previous column) However, if containment pressure increases to 22 PSIG, the emergency procedure essentially instructs the operators to "abandon" the core and use whatever injection systems are available for containment sprays, i.e., save containment. It seems extremely unlikely that the operators would delay venting and put themselves in this position. T-H calculations indicated that it would take about three hours for pressure to increase from 20 to 22 PSIG. Thus, it seemed reasonable to assume that if the operators were going to follow procedure and vent, they would do so within the three hours after the relevant parameters were reached. In actuality, 10 to 14 hours would have to elapse before containment pressure would become great enough to fail containment and the operators could vent at any point prior to that time.

Table B.3.2-26
HEP 3: Operator Action Performance Time

Activities (1)	Location (2)	Travel Time (T_t) (3)	Performance Time (T_p) (4)	Total Action Time (T_a) (5)	Comments/ Source of Information (6)
1. Jumper two different sets of relays in the main control room.	Main control room	Per ASEP HRAP, Table 8-1, Step 5c, 2 minutes travel time was assumed.	Per ASEP HRAP, Table 8-1, Step 5a, five minutes were assumed necessary to retrieve and read Attachment 13. It was conservatively assumed that the two relays could be jumpered within 15 minutes.	Per columns 3 and 4 of this Table, Total action time would be 22 minutes.	Retrieving and reading Attachment 13 and jumpering the two sets of relays in the main control room were assumed to be completely dependent actions.
2. Jumper two different sets of relays in the upper control room.	Upper control room	Travel time to upper control (up one flight of stairs) was conservatively assumed to be 15 minutes.	As noted above, performance time for jumpering two relays could be easily accomplished within 15 minutes.	Travel plus performance time equals about 30 minutes.	Except for retrieving and reading Attachment 13, this set of actions could be accomplished in parallel with the set of actions in activity #1 discussed above. However, given the procedural demands for a second check on each step of the task and the time available, it was assumed that the actions would occur serially. Jumpering the two sets of relays in the upper control room were assumed to be completely dependent actions.
3. Open six valves from the control room to vent containment (per Attachment 13).	Main Control Room	--	6 minutes	6 minutes Total Action Time for the three sets of actions would be equal to: 22 + 30 + 6 minutes, which is equal to 58 minutes (or about 1 hour).	Per ASEP Table 8-1, Step 5b, a 1 min. travel and manipulation time was assumed for actions in the control room. Opening the relevant valves was assumed to be a completely dependent set of actions.

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Table B.3.2-27
HEP 3: Diagnosis Time for Operator Action

Action (1)	Maximum Time Available (T_m) (2)	Total Action Time (T_a) (3)	Time Available to Diagnosis (T_d) (4)	Comments/ Source of Information (5)
Diagnose the need to vent containment.	3 hours	1 hours	2 hours	

Table B.3.2-28
HEP 3: Diagnosis Analysis

Action (1)	Failure to Diagnose (2)	Skill-Based (3)	Adjusted/ Final HEP (4)	Comments/ Source of Information (5)
Diagnose the need to vent containment.	Per ASEP HRAP Table 8-3, the median value from ASEP Figure 8-1 for 2 hours diagnosis time was assigned.	N/A	Median = 6.0E-5 EF = 30 Mean = 5.1E-4	

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Table B.3.2-29
HEP 3: Post-Diagnosis Action Type Identification
per Step 10, Table 8-1 of ASEP HRAP

Action (1)	Safety Systems Failed (2)	EOPs, Training, Use EOPs, Well Designed EOPs (3)	Individual Operator Must Perform Concurrent Tasks (4)	Dynamic or Step-by-Step (5)	Comments/ Source of Information (6)
Carry-out the three sets of actions necessary to vent containment.	N/A - No additional safety systems are assumed to fail at this point.	Although the lower bound value from the diagnosis model was not used, the EOPs and Attachment 13 are clear and the tasks are straightforward. Thus, the actions were not assumed to be dynamic.	No	Step-by-Step	

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Table B.3.2-30
HEP 3: Post-Diagnosis Stress-Level Identification per Step 10, Table 8-1 of ASEP HRAP

Action (1)	T_m <2h After IE (2)	Recirc. Phase in Large LOCA (3)	More Than Two Safety Systems Fail (4)	Operator Familiar W/Sequence (5)	Stress Level (6)	Comments/ Source of Information (7)
Carry-out the three sets of actions necessary to vent containment	N/A[1]	See comments	See comments	Yes - see comments	Moderately High	Given the substantial time available to accomplish the task, the fact that an Emergency Response Team would be in place, and that a verifier would accompany the individual performing the jumpering of the relays, moderate as opposed to extremely high stress was assigned.

[1] At least moderately high stress was assumed for all events.

Table B.3.2-31
HEP 3: Total HEP

Action (1)	Original Operator HEP (HEP_{op}) (2)	Independent Check/Correction HEP (HEP_{rt}) (3)	Total HEP (4)	EF (5)	Comments/ Source of Information (6)
1. Diagnose need to vent containment	Median = 6.0E-5 EF = 30 Mean = 5.1E-4	N/A	Med. Mean 6.0E-5 5.1E-4	(30)	
2. Jumper two different sets of relays in the main control room.	Med. = 0.02 Mean = 0.032	Credit for a second check was given in this instance. Given the initial time available for the tasks, the fact that a verifier would accompany the individual assigned to perform the task and written verification is required, the fact that feedback would be immediate (decrease in pressure), and the seriousness of the action, any initial failures in performing the actions would have some probability of being recovered. Second check HEPs for the action were: Median = 0.2 Mean = 0.323	0.004 0.01	(5)	The second check HEP is multiplied by the original HEP for a given action.
3. Jumper two different sets of relays in the upper control room.	Med. = 0.02 Mean = 0.032	Same as for action 2 above	0.004 0.01	(5)	
4. Open six valves from the control room to vent containment (per Attachment 13).	Med. = 0.02 Mean = 0.032	Same as for action 2 above	0.004 0.01	(5)	
Total HEP and Error Factor			Total Median HEP = 0.012 Total Mean HEP = 0.031	(5)	The error factor associated with the dominant HEPs was assigned.

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Table B.3.2-32
HEP 4 Calculation

Human Action Event (1)	Depressurize Reactor Pressure Vessel (RPV)
Event Description (2)	Operators must depressurize the vessel per emergency operating procedures.
Event Context (3)	An initiating event has occurred, vessel makeup has not been provided for any of several reasons, and the reactor vessel level has dropped to Top of Active Fuel (TAF). As level continues to drop, if vessel pressure is high, emergency depressurization is indicated in several places in EP-2, e.g., vessel level reaching - 210 in. In addition, depressurization is indicated in EP-3 under several different sections related to containment control, e.g., suppression pool, drywell, and containment temperature control and containment pressure control. Given the circumstances, several of the parameters indicating the need to depressurize will be reached before core damage. HEP 4 assesses the probability that the operators and Emergency Response Team would fail to depressurize given that the EPs direct them to do so.
Applicable Procedures (4)	EP-2 (RPV Control, GGNS, Rev. 19), EP-3 (Containment Control, GGNS, Rev. 21)

Table B.3.2-33
HEP 4: Sequence Timing and Indications

Event/Occurrence (of most interest) (1)	Time (T_o) Operator Alerted (2)	Annunciator/Indication (3)	Comments/ Source of Information (4)
Vessel level is dropping and parameters are reached which indicate that the operators should depressurize the vessel	O	Several indicators are likely, including vessel level reaching - 210 in., SP temperature and RPV pressure outside the safe zone of the Heat Capacity Temperature Limit, drywell temperature above 330 degrees F, containment temperature below 180 degrees F, and/or containment pressure outside the safe zone of the Pressure Suppression Pressure (PSP).	Because of the several ways in which the situation of interest could be reached, it was not precisely determined when the relevant parameters would be reached. The question is whether or the not the operators would depressurize in a "reasonable" amount of time given that the relevant parameters were reached.

Table B.3.2-34
HEP 4: Potential Operator Action

Description of Event (1)	Number of Abnormal Events (2)	Activities (Tasks) Required to Perform Action and Procedures (3)	Comments/ Source of Information (4)
For any number of possible reasons, vessel level has dropped to TAF and makeup is not being provided. Vessel temperature and pressure is increasing.	One abnormal event is assumed per Table 8-1, Step 9.b. While failures of several systems may occur in the relevant scenarios, the entry conditions for EP-2 and the EP-3 are clear and any of several different conditions will indicate depressurization (see above). Essentially the original abnormal event is continuing. The EPs should guide the operators to make the appropriate response.	Depressurize the vessel from the control room using SRVs, MSIVs, or any available means.	

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Table B.3.2-35
HEP 4: Time Available to Diagnose and Perform the Task

Action (1)	Time by Which Operator Must Act (T_{cd}) (2)	Time at Which Operator is Alerted that Symptom has Occurred (T_o) (3)	Maximum Time Available to Perform the Identified Operator Activities (T_m) (4)	Comments/ Source of Information (5)
Diagnose the need and carry-out the actions required to depressurize the RPV.	1 hour	0	1 hour	As noted above, the time available for the action was not precisely determined because of the different ways in which the need to depressurize could be reached. The relevant constants are that TAF has been reached, adequate vessel makeup is not being provided, and several different indicators should signal the need to depressurize. Obviously the situation is not good and the concern is whether or the not the operators will follow procedure and depressurize within an hour after the indicators are reached. If the operators follow procedure and depressurize as instructed by the EPs, then the consequences of the accident might be reduced.

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Table B.3.2-36
HEP 4: Operator Action Performance Time

Activities (1)	Location (2)	Travel Time (T_L) (3)	Performance Time (T_P) (4)	Total Action Time (T_a) (5)	Comments/ Source of Information (6)
Open SRVs, MSIVs, etc. to depressurize vessel	Control Room	--	1 minute	1 minute	Per ASEP Table 8-1, Step 5b, a 1 min. travel and manipulation time was assumed for actions in the control room.

Table B.3.2-37
HEP 4: Diagnosis Time for Operator Action

Action (1)	Maximum Time Available (T_m) (2)	Total Action Time (T_a) (3)	Time Available to Diagnosis (T_d) (4)	Comments/ Source of Information (5)
Diagnose the need to depressurize the RPV	1 hour	1 minute	Approx. 59 minutes	

Table B.3.2-38
HEP 4: Diagnosis Analysis

Action (1)	Failure to Diagnose (2)	Skill-Based (3)	Adjusted/ Final HEP (4)	Comments/ Source of Information (5)
Diagnose the need to depressurize the vessel.	Per ASEP HRAP Table 8-3, the median value from ASEP Figure 8-1 for approx. 59 minutes diagnosis time was assigned.		Median = 1.0E-4 EF = 30 Mean = 8.5E-4	

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Table B.3.2-39
HEP 4: Post-Diagnosis Action Type Identification per Step 10, Table 8-1 of ASEP HRAP

Action (1)	**Safety Systems Failed (2)**	**EOPs, Training, Use EOPs, Well Designed EOPs (3)**	**Individual Operator Must Perform Concurrent Tasks (4)**	**Dynamic or Step-by-Step (5)**	**Comments/ Source of Information (6)**
Depressurize the vessel	N/A - No additional safety systems are assumed to fail at this point.	Although the lower bound value from the diagnosis model was not used, the EOPs are clear and the operators need only open SRVs from the control room. Thus, the actions were not assumed to be dynamic.	No	Step-by-Step	

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Table B.3.2-40
HEP 4: Post-Diagnosis Stress-Level Identification per Step 10, Table 8-1 of ASEP HRAP

Action (1)	T_m <2h After IE (2)	Recirc. Phase in Large LOCA (3)	More Than Two Safety Systems Fail (4)	Operator Familiar W/Sequence (5)	Stress Level (6)	Comments/ Source of Information (7)
Depressurize the vessel	N/A[1]	Maybe	Maybe	Maybe	Extremely High	With vessel level dropping to TAF in scenarios where safety systems have failed and possibly in the context of a LOCA, extremely high stress was assumed during the time period of interest.

[1] At least moderately high stress was assumed for all events.

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Table B.3.2-41
HEP 4: Total HEP

Action (1)	Original Operator HEP (HEP_o) (2)	Independent Check/Correction HEP (HEP_{cl}) (3)	Total HEP (4)	EF (5)	Comments/ Source of Information (6)
1. Diagnose need to depressurize the vessel	Median = 1.0E-4 EF = 30 Mean = 8.5E-4		Med. Mean 1.0E-4 8.5E-4	(30)	
2. Open SRVs, MSIVs, etc. to depressurize vessel.	Med. = 0.05 Mean = 0.081	Credit for a second and third check was given in this instance. The EPs are clear, the action is simple, and an initial failure to carry-out the needed actions would be likely to be detected by the crew, i.e., the operators would be monitoring the relevant parameters. Second and third check HEPs for the action were: Median = 0.5 Mean = 0.81	<u>0.013</u> <u>0.053</u> 0.013 0.054	(5)	The second check HEP is multiplied by the original HEP for a given action. The resulting product is multiplied by the HEP for the third check.
Total HEP and Error Factor			Total Median HEP =0.013 Total Mean HEP = 0.054	(5)	The error factor associated with the dominant HEP was assigned.

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Appendix C: Supporting Information for the Source Term Analysis

Appendix C.1 provides a list of the FORTRAN code GGSORP5.FOR that implements the parametric expressions used to estimate the source terms. Appendix C.2 provides a listing of the input file for GGSORP5 that contains the data for the parameters in the parametric expression. Appendix C.3 is a listing of the source term for each source term group defined with using the PARTITION code.

C.1 Listing of GGSORP5.FOR

```
      PROGRAM GGSOR5
C*****ADAPTATION OF RELTRAC INPUT PROCESSOR FOR USE IN GGSOR
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      LOGICAL NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1    EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /KEYS/ NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1    EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /CONTRL/ NLHS, NOBS, NSTART, NBIN, NDM, NTOT
C
C
      OPEN (5, FILE='GGSOR.INP')
      OPEN (6, FILE='GGSOR.OUT')
      OPEN (9, FILE='GGSOR.CFL')
C///      OPEN (10, FILE='GGSOR.ONS')
C*****READ KEYWORDS AND RELATED INFORMATION FROM UNIT 5.
C*****KEYWORDS DETERMINE OPERATION OF RELCLC:
C*****(1) BINNED INPUT WITH SAMPLING
C*****(2) BINNED INPUT WITHOUT SAMPLING
C*****(3) DIRECT INPUT WITH SAMPLING
C*****(4) DIRECT INPUT WITHOUT SAMPLING
      CALL INPUT
C*****CHECK FOR BINNED EXECUTION
      IF (BINNED) THEN
C********CHECK FOR SAMPLING EXECUTION
         IF (SAMPLE) THEN
C***********BINNED INPUT WITH SAMPLING
            CALL BINSMP
         ELSE
C***********BINNED INPUT WITHOUT SAMPLING
            CALL BIN
         ENDIF
      ELSE
C********CHECK FOR SAMPLING EXECUTION
         IF (SAMPLE) THEN
C***********DIRECT INPUT WITH SAMPLING
            CALL DIRSMP
         ELSE
C***********DIRECT INPUT WITHOUT SAMPLING
            CALL DIR
         ENDIF
      ENDIF
      STOP
      END
      BLOCK DATA
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      CHARACTER*7 NAME
      LOGICAL LDEFLT, LREAL
      COMMON /DEFLT1/ NAME(MAXVAR)
      COMMON /DEFLT2/ NVAR, NVAL, NVCB1, NVCB2, NVCB3, NVCB4,
     1                NVCB5, IDIMEN(3,MAXVAR), ISPOS(MAXVAR),
     2                ISMPPS(MAXVAL), IPNT(MAXVAR), LDEFLT(MAXVAL),
     3                LREAL(MAXVAL)
C*****DEFINE VARIABLE NAMES AND CORRESPONDING DIMENSIONS TO BE SET
C*****THROUGH DEFAULT AND SAMPLE VECTOR SUBSTITUTION FOR BINNED
C*****EXECUTION.  VARIABLE NAMES AND DIMENSIONS CORRESPOND EXACTLY
C*****TO ORDER OF VARIABLES IN COMMON BLOCKS:
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C*****(1) BASVAL, (2) BINNED, AND (3) EXPERT
C*****AS IF THESE COMMON BLOCKS ARE CONCATENTATED.
      DATA NAME /
     1   'FCOR', 'FVES', 'DFVPA', 'DFCPA',
     2   'FEVSE', 'FDCH', 'FCCI', 'DFCAV',
     3   'VBPUF', 'FCONV', 'FCONC', 'RBDF',
     4   'DFSPRV', 'DFSPRC', 'FREVO', 'VALISS',
     5   'FLTI1', 'FLTI2', 'NSPEC', 'FLV', 'FHPE', 'EVSE', 'WFAC',
     6   'PFAC', 'FPLBYE', 'FPLBYP', 'FPLBYD', 'FPLBYC', 'FTLPH',
     7   'FTLPL', 'FTLP', 'TC11', 'TC12', 'TB11', 'TB12', 'TB21',
     8   'TB22', 'TBS1', 'TBS2', 'TBR1', 'TBR2', 'TW', 'T1', 'T2',
     9   'DT1', 'DT2', 'DTCDB', 'ELEV', 'PUFF', 'HVSPLT', 'FCD',
     A   'FCOR0', 'FVES0', 'DFVPA0',
     B   'DFCPA0', 'FDCH0', 'FEVSE0',
     C   'FCCI0', 'DFCAV0', 'VBPUF0',
     D   'FCONV0', 'FCONC0', 'RBDF0',
     E   'DFSPRV0', 'DFSPRC0', 'FREVO0', 'FLTI10',
     F   'FLTI20', 'FHPE0', 'EVSE0', 'FPLBY0', 'HVSPLT0',
     G   'TW0', 'T10', 'DT10', 'DT20', 'PUFF0', 'E0',
     H   'FCORL', 'FVESL',
     I   'FREVOL', 'FCCIL',
     J   'FCONVL', 'FCONCL',
     K   'FLTI1L', 'FLTI2L', 'RBDFL',
     L   'FDCHL', 'FEVSEL',
     M   'DFVPAL', 'DFCPAL',
     N   'DFCAVL', 'DFSPRVL',
     O   'DFSPRCL', 'PRBLEV',
     P   5*' ' /
C*****DEFINE 3 DIMENSIONS FOR EACH OF THE VARIABLES
      DATA IDIMEN /
     1   MAXSPC,1,1, MAXSPC,1,1, MAXSPC,1,1, MAXSPC,1,1,
     2   MAXSPC,1,1, MAXSPC,1,1, MAXSPC,1,1, MAXSPC,1,1,
     3   MAXSPC,1,1, MAXSPC,1,1, MAXSPC,1,1, MAXSPC,1,1,
     4   MAXSPC,1,1, MAXSPC,1,1, MAXSPC,1,1, MAXISS,1,1,
     5   1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1,
     6   1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1,
     7   1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1,
     8   1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1,
     9   1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1,
     A   MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1,
     B   MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1,
     C   MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1,
     D   MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1,
     E   MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1, MAXCAS,1,1,
     F   MAXCAS,1,1, MAXCAS,1,1, MAXCAS,1,1, 3,1,1, 1,1,1, MAXTIM,1,1,
     G   MAXTIM,1,1, MAXTIM,1,1, MAXTIM,1,1, MAXTIM,1,1, MAXTIM,1,1,
     H   MAXSPC,MAXLEV,MAXCAS, MAXSPC,MAXLEV,MAXCAS,
     I   MAXSPC,MAXLEV,MAXCAS, MAXSPC,MAXLEV,MAXCAS,
     J   MAXSPC,MAXLEV,MAXCAS, MAXSPC,MAXLEV,MAXCAS,
     K   MAXLEV,MAXCAS,1, MAXLEV,MAXCAS,1, MAXSPC,MAXLEV,MAXCAS,
     L   MAXSPC,MAXLEV,MAXCAS, MAXSPC,MAXLEV,MAXCAS,
     M   MAXSPC,MAXLEV,MAXCAS, MAXSPC,MAXLEV,MAXCAS,
     N   MAXSPC,MAXLEV,MAXCAS, MAXSPC,MAXLEV,MAXCAS,
     O   MAXSPC,MAXLEV,MAXCAS, MAXLEV,1,1,
     P   15*0 /
C*****DEFINE NUMBERS OF VALUES IN COMMON BLOCKS:
C*****(1) BASVAL, (2) BINNED, AND (3) EXPERT
      DATA NVCB1 / 205 /, NVCB2 / 1356 /, NVCB3 / 11370 /, NVCB4 / 0 /,
     1     NVCB5 / 0 /
      END
      SUBROUTINE INPUT
C*****PROCESS KEYWORD INPUT ON UNIT 5
      PARAMETER (MAXLEN=101)
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      COMMON /CONTRL/ NLHS, NOBS, NSTART, NBIN, NDM, NTOT
      LOGICAL NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1    EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /KEYS/ NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
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     1    EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      CHARACTER BINARR*(MAXBD), BTITLE*80, TITLE*80
      COMMON /BINS/ BINARR(MAXBIN), BTITLE, TITLE
      CHARACTER*80 FILNAM
      CHARACTER*80 DEFFIL, SAMFIL, VECFIL
      COMMON /FILBLK/ DEFFIL, SAMFIL, VECFIL
      CHARACTER CARD*(MAXLEN), CVAL*(MAXLEN), KEYWRD*20
C*****SET LOGICAL TYPES FOR FREE FORMAT SUBROUTINE RDSTRG
      LOGICAL EOR, LVAL, TYPE(4)
C
C
C*****INITIALLIZE COLUMN POINTER FOR CURRENT RECORD
      IC=1
C*****READ RECORD
      READ(5,1001) CARD
C*****READ MODE SWITCH
      CALL RDSTRG (CARD, IC, KEYWRD, LVAL, IVAL, RVAL, KLNGTH, TYPE,
     1             EOR)
C*****CHECK FOR BINNED OR DIRECT EXECUTION
      IF (KEYWRD(1:KLNGTH) .EQ. 'BINNED') THEN
C********SET BINNED EXECUTION TYPE
         BINNED=.TRUE.
      ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'DIRECT') THEN
C********SET DIRECT EXECUTION TYPE
         BINNED=.FALSE.
      ELSE
C********MODE SWITCH WAS NEITHER BINNED NOR DIRECT SO PRINT ERROR MESSAGE
         WRITE(6,5030)
         STOP
      ENDIF
C*****SET DEFAULT VALUES
      SAMPLE=.FALSE.
      NOCALC=.FALSE.
      PRTINP=.FALSE.
      NOBS=1
      REPRTB=.FALSE.
      BYRUN=.FALSE.
      CONSFL=.FALSE.
      DIAG=.FALSE.
      EXPERT=.FALSE.
C*****INITIALLIZE NUMBER OF BINS
      NBIN=0
C*****READ TITLE
      READ(5,1001) TITLE
C*****PRINT MESSAGE FOR EXECUTION TYPE AND TITLE
      WRITE(6,1003) KEYWRD(1:KLNGTH), TITLE
      WRITE(6,1002) CARD
      WRITE(6,1002) TITLE
C*****PROCESS KEYWORDS
  666 CONTINUE
C*****READ RECORD
      READ(5,1001,END=6000) CARD
      WRITE(6,1002) CARD
C*****INITIALLIZE COLUMN POINTER FOR CURRENT RECORD
      IC=1
  500 CONTINUE
C*****READ CHARACTER STRING FOR COMPARISON AGAINST KEYWORDS
      CALL RDSTRG (CARD, IC, KEYWRD, LVAL, IVAL, RVAL, KLNGTH, TYPE,
     1             EOR)
C*****CHECK FOR END-OF-RECORD
      IF (EOR) GO TO 666
C*****CHECK CHARACTER STRING AGAINST KEYWORDS
      IF (KEYWRD(1:KLNGTH) .EQ. 'SAMPLE') THEN
C********SET SAMPLE TYPE TO .TRUE.
         SAMPLE=.TRUE.
C********OBTAIN SAMPLE INFORMATION
C********OBTAIN NUMBER OF SAMPLE VECTORS TO BE EXECUTED
         CALL RDSTRG (CARD, IC, CVAL, LVAL, NOBS, RVAL, LENGTH, TYPE,
     1                EOR)
C********CHECK FOR INTEGER VALUE
```

```
         IF (TYPE(3)) THEN
C***********OBTAIN SAMPLE VECTOR NUMBER TO BEGIN EXECUTION
            CALL RDSTRG (CARD, IC, CVAL, LVAL, NSTART, RVAL, LENGTH,
     1                   TYPE, EOR)
C***********CHECK FOR INTEGER VALUE
            IF (TYPE(3)) THEN
C***************OBTAIN NAME OF FILE CONTAINING SAMPLE VECTORS
               CALL RDSTRG (CARD, IC, FILNAM, LVAL, IVAL, RVAL, LENGTH,
     1                      TYPE, EOR)
C***************CHECK FOR CHARACTER VALUE
               IF (TYPE(1)) THEN
C******************CHARACTER VALUE FOUND, OPEN SAMPLE VECTOR FILE
                  OPEN(3, FILE=FILNAM, STATUS='OLD', ERR=8000)
C******************SKIP HEADER RECORDS ON SAMPLE VECTOR FILE
  600             CONTINUE
                  READ(3,1001,ERR=8100) CVAL
                  IND=INDEX(CVAL,'@SAMPLEDATA')
                  IF (IND .EQ. 0) GO TO 600
               ELSE
C******************PRINT ERROR MESSAGE
                  GO TO 9200
               ENDIF
            ELSE
C***************PRINT ERROR MESSAGE
               GO TO 9100
            ENDIF
         ELSE
C***********PRINT ERROR MESSAGE
            GO TO 9100
         ENDIF
      ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'NORUN') THEN
C********SET TYPE FOR VALIDATION OF INPUT ONLY, NO EXECUTION
         NOCALC=.TRUE.
      ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'DEFAULT') THEN
C********READ NAME OF FILE CONTAINING DEFAULT VALUES
         CALL RDSTRG (CARD, IC, DEFFIL, LVAL, IVAL, RVAL, LENGTH, TYPE,
     1                EOR)
C********CHECK FOR CHARACTER VALUE
         IF (.NOT. TYPE(1)) GO TO 9100
      ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'VECPOS') THEN
C********READ NAME OF FILE CONTAINING SAMPLE VECTOR POSTION INFORMATION
         CALL RDSTRG (CARD, IC, SAMFIL, LVAL, IVAL, RVAL, LENGTH, TYPE,
     1                EOR)
C********CHECK FOR CHARACTER VALUE
         IF (.NOT. TYPE(1)) GO TO 9100
      ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'BINFILE') THEN
C********READ A BIN ARRAY FILE
C********CHECK FOR BINNED EXECUTION
         IF (BINNED) THEN
C***********READ NAME OF FILE CONTAINING BIN INFORMATION
            CALL RDSTRG (CARD, IC, FILNAM, LVAL, IVAL, RVAL, LENGTH,
     1                   TYPE, EOR)
C***********CHECK FOR CHARACTER VALUE
            IF (TYPE(1)) THEN
C***************CHARACTER VALUE FOUND SO OPEN BIN FILE
               OPEN(4, FILE=FILNAM, STATUS='OLD', ERR=8000)
            ELSE
C***************PRINT ERROR MESSAGE
               GO TO 9200
            ENDIF
         ELSE
C***********NO BINS USED FOR DIRECT EXECUTION, PRINT ERROR MESSAGE
            GO TO 9300
         ENDIF
      ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'PRTINP') THEN
C********SET CONTROL FLAG PRTINP
         PRTINP=.TRUE.
      ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'REPORTB') THEN
C********SET CONTROL FLAG REPORTB
         REPRTB=.TRUE.
```

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C*****(1) BASVAL, (2) BINNED, AND (3) EXPERT
C*****AS IF THESE COMMON BLOCKS ARE CONCATENTATED.
      DATA NAME /
     1   'FCOR', 'FVES', 'DFVPA', 'DFCPA',
     2   'FEVSE', 'FDCH', 'FCCI', 'DFCAV',
     3   'VBPUF', 'FCONV', 'FCONC', 'RBDF',
     4   'DFSPRV', 'DFSPRC', 'FREVO', 'VALISS',
     5   'FLTI1', 'FLTI2', 'NSPEC', 'FLV', 'FHPE', 'EVSE', 'WFAC',
     6   'PFAC', 'FPLBYE', 'FPLBYP', 'FPLBYD', 'FPLBYC', 'FTLPH',
     7   'FTLPL', 'FTLP', 'TC11', 'TC12', 'TB11', 'TB12', 'TB21',
     8   'TB22', 'TBS1', 'TBS2', 'TBR1', 'TBR2', 'TW', 'T1', 'T2',
     9   'DT1', 'DT2', 'DTCDB', 'ELEV', 'PUFF', 'HVSPLT', 'FCD',
     A   'FCOR0', 'FVES0', 'DFVPA0',
     B   'DFCPA0', 'FDCH0', 'FEVSE0',
     C   'FCCI0', 'DFCAV0', 'VBPUF0',
     D   'FCONV0', 'FCONC0', 'RBDF0',
     E   'DFSPRV0', 'DFSPRC0', 'FREVO0', 'FLTI10',
     F   'FLTI20', 'FHPE0', 'EVSE0', 'FPLBY0', 'HVSPLT0',
     G   'TW0', 'T10', 'DT10', 'DT20', 'PUFF0', 'E0',
     H   'FCORL', 'FVESL',
     I   'FREVOL', 'FCCIL',
     J   'FCONVL', 'FCONCL',
     K   'FLTI1L', 'FLTI2L', 'RBDFL',
     L   'FDCHL', 'FEVSEL',
     M   'DFVPAL', 'DFCPAL',
     N   'DFCAVL', 'DFSPRVL',
     O   'DFSPRCL', 'PRBLEV',
     P   5*' ' /
C*****DEFINE 3 DIMENSIONS FOR EACH OF THE VARIABLES
      DATA IDIMEN /
     1   MAXSPC,1,1, MAXSPC,1,1, MAXSPC,1,1, MAXSPC,1,1,
     2   MAXSPC,1,1, MAXSPC,1,1, MAXSPC,1,1, MAXSPC,1,1,
     3   MAXSPC,1,1, MAXSPC,1,1, MAXSPC,1,1, MAXSPC,1,1,
     4   MAXSPC,1,1, MAXSPC,1,1, MAXSPC,1,1, MAXISS,1,1,
     5   1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1,
     6   1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1,
     7   1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1,
     8   1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1,
     9   1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1, 1,1,1,
     A   MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1,
     B   MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1,
     C   MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1,
     D   MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1,
     E   MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1, MAXSPC,MAXCAS,1, MAXCAS,1,1,
     F   MAXCAS,1,1, MAXCAS,1,1, MAXCAS,1,1, 3,1,1, 1,1,1, MAXTIM,1,1,
     G   MAXTIM,1,1, MAXTIM,1,1, MAXTIM,1,1, MAXTIM,1,1, MAXTIM,1,1,
     H   MAXSPC,MAXLEV,MAXCAS, MAXSPC,MAXLEV,MAXCAS,
     I   MAXSPC,MAXLEV,MAXCAS, MAXSPC,MAXLEV,MAXCAS,
     J   MAXSPC,MAXLEV,MAXCAS, MAXSPC,MAXLEV,MAXCAS,
     K   MAXLEV,MAXCAS,1, MAXLEV,MAXCAS,1, MAXSPC,MAXLEV,MAXCAS,
     L   MAXSPC,MAXLEV,MAXCAS, MAXSPC,MAXLEV,MAXCAS,
     M   MAXSPC,MAXLEV,MAXCAS, MAXSPC,MAXLEV,MAXCAS,
     N   MAXSPC,MAXLEV,MAXCAS, MAXSPC,MAXLEV,MAXCAS,
     O   MAXSPC,MAXLEV,MAXCAS, MAXLEV,1,1,
     P   15*0 /
C*****DEFINE NUMBERS OF VALUES IN COMMON BLOCKS:
C*****(1) BASVAL, (2) BINNED, AND (3) EXPERT
      DATA NVCB1 / 205 /, NVCB2 / 1356 /, NVCB3 / 11370 /, NVCB4 / 0 /,
     1     NVCB5 / 0 /
      END
      SUBROUTINE INPUT
C*****PROCESS KEYWORD INPUT ON UNIT 5
      PARAMETER (MAXLEN=101)
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      COMMON /CONTRL/ NLHS, NOBS, NSTART, NBIN, NDM, NTOT
      LOGICAL NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1    EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /KEYS/ NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
```

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     1    EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      CHARACTER BINARR*(MAXBD), BTITLE*80, TITLE*80
      COMMON /BINS/ BINARR(MAXBIN), BTITLE, TITLE
      CHARACTER*80 FILNAM
      CHARACTER*80 DEFFIL, SAMFIL, VECFIL
      COMMON /FILBLK/ DEFFIL, SAMFIL, VECFIL
      CHARACTER CARD*(MAXLEN), CVAL*(MAXLEN), KEYWRD*20
C*****SET LOGICAL TYPES FOR FREE FORMAT SUBROUTINE RDSTRG
      LOGICAL EOR, LVAL, TYPE(4)
C
C
C*****INITIALLIZE COLUMN POINTER FOR CURRENT RECORD
      IC=1
C*****READ RECORD
      READ(5,1001) CARD
C*****READ MODE SWITCH
      CALL RDSTRG (CARD, IC, KEYWRD, LVAL, IVAL, RVAL, KLNGTH, TYPE,
     1             EOR)
C*****CHECK FOR BINNED OR DIRECT EXECUTION
      IF (KEYWRD(1:KLNGTH) .EQ. 'BINNED') THEN
C********SET BINNED EXECUTION TYPE
         BINNED=.TRUE.
      ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'DIRECT') THEN
C********SET DIRECT EXECUTION TYPE
         BINNED=.FALSE.
      ELSE
C********MODE SWITCH WAS NEITHER BINNED NOR DIRECT SO PRINT ERROR MESSAGE
         WRITE(6,5030)
         STOP
      ENDIF
C*****SET DEFAULT VALUES
      SAMPLE=.FALSE.
      NOCALC=.FALSE.
      PRTINP=.FALSE.
      NOBS=1
      REPRTB=.FALSE.
      BYRUN=.FALSE.
      CONSFL=.FALSE.
      DIAG=.FALSE.
      EXPERT=.FALSE.
C*****INITIALLIZE NUMBER OF BINS
      NBIN=0
C*****READ TITLE
      READ(5,1001) TITLE
C*****PRINT MESSAGE FOR EXECUTION TYPE AND TITLE
      WRITE(6,1003) KEYWRD(1:KLNGTH), TITLE
      WRITE(6,1002) CARD
      WRITE(6,1002) TITLE
C*****PROCESS KEYWORDS
  666 CONTINUE
C*****READ RECORD
      READ(5,1001,END=6000) CARD
      WRITE(6,1002) CARD
C*****INITIALLIZE COLUMN POINTER FOR CURRENT RECORD
      IC=1
  500 CONTINUE
C*****READ CHARACTER STRING FOR COMPARISON AGAINST KEYWORDS
      CALL RDSTRG (CARD, IC, KEYWRD, LVAL, IVAL, RVAL, KLNGTH, TYPE,
     1             EOR)
C*****CHECK FOR END-OF-RECORD
      IF (EOR) GO TO 666
C*****CHECK CHARACTER STRING AGAINST KEYWORDS
      IF (KEYWRD(1:KLNGTH) .EQ. 'SAMPLE') THEN
C********SET SAMPLE TYPE TO .TRUE.
         SAMPLE=.TRUE.
C********OBTAIN SAMPLE INFORMATION
C********OBTAIN NUMBER OF SAMPLE VECTORS TO BE EXECUTED
         CALL RDSTRG (CARD, IC, CVAL, LVAL, NOBS, RVAL, LENGTH, TYPE,
     1                EOR)
C********CHECK FOR INTEGER VALUE
```

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          IF (TYPE(3)) THEN
C***********OBTAIN SAMPLE VECTOR NUMBER TO BEGIN EXECUTION
            CALL RDSTRG (CARD, IC, CVAL, LVAL, NSTART, RVAL, LENGTH,
     1                   TYPE, EOR)
C***********CHECK FOR INTEGER VALUE
            IF (TYPE(3)) THEN
C***************OBTAIN NAME OF FILE CONTAINING SAMPLE VECTORS
              CALL RDSTRG (CARD, IC, FILNAM, LVAL, IVAL, RVAL, LENGTH,
     1                     TYPE, EOR)
C***************CHECK FOR CHARACTER VALUE
              IF (TYPE(1)) THEN
C*****************CHARACTER VALUE FOUND, OPEN SAMPLE VECTOR FILE
                OPEN(3, FILE=FILNAM, STATUS='OLD', ERR=8000)
C*****************SKIP HEADER RECORDS ON SAMPLE VECTOR FILE
  600           CONTINUE
                READ(3,1001,ERR=8100) CVAL
                IND=INDEX(CVAL,'@SAMPLEDATA')
                IF (IND .EQ. 0) GO TO 600
              ELSE
C*****************PRINT ERROR MESSAGE
                GO TO 9200
              ENDIF
            ELSE
C***************PRINT ERROR MESSAGE
              GO TO 9100
            ENDIF
          ELSE
C***********PRINT ERROR MESSAGE
            GO TO 9100
          ENDIF
        ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'NORUN') THEN
C********SET TYPE FOR VALIDATION OF INPUT ONLY, NO EXECUTION
          NOCALC=.TRUE.
        ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'DEFAULT') THEN
C********READ NAME OF FILE CONTAINING DEFAULT VALUES
          CALL RDSTRG (CARD, IC, DEFFIL, LVAL, IVAL, RVAL, LENGTH, TYPE,
     1                 EOR)
C********CHECK FOR CHARACTER VALUE
          IF (.NOT. TYPE(1)) GO TO 9100
        ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'VECPOS') THEN
C********READ NAME OF FILE CONTAINING SAMPLE VECTOR POSTION INFORMATION
          CALL RDSTRG (CARD, IC, SAMFIL, LVAL, IVAL, RVAL, LENGTH, TYPE,
     1                 EOR)
C********CHECK FOR CHARACTER VALUE
          IF (.NOT. TYPE(1)) GO TO 9100
        ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'BINFILE') THEN
C********READ A BIN ARRAY FILE
C********CHECK FOR BINNED EXECUTION
          IF (BINNED) THEN
C***********READ NAME OF FILE CONTAINING BIN INFORMATION
            CALL RDSTRG (CARD, IC, FILNAM, LVAL, IVAL, RVAL, LENGTH,
     1                   TYPE, EOR)
C***********CHECK FOR CHARACTER VALUE
            IF (TYPE(1)) THEN
C***************CHARACTER VALUE FOUND SO OPEN BIN FILE
              OPEN(4, FILE=FILNAM, STATUS='OLD', ERR=8000)
            ELSE
C***************PRINT ERROR MESSAGE
              GO TO 9200
            ENDIF
          ELSE
C***********NO BINS USED FOR DIRECT EXECUTION, PRINT ERROR MESSAGE
            GO TO 9300
          ENDIF
        ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'PRTINP') THEN
C********SET CONTROL FLAG PRTINP
          PRTINP=.TRUE.
        ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'REPORTB') THEN
C********SET CONTROL FLAG REPORTB
          REPRTB=.TRUE.
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      ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'KPBYRUN') THEN
C*********SET CONTROL FLAG KPBYRUN
         BYRUN=.TRUE.
      ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'CONSFL') THEN
C*********SET CONTROL FLAG CONSFL
         CONSFL=.TRUE.
      ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'DIAG') THEN
C*********SET DIAGNOSTIC PRINT CONTROL FLAG DIAG
         DIAG=.TRUE.
      ELSE IF (KEYWRD(1:KLNGTH) .EQ. 'EXPERT') THEN
C*********SET EXPERT OPINION CONTROL FLAG EXPERT
         EXPERT=.TRUE.
      ELSE
C*********INVALID KEYWORD, PRINT ERROR MESSAGE
         WRITE(6,5020) KEYWRD(1:KLNGTH)
         NOCALC=.TRUE.
      ENDIF
      GO TO 500
 6000 CONTINUE
C*****VALIDATE COMBINATION OF FLAGS
      IF (EXPERT .AND. ((.NOT. BINNED) .OR. (.NOT. SAMPLE))) THEN
         WRITE(6,6001)
         STOP
      ENDIF
      IF (NOCALC) STOP
C*****PRINT CONTROL INFORMATION
      IF (SAMPLE) WRITE(6,5025) NOBS, NSTART
C*****CALCULATE TOTAL NUMBER OF SOURCE TERMS FOR BINNED/SAMPLED EXECUTION
      IF (BINNED .AND. BYRUN) THEN
C*********READ BIN FILE TO DETERMINE TOTAL NUMBER OF SOURCE TERMS
         NTOT=0
         NSAMPL=NSTART + NOBS - 1
C*********READ TITLE RECORD
         READ(4,1001) BTITLE
         DO 7000 IOBS=1,NSAMPL
            READ(4,*) I, NDM, NBIN
            IF (IOBS .GE. NSTART) NTOT=NTOT + NBIN
            IF (NBIN .GT. 0) READ(4,1001) (BINARR(1)(1:NDM),I=1,NBIN)
 7000    CONTINUE
         REWIND 4
      ENDIF
      RETURN
C*****FILE OPEN ERROR
 8000 WRITE(6,5022) FILNAM, KEYWRD(1:KLNGTH)
      NOCALC=.TRUE.
      GO TO 500
C*****FILE READ ERROR
 8100 CONTINUE
      WRITE(6,5023) FILNAM, KEYWRD(1:KLNGTH)
      NOCALC=.TRUE.
      GO TO 500
 9100 CONTINUE
C*****PRINT ERROR MESSAGE FOR WRONG TYPE OF VARIABLE
      WRITE(5,9101) KEYWRD(1:KLNGTH)
      NOCALC=.TRUE.
      GO TO 500
 9200 CONTINUE
C*****PRINT ERROR MESSAGE FOR NO FILE NAME
      WRITE(6,9201) KEYWRD(1:KLNGTH)
      NOCALC=.TRUE.
      GO TO 500
 9300 CONTINUE
C*****PRINT ERROR MESSAGE FOR BINNED KEYWORD USED FOR DIRECT EXECUTION
      WRITE(6,9301) KEYWRD(1:KLNGTH)
      NOCALC=.TRUE.
      GO TO 500
C*****FORMAT STATEMENTS
 1001 FORMAT(A)
 1002 FORMAT(11X,A)
 1003 FORMAT(/1X,130('*'),
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     1         /1X,48('*'),5X,'GGSOR ',A,' EXECUTION',5X,48('*'),
     2         /1X,20('*'),5X,A,5X,20('*'),
     3         /1X,130('*'),/)
 1004 FORMAT(1X,A)
 5020 FORMAT(' >>>>>UNRECOGNIZED KEYWORD (',A,')',/)
 5022 FORMAT(' >>>>>OPEN ERROR ON FILE ',A,' FOR KEYWORD ',A,/)
 5023 FORMAT(' >>>>>READ ERROR ON FILE ',A,' FOR KEYWORD ',A,/)
 5025 FORMAT(//1X,'THE INPUT WILL BE SAMPLED WITH ',I4,
     1       ' SAMPLE VECTOR(S) STARTING WITH SAMPLE VECTOR ',I4,//)
 5030 FORMAT(1X,'>>>>>UNRECOGNIZED MODE SWITCH',/)
 6001 FORMAT(1X,'>>>>>BINNED AND SAMPLE FLAGS MUST BE SPECIFIED ',
     1       'TO USE EXPERT OPINION TABLES')
 9101 FORMAT(1X,'>>>>>VALUE(S) FOLLOWING KEYWORD ',A,' INVALID',/)
 9201 FORMAT(1X,'>>>>>NO FILE NAME FOUND FOLLOWING KEYWORD ',A,/)
 9301 FORMAT(1X,'>>>>>INVALID KEYWORD (',A,') SPECIFIED FOR DIRECT ',
     1       'EXECUTION',/)
 9501 FORMAT(1X,'>>>>>UNABLE TO LOCATE EVENT NAME ',A,' DURING ',
     1        'PROCESSING OF KEYWORD ',A,/)
      END
      SUBROUTINE DIR
C*****IMPLEMENTS GGSOR RUNS WHICH INVOLVE DIRECT INPUT WITHOUT
C*****SAMPLING
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      CHARACTER BINARR*(MAXBD), BTITLE*80, TITLE*80
      COMMON /BINS/ BINARR(MAXBIN), BTITLE, TITLE
      COMMON /CONTRL/ NLHS, NOBS, NSTART, NBIN, NDM, NTOT
      CHARACTER*7 NAME
      LOGICAL LDEFLT, LREAL
      COMMON /DEFLT1/ NAME(MAXVAR)
      COMMON /DEFLT2/ NVAR, NVAL, NVCB1, NVCB2, NVCB3, NVCB4,
     1                NVCB5, IDIMEN(3,MAXVAR), ISPOS(MAXVAR),
     2                ISMPPS(MAXVAL), IPNT(MAXVAR), LDEFLT(MAXVAL),
     3                LREAL(MAXVAL)
      LOGICAL NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1    EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /KEYS/ NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1    EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /SRCTRM/ ST(MAXSPC), STE(MAXSPC), STCCI(MAXSPC),
     1                STL(MAXSPC), STIL, STRVOL(MAXSPC),
     2                ST1(MAXSPC), ST2(MAXSPC), RV(MAXSPC)
      COMMON /BASVAL/ FCOR(MAXSPC), FVES(MAXSPC), DFVPA(MAXSPC),
     1                DFCPA(MAXSPC), FEVSE(MAXSPC), FDCH(MAXSPC),
     2                FCCI(MAXSPC), DFCAV(MAXSPC), VBPUF(MAXSPC),
     3                FCONV(MAXSPC), FCONC(MAXSPC), RBDF(MAXSPC),
     4                DFSPRV(MAXSPC), DFSPRC(MAXSPC), FREVO(MAXSPC),
     5                VALISS(MAXISS), FLTI1, FLTI2, NSPEC, FLV, FHPE,
     6                EVSE, WFAC, PFAC, FPLBYE, FPLBYP, FPLBYD,
     7                FPLBYC, FTLPH, FTLPL, FTLP, TC11, TC12, TB11,
     8                TB12, TB21, TB22, TBS1, TBS2, TBR1, TBR2, TW,
     9                T1, T2, DT1, DT2, DTCDB, ELEV, PUFF, HVSPLT, FCD
      COMMON /BINNED/ FCOR0(MAXSPC,MAXCAS), FVES0(MAXSPC,MAXCAS),
     1                DFVPA0(MAXSPC,MAXCAS), DFCPA0(MAXSPC,MAXCAS),
     2                FDCH0(MAXSPC,MAXCAS), FEVSE0(MAXSPC,MAXCAS),
     3                FCCI0(MAXSPC,MAXCAS), DFCAV0(MAXSPC,MAXCAS),
     4                VBPUF0(MAXSPC,MAXCAS), FCONV0(MAXSPC,MAXCAS),
     5                FCONC0(MAXSPC,MAXCAS), RBDF0(MAXSPC,MAXCAS),
     6                DFSPRV0(MAXSPC,MAXCAS), DFSPRC0(MAXSPC,MAXCAS),
     7                FREVO0(MAXSPC,MAXCAS), FLTI10(MAXCAS),
     8                FLTI20(MAXCAS), FHPE0(MAXCAS), EVSE0(MAXCAS),
     9                FPLBY0(3), HVSPLT0, TW0(MAXTIM), T10(MAXTIM),
     A                DT10(MAXTIM), DT20(MAXTIM), PUFF0(MAXTIM),
     B                E0(MAXTIM)
      COMMON /EXPERT/ FCORL(MAXSPC,MAXLEV,MAXCAS),
     1                FVESL(MAXSPC,MAXLEV,MAXCAS),
     2                FREVOL(MAXSPC,MAXLEV,MAXCAS),
     3                FCCIL(MAXSPC,MAXLEV,MAXCAS),
     4                FCONVL(MAXSPC,MAXLEV,MAXCAS),
     5                FCONCL(MAXSPC,MAXLEV,MAXCAS),
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     6                 FLTI1L(MAXLEV,MAXCAS), FLTI2L(MAXLEV,MAXCAS),
     7                 RBDFL(MAXSPC,MAXLEV,MAXCAS),
     8                 FDCHL(MAXSPC,MAXLEV,MAXCAS),
     9                 FEVSEL(MAXSPC,MAXLEV,MAXCAS),
     A                 DFVPAL(MAXSPC,MAXLEV,MAXCAS),
     B                 DFCPAL(MAXSPC,MAXLEV,MAXCAS),
     C                 DFCAVL(MAXSPC,MAXLEV,MAXCAS),
     D                 DFSPRVL(MAXSPC,MAXLEV,MAXCAS),
     E                 DFSPRCL(MAXSPC,MAXLEV,MAXCAS),
     F                 PRBLEV(MAXLEV)
      COMMON /LHSBLK/ XLHS(MAXSMP)
      DATA IOBS / 1 /, IBIN / 1 /
C
C
C*****SET NUMBER OF SAMPLE VALUES
      XLHS(1)=0.0
C*****DEFINE VARIABLE NAMES FOR DEFAULT INPUT
      CALL DEFINE
C*****SET DEFAULT VALUE ARRAY BY READING DEFAULT INPUT FROM FILE DEFFIL
      CALL SETDEF
C*****PRINT DEFAULT VALUE INFORMATION
      IF (PRTINP) CALL WRTPAR
C*****TERMINATE EXECUTION IF ONLY VALIDATING INPUT OR ERROR ENCOUNTERED
C*****DURING READING OF INPUT DATA
      IF (NOCALC) THEN
C********PRINT MESSAGE
         WRITE(6,1010)
         STOP
      ENDIF
C*****TRANSFER DEFAULT VALUE ARRAY TO COMMON BLOCK VALUES
      CALL TRANS (FCOR(1), FCOR0(1,1), FCORL(1,1,1))
C*****PRINT CONTENTS OF COMMON BLOCKS
      IF (REPRTB) CALL WRREL
C*****SET TOTAL NUMBER OF SOURCE TERMS
      NTOT=1
C*****WRITE HEADER TO CONSEQUENCE DATA FILE
      IF (CONSFL) WRITE(9,1006) TITLE, NDM, NSPEC, NTOT, NOBS
C*****WRITE STANDARD HEADER TO SPECIALIZED CONSEQUENCE DATA FILE
C///      IF (CONSFL) WRITE(10,5001) TITLE, NDM, NSPEC, NTOT, NOBS
C*****PERFORM SOURCE TERM CALCULATIONS
      CALL GGSORC (IOBS, IBIN)
C*****PRINT PROCESSING SUMMARY
      WRITE(6,2003)
      RETURN
C*****FORMAT STATEMENTS
 1006 FORMAT(1X,A,/1X,4I10)
 1010 FORMAT(/1X,'EXECUTION TERMINATED FOLLOWING VALIDATION OF INPUT')
 2003 FORMAT(/1X,'SINGLE DIRECT EXECUTION PROCESSED')
 5001 FORMAT(1X,A,/,1X,'NDM = ',I5,'    NSPEC =  ',I5,'    NTOT=  ',I5,
     1'    NOBS = ',I5)
      END
      SUBROUTINE DIRSMP
C*****IMPLEMENTS GGSOR RUNS WHICH INVOLVE DIRECT INPUT WITH
C*****SAMPLING
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      CHARACTER BINARR*(MAXBD), BTITLE*80, TITLE*80
      COMMON /BINS/ BINARR(MAXBIN), BTITLE, TITLE
      COMMON /CONTRL/ NLHS, NOBS, NSTART, NBIN, NDM, NTOT
      CHARACTER*7 NAME
      LOGICAL LDEFLT, LREAL
      COMMON /DEFLT1/ NAME(MAXVAR)
      COMMON /DEFLT2/ NVAR, NVAL, NVCB1, NVCB2, NVCB3, NVCB4,
     1                NVCB5, IDIMEN(3,MAXVAR), ISPOS(MAXVAR),
     2                ISMPPS(MAXVAL), IPNT(MAXVAR), LDEFLT(MAXVAL),
     3                LREAL(MAXVAL)
      LOGICAL NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1   EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /KEYS/ NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
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     1    EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /SRCTRM/ ST(MAXSPC), STE(MAXSPC), STCCI(MAXSPC),
     1                STL(MAXSPC), STIL, STRVOL(MAXSPC),
     2                ST1(MAXSPC), ST2(MAXSPC), RV(MAXSPC)
      COMMON /BASVAL/ FCOR(MAXSPC), FVES(MAXSPC), DFVPA(MAXSPC),
     1                DFCPA(MAXSPC), FEVSE(MAXSPC), FDCH(MAXSPC),
     2                FCCI(MAXSPC), DFCAV(MAXSPC), VBPUF(MAXSPC),
     3                FCONV(MAXSPC), FCONC(MAXSPC), RBDF(MAXSPC),
     4                DFSPRV(MAXSPC), DFSPRC(MAXSPC), FREVO(MAXSPC),
     5                VALISS(MAXISS), FLTI1, FLTI2, NSPEC, FLV, FHPE,
     6                EVSE, WFAC, PFAC, FPLBYE, FPLBYP, FPLBYD,
     7                FPLBYC, FTLPH, FTLPL, FTLP, TC11, TC12, TB11,
     8                TB12, TB21, TB22, TBS1, TBS2, TBR1, TBR2, TW,
     9                T1, T2, DT1, DT2, DTCDB, ELEV, PUFF, HVSPLT, FCD
      COMMON /BINNED/ FCOR0(MAXSPC,MAXCAS), FVES0(MAXSPC,MAXCAS),
     1                DFVPA0(MAXSPC,MAXCAS), DFCPA0(MAXSPC,MAXCAS),
     2                FDCH0(MAXSPC,MAXCAS), FEVSE0(MAXSPC,MAXCAS),
     3                FCCI0(MAXSPC,MAXCAS), DFCAV0(MAXSPC,MAXCAS),
     4                VBPUF0(MAXSPC,MAXCAS), FCONV0(MAXSPC,MAXCAS),
     5                FCONC0(MAXSPC,MAXCAS), RBDF0(MAXSPC,MAXCAS),
     6                DFSPRV0(MAXSPC,MAXCAS), DFSPRC0(MAXSPC,MAXCAS),
     7                FREVO0(MAXSPC,MAXCAS), FLTI10(MAXCAS),
     8                FLTI20(MAXCAS), FHPE0(MAXCAS), EVSE0(MAXCAS),
     9                FPLBY0(3), HVSPLT0, TW0(MAXTIM), T10(MAXTIM),
     A                DT10(MAXTIM), DT20(MAXTIM), PUFF0(MAXTIM),
     B                E0(MAXTIM)
      COMMON /EXPERT/ FCORL(MAXSPC,MAXLEV,MAXCAS),
     1                FVESL(MAXSPC,MAXLEV,MAXCAS),
     2                FREVOL(MAXSPC,MAXLEV,MAXCAS),
     3                FCCIL(MAXSPC,MAXLEV,MAXCAS),
     4                FCONVL(MAXSPC,MAXLEV,MAXCAS),
     5                FCONCL(MAXSPC,MAXLEV,MAXCAS),
     6                FLTI1L(MAXLEV,MAXCAS), FLTI2L(MAXLEV,MAXCAS),
     7                RBDFL(MAXSPC,MAXLEV,MAXCAS),
     8                FDCHL(MAXSPC,MAXLEV,MAXCAS),
     9                FEVSEL(MAXSPC,MAXLEV,MAXCAS),
     A                DFVPAL(MAXSPC,MAXLEV,MAXCAS),
     B                DFCPAL(MAXSPC,MAXLEV,MAXCAS),
     C                DFCAVL(MAXSPC,MAXLEV,MAXCAS),
     D                DFSPRVL(MAXSPC,MAXLEV,MAXCAS),
     E                DFSPRCL(MAXSPC,MAXLEV,MAXCAS),
     F                PRBLEV(MAXLEV)
      COMMON /LHSBLK/ XLHS(MAXSMP)
C
C
C*****DEFINE VARIABLE NAMES AND POSITION INFORMATION FOR DEFAULT
C*****INPUT AND SAMPLE VECTOR SUBSTITUTION
      CALL DEFINE
C*****SET DEFAULT VALUE ARRAY BY READING DEFAULT INPUT FROM FILE DEFFIL
      CALL SETDEF
C*****SET SAMPLE VECTOR POSITIONS
      CALL VECPOS
C*****PRINT DEFAULT VALUE AND SAMPLE VECTOR POSITION INFORMATION
      IF (PRTINP) CALL WRTPAR
C*****TERMINATE EXECUTION IF ONLY VALIDATING INPUT OR ERROR ENCOUNTERED
C*****DURING READING OF INPUT DATA
      IF (NOCALC) THEN
C********PRINT MESSAGE
         WRITE(6,1010)
         STOP
      ENDIF
C*****SKIP TO STARTING SAMPLE VECTOR
      IF (NSTART .NE. 1) THEN
         DO 1000 ISKIP=1,NSTART-1
            READ(3,*) I, NLHS, (XLHS(I),I=1,NLHS)
 1000    CONTINUE
      ENDIF
C*****SET TOTAL NUMBER OF SOURCE TERMS
      NTOT=NOBS
C*****PROCESS SAMPLE VECTORS
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      DO 2000 IOBS=1,NOBS
C*********READ CURRENT SAMPLE VECTOR
         READ(3,*) I, NLHS, (XLHS(I),I=1,NLHS)
C*********TRANSFER SAMPLE VECTOR VALUES TO DEFAULT ARRAY
         CALL SUBVEC
C*********TRANSFER DEFAULT VALUE ARRAY TO COMMON BLOCK VALUES
         CALL TRANS (FCOR(1), FCOR0(1,1), FCORL(1,1,1))
C*********PRINT CONTENTS OF COMMON BLOCKS
         IF (REPRTB) CALL WRREL
C*********WRITE HEADER TO CONSEQUENCE DATA FILE
         IF (CONSFL .AND. (IOBS .EQ. 1))
     1        WRITE(9,1006) TITLE, NDM, NSPEC, NTOT, NOBS
C*********WRITE STANDARD HEADER TO SPECIALIZED CONSEQUENCE DATA FILE
C///        IF (CONSFL .AND. (IOBS .EQ. 1))
C///    1        WRITE(10,5001) TITLE, NDM, NSPEC, NTOT, NOBS
C*********PERFORM SOURCE TERM CALCULATIONS
         CALL GGSORC (IOBS+NSTART-1, IBIN)
 2000 CONTINUE
C*****PRINT PROCESSING SUMMARY
      IF (NOBS .EQ. 1) THEN
         WRITE(6,2003) NOBS
      ELSE
         WRITE(6,2004) NSTART, NSTART+NOBS-1, NOBS
      ENDIF
      RETURN
C*****FORMAT STATEMENTS
 1006 FORMAT(1X,A,/1X,4I10)
 1010 FORMAT(/1X,'EXECUTION TERMINATED FOLLOWING VALIDATION OF INPUT')
 2003 FORMAT(/1X,I5,' SAMPLE VECTOR PROCESSED')
 2004 FORMAT(/1X,'SAMPLE VECTORS ',I4,' THRU ',I5,
     1         ' WERE PROCESSED (TOTAL OF ',I5,')')
 5001 FORMAT(1X,A,/,1X,'NDM = ',I5,'    NSPEC =  ',I5,'    NTOT=  ',I5,
     1'    NOBS = ',I5)
      END
      SUBROUTINE BIN
C*****IMPLEMENTS GGSOR RUNS WHICH INVOLVE BINNED INPUT WITHOUT
C*****SAMPLING
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      CHARACTER BINARR*(MAXBD), BTITLE*80, TITLE*80
      COMMON /BINS/ BINARR(MAXBIN), BTITLE, TITLE
      COMMON /CONTRL/ NLHS, NOBS, NSTART, NBIN, NDM, NTOT
      CHARACTER*7 NAME
      LOGICAL LDEFLT, LREAL
      COMMON /DEFLT1/ NAME(MAXVAR)
      COMMON /DEFLT2/ NVAR, NVAL, NVCB1, NVCB2, NVCB3, NVCB4,
     1                NVCB5, IDIMEN(3,MAXVAR), ISPOS(MAXVAR),
     2                ISMPPS(MAXVAL), IPNT(MAXVAR), LDEFLT(MAXVAL),
     3                LREAL(MAXVAL)
      LOGICAL NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1   EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /KEYS/ NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1   EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /SRCTRM/ ST(MAXSPC), STE(MAXSPC), STCCI(MAXSPC),
     1                STL(MAXSPC), STIL, STRVOL(MAXSPC),
     2                ST1(MAXSPC), ST2(MAXSPC), RV(MAXSPC)
      COMMON /BASVAL/ FCOR(MAXSPC), FVES(MAXSPC), DFVPA(MAXSPC),
     1                DFCPA(MAXSPC), FEVSE(MAXSPC), FDCH(MAXSPC),
     2                FCCI(MAXSPC), DFCAV(MAXSPC), VBPUF(MAXSPC),
     3                FCONV(MAXSPC), FCONC(MAXSPC), RBDF(MAXSPC),
     4                DFSPRV(MAXSPC), DFSPRC(MAXSPC), FREVO(MAXSPC),
     5                VALISS(MAXISS), FLTI1, FLTI2, NSPEC, FLV, FHPE,
     6                EVSE, WFAC, PFAC, FPLBYE, FPLBYP, FPLBYD,
     7                FPLBYC, FTLPH, FTLPL, FTLP, TC11, TC12, TB11,
     8                TB12, TB21, TB22, TBS1, TBS2, TBR1, TBR2, TW,
     9                T1, T2, DT1, DT2, DTCDB, ELEV, PUFF, HVSPLT, FCD
      COMMON /BINNED/ FCOR0(MAXSPC,MAXCAS), FVES0(MAXSPC,MAXCAS),
     1                DFVPA0(MAXSPC,MAXCAS), DFCPA0(MAXSPC,MAXCAS),
     2                FDCH0(MAXSPC,MAXCAS), FEVSE0(MAXSPC,MAXCAS),
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     3                    FCCI0(MAXSPC,MAXCAS), DFCAV0(MAXSPC,MAXCAS),
     4                    VBPUF0(MAXSPC,MAXCAS), FCONV0(MAXSPC,MAXCAS),
     5                    FCONC0(MAXSPC,MAXCAS), RBDF0(MAXSPC,MAXCAS),
     6                    DFSPRV0(MAXSPC,MAXCAS), DFSPRC0(MAXSPC,MAXCAS),
     7                    FREVO0(MAXSPC,MAXCAS), FLTI10(MAXCAS),
     8                    FLTI20(MAXCAS), FHPE0(MAXCAS), EVSE0(MAXCAS),
     9                    FPLBY0(3), HVSPLT0, TW0(MAXTIM), T10(MAXTIM),
     A                    DT10(MAXTIM), DT20(MAXTIM), PUFF0(MAXTIM),
     B                    E0(MAXTIM)
      COMMON /EXPERT/ FCORL(MAXSPC,MAXLEV,MAXCAS),
     1                    FVESL(MAXSPC,MAXLEV,MAXCAS),
     2                    FREVOL(MAXSPC,MAXLEV,MAXCAS),
     3                    FCCIL(MAXSPC,MAXLEV,MAXCAS),
     4                    FCONVL(MAXSPC,MAXLEV,MAXCAS),
     5                    FCONCL(MAXSPC,MAXLEV,MAXCAS),
     6                    FLTI1L(MAXLEV,MAXCAS), FLTI2L(MAXLEV,MAXCAS),
     7                    RBDFL(MAXSPC,MAXLEV,MAXCAS),
     8                    FDCHL(MAXSPC,MAXLEV,MAXCAS),
     9                    FEVSEL(MAXSPC,MAXLEV,MAXCAS),
     A                    DFVPAL(MAXSPC,MAXLEV,MAXCAS),
     B                    DFCPAL(MAXSPC,MAXLEV,MAXCAS),
     C                    DFCAVL(MAXSPC,MAXLEV,MAXCAS),
     D                    DFSPRVL(MAXSPC,MAXLEV,MAXCAS),
     E                    DFSPRCL(MAXSPC,MAXLEV,MAXCAS),
     F                    PRBLEV(MAXLEV)
      COMMON /LHSBLK/ XLHS(MAXSMP)
      DATA IOBS / 1 /
C
C
C*****DEFINE VARIABLE NAMES FOR DEFAULT INPUT
      CALL DEFINE
C*****SET DEFAULT VALUE ARRAY BY READING DEFAULT INPUT FROM FILE DEFFIL
      CALL SETDEF
C*****PRINT DEFAULT VALUE INFORMATION
      IF (PRTINP) CALL WRTPAR
C*****TERMINATE EXECUTION IF ONLY VALIDATING INPUT OR ERROR ENCOUNTERED
C*****DURING READING OF INPUT DATA
      IF (NOCALC) THEN
C********PRINT MESSAGE
         WRITE(6,1010)
         STOP
      ENDIF
C*****TRANSFER DEFAULT VALUE ARRAY TO COMMON BLOCK VALUES
      CALL TRANS (FCOR(1), FCOR0(1,1), FCORL(1,1,1))
C*****ONE SET OF BIN DEFINITIONS IS USED
      READ(4,1001) BTITLE
      READ(4,*) I, NDM, NBIN
      IF (NBIN .GT. MAXBIN) THEN
C********PRINT ERROR MESSAGE
         WRITE(6,1011) NBIN, MAXBIN, NBIN
         STOP
      ENDIF
      IF (NBIN .GT. 0) READ(4,1001) (BINARR(I)(1:NDM),I=1,NBIN)
      WRITE(6,1003) BTITLE, NBIN, (I, BINARR(I)(1:NDM),I=1,NBIN)
      WRITE(6,1004)
C*****SET TOTAL NUMBER OF SOURCE TERMS
      NTOT=NBIN
C*****WRITE HEADER TO CONSEQUENCE DATA FILE
      IF (CONSFL) WRITE(9,1006) TITLE, NDM, NSPEC, NTOT, NOBS
C********
C*****WRITE STANDARD HEADER TO SPECIALIZED CONSEQUENCE DATA FILE
C********
C///      IF (CONSFL) WRITE(10,5001) TITLE, NDM, NSPEC, NTOT, NOBS
C*****LOOP OVER INDIVIDUAL BINS
      DO 1000 IBIN=1,NBIN
C********TRANSLATE CURRENT BIN ID TO PARAMETERS FOR USE IN RELCLC
         CALL BINTRN (IBIN)
C********PRINT CONTENTS OF COMMON BLOCKS
         IF (REPRTB) CALL WRREL
C********PERFORM SOURCE TERM CALCULATIONS
```

```
         CALL GGSORC (IOBS, IBIN)
 1000 CONTINUE
C*****PRINT NUMBER OF BINS PROCESSED
      WRITE(6,2003) NBIN
      RETURN
C*****FORMAT STATEMENTS
 1001 FORMAT(A)
 1002 FORMAT(1X,A)
 1003 FORMAT(//1X,130('='),
     1        //1X,'BINNING INFORMATION',
     2        /1X,A,
     3        //1X,'THE FOLLOWING ',I7,' BIN(S) ARE TO BE PROCESSED:',
     4        //(1X,I7,'-',A))
 1004 FORMAT(/1X,130('='),//)
 1006 FORMAT(1X,A,/1X,4I10)
 1010 FORMAT(/1X,'EXECUTION TERMINATED FOLLOWING VALIDATION OF INPUT')
 1011 FORMAT(/1X,'>>>>>NUMBER OF BINS (',I7,') READ FROM FILE IS ',
     1         'LARGER THAN ALLOWED DIMENSION (',I7,')',
     2        /1X,'>>>>>INCREASE PARAMETER MAXBIN TO AT LEAST ',I7,
     3        /1X,'>>>>>EXECUTION TERMINATED')
 2003 FORMAT(/1X,I7,' BIN(S) PROCESSED')
 5001 FORMAT(1X,A,//,1X,'NDM = ',I5,'     NSPEC =  ',I5,'     NTOT=  ',I5,
     1'    NOBS = ',I5,/)
      END
      SUBROUTINE BINSMP
C*****IMPLEMENTS GGSOR RUNS WHICH INVOLVE BINNED INPUT WITH SAMPLING
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      CHARACTER BINARR*(MAXBD), BTITLE*80, TITLE*80
      COMMON /BINS/ BINARR(MAXBIN), BTITLE, TITLE
      COMMON /CONTRL/ NLHS, NOBS, NSTART, NBIN, NDM, NTOT
      CHARACTER*7 NAME
      LOGICAL LDEFLT, LREAL
      COMMON /DEFLT1/ NAME(MAXVAR)
      COMMON /DEFLT2/ NVAR, NVAL, NVCB1, NVCB2, NVCB3, NVCB4,
     1                NVCB5, IDIMEN(3,MAXVAR), ISPOS(MAXVAR),
     2                ISMPPS(MAXVAL), IPNT(MAXVAR), LDEFLT(MAXVAL),
     3                LREAL(MAXVAL)
      LOGICAL NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1   EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /KEYS/ NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1   EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /SRCTRM/ ST(MAXSPC), STE(MAXSPC), STCCI(MAXSPC),
     1                STL(MAXSPC), STIL, STRVOL(MAXSPC),
     2                ST1(MAXSPC), ST2(MAXSPC), RV(MAXSPC)
      COMMON /BASVAL/ FCOR(MAXSPC), FVES(MAXSPC), DFVPA(MAXSPC),
     1                DFCPA(MAXSPC), FEVSE(MAXSPC), FDCH(MAXSPC),
     2                FCCI(MAXSPC), DFCAV(MAXSPC), VBPUF(MAXSPC),
     3                FCONV(MAXSPC), FCONC(MAXSPC), RBDF(MAXSPC),
     4                DFSPRV(MAXSPC), DFSPRC(MAXSPC), FREVO(MAXSPC),
     5                VALISS(MAXISS), FLTI1, FLTI2, NSPEC, FLV, FHPE,
     6                EVSE, WFAC, PFAC, FPLBYE, FPLBYP, FPLBYD,
     7                FPLBYC, FTLPH, FTLPL, FTLP, TC11, TC12, TB11,
     8                TB12, TB21, TB22, TBS1, TBS2, TBR1, TBR2, TW,
     9                T1, T2, DT1, DT2, DTCDB, ELEV, PUFF, HVSPLT, FCD
      COMMON /BINNED/ FCOR0(MAXSPC,MAXCAS), FVES0(MAXSPC,MAXCAS),
     1                DFVPA0(MAXSPC,MAXCAS), DFCPA0(MAXSPC,MAXCAS),
     2                FDCH0(MAXSPC,MAXCAS), FEVSE0(MAXSPC,MAXCAS),
     3                FCCI0(MAXSPC,MAXCAS), DFCAV0(MAXSPC,MAXCAS),
     4                VBPUF0(MAXSPC,MAXCAS), FCONV0(MAXSPC,MAXCAS),
     5                FCONC0(MAXSPC,MAXCAS), RBDF0(MAXSPC,MAXCAS),
     6                DFSPRV0(MAXSPC,MAXCAS), DFSPRC0(MAXSPC,MAXCAS),
     7                FREVO0(MAXSPC,MAXCAS), FLTI10(MAXCAS),
     8                FLTI20(MAXCAS), FHPE0(MAXCAS), EVSE0(MAXCAS),
     9                FPLBY0(3), HVSPLT0, TW0(MAXTIM), T10(MAXTIM),
     A                DT10(MAXTIM), DT20(MAXTIM), PUFF0(MAXTIM),
     B                E0(MAXTIM)
      COMMON /EXPERT/ FCORL(MAXSPC,MAXLEV,MAXCAS),
     1                FVESL(MAXSPC,MAXLEV,MAXCAS),
```

```
     2                  FREVOL(MAXSPC,MAXLEV,MAXCAS),
     3                  FCCIL(MAXSPC,MAXLEV,MAXCAS),
     4                  FCONVL(MAXSPC,MAXLEV,MAXCAS),
     5                  FCONCL(MAXSPC,MAXLEV,MAXCAS),
     6                  FLTI1L(MAXLEV,MAXCAS), FLTI2L(MAXLEV,MAXCAS),
     7                  RBDFL(MAXSPC,MAXLEV,MAXCAS),
     8                  FDCHL(MAXSPC,MAXLEV,MAXCAS),
     9                  FEVSEL(MAXSPC,MAXLEV,MAXCAS),
     A                  DFVPAL(MAXSPC,MAXLEV,MAXCAS),
     B                  DFCPAL(MAXSPC,MAXLEV,MAXCAS),
     C                  DFCAVL(MAXSPC,MAXLEV,MAXCAS),
     D                  DFSPRVL(MAXSPC,MAXLEV,MAXCAS),
     E                  DFSPRCL(MAXSPC,MAXLEV,MAXCAS),
     F                  PRBLEV(MAXLEV)
      COMMON /LHSBLK/ XLHS(MAXSMP)
C
C
C*****DEFINE VARIABLE NAMES AND POSITION INFORMATION FOR DEFAULT
C*****INPUT AND SAMPLE VECTOR SUBSTITUTION
      CALL DEFINE
C*****SET DEFAULT VALUE ARRAY BY READING DEFAULT INPUT FROM FILE DEFFIL
      CALL SETDEF
C*****SET SAMPLE VECTOR POSITIONS
      CALL VECPOS
C*****PRINT DEFAULT VALUE AND SAMPLE VECTOR POSITION INFORMATION
      IF (PRTINP) CALL WRTPAR
C*****TERMINATE EXECUTION IF ONLY VALIDATING INPUT OR ERROR ENCOUNTERED
C*****DURING READING OF INPUT DATA
      IF (NOCALC) THEN
C********PRINT MESSAGE
         WRITE(6,1010)
         STOP
      ENDIF
C*****READ TITLE RECORD
      READ(4,1001) BTITLE
      IF (.NOT. BYRUN) THEN
C********READ SET OF BINS WHICH WILL BE USED FOR ALL SAMPLES
         READ(4,*) I, NDM, NBIN
         IF (NBIN .GT. MAXBIN) THEN
C***********PRINT ERROR MESSAGE
            WRITE(6,1011) NBIN, MAXBIN, NBIN
            STOP
         ENDIF
         IF (NBIN .GT. 0) READ(4,1001) (BINARR(I)(1:NDM),I=1,NBIN)
         WRITE(6,1003) IOBS, BTITLE, NBIN,
     1                 (I, BINARR(I)(1:NDM),I=1,NBIN)
         WRITE(6,1004)
C********SET TOTAL NUMBER OF SOURCE TERMS
         NTOT=NBIN * NOBS
      ENDIF
C*****CHECK FOR STARTING SAMPLE VECTOR
      IF (NSTART .NE. 1) THEN
C********SKIP TO STARTING SAMPLE VECTOR
         DO 1000 ISKIP=1,NSTART-1
C***********READ SAMPLE VECTOR VALUES
            READ(3,*) I, NLHS, (XLHS(I),I=1,NLHS)
            IF (BYRUN) THEN
C**************READ BIN DEFINITIONS FOR CURRENT SAMPLE VECTOR
               READ(4,*) I, NDM, NBIN
               IF (NBIN .GT. MAXBIN) THEN
C*****************PRINT ERROR MESSAGE
                  WRITE(6,1011) NBIN, ISKIP, MAXBIN, NBIN
                  STOP
               ENDIF
               IF (NBIN .GT. 0) READ(4,1001)
     1                                (BINARR(I)(1:NDM),I=1,NBIN)
            ENDIF
 1000    CONTINUE
      ENDIF
C*****PROCESS SAMPLE VECTOR
```

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      DO 3000 IOBS=1,NOBS
C*********READ CURRENT SAMPLE VECTOR VALUES
         READ(3,*) I, NLHS, (XLHS(I),I=1,NLHS)
C*********VALIDATE NUMBER OF SAMPLE VECTOR VALUES
         IF (NLHS .GT. MAXSMP) THEN
            WRITE(6,1005) NLHS, MAXSMP
            STOP
         ENDIF
C*********TRANSFER SAMPLE VECTOR VALUES
         CALL SUBVEC
C*********TRANSFER DEFAULT VALUE ARRAY TO COMMON BLOCK VALUES
         CALL TRANS (FCOR(1), FCOR0(1,1), FCORL(1,1,1))
C*********WRITE HEADER TO CONSEQUENCE DATA FILE
         IF (CONSFL .AND. (IOBS .EQ. 1))
     1         WRITE(9,1006) TITLE, NDM, NSPEC, NTOT, NOBS
C*********WRITE STANDARD HEADER TO SPECIALIZED CONSEQUENCE DATA FILE
C///        IF (CONSFL .AND. (IOBS .EQ. 1))
C///    1        WRITE(10,5001) TITLE, NDM, NSPEC, NTOT, NOBS
C*********USE EXPERT OPINION TABLES
         IF (EXPERT) CALL EXPTAB
C*********CHECK FOR SEPARATE BIN DEFINITIONS FOR EACH SAMPLE
         IF (BYRUN) THEN
C************READ SET OF BINS FOR CURRENT SAMPLE
            READ(4,*) I, NDM, NBIN
            IF (NBIN .GT. MAXBIN) THEN
C***************PRINT ERROR MESSAGE
               WRITE(6,1011) NBIN, NSTART+IOBS-1, MAXBIN, NBIN
               STOP
            ENDIF
            IF (NBIN .GT. 0) READ(4,1001) (BINARR(I)(1:NDM),I=1,NBIN)
            WRITE(6,1003) IOBS, BTITLE, NBIN,
     1                    (I, BINARR(I)(1:NDM),I=1,NBIN)
            WRITE(6,1004)
         ENDIF
C*********LOOP OVER INDIVIDUAL BINS
         DO 2000 IBIN=1,NBIN
C************TRANSLATE CURRENT BIN ID TO PARAMETERS FOR USE IN RELCLC
            CALL BINTRN (IBIN)
C************PRINT CONTENTS OF COMMON BLOCKS
            IF (REPRTB) CALL WRREL
C************PERFORM SOURCE TERM CALCULATIONS
            CALL GGSORC (IOBS+NSTART-1, IBIN)
 2000    CONTINUE
C*********PRINT NUMBER OF BINS PROCESSED FOR CURRENT SAMPLE VECTOR
         WRITE(6,2003) NBIN, IOBS+NSTART-1
 3000 CONTINUE
      RETURN
C*****FORMAT STATEMENTS
 1001 FORMAT(A)
 1002 FORMAT(1X,A)
 1003 FORMAT(//1X,130('='),
     1       //1X,'BINNING INFORMATION FOR SAMPLE VECTOR ',I4,
     2       /1X,A,
     3       //1X,'THE FOLLOWING ',I7,' BIN(S) ARE TO BE PROCESSED:',
     4       //(1X,I7,'-',A))
 1004 FORMAT(/1X,130('='),//)
 1005 FORMAT(/1X,'>>>>>NUMBER OF SAMPLE VECTOR VALUES (',I4,
     1        ') READ FROM UNIT 3 EXCEEDS ',
     2       /1X,'>>>>>MAXIMUM NUMBER ALLOWED (MAXSMP=',I4 ')',
     3       /1X,'>>>>>EXECUTION TERMINATED')
 1006 FORMAT(1X,A,/1X,4I10)
 1010 FORMAT(/1X,'EXECUTION TERMINATED FOLLOWING VALIDATION OF INPUT')
 1011 FORMAT(/1X,'>>>>>NUMBER OF BINS (',I7,') READ FROM UNIT 4, ',
     1        'SAMPLE VECTOR ',I4,
     2        ', IS LARGER THAN ALLOWED DIMENSION (',I7,')',
     3       /1X,'>>>>>INCREASE PARAMETER MAXBIN TO AT LEAST ',I7,
     4       /1X,'>>>>>EXECUTION TERMINATED')
 2003 FORMAT(/1X,I7,' BIN(S) PROCESSED FOR SAMPLE VECTOR ',I4)
 5001 FORMAT(1X,A,//,1X,'NDM = ',I5,'    NSPEC =  ',I5,'    NTOT=  ',I5,
     1'    NOBS = ',I5,/)
```

```
      END
      SUBROUTINE DEFINE
C*****DEFINE NAMES AND DIMENSIONS OF VARIABLES TO BE SET THROUGH
C*****DEFAULT INPUT AND SAMPLE VECTOR SUBSTITUTION
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      CHARACTER*7 NAME
      LOGICAL LDEFLT, LREAL
      COMMON /DEFLT1/ NAME(MAXVAR)
      COMMON /DEFLT2/ NVAR, NVAL, NVCB1, NVCB2, NVCB3, NVCB4,
     1                NVCB5, IDIMEN(3,MAXVAR), ISPOS(MAXVAR),
     2                ISMPPS(MAXVAL), IPNT(MAXVAR), LDEFLT(MAXVAL),
     3                LREAL(MAXVAL)
      LOGICAL NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1   EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /KEYS/ NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1   EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
C
C
C*****SET DEFAULT TYPES TO .FALSE.
      DO 1000 I=1,MAXVAL
         LDEFLT(I)=.FALSE.
 1000 CONTINUE
C*****INITIALLIZE NUMBER OF VARIABLES
      NVAR=0
C*****INITIALLIZE TOTAL NUMBER OF VALUES
      NVAL=1
C*****SET POSITION INFORMATION
      DO 2000 IVAR=1,MAXVAR
C********CHECK FOR BLANK VARIABLE NAME
         IF (NAME(IVAR) .EQ. ' ') GO TO 3000
C********INCREMENT NUMBER OF VARIABLES
         NVAR=NVAR + 1
C********SAVE STARTING POSITION OF CURRENT VARIABLE
         ISPOS(NVAR)=NVAL
C********CHECK FOR NON-ZERO DIMENSIONS
         IF ((IDIMEN(1,NVAR) .LE. 0) .OR. (IDIMEN(2,NVAR) .LE. 0) .OR.
     1       (IDIMEN(3,NVAR) .LE. 0)) THEN
C***********PRINT ERROR MESSAGE
            WRITE(6,1001) NAME(IVAR), (IDIMEN(I,NVAR),I=1,3)
            NOCALC=.TRUE.
         ENDIF
C********INCREMENT TOTAL NUMBER OF VALUES
         NVAL=NVAL + IDIMEN(1,NVAR)*IDIMEN(2,NVAR)*IDIMEN(3,NVAR)
 2000 CONTINUE
 3000 CONTINUE
C*****SET TOTAL NUMBER OF VALUES
      NVAL=NVAL - 1
C*****VALIDATE TOTAL NUMBER OF VALUES AGAINST MAXIMUM DIMENSION
      IF (NVAL .GT. MAXVAL) THEN
C********PRINT ERROR MESSAGE
         WRITE(6,3001) NVAL, MAXVAL, NVAL
         NOCALC=.TRUE.
      ENDIF
C*****VALIDATE TOTAL NUMBER OF VALUES AGAINST SUM OF VALUES IN
C*****DEFAULTED COMMON BLOCKS
      IF (NVAL .NE. NVCB1+NVCB2+NVCB3+NVCB4+NVCB5) THEN
C********PRINT ERROR MESSAGE
         WRITE(6,3002) NVAL, NVCB1+NVCB2+NVCB3+NVCB4+NVCB5
         NOCALC=.TRUE.
      ENDIF
C*****SET ASCII CODE FOR CHARACTERS I AND N
      ICI=ICHAR('I')
      ICN=ICHAR('N')
C*****INITIALLIZE VALUE POINTER
      IVAL=1
C*****SET VARIABLE TYPES
      DO 5000 IVAR=1,NVAR
C********INITIALLIZE POINTER ARRAY
```

```
         IPNT(IVAR)=IVAR
C*********SET ASCII CODE FOR FIRST CHARACTER OF VARIABLE NAME
         IC=ICHAR(NAME(IVAR)(1:1))
C*********COMPARE ASCII CODE TO I THRU N RANGE (INTEGER VARIABLE)
         IF ((IC .GE. ICI) .AND. (IC .LE. ICN)) THEN
C************SET REAL VARIABLE FLAG TO .FALSE. (INTEGER VARIABLE)
            LREAL(IVAL)=.FALSE.
         ELSE
C************SET REAL VARIABLE FLAG TO .TRUE. (REAL VARIABLE)
            LREAL(IVAL)=.TRUE.
         ENDIF
C*********SET ALL TYPES FOR CURRENT VARIABLE
         IFIRST=IVAL
         ILAST=IVAL - 1 + IDIMEN(1,IVAR)*IDIMEN(2,IVAR)*IDIMEN(3,IVAR)
         DO 4000 I=IFIRST,ILAST
            LREAL(I)=LREAL(IVAL)
 4000    CONTINUE
C*********RESET VALUE POINTER
         IVAL=ILAST + 1
 5000 CONTINUE
C*****ALL INFORMATION FOR VARIABLES THAT MAY BE SET THROUGH DEFAULT
C*****AND VECTOR POSITION HAS BEEN SAVED
C*****SORT LIST OF VARIABLE NAMES USING POINTER IPNT TO FACILITATE
C*****SEARCHING
      CALL CSORT (NVAR, NAME, IPNT)
      RETURN
C*****FORMAT STATEMENTS
 1001 FORMAT(1X,'>>>>>DIMENSIONS FOR VARIABLE ',A,' MUST BE GREATER ',
     1        'THAN 0',
     2     /1X,'>>>>>DIMENSION 1 =',I5,', DIMENSION 2 =',I5,
     3      ', DIMENSION 3 =',I5,
     4     /1X,'>>>>>CHECK VARIABLE DEFINITIONS IN SUBROUTINE DEFINE',/)
 3001 FORMAT(1X,'>>>>>NUMBER OF VARIABLES (NVAR=',I5,') EXCEEDS ',
     1        'DIMENSION (MAXVAR=',I5,')',
     2      /1X,'>>>>>CHECK VARIABLE DEFINITIONS IN SUBROUTINE DEFINE ',
     3        'AND/OR',
     4     /1X,'>>>>>RESET PARAMETER MAXVAR TO AT LEAST ',I5,/)
 3002 FORMAT(1X,'>>>>>NUMBER OF VALUES WHICH CAN BE SET THROUGH ',
     1        'DEFAULT AND VECTOR SUBSTITUTION (',I5,')',
     2      /1X,'>>>>>SHOULD BE EQUAL TO THE TOTAL NUMBER OF VALUES ',
     3       'IN THE COMMON BLOCKS TO BE SET ',
     4      /1X,'>>>>>(NVCB1+NVCB2+NVCB3+NVCB4+NVCB5=',I5,') ',
     5       '-- SUBROUTINE DEFINE',/)
      END
      SUBROUTINE SETDEF
C*****SET DEFAULT VALUES BY READING VARIABLE NAMES AND CORRESPONDING
C*****VALUES FROM FILE DESIGNATED FOR DEFAULT VALUES (DEFFIL)
      PARAMETER (MAXLEN=101, MAXVLN=20)
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      CHARACTER*7 NAME
      LOGICAL LDEFLT, LREAL
      COMMON /DEFLT1/ NAME(MAXVAR)
      COMMON /DEFLT2/ NVAR, NVAL, NVCB1, NVCB2, NVCB3, NVCB4,
     1                NVCB5, IDIMEN(3,MAXVAR), ISPOS(MAXVAR),
     2                ISMPPS(MAXVAL), IPNT(MAXVAR), LDEFLT(MAXVAL),
     3                LREAL(MAXVAL)
      LOGICAL NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1   EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /KEYS/ NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1   EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      CHARACTER*80 DEFFIL, SAMFIL, VECFIL
      COMMON /FILBLK/ DEFFIL, SAMFIL, VECFIL
      CHARACTER*(MAXLEN) CARD
      CHARACTER*(MAXVLN) CVAL, TMPVAL
      DIMENSION INDX(3)
      LOGICAL EOR, TYPE(4), LVAL
      CHARACTER*10 IFRMT
      COMMON /VALUES/ RVL(MAXVAL)
```

```
      DIMENSION IVL(MAXVAL)
      EQUIVALENCE (IVL, RVL)
C
C
C*****PRINT HEADER MESSAGE
      IF (PRTINP) WRITE(6,1003) DEFFIL
C*****WRITE INTEGER FORMAT FOR READING VARIABLE ARRAY INDICES
      WRITE(IFRMT,1004) MAXVLN
C*****INITIALLIZE CURRENT VALUE POSITION
      IVPOS=-1
C*****OPEN DEFAULT FILE
      OPEN(1, FILE=DEFFIL, STATUS='OLD', ERR=9100)
 1000 CONTINUE
C*****READ RECORD
      READ(1,1001,END=8000) CARD
C*****PRINT RECORD
      IF (PRTINP) WRITE(6,1002) CARD
C*****INITIALLIZE COLUMN POINTER FOR CURRENT RECORD
      IC=1
 2000 CONTINUE
C*****READ NEXT VALUE ON RECORD
      CALL RDSTRG (CARD, IC, CVAL, LVAL, IVAL, RVAL, LENGTH, TYPE, EOR)
C*****CHECK FOR END-OF-RECORD
      IF (EOR) GO TO 1000
C*****CHECK FOR CHARACTER VALUE (VARIABLE NAME)
      IF (TYPE(1)) THEN
C********INITIALLIZE ARRAY SPECIFICATIONS
         INDX(1)=1
         INDX(2)=1
         INDX(3)=1
C********CHARACTER VALUE (VARIABLE NAME) FOUND
C********CHECK FOR LEFT PARENTHESIS IN VARIABLE NAME
         ILPAR=INDEX(CVAL, '(')
         IF (ILPAR .NE. 0) THEN
C***********FOUND LEFT PARENTHESIS, CHECK FOR RIGHT PARENTHESIS
            IRPAR=INDEX(CVAL, ')')
            IF (IRPAR .NE. 0) THEN
C**************FOUND RIGHT PARENTHESIS, CHECK FOR COMMA
               ICOMMA=INDEX(CVAL, ',')
               IF (ICOMMA .NE. 0) THEN
C*****************FOUND COMMA, DETERMINE FIRST ARRAY INDEX
                  IS=ILPAR + 1
                  IE=ICOMMA
                  IND=1
 3000             CONTINUE
                  IE=IE - 1
                  IF (CVAL(IE:IE) .EQ. ' ') GO TO 3000
                  IF (IE .GE. IS) THEN
C********************SET CURRENT ARRAY INDEX
                     TMPVAL=' '
                     TMPVAL(MAXVLN+IS-IE:MAXVLN)=CVAL(IS:IE)
                     READ(TMPVAL,IFRMT,ERR=9200) INDX(IND)
                  ELSE
C********************INVALID ARRAY INDEX
                     WRITE(6,3001) IND, CVAL(1:LENGTH)
                     NOCALC=.TRUE.
                  ENDIF
C*****************CHECK FOR FINAL ARRAY INDEX FOUND
                  IF (ICOMMA .GT. 0) THEN
                     IS=ICOMMA + 1
C********************LOCATE NEXT COMMA
                     ICOMMA=INDEX(CVAL(IS:LENGTH), ',')
                     IF (ICOMMA .GT. 0) THEN
                        ICOMMA=IS + ICOMMA - 1
                        IE=ICOMMA
                     ELSE
                        IE=IRPAR
                     ENDIF
C********************INCREMENT COUNTER FOR CURRENT ARRAY INDEX
                     IND=IND + 1
```

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                  IF (IND .GT. 3) THEN
C***********************MORE THAN 3 ARRAY INDICES
                     WRITE(6,3002) CVAL(1:LENGTH)
                     IVPOS=-1
                     NOCALC=.TRUE.
                     GO TO 2000
                  ENDIF
                  GO TO 3000
               ENDIF
            ELSE
C******************NO COMMA FOUND, ONLY ONE ARRAY INDEX
               IS=ILPAR + 1
               IE=IRPAR
               IND=1
 4000          CONTINUE
               IE=IE - 1
               IF (CVAL(IE:IE) .EQ. ' ') GO TO 4000
               IF (IE .GE. IS) THEN
C*********************SET SINGLE ARRAY INDEX
                  TMPVAL=' '
                  TMPVAL(MAXVLN+IS-IE:MAXVLN)=CVAL(IS:IE)
                  READ(TMPVAL,IFRMT,ERR=9200) INDX(IND)
               ELSE
C*********************INVALID ARRAY INDEX
                  WRITE(6,4001) CVAL(1:LENGTH)
                  NOCALC=.TRUE.
               ENDIF
            ENDIF
C***************BLANK OUT ARRAY INDEX PORTION OF VARIABLE NAME
            CVAL(ILPAR:MAXVLN)=' '
         ELSE
C***************ERROR IN ARRAY INDEX SPECIFICATION
            WRITE(6,4001) CVAL(1:LENGTH)
            NOCALC=.TRUE.
         ENDIF
      ENDIF
C*********SEARCH FOR VARIABLE NAME
      CALL SEARCH (NVAR, CVAL(1:LENGTH), NAME, IPNT, IPOINT)
C*********CHECK FOR VARIABLE NAME FOUND
      IF (IPOINT .GT. 0) THEN
C***********VARIABLE NAME FOUND, VALIDATE ARRAY INDICES
         IF ((INDX(1) .GE. 1) .AND. (INDX(2) .GE. 1) .AND.
     1       (INDX(3) .GE. 1) .AND.
     2       (INDX(1) .LE. IDIMEN(1,IPNT(IPOINT))) .AND.
     3       (INDX(2) .LE. IDIMEN(2,IPNT(IPOINT))) .AND.
     4       (INDX(3) .LE. IDIMEN(3,IPNT(IPOINT)))) THEN
C***************VALID ARRAY INDICES, SET CURRENT VALUE POSITION
            IVPOS=ISPOS(IPNT(IPOINT)) + INDX(1) +
     1             (INDX(2)-1)*IDIMEN(1,IPNT(IPOINT)) +
     2             (INDX(3)-1)*IDIMEN(1,IPNT(IPOINT))*
     3              IDIMEN(2,IPNT(IPOINT)) - 1
         ELSE
C***************INVALID ARRAY INDICES
            WRITE(6,4002) NAME(IPNT(IPOINT)),
     1                    (I, INDX(I), IDIMEN(I,IPNT(IPOINT)),I=1,3)
C***************SET INVALID VALUE POSITION
            IVPOS=-1
            NOCALC=.TRUE.
         ENDIF
      ELSE
C************VARIABLE NAME NOT FOUND
         WRITE(6,4003) CVAL(1:LENGTH)
C************SET INVALID VALUE POSITION
         IVPOS=-1
         NOCALC=.TRUE.
      ENDIF
   ELSE IF (TYPE(3)) THEN
C*********CHECK FOR VALID VALUE POSITION
      IF (IVPOS .GE. 0) THEN
C***********SET STARTING POSITION OF VARIABLE FOLLOWING CURRENT VARIABLE
```

```
            IP=IPNT(IPOINT)
            IPNXTV=ISPOS(IP) + IDIMEN(1,IP)*IDIMEN(2,IP)*IDIMEN(3,IP)
C***********CHECK FOR VALID ARRAY POSITION
            IF (IVPOS .LT. IPNXTV) THEN
C**************INTEGER VALUE FOUND SO CHECK FOR INTEGER VARIABLE TYPE
               IF (.NOT. LREAL(IVPOS)) THEN
C*****************INTEGER VARIABLE SO TRANSFER INTEGER VALUE
                  IVL(IVPOS)=IVAL
C*****************SET DEFAULT FLAG
                  LDEFLT(IVPOS)=.TRUE.
               ELSE
C*****************REAL VARIABLE, PRINT ERROR MESSAGE
                  WRITE(6,4004) NAME(IPNT(IPOINT)), IVAL
                  NOCALC=.TRUE.
               ENDIF
            ELSE
C**************INVALID ARRAY POSITION, PRINT ERROR MESSAGE
               WRITE(6,4006) NAME(IPNT(IPOINT)),
     1                       (IDIMEN(I,IPNT(IPOINT)),I=1,3)
               NOCALC=.TRUE.
            ENDIF
C***********INCREMENT VALUE POSITION
            IVPOS=IVPOS + 1
         ENDIF
      ELSE IF (TYPE(4)) THEN
C********CHECK FOR VALID VALUE POSITION
         IF (IVPOS .GE. 0) THEN
C***********SET STARTING POSITION OF VARIABLE FOLLOWING CURRENT VARIABLE
            IP=IPNT(IPOINT)
            IPNXTV=ISPOS(IP) + IDIMEN(1,IP)*IDIMEN(2,IP)*IDIMEN(3,IP)
C***********CHECK FOR VALID ARRAY POSITION
            IF (IVPOS .LT. IPNXTV) THEN
C**************REAL VALUE FOUND SO CHECK FOR REAL VARIABLE TYPE
               IF (LREAL(IVPOS)) THEN
C*****************REAL VARIABLE SO TRANSFER REAL VALUE
                  RVL(IVPOS)=RVAL
C*****************SET DEFAULT FLAG
                  LDEFLT(IVPOS)=.TRUE.
               ELSE
C*****************INTEGER VARIABLE, PRINT ERROR MESSAGE
                  WRITE(6,4005) NAME(IPNT(IPOINT)), RVAL
                  NOCALC=.TRUE.
               ENDIF
            ELSE
C**************INVALID ARRAY POSITION, PRINT ERROR MESSAGE
               WRITE(6,4006) NAME(IPNT(IPOINT)),
     1                       (IDIMEN(I,IPNT(IPOINT)),I=1,3)
               NOCALC=.TRUE.
            ENDIF
C***********INCREMENT VALUE POSITION
            IVPOS=IVPOS + 1
         ENDIF
      ENDIF
      GO TO 2000
 8000 CONTINUE
C*****CLOSE DEFAULT FILE
      CLOSE (1)
      RETURN
 9100 CONTINUE
C*****ERROR IN OPENING DEFAULT FILE
      WRITE(6,9101) DEFFIL
      STOP
 9200 CONTINUE
C*****ERROR IN READING ARRAY INDEX
      WRITE(6,9201)
      NOCALC=.TRUE.
      GO TO 1000
C*****FORMAT STATEMENTS
 1001 FORMAT(A)
 1002 FORMAT(11X,A)
```

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```
 1003 FORMAT('1',/1X,130('*'),
     1       /1X,53('*'),5X,'DEFAULT INPUT',5X,54('*'),
     2       /1X,17('*'),5X,'FILE =',A,5X,17('*'),
     3       /1X,130('*'),/)
 1004 FORMAT('(I',I2,')')
 3001 FORMAT(1X,'>>>>>DEFAULT VARIABLE ARRAY INDEX ',I1,' FOR ',
     1       'VARIABLE ',A,' IS INVALID',/)
 3002 FORMAT(1X,'>>>>>MORE THAN 3 ARRAY INDICES GIVEN FOR VARIABLE ',A,
     1       'ON DEFAULT FILE')
 4001 FORMAT(1X,'>>>>>DEFAULT VARIABLE ARRAY INDEX FOR ',
     1       'VARIABLE ',A,' IS INVALID',/)
 4002 FORMAT(1X,'>>>>>ARRAY INDICES FOR DEFAULT VARIABLE ',A,
     1       ' ARE OUT OF RANGE:'
     2   /(1X,'>>>>>  INDEX',I2,' =',I5,', VALID RANGE = 1 TO ',I5,/:))
 4003 FORMAT(1X,'>>>>>DEFAULT VARIABLE NAME ',A,
     1       ' NOT FOUND IN DEFAULT VARIABLE LIST',/)
 4004 FORMAT(1X,'>>>>>ATTEMPT TO DEFAULT REAL VARIABLE (',A,
     1       ') TO INTEGER VALUE (',I10,')',/)
 4005 FORMAT(1X,'>>>>>ATTEMPT TO DEFAULT INTEGER VARIABLE (',A,
     1       ') TO REAL VALUE (',1PE10.3,')',/)
 4006 FORMAT(1X,'>>>>>INVALID ARRAY POSITION ENCOUNTERED WHILE ',
     1       'SETTING DEFAULT VALUES FOR VARIABLE--',
     2       A,'(',I2,',',I2,',',I2,')')
 8001 FORMAT('1')
 9101 FORMAT(1X,'>>>>>ERROR OPENING DEFAULT FILE ',A,/)
 9201 FORMAT(1X,'>>>>>ERROR IN READING PREVIOUS ARRAY INDEX')
      END
      SUBROUTINE VECPOS
C*****SET SAMPLE VECTOR POSITIONS BY READING VARIABLE NAMES AND
C*****CORRESPONDING SAMPLE VECTOR POSITIONS FROM FILE DESIGNATED
C*****FOR SAMPLE VECTOR POSITIONS (SAMFIL)
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      PARAMETER (MAXLEN=101, MAXVLN=20)
      CHARACTER*7 NAME
      LOGICAL LDEFLT, LREAL
      COMMON /DEFLT1/ NAME(MAXVAR)
      COMMON /DEFLT2/ NVAR, NVAL, NVCB1, NVCB2, NVCB3, NVCB4,
     1                NVCB5, IDIMEN(3,MAXVAR), ISPOS(MAXVAR),
     2                ISMPPS(MAXVAL), IPNT(MAXVAR), LDEFLT(MAXVAL),
     3                LREAL(MAXVAL)
      LOGICAL NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1    EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /KEYS/ NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1    EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      CHARACTER*80 DEFFIL, SAMFIL, VECFIL
      COMMON /FILBLK/ DEFFIL, SAMFIL, VECFIL
      CHARACTER*(MAXLEN) CARD
      CHARACTER*(MAXVLN) CVAL, TMPVAL
      CHARACTER*10 IFRMT
      DIMENSION INDX(3)
      LOGICAL EOR, TYPE(4), LVAL
C
C
C*****PRINT HEADER MESSAGE
      IF (PRTINP) WRITE(6,1003) SAMFIL
C*****WRITE INTEGER FORMAT FOR READING VARIABLE ARRAY INDICES
      WRITE(IFRMT,1004) MAXVLN
C*****INITIALLIZE CURRENT VALUE POSITION
      IVPOS=-1
C*****OPEN SAMPLE VECTOR POSITION FILE
      OPEN(1, FILE=SAMFIL, STATUS='OLD', ERR=9100)
 1000 CONTINUE
C*****READ RECORD
      READ(1,1001,END=9000) CARD
C*****PRINT RECORD
      IF (PRTINP) WRITE(6,1002) CARD
C*****INITIALLIZE COLUMN POINTER FOR CURRENT RECORD
      IC=1
```

```
 2000 CONTINUE
C*****READ NEXT VALUE ON RECORD
      CALL RDSTRG (CARD, IC, CVAL, LVAL, IVAL, RVAL, LENGTH, TYPE, EOR)
C*****CHECK FOR END-OF-RECORD
      IF (EOR) GO TO 1000
C*****CHECK FOR CHARACTER VALUE (VARIABLE NAME)
      IF (TYPE(1)) THEN
C********INITIALLIZE ARRAY SPECIFICATIONS
         INDX(1)=1
         INDX(2)=1
         INDX(3)=1
C********CHARACTER VALUE (VARIABLE NAME) FOUND
C********CHECK FOR LEFT PARENTHESIS IN VARIABLE NAME
         ILPAR=INDEX(CVAL, '(')
         IF (ILPAR .NE. 0) THEN
C***********FOUND LEFT PARENTHESIS, CHECK FOR RIGHT PARENTHESIS
            IRPAR=INDEX(CVAL, ')')
            IF (IRPAR .NE. 0) THEN
C**************FOUND RIGHT PARENTHESIS, CHECK FOR COMMA
               ICOMMA=INDEX(CVAL, ',')
               IF (ICOMMA .NE. 0) THEN
C*****************FOUND COMMA, DETERMINE FIRST ARRAY INDEX
                  IS=ILPAR + 1
                  IE=ICOMMA
                  IND=1
 3000             CONTINUE
                  IE=IE - 1
                  IF (CVAL(IE:IE) .EQ. ' ') GO TO 3000
                  IF (IE .GE. IS) THEN
C********************SET CURRENT ARRAY INDEX
                     TMPVAL=' '
                     TMPVAL(MAXVLN+IS-IE:MAXVLN)=CVAL(IS:IE)
                     READ(TMPVAL,IFRMT,ERR=9200) INDX(IND)
                  ELSE
C********************INVALID ARRAY INDEX
                     WRITE(6,3001) IND, CVAL(1:LENGTH)
                     NOCALC=.TRUE.
                  ENDIF
C*****************CHECK FOR FINAL ARRAY INDEX FOUND
                  IF (ICOMMA .GT. 0) THEN
                     IS=ICOMMA + 1
C********************LOCATE NEXT COMMA
                     ICOMMA=INDEX(CVAL(IS:LENGTH), ',')
                     IF (ICOMMA .GT. 0) THEN
                        ICOMMA=IS + ICOMMA - 1
                        IE=ICOMMA
                     ELSE
                        IE=IRPAR
                     ENDIF
C********************INCREMENT COUNTER FOR CURRENT ARRAY INDEX
                     IND=IND + 1
                     IF (IND .GT. 3) THEN
C***********************MORE THAN 3 ARRAY INDICES
                        WRITE(6,3002) CVAL(1:LENGTH)
                        IVPOS=-1
                        NOCALC=.TRUE.
                        GO TO 2000
                     ENDIF
                     GO TO 3000
                  ENDIF
               ELSE
C*****************NO COMMA FOUND, ONLY ONE ARRAY INDEX
                  IS=ILPAR + 1
                  IE=IRPAR
                  IND=1
 4000             CONTINUE
                  IE=IE - 1
                  IF (C AL(IE:IE) .EQ. ' ') GO TO 4000
                  IF (IE .GE. IS) THEN
C********************SET SINGLE ARRAY INDEX
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                     TMPVAL=' '
                     TMPVAL(MAXVLN+IS-IE:MAXVLN)=CVAL(IS:IE)
                     READ(TMPVAL,IFRMT,ERR=9200) INDX(IND)
                  ELSE
C*******************INVALID ARRAY INDEX
                     WRITE(6,4001) CVAL(1:LENGTH)
                     NOCALC=.TRUE.
                  ENDIF
               ENDIF
C***************BLANK OUT ARRAY INDEX PORTION OF VARIABLE NAME
               CVAL(ILPAR:MAXVLN)=' '
            ELSE
C***************ERROR IN ARRAY INDEX SPECIFICATION
               WRITE(6,4001) CVAL(1:LENGTH)
               NOCALC=.TRUE.
            ENDIF
         ENDIF
C*********SEARCH FOR VARIABLE NAME
         CALL SEARCH (NVAR, CVAL(1:LENGTH), NAME, IPNT, IPOINT)
C*********CHECK FOR VARIABLE NAME FOUND
         IF (IPOINT .GT. 0) THEN
C************VARIABLE NAME FOUND, VALIDATE ARRAY INDICES
            IF ((INDX(1) .GE. 1) .AND. (INDX(2) .GE. 1) .AND.
     1          (INDX(2) .GE. 1) .AND.
     2          (INDX(1) .LE. IDIMEN(1,IPNT(IPOINT))) .AND.
     3          (INDX(2) .LE. IDIMEN(2,IPNT(IPOINT))) .AND.
     4          (INDX(3) .LE. IDIMEN(3,IPNT(IPOINT)))) THEN
C***************VALID ARRAY INDICES, SET CURRENT VALUE POSITION
               IVPOS=ISPOS(IPNT(IPOINT)) + INDX(1) +
     1               (INDX(2)-1)*IDIMEN(1,IPNT(IPOINT)) +
     2               (INDX(3)-1)*IDIMEN(1,IPNT(IPOINT))*
     3                IDIMEN(2,IPNT(IPOINT)) - 1
            ELSE
C***************INVALID ARRAY INDICES
               WRITE(6,4002) CVAL(1:LENGTH),
     1                       (I, INDX(I), IDIMEN(I,IPNT(IPOINT)),I=1,3)
C***************SET INVALID VALUE POSITION
               IVPOS=-1
               NOCALC=.TRUE.
            ENDIF
         ELSE
C************VARIABLE NAME NOT FOUND
            WRITE(6,4003) CVAL(1:LENGTH)
            NOCALC=.TRUE.
         ENDIF
      ELSE IF (TYPE(3)) THEN
C*********CHECK FOR VALID VALUE POSITION
         IF (IVPOS .GE. 0) THEN
C************SET STARTING POSITION OF VARIABLE FOLLOWING CURRENT VARIABLE
            IP=IPNT(IPOINT)
            IPNXTV=ISPOS(IP) + IDIMEN(1,IP)*IDIMEN(2,IP)*IDIMEN(3,IP)
C************CHECK FOR VALID ARRAY POSITION
            IF (IVPOS .LT. IPNXTV) THEN
C***************TRANSFER INTEGER VALUE TO SAMPLE VECTOR POSITION
               ISMPPS(IVPOS)=IVAL
            ELSE
C***************INVALID ARRAY POSITION, PRINT ERROR MESSAGE
               WRITE(6,4006) NAME(IPNT(IPOINT)),
     1                       (IDIMEN(I,IPNT(IPOINT)),I=1,2)
               NOCALC=.TRUE.
            ENDIF
C************INCREMENT VALUE POSITION
            IVPOS=IVPOS + 1
         ENDIF
      ELSE IF (TYPE(4)) THEN
C*********REAL VALUE TYPE, INVALID FOR SAMPLE VECTOR POSITION
         WRITE(6,4004) RVAL, NAME(IPNT(IPOINT))
         NOCALC=.TRUE.
      ENDIF
      GO TO 2000
```

```
 9000 CONTINUE
C*****CLOSE SAMPLE VECTOR POSITION FILE
      CLOSE (1)
      RETURN
 9100 CONTINUE
C*****ERROR IN OPENING SAMPLE VECTOR POSITION FILE
      WRITE(6,9101) SAMFIL
      STOP
 9200 CONTINUE
C*****ERROR IN READING ARRAY INDEX
      WRITE(6,9201)
      NOCALC=.TRUE.
      GO TO 1000
C*****FORMAT STATEMENTS
 1001 FORMAT(A)
 1002 FORMAT(11X,A)
 1003 FORMAT('1',/1X,130('*'),
     1       /1X,46('*'),5X,'SAMPLE VECTOR POSITION INPUT',5X,46('*'),
     2       /1X,17('*'),5X,'FILE =',A,5X,17('*'),
     3       /1X,130('*'),/)
 1004 FORMAT('(I',I2,')')
 3001 FORMAT(1X,'>>>>>SAMPLE VECTOR POSITION VARIABLE ARRAY INDEX ',
     1         I1,' FOR VARIABLE ',A,' IS INVALID',/)
 3002 FORMAT(1X,'>>>>>MORE THAN 3 ARRAY INDICES GIVEN FOR VARIABLE ',A,
     1         'ON SAMPLE VECTOR POSITION FILE')
 4001 FORMAT(1X,'>>>>>SAMPLE VECTOR POSITION VARIABLE ARRAY INDEX ',
     1         'FOR VARIABLE ',A,' IS INVALID',/)
 4002 FORMAT(1X,'>>>>>ARRAY INDICES FOR SAMPLE VECTOR POSITION ',
     1         'VARIABLE ',A,' ARE OUT OF RANGE:'
     2    /(1X,'>>>>>  INDEX',I2,' =',I5,', VALID RANGE = 1 TO ',I5,/:))
 4003 FORMAT(1X,'>>>>>SAMPLE VECTOR POSITION VARIABLE NAME ',A,
     1          ' NOT FOUND IN DEFAULT VARIABLE LIST',/)
 4004 FORMAT(1X,'>>>>>ATTEMPT TO USE REAL VALUE (',1PE10.2,') TO ',
     1         'SPECIFY SAMPLE VECTOR POSITION FOR VARIABLE ',A,/)
 4006 FORMAT(1X,'>>>>>INVALID ARRAY POSITION ENCOUNTERED WHILE ',
     1         'SETTING DEFAULT VALUES FOR VARIABLE--',
     2         A,'(',I2,',',I2,',',I2,')')
 9101 FORMAT(1X,'>>>>>ERROR OPENING SAMPLE VECTOR POSITION FILE ',A,/)
 9201 FORMAT(1X,'>>>>>ERROR IN READING PREVIOUS ARRAY INDEX')
      END
      SUBROUTINE RDSTRG (CARD, IC, CVAL, LVAL, IVAL, RVAL,
     1                   LENGTH, TYPE, EOR)
C*****CONVERTS A RECORD STRING TO A CHARACTER VALUE, A LOGICAL VALUE,
C*****A REAL VALUE, AND AN INTEGER VALUE
      PARAMETER (IL=100)
      CHARACTER*(*) CARD, CVAL
      CHARACTER*(IL) TMPCRD
      CHARACTER*8 RFRMT
      LOGICAL EOR, FIRST, LVAL, TYPE(4)
      DATA FIRST / .TRUE. /
C
C
C*****CHECK FOR FIRST TIME INTO ROUTINE
      IF (FIRST) THEN
C*********WRITE INTEGER AND REAL FORMATS
          WRITE(RFRMT,1002) IL
C*********RESET INITIALLIZATION TYPE
          FIRST=.FALSE.
      ENDIF
      LVAL=.FALSE.
C*****SET LENGTH OF INCOMING RECORD
      ILMAX=LEN(CARD)
C*****SET LENGTH OF CHARACTER VARIABLE
      LENCVAL=LEN(CVAL)
C*****INITIALLIZE VARIABLE FLAG TYPES (1=CHAR, 2=LOGIC, 3=INTEG, 4=REAL)
      DO 1000 I=1,4
          TYPE(I)=.FALSE.
 1000 CONTINUE
C*****INTIALLIZE END-OF-RECORD TYPE
      EOR=.FALSE.
```

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C*****RESET STARTING POSITION FOR CHARACTER POINTER
      IC=IC - 1
C*****SEARCH FOR FIRST NON-BLANK CHARACTER
 2000 CONTINUE
C*****INCREMENT CHARACTER POINTER
      IC=IC + 1
C*****CHECK FOR END OF RECORD
      IF (IC .GT. ILMAX) GO TO 9100
C*****CHECK FOR BLANK CHARACTER (STRING DELIMITER)
      IF (CARD(IC:IC) .EQ. ' ') GO TO 2000
C*****CHECK FOR BEGINNING OF COMMENT
      IF (CARD(IC:IC) .EQ. '$') GO TO 9100
C*****CHECK FOR COMMA CHARACTER (STRING DELIMITER)
      IF (CARD(IC:IC) .EQ. ',') GO TO 2000
C*****CHECK FOR QUOTE CHARACTER (CHARACTER STRING DELIMITER)
      IF (CARD(IC:IC) .EQ. '''') THEN
C********SAVE STARTING POSITION OF CHARACTER STRING
         IS=IC + 1
C********SEARCH FOR ANOTHER QUOTE
         IC=INDEX(CARD(IS:ILMAX), '''')
         IF (IC .EQ. 0) THEN
C***********QUOTE NOT FOUND SO CONTINUE SEARCH FOR BLANK TO TERMINATE
C***********CHARACTER STRING
            IC=IS - 1
         ELSE
C***********QUOTE FOUND
            IC=IS + IC - 1
            GO TO 3100
         ENDIF
C********SEARCH FOR END OF CHARACTER STRING (' SIGNIFIES BEGINNING AND
C********END OF CHARACTER STRING)
 3000    CONTINUE
C********INCREMENT CHARACTER POINTER
         IC=IC + 1
C********CHECK FOR END OF RECORD
         IF (IC .GT. ILMAX) GO TO 9100
C********CHECK FOR BEGINNING OF COMMENT
         IF (CARD(IC:IC) .EQ. '$') GO TO 3100
C********CHECK FOR BLANK TO TERMINATE CHARACTER STRING
         IF (CARD(IC:IC) .NE. ' ') GO TO 3000
 3100    CONTINUE
C********END OF CHARACTER STRING FOUND
C********COMPARE STRING LENGTH TO CHARACTER VARIABLE LENGTH
         IE=IC - 1
         IF (IE-IS+1 .GT. LENCVAL) GO TO 9300
C********TRANSFER CHARACTER STRING
         CVAL=CARD(IS:IE)
C********SET LENGTH OF CHARACTER STRING
         LENGTH=IE - IS + 1
C********SET VARIABLE FLAG TYPE FOR CHARACTER VARIABLE FOUND
         TYPE(1)=.TRUE.
      ELSE
C********SAVE STARTING POSITION FOR STRING
         IS=IC
C********SEARCH FOR END OF STRING (BLANK OR , SIGNIFY END OF STRING)
 4000    CONTINUE
C********INCREMENT CHARACTER POINTER
         IC=IC + 1
C********CHECK FOR END OF RECORD
         IF (IC .GT. ILMAX) GO TO 9100
C********CHECK FOR BEGINNING OF COMMENT
         IF (CARD(IC:IC) .EQ. '$') GO TO 4100
C********CHECK FOR COMMA CHARACTER
         IF (CARD(IC:IC) .EQ. ',') GO TO 4000
C********CHECK FOR BLANK CHARACTER
         IF (CARD(IC:IC) .NE. ' ') GO TO 4000
 4100    CONTINUE
C********END OF STRING FOUND
C********COMPARE STRING LENGTH TO FORMAT LENGTH
         IE=IC - 1
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          IF (IE-IS+1 .GT. IL) GO TO 9200
C*********RIGHT JUSTIFY STRING FOR INTERNAL FORMATTED READS
          TMPCRD=' '
          TMPCRD(IL+IS-IE:IL)=CARD(IS:IE)
C*********READ STRING WITH LOGICAL FORMAT (NOT USED IN GGSOR)
C          READ(TMPCRD,LFRMT,ERR=5000) LVAL
C          TYPE(2)=.TRUE.
C 5000     CONTINUE
C*********READ STRING WITH INTEGER FORMAT
C          READ(TMPCRD,IFRMT,ERR=6000) IVAL
C          TYPE(3)=.TRUE.
C 6000     CONTINUE
C*********READ STRING WITH REAL FORMAT
          READ(TMPCRD,RFRMT,ERR=7000) RVAL
          TYPE(4)=.TRUE.
C*********CHECK FOR DECIMAL POINT IN VALUE AND MAGNITUDE OF VALUE
          IF ((INDEX(TMPCRD, '.') .EQ. 0) .AND.
     1          (ABS(RVAL) .LE. 1.0E10)) THEN
             IVAL=NINT (RVAL)
             TYPE(3)=.TRUE.
          ENDIF
          GO TO 8000
 7000     CONTINUE
C*********STRING IS NOT LOGICAL, INTEGER, OR REAL SO ASSUMED TO BE CHAR
C*********COMPARE STRING LENGTH TO CHARACTER VARIABLE LENGTH
          IF (IE-IS+1 .GT. LENCVAL) GO TO 9300
C*********TRANSFER CHARACTER STRING
          CVAL=CARD(IS:IE)
C*********SET LENGTH OF CHARACTER STRING
          LENGTH=IE - IS + 1
C*********SET VARIABLE FLAG TYPE FOR CHARACTER VARIABLE FOUND
          TYPE(1)=.TRUE.
 8000     CONTINUE
       ENDIF
C*****CHECK FOR BEGINNING OF COMMENT
       IF (CARD(IC:IC) .NE. '$') IC=IC + 1
       RETURN
 9100 CONTINUE
C*****END OF RECORD ENCOUNTERED SEARCHING FOR VALUE POSITION
C*****SET END-OF-RECORD TYPE
       EOR=.TRUE.
       RETURN
 9200 CONTINUE
C*****LENGTH OF STRING TOO LONG FOR EITHER CHARACTER STORAGE OR INTERNAL
C*****FORMATTED READ
       WRITE(6,9201) CARD, IL
       RETURN
 9300 CONTINUE
C*****LENGTH OF STRING TOO LONG FOR CHARACTER VARIABLE
       WRITE(6,9301) CARD(IS:IE), LENCVAL
       RETURN
C*****FORMAT STATEMENTS
 1001 FORMAT('(I',I3,')')
 1002 FORMAT('(E',I3,'.0)')
 1003 FORMAT('(L',I3,')')
 9201 FORMAT(1X,'>>>>>LENGTH OF STRING TOO LONG FOR EITHER CHARACTER ',
     1         'STORAGE OR INTERNAL FORMATTED READ',
     2        /1X,'>>>>>',A,
     3        /1X,'>>>>>RESET PARAMETER IL IN RDSTRG TO A VALUE ',
     4         'GREATER THAN ',I3,' TO ACCOUNT FOR ',
     5        /1X,'>>>>>LARGER STRING SIZE FOR VALUES ON INPUT FILE',/)
 9301 FORMAT(1X,'>>>>>LENGTH OF STRING TOO LONG FOR CHARACTER ',
     1         'VARIABLE STORAGE ',
     2        /1X,'>>>>>',A,
     3        /1X,'>>>>>RESET CORRESPONDING CHARACTER VARIABLE LENGTH ',
     4         ' IN RDSTRG TO A VALUE ',
     4        /1X,'>>>>>GREATER THAN ',I3,' TO ACCOUNT FOR ',
     5         'LARGER STRING SIZE FOR VALUES ON INPUT FILE',/)
       END
       SUBROUTINE SEARCH (NVAR, CVAL, NAME, IPNT, IPOINT)
```

```
C*****LOCATE VARIABLE NAME CVAL USING BINARY SEARCH RETURNING IPOINT
C*****AS POSITION IN IPNT OF NAME (IPOINT=0 IF NOT LOCATED)
      CHARACTER*(*) CVAL, NAME(NVAR)
      DIMENSION IPNT(NVAR)
C
C
C*****SET LOWER LIMIT POINTER FOR SEARCH RANGE
      IL=1
C*****SET UPPER LIMIT POINTER FOR SEARCH RANGE
      IH=NVAR
C*****SET MIDPOINT POINTER FOR SEARCH RANGE
      IM=IH / 2
C*****BEGINNING OF BINARY SEARCH LOOP
 1000 CONTINUE
C*****COMPARE SEARCH ID TO CURRENT MIDPOINT ID
      IF (CVAL .EQ. NAME(IPNT(IM))) GO TO 2000
C*****CHECK TO SEE IF MIDPOINT ID IS GREATER THAN SEARCH ID
      IF (CVAL .GT. NAME(IPNT(IM))) THEN
C********SEARCH ID IS IN UPPER HALF OF SEARCH RANGE
C********RESET LOWER LIMIT POINTER TO FORMER MIDPOINT
         IL=IM
C********RESET MIDPOINT TO CURRENT INTERVAL
         IM=(IL+IH+1) / 2
      ELSE
C********SEARCH ID IS IN LOWER HALF OF SEARCH RANGE
C********RESET UPPER LIMIT POINTER TO FORMER MIDPOINT
         IH=IM
C********RESET MIDPOINT TO CURRENT INTERVAL
         IM=(IL+IH) / 2
      ENDIF
      IF (IL+1 .EQ. IH) THEN
         IF ((CVAL .NE. NAME(IPNT(IL))) .AND.
     1        (CVAL .NE. NAME(IPNT(IH)))) THEN
C***********VALUE NOT FOUND SO RETURN 0 FOR POINTER
            IPOINT=0
            RETURN
         ENDIF
      ENDIF
      GO TO 1000
 2000 CONTINUE
C*****VALUE FOUND SO RETURN MIDPOINT FOR POINTER
      IPOINT=IM
      RETURN
      END
      SUBROUTINE CSORT (NVAR, NAME, IPNT)
C*****SORT NVAR VALUES OF CHARACTER ARRAY NAME IN INCREASING ORDER
C*****USING POINTER ARRAY IPNT
      CHARACTER*(*) NAME(NVAR)
      DIMENSION IPNT(NVAR)
C
C
      N=NVAR
      L=N/2+1
      IR=N
  100 CONTINUE
      IF (L.LE.1) GO TO 700
      L=L-1
      LHOLD=IPNT(L)
  200 CONTINUE
      J=L
  300 CONTINUE
      I=J
      J=2*J
      IF (J-IR) 400, 500, 600
  400 CONTINUE
      IF (NAME(IPNT(J)) .LT. NAME(IPNT(J+1))) J=J+1
  500 CONTINUE
      IF (NAME(LHOLD) .GE. NAME(IPNT(J))) GO TO 600
      IPNT(I)=IPNT(J)
      GO TO 300
```

```
  600 CONTINUE
      IPNT(I)=LHOLD
      GO TO 100
  700 CONTINUE
      LHOLD=IPNT(IR)
      IPNT(IR)=IPNT(1)
      IR=IR - 1
      IF (IR .GT. 1) GO TO 200
      IPNT(1)=LHOLD
      RETURN
      END
      SUBROUTINE SUBVEC
C*****SUBSTITUTE SAMPLE VECTOR VALUES INTO DEFAULT VALUE ARRAY
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      COMMON /LHSBLK/ XLHS(MAXSMP)
      CHARACTER*7 NAME
      LOGICAL LDEFLT, LREAL
      COMMON /DEFLT1/ NAME(MAXVAR)
      COMMON /DEFLT2/ NVAR, NVAL, NVCB1, NVCB2, NVCB3, NVCB4,
     1                NVCB5, IDIMEN(3,MAXVAR), ISPOS(MAXVAR),
     2                ISMPPS(MAXVAL), IPNT(MAXVAR), LDEFLT(MAXVAL),
     3                LREAL(MAXVAL)
      COMMON /VALUES/ RVL(MAXVAL)
      DIMENSION IVL(MAXVAL)
      EQUIVALENCE (IVL, RVL)
C
C
C*****MAKE SAMPLE VECTOR SUBSTITUTIONS
      DO 1000 IVAL=1,NVAL
C********CHECK FOR POSITIVE SAMPLE VECTOR SUBSTITUTION POSITION
         IF (ISMPPS(IVAL) .GT. 0) THEN
C***********CHECK FOR REAL VALUE
            IF (LREAL(IVAL)) THEN
C**************TRANSFER REAL VALUE
               RVL(IVAL)=XLHS(ISMPPS(IVAL))
            ELSE
C**************TRANSFER INTEGER VALUE
               IVL(IVAL)=NINT(XLHS(ISMPPS(IVAL)))
            ENDIF
         ENDIF
 1000 CONTINUE
      RETURN
      END
      SUBROUTINE WRTPAR
C*****PRINT DEFAULT AND SAMPLE VECTOR SUBSTITUTION INFORMATION FOR
C*****BINNED AND DIRECT EXECUTIONS
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      PARAMETER (MAXPR=132)
      CHARACTER*(MAXPR) RECOUT
      CHARACTER*7 NAME
      LOGICAL LDEFLT, LREAL
      COMMON /DEFLT1/ NAME(MAXVAR)
      COMMON /DEFLT2/ NVAR, NVAL, NVCB1, NVCB2, NVCB3, NVCB4,
     1                NVCB5, IDIMEN(3,MAXVAR), ISPOS(MAXVAR),
     2                ISMPPS(MAXVAL), IPNT(MAXVAR), LDEFLT(MAXVAL),
     3                LREAL(MAXVAL)
      COMMON /VALUES/ RVL(MAXVAL)
      DIMENSION IVL(MAXVAL)
      EQUIVALENCE (IVL, RVL)
C
C
C*****PRINT HEADER RECORD
      WRITE(6,1001)
C*****LOAD OUTPUT RECORD BEFORE PRINTING (10 OR FEWER VALUES PER RECORD)
      IVAL=0
C*****LOOP OVER VARIABLES, PRINTING IN SORTED ORDER
```

```
      DO 4000 IVR=1,NVAR
         IVAR=IPNT(IVR)
C*********START NEW RECORD FOR EACH VARIABLE
         WRITE(RECOUT,2001) NAME(IVAR)
C*********INITIALLIZE VALUE POSITION FOR CURRENT VARIABLE
         IVAL=ISPOS(IVAR) - 1
C*********SET COLUMN POINTER
         IC=1
C*********LOOP OVER VALUES FOR CURRENT VARIABLE (3RD DIMENSION)
         DO 3000 IDM3=1,IDIMEN(3,IVAR)
C************LOOP OVER VALUES FOR CURRENT VARIABLE (2ND DIMENSION)
            DO 2000 IDM2=1,IDIMEN(2,IVAR)
C***************LOOP OVER VALUES FOR CURRENT VARIABLE (1ST DIMENSION)
               DO 1000 IDM1=1,IDIMEN(1,IVAR)
C******************INCREMENT VALUE POINTER
                  IVAL=IVAL + 1
C******************INCREMENT COLUMN
                  IC=IC + 11
C******************CHECK FOR DEFAULT AND SAMPLE VECTOR SUBSTITUTION
                  IF (ISMPPS(IVAL) .GT. 0) THEN
C*********************TRANSFER SAMPLE VECTOR POSITION TO OUTPUT RECORD
                     WRITE(RECOUT(IC:IC+10),2002) ISMPPS(IVAL)
                  ELSE IF (LDEFLT(IVAL)) THEN
C*********************CHECK FOR REAL DEFAULT VALUE
                     IF (LREAL(IVAL)) THEN
C************************TRANSFER REAL VALUE TO OUTPUT RECORD
                        WRITE(RECOUT(IC:IC+10),2003) RVL(IVAL)
                     ELSE
C************************TRANSFER INTEGER VALUE TO OUTPUT RECORD
                        WRITE(RECOUT(IC:IC+10),2004) IVL(IVAL)
                     ENDIF
                  ELSE
C*********************NO DEFAULT OR SAMPLE VECTOR SUBSTITUTION
                     WRITE(RECOUT(IC:IC+10),2005)
                  ENDIF
C******************CHECK FOR OUTPUT RECORD WITH MORE THAN 104 COLUMNS
                  IF (IC .GT. 104) THEN
C*********************PRINT OUTPUT RECORD
                     WRITE(6,2010) RECOUT
C*********************INITIALLIZE OUTPUT RECORD
                     RECOUT=' '
C*********************RESET COLUMN POINTER
                     IC=1
                  ENDIF
 1000          CONTINUE
 2000       CONTINUE
 3000    CONTINUE
C*********IF INFORMATION IS STORED ON OUTPUT RECORD, PRINT OUTPUT RECORD
         IF (IC .GT. 1) WRITE(6,2010) RECOUT
 4000 CONTINUE
C*****START NEW PAGE
      WRITE(6,3001)
      RETURN
C*****FORMAT STATEMENTS
 1001 FORMAT('1',/1X,130('*'),
     1       /1X,30('*'),5X,'DEFAULT INPUT AND SAMPLE VECTOR ',
     2        'SUBSTITUTION INFORMATION',5X,34('*'),
     3       /1X,130('*'),/)
 2001 FORMAT(1X,A9)
 2002 FORMAT('  V-POS-',I3.3)
 2003 FORMAT(1PE11.
 2004 FORMAT(I11)
 2005 FORMAT('    NO VALUE')
 2010 FORMAT(A)
 3001 FORMAT('1')
      END
      SUBROUTINE TRANS (CB1, CB2, CB3)
C*****TRANSFER VALUES FROM ARRAY RVAL TO COMMON BLOCKS CB1, CB2, AND CB3
C*****IN THIS ORDER
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
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     1             MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2             MAXSPC=10, MAXTIM=20)
      CHARACTER*7 NAME
      LOGICAL LDEFLT, LREAL
      COMMON /DEFLT1/ NAME(MAXVAR)
      COMMON /DEFLT2/ NVAR, NVAL, NVCB1, NVCB2, NVCB3, NVCB4,
     1                NVCB5, IDIMEN(3,MAXVAR), ISPOS(MAXVAR),
     2                ISMPPS(MAXVAL), IPNT(MAXVAR), LDEFLT(MAXVAL),
     3                LREAL(MAXVAL)
      COMMON /VALUES/ RVL(MAXVAL)
      DIMENSION IVL(MAXVAL)
      EQUIVALENCE (IVL, RVL)
      DIMENSION CB1(NVCB1), CB2(NVCB2), CB3(NVCB3)
C
C
C*****INITIALLIZE VALUE COUNTER
      IVAL=0
C*****CHECK NUMBER OF VALUES ASSIGNED TO COMMON BLOCK 1
      IF (NVCB1 .GT. 0) THEN
C********TRANSFER VALUES FOR COMMON BLOCK 1
         DO 1000 I=1,NVCB1
C***********INCREMENT VALUE COUNTER
            IVAL=IVAL + 1
            CB1(I)=RVL(IVAL)
 1000    CONTINUE
      ENDIF
C*****CHECK NUMBER OF VALUES ASSIGNED TO COMMON BLOCK 2
      IF (NVCB2 .GT. 0) THEN
C********TRANSFER VALUES FOR COMMON BLOCK 2
         DO 2000 I=1,NVCB2
C***********INCREMENT VALUE COUNTER
            IVAL=IVAL + 1
            CB2(I)=RVL(IVAL)
 2000    CONTINUE
      ENDIF
C*****CHECK NUMBER OF VALUES ASSIGNED TO COMMON BLOCK 3
      IF (NVCB3 .GT. 0) THEN
C********TRANSFER VALUES FOR COMMON BLOCK 3
         DO 3000 I=1,NVCB3
C***********INCREMENT VALUE COUNTER
            IVAL=IVAL + 1
            CB3(I)=RVL(IVAL)
 3000    CONTINUE
      ENDIF
      RETURN
      END
      SUBROUTINE WRREL
C*****PRINT CONTENTS OF COMMON BLOCKS
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1             MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2             MAXSPC=10, MAXTIM=20)
      COMMON /BASVAL/ FCOR(MAXSPC), FVES(MAXSPC), DFVPA(MAXSPC),
     1                DFCPA(MAXSPC), FEVSE(MAXSPC), FDCH(MAXSPC),
     2                FCCI(MAXSPC), DFCAV(MAXSPC), VBPUF(MAXSPC),
     3                FCONV(MAXSPC), FCONC(MAXSPC), RBDF(MAXSPC),
     4                DFSPRV(MAXSPC), DFSPRC(MAXSPC), FREVO(MAXSPC),
     5                VALISS(MAXISS), FLTI1, FLTI2, NSPEC, FLV, FHPE,
     6                EVSE, WFAC, PFAC, FPLBYE, FPLBYP, FPLBYD,
     7                FPLBYC, FTLPH, FTLPL, FTLP, TC11, TC12, TB11,
     8                TB12, TB21, TB22, TBS1, TBS2, TBR1, TBR2, TW,
     9                T1, T2, DT1, DT2, DTCDB, ELEV, PUFF, HVSPLT, FCD
C
C
      WRITE(6,1001)
C*****PRINT COMMON BLOCK BASVAL ARRAY VARIABLES
      WRITE(6,1002) 'BASVAL', 'BASE VALUES FOR GGSOR'
      WRITE(6,1003) 'FCOR   ', 'FVES   ', 'DFVPA  ', 'DFCPA  ',
     1              'FEVSE  ', 'FDCH   ', 'FCCI   ', 'DFCAV  ',
     2              'VBPUF  ', 'FCONV  '
      DO 1000 K=1,NSPEC
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         WRITE(6,1004) FCOR(K), FVES(K), DFVPA(K), DFCPA(K),
     1                 FEVSE(K), FDCH(K), FCCI(K), DFCAV(K),
     2                 VBPUF(K), FCONV(K)
 1000 CONTINUE
      WRITE(6,1003) 'FCONC  ', 'RBDF   ', 'DFSPRV ', 'DFSPRC ',
     1              'FREVO  '
      DO 2000 K=1,NSPEC
         WRITE(6,1004) FCONC(K), RBDF(K), DFSPRV(K), DFSPRC(K),
     1                 FREVO(K)
 2000 CONTINUE
      WRITE(6,1003) 'VALISS '
      DO 3000 IISS=1,MAXISS
         WRITE(6,1004) VALISS(IISS)
 3000 CONTINUE
C*****PRINT COMMON BLOCK BASVAL SINGLE VARIABLES
      WRITE(6,1003) 'FLTI1  ', 'FLTI2  ', 'NSPEC  ', 'FLV    ',
     1              'FHPE   ', 'EVSE   ', 'WFAC   ', 'PFAC   ',
     2              'FPLBYE ', 'FPLBYP '
      WRITE(6,1004) FLTI1, FLTI2, FLOAT(NSPEC), FLV,
     1              FHPE, EVSE, WFAC, PFAC,
     2              FPLBYE, FPLBYP
      WRITE(6,1003) 'FPLBYD ', 'FPLBYC ', 'FTLPH  ', 'FTLPL  ',
     1              'FTLP   ', 'TC11   ', 'TC12   ', 'TB11   ',
     2              'TB12   ', 'TB21   '
      WRITE(6,1004) FPLBYD, FPLBYC, FTLPH, FTLPL,
     1              FTLP, TC11, TC12, TB11,
     2              TB12, TB21
      WRITE(6,1003) 'TB22   ', 'TBS1   ', 'TBS2   ', 'TBR1   ',
     1              'TBR2   ', 'TW     ', 'T1     ', 'T2     ',
     2              'DT1    ', 'DT2    '
      WRITE(6,1004) TB22, TBS1, TBS2, TBR1,
     1              TBR2, TW, T1, T2,
     2              DT1, DT2
      WRITE(6,1003) 'DTCDB  ', 'ELEV   ', 'PUFF   '
      WRITE(6,1004) DTCDB, ELEV, PUFF
      RETURN
C*****FORMAT STATEMENTS
 1001 FORMAT('1')
 1002 FORMAT(//1X,130('='),
     1       /1X,5('*'),' CONTENTS OF COMMON BLOCK ',A,' ',5('*'),
     2       /7X,A,
     3       /1X,130('='))
 1003 FORMAT(/3X,10(A7,4X))
 1004 FORMAT(1X,1P,10E11.3)
      END
      SUBROUTINE BINTRN (IBIN)
C*****PERFORM BIN TRANSLATION
C
C=====BIN DIMENSIONS
C
C---------------------------------------------------------------
C===== INDX(1): PLANT DAMAGE STATES
C          1: PLANT DAMAGE STATE 1
C          2: PLANT DAMAGE STATE 2
C          3: PLANT DAMAGE STATE 3
C                   .
C                   .
C                   .
C         12: PLANT DAMAGE STATE 12
C===== INDX(2): CONTAINMENT STATUS
C          1: CONTAINMENT EQUIPMENT HATCH IS OPEN
C          2: CONTAINMENT VENTED DURING CORE DAMAGE
C          3: CONTAINMENT RUPTURE ABOVE AUX BLDG DURING CORE DAMAGE
C          4: CONTAINMENT LEAK ABOVE AUX BLDG DURING CORE DAMAGE
C          5: CONTAINMENT RUPTURE ABOVE AUX BLDG DURING VESSEL FAILURE
C          6: CONTAINMENT LEAK ABOVE AUX BLDG DURING VESSEL FAILURE
C          7: CONTAINMENT VENTED DURING LATE TIME REGIME
C          8: CONTAINMENT RUPTURE LATE IN ACCIDENT
C          9: CONTAINMENT LEAK LATE IN ACCIDENT
C         10: CONTAINMENT CLOSED
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C===== INDX(3): AUXILIARY BUILDING PRESSURE INTEGRITY STATUS
C          1: CONTAINMENT OPEN AND AUX BLDG FAILS PRIOR TO CORE DAMAGE
C          2: CONTAINMENT OPEN AND AUX BLDG FAILS DURING CORE DAMAGE
C          3: CONTAINMENT OPEN AND AUX BLDG FAILS AFTER VESSEL FAILURE
C          4: CONTAINMENT CLOSED
C===== INDX(4): DRYWELL PRESSURE INTEGRITY STATUS
C          1: DRYWELL EQUIPMENT HATCH OPEN
C          2: DRYWELL EQUIPMENT HATCH CLOSED
C===== INDX(5): REACTOR VESSEL PRESSURE BOUNDARY STATUS
C          1: ISOLATED
C          2: PIPE BREAK IN DRYWELL (LOCA) PRIOR TO CORE DAMAGE
C          3: OPEN MSIV OR UNISOLATED INTERFACING SYSTEMS LOCA
C          4: UNISOLATED PRIOR TO CORE DAMAGE, ISOLATED DURING CORE DAMAGE
C===== INDX(6): REACTOR HEAD VENT STATUS PRIOR TO CORE DAMAGE
C          1: CLOSED
C          2: OPEN
C===== INDX(7): THE SRV TAILPIPE VACUUM BREAKERS STATUS
C          1: STICKS OPEN
C          2: DOES NOT STICK OPEN
C===== INDX(8): REACTOR VESSEL STATUS PRIOR TO VESSEL FAILURE
C          1: PRESSURIZED, NO COOLANT INJECTION
C          2: UNPRESSURIZED, NO COOLANT INJECTION
C          3: PRESSURIZED, COOLANT INJECTION
C          4: UNPRESSURIZED, COOLANT INJECTION
C          5: PRESSURIZED, CORE COOLING RESTORED, CORE DAMAGE ARRESTED
C          6: UNPRESSURIZED, CORE COOLING RESTORED, CORE DAMAGE ARRESTED
C===== INDX(9): CONTAINMENT SPRAYS STATUS
C          1: NOT USED
C          2: USED
C===== INDX(10): ZIRCONIUM FRACTION OXIDIZED PRIOR TO VESSEL FAILURE
C          1: GREATER THAN 0.21
C          2: LESS THAN 0.21
C===== INDX(11): CORE FRACTION PARTICIPATING IN HPME OR STEAM EXPLOSION
C          1: 40% OF CORE PARTICIPATES IN HPME
C          2: 10% OF CORE PARTICIPATES IN HPME
C          3: 40% OF CORE PARTICIPATES IN EX-VESSEL STEAM EXPLOSION
C          4: 10% OF CORE PARTICIPATES IN EX-VESSEL STEAM EXPLOSION
C          5: NO HPME, NO EX-VESSEL STEAM EXPLOSION
C===== INDX(12): CORE DEBRIS STATUS IN THE REACTOR PEDESTAL CAVITY
C          1: CORE-CONCRETE INTERACTIONS IN DRY CAVITY
C          2: CORE-CONCRETE INTERACTIONS IN FLOODED CAVITY
C          3: NO CORE-CONCRETE INTERACTIONS, CAVITY CORE DEBRIS QUENCHED
C===== INDX(13): INITIATING EVENT TIME WINDOW
C          1: 14 TO 24 HOURS AFTER SHUTDOWN
C          2: 24 TO 94 HOURS AFTER SHUTDOWN
C          3: 40 TO 50 DAYS AFTER SHUTDOWN
C===== INDX(14): SUPPRESSION POOL TEMPERATURE
C          1: SUBCOOLED SUPPRESSION POOL
C          2: SATURATED SUPPRESSION POOL
C
C ---------------------------------------------------------------------
C
C   SPECIES INDEX = ISP, 1 TO NSPEC; ORDER IS NG,I,CS,TE,SR,RU,LA,CE,BA
C
C =====================================================================
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      CHARACTER BINARR*(MAXBD), BTITLE*80, TITLE*80
      COMMON /BINS/ BINARR(MAXBIN), BTITLE, TITLE
      LOGICAL NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1   EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /KEYS/ NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1   EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /SRCTRM/ ST(MAXSPC), STE(MAXSPC), STCCI(MAXSPC),
     1                STL(MAXSPC), STIL, STRVOL(MAXSPC),
     2                ST1(MAXSPC), ST2(MAXSPC), RV(MAXSPC)
      COMMON /BASVAL/ FCOR(MAXSPC), FVES(MAXSPC), DFVPA(MAXSPC),
     1                DFCPA(MAXSPC), FEVSE(MAXSPC), FDCH(MAXSPC),
     2                FCCI(MAXSPC), DFCAV(MAXSPC), VBPUF(MAXSPC),
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     3                 FCONV(MAXSPC), FCONC(MAXSPC), RBDF(MAXSPC),
     4                 DFSPRV(MAXSPC), DFSPRC(MAXSPC), FREVO(MAXSPC),
     5                 VALISS(MAXISS), FLTI1, FLTI2, NSPEC, FLV, FHPE,
     6                 EVSE, WFAC, PFAC, FPLBYE, FPLBYP, FPLBYD,
     7                 FPLBYC, FTLPH, FTLPL, FTLP, TC11, TC12, TB11,
     8                 TB12, TB21, TB22, TBS1, TBS2, TBR1, TBR2, TW,
     9                 T1, T2, DT1, DT2, DTCDB, ELEV, PUFF, HVSPLT, FCD
      COMMON /BINNED/ FCOR0(MAXSPC,MAXCAS), FVES0(MAXSPC,MAXCAS),
     1                 DFVPA0(MAXSPC,MAXCAS), DFCPA0(MAXSPC,MAXCAS),
     2                 FDCH0(MAXSPC,MAXCAS), FEVSE0(MAXSPC,MAXCAS),
     3                 FCCI0(MAXSPC,MAXCAS), DFCAV0(MAXSPC,MAXCAS),
     4                 VBPUFC(MAXSPC,MAXCAS), FCONV0(MAXSPC,MAXCAS),
     5                 FCONC0(MAXSPC,MAXCAS), RBDF0(MAXSPC,MAXCAS),
     6                 DFSPRV0(MAXSPC,MAXCAS), DFSPRC0(MAXSPC,MAXCAS),
     7                 FREVO0(MAXSPC,MAXCAS), FLTI10(MAXCAS),
     8                 FLTI20(MAXCAS), FHPE0(MAXCAS), EVSE0(MAXCAS),
     9                 FPLBY0(3), HVSPLT0, TW0(MAXTIM), T10(MAXTIM),
     A                 DT10(MAXTIM), DT20(MAXTIM), PUFF0(MAXTIM),
     B                 E0(MAXTIM)
      COMMON /BININD/ INDX(MAXBD)
C
C
      ICAM1=ICHAR('A') - 1
C*****PLANT DAMAGE STATE
      INDX(1) = ICHAR(BINARR(IBIN)(1:1)) - ICAM1
C*****CONTAINMENT STATUS
      INDX(2) = ICHAR(BINARR(IBIN)(2:2)) - ICAM1
C*****AUXILIARY BUILIDING PRESSURE INTEGRITY STATUS
      INDX(3) = ICHAR(BINARR(IBIN)(3:3)) - ICAM1
C*****DRYWELL PRESSURE INTEGRITY STATUS
      INDX(4) = ICHAR(BINARR(IBIN)(4:4)) - ICAM1
C*****REACTOR VESSEL PRESSURE BOUNDARY STATUS
      INDX(5) = ICHAR(BINARR(IBIN)(5:5)) - ICAM1
C*****REACTOR HEAD VENT STATUS PRIOR TO CORE DAMAGE
      INDX(6) = ICHAR(BINARR(IBIN)(6:6)) - ICAM1
C*****SRV TAILPIPE VACUUM BREAKERS STATUS
      INDX(7) = ICHAR(BINARR(IBIN)(7:7)) - ICAM1
C*****REACTOR VESSEL STATUS PRIOR TO VESSEL FAILURE
      INDX(8) = ICHAR(BINARR(IBIN)(8:8)) - ICAM1
C*****CONTAINMENT SPRAYS STATUS
      INDX(9) = ICHAR(BINARR(IBIN)(9:9)) - ICAM1
C*****ZIRCONIUM FRACTION OXIDIZED PRIOR TO VESSEL FAILURE
      INDX(10)=ICHAR(BINARR(IBIN)(10:10)) - ICAM1
C*****CORE FRACTION PARTICIPATING IN HPME OR STEAM EXPLOSION
      INDX(11)=ICHAR(BINARR(IBIN)(11:11)) - ICAM1
C*****CORE DEBRIS STATUS IN THE REACTOR PEDESTAL CAVITY
      INDX(12)=ICHAR(BINARR(IBIN)(12:12)) - ICAM1
C*****INITIATING EVENT TIME WINDOW
      INDX(13)=ICHAR(BINARR(IBIN)(13:13)) - ICAM1
C*****SUPPRESSION POOL TEMPERATURE
      INDX(14)=ICHAR(BINARR(IBIN)(14:14)) - ICAM1
C*****SET LOGICAL FLAGS TO BE PASSED TO GGSORC
C*****TEMPORARY COOLABLE DEBRIS BED OR COOLABLE DEBRIS BED
      TMPCDB=.FALSE.
      CDB=(INDX(12) .EQ. 3)
C*****VESSEL BREACH
      VB=(INDX(8) .LE. 4)
C*****SUPPRESSION POOL TEMPERATURE
      SUBCL=(INDX(14) .EQ. 1)
C*****NO CONTAINMENT FAILURE FLAG
      NOCF=(INDX(2) .EQ. 10)
C*****EARLY CF BEFORE VB INCLUDES CASE WITH ILOCA OR OPEN MSIVS
      ECF=((INDX(2) .LE. 4) .OR. (INDX(5) .EQ. 3))
C*****INTERMEDIATE CF AT VB
      ICF=(INDX(2) .EQ. 5) .OR. (INDX(2) .EQ. 6)
C*****TAIL PIPE VACUUM BREAKER STUCK OPEN FLAG
      BRKOPN=(INDX(7) .EQ. 1)
C*****REACTOR HEAD VENT SPLIT FRACTION
      IF ((INDX(6) .EQ. 2) .AND. (INDX(5) .EQ. 1)) THEN
         HVSPLT=HVSPLT0
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      ELSE
         HVSPLT=0.0
      ENDIF
C*****LOOP OVER SPECIES
      DO 500 ISP=1,NSPEC
C*********FCOR *****************
         IF (INDX(8) .GE. 5) THEN
            FCOR(ISP)=FCD * FCOR0(ISP,2)
         ELSE IF (INDX(10) .EQ. 1) THEN
            FCOR(ISP)=FCOR0(ISP,1)
         ELSE IF (INDX(10) .EQ. 2) THEN
            FCOR(ISP)=FCOR0(ISP,2)
         ENDIF
C*********FVES ****************
         IF ((INDX(8) .EQ. 1) .OR. (INDX(8) .EQ. 3) .OR.
     1                             (INDX(8) .EQ. 5)) THEN
C************HIGH PRESSURE AT VB
            FVES(ISP)=FVES0(ISP,1)
         ELSE
C************LOW PRESSURE AT VB
            FVES(ISP)=FVES0(ISP,2)
         ENDIF
C*********REVOLATILIZATION AFTER VESSEL BREACH
         IF (INDX(8) .GE. 5) THEN
C************NO VESSEL BREACH, NO REVOLATILIZATION
            FREVO(ISP)=0.0
         ELSE IF ((INDX(8) .EQ. 3) .OR. (INDX(8) .EQ. 4)) THEN
C************LPI RECOVERY, AFTER VB
            FREVO(ISP)=FREVO0(ISP,3)
         ELSE
C************NO LPI RECOVERY AFTER VB
            FREVO(ISP)=FREVO0(ISP,1)
         ENDIF
C*********FCCI ************
         IF (CDB) THEN
C************COOLABLE DEBRIS BED: NO CCI RELEASE
            FCCI(ISP)=0.0
         ELSE IF ((INDX(12) .EQ. 1) .OR. TMPCDB) THEN
            IF (INDX(10) .EQ. 1) THEN
C***************HIGH ZR OXIDATION .IE., LOW ZR CONTENT IN MCCI
               FCCI(ISP)=FCCI0(ISP,1)
            ELSE
C***************LOW ZR OXIDATION IE., HIGH ZR CONTENT IN MCCI
               FCCI(ISP)=FCCI0(ISP,3)
            ENDIF
         ELSE
            IF (INDX(10) .EQ. 1) THEN
C***************HIGH ZR OXIDATION .IE., LOW ZR CONTENT IN MCCI
               FCCI(ISP)=FCCI0(ISP,2)
            ELSE
C***************LOW ZR OXIDATION IE., HIGH ZR CONTENT IN MCCI
               FCCI(ISP)=FCCI0(ISP,4)
            ENDIF
         ENDIF
C*********FCONV: CONTAINMENT RETENTION FOR IN-VESSEL RELEASE, OUTER
C*********       CONTAINMENT ONLY
C*********FCONC: CONTAINMENT RETENTION FOR EX-VESSEL RELEASE, INCLUDING
C*********       DRYWELL AND OUTER CONTAINMENT
C*********THIS IS RETENTION WITHOUT CONSIDERING OTHER EFFECTS SUCH AS:
C*********POOL BYPASS, CONTAINMENT SPRAYS, ETC.
         IF ((INDX(2) .EQ. 10) .AND. (INDX(5) .NE. 3)) THEN
C************NO CONTAINMENT FAILURE
            FCONV(ISP)=FCONV0(ISP,7)
            FCONC(ISP)=FCONC0(ISP,7)
         ELSE IF (((INDX(2) .EQ. 4) .OR. (INDX(2) .EQ. 6)) .AND.
     1            (INDX(5) .NE. 3)) THEN
C************EARLY LEAK
            IF (SUBCL) THEN
C***************SUPPRESSION POOL IS SUBCOOLED
               FCONV(ISP)=FCONV0(ISP,1)
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               FCONC(ISP)=FCONC0(ISP,1)
            ELSE
               FCONV(ISP)=FCONV0(ISP,2)
               FCONC(ISP)=FCONC0(ISP,2)
            ENDIF
         ELSE IF ((INDX(2) .EQ. 1) .OR. (INDX(2) .EQ. 2) .OR.
     1            (INDX(2) .EQ. 3) .OR. (INDX(2) .EQ. 5) .OR.
     1            (INDX(5) .EQ. 3)) THEN
C***********EARLY RUPTURE OR ILOCA IN AUX. BLDG
            IF (SUBCL) THEN
C**************SUPPRESSION POOL SUBCOOLED
               FCONV(ISP)=FCONV0(ISP,3)
               FCONC(ISP)=FCONC0(ISP,3)
            ELSE
C**************SUPPRESSION POOL SATURATED
               FCONV(ISP)=FCONV0(ISP,4)
               FCONC(ISP)=FCONC0(ISP,4)
            ENDIF
         ELSE IF (INDX(2) .EQ. 9) THEN
C***********LATE LEAK
            FCONV(ISP)=FCONV0(ISP,5)
            FCONC(ISP)=FCONC0(ISP,5)
         ELSE
C***********LATE RUPTURE OR LATE VENTING
            FCONV(ISP)=FCONV0(ISP,6)
            FCONC(ISP)=FCONC0(ISP,6)
         ENDIF
         IF ((INDX(5) .EQ. 3) .OR. (INDX(5) .EQ. 4)) THEN
           FCONV(ISP)= 1.0
         ENDIF
C*********RBDF *************
         IF ((INDX(3) .GE. 1) .AND. (INDX(3) .LE. 3)) THEN
C***********RELEASE GOES THROUGH REACTOR BUILDING
            RBDF(ISP)=RBDF0(ISP,1)
         ELSE
C***********REACTOR BUILDING BYPASSED
            RBDF(ISP)=1.0
         ENDIF
C*********FDCH OR EX-VESSEL STEAM EXPLOSION ***********
         IF (INDX(11) .EQ. 5) THEN
C***********NO DCH, NO STEAM EXPLOSION
            FHPE=0.0
            EVSE=0.0
            FDCH(ISP)=0.0
            FEVSE(ISP)=0.0
         ELSE IF (INDX(11) .LE. 2) THEN
C***********DCH, NO STEAM EXPLOSION
            EVSE=0.
            FEVSE(ISP)=0.
            FDCH(ISP)=FDCH0(ISP,1)
            IF (INDX(11) .EQ. 1) THEN
               FHPE=FHPE0(1)
            ELSE
               FHPE=FHPE0(2)
            ENDIF
         ELSE
C***********NO DCH, BUT EX-VESSEL STEAM EXPLOSION
            FHPE=0.0
            FDCH(ISP)=0.0
            FEVSE(ISP)=FEVSE0(ISP,1)
            IF (INDX(11) .EQ. 3) THEN
               EVSE=EVSE0(1)
            ELSE
               EVSE=EVSE0(2)
            ENDIF
         ENDIF
C*********POOL BYPASS
C*********FPLBYE, FPLBYI, AND FPLBYL:
C*********FOR EARLY PHASE, ASSUME VACUUM BREAKER STICKS OPEN FOR ASSIGNING
C*********FPLBYE.  IF BRKOPEN IS FALSE, SET FPLBYE TO 0.0 LATER
```

```
         IF (INDX(4) .EQ. 1) THEN
C************COMPLETE POOL BYPASS
            FPLBYE=FPLBY0(3)
            FPLBYI=FPLBY0(3)
            FPLBYL=FPLBY0(3)
         ELSE
C************NOMINAL POOL BYPASS
            FPLBYE=FPLBY0(1)
            FPLBYI=FPLBY0(1)
            FPLBYL=FPLBY0(1)
         ENDIF
C*********IF BRKOPN IS FALSE, THEN FPLBYE=0.0 IRREGARDLESS OF DRYWELL
C*********LEAKAGE SINCE EVERYTHING GOES THROUGH POOL
         IF (.NOT. BRKOPN) FPLBYE=0.0
         IF (BRKOPN) THEN
C************FOR VACUUM BREAKER STUCK OPEN CASES, ASSIGN TAIL PIPE FLOW
C************FRACTION FOR HIGH PRESSURE VERSUS LOW PRESSURE SEQUENCES
            IF ((INDX(8) .EQ. 1) .OR. (INDX(8) .EQ. 3) .OR.
     1                               (INDX(8) .EQ. 5)) THEN
C***************VESSEL AT HIGH PRESSURE
               FTLP=FTLPH
            ELSE
C***************VESSEL AT LOW PRESSURE
               FTLP=FTLPL
            ENDIF
         ELSE
C************VACUUM BREAKER STAYS CLOSED, NO POOL BYPASS
            FTLP=0.0
         ENDIF
C*********THE THREE POOL BYPASS FRACTIONS ARE FOR A "DRY CAVITY" AND
C*********"FAILED CONTAINMENT". IT IS MULTIPLIED BY 'PFAC' IF CONTAINMENT
C*********HAS NOT FAILED AND DIVIDED BY 'WFAC' IF THE CAVITY IS FLOODED.
C*********
C*********ESTIMATE BYPASS FRACTION FOR THE VESSEL BREACH PUFF (FPLBYP),
C*********DCH (FPLBYD) AND CCI RELEASES (FPLBYC)
C*********
C*********FOR IN-VESSEL RELEASE PHASE, ASSUMES NO PRESSURE FACTOR (PFAC)
C*********APPLIES BUT STEAMING FACTOR (WFAC) ALWAYS APPLIES
         FPLBYE=FPLBYE / WFAC
C*********FOR THE PUFF CASE, IT IS ASSUMED VALUES WITH STEAM ALWAYS APPLY
C*********ROUGHLY CONSISTENT WITH TB2
C*********FOR DCH, POOL BYPASS IS TREATED LIKE PUFF RELEASE
         FPLBYP=FPLBYI / WFAC
         FPLBYD=FPLBYI / WFAC
C*********LATE CONTAINMENT FAILURE CASES, APPLY PRESSURE CORRECTION
         IF (INDX(2) .GE. 7) THEN
            FPLBYP=FPLBYP * PFAC
            FPLBYD=FPLBYD * PFAC
         ENDIF
C*********CCI RELEASE
         FPLBYC=FPLBYL
C*********FOR WET OR FLOODED CAVITY CASES, STEAMING FACTOR APPLIES
         IF (INDX(12) .EQ. 2) FPLBYC=FPLBYC / WFAC
         IF (INDX(2) .GE. 7) FPLBYC=FPLBYC * PFAC
C*********ALL FRACTIONS OF POOL BYPASS SHOULD BE <= 1.0
         FPLBYE=MIN (FPLBYE, 1.0)
         FPLBYP=MIN (FPLBYP, 1.0)
         FPLBYD=MIN (FPLBYD, 1.0)
         FPLBYC=MIN (FPLBYC, 1.0)
C*********LATE IODINE RELEASE FROM POOL
         IF (SUBCL) THEN
            FLTI1=FLTI10(1)
         ELSE
            FLTI1=FLTI10(2)
         ENDIF
C*********LATE IODINE RELEASE FROM CAVITY WATER
         IF ((INDX(12) .EQ. 1) .OR. TMPCDB) THEN
C************DRY CAVITY CASES
            FLTI2=1.0
         ELSE IF (INDX(12) .EQ. 2) THEN
```

```
C***********FLOODED CAVITY CASE LIKE TC
            FLTI2=FLTI20(2)
          ELSE
C***********NO CCI RELEASE CASE
            FLTI2=0.0
          ENDIF
C*********IN-VESSEL RELEASE POOL SCRUBBING
          IF (INDX(5) .EQ. 1) THEN
            DFVPA(ISP)=DFVPA0(ISP,1)
          ELSE
            DFVPA(ISP)=1.0
          ENDIF
C*********EX-VESSEL RELEASE POOL SCRUBBING
          DFCPA(ISP) = DFCPA0(ISP,1)
C*********CONTAINMENT (WETWELL) SPRAY DF
          IF (INDX(9) .EQ. 1) THEN
C***********NO SPRAYS
            DFSPRV(ISP)=1.0
            DFSPRC(ISP)=1.0
          ELSE
C***********EARLY SPRAYS AND LATE SPRAYS
            DFSPRV(ISP)=DFSPRV0(ISP,1)
            DFSPRC(ISP)=DFSPRC0(ISP,1)
          ENDIF
          IF ((INDX(5) .EQ. 3) .OR. (INDX(5) .EQ. 4)) THEN
            DFSPRV(ISP)=1.0
          ENDIF
C*********REACTOR CAVITY WATER SCRUBBING OF FISSION PRODUCTS
          IF ((INDX(12) .EQ. 1) .OR. TMPCDB) THEN
C***********DRY CAVITY OR DELAYED CCI RELEASE CASE
            DFCAV(ISP)=1.0
          ELSE
C***********FLOODED CAVITY  CDB: LIKE BMI-2139 TC
            DFCAV(ISP)=DFCAV0(ISP,2)
          ENDIF
C*********OTHER VARIABLES NOT SAMPLED IN LHS
C*********ASSUMES ALL CORE ULTIMATELY LEAVE VESSEL AFTER VESSEL BREACH
          FLV=1.0
          IF (.NOT. VB) FLV=0.0
C*********VESSEL BREACH PUFF RELEASE
          VBPUF(ISP)=VBPUF0(ISP,1)
  500 CONTINUE
C***
C***ST TIMING PARAMETERS
C***
C***DURATION OF 2 RELEASE SEGMENT ALWAYS SET TO 24 HOURS UNLESS NO VB
      IF (INDX(8) .LE. 4) THEN
        DT2=DT20(1)
      ELSE
        DT2=DT20(2)
      ENDIF
C***WARNING TIME
C***PDS1-1: LOCA TIME WINDOW 1
      IF (INDX(1) .EQ. 1) THEN
        TW=TW0(1)
        IF ((INDX(3) .EQ. 1) .OR. (INDX(3) .EQ. 2)) THEN
          T1=T10(1)
          DT1=DT10(5)
        ELSE
          T1=T10(2)
          DT1=DT10(2)
        ENDIF
C***PDS1-2 & PDS1-4: SBO W/ SDC BREAK
      ELSEIF ((INDX(1) .EQ. 2) .OR. (INDX(1) .EQ. 4)) THEN
          TW=TW0(2)
          T1=T10(3)
          DT1=DT10(4)
C***PDS1-3: SBO W/FIREWATER FOR 10 HOURS
      ELSEIF (INDX(1) .EQ. 3) THEN
          TW=TW0(3)
```

```
          T1=T10(4)
          DT1=DT10(3)
C***PDS1-5: RPV@LoP, FLOODED CNMT
      ELSEIF (INDX(1) .EQ. 5) THEN
        TW=TW0(4)
        IF ((INDX(3) .EQ. 1) .OR. (INDX(3) .EQ. 2)) THEN
          T1=T10(5)
          DT1=DT10(4)
        ELSE
          T1=T10(6)
          DT1=DT10(2)
        ENDIF
C***PDS2-1: LOCA TIME WINDOW 2
      ELSE IF (INDX(1) .EQ. 6) THEN
        TW=TW0(5)
        IF ((INDX(3) .EQ. 1) .OR. (INDX(3) .EQ. 2)) THEN
          T1=T10(7)
          DT1=DT10(5)
        ELSE
          T1=T10(8)
          DT1=DT10(2)
        ENDIF
C***PDS2-2 & PDS2-3: SBO W/ SDC BREAK
      ELSE IF ((INDX(1) .EQ. 7) .OR. (INDX(1) .EQ. 8)) THEN
          TW=TW0(6)
          T1=T10(9)
          DT1=DT10(3)
C***PDS2-4: RPV@LoP, FLOODED CNMT
      ELSE IF (INDX(1) .EQ. 9) THEN
        TW=TW0(7)
        IF ((INDX(3) .EQ. 1) .OR. (INDX(3) .EQ. 2)) THEN
          T1=T10(10)
          DT1=DT10(5)
        ELSE
          T1=T10(11)
          DT1=DT10(2)
        ENDIF
C***PDS2-5: RPV@HiP
      ELSE IF (INDX(1) .EQ. 10) THEN
C***CNMT OPEN TO AUX. BLDG
        IF (INDX(3) .NE. 4) THEN
          TW=TW0(8)
          T1=T10(12)
          DT1=DT10(3)
C***CNMT CLOSED PRIOR TO CD
        ELSE
C***CNMT FAILS DURING CD
          IF ((INDX(2) .GE. 2) .AND. (INDX(2) .LE. 4)) THEN
            TW=TW0(9)
            T1=T10(13)
            DT1=DT10(3)
C********** CNMT FAILS AT VB
          ELSE IF ((INDX(2) .EQ. 5) .OR. (INDX(2) .EQ. 6)) THEN
            TW=TW0(10)
            T1=T10(14)
C***CNMT FAILS AS A RUPTURE
            IF (INDX(2) .EQ. 5) THEN
              DT1=DT10(1)
C***CNMT FAILS AS A LEAK
            ELSE
              DT1=DT10(3)
            ENDIF
C********** CNMT FAILS LATE OR NOT AT ALL
          ELSE
            TW=TW0(11)
            T1=T10(15)
C***CNMT FAILS AS A RUPTURE
            IF ((INDX(2) .EQ. 7) .OR. (INDX(2) .EQ. 8)) THEN
              DT1=DT10(1)
C***CNMT FAILS AS A LEAK OR NOT AT ALL
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          ELSE
            DT1=DT10(3)
          ENDIF
        ENDIF
      ENDIF
C*****PDS2-6: RPV@LoP, OPEN MSIVs
      ELSE IF (INDX(1) .EQ. 11) THEN
        TW=TW0(12)
        T1=T10(16)
        DT1=DT10(5)
C*****PDS3-1: LOCA TIME WINDOW 3
      ELSE IF (INDX(1) .EQ. 12) THEN
        TW=TW0(13)
        IF ((INDX(3) .EQ. 1) .OR. (INDX(3) .EQ. 2)) THEN
          T1=T10(17)
          DT1=DT10(5)
        ELSE
          T1=T10(18)
          DT1=DT10(2)
        ENDIF
      ENDIF
C*****START OF SECOND RELEASE
C*****NO TEMPORARY COOLABLE DEBRIS BED
      T2=T1 + DT1
C*****TEMPORARY COOLABLE DEBRIS BED
      IF (TMPCDB) T2=T2 + DTCDB
C*****FOR LATE CONTAINMENT FAILURE CASES, ASSIGN FRACTION
C*****OF TOTAL RELEASE TO THE FIRST RELEASE SEGMENT
C*****SET DEFAULT OF PUFF TO 1.0
      IF ((INDX(2) .EQ. 7) .OR. (INDX(2) .EQ. 8)) THEN
C*********LATE RUPTURE OR LATE VENT
        PUFF=PUFF0(1)
      ELSE IF ((INDX(2) .EQ. 9) .OR. (INDX(2) .EQ. 10)) THEN
C*********LATE LEAK OR NO CONTAINMENT FAILURE
        PUFF=PUFF0(2)
      ELSE
C*********SET DEFAULT
        PUFF=1.0
      ENDIF
      RETURN
      END
      SUBROUTINE EXPTAB
C*****SET VARIABLES IN COMMON BLOCK BINNED BY INTERPOLATION OF
C*****EXPERT OPINION TABLES
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      COMMON /BASVAL/ FCOR(MAXSPC), FVES(MAXSPC), DFVPA(MAXSPC),
     1                DFCPA(MAXSPC), FEVSE(MAXSPC), FDCH(MAXSPC),
     2                FCCI(MAXSPC), DFCAV(MAXSPC), VBPUF(MAXSPC),
     3                FCONV(MAXSPC), FCONC(MAXSPC), RBDF(MAXSPC),
     4                DFSPRV(MAXSPC), DFSPRC(MAXSPC), FREVO(MAXSPC),
     5                VALISS(MAXISS), FLTI1, FLTI2, NSPEC, FLV, FHPE,
     6                EVSE, WFAC, PFAC, FPLBYE, FPLBYP, FPLBYD,
     7                FPLBYC, FTLPH, FTLPL, FTLP, TC11, TC12, TB11,
     8                TB12, TB21, TB22, TBS1, TBS2, TBR1, TBR2, TW,
     9                T1, T2, DT1, DT2, DTCDB, ELEV, PUFF, HVSPLT, FCD
      COMMON /BINNED/ FCOR0(MAXSPC,MAXCAS), FVES0(MAXSPC,MAXCAS),
     1                DFVPA0(MAXSPC,MAXCAS), DFCPA0(MAXSPC,MAXCAS),
     2                FDCH0(MAXSPC,MAXCAS), FEVSE0(MAXSPC,MAXCAS),
     3                FCCI0(MAXSPC,MAXCAS), DFCAV0(MAXSPC,MAXCAS),
     4                VBPUF0(MAXSPC,MAXCAS), FCONV0(MAXSPC,MAXCAS),
     5                FCONC0(MAXSPC,MAXCAS), RBDF0(MAXSPC,MAXCAS),
     6                DFSPRV0(MAXSPC,MAXCAS), DFSPRC0(MAXSPC,MAXCAS),
     7                FREVO0(MAXSPC,MAXCAS), FLTI10(MAXCAS),
     8                FLTI20(MAXCAS), FHPE0(MAXCAS), EVSE0(MAXCAS),
     9                FPLBY0(3), HVSPLT0, TW0(MAXTIM), T10(MAXTIM),
     A                DT10(MAXTIM), DT20(MAXTIM), PUFF0(MAXTIM),
     B                E0(MAXTIM)
      COMMON /EXPERT/ FCORL(MAXSPC,MAXLEV,MAXCAS),
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     1                  FVESL(MAXSPC,MAXLEV,MAXCAS),
     2                  FREVOL(MAXSPC,MAXLEV,MAXCAS),
     3                  FCCIL(MAXSPC,MAXLEV,MAXCAS),
     4                  FCONVL(MAXSPC,MAXLEV,MAXCAS),
     5                  FCONCL(MAXSPC,MAXLEV,MAXCAS),
     6                  FLTI1L(MAXLEV,MAXCAS), FLTI2L(MAXLEV,MAXCAS),
     7                  RBDFL(MAXSPC,MAXLEV,MAXCAS),
     8                  FDCHL(MAXSPC,MAXLEV,MAXCAS),
     9                  FEVSEL(MAXSPC,MAXLEV,MAXCAS),
     A                  DFVPAL(MAXSPC,MAXLEV,MAXCAS),
     B                  DFCPAL(MAXSPC,MAXLEV,MAXCAS),
     C                  DFCAVL(MAXSPC,MAXLEV,MAXCAS),
     D                  DFSPRVL(MAXSPC,MAXLEV,MAXCAS),
     E                  DFSPRCL(MAXSPC,MAXLEV,MAXCAS),
     F                  PRBLEV(MAXLEV)
      DATA I1 / 1 /
C
C
C*****SET VALUES FOR RELEASE FRACTIONS DURING IN-VESSEL RELEASE
      CALL INTERP (MAXSPC, MAXLEV, MAXCAS, VALISS(1), FCORL, FCOR0,
     1             PRBLEV)
C*****SET VALUES FOR RELEASE FRACTIONS FROM VESSEL
      CALL INTERP (MAXSPC, MAXLEV, MAXCAS, VALISS(2), FVESL, FVES0,
     1             PRBLEV)
C*****SET VALUES FOR REVOLATILIZATION RELEASE AFTER VESSEL BREACH
      CALL INTERP (MAXSPC, MAXLEV, MAXCAS, VALISS(3), FREVOL, FREVO0,
     1             PRBLEV)
C*****SET VALUES FOR RELEASE FRACTIONS DURING CCI RELEASE
      CALL INTERP (MAXSPC, MAXLEV, MAXCAS, VALISS(4), FCCIL, FCCI0,
     1             PRBLEV)
C*****SET VALUES FOR RELEASE FRACTIONS FROM CONTAINMENT TO ENVIRONMENT
C*****FOR IN-VESSEL RELEASE SOURCE TERMS
      CALL INTERP (MAXSPC, MAXLEV, MAXCAS, VALISS(5), FCONVL, FCONV0,
     1             PRBLEV)
C*****SET VALUES FOR RELEASE FRACTIONS FROM CONTAINMENT TO ENVIRONMENT
C*****FOR EX-VESSEL RELEASE SOURCE TERMS
      CALL INTERP (MAXSPC, MAXLEV, MAXCAS, VALISS(6), FCONCL, FCONC0,
     1             PRBLEV)
C*****SET VALUES FOR RELEASE FRACTIONS FOR LATE IODINE RELEASE FROM
C*****SUPPRESSION POOL
      CALL INTERP (I1, MAXLEV, MAXCAS, VALISS(7), FLTI1L, FLTI10,
     1             PRBLEV)
C*****SET VALUES FOR RELEASE FRACTIONS FOR LATE IODINE RELEASE FROM
C*****CAVITY WATER
      CALL INTERP (I1, MAXLEV, MAXCAS, VALISS(8), FLTI2L, FLTI20,
     1             PRBLEV)
C*****SET VALUES FOR REACTOR BUILDING DECONTAMINATION FACTOR
      CALL INTERP (MAXSPC, MAXLEV, MAXCAS, VALISS(9), RBDFL, RBDF0,
     1             PRBLEV)
C*****SET VALUES FOR RELEASE FRACTIONS DUE TO DIRECT CONTAINMENT HEATING
      CALL INTERP (MAXSPC, MAXLEV, MAXCAS, VALISS(10), FDCHL, FDCH0,
     1             PRBLEV)
C*****SET VALUES FOR SUPPRESSION POOL DF FOR IN-VESSEL RELEASE
      CALL INTERP (MAXSPC, MAXLEV, MAXCAS, VALISS(11), DFVPAL, DFVPA0,
     1             PRBLEV)
C*****SET VALUES FOR SUPPRESSION POOL DF AFTER VESSEL BREACH
      CALL INTERP (MAXSPC, MAXLEV, MAXCAS, VALISS(12), DFCPAL, DFCPA0,
     1             PRBLEV)
C*****SET VALUES FOR CAVITY WATER DF FOR CCI RELEASE
      CALL INTERP (MAXSPC, MAXLEV, MAXCAS, VALISS(13), DFCAVL, DFCAV0,
     1             PRBLEV)
C*****SET VALUES FOR CONTAINMENT SPRAYS DF FOR IN-VESSEL RELEASE
      CALL INTERP (MAXSPC, MAXLEV, MAXCAS, VALISS(14), DFSPRVL, DFSPRV0,
     1             PRBLEV)
C*****SET VALUES FOR CONTAINMENT SPRAYS DF FOR EX-VESSEL RELEASE
      CALL INTERP (MAXSPC, MAXLEV, MAXCAS, VALISS(15), DFSPRCL, DFSPRC0,
     1             PRBLEV)
C*****SET VALUES FOR EX-VESSEL STEAM EXPLOSION RELEASE
      CALL INTERP (MAXSPC, MAXLEV, MAXCAS, VALISS(16), FEVSEL, FEVSE0,
     1             PRBLEV)
```

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      RETURN
      END
      SUBROUTINE INTERP (MAXSPC, MAXLEV, MAXCAS, PROB, RL, R0, PRBLEV)
C*****PERFORM INTERPOLATION IN SPECIFIED EXPERT OPINION TABLE
      DIMENSION RL(MAXSPC,MAXLEV,MAXCAS), R0(MAXSPC,MAXCAS),
     1          PRBLEV(MAXLEV)
      LOGICAL FIRST
      DATA FIRST / .TRUE. /
C
C
      IF (FIRST) THEN
C********DETERMINE NUMBER OF LEVELS
         DO 100 ILEV=2,MAXLEV
            IF (PRBLEV(ILEV) .LE. 0.0) THEN
               NLEV=ILEV - 1
               GO TO 200
            ENDIF
  100    CONTINUE
         NLEV=MAXLEV
  200    CONTINUE
         IF (NLEV .LE. 1) THEN
            WRITE(6,1002)
            STOP
         ENDIF
         FIRST=.FALSE.
      ENDIF
C*****VALIDATE PROBABILITY
      IF (PROB .LT. PRBLEV(1)) THEN
         WRITE(6,1001) PROB, (PRBLEV(I),I=1,NLEV)
         STOP
      ENDIF
C*****LOCATE PROBABILITY LEVELS TO INTERPOLATE BETWEEN
      DO 1000 ILEV=2,NLEV
         IF (PROB .LE. PRBLEV(ILEV)) THEN
            JLEV=ILEV
            GO TO 2000
         ENDIF
 1000 CONTINUE
C*****PROBABILITY VALUE OUTSIDE OF TABLE RANGE
      WRITE(6,1001) PROB, (PRBLEV(I),I=1,NLEV)
      STOP
 2000 CONTINUE
C*****LOOP OVER CASES
      DO 4000 ICAS=1,MAXCAS
C********LOOP OVER SPECIES
         DO 3000 ISPEC=1,MAXSPC
C***********PERFORM INTERPOLATION FOR CURRENT SPECIES AND CASE
            IF ((RL(ISPEC,1,ICAS) .GT. 0.0) .AND.
     1          (RL(ISPEC,NLEV,ICAS)/RL(ISPEC,1,ICAS) .GT. 10.)) THEN
C**************PERFORM LOGARITHMIC INTERPOLATION
               R0(ISPEC,ICAS)=10.**(LOG10(RL(ISPEC,JLEV-1,ICAS)) +
     1                        (PROB-PRBLEV(JLEV-1)) *
     2    (LOG10(RL(ISPEC,JLEV,ICAS))-LOG10(RL(ISPEC,JLEV-1,ICAS))) /
     3                     (PRBLEV(JLEV)-PRBLEV(JLEV-1)))
            ELSE
C**************PERFORM LINEAR INTERPOLATION
               R0(ISPEC,ICAS)=RL(ISPEC,JLEV-1,ICAS) +
     1                        (PROB-PRBLEV(JLEV-1)) *
     2             (RL(ISPEC,JLEV,ICAS)-RL(ISPEC,JLEV-1,ICAS)) /
     3                     (PRBLEV(JLEV)-PRBLEV(JLEV-1))
            ENDIF
 3000    CONTINUE
 4000 CONTINUE
      RETURN
C*****FORMAT STATEMENTS
 1001 FORMAT(/1X,'>>>>>PROBABILITY VALUE (',F5.2,') OUT OF RANGE FOR ',
     1       'INTERPOLATION OF LEVELS',
     2       /1X,'>>>>>PRBLEV(I)=',20F6.3)
 1002 FORMAT(/1X,'>>>>>FEWER THAN 2 PROBABILITY LEVELS (PRBLEV) ',
     1       'SPECIFIED')
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      END
      SUBROUTINE GGSORC (IOBS, IBIN)
C*****CALCULATE XXSOR TYPE OF SOURCE TERMS FOR THE GRAND GULF
C-------------------------------------------------------------
C
C                       - - - - OUTPUT - - -
C
C  ST(ISP)      == TOTAL ENVIRONMENTAL RELEASE FRACTIONS FOR SPECIES 'ISP'
C                  (EARLY + LATE)
C  STE(ISP)     == RELEASES UP THROUGH VESSEL BREACH. THE DEFINING TIME
C                  IS RELEASE TO THE CONTAINMENT; ACTUAL RELEASE TO THE
C                  ENVIRONMENT WILL BE LATER IF CONTAINMENT FAILURE IS LATER
C  STCCI(ISP) == CCI RELEASE SOURCE TERMS
C  STL(ISP)     == LATE RELEASE SOURCE TERMS (CCI+STIL+STRVOL)
C  STIL         == "LATE" IODINE COMPONENT, TREATED AS GASEOUS (E.G., ORGANIC)
C                  IODINE RELEASED FROM POOL AND FLOODED CAVITY;
C                  NO DF'S OR CONTAINMENT RETENTION FACTORS APPLY
C  STRVOL(ISP)== I, CS AND TE COMPONENT REVOLATILIZED FROM PRIMARY SYSTEM:
C                  TREATED AS AEROSOL; DF'S FOR SPRAYS, SUPPRESSION POOL
C                  SCRUBBING, AND CONTAINMENT RETENTION APPLY
C=====================================================================
C
C   SPECIES INDEX=ISP, 1 TO NSPEC; ORDER IS NG,I,CS,TE,SR,RU,LA,CE,BA
C
C  FCOR  == RELEASE FRACTION OF EACH ELEMENT GROUP FROM THE FUEL DURING
C           DURING IN-VESSEL RELEASE
C  FVES  == RELEASE FRACTION FROM THE VESSEL (FRACTION OF FCOR)
C  DFVPA == POOL DF'S DURING IN-VESSEL RELEASE
C  DFCPA == POOL DF'S DURING CCI RELEASE
C  VBPUF == PUFF RELEASE FRACTION OF THE TOTAL CORE AT VESSEL BREACH
C                - - - POOL BYPASS PARAMETERS - - -
C  FPLBYE, FPLBYP, FPLBYD, FPLBYC  ==
C           FRACTION OF POOL BYPASS AT DIFFERENT TIME STEPS:
C           EARLY (BEFORE VB), PUFF SOURCE TERMS,
C           DCH SOURCE TERMS, AND CCI SOURCE TERMS.
C           THIS FRACTION DO NOT GO THROUGH SUPPRESSION POOL
C  FCONV == FRACTIONS OF AEROSOL SPECIES RELEASED FROM THE RCS TO THE
C           CONTAINMENT AND THEN TO THE ENVIRONMENT
C  FCONC == FRACTIONS OF AEROSOL SPECIES RELEASED TO FROM CCI TO THE
C           CONTAINMENT AND THEN TO THE ENVIRONMENT (INCLUDES DRYWELL
C           RETENTION AND OUTER CONTAINMENT RETENTION)
C  DFSPRV== DF'S FOR SPRAYS (ESTIMATED FROM CALCULATED CS AND I RELEASES)
C  DFSPRC== DF'S FOR SPRAYS (ESTIMATED FROM CALCULATED SR AND CE RELEASES)
C  FLTI1 == LATE IODINE RELEASE FROM SUPPRESSION POOL
C  FLTI2 == LATE IODINE RELEASE FROM CAVITY WATER
C
C  RBDF  == REACTOR BUILDING DECONTAMINATION FACTOR, ADDED TO THE CODE FOR
C           USE IN THE GRAND GULF LOW POWER SHUTDOWN STUDY BY LANNY SMITH,
C           8 APR 92.
C
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      CHARACTER BINARR*(MAXBD), BTITLE*80, TITLE*80
      COMMON /BINS/ BINARR(MAXBIN), BTITLE, TITLE
      LOGICAL NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1    EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /KEYS/ NOCALC, SAMPLE, REPRTB, BINNED, BYRUN, CONSFL, DIAG,
     1    EXPERT, PRTINP, NOCF, SUBCL, CDB, TMPCDB, BRKOPN, VB, ECF, ICF
      COMMON /SRCTRM/ ST(MAXSPC), STE(MAXSPC), STCCI(MAXSPC),
     1                STL(MAXSPC), STIL, STRVOL(MAXSPC),
     2                ST1(MAXSPC), ST2(MAXSPC), RV(MAXSPC)
      COMMON /BASVAL/ FCOR(MAXSPC), FVES(MAXSPC), DFVPA(MAXSPC),
     1                DFCPA(MAXSPC), FEVSE(MAXSPC), FDCH(MAXSPC),
     2                FCCI(MAXSPC), DFCAV(MAXSPC), VBPUF(MAXSPC),
     3                FCONV(MAXSPC), FCONC(MAXSPC), RBDF(MAXSPC),
     4                DFSPRV(MAXSPC), DFSPRC(MAXSPC), FREVO(MAXSPC),
     5                VALISS(MAXISS), FLTI1, FLTI2, NSPEC, FLV, FHPE,
     6                EVSE, WFAC, PFAC, FPLBYE, FPLBYP, FPLBYD,
     7                FPLBYC, FTLPH, FTLPL, FTLP, TC11, TC12, TB11,
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     8                 TB12, TB21, TB22, TBS1, TBS2, TBR1, TBR2, TW,
     9                 T1, T2, DT1, DT2, DTCDB, ELEV, PUFF, HVSPLT, FCD
      COMMON /BININD/ INDX(MAXBD)
      COMMON /CONTRL/ NLHS, NOBS, NSTART, NBIN, NDM, NTOT
C*****NEW VARIABLES HAVE BEEN ADDED TO THE FOLLOWING DIMENSION
C*****STATEMENT TO ACCOUNT FOR THE RELEASE FRACTIONS TO CONTAINMENT
C*****AND TO THE AUX BLDG FOR EACH RELEASE SPECIES
C*****(LANNY SMITH, 15 APR 92)
      DIMENSION RFDCH(MAXSPC), RFCCI(MAXSPC),
     1          RFBVB(MAXSPC), RFEVSE(MAXSPC),
     2          STC(MAXSPC), STA(MAXSPC),
     3          STEC(MAXSPC), STEA(MAXSPC),
     4          STLC(MAXSPC), STLA(MAXSPC),
     5          STCCIC(MAXSPC), STCCIA(MAXSPC),
     6          STRVLC(MAXSPC), STRVLA(MAXSPC)
C///    DIMENSION  ST1C(MAXSPC), ST1A(MAXSPC),
C///    1          ST2C(MAXSPC), ST2A(MAXSPC)
C
C
C*****ZERO OUT THE SOURCE TERM ARRAYS
C*****NEW VARIABLES HAVE BEEN ADDED TO THE FOLLOWING DO LOOP
C*****LOGIC TO ACCOUNT FOR THE RELEASE FRACTIONS TO CONTAINMENT
C*****AND TO THE AUX BLDG FOR EACH RELEASE SPECIES
C*****(LANNY SMITH, 15 APR 92)
      DO 1000 ISP=1,NSPEC
         RFBVB(ISP)=0.0
         RFEVSE(ISP)=0.0
         RFDCH(ISP)=0.0
         RFCCI(ISP)=0.0
         STC(ISP)=0.0
         STA(ISP)=0.0
         ST(ISP)=0.0
         STEC(ISP)=0.0
         STEA(ISP)=0.0
         STE(ISP)=0.0
         STCCIC(ISP)=0.0
         STCCIA(ISP)=0.0
         STCCI(ISP)=0.0
         STRVLC(ISP)=0.0
         STRVLA(ISP)=0.0
         STRVOL(ISP)=0.0
         STLC(ISP)=0.0
         STLA(ISP)=0.0
         STL(ISP)=0.0
C///        ST1C(ISP)=0.0
C///        ST1A(ISP)=0.0
         ST1(ISP)=0.0
C///        ST2C(ISP)=0.0
C///        ST2A(ISP)=0.0
         ST2(ISP)=0.0
 1000 CONTINUE
      POOLI=0.0
      CAVWI=0.0
      STIL=0.0
C*****SAVE I, CS, AND TE IN VESSEL FOR REVOLATILIZATION IN LATE RELEASES
      DO 1200 ISP=2,4
         RV(ISP)=FCOR(ISP) * (1.0-FVES(ISP))
 1200 CONTINUE
C*****RELEASE FROM VESSEL PRIOR TO VESSEL BREACH
      DO 2000 ISP=1,NSPEC
C********RELEASE FRACTION THRU TAIL PIPE THAT BYPASSES POOL
         RELF1=FTLP * FPLBYE / DFSPRV(ISP)
C********RELEASE FRACTION THRU TAIL PIPE THAT GOES THRU POOL
         RELF2=FTLP * (1.0-FPLBYE) / MAX (DFCPA(ISP), DFSPRV(ISP))
C********RELEASE FRACTION THRU T-QUENCHER
         RELF3=(1.0-FTLP) / MAX (DFVPA(ISP), DFSPRV(ISP))
C********EARLY RELEASE FRACTION (INCLUDE HEAD VENT RELEASE)
         STE(ISP)=(FCOR(ISP)*FVES(ISP)*(1.0-HVSPLT)*(RELF1+RELF2+RELF3)+
     1            FCOR(ISP)*FVES(ISP)*HVSPLT/DFCAV(ISP)) *
     2            FCONV(ISP) / RBDF(ISP)
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C*********NEW VARIABLES STEC(ISP) AND STEA(ISP) HAVE BEEN
C*********ADDED TO ACCOUNT FOR THE RELEASE FRACTIONS TO CONTAINMENT
C*********AND TO THE AUX BLDG FOR EACH RELEASE SPECIES
C*********(LANNY SMITH, 15 APR 92)
C*********STEC(ISP)=(STE(ISP) / FCONV(ISP)) * RBDF(ISP)
          STEC(ISP)=(STE(ISP) / 1.0) * RBDF(ISP)
          STEA(ISP)=STE(ISP) * RBDF(ISP)
          RFBVB(ISP)=STE(ISP)
C*********SAVE IODINE IN POOL AND IN CAVITY
          IF (ISP .EQ. 2) THEN
             POOLI=FCOR(ISP) * FVES(ISP) *
     1             MAX (0.0, (1.0-RELF1-RELF2-RELF3))
             CAVWI=FCOR(ISP) * FVES(ISP) * HVSPLT *
     1              (1.0 - 1.0/DFCAV(ISP))
          ENDIF
 2000 CONTINUE
      IF (DIAG) THEN
C*********DIAGNOSTIC PRINT
          WRITE(6,2001)
          WRITE(6,4202)  (STE(ISP),ISP=1,NSPEC)
          WRITE(6,4203)  (STL(ISP),ISP=1,NSPEC)
          WRITE(6,4204)  (ST(ISP),ISP=1,NSPEC)
          WRITE(6,4205)  (RFBVB(ISP),ISP=1,NSPEC)
          WRITE(6,4206)  (RFEVSE(ISP),ISP=1,NSPEC)
          WRITE(6,4207)  (RFDCH(ISP),ISP=1,NSPEC)
          WRITE(6,4208)  (RFCCI(ISP),ISP=1,NSPEC)
          WRITE(6,4209)  (STCCI(ISP),ISP=1,NSPEC)
          WRITE(6,4210)  (RV(I),I=2,4), (STRVOL(I),I=2,4), POOLI,
     1                   CAVWI, STIL
      ENDIF
C*****ADD FOR LP&S POS-5 ANALYSIS
C*****IF CNMT CLOSE WITH ILOCA OR OPEN MSIV AND LATE CNMT FAILURE, SET
C
C*****RBDF FOR RELEASE AT OR AFTER VB TO 1.0
      IF(((INDX(2).EQ.2) .OR. (INDX(2).EQ. 3) .OR. (INDX(2).EQ. 5))
     1    .AND. (INDX(5).EQ.3)) THEN
        DO 2099 ISP=1,9
         RBDF(ISP) = 1.0
 2099   CONTINUE
      ENDIF
C*****
C*****IF EVSE, STEAM EXPLOSIONS ARE CONSIDERED
      IF ((INDX(11) .LT. 3) .OR. (INDX(11) .EQ. 5) ) GO TO 7250
C*****ADD EX-VESSEL STEAM EXPLOSION
      DO 4500 ISP=1,NSPEC
C*********RELEASE FRACTION DUE TO EX-VESSEL STEAM EXPLOSION
          RFEVSE(ISP)=MAX (0.0, (1.0-FCOR(ISP)-VBPUF(ISP))) * FLV *
     1                EVSE * FEVSE(ISP)
          IF (RFEVSE(ISP) .GT. 0.0) THEN
C************RELEASE FRACTION DUE TO EX-VESSEL STEAM EXPLOSION, THAT
C************BYPASSES POOL
             RELF1=FPLBYD / DFSPRC(ISP)
C************RELEASE FRACTION DUE TO EX-VESSEL STEAM EXPLOSION THAT
C************GOES THRU POOL
             RELF2=(1.0-FPLBYD) / MAX (DFCPA(ISP), DFSPRC(ISP))
C************EARLY RELEASE FRACTION
C************NEW VARIABLES 'STEC(ISP)' AND 'STEA(ISP)' HAVE BEEN
C************ADDED TO ACCOUNT FOR THE RELEASE FRACTIONS TO CONTAINMENT
C************AND TO THE AUX BLDG FOR EACH RELEASE SPECIES
C************(LANNY SMITH, 15 APR 92)
             STEC(ISP)=MIN (1.0, STEC(ISP) + RFEVSE(ISP)*(RELF1+RELF2))
             STEA(ISP)=MIN (1.0, STEA(ISP) +
     1                     RFEVSE(ISP)*(RELF1+RELF2)*FCONC(ISP))
             STE(ISP)=MIN (1.0, STE(ISP) +
     1                     RFEVSE(ISP)*(RELF1+RELF2)*FCONC(ISP)/RBDF(ISP))
C************SAVE IODINE IN POOL
             IF (ISP .EQ. 2) THEN
                POOLI=POOLI + RFEVSE(ISP)*MAX (0.0, (1.0-RELF1-RELF2))
             ENDIF
          ENDIF
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 4500 CONTINUE
      IF (DIAG) THEN
C*********DIAGNOSTIC PRINT
         WRITE(6,4501)
         WRITE(6,4202)  (STE(ISP),ISP=1,NSPEC)
         WRITE(6,4203)  (STL(ISP),ISP=1,NSPEC)
         WRITE(6,4204)  (ST(ISP),ISP=1,NSPEC)
         WRITE(6,4205)  (RFBVB(ISP),ISP=1,NSPEC)
         WRITE(6,4206)  (RFEVSE(ISP),ISP=1,NSPEC)
         WRITE(6,4207)  (RFDCH(ISP),ISP=1,NSPEC)
         WRITE(6,4208)  (RFCCI(ISP),ISP=1,NSPEC)
         WRITE(6,4209)  (STCCI(ISP),ISP=1,NSPEC)
         WRITE(6,4210)  (RV(I),I=2,4), (STRVOL(I),I=2,4), POOLI,
     1                  CAVWI, STIL
      ENDIF
 7250 CONTINUE
C*****IF NO VB, THEN NO PUFF, NO CCI, NO DCH SOURCE TERMS
      IF (.NOT. VB) GO TO 7500
C*****ADD VESSEL BREACH PUFF RELEASE TO EARLY SOURCE TERM
      DO 3000 ISP=1,NSPEC
C*********RELEASE FRACTION DUE TO VESSEL BREACH PUFF THAT BYPASSES POOL
         RELF1=FPLBYP / DFSPRC(ISP)
C*********RELEASE FRACTION DUE TO VESSEL BREACH PUFF THAT GOES THRU POOL
         RELF2=(1.0-FPLBYP) / MAX (DFCPA(ISP), DFSPRC(ISP))
C*********EARLY RELEASE FRACTION
C*********NEW VARIABLES STEC(ISP) AND STEA(ISP) HAVE BEEN
C*********ADDED TO ACCOUNT FOR THE RELEASE FRACTIONS TO CONTAINMENT
C*********AND TO THE AUX BLDG FOR EACH RELEASE SPECIES
C*********(LANNY SMITH, 15 APR 92)
         STEC(ISP)=MIN (1.0, STEC(ISP) + VBPUF(ISP)*(RELF1+RELF2) )
         STEA(ISP)=MIN (1.0, STEA(ISP) +
     1                  VBPUF(ISP)*(RELF1+RELF2)*FCONC(ISP))
         STE(ISP)=MIN (1.0, STE(ISP) +
     1                 VBPUF(ISP)*(RELF1+RELF2)*FCONC(ISP)/RBDF(ISP))
C*********SAVE IODINE IN POOL
         IF (ISP .EQ. 2) THEN
            POOLI=POOLI + VBPUF(ISP)*MAX (0.0, (1.0-RELF1-RELF2))
         ENDIF
 3000 CONTINUE
      IF (DIAG) THEN
C*********DIAGNOSTIC PRINT
         WRITE(6,3001)
         WRITE(6,4202)  (STE(ISP),ISP=1,NSPEC)
         WRITE(6,4203)  (STL(ISP),ISP=1,NSPEC)
         WRITE(6,4204)  (ST(ISP),ISP=1,NSPEC)
         WRITE(6,4205)  (RFBVB(ISP),ISP=1,NSPEC)
         WRITE(6,4206)  (RFEVSE(ISP),ISP=1,NSPEC)
         WRITE(6,4207)  (RFDCH(ISP),ISP=1,NSPEC)
         WRITE(6,4208)  (RFCCI(ISP),ISP=1,NSPEC)
         WRITE(6,4209)  (STCCI(ISP),ISP=1,NSPEC)
         WRITE(6,4210)  (RV(I),I=2,4), (STRVOL(I),I=2,4), POOLI,
     1                  CAVWI, STIL
      ENDIF
C*****ADD DIRECT CONTAINMENT HEATING RELEASE TO EARLY SOURCE TERM
      DO 4000 ISP=1,NSPEC
C*********RELEASE FRACTION DUE TO DIRECT CONTAINMENT HEATING
         RFDCH(ISP)=MAX (0.0, (1.0-FCOR(ISP)-VBPUF(ISP))) * FLV *
     1              FHPE * FDCH(ISP)
         IF (RFDCH(ISP) .GT. 0.0) THEN
C************RELEASE FRACTION DUE TO DIRECT CONTAINMENT HEATING THAT
C************BYPASSES POOL
            RELF1=FPLBYD / DFSPRC(ISP)
C************RELEASE FRACTION DUE TO DIRECT CONTAINMENT HEATING THAT
C************GOES THRU POOL
            RELF2=(1.0-FPLBYD) / MAX (DFCPA(ISP), DFSPRC(ISP))
C************EARLY RELEASE FRACTION
C************NEW VARIABLES STEC(ISP) AND STEA(ISP) HAVE BEEN
C************ADDED TO ACCOUNT FOR THE RELEASE FRACTIONS TO CONTAINMENT
C************AND TO THE AUX BLDG FOR EACH RELEASE SPECIES
C************(LANNY SMITH, 15 APR 92)
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          STEC(ISP)=MIN (1.0, STEC(ISP) + RFDCH(ISP)*(RELF1+RELF2))
          STEA(ISP)=MIN (1.0, STEA(ISP) +
     1                RFDCH(ISP)*(RELF1+RELF2)*FCONC(ISP))
          STE(ISP)=MIN (1.0, STE(ISP) +
     1                RFDCH(ISP)*(RELF1+RELF2)*FCONC(ISP)/RBDF(ISP))
C***********SAVE IODINE IN POOL
          IF (ISP .EQ. 2) THEN
             POOLI=POOLI + RFDCH(ISP)*MAX (0.0, (1.0-RELF1-RELF2))
          ENDIF
       ENDIF
 4000 CONTINUE
      IF (DIAG) THEN
C********DIAGNOSTIC PRINT
         WRITE(6,4201)
         WRITE(6,4202) (STE(ISP),ISP=1,NSPEC)
         WRITE(6,4203) (STL(ISP),ISP=1,NSPEC)
         WRITE(6,4204) (ST(ISP),ISP=1,NSPEC)
         WRITE(6,4205) (RFBVB(ISP),ISP=1,NSPEC)
         WRITE(6,4206) (RFEVSE(ISP),ISP=1,NSPEC)
         WRITE(6,4207) (RFDCH(ISP),ISP=1,NSPEC)
         WRITE(6,4208) (RFCCI(ISP),ISP=1,NSPEC)
         WRITE(6,4209) (STCCI(ISP),ISP=1,NSPEC)
         WRITE(6,4210) (RV(I),I=2,4), (STRVOL(I),I=2,4), POOLI,
     1                 CAVWI, STIL
      ENDIF
C*****EX-VESSEL STEAM EXPLOSION
      IF (EVSE .GT. 0.0) THEN
         XCCI=1.0 - EVSE
      ELSE IF (FHPE .GT. 0.0) THEN
         XCCI=1.0 - FHPE
      ELSE
         XCCI=1.0
      ENDIF
      IF (.NOT. CDB) THEN
C********CORE-CONCRETE INTERACTION RELEASES AND CAVITY SCRUBBING
         DO 5000 ISP=1,NSPEC
C***********RELEASE FRACTION DUE TO CORE-CONCRETE INTERACTIONS
            RFCCI(ISP)=MAX (0.0, (1.0-FCOR(ISP)-VBPUF(ISP))) * FLV *
     1                 XCCI * FCCI(ISP)
C***********RELEASE FRACTION DUE TO CORE-CONCRETE INTERACTIONS THAT
C***********BYPASSES POOL
            RELF1=FPLBYC / MAX (DFCAV(ISP), DFSPRC(ISP))
C***********RELEASE FRACTION DUE TO CORE-CONCRETE INTERACTIONS THAT
C***********GOES THRU POOL
            RELF2=(1.0-FPLBYC) /
     1            MAX (DFCAV(ISP), DFCPA(ISP), DFSPRC(ISP))
C***********CORE-CONCRETE RELEASE FRACTION
            STCCI(ISP)=RFCCI(ISP) * (RELF1+RELF2) *
     1                 FCONC(ISP) / RBDF(ISP)
C***********NEW VARIABLES STCCIC(ISP) AND STCCIA(ISP) HAVE
C***********BEEN ADDED TO ACCOUNT FOR THE RELEASE FRACTIONS TO THE
C***********CONTAINMENT AND TO THE AUX BLDG FOR EACH RELEASE SPECIES
C***********(LANNY SMITH, 15 APR 92)
            IF (FCONC(ISP) .GT. 0.0) THEN
               STCCIC(ISP)= STCCI(ISP) / FCONC(ISP) * RBDF(ISP)
            ELSE
               STCCIC(ISP)=0.0
            ENDIF
            STCCIA(ISP)= STCCI(ISP) * RBDF(ISP)
C***********SAVE IODINE IN CAVITY WATER AND IN POOL
            IF (ISP .EQ. 2) THEN
               CAVWI2=1.0 - 1.0/DFCAV(ISP)
               POOLI=POOLI + RFCCI(ISP) *
     1               MAX (0.0, (1.0-RELF1-RELF2-CAVWI2))
               CAVWI=CAVWI + RFCCI(ISP) * CAVWI2
            ENDIF
 5000    CONTINUE
      ENDIF
C*****REVOLATIZATION RELEASE OF I, CS, AND TE
C*****(SIMILAR TO VESSEL BREACH PUFF RELEASE)
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      DO 6000 ISP=2,4
C*********RELEASE FRACTION DUE TO REVOLATILIZATION THAT BYPASSES POOL
         RELF1=FPLBYC / DFSPRC(ISP)
C*********RELEASE FRACTION DUE TO REVOLATILIZATION THAT GOES THRU POOL
         RELF2=(1.0-FPLBYC) / MAX (DFCPA(ISP), DFSPRC(ISP))
C*********REVOLATILIZATION RELEASE FRACTION
         STRVOL(ISP)=FREVO(ISP) * RV(ISP) * (RELF1+RELF2) *
     1               FCONC(ISP) / RBDF(ISP)
C*********NEW VARIABLES STRVLC(ISP) AND STRVLA(ISP) HAVE BEEN
C*********ADDED TO ACCOUNT FOR THE RELEASE FRACTIONS TO CONTAINMENT
C*********AND TO THE AUX BLDG FOR EACH RELEASE SPECIES
C*********(LANNY SMITH, 15 APR 92)
         IF (FCONC(ISP) .GT. 0.0) THEN
            STRVLC(ISP)= STRVOL(ISP) / FCONC(ISP) * RBDF(ISP)
         ELSE
            STRVLC(ISP)=0.0
         ENDIF
         STRVLA(ISP)= STRVOL(ISP) * RBDF(ISP)
C*********SAVE IODINE IN POOL
         IF (ISP .EQ. 2) THEN
            POOLI=POOLI + FREVO(ISP)*RV(ISP)*
     1            MAX (0.0, (1.0-RELF1-RELF2))
         ENDIF
 6000 CONTINUE
C*****CCI, RCS REVOLATILIZATION WERE SKIPPED IF VESSEL BREACH WAS PREVENTED,
C*****BUT LATE IODINE RELEASE FROM THE POOL CAN STILL OCCUR
 7500 CONTINUE
C*****CALCULATE THE IODINE REVOLATILIZED FROM THE POOL,
C*****WHICH IS NOT SUBJECT TO ANY DF OR CONTAINMENT RETENTION IF
C*****CONTAINMENT FAILS. HOWEVER, IF NO CONTAINMENT FAILURE, ASSUME
C*****ONLY SMALL FRACTION RELEASED TO ENVIRONMENT.
C*****FOR LATE IODINE RELEASE FROM CAVITY WATER, POOL BYPASS FRACTION
C*****APPLIES. POOL DF OF IODINE APPLIES TO FRACTION GO THROUGH POOL
      STIL1=FLTI1 * POOLI
      STIL2=FLTI2 * CAVWI * (FPLBYC+(1.0-FPLBYC)/DFCPA(2))
      STIL=STIL1 + STIL2
C*****IF NO CONTAINMENT FAILURE, LATE IODINE RELEASE IS TREATED SIMILAR
C*****TO NOBLE GASES SINCE IODINE IS VOLATILE
      IF (NOCF) STIL=STIL * FCONC(1)
C*****ADD ALL SOURCE TERMS UP TO GET TOTAL SOURCE TERMS
      DO 8000 ISP=1,NSPEC
C*********NEW VARIABLES STLC(ISP), STLA(ISP), STC(ISP), AND
C*********STA(ISP) HAVE BEEN ADDED TO ACCOUNT FOR THE RELEASE
C*********FRACTIONS TO CONTAINMENT AND TO THE AUX BLDG FOR EACH
C*********RELEASE SPECIES
C*********(LANNY SMITH, 15 APR 92)
         STLC(ISP)=STCCIC(ISP) + STRVLC(ISP)
         STLA(ISP)=STCCIA(ISP) + STRVLA(ISP)
         STL(ISP)=STCCI(ISP) + STRVOL(ISP)
         STC(ISP)=STEC(ISP) + STLC(ISP)
         STA(ISP)=STEA(ISP) + STLA(ISP)
         ST(ISP)=STE(ISP) + STL(ISP)
8000  CONTINUE
      STC(2)=STC(2) + STIL
      STA(2)=STA(2) + STIL
      ST(2)=ST(2) + STIL
      STLC(2)=STLC(2) + STIL
      STLA(2)=STLA(2) + STIL
      STL(2)=STL(2) + STIL
C*****REALLOCATE RELEASE FRACTIONS
      DO 9000 ISP=1,NSPEC
         IF (ECF) THEN
C************CF BEFORE VB
C************NEW VARIABLES ST1C(ISP), ST1A(ISP), ST2C(ISP), AND
C************ST2A(ISP) HAVE BEEN ADDED TO ACCOUNT FOR THE RELEASE
C************FRACTIONS TO CONTAINMENT AND TO THE AUX BLDG FOR EACH
C************RELEASE SPECIES
C************(LANNY SMITH, 15 APR 92)
C///           ST1C(ISP)=STEC(ISP)
C///           ST1A(ISP)=STEA(ISP)
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            ST1(ISP)=STE(ISP)
C///            ST2C(ISP)=STLC(ISP)
C///            ST2A(ISP)=STLA(ISP)
            ST2(ISP)=STL(ISP)
         ELSE IF (ICF) THEN
C***********CF AT VB
C///            ST1C(ISP)=STEC(ISP)
C///            ST1A(ISP)=STEA(ISP)
            ST1(ISP)=STE(ISP)
C///            ST2C(ISP)=STLC(ISP)
C///            ST2A(ISP)=STLA(ISP)
            ST2(ISP)=STL(ISP)
         ELSE
C***********LATE LEAK OR RUPTURE OR NO CONTAINMENT FAILURE
            STEC(ISP)=0.0
            STEA(ISP)=0.0
            STE(ISP)=0.0
            STLC(ISP)=ST(ISP)
            STLA(ISP)=ST(ISP)
            STL(ISP)=ST(ISP)
C///            ST1C(ISP)=PUFF * STC(ISP)
C///            ST1A(ISP)=PUFF * STA(ISP)
            ST1(ISP)=PUFF * ST(ISP)
C///            ST2C(ISP)=(1.0-PUFF) * STC(ISP)
C///            ST2A(ISP)=(1.0-PUFF) * STA(ISP)
            ST2(ISP)=(1.0-PUFF) * ST(ISP)
         ENDIF
 9000 CONTINUE
C*****CALCULATE ENERGY RELEASES
      CALL ENERGY (E1, E2)
C*****CALCULATE ENERGY RELEASE RATES
      IF (DT1 .GT. 0.0) THEN
         ER1=E1 / DT1
      ELSE
         ER1=0.0
      ENDIF
      IF (DT2 .GT. 0.0) THEN
        ER2=E2 / DT2
      ELSE
        ER2=0.0
      END IF
      IF (DIAG) THEN
         WRITE(6,8001)
         WRITE(6,4202) (STE(ISP),ISP=1,NSPEC)
         WRITE(6,4203) (STL(ISP),ISP=1,NSPEC)
         WRITE(6,4204) (ST(ISP),ISP=1,NSPEC)
         WRITE(6,4205) (RFBVB(ISP),ISP=1,NSPEC)
         WRITE(6,4206) (RFEVSE(ISP),ISP=1,NSPEC)
         WRITE(6,4207) (RFDCH(ISP),ISP=1,NSPEC)
         WRITE(6,4208) (RFCCI(ISP),ISP=1,NSPEC)
         WRITE(6,4209) (STCCI(ISP),ISP=1,NSPEC)
         WRITE(6,4210) (RV(I),I=2,4), (STRVOL(I),I=2,4), POOLI,
     1                 CAVWI, STIL
         WRITE(6,4211) TW, T1, DT1, T2, DT2, ELEV, ER1, FR2
         WRITE(6,4212) (ST1(ISP),ISP=1,NSPEC)
         WRITE(6,4213) (ST2(ISP),ISP=1,NSPEC)
      ENDIF
      IF (CONSFL) THEN
C*********WRITE SOURCE TERM TO FILE
         WRITE(9,1003) IOBS, BINARR(IBIN)(1:NDM)
         WRITE(9,1004) TW, T1, DT1, T2, DT2, ELEV, FLOAT(INDX(13))
         WRITE(9,1004) ER1, (ST1(ISP),ISP=1,NSPEC)
         WRITE(9,1004) ER2, (ST2(ISP),ISP=1,NSPEC)
      ENDIF
C*****NEW WRITE STATEMENTS HAVE BEEN ADDED TO ACCOUNT FOR THE
C*****RELEASE FRACTIONS TO CONTAINMENT, THE AUX BLDG, AND THE
C*****ENVIRONMENT FOR EACH RELEASE SPECIES
C*****(LANNY SMITH, 15 APR 92)
      IF (CONSFL) THEN
C*********WRITE CONTAIMENT AND AUX BLDG SOURCE TERMS FOR EARLY AND
```

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C*********LATE SEGMENTS
C///          WRITE(10,1004)  (ST1C(ISP),ISP=1,NSPEC)
C///          WRITE(10,1004)  (ST1A(ISP),ISP=1,NSPEC)
C///          WRITE(10,1004)  (ST2C(ISP),ISP=1,NSPEC)
C///          WRITE(10,1004)  (ST2A(ISP),ISP=1,NSPEC)
      ENDIF
      RETURN
C*****FORMAT STATEMENTS
 1003 FORMAT(I4,2X,A)
 1004 FORMAT(1P10E9.2)
 2001 FORMAT(//5X,'***** DIAGNOSTIC PRINT *****',
     1        /10X,'===== PARAMETER VALUES UP TO VESSEL BREACH =====')
 3001 FORMAT(//5X,'***** DIAGNOSTIC PRINT *****',
     1        /10X,'===== PARAMETER VALUES AFTER VESSEL BREACH =====')
 4201 FORMAT(//5X,'***** DIAGNOSTIC PRINT *****',
     1        /10X,'===== PARAMETER VALUES AFTER DCH =====')
 4202 FORMAT(/5X,'STE:',1P,/(5X,10E10.2))
 4203 FORMAT(5X,'STL:',1P,/(5X,10E10.2))
 4204 FORMAT(5X,'ST :',1P,/(5X,10E10.2))
 4205 FORMAT(5X,'RFBVB:',1P,/(5X,10E10.2))
 4206 FORMAT(5X,'RFEVSE:',1P,/(5X,10E10.2))
 4207 FORMAT(5X,'RFDCH:',1P,/(5X,10E10.2))
 4208 FORMAT(5X,'RFCCI:',1P,/(5X,10E10.2))
 4209 FORMAT(5X,'STCCI:',1P,/(5X,10E10.2))
 4210 FORMAT(5X,1P,'RVI = ',E10.2,5X,'RVCS = ',E10.2,5X,'RVTE = ',E10.2,
     1        /5X,'STRVOL(2) = ',E10.2,5X,'STRVOL(3) = ',E10.2,
     2         5X,'STRVOL(4) = ',E10.2,
     3        /5X,'POOLI = ',E10.2,5X,'CAVWI = ',E10.2,
     4         5X,'STIL = ',E10.2)
 4211 FORMAT(/5X,'SOURCE TERM INFORMATION:',
     1        /5X,1P,'TW =',E10.2,5X,'T1 =',E10.2,5X,'DT1 =',E10.2,5X,
     2         'T2 =',E10.2,5X,'DT2 =',E10.2,
     3        /5X,'ELEV =',E10.2,5X,'ER1 =',E10.2,5X,'ER2 =',E10.2)
 4212 FORMAT(5X,'ST1:',1P,/(5X,10E10.2))
 4213 FORMAT(5X,'ST2:',1P,/(5X,10E10.2))
 4501 FORMAT(//5X,'***** DIAGNOSTIC PRINT *****',
     1        /10X,'===== PARAMETER VALUES AFTER EVSE =====')
 5002 FORMAT(1X,'OBS: ',I4,2X,'BIN:  ',A,/)
 5003 FORMAT(1X,'T1 =',1PE12.4,'  DT1 =',1PE12.4,'  T2 =',1PE12.4,'
     1 DT2 =',1PE12.4,/)
 5004 FORMAT(1X,'EARLY CONT',1P9E12.4)
 5005 FORMAT(1X,'EARLY AUX ',1P9E12.4)
 5006 FORMAT(1X,'EARLY ENV ',1P9E12.4)
 5007 FORMAT(1X,'LATE  CONT',1P9E12.4)
 5008 FORMAT(1X,'LATE  AUX ',1P9E12.4)
 5009 FORMAT(1X,'LATE  ENV ',1P9E12.4,/)
 8001 FORMAT(//5X,'***** DIAGNOSTIC PRINT *****',
     1        /10X,'===== PARAMETER VALUES AT END OF GGSORC =====')
      END
      SUBROUTINE ENERGY (EARLY, TAIL)
C*****ESTIMATE ENERGY RELEASES FOR BOTH EARLY PUFF AND LATE
C*****TAIL. DATA BASE ARE TAKEN FROM RESULTS OF MELCOR CALCULATIONS
C*****FOR GRAND GULF (ENERGIES ARE IN JOULES).
C*****EARLY: CALCULATED PUFF ENERGY RELEASE (JOULES)
C*****TAIL: CALCULATED ENERGY RELEASE AFTER PUFF (JOULES)
C*****RLATCF: CORRECTION FACTOR FOR LATE CONTAINMENT FAILURE,
C*****          NOT USED FOR POS 5
C*****SPRFAC: CONTAINMENT SPRAY FACTOR FOR BOTH EARLY AND TAIL
C*****          NOT USED FOR POS 5
      PARAMETER (MAXBD=20, MAXBIN=10000, MAXSMP=300, MAXCAS=8,
     1           MAXISS=20, MAXLEV=10, MAXVAR=100, MAXVAL=13000,
     2           MAXSPC=10, MAXTIM=20)
      COMMON /BASVAL/ FCOR(MAXSPC), FVES(MAXSPC), DFVPA(MAXSPC),
     1                DFCPA(MAXSPC), FEVSE(MAXSPC), FDCH(MAXSPC),
     2                FCCI(MAXSPC), DFCAV(MAXSPC), VBPUF(MAXSPC),
     3                FCONV(MAXSPC), FCONC(MAXSPC), RBDF(MAXSPC),
     4                DFSPRV(MAXSPC), DFSPRC(MAXSPC), FREVO(MAXSPC),
     5                VALISS(MAXISS), FLTI1, FLTI2, NSPEC, FLV, FHPE,
     6                FVSE, WFAC, PFAC, FPLBYE, FPLBYP, FPLBYD,
     7                FPLBYC, FTLPH, FTLPL, FTLP, TC11, TC12, TB11,
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     8                 TB12, TB21, TB22, TBS1, TBS2, TBR1, TBR2, TW,
     9                 T1, T2, DT1, DT2, DTCDB, ELEV, PUFF, HVSPLT, FCD
      COMMON /BININD/ INDX(MAXBD)
C
C
      RLATCF=1.0
      SPRFAC=1.0
      SPRAYV=1.0
      SPRAYC=1.0
C*****IF CONTAINMENT DOES NOT FAIL, BYPASS CALCULATION
      IF (INDX(3) .NE. 4) THEN
C*** BASE CASE ENERGIES-LATER CASES WILL OVER WRITE IF NECESSARY
         EARLY = TC12
         TAIL  = TB21
C*** CNMT FLOODED SCENARIOS
        IF ((INDX(1) .EQ. 1) .OR. (INDX(1) .EQ. 5) .OR.
     +       (INDX(1) .EQ. 6) .OR. (INDX(1) .EQ. 9) .OR.
     +       (INDX(1) .EQ. 12))THEN
            IF (INDX(3) .EQ. 3) THEN
               EARLY = TC11
               TAIL  = TB11
            ELSE
               EARLY = TB11
               TAIL  = TB21
            ENDIF
        ENDIF
C*** SBO SCENARIOS
        IF ((INDX(1) .EQ. 2) .OR. (INDX(1) .EQ. 3) .OR.
     +       (INDX(1) .EQ. 4) .OR. (INDX(1) .EQ. 7) .OR.
     +       (INDX(1) .EQ. 8)) THEN
            EARLY = TC12
            TAIL  = TB21
        ENDIF
C*** OPEN MSIV SCENARIO
        IF (INDX(1) .EQ. 11) THEN
           EARLY = TB21
           EARLY = TB21
        ENDIF
C*** CNMT CLOSED
      ELSE
C*** CNMT RUPTURED OR VENTED
        IF ((INDX(2) .EQ. 2) .OR. (INDX(2) .EQ. 3) .OR.
     +       (INDX(2) .EQ. 5) .OR. (INDX(2) .EQ. 7) .OR.
     +       (INDX(2) .EQ. 8)) THEN
            EARLY = TB11
            TAIL  = TB21
C*** CNMT DOES NOT FAIL
        ELSEIF (INDX(2) .EQ. 10) THEN
            EARLY = 0.0
            TAIL  = 0.0
C*** CNMT LEAKS
        ELSE
            EARLY = TC12
            TAIL  = TB21
        ENDIF
      ENDIF
C*** NO VESSEL FAILURE
      IF ((INDX(8) .EQ. 5) .OR. (INDX(8) .EQ. 6)) THEN
         TAIL = 0.0
      ENDIF
C*****CONVERT BTU TO JOULES
C       EARLY=1055. * EARLY
C       TAIL=1055. * TAIL
      RETURN
      END
```

C.2 Input Data for GGSORP5

The following is the input file to GGSORP5 that provides the values used to quantify the parameters in the XSOR expression.

```
$  GGSOR POS5 DATA BASE : 15 APRIL 94 (BASED ON GGSOR NUREG-1150 DATA BASE)
$*******************************************************************
$    ENERGY RELEASE PARAMETERS BASED ON MELCOR VALUES (J)
$    FOR USE IN LP&S POS 5 STUDY
TC11  5.00E+8         $ AUX. BLDG FAILURE AT VB
TC12  1.50E+9         $ EARLY RELEASE FOR SBO  (OPEN CNMT)
TB11  5.00E+9         $ AUX. BLDG FAILS FROM BRN OR CNMT RUPT
TB12  1.00E+10        $ NOT USED
TB21  3.00E+10        $ FIRST RELEASE FOR OPEN MSIV OR SECOND REL. W/ BRN
TB22  0.00E+0         $ NOT USED
TBS1  0.00E+0         $ NOT USED
TBS2  0.00E+0         $ NOT USED
TBR1  0.00E+0         $ NOT USED
TBR2  0.00E+0         $ NOT USED
$    ENERGY RELEASE PARAMETERS ADDED FOR GGSOR POS5
E0(1) 1.0E8           $ NOT USED
E0(2) 1.0E7           $ NOT USED
E0(3) 1.0E6           $ NOT USED
E0(4) 1.0E5           $ NOT USED
$     WARNING TIME (S)  Fuel Heatup or CNMT Failure
TW0(1)    5600. $ PDS1-1: LOCA
TW0(2)   13500. $ PDS1-2 & PDS1-4: SBO w/ SDC Break
TW0(3)   56000. $ PDS1-3: SBO w/ FW
TW0(4)   25600. $ PDS1-5: Flooded CNMT
TW0(5)    6600. $ PDS2-1: LOCA
TW0(6)   20000. $ PDS2-2 & PDS2-3: SBO w/SDC Break
TW0(7)   32100. $ PDS2-4: Flooded CNMT
TW0(8)   43500. $ PDS2-5: HiP, CNMT Equip. Hatch Open
TW0(9)   49857. $ PDS2-5: HiP, CNMT Fails during CD
TW0(10) 73712. $ PDS2-5: HiP, CNMT Fails at VB
TW0(11) 73712. $ PDS2-5: HiP, CNMT Vented or Fails Late
TW0(12) 30000. $ PDS2-6: Open MSIV
TW0(13)  8100. $ PDS3-1: LOCA
$     FIRST RELEASE TIME (S): Start of Gap Release or CNMT Failure
T10(1)    12900. $ PDS1-1: LOCA W/ AUX. BLDG FAILURE DURING CD
T10(2)    96100. $ PDS1-1: LOCA W/ AUX. BLDG FAILURE AT VB
T10(3)    15670. $ PDS1-2 & PDS1-3: SBO W/ SDC BREAK
T10(4)    63086. $ PDS1-4: SBO W/ FW
T10(5)    27055. $ PDS1-5: FLOODED CNMT W/ AUX. BLDG FAILURE DURING CD
T10(6)    82397. $ PDS1-5: FLOODED CNMT W/ AUX. BLDG FAILURE DURING VB
T10(7)    18200. $ PDS2-1: LOCA W/ AUX. BLDG FAILURE DURING CD
T10(8)   143700. $ PDS2-1: LOCA W/ AUX. BLDG FAILURE DURING VB
T10(9)    22820. $ PDS2-2 & PDS2-3: SBO W/ SDC BREAK
T10(10)   39860. $ PDS2-4: FLOODED CNMT W/ AUX. BLDG FAILURE DURING CD
T10(11) 113625. $ PDS2-4: FLOODED CNMT W/ AUX. BLDG FAILURE DURING VB
T10(12)   49857. $ PDS2-5: HiP W/ CNMT HATCH OPEN
T10(13)   49857. $ PDS2-5: HiP W/ CNMT FAILURE DURING CD
T10(14)   73712. $ PDS2-5: HiP W/ CNMT FAILURE AT VB
T10(15) 104400. $ PDS2-5: HiP W/ CNMT VENTING (OR LATE CNMT FAILURE)
T10(16)   35290. $ PDS2-6: OPEN MSIV
T10(17)   25700. $ PDS3-1: LOCA W/ AUX. BLDG FAILURE DURING CD
T10(18) 235400. $ PDS3-1: LOCA W/ AUX. BLDG FAILURE AT VB
$     RELEASE DURATION FOR FIRST RELEASE (S)
DT10(1)    180. $ CF RUPTURE, VENTING
DT10(2)   1800. $ AUX. BLDG FAILURE AT VB
DT10(3) 21600. $ SHORT DURATION (6 HRS)
DT10(4) 36000. $ MEDIUM DURATION (10 HRS)
DT10(5) 72000. $ LONG DURATION   (20 HRS)
$     RELEASE DURATION FOR SECOND RELEASE (S)
DT20(1) 86400. $ VB
DT20(2)      0. $ No VB
$     DELAY TIME FOR SECOND RELEASE (S) FOR TEMPORARY COOLABLE DEBRIS BED
DTCDB  10800.
$    FIRST RELEASE (PUFF) FRACTION FOR LATE CONTAINMENT FAILURE
PUFF0(1) 0.90  $ LATE CONTAINMENT FAILURE
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PUFF0(2) 0.50  $ LATE LEAK OR NO CONTAINMENT FAILURE
$    RELEASE ELEVATION (M)
ELEV  32.
$ ****************************************************************
$ FPLBY0: FRACTION OF POOL BYPASS HAS THREE CASES
FPLBY0(1)  0.0564  1.32  1.E+06
$ DRY CAVITY AND CONTAINMENT FAILURE CASES DERIVED FROM BMI-2139 GG STCP CALC
$   IF CAVITY IS WET, DIVIDED BY WFAC
$   IF LATE CF, MULTIPLIED BY PFAC
$ STEAMING CORRECTION FACTOR FOR FPLBY0 IF CAVITY IS NOT DRY
WFAC  3.1
$ PRESSURE CORRECTION FACTOR FOR FPLBY0 IF LATE CONTAINMENT FAILURE
PFAC  3.9
$ SPLIT FRACTION BETWEEN TAIL PIPE VACUUM BREAKER OPENING AND T-QUENCHER
$   HIGH PRESSURE SEQUENCES
FTLPH  0.39
$   LOW PRESSURE SEQUENCES
FTLPL  1.0
$ SPLIT FRACTION TO REACTOR HEAD VENT VS PIPE (ADDED FOR GGSOR POS5)
HVSPLT0  1.0
$ FRACTION CORE DAMAGE (ADDED FOR GGSOR POS5)
FCD  1.0
$**********************************************************
$  FHPE: FRACTION OF CORE PARTICIPATING IN DCH OR STEAM EXPLOSION
$       TWO CASES:  (1) HIGH, (2) LOW
FHPE0(1)  0.4  0.1
$ *********************************************************************
$  EVSE:  FRACTION OF CORE PARTICIPATING IN EX-VESSEL STEAM EXPLOSION
EVSE0(1)  0.2  0.05
$ *********************************************************************
$    PUFF RELEASE AT VESSEL BREACH: ONE SET FOR ALL => USE GG TB1/TB2
VBPUF0(1,1) 7.55E-5  5.92E-5  6.83E-5  5.30E-5  1.87E-7  2.31E-10  7.64E-12
            0.0       5.63E-6
$ *********************************************************************
$  THE FOLLOWING DATA BLOCKS WHICH HAVE VARIABLES ENDING WITH "0"
$  ARE TAKEN FROM MEDIAN VALUES FROM EXPERT OPINION VALUES FOR GRAND GULF
$  UNLESS OTHERWISE NOTED.
$         (1) FIRST DIMENSION IS CHEMICAL SPECIES
$         (2) SECOND DIMENSION IS CASE
$  *****************************************************************
$   NUMBER OF CHEMICAL SPECIES (NG, I, CS, TE, SR, RU, LA, CE, BA)
NSPEC   9
$ *********************************************************************
$ FCOR0 : IN-VESSEL RELEASE FRACTION FROM CORE TO RPV ATMOS.
$    BWR CASE 1: HIGH ZR OXIDATION
FCOR0(1,1)  .9    .74   .59   .15  6.4E-3  4.6E-3  1.0E-4  1.5E-4  8.6E-3
$     BWR CASE 2: LOW ZR OXIDATION
FCOR0(1,2)  .90   .69   .59   .14  4.0E-3  2.0E-3  1.0E-4  1.5E-4  6.5E-3
$ *********************************************************************
$ FVES0:  FRACTION OF RADIONUCLIDE LEAVING VESSEL DUIRNG IN-VESSEL
$         RELEASE PHASE
$     FVES BWR CASE 1: TBUX (FAST, HIGH PRESSURE)
$                      USED FOR POS 5 HIGH PRESSURE CASE
FVES0(1,1)  1.   .086  .033  .033  .033  .033  .033  .033  .033
$     FVES BWR CASE 2: TBU (FAST, LOW PRESSURE)
$                      USED FOR POS 5 LOW PRESSURE CASE
FVES0(1,2)  1.   .41   .30   .27   .26   .26   .26   .26   .26
$     FVES BWR CASE 3: TCUX (SLOW, HIGH PRESSURE, CRD)
$                      NOT USED IN POS5
FVES0(1,3)  1.   .28   .25   .10   .078  .078  .078  .078  .078
$ *********************************************************************
$ FCCI0: RELEASE FRACTIONS FROM MOLTEN CORE CONCRETE INTERACTION
$      FCCI BWR CASE 1:  LOW ZR CONTENTS AND DRY CAVITY
FCCI0(1,1)  1.  1.  1.  .66  .052  5.6E-9  2.2E-3  2.9E-3  .061
$      FCCI BWR CASE 2: LOW ZR CONTENTS AND WATER OVER DEBRIS
FCCI0(1,2)  1.  1.  1.  .64  .036  1.7E-9  2.1E-3  2.5E-3  .032
$     FCCI BWR CASE 3:  HIGH ZR CONTENTS AND DRY CAVITY
FCCI0(1,3)  1.  1.  1.  .67  .052  5.6E-9  2.2E-3  2.9E-3  .061
$     FCCI BWR CASE 4:  HIGH ZR CONTENTS AND WATER OVER DEBRIS
FCCI0(1,4)  1.  1.  1.  .64  .036  1.7E-9  2.1E-3  2.5E-3  .032
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$**********************************************************
$ FDCH: DIRECT CONTAINMENT HEATING RELEASE
$ FDCH: BWR ONE CASE ONLY: FOR HIGH PRESSURE SEQUENCES
FDCH0(1,1)  1.0  1.0  1.0   .043  .012  .020  .011  .011  .012
$*******************************************************************
$ FEVSE:  EX-VESSEL STEAM EXPLOSION RELEASE
FEVSE0(1,1) 1.   1.   1.    .043  .012  .020  .011  .011  .012
$**********************************************************
$ FLTI1: LATE IODINE RELEASE FROM SUPPRESSION POOL: IODINE ONLY
$      FLTI1 CASE 1: SUBCOOLED SUPPRESSION POOL
FLTI10(1)  1.55E-3
$      FLTI1 CASE 2: SATURATED SUPPRESSION POOL
FLTI10(2)  4.63E-3
$*********************************************************************
$ FLTI2: LATE IODINE RELEASE FROM CAVITY WATER: IODINE ONLY
$      FLTI2 CASE 1: WET CAVITY (LIKE TBS)
FLTI20(1)  .847
$      FLTI2 CASE 2: FLOODED CAVITY LIKE TC (REPLENISHABLE WATER SUPPLY)
FLTI20(2)  .435
$****************************************************************
$ FREVO0: REVOLATILIZATION RELEASE AFTER VESSEL BREACH: I,CS AND TE
$ SET ALL OTHER NUCLIDE GROUPS TO ZERO
$    BWR CASE 1: STATION BLACKOUT AND HIGH DRYWELL TEMPERATURE
$                USED FOR POS 5 CASES WITH NO RECOVERY OF INJECTION
FREVO0(1,1)  1.   .115  .051  0.    0.     0.   0.    0.    0.
$      BWR CASE 2: STATION BLACKOUT AND LOW DRYWELL TEMPERATURE
$      (NOT APPLICABLE TO GRAND GULF SINCE GRAND GULF CONTAINMENT SPRAY
$       IS IN OUTER CONTAINMENT, NOT DRYWELL)
FREVO0(1,2)  1.   .114  .050  0.    0.     0.   0.    0.    0.
$      BWR CASE 3: ATWS HIGH PRESSURE (TCUX) AND LOW PRESS. SYSTEMS
$                  AVAILABLE FOR INJECTION AFTER VESSEL BREACH
$                  USED FOR POS 5 CASES WITH INJECTION RESTORED
FREVO0(1,3)  1.   .03   .001  0.    0.     0.   0.    0.    0.
$****************************************************************
$ RBDF: REACTOR BUILDING DF FOR GRAND GULF : ALL NINE GROUPS
$ DF FROM PEACH BOTTOM-DW SHELL FAILURE INTO REACTOR BUILDING SAT. POOL.
$      GG CASE 1:
RBDF0(1,1)  1.   4.05  4.05  4.02  4.02  4.02  4.02  4.02  4.02
$****************************************************************
$ FCONV: CONTAINMENT RELEASE FRACTION BEFORE VESSEL BREACH
$    FCONV GG CASE 1: EARLY LEAK SUBCOOLED POOL
FCONV0(1,1)  1.  .233  .233  .233  .233  .233  .233  .233  .233
$    FCONV GG CASE 2: EARLY LEAK SATUARATED POOL
FCONV0(1,2)  1.  .245  .245  .245  .245  .245  .245  .245  .245
$    FCONV GG CASE 3: EARLY RUPTURE SUBCOOLED POOL
$                      (ALSO USED FOR POS 5 CASES WITH OPEN OPEN CNMT)
FCONV0(1,3)  1.  .639  .639  .639  .639  .639  .639  .639  .639
$    FCONV GG CASE 4: EARLY RUPTURE SATURATED POOL
$                      (ALSO USED FOR POS 5 CASES WITH OPEN OPEN CNMT)
FCONV0(1,4)  1.  .639  .639  .639  .639  .639  .639  .639  .639
$    FCONV GG CASE 5: LATE LEAK
FCONV0(1,5)  1.  .052  .052  .052  .052  .052  .052  .052  .052
$    FCONV GG CASE 6: LATE RUPTURE
FCONV0(1,6)  1.  .084  .084  .084  .084  .084  .084  .084  .084
$    NO CONTAINMENT FAILURE CASE
FCONV0(1,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
$*************************************************************
$ FCONC: CONTAINMENT RELEASE FRACTION AFTER VESSEL BREACH
$      FCONC GG CASE 1: EARLY LEAK SUBCOOLED POOL
FCONC0(1,1)  1.  .280  .280  .251  .251  .251  .251  .251  .251
$      FCONC GG CASE 2: EARLY LEAK SATURATED POOL
FCONC0(1,2)  1.  .251  .251  .231  .231  .231  .231  .231  .231
$      FCONC GG CASE 3: EARLY RUPTURE SUBCOOLED POOL
$                      (ALSO USED FOR POS 5 CASES WITH OPEN OPEN CNMT)
FCONC0(1,3)  1.  .743  .743  .720  .720  .720  .720  .720  .720
$      FCONC GG CASE 4: EARLY RUPTURE SATURATED POOL
$                      (ALSO USED FOR POS 5 CASES WITH OPEN OPEN CNMT)
FCONC0(1,4)  1.  .719  .719  .675  .675  .675  .675  .675  .675
$      FCONC GG CASE 5: LATE LEAK
FCONC0(1,5)  1.  .052  .052  .082  .063  .082  .063  .072  .072
```

```
$     FCONC GG CASE 6: LATE RUPTURE
FCONC0(1,6)  1.   .084  .084  .107  .094  .107  .094  .094  .094
$     NO CONTAINMENT FAILURE CASE
FCONC0(1,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
$ ***********************************************************************
$ SUPPRESSION POOL DF VALUES BASED ON VALUES FROM DRAFT NUREG/CR-4551
$ EXPERT MEDIAN VALUES
$     SUPPRESSION POOL DF THROUGH SRV T-QUENCHERS
DFVPA0(1,1)      1.0  56.     56.     56.     56.     56.     56.     56.     56.
$     SUPPRESSION POOL DF THROUGH DOWNCOMERS
DFCPA0(1,1)      1.0  6.8     6.8     6.8     6.8     6.8     6.8     6.8     6.8
$ ***********************************************************************
$ CONTAINMENT SPRAYS DF BASED ON VALUES FROM DRAFT NUREG/CR-4551
DFSPRV0(1,1)     1.0  11.     11.     11.     11.     11.     11.     11.     11.
DFSPRC0(1,1)     1.0  17.     17.     17.     17.     17.     17.     17.     17.
$ ***********************************************************************
$ CAVITY WATER DF VALUES BASED ON VALUES FROM DRAFT NUREG/CR-4551
$     EXPERT MEDIAN VALUES
$             CASE 1: WET CAVITY LIKE GRAND GULF TBS CASE
DFCAV0(1,1)      1.0  4.4     4.4     4.4     4.4     4.4     4.4     4.4     4.4
$             CASE 2: FLOODED CAVITY LIKE GRAND GULF TC CASE
DFCAV0(1,2)      1.0  6.0     6.0     6.0     6.0     6.0     6.0     6.0     6.0
$ ****************************************************************
$ ================================================================
$ ****************************************************************
$ GRAND GULF LATIN HYPERCUBE SAMPLE INTERPOLATION DATA BASE
$ ALL VARIABLE ARRAYS END WITH "L" TO REPRESENT LHS VARIABLES
$ STANDARD ARRAYS HAVE THREE DIMENSIONS:
$ FIRST DIMENSION  = RADIONUCLIDE GROUP 1 THROUGH 9
$ SECOND DIMENSION = CUMULATIVE PROBABILITY POINTS
$ THIRD DIMENSION  = DIFFERENT CASES
$ NINE NUCLIDE GROUPS GOING ACROSS: NG,I,CS,TE,SR,RU,LA,CE,BA
$ NINE CUMULATIVE PROBABILITY POINTS GOING DOWN:
$ 0.,0.01,0.05,0.25,0.5,0.75,0.95,0.99,1.0
PRBLEV  0.0  0.01  0.05  0.25  0.50  0.75  0.95  0.99  1.00
$ EACH CASE CONSISTS OF A BLOCK OF DATA OF 9 BY 9
$ *******************************************************
$ FCORL : IN-VESSEL RELEASE FRACTION FROM CORE TO RPV ATMOS.
$     BWR CASE 1: HIGH ZR OXIDATION
FCORL(1,1,1)  .05     .03     .02     0.      0.      0.      0.      0.      0.
FCORL(1,2,1)  .073    .049    .033    3.0E-3  3.0E-5  0.      0.      0.      2.2E-4
FCORL(1,3,1)  .17     .13     .07     .018    2.5E-4  0.      0.      0.      1.2E-3
FCORL(1,4,1)  .56     .34     .26     .071    2.1E-3  5.0E-5  2.0E-5  2.0E-5  4.2E-3
FCORL(1,5,1)  .9      .74     .59     .15     6.4E-3  4.6E-3  1.0E-4  1.5E-4  8.6E-3
FCORL(1,6,1)  1.      .96     .89     .59     .018    .02     1.2E-3  3.0E-3  .03
FCORL(1,7,1)  1.      1.      1.      .91     .52     .081    .021    .085    .52
FCORL(1,8,1)  1.      1.      1.      .99     1.      .14     .1      .51     1.
FCORL(1,9,1)  1.      1.      1.      1.      1.      .27     .11     1.      1.
$      BWR CASE 2: LOW ZR OXIDATION
FCORL(1,1,2)  .02     6.0E-3  5.0E-3  0.      0.      0.      0.      0.      0.
FCORL(1,2,2)  .033    6.6E-3  5.8E-3  2.9E-3  3.0E-5  0.      0.      0.      1.1E-4
FCORL(1,3,2)  .084    9.2E-3  9.0E-3  7.3E-3  1.5E-4  0.      0.      0.      2.2E-4
FCORL(1,4,2)  .41     .16     .088    .049    7.6E-4  5.0E-5  2.0E-5  2.0E-5  1.7E-3
FCORL(1,5,2)  .90     .69     .59     .14     4.0E-3  2.0E-3  1.0E-4  1.5E-4  6.5E-3
FCORL(1,6,2)  1.      .91     .83     .46     .013    .012    9.5E-4  2.5E-3  .027
FCORL(1,7,2)  1.      1.      1.      .89     .52     .058    .021    .085    .52
FCORL(1,8,2)  1.      1.      1.      .98     1.      .14     .10     .51     1.
FCORL(1,9,2)  1.      1.      1.      1.      1.      .27     .11     1.      1.
$****************************************************************
$ FVESL:  FRACTION OF RADIONUCLIDE LEAVING VESSEL DURING IN-VESSEL
$         RELEASE PHASE
$      FVESL BWR CASE 1: TBUX (FAST, HIGH PRESSURE)
$                        USED FOR POS 5 HIGH PRESSURE CASES
FVESL(1,1,1)  1.      0.      0.      0.      0.      0.      0.      0.      0.
FVESL(1,2,1)  1.      2.0E-5  2.0E-5  1.0E-5  1.0E-5  1.0E-5  1.0E-5  1.0E-5  1.0E-5
FVESL(1,3,1)  1.      8.0E-5  8.0E-5  5.0E-5  5.0E-5  5.0E-5  5.0E-5  5.0E-5  5.0E-5
FVESL(1,4,1)  1.      9.6E-3  5.1E-3  1.9E-3  1.9E-3  1.9E-3  1.9E-3  1.9E-3  1.9E-3
FVESL(1,5,1)  1.      .086    .033    .033    .033    .033    .033    .033    .033
FVESL(1,6,1)  1.      .33     .32     .31     .25     .25     .25     .25     .25
FVESL(1,7,1)  1.      .79     .79     .78     .77     .77     .77     .77     .77
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FVESL(1,8,1)  1.     .96     .96     .96     .95     .95     .95     .95     .95
FVESL(1,9,1)  1.     1.      1.      1.      1.      1.      1.      1.      1.
$     FVESL BWR CASE 2: TBU (FAST, LOW PRESSURE)
$                      USED FOR POS 5 LOW PRESSURE CASES
FVESL(1,1,2)  1.     0.      0.      0.      0.      0.      0.      0.      0.
FVESL(1,2,2)  1.     5.9E-3  3.3E-3  3.3E-3  3.3E-3  3.3E-3  3.3E-3  3.3E-3  3.3E-3
FVESL(1,3,2)  1.     .041    .023    .023    .023    .023    .023    .023    .023
FVESL(1,4,2)  1.     .23     .14     .14     .13     .13     .13     .13     .13
FVESL(1,5,2)  1.     .41     .30     .27     .26     .26     .26     .26     .26
FVESL(1,6,2)  1.     .63     .60     .59     .58     .58     .58     .58     .58
FVESL(1,7,2)  1.     .99     .99     .99     .99     .99     .99     .99     .99
FVESL(1,8,2)  1.     1.      1.      1.      1.      1.      1.      1.      1.
FVESL(1,9,2)  1.     1.      1.      1.      1.      1.      1.      1.      1.
$     FVESL BWR CASE 3: TCUX (SLOW, HIGH PRESSURE, CRD)
$                      NOT USED IN POS 5
FVESL(1,1,3)  1.     0.      1.0E-5  0.      0.      0.      0.      0.      0.
FVESL(1,2,3)  1.     8.0E-5  8.0E-5  2.0E-5  2.0E-5  2.0E-5  2.0E-5  2.0E-5  2.0E-5
FVESL(1,3,3)  1.     .018    7.6E-3  1.0E-4  1.0E-4  1.0E-4  1.0E-4  1.0E-4  1.0E-4
FVESL(1,4,3)  1.     .089    .052    4.9E-3  4.8E-3  4.8E-3  4.8E-3  4.8E-3  4.8E-3
FVESL(1,5,3)  1.     .28     .25     .10     .078    .078    .078    .078    .078
FVESL(1,6,3)  1.     .75     .63     .39     .29     .29     .29     .29     .29
FVESL(1,7,3)  1.     .95     .9      .7      .7      .7      .7      .7      .7
FVESL(1,8,3)  1.     .99     .99     .88     .88     .88     .88     .88     .88
FVESL(1,9,3)  1.     1.      1.      .98     .98     .98     .98     .98     .98
$***************************************************************************
$ FCCIL: RELEASE FRACTIONS FROM MOLTEN CORE CONCRETE INTERACTION
$      FCCI GG CASE 1:  LOW ZR CONTENTS AND DRY CAVITY
FCCIL(1,1,1)  1.     1.      1.      4.4E-3  0.      1.0E-9  0.      0.      3.0E-5
FCCIL(1,2,1)  1.     1.      1.      .012    5.0E-5  1.0E-9  0.      0.      1.2E-4
FCCIL(1,3,1)  1.     1.      1.      .069    3.1E-4  1.2E-9  1.0E-5  3.0E-5  4.9E-4
FCCIL(1,4,1)  1.     1.      1.      .32     2.6E-3  2.4E-9  2.1E-4  3.2E-4  3.2E-3
FCCIL(1,5,1)  1.     1.      1.      .66     .052    5.6E-9  2.2E-3  2.9E-3  .061
FCCIL(1,6,1)  1.     1.      1.      .76     .62     5.0E-6  .013    .026    .45
FCCIL(1,7,1)  1.     1.      1.      .94     .95     7.3E-3  .086    .018    .88
FCCIL(1,8,1)  1.     1.      1.      .99     .99     9.7E-2  .1      .2      .98
FCCIL(1,9,1)  1.     1.      1.      1.      1.      .25     .1      .2      1.
$      FCCI GG CASE 2: LOW ZR CONTENTS AND WET CAVITY
FCCIL(1,1,2)  1.     1.      1.      1.2E-3  0.      1.0E-9  0.      0.      1.0E-5
FCCIL(1,2,2)  1.     1.      1.      4.8E-3  2.0E-5  1.0E-9  0.      0.      8.0E-5
FCCIL(1,3,2)  1.     1.      1.      .032    2.7E-4  1.1E-9  0.      1.0E-5  3.6E-4
FCCIL(1,4,2)  1.     1.      1.      .26     2.0E-3  1.3E-9  1.9E-4  2.6E-4  2.3E-3
FCCIL(1,5,2)  1.     1.      1.      .64     .036    1.7E-9  2.1E-3  2.5E-3  .032
FCCIL(1,6,2)  1.     1.      1.      .74     .59     1.0E-6  .012    .02     .41
FCCIL(1,7,2)  1.     1.      1.      .93     .94     2.5E-3  .084    .17     .87
FCCIL(1,8,2)  1.     1.      1.      .99     .99     5.8E-2  .099    .2      .98
FCCIL(1,9,2)  1.     1.      1.      1.      1.      .15     .1      .2      1.
$      FCCI GG CASE 3:  HIGH ZR CONTENTS AND DRY CAVITY
FCCIL(1,1,3)  1.     1.      1.      4.4E-3  0.      1.0E-9  0.      0.      3.0E-5
FCCIL(1,2,3)  1.     1.      1.      .012    5.0E-5  1.0E-9  0.      0.      1.2E-4
FCCIL(1,3,3)  1.     1.      1.      .069    3.1E-4  1.2E-9  1.0E-5  3.0E-5  4.9E-4
FCCIL(1,4,3)  1.     1.      1.      .40     2.6E-3  2.4E-9  2.1E-4  3.2E-4  3.2E-3
FCCIL(1,5,3)  1.     1.      1.      .67     .052    5.6E-9  2.2E-3  2.9E-3  .061
FCCIL(1,6,3)  1.     1.      1.      .79     .65     5.0E-6  .02     .031    .51
FCCIL(1,7,3)  1.     1.      1.      .96     .97     7.3E-3  .11     .18     .9
FCCIL(1,8,3)  1.     1.      1.      .99     1.      9.7E-2  .15     .2      .98
FCCIL(1,9,3)  1.     1.      1.      1.      1.      .25     .16     .2      1.
$      FCCI GG CASE 4:  HIGH ZR CONTENTS AND WATER OVER DEBRIS
FCCIL(1,1,4)  1.     1.      1.      1.2E-3  0.      1.0E-9  0.      0.      1.0E-5
FCCIL(1,2,4)  1.     1.      1.      4.8E-3  2.0E-5  1.0E-9  0.      0.      8.0E-5
FCCIL(1,3,4)  1.     1.      1.      .032    2.7E-4  1.1E-9  0.      1.0E-5  3.6E-4
FCCIL(1,4,4)  1.     1.      1.      .26     2.0E-3  1.3E-9  1.9E-4  2.6E-4  2.3E-3
FCCIL(1,5,4)  1.     1.      1.      .64     .036    1.7E-9  2.1E-3  2.5E-3  .032
FCCIL(1,6,4)  1.     1.      1.      .74     .59     1.0E-6  .012    .02     .41
FCCIL(1,7,4)  1.     1.      1.      .93     .94     2.5E-3  .084    .17     .87
FCCIL(1,8,4)  1.     1.      1.      .99     .99     5.8E-2  .099    .2      .98
FCCIL(1,9,4)  1.     1.      1.      1.      1.      .15     .1      .2      1.
$***************************************************************************
$ FDCH: BWR ONE CASE ONLY: FOR HIGH PRESSURE SEQUENCES
$       FISRT DIMENSION  = RADIONUCLIDE GROUP
$       SECOND DIMENSION = PROBABILITY POINTS
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FDCHL(1,1,1)  1.     .063   .063   0.     0.     0.     0.     0.     0.
FDCHL(1,2,1)  1.     .15    .15    0.     0.     0.     0.     0.     0.
FDCHL(1,3,1)  1.     .50    .50    .001   .001   .001   .001   .001   .001
FDCHL(1,4,1)  1.     1.     1.     .008   .002   .007   .002   .002   .004
FDCHL(1,5,1)  1.     1.     1.     .043   .012   .020   .011   .011   .012
FDCHL(1,6,1)  1.     1.     1.     .600   .030   .063   .040   .040   .067
FDCHL(1,7,1)  1.     1.     1.     .975   .751   .700   .087   .087   .863
FDCHL(1,8,1)  1.     1.     1.     1.     .980   .900   .200   .280   .980
FDCHL(1,9,1)  1.     1.     1.     1.     1.     .950   .230   .330   1.
$*********************************************************************
$ FEVSE:  EX-VESSEL STEAM EXPLOSION RELEASE
FEVSEL(1,1,1) 1.     .063   .063   0.     0.     0.     0.     0.     0.
FEVSEL(1,2,1) 1.     .15    .15    0.     0.     0.     0.     0.     0.
FEVSEL(1,3,1) 1.     .50    .50    .001   .001   .001   .001   .001   .001
FEVSEL(1,4,1) 1.     1.     1.     .008   .002   .007   .002   .002   .004
FEVSEL(1,5,1) 1.     1.     1.     .043   .012   .020   .011   .011   .012
FEVSEL(1,6,1) 1.     1.     1.     .600   .030   .063   .040   .040   .067
FEVSEL(1,7,1) 1.     1.     1.     .975   .751   .700   .087   .087   .863
FEVSEL(1,8,1) 1.     1.     1.     1.     .980   .900   .200   .280   .980
FEVSEL(1,9,1) 1.     1.     1.     1.     1.     .950   .230   .330   1.
$**********************************************************************
$ FLTI1: LATE IODINE RELEASE FROM SUPPRESSION POOL: IODINE ONLY
$          THEREFORE, PROBABILITY GOING ACROSS
$          FIRST DIMENSION  = PROBABILITY POINTS
$          SECOND DIMENSION = CASES
$     FLTI1 CASE 1: SUBCOOLED SUPPRESSION POOL
FLTI1L(1,1)  0.  0.     0.       5.00E-4  1.55E-3  .0278  .085  .097  .10
$     FLTI1 CASE 2: SATURATED SUPPRESSION POOL
FLTI1L(1,2)  0.  1.E-6  4.06E-5  9.36E-4  4.63E-3  .173   .759  .95   1.
$***********************************************************************
$ FLTI2: LATE IODINE RELEASE FROM CAVITY WATER: IODINE ONLY
$     FLTI2 CASE 1: WET CAVITY (LIKE TBS)
FLTI2L(1,1)  .08   .109 .153  .365  .847  .957  1.    1.    1.
$     FLTI2 CASE 2: FLOODED CAVITY LIKE TC (REPLENISHABLE WATER SUPPLY)
FLTI2L(1,2)  .004  .04  .109  .247  .435  .670  .936  .985  1.
$********************************************************************
$ FREVOL: REVOLATILIZATION RELEASE AFTER VESSEL BREACH: I,CS AND TE
$ SET ALL OTHER NUCLIDE GROUPS TO ZERO
$    BWR CASE 1: SBO AND HIGH DW TEMP
$                USED FOR POS 5 CASES WITH NO LPI RECOVERED
FREVOL(1,1,1)  1.   0.    0.    0.    0.   0.   0.   0.   0.
FREVOL(1,2,1)  1.   0.    0.    0.    0.   0.   0.   0.   0.
FREVOL(1,3,1)  1.   0.    0.    0.    0.   0.   0.   0.   0.
FREVOL(1,4,1)  1.   .03   .001  0.    0.   0.   0.   0.   0.
FREVOL(1,5,1)  1.   .115  .051  0.    0.   0.   0.   0.   0.
FREVOL(1,6,1)  1.   .306  .132  .024  0.   0.   0.   0.   0.
FREVOL(1,7,1)  1.   .557  .284  .224  0.   0.   0.   0.   0.
FREVOL(1,8,1)  1.   .800  .535  .413  0.   0.   0.   0.   0.
FREVOL(1,9,1)  1.   1.    .750  .800  0.   0.   0.   0.   0.
$     BWR CASE 2: STATION BLACKOUT AND LOW DRYWELL TEMPERATURE
$     (NOT APPLICABLE TO GRAND GULF SINCE GRAND GULF CONTAINMENT SPRAY
$      IS IN OUTER CONTAINMENT, NOT DRYWELL)
FREVOL(1,1,2)  1.   0.    0.    0.    0.   0.   0.   0.   0.
FREVOL(1,2,2)  1.   0.    0.    0.    0.   0.   0.   0.   0.
FREVOL(1,3,2)  1.   0.    0.    0.    0.   0.   0.   0.   0.
FREVOL(1,4,2)  1.   .03   .001  0.    0.   0.   0.   0.   0.
FREVOL(1,5,2)  1.   .114  .050  0.    0.   0.   0.   0.   0.
FREVOL(1,6,2)  1.   .261  .122  .024  0.   0.   0.   0.   0.
FREVOL(1,7,2)  1.   .486  .236  .209  0.   0.   0.   0.   0.
FREVOL(1,8,2)  1.   .800  .438  .413  0.   0.   0.   0.   0.
FREVOL(1,9,2)  1.   1.    .750  .800  0.   0.   0.   0.   0.
$     BWR CASE 3: ATWS HIGH PRES.(TCUX) & LOW PRESS. SYSTEMS AVAILABLE
$                 FOR INJECTION AFTER VB
$                 USED FOR POS 5 CASES WITH INJECTION RESTORED
FREVOL(1,1,3)  1.   0.    0.    0.    0.   0.   0.   0.   0.
FREVOL(1,2,3)  1.   0.    0.    0.    0.   0.   0.   0.   0.
FREVOL(1,3,3)  1.   0.    0.    0.    0.   0.   0.   0.   0.
FREVOL(1,4,3)  1.   0.    0.    0.    0.   0.   0.   0.   0.
FREVOL(1,5,3)  1.   .03   .001  0.    0.   0.   0.   0.   0.
FREVOL(1,6,3)  1.   .117  .061  .024  0.   0.   0.   0.   0.
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FREVOL(1,7,3)  1.   .439  .200  .209  0.    0.    0.    0.    0.
FREVOL(1,8,3)  1.   .800  .287  .413  0.    0.    0.    0.    0.
FREVOL(1,9,3)  1.   1.00  .750  .800  0.    0.    0.    0.    0.
$*********************************************************************
$ RBDF: REACTOR BUILDING DF FOR GRAND GULF : ALL NINE GROUPS
$ DF FROM PEACH BOTTOM-DW SHELL FAILURE INTO REACTOR BUILDING SAT. POOL.
$     GG CASE 1:
RBDFL(1,1,1)  1.   1.    1.    1.    1.    1.    1.    1.    1.
RBDFL(1,2,1)  1.   1.06  1.06  1.08  1.08  1.08  1.08  1.08  1.08
RBDFL(1,3,1)  1.   1.23  1.23  1.25  1.25  1.25  1.25  1.25  1.25
RBDFL(1,4,1)  1.   2.29  2.29  2.22  2.22  2.22  2.22  2.22  2.22
RBDFL(1,5,1)  1.   4.05  4.05  4.02  4.02  4.02  4.02  4.02  4.02
RBDFL(1,6,1)  1.   6.86  6.86  6.86  6.86  6.86  6.86  6.86  6.86
RBDFL(1,7,1)  1.   14.5  14.5  17.0  17.0  17.0  17.0  17.0  17.0
RBDFL(1,8,1)  1.   87.9  87.9  92.1  92.1  92.1  92.1  92.1  92.1
RBDFL(1,9,1)  1.   551.  551.  590.  590.  590.  590.  590.  590.
$********************************************************************
$ FCONV: CONTAINMENT RELEASE FRACTION BEFORE VESSEL BREACH: ALL NINE GROUPS
$   FCONVL GG CASE 1: EARLY LEAK, SUBCOOLED POOL
FCONVL(1,1,1)  1.  .001  .001  .001  .001  .001  .001  .001  .001
FCONVL(1,2,1)  1.  .003  .003  .003  .003  .003  .003  .003  .003
FCONVL(1,3,1)  1.  .012  .012  .012  .012  .012  .012  .012  .012
FCONVL(1,4,1)  1.  .117  .117  .117  .117  .117  .117  .117  .117
FCONVL(1,5,1)  1.  .233  .233  .233  .233  .233  .233  .233  .233
FCONVL(1,6,1)  1.  .417  .417  .417  .417  .417  .417  .417  .417
FCONVL(1,7,1)  1.  .676  .676  .676  .676  .676  .676  .676  .676
FCONVL(1,8,1)  1.  .784  .784  .784  .784  .784  .784  .784  .784
FCONVL(1,9,1)  1.  .949  .949  .949  .949  .949  .949  .949  .949
$   FCONVL GG CASE 2: EARLY LEAK, SATURATED POOL
FCONVL(1,1,2)  1.  .002  .002  .002  .002  .002  .002  .002  .002
FCONVL(1,2,2)  1.  .008  .008  .008  .008  .008  .008  .008  .008
FCONVL(1,3,2)  1.  .030  .030  .030  .030  .030  .030  .030  .030
FCONVL(1,4,2)  1.  .151  .151  .151  .151  .151  .151  .151  .151
FCONVL(1,5,2)  1.  .245  .245  .245  .245  .245  .245  .245  .245
FCONVL(1,6,2)  1.  .447  .447  .447  .447  .447  .447  .447  .447
FCONVL(1,7,2)  1.  .695  .695  .695  .695  .695  .695  .695  .695
FCONVL(1,8,2)  1.  .792  .792  .792  .792  .792  .792  .792  .792
FCONVL(1,9,2)  1.  .953  .953  .953  .953  .953  .953  .953  .953
$   FCONVL GG CASE 3: EARLY RUPTURE, SUBCOOLED POOL
$                    (ALSO USED FOR POS 5 OPEN CNMT CASES)
FCONVL(1,1,3)  1.  .021  .021  .021  .021  .021  .021  .021  .021
FCONVL(1,2,3)  1.  .090  .090  .090  .090  .090  .090  .090  .090
FCONVL(1,3,3)  1.  .197  .197  .197  .197  .197  .197  .197  .197
FCONVL(1,4,3)  1.  .437  .437  .437  .437  .437  .437  .437  .437
FCONVL(1,5,3)  1.  .639  .639  .639  .639  .639  .639  .639  .639
FCONVL(1,6,3)  1.  .790  .790  .770  .770  .770  .770  .770  .770
FCONVL(1,7,3)  1.  .915  .915  .892  .892  .892  .892  .892  .892
FCONVL(1,8,3)  1.  .966  .966  .966  .966  .966  .966  .966  .966
FCONVL(1,9,3)  1.  .996  .996  .996  .996  .996  .996  .996  .996
$   FCONVL GG CASE 4: EARLY RUPTURE, SATURATED POOL
$                    (ALSO USED FOR POS 5 OPEN CNMT CASES)
FCONVL(1,1,4)  1.  .021  .021  .021  .021  .021  .021  .021  .021
FCONVL(1,2,4)  1.  .090  .090  .090  .090  .090  .090  .090  .090
FCONVL(1,3,4)  1.  .197  .197  .197  .197  .197  .197  .197  .197
FCONVL(1,4,4)  1.  .437  .437  .437  .437  .437  .437  .437  .437
FCONVL(1,5,4)  1.  .639  .639  .639  .639  .639  .639  .639  .639
FCONVL(1,6,4)  1.  .790  .790  .770  .770  .770  .770  .770  .770
FCONVL(1,7,4)  1.  .915  .915  .892  .892  .892  .892  .892  .892
FCONVL(1,8,4)  1.  .966  .966  .966  .966  .966  .966  .966  .966
FCONVL(1,9,4)  1.  .996  .996  .996  .996  .996  .996  .996  .996
$   FCONVL GG CASE 5: LATE LEAK
FCONVL(1,1,5)  1.  0.    0.    0.    0.    0.    0.    0.    0.
FCONVL(1,2,5)  1.  0.    0.    0.    0.    0.    0.    0.    0.
FCONVL(1,3,5)  1.  .001  .001  .001  .001  .001  .001  .001  .001
FCONVL(1,4,5)  1.  .008  .008  .008  .008  .008  .008  .008  .008
FCONVL(1,5,5)  1.  .052  .052  .052  .052  .052  .052  .052  .052
FCONVL(1,6,5)  1.  .128  .128  .128  .128  .128  .128  .128  .128
FCONVL(1,7,5)  1.  .330  .330  .330  .330  .330  .330  .330  .330
FCONVL(1,8,5)  1.  .510  .510  .510  .510  .510  .510  .510  .510
FCONVL(1,9,5)  1.  .814  .814  .814  .814  .814  .814  .814  .814
```

```
$      FCONVL GG CASE 6: LATE RUPTURE
FCONVL(1,1,6)  1.  0.    0.    0.    0.    0.    0.    0.    0.
FCONVL(1,2,6)  1.  0.    0.    0.    0.    0.    0.    0.    0.
FCONVL(1,3,6)  1.  .002  .002  .002  .002  .002  .002  .002  .002
FCONVL(1,4,6)  1.  .017  .017  .017  .017  .017  .017  .017  .017
FCONVL(1,5,6)  1.  .084  .084  .084  .084  .084  .084  .084  .084
FCONVL(1,6,6)  1.  .186  .186  .186  .186  .186  .186  .186  .186
FCONVL(1,7,6)  1.  .338  .338  .338  .338  .338  .338  .338  .338
FCONVL(1,8,6)  1.  .540  .540  .540  .540  .540  .540  .540  .540
FCONVL(1,9,6)  1.  .969  .969  .969  .969  .969  .969  .969  .969
$      FCONVL: NO CONTAINMENT FAILURE CASE
FCONVL(1,1,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONVL(1,2,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONVL(1,3,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONVL(1,4,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONVL(1,5,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONVL(1,6,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONVL(1,7,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONVL(1,8,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONVL(1,9,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
$*************************************************************************
$ FCONC: CONTAINMENT RELEASE FRACTION AFTER VESSEL BREACH: ALL NINE GROUPS
$      FCONC GG CASE 1: EARLY LEAK, SUBCOOLED POOL
FCONCL(1,1,1)  1.  .001  .001  .001  .001  .001  .001  .001  .001
FCONCL(1,2,1)  1.  .003  .003  .003  .003  .003  .003  .003  .003
FCONCL(1,3,1)  1.  .012  .012  .012  .012  .012  .012  .012  .012
FCONCL(1,4,1)  1.  .115  .115  .088  .088  .088  .088  .088  .088
FCONCL(1,5,1)  1.  .280  .280  .251  .251  .251  .251  .251  .251
FCONCL(1,6,1)  1.  .461  .461  .428  .428  .428  .428  .428  .428
FCONCL(1,7,1)  1.  .672  .672  .672  .672  .672  .672  .672  .672
FCONCL(1,8,1)  1.  .779  .779  .779  .779  .779  .779  .779  .779
FCONCL(1,9,1)  1.  .876  .876  .876  .876  .876  .876  .876  .876
$      FCONC GG CASE 2: EARLY LEAK, SATURATED POOL
FCONCL(1,1,2)  1.  .002  .002  .002  .002  .002  .002  .002  .002
FCONCL(1,2,2)  1.  .008  .008  .008  .008  .008  .008  .008  .008
FCONCL(1,3,2)  1.  .030  .030  .024  .024  .024  .024  .024  .024
FCONCL(1,4,2)  1.  .141  .141  .115  .115  .115  .115  .115  .115
FCONCL(1,5,2)  1.  .251  .251  .231  .231  .231  .231  .231  .231
FCONCL(1,6,2)  1.  .449  .449  .405  .405  .405  .405  .405  .405
FCONCL(1,7,2)  1.  .689  .689  .689  .689  .689  .689  .689  .689
FCONCL(1,8,2)  1.  .789  .789  .789  .789  .789  .789  .789  .789
FCONCL(1,9,2)  1.  .892  .892  .892  .892  .892  .892  .892  .892
$      FCONC GG CASE 3: EARLY RUPTURE, SUBCOOLED POOL
$                      (ALSO USED FOR POS 5 OPEN CNMT CASES)
FCONCL(1,1,3)  1.  .038  .038  .015  .015  .015  .015  .015  .015
FCONCL(1,2,3)  1.  .148  .148  .054  .054  .054  .054  .054  .054
FCONCL(1,3,3)  1.  .218  .218  .169  .169  .169  .169  .169  .169
FCONCL(1,4,3)  1.  .512  .512  .451  .451  .451  .451  .451  .451
FCONCL(1,5,3)  1.  .743  .743  .720  .720  .720  .720  .720  .720
FCONCL(1,6,3)  1.  .882  .882  .855  .855  .855  .855  .855  .855
FCONCL(1,7,3)  1.  .985  .985  .985  .985  .985  .985  .985  .985
FCONCL(1,8,3)  1.  .990  .990  .990  .990  .990  .990  .990  .990
FCONCL(1,9,3)  1.  1.    1.    1.    1.    1.    1.    1.    1.
$      FCONC GG CASE 4: EARLY RUPTURE, SATURATED POOL
$                      (ALSO USED FOR POS 5 OPEN CNMT CASES)
FCONCL(1,1,4)  1.  .038  .038  .013  .013  .013  .013  .013  .013
FCONCL(1,2,4)  1.  .148  .148  .042  .042  .042  .042  .042  .042
FCONCL(1,3,4)  1.  .218  .218  .153  .153  .153  .153  .153  .153
FCONCL(1,4,4)  1.  .491  .491  .435  .435  .435  .435  .435  .435
FCONCL(1,5,4)  1.  .719  .719  .675  .675  .675  .675  .675  .675
FCONCL(1,6,4)  1.  .859  .859  .828  .828  .828  .828  .828  .828
FCONCL(1,7,4)  1.  .940  .940  .936  .936  .936  .936  .936  .936
FCONCL(1,8,4)  1.  .974  .974  .974  .974  .974  .974  .974  .974
FCONCL(1,9,4)  1.  .994  .994  .994  .994  .994  .994  .994  .994
$      FCONC GG CASE 5: LATE LEAK
FCONCL(1,1,5)  1.  0.    0.    0.    0.    0.    0.    0.    0.
FCONCL(1,2,5)  1.  0.    0.    .001  .001  .001  .001  .001  .001
FCONCL(1,3,5)  1.  .001  .001  .002  .002  .002  .002  .002  .002
FCONCL(1,4,5)  1.  .008  .008  .023  .014  .023  .014  .014  .014
FCONCL(1,5,5)  1.  .052  .052  .082  .063  .082  .063  .072  .072
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FCONCL(1,6,5)  1.  .128  .128  .183  .149  .183  .149  .164  .164
FCONCL(1,7,5)  1.  .330  .330  .423  .392  .423  .392  .404  .404
FCONCL(1,8,5)  1.  .510  .510  .595  .510  .595  .510  .510  .510
FCONCL(1,9,5)  1.  .814  .814  .820  .814  .820  .814  .814  .814
$    FCONC GG CASE 6: LATE RUPTURE
FCONCL(1,1,6)  1.  0.    0.    0.    0.    0.    0.    0.    0.
FCONCL(1,2,6)  1.  0.    0.    .001  .001  .001  .001  .001  .001
FCONCL(1,3,6)  1.  .002  .002  .003  .003  .003  .003  .003  .003
FCONCL(1,4,6)  1.  .017  .017  .037  .020  .037  .020  .020  .020
FCONCL(1,5,6)  1.  .084  .084  .107  .094  .107  .094  .094  .094
FCONCL(1,6,6)  1.  .186  .186  .256  .226  .256  .226  .226  .226
FCONCL(1,7,6)  1.  .338  .338  .775  .771  .775  .771  .771  .771
FCONCL(1,8,6)  1.  .540  .540  .920  .920  .920  .920  .920  .920
FCONCL(1,9,6)  1.  .969  .969  .973  .973  .973  .973  .973  .973
$     FCONCL: NO CONTAINMENT FAILURE CASE
FCONCL(1,1,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONCL(1,2,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONCL(1,3,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONCL(1,4,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONCL(1,5,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONCL(1,6,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONCL(1,7,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONCL(1,8,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
FCONCL(1,9,7)  0.005 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6 1.0E-6
$*************************************************************************
$  DFVPA:  SUPPRESSION POOL DF DURING IN-VESSEL RELEASE PHASE
$          (THROUGH T-QUENCHER)
$      DFVPA GG CASE 1:  DRAFT NUREG/CR-4551
DFVPAL(1,1,1)  1.0  1.0    1.0    1.0    1.0    1.0    1.0    1.0    1.0
DFVPAL(1,2,1)  1.0  1.1    1.1    1.1    1.1    1.1    1.1    1.1    1.1
DFVPAL(1,3,1)  1.0  1.8    1.8    1.8    1.8    1.8    1.8    1.8    1.8
DFVPAL(1,4,1)  1.0  16.    16.    16.    16.    16.    16.    16.    16.
DFVPAL(1,5,1)  1.0  56.    56.    56.    56.    56.    56.    56.    56.
DFVPAL(1,6,1)  1.0  180.   180.   180.   180.   180.   180.   180.   180.
DFVPAL(1,7,1)  1.0  2500.  2500.  2500.  2500.  2500.  2500.  2500.  2500.
DFVPAL(1,8,1)  1.0  4300.  4300.  4300.  4300.  4300.  4300.  4300.  4300.
DFVPAL(1,9,1)  1.0  5000.  5000.  5000.  5000.  5000.  5000.  5000.  5000.
$*************************************************************************
$  DFCPA:  SUPPRESSION POOL DF THRU VENT PIPES
$      DFCPA GG CASE 1:  DRAFT NUREG/CR-4551
DFCPAL(1,1,1)  1.0  1.0    1.0    1.0    1.0    1.0    1.0    1.0    1.0
DFCPAL(1,2,1)  1.0  1.0    1.0    1.0    1.0    1.0    1.0    1.0    1.0
DFCPAL(1,3,1)  1.0  1.2    1.2    1.2    1.2    1.2    1.2    1.2    1.2
DFCPAL(1,4,1)  1.0  2.6    2.6    2.6    2.6    2.6    2.6    2.6    2.6
DFCPAL(1,5,1)  1.0  6.8    6.8    6.8    6.8    6.8    6.8    6.8    6.8
DFCPAL(1,6,1)  1.0  20.    20.    20.    20.    20.    20.    20.    20.
DFCPAL(1,7,1)  1.0  72.    72.    72.    72.    72.    72.    72.    72.
DFCPAL(1,8,1)  1.0  94.    94.    94.    94.    94.    94.    94.    94.
DFCPAL(1,9,1)  1.0  100.   100.   100.   100.   100.   100.   100.   100.
$*************************************************************************
$  DFCAV:  CAVITY WATER DF FOR CCI RELEASE
$       DFCAV GG CASE 1:  WET CAVITY SIMILAR TO BMI-2139 GG TBS
$                         (DRAFT NUREG/CR-4551)
DFCAVL(1,1,1)  1.0  1.0    1.0    1.0    1.0    1.0    1.0    1.0    1.0
DFCAVL(1,2,1)  1.0  1.0    1.0    1.0    1.0    1.0    1.0    1.0    1.0
DFCAVL(1,3,1)  1.0  1.1    1.1    1.1    1.1    1.1    1.1    1.1    1.1
DFCAVL(1,4,1)  1.0  2.0    2.0    2.0    2.0    2.0    2.0    2.0    2.0
DFCAVL(1,5,1)  1.0  4.4    4.4    4.4    4.4    4.4    4.4    4.4    4.4
DFCAVL(1,6,1)  1.0  11.    11.    11.    11.    11.    11.    11.    11.
DFCAVL(1,7,1)  1.0  41.    41.    41.    41.    41.    41.    41.    41.
DFCAVL(1,8,1)  1.0  65.    65.    65.    65.    65.    65.    65.    65.
DFCAVL(1,9,1)  1.0  73.    73.    73.    73.    73.    73.    73.    73.
$       DFCAV GG CASE 2:  FLOODED CAVITY SIMILAR TO BMI-2139 GG TC
DFCAVL(1,1,2)  1.0  1.0    1.0    1.0    1.0    1.0    1.0    1.0    1.0
DFCAVL(1,2,2)  1.0  1.0    1.0    1.0    1.0    1.0    1.0    1.0    1.0
DFCAVL(1,3,2)  1.0  1.2    1.2    1.2    1.2    1.2    1.2    1.2    1.2
DFCAVL(1,4,2)  1.0  2.8    2.8    2.8    2.8    2.8    2.8    2.8    2.8
DFCAVL(1,5,2)  1.0  6.0    6.0    6.0    6.0    6.0    6.0    6.0    6.0
DFCAVL(1,6,2)  1.0  15.    15.    15.    15.    15.    15.    15.    15.
DFCAVL(1,7,2)  1.0  56.    56.    56.    56.    56.    56.    56.    56.
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DFCAVL(1,8,2)  1.0  89.    89.    89.    89.    89.    89.    89.    89.
DFCAVL(1,9,2)  1.0  100.   100.   100.   100.   100.   100.   100.   100.
$****************************************************************************
$  DFSPRV:  SPRAY DF FOR IN-VESSEL RELEASES
$       DFSPRV GG CASE 1 DRAFT NUREG/CR-4551 (SURRY)
DFSPRVL(1,1,1)  1.0  1.0   1.0    1.0    1.0    1.0    1.0    1.0    1.0
DFSPRVL(1,2,1)  1.0  1.1   1.1    1.1    1.1    1.1    1.1    1.1    1.1
DFSPRVL(1,3,1)  1.0  1.3   1.3    1.3    1.3    1.3    1.3    1.3    1.3
DFSPRVL(1,4,1)  1.0  4.2   4.2    4.2    4.2    4.2    4.2    4.2    4.2
DFSPRVL(1,5,1)  1.0  11.   11.    11.    11.    11.    11.    11.    11.
DFSPRVL(1,6,1)  1.0  29.   29.    29.    29.    29.    29.    29.    29.
DFSPRVL(1,7,1)  1.0  78.   78.    78.    78.    78.    78.    78.    78.
DFSPRVL(1,8,1)  1.0  95.   95.    95.    95.    95.    95.    95.    95.
DFSPRVL(1,9,1)  1.0  100.  100.   100.   100.   100.   100.   100.   100.
$****************************************************************************
$  DFSPRC:  SPRAY DF FOR CCI RELEASES
$       DFSPRC GG CASE 1 DRAFT NUREG/CR-4551 (SURRY)
DFSPRCL(1,1,1)  1.0  1.0   1.0    1.0    1.0    1.0    1.0    1.0    1.0
DFSPRCL(1,2,1)  1.0  1.1   1.1    1.1    1.1    1.1    1.1    1.1    1.1
DFSPRCL(1,3,1)  1.0  1.5   1.5    1.5    1.5    1.5    1.5    1.5    1.5
DFSPRCL(1,4,1)  1.0  7.8   7.8    7.8    7.8    7.8    7.8    7.8    7.8
DFSPRCL(1,5,1)  1.0  17.   17.    17.    17.    17.    17.    17.    17.
DFSPRCL(1,6,1)  1.0  29.   29.    29.    29.    29.    29.    29.    29.
DFSPRCL(1,7,1)  1.0  480.  480.   480.   480.   480.   480.   480.   480.
DFSPRCL(1,8,1)  1.0  860.  860.   860.   860.   860.   860.   860.   860.
DFSPRCL(1,9,1)  1.0  1000. 1000.  1000.  1000.  1000.  1000.  1000.  1000.
```

C.3 Partitioned Source Terms

The partitioning process generated 55 source term groups (STG). The source terms associated with each STG is provided in Table C.3-1. A 55th STG is the special case where there is no release. A 5x5 early health effect versus chronic healt effect partition grid was used in this analysis. The source terms, as listed in Table C.3-1, provide the following informtion about the release:

$$ST = [STG, Freq, TW, T1, D1, EL, EVT, E1\ (RF1_i\ i=1,...9), T2, D2, E2, (RF2_i\ i=1,...9)],$$

where

STG	=	Source term group
Freq	=	Frequency (1/yr) of source term group
TW	=	the time (seconds), relative to the start of the accident, when a general emergency is declared,
T1	=	the time (seconds), relative to the start of the accident, when the first release segment begins,
D1	=	the duration (seconds) of the first release segment,
EL	=	the elevation (meters), relative to ground level, from which the radionuclides are released from the containment,
EVT	=	the event type which identifies the time window in which the accident occurs and is used to select the appropriate inventory in the consequence analysis,
E1	=	the energy release rate (MW) associated with the first release segment,
$RF1_i$	=	release fraction for radionuclide class i, i = 1,...9, in the first release segment,
T2	=	the time (seconds), relative to the start of the accident, when the second release segment begins (in this analysis the second release segment immediately follows the first release segment),
D2	=	the duration (seconds) of the second release segment,
E2	=	the energy release rate (MW) associated with the second release segment,
$RF2_i$	=	release fraction for radionuclide class i, i = 1,...9, in the second release segment.

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Table C.3-1
Source Terms for Partitioned Source Term Groups

GG POS5 PARTITION DEFINITIONS

SOURCE TERM	FREQ (/YR)	TW (S)	T (S)	DT (S)	EL (M)	EVT	ENERGY (W)	RF1 XE/KR	RF2 I	RF3 CS	RF4 TE	RF5 SR	RF6 RU	RF7 LA	RF8 CE	RF9 BA
GG-001	1.1E-08	5.6E+04	6.3E+04	2.2E+04	32.	1	6.9E+04	6.8E-01	4.0E-02	3.5E-02	1.9E-02	5.8E-03	5.0E-03	9.8E-04	1.2E-03	6.6E-03
			8.5E+04	8.6E+04			3.5E+05	3.2E-01	1.6E-01	1.5E-01	1.1E-01	7.3E-02	2.3E-04	5.1E-03	4.7E-03	5.6E-02
GG-002	5.5E-09	1.3E+04	1.6E+04	3.6E+04	32.	1	4.2E+04	8.1E-01	1.2E-01	9.0E-02	5.6E-02	1.7E-02	3.3E-03	1.3E-03	6.5E-03	1.7E-02
			5.2E+04	8.6E+04			3.5E+05	1.9E-01	1.2E-01	1.3E-01	1.1E-01	8.7E-02	1.6E-03	1.4E-02	1.1E-02	6.9E-02
GG-003	6.5E-09	1.3E+04	1.6E+04	3.6E+04	32.	1	4.2E+04	4.6E-01	4.3E-02	2.7E-02	1.2E-02	1.7E-03	3.7E-04	7.1E-05	2.6E-04	2.0E-03
			5.2E+04	8.6E+04			3.5E+05	5.4E-01	1.9E-01	2.0E-01	1.2E-01	6.3E-02	4.2E-04	4.4E-03	3.2E-03	5.0E-02
GG-004	1.7E-09	1.3E+04	1.6E+04	3.6E+04	32.	1	4.2E+04	9.3E-01	3.9E-01	3.7E-01	3.1E-01	1.5E-01	2.4E-02	8.3E-03	3.7E-02	1.6E-01
			5.2E+04	8.6E+04			3.5E+05	7.0E-02	7.5E-02	8.7E-02	9.8E-02	9.8E-02	2.7E-04	5.2E-03	5.5E-03	7.6E-02
GG-005	2.1E-09	1.3E+04	1.6E+04	3.6E+04	32.	1	4.2E+04	6.4E-01	2.7E-01	2.3E-01	1.2E-01	1.1E-02	4.2E-03	4.9E-04	1.7E-03	1.3E-02
			5.2E+04	8.6E+04			3.5E+05	3.6E-01	2.3E-01	2.7E-01	2.1E-01	7.2E-02	4.3E-08	1.9E-03	3.3E-03	3.6E-02
GG-006	8.7E-09	5.6E+03	1.3E+04	7.2E+04	32.	1	6.9E+04	6.5E-01	4.2E-02	3.1E-02	1.3E-02	4.2E-03	2.8E-03	5.8E-04	9.7E-04	4.6E-03
			8.5E+04	8.6E+04			3.5E+05	3.0E-01	1.7E-01	4.5E-02	1.6E-02	3.2E-03	1.8E-06	1.4E-04	2.6E-04	2.0E-03
GG-007	5.7E-09	5.6E+03	1.3E+04	7.2E+04	32.	1	6.9E+04	8.0E-01	4.8E-02	3.2E-02	2.0E-02	8.6E-03	4.4E-03	8.6E-04	1.9E-03	9.2E-03
			8.5E+04	8.6E+04			3.5E+05	2.0E-01	1.8E-01	2.1E-02	2.5E-02	3.8E-02	1.3E-03	3.8E-03	7.5E-03	3.5E-02
GG-008	6.6E-09	5.6E+03	1.3E+04	7.2E+04	32.	1	6.9E+04	8.1E-01	1.5E-01	1.3E-01	6.4E-02	1.2E-02	6.0E-03	1.2E-03	2.3E-03	1.4E-02
			8.5E+04	8.6E+04			3.5E+05	1.8E-01	1.3E-01	5.4E-02	3.3E-02	1.2E-02	7.6E-06	5.1E-04	9.2E-04	6.4E-03
GG-009	1.6E-09	5.6E+04	6.3E+04	2.2E+04	32.	1	6.9E+04	7.1E-01	1.0E-01	9.9E-02	7.9E-02	4.8E-02	2.4E-02	5.2E-03	1.4E-02	5.0E-02
			8.5E+04	8.6E+04			3.5E+05	2.9E-01	2.2E-01	1.9E-01	1.4E-01	7.6E-02	4.2E-05	4.1E-03	4.5E-03	5.4E-02
GG-010	7.2E-09	1.3E+04	1.6E+04	3.6E+04	32.	1	4.2E+04	7.8E-01	2.7E-02	1.6E-02	6.8E-03	6.2E-04	2.1E-04	2.7E-05	9.1E-05	7.5E-04
			5.2E+04	8.6E+04			3.5E+05	2.2E-01	4.3E-02	4.7E-02	4.2E-02	2.7E-02	1.4E-04	1.0E-03	1.6E-03	1.8E-02
GG-011	2.4E-09	1.3E+04	1.6E+04	3.6E+04	32.	1	4.2E+04	9.9E-01	8.1E-02	5.6E-02	3.8E-02	1.6E-02	2.8E-03	1.5E-03	7.6E-03	1.6E-02
			5.2E+04	8.6E+04			3.5E+05	1.2E-02	2.0E-02	2.2E-02	3.4E-02	2.1E-02	5.2E-04	1.0E-03	1.7E-03	1.3E-02
GG-012	7.5E-10	5.6E+04	6.3E+04	2.2E+04	32.	1	6.9E+04	5.2E-01	6.1E-02	5.8E-02	3.7E-02	2.3E-02	2.1E-02	3.3E-03	3.5E-03	2.6E-02
			8.5E+04	8.6E+04			3.5E+05	4.8E-01	3.6E-01	3.8E-01	2.9E-01	2.1E-01	1.4E-05	7.0E-03	8.7E-03	1.4E-01
GG-013	7.6E-10	5.6E+03	1.3E+04	7.2E+04	32.	1	6.9E+04	8.0E-01	1.7E-01	1.6E-01	1.3E-01	7.8E-02	1.9E-02	6.5E-03	2.5E-02	7.9E-02
			8.5E+04	8.6E+04			3.5E+05	2.0E-01	1.2E-01	5.7E-02	5.8E-02	6.7E-02	6.6E-05	5.6E-03	1.1E-02	5.8E-02
GG-014	4.3E-10	5.6E+03	1.3E+04	7.2E+04	32.	1	6.9E+04	1.0E+00	2.6E-01	2.5E-01	2.4E-01	2.2E-01	3.2E-02	2.1E-02	1.1E-01	2.2E-01
			8.5E+04	8.6E+04			3.5E+05	0.0E+00	2.8E-02	1.5E-02	9.5E-03	1.9E-03	3.4E-09	4.2E-04	4.3E-04	7.6E-04
GG-015	4.5E-09	5.6E+04	6.3E+04	2.2E+04	32.	1	6.9E+04	9.1E-01	1.8E-02	1.6E-02	1.1E-02	4.0E-03	2.2E-03	4.4E-04	7.2E-04	4.4E-03
			8.5E+04	8.6E+04			3.5E+05	8.9E-02	5.5E-02	3.8E-02	3.8E-02	3.0E-02	2.7E-04	1.8E-03	2.1E-03	2.1E-02
GG-016	2.0E-09	5.6E+04	6.3E+04	2.2E+04	32.	1	6.9E+04	9.9E-01	3.2E-02	3.1E-02	2.9E-02	1.2E-02	6.5E-03	2.0E-03	3.2E-03	1.4E-02
			8.5E+04	8.6E+04			3.5E+05	7.3E-03	3.2E-01	2.2E-02	2.2E-02	2.3E-02	5.5E-04	2.5E-03	3.3E-03	1.8E-02

Table C.3-1 (continued)

GG POS5 PARTITION DEFINITIONS

SOURCE TERM	FREQ (/YR)	TW (S)	T (S)	DT (S)	EL (M)	EVT	ENERGY (W)	RF1 XE/KR	RF2 I	RF3 CS	RF4 TE	RF5 SR	RF6 RU	RF7 LA	RF8 CE	RF9 BA
GG-017	1.4E-10	5.6E+04	6.3E+04	2.2E+04	32.	1	6.9E+04	5.0E-01	6.3E-02	6.4E-02	3.6E-02	1.9E-02	1.9E-02	3.7E-03	4.3E-03	2.2E-02
			8.5E+04	8.6E+04			3.5E+05	5.0E-01	3.9E-01	4.2E-01	4.4E-01	4.7E-01	2.1E-02	4.6E-02	6.7E-02	4.3E-01
GG-018	1.3E-08	1.4E+04	2.0E+04	5.2E+04	32.	1	7.7E+04	8.0E-01	1.7E-02	1.2E-02	7.2E-03	2.2E-03	1.6E-03	3.5E-04	5.5E-04	2.5E-03
			7.3E+04	8.6E+04			3.5E+05	1.6E-01	6.0E-02	7.8E-03	3.2E-03	1.6E-03	5.3E-06	8.0E-05	1.4E-04	1.1E-03
GG-019	9.8E-09	4.1E+04	4.7E+04	3.1E+04	32.	1	6.8E+04	6.3E-01	1.7E-02	1.3E-02	3.9E-03	5.0E-04	3.8E-04	1.4E-04	1.7E-04	6.3E-04
			7.9E+04	8.6E+04			3.5E+05	3.3E-01	5.1E-02	5.0E-02	2.2E-02	6.3E-03	1.7E-05	3.2E-04	3.6E-04	4.7E-03
GG-020	1.8E-09	5.6E+04	6.3E+04	2.2E+04	32.	1	6.9E+04	7.5E-01	1.5E-02	1.7E-02	2.7E-03	4.6E-04	6.5E-04	2.6E-04	2.6E-04	6.4E-04
			8.5E+04	8.6E+04			3.5E+05	2.5E-01	1.8E-01	1.7E-01	7.1E-02	7.3E-03	1.8E-07	2.5E-04	4.0E-04	4.7E-03
GG-021	2.5E-08	3.3E+04	4.0E+04	4.0E+04	32.	1	7.9E+04	8.0E-01	6.1E-03	4.4E-03	1.7E-03	6.4E-04	4.7E-04	1.3E-04	2.1E-04	7.1E-04
			8.1E+04	8.6E+04			3.5E+05	3.8E-02	1.0E-02	6.0E-03	1.5E-03	2.0E-04	1.1E-07	1.6E-05	1.7E-05	1.3E-04
GG-022	2.3E-09	5.6E+03	1.3E+04	7.2E+04	32.	1	6.9E+04	7.3E-01	1.9E-02	1.4E-02	2.7E-03	1.6E-04	2.3E-04	1.1E-04	1.1E-04	2.4E-04
			8.5E+04	8.6E+04			3.5E+05	3.3E-02	2.3E-02	1.6E-02	2.4E-03	1.7E-05	2.5E-11	1.3E-06	1.9E-06	1.4E-05
GG-023	2.6E-09	5.6E+04	6.3E+04	2.2E+04	32.	1	6.9E+04	1.0E+00	3.0E-04	1.7E-04	6.3E-05	4.8E-05	2.3E-05	9.0E-06	2.2E-05	5.0E-05
			8.5E+04	8.6E+04			3.5E+05	6.3E-04	5.7E-04	1.7E-04	7.9E-04	5.9E-04	2.8E-10	8.3E-05	9.4E-05	3.6E-04
GG-024	3.2E-09	2.4E+04	2.9E+04	4.6E+04	32.	1	1.0E+05	5.6E-01	7.1E-03	6.1E-03	3.6E-03	2.2E-03	3.7E-03	7.4E-04	1.1E-03	2.5E-03
			7.5E+04	8.6E+04			3.5E+05	4.4E-01	2.1E-01	8.2E-03	4.4E-03	3.5E-03	9.1E-04	5.1E-03	2.8E-03	2.7E-03
GG-025	1.8E-09	2.6E+04	2.7E+04	3.6E+04	32.	1	1.4E+05	7.4E-01	1.8E-02	1.8E-02	8.6E-03	3.0E-03	3.1E-03	7.2E-04	8.0E-04	3.6E-03
			6.3E+04	8.6E+04			3.5E+05	2.4E-01	1.4E-01	4.2E-02	2.4E-02	1.4E-02	8.7E-06	5.5E-04	1.0E-03	8.4E-03
GG-026	1.8E-09	2.6E+04	2.7E+04	3.6E+04	32.	1	1.4E+05	8.3E-01	5.2E-04	4.8E-04	2.5E-04	1.1E-04	4.7E-04	1.0E-04	9.9E-05	1.2E-04
			6.3E+04	8.6E+04			3.5E+05	1.3E-05	7.5E-03	3.4E-04	6.9E-05	1.3E-05	1.8E-12	1.7E-06	1.8E-06	7.0E-06
GG-027	2.0E-09	5.6E+04	6.3E+04	2.2E+04	32.	1	6.9E+04	7.6E-01	8.3E-03	6.9E-03	1.6E-03	2.8E-04	3.0E-04	8.0E-05	8.9E-05	3.4E-04
			8.5E+04	8.6E+04			3.5E+05	2.4E-01	3.1E-02	3.5E-02	9.4E-03	1.3E-03	3.0E-06	1.1E-04	8.8E-05	1.1E-03
GG-028	1.4E-07	2.0E+04	2.3E+04	2.2E+04	32.	2	6.9E+04	9.9E-01	2.1E-01	1.9E-01	1.6E-01	8.5E-02	1.3E-02	5.6E-03	2.6E-02	8.6E-02
			4.4E+04	8.6E+04			3.5E+05	1.2E-02	3.5E-02	3.0E-02	5.4E-02	1.2E-01	7.3E-04	1.5E-02	1.1E-02	9.6E-02
GG-029	2.6E-07	2.0E+04	2.3E+04	2.2E+04	32.	2	6.9E+04	4.8E-01	6.2E-02	4.5E-02	2.4E-02	4.4E-03	1.1E-03	2.1E-04	8.5E-04	4.7E-03
			4.4E+04	8.6E+04			3.5E+05	5.2E-01	1.2E-01	1.2E-01	9.5E-02	8.0E-02	1.2E-04	3.9E-03	4.1E-03	6.2E-02
GG-030	2.9E-08	2.0E+04	2.3E+04	2.2E+04	32.	2	6.9E+04	8.6E-01	3.8E-01	3.4E-01	2.1E-01	5.1E-02	1.1E-02	2.1E-03	8.3E-03	5.4E-02
			4.4E+04	8.6E+04			3.5E+05	1.4E-01	1.5E-01	1.8E-01	1.3E-01	5.4E-02	1.1E-06	1.8E-03	2.6E-03	3.5E-02
GG-031	8.4E-08	6.6E+03	1.8E+04	7.2E+04	32.	2	6.9E+04	7.1E-01	4.3E-02	3.1E-02	1.6E-02	5.6E-03	2.9E-03	6.4E-04	1.4E-03	6.1E-03
			9.0E+04	8.6E+04			3.5E+05	2.7E-01	2.0E-01	4.1E-02	2.4E-02	1.8E-02	4.5E-04	1.6E-03	3.1E-03	1.6E-02
GG-032	5.9E-09	2.0E+04	2.3E+04	2.2E+04	32.	2	6.9E+04	7.9E-01	3.7E-01	3.3E-01	2.2E-01	9.1E-02	1.8E-02	8.6E-03	5.0E-02	9.4E-02
			4.4E+04	8.6E+04			3.5E+05	2.1E-01	2.0E-01	2.3E-01	2.9E-01	3.8E-01	6.6E-03	3.3E-02	3.3E-02	3.3E-01

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Table C.3-1 (continued)

GG POS5 PARTITION DEFINITIONS

SOURCE TERM	FREQ (/YR)	TW (S)	T (S)	DT (S)	EL (M)	EVT	ENERGY (W)	RF1 XE/KR	RF2 I	RF3 CS	RF4 TE	RF5 SR	RF6 RU	RF7 LA	RF8 CE	RF9 BA
GG-033	4.0E-08	6.6E+03	1.8E+04	7.2E+04	32.	2	6.9E+04	8.2E-01	1.7E-01	1.5E-01	7.7E-02	1.5E-02	7.0E-03	1.4E-03	2.8E-03	1.7E-02
			9.0E+04	8.6E+04			3.5E+05	1.7E-01	1.3E-01	5.6E-02	4.0E-02	1.9E-02	1.8E-05	9.3E-04	1.7E-03	1.1E-02
GG-034	6.5E-08	2.0E+04	2.3E+04	2.2E+04	32.	2	6.9E+04	9.0E-01	4.8E-02	3.3E-02	1.7E-02	2.3E-03	6.8E-04	1.2E-04	4.6E-04	2.6E-03
			4.4E+04	8.6E+04			3.5E+05	1.0E-01	4.0E-02	4.2E-02	4.6E-02	2.9E-02	2.2E-04	1.6E-03	1.9E-03	2.1E-02
GG-035	9.8E-09	6.6E+03	1.8E+04	7.2E+04	32.	2	6.9E+04	8.4E-01	1.9E-01	1.8E-01	1.7E-01	1.4E-01	2.6E-02	1.3E-02	5.8E-02	1.4E-01
			9.0E+04	8.6E+04			3.5E+05	1.6E-01	9.9E-02	4.6E-02	4.3E-02	4.3E-02	4.9E-05	3.9E-03	7.6E-03	3.8E-02
GG-036	2.1E-08	2.0E+04	2.3E+04	2.2E+04	32.	2	6.9E+04	7.6E-01	5.0E-02	2.9E-02	9.8E-03	3.2E-04	2.1E-04	9.2E-06	1.8E-05	5.0E-04
			4.4E+04	8.6E+04			3.5E+05	2.4E-01	1.6E-01	1.6E-01	7.7E-02	1.2E-02	5.4E-08	3.2E-04	5.7E-04	7.9E-03
GG-037	8.1E-08	2.0E+04	2.3E+04	2.2E+04	32.	2	6.9E+04	6.2E-01	2.4E-02	1.6E-02	6.7E-03	8.1E-04	2.1E-04	3.3E-05	1.2E-04	9.3E-04
			4.4E+04	8.6E+04			3.5E+05	3.8E-01	4.4E-02	5.0E-02	2.5E-02	9.7E-03	1.5E-06	3.2E-04	4.8E-04	6.4E-03
GG-038	1.2E-08	2.9E+04	3.6E+04	5.9E+04	32.	2	8.6E+04	5.8E-01	7.5E-02	6.7E-02	4.3E-02	1.9E-02	4.3E-03	1.3E-03	3.7E-03	1.9E-02
			1.1E+05	8.6E+04			3.4E+05	4.1E-01	2.4E-01	8.4E-02	7.0E-02	5.2E-02	2.9E-04	2.2E-03	4.2E-03	3.7E-02
GG-039	1.7E-08	1.6E+04	2.6E+04	5.9E+04	32.	2	1.1E+06	8.8E-01	5.9E-02	4.3E-02	3.8E-02	2.8E-02	9.4E-03	2.9E-03	9.2E-03	2.9E-02
			8.6E+04	8.6E+04			3.5E+05	1.1E-01	1.1E-01	1.6E-02	1.9E-02	2.8E-02	2.0E-03	3.2E-03	5.9E-03	2.5E-02
GG-040	2.4E-08	1.2E+04	2.6E+04	6.8E+04	32.	2	1.1E+06	7.6E-01	1.1E-01	8.8E-02	3.7E-02	3.8E-03	3.0E-03	6.3E-04	7.5E-04	4.8E-03
			9.4E+04	8.6E+04			3.4E+05	2.2E-01	1.3E-01	6.8E-02	3.5E-02	9.6E-03	1.2E-05	3.0E-04	5.0E-04	4.8E-03
GG-041	4.2E-08	6.6E+03	1.8E+04	7.2E+04	32.	2	6.9E+04	7.0E-01	5.0E-02	3.6E-02	1.5E-02	2.8E-03	2.2E-03	4.2E-04	5.9E-04	3.3E-03
			9.0E+04	8.6E+04			3.5E+05	2.0E-01	9.9E-02	2.1E-02	5.1E-03	1.4E-03	8.1E-08	6.8E-05	1.2E-04	8.8E-04
GG-042	2.0E-07	2.1E+04	3.1E+04	6.3E+04	32.	2	3.5E+05	7.2E-01	8.3E-03	6.1E-03	2.9E-03	8.3E-04	6.6E-04	1.6E-04	2.1E-04	9.4E-04
			9.5E+04	8.6E+04			3.4E+05	9.3E-02	3.3E-02	4.3E-03	1.4E-03	3.8E-04	4.6E-06	6.6E-05	8.3E-05	2.4E-04
GG-043	2.6E-08	3.2E+04	4.0E+04	7.2E+04	32.	2	6.9E+04	6.4E-01	1.5E-02	1.6E-02	9.4E-03	5.8E-03	3.5E-03	9.4E-04	2.0E-03	6.3E-03
			1.1E+05	8.6E+04			3.5E+05	3.5E-01	2.9E-01	5.2E-02	2.8E-02	1.3E-02	1.1E-04	6.0E-04	1.1E-03	8.2E-03
GG-044	5.4E-08	1.6E+04	2.6E+04	5.8E+04	32.	2	6.9E+05	7.6E-01	2.6E-02	1.9E-02	5.5E-03	5.4E-04	4.3E-04	1.2E-04	1.4E-04	7.0E-04
			8.4E+04	8.6E+04			3.4E+05	8.0E-02	2.6E-02	1.8E-02	4.0E-03	2.7E-04	1.0E-06	1.1E-05	1.6E-05	1.4E-04
GG-045	4.2E-08	6.6E+03	1.8E+04	7.2E+04	32.	2	6.9E+04	6.2E-01	1.7E-02	1.1E-02	6.9E-03	2.7E-03	2.2E-03	4.0E-04	5.1E-04	3.0E-03
			9.0E+04	8.6E+04			3.5E+05	3.3E-01	1.3E-01	8.8E-03	3.6E-03	1.6E-03	1.4E-05	1.1E-04	2.1E-04	1.2E-03
GG-046	4.4E-08	3.2E+04	4.0E+04	7.2E+04	32.	2	6.9E+04	8.4E-01	6.0E-04	5.4E-04	2.5E-04	1.0E-04	4.1E-04	9.0E-05	8.4E-05	1.1E-04
			1.1E+05	8.6E+04			3.5E+05	1.2E-03	7.3E-03	3.3E-04	5.3E-05	8.5E-06	1.0E-12	1.0E-06	1.1E-06	5.5E-06
GG-047	2.5E-08	3.2E+04	4.0E+04	7.2E+04	32.	2	6.9E+04	5.3E-01	6.7E-03	6.8E-03	5.0E-03	3.3E-03	2.5E-03	5.8E-04	1.5E-03	3.6E-03
			1.1E+05	8.6E+04			3.5E+05	4.7E-01	1.7E-01	9.9E-03	4.2E-03	2.7E-03	2.2E-06	1.9E-04	3.4E-04	1.9E-03
GG-048	1.3E-08	3.2E+04	4.0E+04	7.2E+04	32.	2	6.9E+04	8.0E-01	1.2E-02	1.3E-02	6.4E-03	2.7E-03	2.5E-03	5.8E-04	6.7E-04	3.1E-03
			1.1E+05	8.6E+04			3.5E+05	1.6E-01	1.1E-01	4.6E-02	2.5E-02	1.0E-02	3.5E-08	3.3E-04	5.6E-04	6.2E-03

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Table C.3-1 (continued)

GG POS5 PARTITION DEFINITIONS

SOURCE TERM	FREQ (/YR)	TW (S)	T (S)	DT (S)	EL (M)	EVT	ENERGY (W)	RF1 XE/KR	RF2 I	RF3 CS	RF4 TE	RF5 SR	RF6 RU	RF7 LA	RF8 CE	RF9 BA
GG-049	4.9E-09	8.1E+03	2.6E+04	7.2E+04	32.	3	6.9E+04	8.1E-01	2.1E-01	2.0E-01	1.9E-01	1.8E-01	4.4E-02	2.0E-02	1.1E-01	1.8E-01
			9.8E+04	8.6E+04			3.5E+05	1.8E-01	1.2E-01	5.7E-02	5.2E-02	4.9E-02	6.0E-05	4.3E-03	8.5E-03	4.4E-02
GG-050	1.1E-08	8.1E+03	2.6E+04	7.2E+04	32.	3	6.9E+04	9.3E-01	1.8E-01	1.8E-01	1.5E-01	1.0E-01	1.8E-02	7.8E-03	3.6E-02	1.0E-01
			9.8E+04	8.6E+04			3.5E+05	7.2E-02	5.3E-02	3.3E-02	3.8E-02	4.8E-02	7.9E-05	3.4E-03	6.4E-03	3.8E-02
GG-051	2.4E-09	8.1E+03	2.6E+04	7.2E+04	32.	3	6.9E+04	4.9E-01	9.1E-02	7.2E-02	4.7E-02	4.0E-02	3.7E-02	7.9E-03	1.1E-02	4.1E-02
			9.8E+04	8.6E+04			3.5E+05	4.8E-01	3.3E-01	5.4E-02	5.1E-02	5.1E-02	2.3E-03	5.0E-03	1.0E-02	5.0E-02
GG-052	3.8E-07	8.1E+03	2.6E+04	7.2E+04	32.	3	6.9E+04	7.5E-01	8.0E-02	6.2E-02	3.0E-02	7.0E-03	3.8E-03	8.2E-04	1.5E-03	7.8E-03
			9.8E+04	8.6E+04			3.5E+05	2.2E-01	1.3E-01	4.6E-02	2.1E-02	1.0E-02	1.4E-04	6.2E-04	1.2E-03	7.0E-03
GG-053	3.0E-07	8.1E+03	2.6E+04	7.2E+04	32.	3	6.9E+04	6.7E-01	1.7E-02	1.2E-02	4.9E-03	1.2E-03	8.7E-04	1.9E-04	2.7E-04	1.4E-03
			9.8E+04	8.6E+04			3.5E+05	1.5E-01	8.6E-02	6.3E-03	2.2E-03	1.1E-03	1.0E-05	8.1E-05	1.6E-04	8.4E-04
GG-054	3.1E-08	8.1E+03	2.6E+04	7.2E+04	32.	3	6.9E+04	6.0E-01	3.0E-03	2.0E-03	7.2E-04	2.3E-04	1.7E-04	3.1E-05	4.3E-05	2.5E-04
			9.8E+04	8.6E+04			3.5E+05	6.2E-02	3.6E-02	4.3E-04	9.4E-05	2.3E-05	2.5E-08	1.5E-06	3.0E-06	1.8E-05
GG-055	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.	0	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
			0.0E+00	0.0E+00			0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

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Appendix D: Supporting Information for the Consequence Analysis

D.1 Radionuclide Inventories

A unique radionuclide inventory was generated with ORIGEN2 for each time window. The inventory for time window 1 corresponds to 7 hours after shutdown, the inventory for time window 2 corresponds to 24 hours after shutdown, and the inventory for time window 3 corresponds to 40 days after shutdown. The inventories, in the form of a input file for MACCS, for the sixty radionuclides considered in the MACCS code for time window 1, time window 2, and time window 3 are presented in Sections D.1.1, D.1.2, and D.1.3, respectively.

D.1.1 Inventory for Time Window 1

```
*
* GRAND GULF LOW POWER AND SHUTDOWN PRA
*  Grand Gulf Inventory for Time Window 1 (7 hrs after Shutdown)

*  Inventories obtained from ORIGENPC;   L. A. Miller (4/05/94)

*
*              NUCNAM      CORINV (BECQUERALs)
*
RDCORINV001    CO-58        1.207E+17
RDCORINV002    CO-60        8.362E+14
RDCORINV003    KR-85        2.656E+16
RDCORINV004    KR-85M       3.884E+17
RDCORINV005    KR-87        4.952E+16
RDCORINV006    KR-88        5.692E+17
RDCORINV007    RB-86        3.235E+15
RDCORINV008    SR-89        4.106E+18
RDCORINV009    SR-90        2.082E+17
RDCORINV010    SR-91        3.123E+18
RDCORINV011    SR-92        9.185E+17
RDCORINV012    Y-90         2.199E+17
RDCORINV013    Y-91         5.491E+18
RDCORINV014    Y-92         2.966E+18
RDCORINV015    Y-93         3.895E+18
RDCORINV016    ZR-95        7.042E+18
RDCORINV017    ZR-97        5.053E+18
RDCORINV018    NB-95        7.060E+18
RDCORINV019    MO-99        6.408E+18
RDCORINV020    TC-99M       5.902E+18
RDCORINV021    RU-103       5.109E+18
RDCORINV022    RU-105       1.057E+18
RDCORINV023    RU-106       1.080E+18
RDCORINV024    RH-105       2.761E+18
RDCORINV025    SB-127       3.268E+17
RDCORINV026    SB-129       3.602E+17
RDCORINV027    TE-127       3.336E+17
RDCORINV028    TE-127M      4.399E+16
RDCORINV029    TE-129       5.278E+17
RDCORINV030    TE-129M      1.614E+17
RDCORINV031    TE-131M      4.375E+17
RDCORINV032    TE-132       4.893E+18
RDCORINV033    I-131        3.564E+18
RDCORINV034    I-132        5.029E+18
RDCORINV035    I-133        6.228E+18
RDCORINV036    I-134        1.113E+17
RDCORINV037    I-135        3.437E+18
RDCORINV038    XE-133       7.652E+18
RDCORINV039    XE-135       3.401E+18
RDCORINV040    CS-134       1.831E+17
RDCORINV041    CS-136       1.000E+17
RDCORINV042    CS-137       2.544E+17
RDCORINV043    BA-139       2.341E+17
RDCORINV044    BA-140       6.630E+18
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RDCORINV045      LA-140        6.808E+18
RDCORINV046      LA-141        2.026E+18
RDCORINV047      LA-142        3.049E+17
RDCORINV048      CE-141        6.388E+18
RDCORINV049      CE-143        5.298E+18
RDCORINV050      CE-144        4.425E+18
RDCORINV051      PR-143        6.062E+18
RDCORINV052      ND-147        2.472E+18
RDCORINV053      NP-239        5.852E+19
RDCORINV054      PU-238        1.428E+15
RDCORINV055      PU-239        1.234E+15
RDCORINV056      PU-240        1.032E+15
RDCORINV057      PU-241        2.661E+17
RDCORINV058      AM-241        1.584E+14
RDCORINV059      CM-242        2.371E+16
RDCORINV060      CM-244        1.764E+14
```

D.1.2 Inventory for Time Window 2

```
*
* GRAND GULF LOW POWER AND SHUTDOWN PRA
*  Grand Gulf Inventory for Time Window 2 (24 hrs after Shutdown)

*  Inventories obtained from ORIGENPC;   L. A. Miller (4/05/94)

*
*           NUCNAM      CORINV (BECQUERALs)
*
RDCORINV001  CO-58      1.199E+17
RDCORINV002  CO-60      8.359E+14
RDCORINV003  KR-85      2.656E+16
RDCORINV004  KR-85M     2.799E+16
RDCORINV005  KR-87      4.686E+12
RDCORINV006  KR-88      8.966E+15
RDCORINV007  RB-86      3.149E+15
RDCORINV008  SR-89      4.064E+18
RDCORINV009  SR-90      2.082E+17
RDCORINV010  SR-91      9.034E+17
RDCORINV011  SR-92      1.188E+16
RDCORINV012   Y-90      2.180E+17
RDCORINV013   Y-91      5.461E+18
RDCORINV014   Y-92      1.749E+17
RDCORINV015   Y-93      1.213E+18
RDCORINV016  ZR-95      6.989E+18
RDCORINV017  ZR-97      2.517E+18
RDCORINV018  NB-95      7.057E+18
RDCORINV019  MO-99      5.361E+18
RDCORINV020  TC-99M     5.127E+18
RDCORINV021  RU-103     5.047E+18
RDCORINV022  RU-105     7.433E+16
RDCORINV023  RU-106     1.079E+18
RDCORINV024  RH-105     2.076E+18
RDCORINV025  SB-127     2.881E+17
RDCORINV026  SB-129     2.353E+16
RDCORINV027  TE-127     3.123E+17
RDCORINV028  TE-127M    4.399E+16
RDCORINV029  TE-129     1.317E+17
RDCORINV030  TE-129M    1.593E+17
RDCORINV031  TE-131M    2.954E+17
RDCORINV032  TE-132     4.209E+18
RDCORINV033   I-131     3.374E+18
RDCORINV034   I-132     4.336E+18
RDCORINV035   I-133     3.534E+18
RDCORINV036   I-134     1.961E+11
RDCORINV037   I-135     5.781E+17
RDCORINV038  XE-133     7.400E+18
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RDCORINV039  XE-135       1.899E+18
RDCORINV040  CS-134       1.830E+17
RDCORINV041  CS-136       9.632E+16
RDCORINV042  CS-137       2.544E+17
RDCORINV043  BA-139       4.535E+13
RDCORINV044  BA-140       6.382E+18
RDCORINV045  LA-140       6.731E+18
RDCORINV046  LA-141       1.011E+17
RDCORINV047  LA-142       1.486E+14
RDCORINV048  CE-141       6.302E+18
RDCORINV049  CE-143       3.706E+18
RDCORINV050  CE-144       4.419E+18
RDCORINV051  PR-143       6.003E+18
RDCORINV052  ND-147       2.364E+18
RDCORINV053  NP-239       4.751E+19
RDCORINV054  PU-238       1.432E+15
RDCORINV055  PU-239       1.237E+15
RDCORINV056  PU-240       1.032E+15
RDCORINV057  PU-241       2.661E+17
RDCORINV058  AM-241       1.592E+14
RDCORINV059  CM-242       2.373E+16
RDCORINV060  CM-244       1.764E+14
```

D.1.3 Inventory for Time Window 3

```
*
* GRAND GULF LOW POWER AND SHUTDOWN PRA
*  Grand Gulf Inventory for Time Window 3 (40 days after Shutdown)
*  Time Window 3 is after refueling, thus, only 2/3 of core used.
*  Inventories obtained from ORIGENPC;    L. A. Miller (4/05/94)
*
*           NUCNAM       CORINV (BECQUERALs)
*
RDCORINV001      CO-58         5.461E+16
RDCORINV002      CO-60         5.503E+14
RDCORINV003      KR-85         1.761E+16
RDCORINV004      KR-85M        0.000E+00
RDCORINV005      KR-87         0.000E+00
RDCORINV006      KR-88         0.000E+00
RDCORINV007      RB-86         4.938E+14
RDCORINV008      SR-89         1.589E+18
RDCORINV009      SR-90         1.386E+17
RDCORINV010      SR-91         0.000E+00
RDCORINV011      SR-92         0.000E+00
RDCORINV012      Y-90          1.387E+17
RDCORINV013      Y-91          2.300E+18
RDCORINV014      Y-92          0.000E+00
RDCORINV015      Y-93          0.000E+00
RDCORINV016      ZR-95         3.059E+18
RDCORINV017      ZR-97         3.572E+01
RDCORINV018      NB-95         4.171E+18
RDCORINV019      MO-99         1.926E+14
RDCORINV020      TC-99M        1.855E+14
RDCORINV021      RU-103        1.692E+18
RDCORINV022      RU-105        0.000E+00
RDCORINV023      RU-106        6.692E+17
RDCORINV024      RH-105        1.499E+10
RDCORINV025      SB-127        1.715E+14
RDCORINV026      SB-129        0.000E+00
RDCORINV027      TE-127        2.335E+16
RDCORINV028      TE-127M       2.367E+16
RDCORINV029      TE-129        3.096E+16
RDCORINV030      TE-129M       4.758E+16
RDCORINV031      TE-131M       7.994E+07
RDCORINV032      TE-132        7.000E+14
RDCORINV033      I-131         7.931E+16
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RDCORINV034	I-132	7.214E+14
RDCORINV035	I-133	6.706E+04
RDCORINV036	I-134	0.000E+00
RDCORINV037	I-135	0.000E+00
RDCORINV038	XE-133	3.189E+16
RDCORINV039	XE-135	0.000E+00
RDCORINV040	CS-134	1.178E+17
RDCORINV041	CS-136	8.168E+15
RDCORINV042	CS-137	1.694E+17
RDCORINV043	BA-139	0.000E+00
RDCORINV044	BA-140	5.145E+17
RDCORINV045	LA-140	5.921E+17
RDCORINV046	LA-141	0.000E+00
RDCORINV047	LA-142	0.000E+00
RDCORINV048	CE-141	1.832E+18
RDCORINV049	CE-143	7.162E+09
RDCORINV050	CE-144	2.681E+18
RDCORINV051	PR-143	5.842E+17
RDCORINV052	ND-147	1.370E+17
RDCORINV053	NP-239	3.317E+14
RDCORINV054	PU-238	9.780E+14
RDCORINV055	PU-239	8.344E+14
RDCORINV056	PU-240	6.886E+14
RDCORINV057	PU-241	1.767E+17
RDCORINV058	AM-241	1.366E+14
RDCORINV059	CM-242	1.347E+16
RDCORINV060	CM-244	1.172E+14

D.2 Listing of Code Used to Estimate Onsite Consequences

```
      PROGRAM ONSITE5
C   CALCULATE PARKING LOT DOSES DUE TO IMMERSION AND INHALATION AT
C   VARIOUS DISTANCES AT GRAND GULF BASED ON SOURCE TERMS PRODUCED
C   BY GGSOR FOLLOWED BY PARTITION
C
C   NTOT = NUMBER OF SOURCE TERMS EVALUATED
C   RF(I,J) = RELEASE FRACTION TO ENVIRONMENT FOR ITH CHEMICAL GROUP AND
C             JTH SEGMENT
C   T1 = TIME AT WHICH FIRST RELEASE TO ENVIRONMENT BEGINS
C   DT1 = DURATION OF FIRST RELEASE
C   T2 = TIME AT WHICH SECOND RELEASE TO ENVIRONMENT BEGINS
C   DT2 = DURATION OF SECOND RELEASE
C   BR = BREATHING RATE (CURRENTLY HARD-WIRED, NUMBER OBTAINED FROM 4551
C        FOR A MAN BREATHING HEAVILY)
C   AREA = MINIMUM AREA OF BUILDING, TAKEN AS MINIMUM WIDTH
C          OF AUX BUILDING TIMES HEIGHT OF CONTAINMENT (M^2)
C   ISTAB = STABILITY CLASS (1 = A, 2 = B, ..., 6 = F)
C   U = WIND SPEED AT 10 M HEIGHT (M/S) (CURRENTLY HARD-WIRED)
C   ICHEM(I) = CHEMICAL GROUP THAT ITH RADIONUCLIDE BELONGS TO
C   XINV(I,J) = INVENTORY OF ITH RADIONUCLIDE AND JTH EVENT TYPE
C               (SHUTDOWN TIME) AS DETERMINED WITH ORIGEN2
C   NUCNAM(I) = NUCLIDE NAME
C   CHIQ(I,J,K) = RELATIVE CONCENTRATION CORRESPONDING TO ITH DISTANCE,
C                 JTH MET, AND KTH MODEL (1=RAMSDELL MODEL, 2=WILSON/REG GUIDE),
C   DCFIMM(I) = DOSE CONVERSION FACTOR FOR IMMERSION FOR ITH NUCLIDE
C               IN (SV/DAY)/(BQ/M^3)
C   DCFINH(I) = DOSE CONVERSION FACTOR FOR INHALATION FOR ITH NUCLIDE
C               IN (SV/BQ)
C   DCFGRD(I) = DOSE CONVERSION FACTOR FOR GROUNDSHINE FOR ITH NUCLIDE
C               IN (SV/DAY)/(BQ/M^2)
C   DCFING(I) = DOSE CONVERSION FACTOR FOR INGESTION FOR ITH NUCLIDE
C               IN (SV/BQ)
C
      PARAMETER (MAXGRP=10, MAXMET=2, MAXMOD=2, MAXISO=100, MAXDST=10,
     1           MAXSEG=2, MAXEVT=3, MAXSTB=6)
      COMMON /DUM1/ FREQ, CPROB, TEVAC, DEVAC, TW, T1, T2, DT2, ELEV,
     1              E1, E2
      CHARACTER*80 TITLE
      COMMON /DUM2/ TITLE
      DIMENSION RF(MAXGRP,MAXSEG), ICHEM(MAXISO), XINV(MAXISO,MAXEVT),
     1          CHIQ(MAXDST,MAXMET,MAXMOD),
     2          DOSEXT(2,MAXSEG,MAXDST,MAXMET,MAXMOD),
     3          DOSINH(2,MAXSEG,MAXDST,MAXMET,MAXMOD),
     4          DOSE(2,MAXDST,MAXMET,MAXMOD),
     5          DOSRAT(MAXSEG,MAXDST,MAXMET,MAXMOD),
     6          DCFIMM(MAXISO), DCFINH(MAXISO), DCFGRD(MAXISO),
     7          DCFING(MAXISO)
      DIMENSION XDIST(MAXDST), YSIGA(MAXSTB), YSIGB(MAXSTB),
     1          ZSIGA(MAXSTB), ZSIGB(MAXSTB)
      CHARACTER*7 NUCNAM(MAXISO)
      CHARACTER*140 REC
      DATA ZERO / 0.0 /
C*****DISTANCES (M)
      DATA NDIST / 5 /
      DATA XDIST / 10., 50., 100., 250., 500., 5*0.0 /
C*****DISPERSION PARAMETERS FOR REG GUIDE 1.145 MODEL
      DATA YSIGA / 0.3658, 0.2751, 0.2089, 0.1474, 0.1046, 0.0722 /
      DATA YSIGB / 6*0.9031 /
      DATA ZSIGA / 2.5E-4, 1.9E-3, 0.2, 0.3, 0.4, 0.2 /
      DATA ZSIGB / 2.125, 1.6021, 0.8543, 0.6532, 0.6021, 0.6020 /
C*****SET PI AS CONSTANT
      DATA PI / 3.141593 /
C*****BREATHING RATE (M3/S)
      DATA BR / 2.66E-4 /
```

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```
C*****GG REACTOR POWER LEVEL
      DATA PWRLVL / 1.071 /
C
C
C*****OPEN OUTPUT FILE
      OPEN (6, FILE='ONSITE5.OUT')
C*****OPEN NUCLIDE LIST WITH CHEMICAL GROUP INDICES
      OPEN (1, FILE='CHEMGRP.DAT')
      ISO=0
 1100 CONTINUE
      READ(1,1001,END=1500) REC
      IF (REC(1:8) .NE. 'ISOTPGRP') GO TO 1100
      IC=11
 1200 CONTINUE
      IC=IC + 1
      IF (REC(IC:IC) .EQ. ' ') GO TO 1200
      ISO=ISO + 1
      NUCNAM(ISO)=REC(IC:IC+6)
      IC=IC + 7
 1300 CONTINUE
      IC=IC + 1
      IF (REC(IC:IC) .EQ. ' ') GO TO 1300
 1400 CONTINUE
      IC=IC + 1
      IF (REC(IC:IC) .NE. ' ') GO TO 1400
      READ(REC(IC:),*) ICHEM(ISO)
      GO TO 1100
 1500 CONTINUE
      CLOSE (1)
      NISO=ISO
C*****READ INVENTORIES FOR DIFFERENT SHUTDOWN TIMES
      DO 2000 IFILE=1,3
         IF (IFILE .EQ. 1) THEN
            OPEN (1, FILE='INV1')
         ELSE IF (IFILE .EQ. 2) THEN
            OPEN (1, FILE='INV2')
         ELSE
            OPEN (1, FILE='INV3')
         ENDIF
 1600    CONTINUE
         READ(1,1001,END=1950) REC
         IF (REC(1:8) .NE. 'RDCORINV') GO TO 1600
         IC=11
 1700    CONTINUE
         IC=IC + 1
         IF (REC(IC:IC) .EQ. ' ') GO TO 1700
         DO 1800 ISO=1,60
            IF (REC(IC:IC+6) .EQ. NUCNAM(ISO)) THEN
               READ(REC(IC+8:),*) XINV(ISO,IFILE)
               XINV(ISO,IFILE)=PWRLVL * XINV(ISO,IFILE)
               GO TO 1900
            ENDIF
 1800    CONTINUE
         WRITE(*,*) '>>>>>ISOTOPE INVENTORIES DO NOT MATCH'
         WRITE(6,*) '>>>>>ISOTOPE INVENTORIES DO NOT MATCH'
         STOP
 1900    CONTINUE
         GO TO 1600
 1950    CONTINUE
         CLOSE (1)
 2000 CONTINUE
      WRITE(6,2001) (I,I=1,3)
      DO 2200 ISO=1,NISO
         WRITE(6,2002) NUCNAM(ISO), (XINV(ISO,IEV),IEV=1,3)
 2200 CONTINUE
C*****READ DOSE CONVERSION FACTORS FOR THE INVENTORY
      OPEN (10, FILE='INDEXR.DAT')
```

```
      OPEN (11, FILE='DFNRC.INH')
      OPEN (12, FILE='DFNRC.ING')
      OPEN (13, FILE='DFNRC.SUB')
      OPEN (14, FILE='DFNRC.GRD')
      WRITE(*,2101)
      WRITE(6,2301)
      DO 2100 I=1,NISO
         WRITE(*,2102) I, NISO
         CALL GETDF (NUCNAM(I), DCFIMM(I), DCFGRD(I), DCFINH(I),
     *               DCFING(I))
 2100 CONTINUE
      CLOSE (10)
      CLOSE (11)
      CLOSE (12)
      CLOSE (13)
      CLOSE (14)
C*****DETERMINE CHI/Q FOR EACH DISTANCE, METEOROLOGY, AND MODEL
      AREA=2902.
      DO 2400 J=1,2
         IF (J .EQ. 1) THEN
C***********A-STABILITY, 5 M/S
            ISTAB=1
            U=5.0
         ELSE
C***********F-STABILITY, 1 M/S
            ISTAB=6
            U=1.0
         ENDIF
C********DETERMINE CHI/Q FOR EACH DISTANCE
         DO 2300 I=1,NDIST
            X=XDIST(I)
C***********RAMSDELL MODEL
            CHIQ(I,J,1)=84.5 * (X**(-1.13)) * (AREA**(-1.25)) *
     *                  (U**.720) * (ISTAB**.473)
            IF (X .LT. 100.) THEN
C**************WILSON MODEL (<100M, INCLUDE 5 AS MULTIPLIER FOR LOW RELEASES)
               CHIQ(I,J,2)=5.0 / (0.11*U*X**2.)
            ELSE
C**************REG GUIDE 1.145 MODEL (>=100M)
               SIGY=YSIGA(ISTAB) * X**YSIGB(ISTAB)
               SIGZ=ZSIGA(ISTAB) * X**ZSIGB(ISTAB)
               IF (X .LE. 800.) THEN
                  CAPSIGY=4.0 * SIGY
               ELSE
                  CAPSIGY=3.0 * YSIGA(ISTAB)*800.**YSIGB(ISTAB) + SIGY
               ENDIF
               CHIQ1=1.0 / (U*(PI*SIGY*SIGZ + AREA/2.))
               CHIQ2=1.0 / (U*(3*PI*SIGY*SIGZ))
               CHIQ3=1.0 / (U*PI*CAPSIGY*SIGZ)
               CHIQ(I,J,2) = MIN (MAX (CHIQ1, CHIQ2), CHIQ3)
            ENDIF
 2300    CONTINUE
 2400 CONTINUE
      WRITE(6,2501)
      DO 2500 K=1,NDIST
         WRITE(6,2502) NINT(XDIST(K)), ((CHIQ(K,L,M),M=1,2),L=1,2)
 2500 CONTINUE
C*****OPEN RESULT FILE FOR RAMSDELL MODEL
      OPEN (1, FILE='GGR.OUT')
      OPEN (11, FILE='GGRCON.OUT')
      WRITE(1,3001) 'RAMSDELL'
      WRITE(11,4001) 'RAMSDELL'
C*****OPEN RESULT FILE FOR WILSON/REG GUIDE MODEL
      OPEN (2, FILE='GGW.OUT')
      OPEN (12, FILE='GGWCON.OUT')
      WRITE(2,3001) 'WILSON/REG GUIDE'
      WRITE(12,4001) 'WILSON/REG GUIDE'
```

```
C*****SOURCE TERM FILE CREATED BY PARTITIONING
      OPEN (3, FILE='MACCS.INP')
C*****READ SOURCE TERMS PRODUCED BY GGSOR5 FOLLOWING PARTITIONING
      READ(3,1001) TITLE
      READ(3,*) NFRC, NSEG, NPAR
      WRITE(*,2401)
      DO 9000 IPAR=1,NPAR
         WRITE(*,2402) IPAR, NPAR
         READ(3,*) IPART, FREQ, CPROB
         IF (IPAR .NE. IPART) THEN
            WRITE(*,*) '>>>>>ERROR IN READING PARTITION SOURCE TERMS'
            WRITE(6,*) '>>>>>ERROR IN READING PARTITION SOURCE TERMS'
            STOP
         ENDIF
         READ(3,*) TEVAC, DEVAC, TW, T1, DT1, T2, DT2, ELEV, EVTYPE,
     1             E1, (RF(I,1),I=1,NFRC), E2, (RF(I,2),I=1,NFRC)
         IEV=NINT(EVTYPE)
C********INITIALIZE DOSE RATES AND DOSES
         DO 3000 M=1,2
            DO 2900 L=1,2
               DO 2800 K=1,NDIST
                  DO 2700 J=1,NSEG
                     DO 2600 I=1,2
                        DOSEXT(I,J,K,L,M)=0.0
                        DOSINH(I,J,K,L,M)=0.0
 2600                CONTINUE
 2700             CONTINUE
 2800          CONTINUE
 2900       CONTINUE
 3000    CONTINUE
C********CALCULATE TIME-INTEGRATED CONCENTRATION OF EACH CHEMICAL
C********GROUP AT THE VARIOUS DISTANCES
C********LOOP OVER MODELS
         DO 3500 M=1,2
C***********LOOP OVER METEOROLOGY
            DO 3400 L=1,2
C**************LOOP OVER DISTANCE
               DO 3300 K=1,NDIST
C*****************LOOP OVER RELEASE SEGMENTS
                  DO 3200 J=1,NSEG
C********************LOOP OVER ISOTOPE INVENTORY
                     DO 3100 I=1,NISO
C***********************TOTAL EXPOSURE (IMMERSION + INHALATION)
                        DOSEXT(2,J,K,L,M)=DOSEXT(2,J,K,L,M) +
     1           RF(ICHEM(I),J)*XINV(I,IEV)*CHIQ(K,L,M)*DCFIMM(I)/86400.
                        DOSINH(2,J,K,L,M)=DOSINH(2,J,K,L,M) +
     1           RF(ICHEM(I),J)*XINV(I,IEV)*CHIQ(K,L,M)*DCFINH(I)*BR
 3100                CONTINUE
 3200             CONTINUE
C*****************15-MIN EXPOSURE (IMMERSION + INHALATION)
                  IF (DT1 .GE. 900.) THEN
                     DOSEXT(1,1,K,L,M)=DOSEXT(2,1,K,L,M) * 900./DT1
                     DOSINH(1,1,K,L,M)=DOSINH(2,1,K,L,M) * 900./DT1
                  ELSE
                     IF (DT1 .GT. 0.0) THEN
                        DOSEXT(1,1,K,L,M)=DOSEXT(2,1,K,L,M)
                        DOSINH(1,1,K,L,M)=DOSINH(2,1,K,L,M)
                     ENDIF
                     IF (DT2 .GT. 0.0) THEN
                        DOSEXT(1,1,K,L,M)=DOSEXT(1,1,K,L,M) +
     1                           DOSEXT(2,2,K,L,M) * (900.-DT1)/DT2
                        DOSINH(1,1,K,L,M)=DOSINH(1,1,K,L,M) +
     1                           DOSINH(2,2,K,L,M) * (900.-DT1)/DT2
                     ENDIF
                  ENDIF
C*****************CALCULATE DOSE RATE FOR EACH RELEASE SEGMENT
                  IF (DT1 .GT. 0.0) THEN
```

```
                     DOSRAT(1,K,L,M)=DOSEXT(2,1,K,L,M) / DT1
                  ENDIF
                  IF (DT2 .GT. 0.0) THEN
                     DOSRAT(2,K,L,M)=DOSEXT(2,2,K,L,M) / DT2
                  ENDIF
C*****************CALCULATE 15-MIN DOSE
                  DOSE(1,K,L,M)=DOSEXT(1,1,K,L,M) + DOSINH(1,1,K,L,M)
C*****************CALCULATE TOTAL DOSE
                  DOSE(2,K,L,M)=DOSEXT(2,1,K,L,M) + DOSINH(2,1,K,L,M) +
     1                          DOSEXT(2,2,K,L,M) + DOSINH(2,2,K,L,M)
 3300           CONTINUE
 3400        CONTINUE
 3500     CONTINUE
C*********WRITE DOSES AND DOSE RATES FOR EACH DISTANCE, METEOROLOGY,
C*********AND MODEL
          DO 4000 M=1,2
             WRITE(M,3002) 'GG', IPAR, (NINT(XDIST(K)),
     1                     ((DOSRAT(J,K,L,M),J=1,2),
     2                     (DOSE(J,K,L,M),J=1,2),L=1,2),K=1,NDIST)
             WRITE(M+10,4002) 'GG', IPAR,
     1                    (((DOSRAT(J,K,L,M),J=1,2),
     2                     (DOSE(J,K,L,M),J=1,2),L=1,2),K=1,NDIST)
 4000     CONTINUE
 9000 CONTINUE
C*********WRITE ZERO SOURCE TERM PARTITION DOSES AND DOSE RATES FOR
C*********EACH DISTANCE, METEOROLOGY, AND MODEL
          DO 9100 M=1,2
             WRITE(M,3002) 'GG', NPAR+1, (NINT(XDIST(K)),
     1                     ((ZERO,J=1,2), (ZERO,J=1,2),L=1,2),K=1,NDIST)
             WRITE(M+10,4002) 'GG', NPAR+1,
     1                    (((ZERO,J=1,2), (ZERO,J=1,2),L=1,2),K=1,NDIST)
 9100     CONTINUE
      CLOSE (1)
      CLOSE (2)
      CLOSE (3)
      CLOSE (11)
      CLOSE (12)
C*****FORMAT STATEMENTS
 1001 FORMAT(A)
 2001 FORMAT(/5X,'INVENTORIES (BQ) FOR EACH EVENT TYPE',
     1       /1X,'NUCLIDE',4X,3('EVENT',I2,5X))
 2002 FORMAT(1X,A,1P3E12.3)
 2101 FORMAT(' PROCESSING DOSE FACTORS FOR ISOTOPE')
 2102 FORMAT('+PROCESSING DOSE FACTORS FOR ISOTOPE ',I3,' OUT OF ',I3)
 2301 FORMAT(/20X,'DOSE FACTORS')
 2401 FORMAT(' PROCESSING PARTITION SOURCE TERM')
 2402 FORMAT('+PROCESSING PARTITION SOURCE TERM ',I3,' OUT OF ',I3)
 2501 FORMAT(/25X,'CHI/Q VALUES',
     1       /16X,'A STABILITY, 5 M/S',6X,'F STABILITY, 1 M/S',
     2       /2X,'DIST (M)',2(5X,'RAMSDELL',2X,'WILSON/REG'))
 2502 FORMAT(I6,2X,2(5X,1P2E10.2))
 3001 FORMAT(50X,A,' MODEL',
     1      /28X,'A STABILITY, 5 M/S',24X,'F STABILITY, 1 M/S',
     2      /2X,'SOURCE',2X,'DISTANCE',2(2X,'REL1',6X,'REL2',5X,
     3       '15-MIN',5X,'TOTAL',5X),
     4      /3X,'TERM',5X,'(M)',1X,2(1X,'DOSE RATE',2X,'DOSE RATE',3X,
     5       'DOSE',6X,'DOSE',4X),
     6      /1X,'PARTITION',6X,2(3X,'(SV/S)',4X,'(SV/S)',5X,
     7       '(SV)',6X,'(SV)',4X),/)
 3002 FORMAT(2X,A,'-',I3.3,I6,2X,1P4E10.2,2X,1P4E10.2,
     1      /(8X,I6,2X,1P4E10.2,2X,1P4E10.2))
 4001 FORMAT(A,' MODEL')
 4002 FORMAT(A,'-',I3.3,
     1      /(1PE10.2))
      END
      SUBROUTINE GETDF (NUCNAM, DCLD, DGRD, DINH, DING)
      CHARACTER*(*) NUCNAM
```

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```
      CHARACTER*7 NEWNAM
      CHARACTER*26 UPPER, LOWER
      DATA UPPER /'ABCDEFGHIJKLMNOPQRSTUVWXYZ'/
      DATA LOWER /'abcdefghijklmnopqrstuvwxyz'/
C
C
      NEWNAM=NUCNAM
      DO 1000 IC=2,7
         IP=INDEX (UPPER, NEWNAM(IC:IC))
         IF (IP .GT. 0) NEWNAM(IC:IC)=LOWER(IP:IP)
 1000 CONTINUE
      DCLD = 0.0
      DGRD = 0.0
      DINH = 0.0
      DING = 0.0
      CALL READIT (NEWNAM, DCLD, DGRD, DINH, DING)
      RETURN
      END
      SUBROUTINE READIT (NEWNAM, DCLD, DGRD, DINH, DING)
C
* MAIN    READID.FOR
* Program illustrates the use of INDEXR.DAT file to coordinate the reading
* of the dose factor files DFNRC.INH, DFNRC.ING, DFNRC.SUB, and DFNRC.GRD.
* All files are direct access formatted files.  The INDEXR.DAT file has a
* record length 102, the others are 80.  See the OPEN statements below for
* other details.                      K.F. Eckerman  4/05/89
*                                     Last mods      5/22/90.
* The file INDEXR.DAT contains the following information:
*
* Variable          Description              Format
* Nuke              Name of nuclide            A7
* T                 Halflife                   A8
* IX                Halflife units             A2
* Mode              Decay Modes                A6
* D1                Name of daughter           A7
* F1                Branching fraction         E10
* D2                Name of daughter           A7
* F2                Branching fraction         E10
* D3                Name of daughter           A7
* F3                Branching fraction         E10
* ID()              Pointers into files        7I4
* ID(1)             Record # of D1 in INDEXR.DAT
* ID(2)             Record # of D2 "             "
* ID(3)             Record # of D3 "             "
* ID(4)             Record # of Nuke in DFNRC.INH
* ID(5)             "                 " DFNRC.ING
* ID(6)             "                 " DFNRC.SUB
* ID(7)             "                 " DFNRC.GRD
*
* Local variables
      CHARACTER*8 T
      CHARACTER*7 Nuke, D1, D2, D3, Nuke1, Nuke2, Nuke3, Nuke4
      CHARACTER*7 NEWNAM
      CHARACTER*6 Mode
      CHARACTER*2 IX
      COMMON /DUMMY1/ D1, D2, D3, NUKE1, NUKE2, NUKE3, NUKE4, MODE
      COMMON /DUMMY2/ F1, F2, F3, IX, T
      DIMENSION DFinh(8),DFing(8), DFsub(8), Dfgrd(8)
      DIMENSION ID(7)
      PARAMETER(Idev=10,Idf1=11,Idf2=12,Idf3=13,Idf4=14,Igrd=4,Isub=4,
     .          Iing=4,Iinh=4,Ngrd=826,Nsub=826,Ning=738,Ninh=738,
     .          Ifile=1, Nfile=825)
C       Search dose factor data files for user-specified nuclide and
C       print dose factors for inhalation, ingestion, submersion and
C       ground plane.
C       Organs: Gonads, Breast, Lung, R. Marrow,
C               Bone Surface, Thyroid, Remainder, and Effective.
```

```
C       Units are:
C           Inhalation & Ingestion: Committed dose per unit intake (Sv/Bq).
C           Submersion:  Dose rate (Sv/d per Bq/m3).
C           Ground plane:  Dose rate (Sv/d per Bq/m2).
C
C
      DO 100 J=1,8
         DFinh(J)=0.0
         DFing(J)=0.0
         DFsub(J)=0.0
         DFgrd(J)=0.0
  100 CONTINUE
      REWIND 10
      REWIND 11
      REWIND 12
      REWIND 13
      REWIND 14
 1000 CONTINUE
      READ(10,'(A7,A8,A2,A6,3(A7,E10.0),7I4)',END=9000)
     .         Nuke, T, IX, MODE, D1, F1, D2, F2, D3, F3, (ID(J),J=1,7)
      IF (Nuke .NE. NEWNAM) GO TO 1000
      IF (ID(4) .NE. 0) THEN
         DO 2000 I=1,ID(4)-1
            READ(11,*)
 2000    CONTINUE
         READ(11,'(1x,A7,1P8E9.2)') Nuke1, (DFinh(j), j = 1, 8)
      ENDIF
      IF (ID(5) .NE. 0) THEN
         DO 3000 I=1,ID(5)-1
            READ(12,*)
 3000    CONTINUE
         READ(12,'(1x,A7,1P8E9.2)') Nuke2, (DFing(j),j=1,8)
      ENDIF
      IF (ID(6) .NE. 0) THEN
         DO 4000 I=1,ID(6)-1
            READ(13,*)
 4000    CONTINUE
         READ(13,'(1x,A7,1P8E9.2)') Nuke3, (DFsub(j),j=1,8)
      ENDIF
      IF (ID(7) .NE. 0) THEN
         DO 5000 I=1,ID(7)-1
            READ(14,*)
 5000    CONTINUE
         READ(14,'(1x,A7,1P8E9.2)') Nuke4, (DFgrd(j),j=1,8)
      ENDIF
      DCLD=DFsub(8)
      DGRD=DFgrd(8)
      DINH=DFinh(8)
      DING=DFing(8)
      WRITE(6,81) NUKE, DCLD, DGRD, DINH, DING
   81 FORMAT(1X,A,' CLD=',1PE8.2,', GRD=',E8.2,', INH=',E8.2,
     1       ', ING=',E8.2)
      RETURN
 9000 CONTINUE
C*****NUCLIDE NOT LOCATED
      WRITE(*,*) '>>>>>NUCLIDE NOT LOCATED--', NEWNAM
      WRITE(6,*) '>>>>>NUCLIDE NOT LOCATED--', NEWNAM
      STOP
      END
```

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D.3 Onsite Consequences for Source Term Groups

Doses and dose rates were calculated for a variety of distances from the containment using the Wilson/Reg. Guide Model and the Ramsdell Model as described in Section 8. These calculations were performed using the mean source term defined for each source term group. The doses and dose rates calculated using the Wilson/Reg. Guide 1.145 model are presented in Table D.3-1. Similar results calculated using the Ramsdell Model are presented in Table D.3-2.

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TABLE D.3-1. Dose and Dose Rates Calculated Using Wilson/Reg. Guide Model

SOURCE TERM PARTITION	DISTANCE (M)	A STABILITY, 5 M/S REL1 DOSE RATE (SV/S)	REL2 DOSE RATE (SV/S)	15-MIN DOSE (SV)	TOTAL DOSE (SV)	F STABILITY, 1 M/S REL1 DOSE RATE (SV/S)	REL2 DOSE RATE (SV/S)	15-MIN DOSE (SV)	TOTAL DOSE (SV)
GG-001	10	4.26E-01	2.09E-01	6.70E+03	8.84E+05	2.13E+00	1.04E+00	3.35E+04	4.42E+06
	50	1.70E-02	8.36E-03	2.68E+02	3.54E+04	8.51E-02	4.18E-02	1.34E+03	1.77E+05
	100	7.16E-04	3.51E-04	1.13E+01	1.49E+03	2.52E-02	1.24E-02	3.97E+02	5.24E+04
	250	4.46E-05	2.19E-05	7.03E-01	9.28E+01	6.35E-03	3.11E-03	1.00E+02	1.32E+04
	500	5.47E-06	2.69E-06	8.62E-02	1.14E+01	2.24E-03	1.10E-03	3.52E+01	4.65E+03
GG-002	10	4.88E-01	1.73E-01	1.30E+04	1.56E+06	2.44E+00	8.64E-01	6.51E+04	7.82E+06
	50	1.95E-02	6.91E-03	5.21E+02	6.25E+04	9.76E-02	3.46E-02	2.61E+03	3.13E+05
	100	8.21E-04	2.91E-04	2.19E+01	2.63E+03	2.89E-02	1.02E-02	7.71E+02	9.26E+04
	250	5.12E-05	1.81E-05	1.37E+00	1.64E+02	7.28E-03	2.58E-03	1.94E+02	2.33E+04
	500	6.28E-06	2.22E-06	1.68E-01	2.01E+01	2.56E-03	9.08E-04	6.84E+01	8.21E+03
GG-003	10	2.03E-01	2.57E-01	2.05E+03	7.60E+05	1.01E+00	1.29E+00	1.02E+04	3.80E+06
	50	8.11E-03	1.03E-02	8.19E+01	3.04E+04	4.06E-02	5.15E-02	4.09E+02	1.52E+05
	100	3.41E-04	4.33E-04	3.44E+00	1.28E+03	1.20E-02	1.52E-02	1.21E+02	4.50E+04
	250	2.13E-05	2.70E-05	2.15E-01	7.97E+01	3.02E-03	3.84E-03	3.05E+01	1.13E+04
	500	2.61E-06	3.31E-06	2.63E-02	9.78E+00	1.07E-03	1.35E-03	1.08E+01	3.99E+03
GG-004	10	1.29E+00	1.10E-01	6.92E+04	3.49E+06	6.45E+00	5.52E-01	3.46E+05	1.74E+07
	50	5.16E-02	4.41E-03	2.77E+03	1.40E+05	2.58E-01	2.21E-02	1.38E+04	6.98E+05
	100	2.17E-03	1.86E-04	1.16E+02	5.87E+03	7.63E-02	6.53E-03	4.10E+03	2.07E+05
	250	1.35E-04	1.16E-05	7.26E+00	3.66E+02	1.92E-02	1.65E-03	1.03E+03	5.20E+04
	500	1.66E-05	1.42E-06	8.91E-01	4.49E+01	6.77E-03	5.80E-04	3.64E+02	1.83E+04
GG-005	10	7.77E-01	2.81E-01	1.32E+04	1.30E+06	3.88E+00	1.41E+00	6.62E+04	6.49E+06
	50	3.11E-02	1.13E-02	5.30E+02	5.19E+04	1.55E-01	5.63E-02	2.65E+03	2.60E+05
	100	1.31E-03	4.73E-04	2.23E+01	2.18E+03	4.60E-02	1.67E-02	7.84E+02	7.69E+04
	250	8.15E-05	2.95E-05	1.39E+00	1.36E+02	1.16E-02	4.20E-03	1.97E+02	1.94E+04
	500	9.99E-06	3.62E-06	1.70E-01	1.67E+01	4.08E-03	1.48E-03	6.96E+01	6.82E+03
GG-006	10	1.24E-01	1.80E-01	1.65E+03	3.66E+05	6.21E-01	8.99E-01	8.27E+03	1.83E+06
	50	4.97E-03	7.19E-03	6.62E+01	1.46E+04	2.48E-02	3.60E-02	3.31E+02	7.32E+04
	100	2.09E-04	3.03E-04	2.78E+00	6.15E+02	7.35E-03	1.06E-02	9.80E+01	2.17E+04
	250	1.30E-05	1.89E-05	1.74E-01	3.84E+01	1.85E-03	2.68E-03	2.47E+01	5.45E+03
	500	1.60E-06	2.31E-06	2.13E-02	4.71E+00	6.52E-04	9.45E-04	8.69E+00	1.92E+03
GG-007	10	1.51E-01	2.01E-01	2.54E+03	8.94E+05	7.54E-01	1.01E+00	1.27E+04	4.47E+06
	50	6.03E-03	8.05E-03	1.02E+02	3.58E+04	3.02E-02	4.03E-02	5.08E+02	1.79E+05
	100	2.54E-04	3.39E-04	4.27E+00	1.50E+03	8.93E-03	1.19E-02	1.50E+02	5.30E+04
	250	1.59E-05	2.11E-05	2.66E-01	9.38E+01	2.25E-03	3.00E-03	3.79E+01	1.33E+04
	500	1.94E-06	2.59E-06	3.27E-02	1.15E+01	7.92E-04	1.06E-03	1.33E+01	4.70E+03
GG-008	10	2.74E-01	1.39E-01	4.93E+03	6.48E+05	1.37E+00	6.97E-01	2.47E+04	3.24E+06
	50	1.10E-02	5.58E-03	1.97E+02	2.59E+04	5.49E-02	2.79E-02	9.86E+02	1.30E+05
	100	4.62E-04	2.35E-04	8.29E+00	1.09E+03	1.63E-02	8.26E-03	2.92E+02	3.83E+04
	250	2.88E-05	1.46E-05	5.17E-01	6.79E+01	4.09E-03	2.08E-03	7.35E+01	9.65E+03
	500	3.53E-06	1.79E-06	6.34E-02	8.33E+00	1.44E-03	7.32E-04	2.59E+01	3.40E+03
GG-009	10	7.84E-01	2.60E-01	4.26E+04	1.82E+06	3.92E+00	1.30E+00	2.13E+05	9.11E+06
	50	3.14E-02	1.04E-02	1.70E+03	7.29E+04	1.57E-01	5.20E-02	8.51E+03	3.64E+05
	100	1.32E-03	4.37E-04	7.16E+01	3.07E+03	4.64E-02	1.54E-02	2.52E+03	1.08E+05
	250	8.23E-05	2.73E-05	4.47E+00	1.91E+02	1.17E-02	3.87E-03	6.35E+02	2.72E+04
	500	1.01E-05	3.34E-06	5.47E-01	2.34E+01	4.12E-03	1.37E-03	2.24E+02	9.57E+03
GG-010	10	2.38E-01	7.07E-02	1.24E+03	2.83E+05	1.19E+00	3.53E-01	6.22E+03	1.41E+06
	50	9.54E-03	2.83E-03	4.98E+01	1.13E+04	4.77E-02	1.41E-02	2.49E+02	5.66E+04
	100	4.01E-04	1.19E-04	2.09E+00	4.76E+02	1.41E-02	4.18E-03	7.37E+01	1.67E+04
	250	2.50E-05	7.41E-06	1.31E-01	2.97E+01	3.56E-03	1.05E-03	1.86E+01	4.22E+03
	500	3.07E-06	9.09E-07	1.60E-02	3.64E+00	1.25E-03	3.71E-04	6.54E+00	1.49E+03
GG-011	10	4.30E-01	2.84E-02	1.25E+04	6.83E+05	2.15E+00	1.42E-01	6.24E+04	3.42E+06
	50	1.72E-02	1.14E-03	4.99E+02	2.73E+04	8.61E-02	5.68E-03	2.50E+03	1.37E+05
	100	7.24E-04	4.77E-05	2.10E+01	1.15E+03	2.55E-02	1.68E-03	7.39E+02	4.05E+04
	250	4.52E-05	2.98E-06	1.31E+00	7.17E+01	6.42E-03	4.23E-04	1.86E+02	1.02E+04
	500	5.54E-06	3.65E-07	1.61E-01	8.79E+00	2.26E-03	1.49E-04	6.56E+01	3.59E+03
GG-012	10	4.93E-01	4.48E-01	1.78E+04	2.09E+06	2.46E+00	2.24E+00	8.92E+04	1.04E+07
	50	1.97E-02	1.79E-02	7.13E+02	8.35E+04	9.85E-02	8.95E-02	3.57E+03	4.18E+05
	100	8.28E-04	7.53E-04	3.00E+01	3.51E+03	2.92E-02	2.65E-02	1.06E+03	1.24E+05
	250	5.17E-05	4.70E-05	1.87E+00	2.19E+02	7.34E-03	6.67E-03	2.66E+02	3.11E+04
	500	6.33E-06	5.76E-06	2.29E-01	2.69E+01	2.59E-03	2.35E-03	9.37E+01	1.10E+04

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TABLE D.3-1. (continued)

SOURCE TERM PARTITION	DISTANCE (M)	A STABILITY, 5 M/S REL1 DOSE RATE (SV/S)	A STABILITY, 5 M/S REL2 DOSE RATE (SV/S)	A STABILITY, 5 M/S 15-MIN DOSE (SV)	A STABILITY, 5 M/S TOTAL DOSE (SV)	F STABILITY, 1 M/S REL1 DOSE RATE (SV/S)	F STABILITY, 1 M/S REL2 DOSE RATE (SV/S)	F STABILITY, 1 M/S 15-MIN DOSE (SV)	F STABILITY, 1 M/S TOTAL DOSE (SV)
GG-013	10	3.32E-01	1.57E-01	2.06E+04	2.53E+06	1.66E+00	7.84E-01	1.03E+05	1.26E+07
	50	1.33E-02	6.27E-03	8.26E+02	1.01E+05	6.64E-02	3.14E-02	4.13E+03	5.06E+05
	100	5.59E-04	2.64E-04	3.47E+01	4.25E+03	1.97E-02	9.29E-03	1.22E+03	1.50E+05
	250	3.49E-05	1.65E-05	2.17E+00	2.65E+02	4.95E-03	2.34E-03	3.08E+02	3.77E+04
	500	4.27E-06	2.02E-06	2.66E-01	3.25E+01	1.75E-03	8.24E-04	1.08E+02	1.33E+04
GG-014	10	5.74E-01	2.70E-02	7.08E+04	5.73E+06	2.87E+00	1.35E-01	3.54E+05	2.87E+07
	50	2.29E-02	1.08E-03	2.83E+03	2.29E+05	1.15E-01	5.39E-03	1.42E+04	1.15E+06
	100	9.65E-04	4.54E-05	1.19E+02	9.64E+03	3.40E-02	1.60E-03	4.20E+03	3.39E+05
	250	6.02E-05	2.83E-06	7.43E+00	6.01E+02	8.55E-03	4.02E-04	1.06E+03	8.55E+04
	500	7.38E-06	3.47E-07	9.11E-01	7.37E+01	3.01E-03	1.42E-04	3.72E+02	3.01E+04
GG-015	10	4.25E-01	7.09E-02	3.74E+03	3.73E+05	2.13E+00	3.54E-01	1.87E+04	1.86E+06
	50	1.70E-02	2.83E-03	1.50E+02	1.49E+04	8.50E-02	1.42E-02	7.49E+02	7.45E+04
	100	7.15E-04	1.19E-04	6.30E+00	6.27E+02	2.52E-02	4.20E-03	2.22E+02	2.21E+04
	250	4.46E-05	7.43E-06	3.93E-01	3.91E+01	6.34E-03	1.06E-03	5.58E+01	5.56E+03
	500	5.47E-06	9.11E-07	4.82E-02	4.79E+00	2.23E-03	3.72E-04	1.97E+01	1.96E+03
GG-016	10	5.34E-01	2.99E-01	1.13E+04	8.95E+05	2.67E+00	1.50E+00	5.65E+04	4.47E+06
	50	2.14E-02	1.20E-02	4.52E+02	3.58E+04	1.07E-01	5.99E-02	2.26E+03	1.79E+05
	100	8.99E-04	5.04E-04	1.90E+01	1.50E+03	3.16E-02	1.77E-02	6.69E+02	5.30E+04
	250	5.61E-05	3.14E-05	1.19E+00	9.39E+01	7.97E-03	4.46E-03	1.69E+02	1.33E+04
	500	6.87E-06	3.85E-06	1.45E-01	1.15E+01	2.81E-03	1.57E-03	5.94E+01	4.70E+03
GG-017	10	4.93E-01	6.19E-01	1.84E+04	5.78E+06	2.46E+00	3.09E+00	9.22E+04	2.89E+07
	50	1.97E-02	2.47E-02	7.38E+02	2.31E+05	9.86E-02	1.24E-01	3.69E+03	1.16E+06
	100	8.29E-04	1.04E-03	3.10E+01	9.73E+03	2.92E-02	3.66E-02	1.09E+03	3.42E+05
	250	5.17E-05	6.49E-05	1.93E+00	6.07E+02	7.35E-03	9.23E-03	2.75E+02	8.62E+04
	500	6.34E-06	7.96E-06	2.37E-01	7.44E+01	2.59E-03	3.25E-03	9.69E+01	3.04E+04
GG-018	10	1.53E-01	6.97E-02	1.19E+03	1.56E+05	7.65E-01	3.49E-01	5.97E+03	7.80E+05
	50	6.12E-03	2.79E-03	4.78E+01	6.24E+03	3.06E-02	1.39E-02	2.39E+02	3.12E+04
	100	2.57E-04	1.17E-04	2.01E+00	2.62E+02	9.06E-03	4.13E-03	7.07E+01	9.23E+03
	250	1.61E-05	7.31E-06	1.25E-01	1.64E+01	2.28E-03	1.04E-03	1.78E+01	2.32E+03
	500	1.97E-06	8.97E-07	1.54E-02	2.01E+00	8.04E-04	3.66E-04	6.27E+00	8.19E+02
GG-019	10	2.10E-01	8.18E-02	1.13E+03	1.56E+05	1.05E+00	4.09E-01	5.67E+03	7.79E+05
	50	8.40E-03	3.27E-03	4.54E+01	6.23E+03	4.20E-02	1.64E-02	2.27E+02	3.12E+04
	100	3.53E-04	1.38E-04	1.91E+00	2.62E+02	1.24E-02	4.84E-03	6.72E+01	9.23E+03
	250	2.20E-05	8.58E-06	1.19E-01	1.64E+01	3.13E-03	1.22E-03	1.69E+01	2.32E+03
	500	2.70E-06	1.05E-06	1.46E-02	2.00E+00	1.10E-03	4.30E-04	5.96E+00	8.19E+02
GG-020	10	3.44E-01	2.01E-01	1.84E+03	3.58E+05	1.72E+00	1.01E+00	9.20E+03	1.79E+06
	50	1.38E-02	8.04E-03	7.36E+01	1.43E+04	6.89E-02	4.02E-02	3.68E+02	7.16E-04
	100	5.79E-04	3.38E-04	3.10E+00	6.02E+02	2.04E-02	1.19E-02	1.09E+02	2.12E+04
	250	3.61E-05	2.11E-05	1.93E-01	3.75E+01	5.14E-03	3.00E-03	2.74E+01	5.33E+03
	500	4.43E-06	2.59E-06	2.37E-02	4.60E+00	1.81E-03	1.06E-03	9.67E+00	1.88E+03
GG-021	10	1.80E-01	1.33E-02	6.51E+02	4.42E+04	8.99E-01	6.65E-02	3.26E+03	2.21E+05
	50	7.19E-03	5.32E-04	2.61E+01	1.77E+03	3.60E-02	2.66E-03	1.30E+02	8.84E+03
	100	3.03E-04	2.24E-05	1.10E+00	7.44E+01	1.07E-02	7.88E-04	3.86E+01	2.62E+03
	250	1.89E-05	1.40E-06	6.83E-02	4.64E+00	2.68E-03	1.98E-04	9.71E+00	6.59E+02
	500	2.31E-06	1.71E-07	8.38E-03	5.69E-01	9.45E-04	6.99E-05	3.42E+00	2.32E+02
GG-022	10	1.04E-01	2.41E-02	4.82E+02	6.92E+04	5.22E-01	1.21E-01	2.41E+03	3.46E+05
	50	4.18E-03	9.65E-04	1.93E+01	2.77E+03	2.09E-02	4.83E-03	9.63E+01	1.38E+04
	100	1.76E-04	4.06E-05	8.10E-01	1.16E+02	6.18E-03	1.43E-03	2.85E+01	4.09E+03
	250	1.10E-05	2.53E-06	5.05E-02	7.26E+00	1.56E-03	3.60E-04	7.18E+00	1.03E+03
	500	1.34E-06	3.10E-07	6.19E-03	8.89E-01	5.49E-04	1.27E-04	2.53E+00	3.63E+02
GG-023	10	3.81E-01	8.89E-04	4.08E+02	1.71E+04	1.90E+00	4.45E-03	2.04E+03	8.55E+04
	50	1.52E-02	3.56E-05	1.63E+01	6.84E+02	7.62E-02	1.78E-06	8.17E+01	3.42E+03
	100	6.41E-04	1.50E-06	6.87E-01	2.88E+01	2.26E-02	5.27E-05	2.42E+01	1.01E+03
	250	4.00E-05	9.33E-08	4.29E-02	1.79E+00	5.68E-03	1.33E-05	6.09E+00	2.55E+02
	500	4.90E-06	1.14E-08	5.25E-03	2.20E-01	2.00E-03	4.67E-06	2.15E+00	8.98E+01
GG-024	10	1.17E-01	2.41E-01	1.66E+03	5.14E+05	5.84E-01	1.21E+00	8.31E+03	2.57E+06
	50	4.67E-03	9.66E-03	6.65E+01	2.05E+04	2.33E-02	4.83E-02	3.32E+02	1.03E+05
	100	1.96E-04	4.06E-04	2.80E+00	8.64E+02	6.91E-03	1.43E-02	9.84E+01	3.04E+04
	250	1.22E-05	2.53E-05	1.74E-01	5.39E+01	1.74E-03	3.60E-03	2.48E+01	7.66E+03
	500	1.50E-06	3.11E-06	2.14E-02	6.61E+00	6.13E-04	1.27E-03	8.73E+00	2.70E+03

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TABLE D.3-1. (continued)

SOURCE TERM PARTITION	DISTANCE (M)	A STABILITY, 5 M/S REL1 DOSE RATE (SV/S)	A STABILITY, 5 M/S REL2 DOSE RATE (SV/S)	A STABILITY, 5 M/S 15-MIN DOSE (SV)	A STABILITY, 5 M/S TOTAL DOSE (SV)	F STABILITY, 1 M/S REL1 DOSE RATE (SV/S)	F STABILITY, 1 M/S REL2 DOSE RATE (SV/S)	F STABILITY, 1 M/S 15-MIN DOSE (SV)	F STABILITY, 1 M/S TOTAL DOSE (SV)
GG-025	10	2.17E-01	1.59E-01	2.34E+03	3.72E+05	1.08E+00	7.94E-01	1.17E+04	1.86E+06
	50	8.66E-03	6.35E-03	9.36E+01	1.49E+04	4.33E-02	3.18E-02	4.68E+02	7.44E+04
	100	3.64E-04	2.67E-04	3.93E+00	6.26E+02	1.28E-02	9.41E-03	1.38E+02	2.20E+04
	250	2.27E-05	1.67E-05	2.45E-01	3.90E+01	3.23E-03	2.37E-03	3.49E+01	5.55E+03
	500	2.79E-06	2.04E-06	3.01E-02	4.79E+00	1.14E-03	8.34E-04	1.23E+01	1.95E+03
GG-026	10	1.90E-01	6.79E-03	3.66E+02	2.38E+04	9.50E-01	3.40E-02	1.83E+03	1.19E+05
	50	7.60E-03	2.72E-04	1.46E+01	9.54E+02	3.80E-02	1.36E-03	7.31E+01	4.77E+03
	100	3.20E-04	1.14E-05	6.15E-01	4.01E+01	1.13E-02	4.02E-04	2.16E+01	1.41E+03
	250	1.99E-05	7.13E-07	3.83E-02	2.50E+00	2.83E-03	1.01E-04	5.45E+00	3.55E+02
	500	2.44E-06	8.74E-08	4.70E-03	3.07E-01	9.98E-04	3.57E-05	1.92E+00	1.25E+02
GG-027	10	3.19E-01	5.27E-02	9.97E+02	7.94E+04	1.60E+00	2.63E-01	4.98E+03	3.97E+05
	50	1.28E-02	2.11E-03	3.99E+01	3.18E+03	6.38E-02	1.05E-02	1.99E+02	1.59E+04
	100	5.37E-04	8.86E-05	1.68E+00	1.34E+02	1.89E-02	3.12E-03	5.90E+01	4.70E+03
	250	3.35E-05	5.53E-06	1.05E-01	8.33E+00	4.76E-03	7.85E-04	1.49E+01	1.18E+03
	500	4.10E-06	6.77E-07	1.28E-02	1.02E+00	1.68E-03	2.77E-04	5.24E+00	4.17E+02
GG-028	10	7.29E-01	5.21E-02	7.03E+04	2.68E+06	3.64E+00	2.61E-01	3.51E+05	1.34E+07
	50	2.91E-02	2.09E-03	2.81E+03	1.07E+05	1.46E-01	1.04E-02	1.41E+04	5.36E+05
	100	1.23E-03	8.77E-05	1.18E+02	4.51E+03	4.31E-02	3.09E-03	4.16E+03	1.59E+05
	250	7.64E-05	5.47E-06	7.37E+00	2.81E+02	1.09E-02	7.78E-04	1.05E+03	4.00E+04
	500	9.37E-06	6.71E-07	9.04E-01	3.45E+01	3.83E-03	2.74E-04	3.69E+02	1.41E+04
GG-029	10	2.06E-01	9.88E-02	5.41E+03	7.42E+05	1.03E+00	4.94E-01	2.71E+04	3.71E+06
	50	8.25E-03	3.95E-03	2.16E+02	2.97E+04	4.12E-02	1.98E-02	1.08E+03	1.48E+05
	100	3.47E-04	1.66E-04	9.10E+00	1.25E+03	1.22E-02	5.85E-03	3.20E+02	4.39E+04
	250	2.16E-05	1.04E-05	5.68E-01	7.78E+01	3.07E-03	1.47E-03	8.07E+01	1.11E+04
	500	2.65E-06	1.27E-06	6.96E-02	9.54E+00	1.08E-03	5.19E-04	2.84E+01	3.90E+03
GG-030	10	1.08E+00	1.07E-01	4.36E+04	1.55E+06	5.41E+00	5.35E-01	2.18E+05	7.76E+06
	50	4.33E-02	4.28E-03	1.74E+03	6.21E+04	2.16E-01	2.14E-02	8.71E+03	3.10E+05
	100	1.82E-03	1.80E-04	7.33E+01	2.61E+03	6.41E-02	6.33E-03	2.58E+03	9.19E+04
	250	1.14E-04	1.12E-05	4.57E+00	1.63E+02	1.61E-02	1.59E-03	6.50E+02	2.31E+04
	500	1.39E-05	1.38E-06	5.60E-01	2.00E+01	5.68E-03	5.62E-04	2.29E+02	8.15E+03
GG-031	10	5.68E-02	1.26E-01	1.78E+03	5.45E+05	2.84E-01	6.30E-01	8.88E+03	2.73E+06
	50	2.27E-03	5.04E-03	7.10E+01	2.18E+04	1.14E-02	2.52E-02	3.55E+02	1.09E+05
	100	9.56E-05	2.12E-04	2.99E+00	9.17E+02	3.36E-03	7.46E-03	1.05E+02	3.23E+04
	250	5.96E-06	1.32E-05	1.86E-01	5.72E+01	8.47E-04	1.88E-03	2.65E+01	8.13E+03
	500	7.31E-07	1.62E-06	2.28E-02	7.01E+00	2.98E-04	6.62E-04	9.32E+00	2.86E+03
GG-032	10	1.15E+00	2.13E-01	1.19E+05	5.99E+06	5.77E+00	1.07E+00	5.97E+05	2.99E+07
	50	4.61E-02	8.53E-03	4.78E+03	2.40E+05	2.31E-01	4.27E-02	2.39E+04	1.20E+06
	100	1.94E-03	3.59E-04	2.01E+02	1.01E+04	6.83E-02	1.26E-02	7.07E+03	3.55E+05
	250	1.21E-04	2.24E-05	1.25E+01	6.28E+02	1.72E-02	3.18E-03	1.78E+03	8.93E+04
	500	1.48E-05	2.74E-06	1.54E+00	7.70E+01	6.06E-03	1.12E-03	6.27E+02	3.15E+04
GG-033	10	1.59E-01	8.35E-02	5.18E+03	6.98E+05	7.96E-01	4.18E-01	2.59E+04	3.49E+06
	50	6.37E-03	3.34E-03	2.07E+02	2.79E+04	3.19E-02	1.67E-02	1.04E+03	1.40E+05
	100	2.68E-04	1.41E-04	8.72E+00	1.17E+03	9.43E-03	4.95E-03	3.07E+02	4.13E+04
	250	1.67E-05	8.76E-06	5.44E-01	7.32E+01	2.37E-03	1.25E-03	7.73E+01	1.04E+04
	500	2.05E-06	1.07E-06	6.67E-02	8.98E+00	8.37E-04	4.39E-04	2.72E+01	3.67E+03
GG-034	10	2.14E-01	3.30E-02	3.68E+03	3.29E+05	1.07E+00	1.65E-01	1.84E+04	1.64E+06
	50	8.56E-03	1.32E-03	1.47E+02	1.32E+04	4.28E-02	6.59E-03	7.36E+02	6.58E+04
	100	3.60E-04	5.54E-05	6.19E+00	5.53E+02	1.27E-02	1.95E-03	2.18E+02	1.95E+04
	250	2.25E-05	3.46E-06	3.86E-01	3.45E+01	3.19E-03	4.91E-04	5.49E+01	4.90E+03
	500	2.75E-06	4.24E-07	4.73E-02	4.23E+00	1.12E-03	1.73E-04	1.93E+01	1.73E+03
GG-035	10	2.31E-01	7.35E-02	4.03E+04	3.81E+06	1.15E+00	3.68E-01	2.01E+05	1.90E+07
	50	9.24E-03	2.94E-03	1.61E+03	1.52E+05	4.62E-02	1.47E-02	8.05E+03	7.61E+05
	100	3.88E-04	1.24E-04	6.77E+01	6.40E+03	1.37E-02	4.35E-03	2.38E+03	2.25E+05
	250	2.42E-05	7.71E-06	4.22E+00	3.99E+02	3.44E-03	1.10E-03	6.00E+02	5.67E+04
	500	2.97E-06	9.45E-07	5.18E-01	4.89E+01	1.21E-03	3.86E-04	2.11E+02	2.00E+04
GG-036	10	2.01E-01	1.06E-01	2.53E+03	3.29E+05	1.00E+00	5.31E-01	1.27E+04	1.65E+06
	50	8.03E-03	4.25E-03	1.01E+02	1.32E+04	4.01E-02	2.13E-02	5.07E+02	6.59E+04
	100	3.37E-04	1.79E-04	4.26E+00	5.54E+02	1.19E-02	6.29E-03	1.50E+02	1.95E+04
	250	2.10E-05	1.12E-05	2.66E-01	3.46E+01	2.99E-03	1.58E-03	3.78E+01	4.91E+03
	500	2.58E-06	1.37E-06	3.26E-02	4.24E+00	1.05E-03	5.58E-04	1.33E+01	1.73E+03

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TABLE D.3-1. (continued)

SOURCE TERM PARTITION	DISTANCE (M)	A STABILITY, 5 M/S REL1 DOSE RATE (SV/S)	REL2 DOSE RATE (SV/S)	15-MIN DOSE (SV)	TOTAL DOSE (SV)	F STABILITY, 1 M/S REL1 DOSE RATE (SV/S)	REL2 DOSE RATE (SV/S)	15-MIN DOSE (SV)	TOTAL DOSE (SV)
GG-037	10	1.25E-01	3.89E-02	1.58E+03	1.50E+05	6.27E-01	1.95E-01	7.92E+03	7.49E+05
	50	5.01E-03	1.56E-03	6.34E+01	5.99E+03	2.51E-02	7.78E-03	3.17E+02	3.00E+04
	100	2.11E-04	6.54E-05	2.66E+00	2.52E+02	7.42E-03	2.30E-03	9.38E+01	8.87E+03
	250	1.31E-05	4.08E-06	1.66E-01	1.57E+01	1.87E-03	5.80E-04	2.36E+01	2.23E+03
	500	1.61E-06	5.00E-07	2.04E-02	1.93E+00	6.58E-04	2.04E-04	8.32E+00	7.87E+02
GG-038	10	9.73E-02	1.62E-01	5.02E+03	9.54E+05	4.86E-01	8.08E-01	2.51E+04	4.77E+06
	50	3.89E-03	6.47E-03	2.01E+02	3.82E+04	1.95E-02	3.23E-02	1.00E+03	1.91E+05
	100	1.64E-04	2.72E-04	8.44E+00	1.60E+03	5.76E-03	9.57E-03	2.97E+02	5.65E+04
	250	1.02E-05	1.70E-05	5.27E-01	1.00E+02	1.45E-03	2.41E-03	7.48E+01	1.42E+04
	500	1.25E-06	2.08E-06	6.46E-02	1.23E+01	5.11E-04	8.49E-04	2.64E+01	5.01E+03
GG-039	10	1.03E-01	7.26E-02	9.06E+03	1.06E+06	5.14E-01	3.63E-01	4.53E+04	5.29E+06
	50	4.11E-03	2.90E-03	3.63E+02	4.23E+04	2.06E-02	1.45E-02	1.81E+03	2.12E+05
	100	1.73E-04	1.22E-04	1.52E+01	1.78E+03	6.08E-03	4.30E-03	5.37E+02	6.27E+04
	250	1.08E-05	7.61E-06	9.51E-01	1.11E+02	1.53E-03	1.08E-03	1.35E+02	1.58E+04
	500	1.32E-06	9.33E-07	1.17E-01	1.36E+01	5.40E-04	3.81E-04	4.76E+01	5.56E+03
GG-040	10	1.14E-01	8.33E-02	2.58E+03	3.94E+05	5.70E-01	4.17E-01	1.29E+04	1.97E+06
	50	4.56E-03	3.33E-03	1.03E+02	1.58E+04	2.28E-02	1.67E-02	5.16E+02	7.88E+04
	100	1.92E-04	1.40E-04	4.34E+00	6.63E+02	6.75E-03	4.93E-03	1.53E+02	2.33E+04
	250	1.20E-05	8.74E-06	2.70E-01	4.14E+01	1.70E-03	1.24E-03	3.84E+01	5.88E+03
	500	1.47E-06	1.07E-06	3.32E-02	5.07E+00	5.99E-04	4.38E-04	1.35E+01	2.07E+03
GG-041	10	6.02E-02	6.24E-02	1.31E+03	2.17E+05	3.01E-01	3.12E-01	6.56E+03	1.09E+06
	50	2.41E-03	2.50E-03	5.25E+01	8.69E+03	1.20E-02	1.25E-02	2.62E+02	4.34E+04
	100	1.01E-04	1.05E-04	2.21E+00	3.65E+02	3.56E-03	3.70E-03	7.77E+01	1.29E+04
	250	6.32E-06	6.55E-06	1.38E-01	2.28E+01	8.98E-04	9.31E-04	1.96E+01	3.24E+03
	500	7.74E-07	8.03E-07	1.69E-02	2.79E+00	3.16E-04	3.28E-04	6.89E+00	1.14E+03
GG-042	10	3.39E-02	2.13E-02	3.78E+02	6.49E+04	1.69E-01	1.06E-01	1.89E+03	3.24E+05
	50	1.36E-03	8.52E-04	1.51E+01	2.60E+03	6.78E-03	4.26E-03	7.56E+01	1.30E+04
	100	5.70E-05	3.58E-05	6.35E-01	1.09E+02	2.01E-03	1.26E-03	2.24E+01	3.84E+03
	250	3.56E-06	2.23E-06	3.96E-02	6.81E+00	5.05E-04	3.18E-04	5.63E+00	9.68E+02
	500	4.36E-07	2.74E-07	4.86E-03	8.35E-01	1.78E-04	1.12E-04	1.98E+00	3.41E+02
GG-043	10	3.55E-02	1.81E-01	1.78E+03	5.41E+05	1.78E-01	9.05E-01	8.88E+03	2.70E+06
	50	1.42E-03	7.24E-03	7.10E-01	2.16E+04	7.11E-03	3.62E-02	3.55E+02	1.08E+05
	100	5.98E-05	3.04E-04	2.99E+00	9.10E+02	2.10E-03	1.07E-02	1.05E+02	3.20E+04
	250	3.73E-06	1.90E-05	1.86E-01	5.67E+01	5.30E-04	2.70E-03	2.65E+01	8.06E+03
	500	4.57E-07	2.33E-06	2.28E-02	6.96E+00	1.87E-04	9.50E-04	9.33E+00	2.84E+03
GG-044	10	5.46E-02	1.74E-02	6.52E-02	7.25E+04	2.73E-01	8.71E-02	3.26E+03	3.63E+05
	50	2.18E-03	6.96E-04	2.61E+01	7.90E+03	1.09E-02	3.48E-03	1.30E+02	1.45E+04
	100	9.19E-05	2.93E-05	1.10E+00	1.22E+02	3.23E-03	1.03E-03	3.86E+01	4.29E+03
	250	5.73E-06	1.83E-06	6.84E-02	7.61E+00	8.14E-04	2.60E-04	9.73E+00	1.08E+03
	500	7.02E-07	2.24E-07	8.39E-03	9.33E-01	2.87E-04	9.15E-05	3.43E+00	3.81E+02
GG-045	10	3.35E-02	8.36E-02	7.84E+02	2.09E+05	1.68E-01	4.18E-01	3.92E+03	1.05E+06
	50	1.34E-03	3.35E-03	3.14E+01	8.36E+03	6.70E-03	1.67E-02	1.57E+02	4.18E+04
	100	5.64E-05	1.41E-04	1.32E+00	3.52E+02	1.98E-03	4.95E-03	4.64E+01	1.24E+04
	250	3.52E-06	8.77E-06	8.23E-02	2.19E+01	5.00E-04	1.25E-03	1.17E+01	3.12E+03
	500	4.31E-07	1.08E-06	1.01E-02	2.69E+00	1.76E-04	4.39E-04	4.12E+00	1.10E+03
GG-046	10	2.79E-02	4.18E-03	1.09E+02	1.61E+04	1.40E-01	2.09E-02	5.46E+02	8.04E+04
	50	1.12E-03	1.67E-04	4.37E+00	6.43E+02	5.59E-03	8.36E-04	2.18E+01	3.21E+03
	100	4.70E-05	7.03E-06	1.84E-01	2.70E+01	1.65E-03	2.47E-04	6.47E+00	9.52E+02
	250	2.93E-06	4.38E-07	1.15E-02	1.69E+00	4.17E-04	6.23E-05	1.63E+00	2.40E+02
	500	3.59E-07	5.37E-08	1.40E-03	2.07E-01	1.47E-04	2.19E-05	5.74E-01	8.44E+01
GG-047	10	2.43E-02	1.10E-01	1.18E+03	2.90E+05	1.22E-01	5.49E-01	5.92E+03	1.45E+06
	50	9.74E-04	4.39E-03	4.74E+01	1.16E+04	4.87E-03	2.20E-02	2.37E+02	5.79E+04
	100	4.09E-05	1.85E-04	1.99E+00	4.87E+02	1.44E-03	6.50E-03	7.01E+01	1.72E+04
	250	2.55E-06	1.15E-05	1.24E-01	3.04E+01	3.63E-04	1.64E-03	1.77E+01	4.32E+03
	500	3.13E-07	1.41E-06	1.52E-02	3.73E+00	1.28E-04	5.77E-04	6.22E+00	1.52E+03
GG-048	10	3.68E-02	7.22E-02	8.47E+02	2.52E+05	1.84E-01	3.61E-01	4.24E+03	1.26E+06 +21 6.4
	50	1.47E-03	2.89E-03	3.39E+01	1.01E+04	7.37E-03	1.44E-02	1.69E+02	5.04E+04
	100	6.19E-05	1.21E-04	1.42E+00	4.23E+02	2.18E-03	4.27E-03	5.02E+01	1.49E+04
	250	3.86E-06	7.57E-06	8.89E-02	2.64E+01	5.49E-04	1.08E-03	1.26E-01	3.75E+03
	500	4.74E-07	9.28E-07	1.09E-02	3.24E+00	1.93E-04	3.79E-04	4.45E+00	1.32E+03

TABLE D.3-1. (continued)

SOURCE TERM PARTITION	DISTANCE (M)	A STABILITY, 5 M/S REL1 DOSE RATE (SV/S)	REL2 DOSE RATE (SV/S)	15-MIN DOSE (SV)	TOTAL DOSE (SV)	F STABILITY, 1 M/S REL1 DOSE RATE (SV/S)	REL2 DOSE RATE (SV/S)	15-MIN DOSE (SV)	TOTAL DOSE (SV)
GG-049	10	1.22E-02	2.[illegible]-03	4.08E+04	3.58E+06	6.08E-02	1.01E-02	2.04E+05	1.79E+07
	50	4.86E-04	8.0[illegible]E-05	1.63E+03	1.43E+05	2.43E-03	4.04E-04	8.15E+03	7.16E+05
	100	2.04E-05	3.40E-06	6.86E+01	6.02E+03	7.20E-04	1.20E-04	2.41E+03	2.12E+05
	250	1.28E-06	2.12E-07	4.28E+00	3.76E+02	1.81E-04	3.01E-05	6.08E+02	5.34E+04
	500	1.56E-07	2.60E-08	5.24E-01	4.61E+01	6.38E-05	1.06E-05	2.14E+02	1.88E+04
GG-050	10	6.05E-03	1.45E-03	1.44E+04	1.41E+06	3.03E-02	7.27E-03	7.20E+04	7.05E+06
	50	2.42E-04	5.81E-05	5.76E+02	5.64E+04	1.21E-03	2.91E-04	2.88E+03	2.82E+05
	100	1.02E-05	2.45E-06	2.42E+01	2.37E+03	3.58E-04	8.61E-05	8.53E+02	8.35E+04
	250	6.35E-07	1.53E-07	1.51E+00	1.48E+02	9.02E-05	2.17E-05	2.15E+02	2.10E+04
	500	7.78E-08	1.87E-08	1.85E-01	1.81E+01	3.18E-05	7.64E-06	7.57E+01	7.40E+03
GG-051	10	5.16E-03	2.56E-03	5.71E+03	8.32E+05	2.58E-02	1.28E-02	2.86E+04	4.16E+06
	50	2.06E-04	1.03E-04	2.29E+02	3.33E+04	1.03E-03	5.13E-04	1.14E+03	1.66E+05
	100	8.67E-06	4.31E-06	9.61E+00	1.40E+03	3.05E-04	1.52E-04	3.38E+02	4.93E+04
	250	5.41E-07	2.69E-07	5.99E-01	8.73E+01	7.69E-05	3.82E-05	8.52E+01	1.24E+04
	500	6.63E-08	3.30E-08	7.35E-02	1.07E+01	2.71E-05	1.35E-05	3.00E+01	4.37E+03
GG-0[illegible]2	10	1.21E-03	7.47E-04	8.41E+02	1.22E+05	6.07E-03	3.74E-03	4.20E+03	6.12E+05
	50	4.86E-05	2.99E-05	3.36E+01	4.90E+03	2.43E-04	1.49E-04	1.68E+02	2.45E+04
	100	2.04E-06	1.26E-06	1.41E+00	2.06E+02	7.19E-05	4.42E-05	4.98E+01	7.25E+03
	250	1.27E-07	7.84E-08	8.82E-02	1.28E+01	1.81E-05	1.11E-05	1.25E+01	1.82E+03
	500	1.56E-08	9.61E-09	1.08E-02	1.57E+00	6.38E-06	3.93E-06	4.42E+00	6.43E+02
GG-053	10	2.69E-04	1.87E-04	1.58E+02	2.08E+04	1.35E-03	9.36E-04	7.92E+02	1.04E+05
	50	1.08E-05	7.48E-06	6.34E+00	8.34E+02	5.38E-05	3.74E-05	3.17E+01	4.17E+03
	100	4.53E-07	3.15E-07	2.66E-01	3.51E+01	1.59E-05	1.11E-05	9.38E+00	1.23E+03
	250	2.82E-08	1.96E-08	1.66E-02	2.19E+00	4.01E-06	2.79E-06	2.36E+00	3.11E+02
	500	3.46E-09	2.41E-09	2.04E-03	2.68E-01	1.41E-06	9.83E-07	8.32E-01	1.09E+02
GG-054	10	6.87E-05	4.96E-05	2.74E+01	3.00E+03	3.44E-04	2.48E-04	1.37E+02	1.50E+04
	50	2.75E-06	1.98E-06	1.10E+00	1.20E+02	1.37E-05	9.91E-06	5.48E+00	6.00E+02
	100	1.16E-07	8.33E-08	4.61E-02	5.05E+00	4.07E-06	2.93E-06	1.62E+00	1.78E+02
	250	7.21E-09	5.20E-09	2.87E-03	3.15E-01	1.02E-06	7.39E-07	4.08E-01	4.47E+01
	500	8.64E-10	6.37E-10	3.52E-04	3.86E-02	3.61E-07	2.60E-07	1.44E-01	1.58E+01
GG-055	10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	50	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	100	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	250	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	500	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

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TABLE D.3-2. Dose and Dose Rates Calculated Using Ramsdell Model

SOURCE TERM PARTITION	DISTANCE (M)	A STABILITY, 5 M/S REL1 DOSE RATE (SV/S)	REL2 DOSE RATE (SV/S)	15-MIN DOSE (SV)	TOTAL DOSE (SV)	F STABILITY, 1 M/S REL1 DOSE RATE (SV/S)	REL2 DOSE RATE (SV/S)	15-MIN DOSE (SV)	TOTAL DOSE (SV)
GG-001	10	4.39E-03	2.15E-03	6.91E+01	9.12E+03	3.21E-03	1.58E-03	5.06E+01	6.68E+03
	50	7.12E-04	3.49E-04	1.12E+01	1.48E+03	5.21E-04	2.56E-04	8.21E+00	1.08E+03
	100	3.25E-04	1.60E-04	5.12E+00	6.76E+02	2.38E-04	1.17E-04	3.75E+00	4.95E+02
	250	1.15E-04	5.67E-05	1.82E+00	2.40E+02	8.46E-05	4.15E-05	1.33E+00	1.76E+02
	500	5.28E-05	2.59E-05	8.31E-01	1.10E+02	3.86E-05	1.90E-05	6.09E-01	8.03E+01
GG-002	10	5.03E-03	1.78E-03	1.34E+02	1.61E+04	3.68E-03	1.31E-03	9.84E+01	1.18E+04
	50	8.16E-04	2.89E-04	2.18E+01	2.61E+03	5.98E-04	2.12E-04	1.60E+01	1.91E+03
	100	3.73E-04	1.32E-04	9.95E+00	1.19E+03	2.73E-04	9.67E-05	7.29E+00	8.75E+02
	250	1.32E-04	4.69E-05	3.53E+00	4.24E+02	9.70E-05	3.44E-05	2.59E+00	3.11E+02
	500	6.05E-05	2.14E-05	1.62E+00	1.94E+02	4.43E-05	1.57E-05	1.18E+00	1.42E+02
GG-003	10	2.09E-03	2.65E-03	2.11E+01	7.83E+03	1.53E-03	1.94E-03	1.55E+01	5.74E+03
	50	3.39E-04	4.30E-04	3.42E+00	1.27E+03	2.48E-04	3.15E-04	2.51E+00	9.31E+02
	100	1.55E-04	1.97E-04	1.56E+00	5.81E+02	1.13E-04	1.44E-04	1.15E+00	4.25E+02
	250	5.50E-05	6.98E-05	5.55E-01	2.06E+02	4.03E-05	5.11E-05	4.07E-01	1.51E+02
	500	2.51E-05	3.19E-05	2.54E-01	9.42E+01	1.84E-05	2.34E-05	1.86E-01	6.90E+01
GG-004	10	1.33E-02	1.14E-03	7.14E+02	3.60E+04	9.73E-03	8.33E-04	5.23E+02	2.63E+04
	50	2.16E-03	1.85E-04	1.16E+02	5.83E+03	1.58E-03	1.35E-04	8.48E+01	4.27E+03
	100	9.85E-04	8.43E-05	5.29E+01	2.67E+03	7.21E-04	6.18E-05	3.88E+01	1.95E+03
	250	3.50E-04	2.99E-05	1.88E+01	9.46E+02	2.56E-04	2.19E-05	1.38E+01	6.93E+02
	500	1.60E-04	1.37E-05	8.58E+00	4.32E+02	1.17E-04	1.00E-05	6.29E+00	3.17E+02
GG-005	10	8.01E-03	2.90E-03	1.37E+02	1.34E+04	5.86E-03	2.13E-03	1.00E+02	9.80E+03
	50	1.30E-03	4.71E-04	2.21E+01	[illegible]	9.51E-04	3.45E-04	1.62E+01	1.59E+03
	100	5.93E-04	2.15E-04	1.01E+01	9.92E+02	4.35E-04	1.58E-04	7.41E+00	7.27E+02
	250	2.11E-04	7.64E-05	3.59E+00	[illegible]	1.54E-04	5.59E-05	2.63E+00	2.58E+02
	500	9.63E-05	3.49E-05	1.64E+00	[illegible]	7.05E-05	2.56E-05	1.20E+00	1.18E+02
GG-006	10	1.28E-03	1.85E-03	1.71E+01	3.77E+03	9.38E-04	1.36E-03	1.25E+01	2.76E+03
	50	2.08E-04	3.01E-04	2.77E+00	6.12E+02	1.52E-04	2.20E-04	2.03E+00	4.48E+02
	100	9.49E-05	1.37E-04	1.26E+00	2.80E+02	6.95E-05	1.01E-04	9.26E-01	2.05E+02
	250	3.37E-05	4.88E-05	4.49E-01	9.93E+01	2.47E-05	3.57E-05	3.29E-01	7.27E+01
	500	1.54E-05	2.23E-05	2.05E-01	4.54E+01	1.13E-05	1.63E-05	1.50E-01	3.32E+01
GG-007	10	1.55E-03	2.08E-03	2.62E+01	9.22E+03	1.14E-03	1.52E-03	1.92E+01	6.75E+03
	50	2.52E-04	3.37E-04	4.25E+00	1.50E+03	1.85E-04	2.47E-04	3.11E+00	1.10E+03
	100	1.15E-04	1.54E-04	1.94E+00	6.83E+02	8.44E-05	1.13E-04	1.42E+00	5.01E+02
	250	4.09E-05	5.46E-05	6.89E-01	2.43E+02	3.00E-05	4.00E-05	5.05E-01	1.78E+02
	500	1.87E-05	2.50E-05	3.15E-01	1.11E+02	1.37E-05	1.83E-05	2.31E-01	8.12E+01
GG-008	10	2.83E-03	1.44E-03	5.08E+01	6.67E+03	2.07E-03	1.05E-03	3.72E+01	4.89E+03
	50	4.59E-04	2.33E-04	8.25E+00	1.08E+03	3.36E-04	1.71E-04	6.04E+00	7.93E+02
	100	2.10E-04	1.07E-04	3.77E+00	4.95E+02	1.54E-04	7.80E-05	2.76E+00	3.62E+02
	250	7.45E-05	3.78E-05	1.34E+00	1.76E+02	5.45E-05	2.77E-05	9.80E-01	1.29E+02
	500	3.40E-05	1.73E-05	6.11E-01	8.03E+01	2.49E-05	1.27E-05	4.48E-01	5.88E+01
GG-009	10	8.08E-03	2.68E-03	4.39E+02	1.88E+04	5.92E-03	1.96E-03	3.21E+02	1.38E+04
	50	1.31E-03	4.35E-04	7.12E+01	3.05E+03	9.60E-04	3.18E-04	5.21E+01	2.23E+03
	100	5.99E-04	1.99E-04	3.25E+01	1.39E+03	4.39E-04	1.45E-04	2.38E+01	1.02E+03
	250	2.13E-04	7.05E-05	1.15E+01	4.94E+02	1.56E-04	5.16E-05	8.46E+00	3.62E+02
	500	9.72E-05	3.22E-05	5.28E+00	2.26E+02	7.12E-05	2.36E-05	3.87E+00	1.65E+02
GG-010	10	2.46E-03	7.28E-04	1.28E+01	2.92E+03	1.80E-03	5.34E-04	9.40E+00	2.14E+03
	50	3.99E-04	1.18E-04	2.08E+00	4.73E+02	2.92E-04	8.66E-05	1.52E+00	3.47E+02
	100	1.82E-04	5.40E-05	9.51E-01	2.16E+02	1.33E-04	3.96E-05	6.97E-01	1.58E+02
	250	6.47E-05	1.92E-05	3.38E-01	7.67E+01	4.74E-05	1.40E-05	2.47E-01	5.62E+01
	500	2.96E-05	8.76E-06	1.54E-01	3.51E+01	2.17E-05	6.42E-06	1.13E-01	2.57E+01
GG-011	10	4.44E-03	2.92E-04	1.29E+02	7.04E+03	3.25E-03	2.14E-04	9.43E+01	5.16E+03
	50	7.20E-04	4.75E-05	2.09E+01	1.14E+03	5.27E-04	3.48E-05	1.53E+01	8.37E+02
	100	3.29E-04	2.17E-05	9.54E+00	5.22E+02	2.41E-04	1.59E-05	6.99E+00	3.83E+02
	250	1.17E-04	7.70E-06	3.39E+00	1.85E+02	8.55E-05	5.64E-06	2.48E+00	1.36E+02
	500	5.34E-05	3.52E-06	1.55E+00	8.47E+01	3.91E-05	2.58E-06	1.13E+00	6.21E+01
GG-012	10	5.08E-03	4.61E-03	1.84E+02	2.15E+04	3.72E-03	3.38E-03	1.35E+02	1.58E+04
	50	8.24E-04	7.49E-04	2.98E+01	3.49E+03	6.03E-04	5.48E-04	2.18E+01	2.56E+03
	100	3.76E-04	3.42E-04	1.36E+01	1.60E+03	2.76E-04	2.51E-04	9.98E+00	1.17E+03
	250	1.34E-04	1.21E-04	4.84E+00	5.57E+02	9.79E-05	8.90E-05	3.54E+00	4.15E+02
	500	6.11E-05	5.55E-05	2.21E+00	[illegible]	4.47E-05	4.06E-05	1.62E+00	1.90E+02

TABLE D.3-2. (continued)

SOURCE TERM PARTITION	DISTANCE (M)	A STABILITY, 5 M/S REL1 DOSE RATE (SV/S)	A STABILITY, 5 M/S REL2 DOSE RATE (SV/S)	A STABILITY, 5 M/S 15-MIN DOSE (SV)	A STABILITY, 5 M/S TOTAL DOSE (SV)	F STABILITY, 1 M/S REL1 DOSE RATE (SV/S)	F STABILITY, 1 M/S REL2 DOSE RATE (SV/S)	F STABILITY, 1 M/S 15-MIN DOSE (SV)	F STABILITY, 1 M/S TOTAL DOSE (SV)
GG-013	10	3.42E-03	1.62E-03	2.13E+02	2.81E+04	2.51E-03	1.18E-03	1.56E+02	1.91E+04
	50	5.56E-04	2.62E-04	3.45E+01	4.23E+03	4.07E-04	1.92E-04	2.53E+01	3.10E+03
	100	2.54E-04	1.20E-04	1.58E+01	1.9[illegible]E+03	1.86E-04	8.78E-05	1.16E+01	1.42E+03
	250	9.01E-05	4.26E-05	5.60E+00	6.86E+02	6.60E-05	3.12E-05	4.10E+00	5.03E+02
	500	4.12E-05	1.94E-05	2.56E+00	3.13E+02	3.02E-05	1.42E-05	1.87E+00	2.30E+02
GG-014	10	5.91E-03	2.78E-04	7.30E+02	5.91E+04	4.33E-03	2.04E-04	5.35E+02	4.33E+04
	50	9.59E-04	4.51E-05	1.18E+02	9.59E+03	7.03E-04	3.30E-05	8.68E+01	7.02E+03
	100	4.38E-04	2.06E-05	5.41E+01	4.38E+03	3.21E-04	1.51E-05	3.97E+01	3.21E+03
	250	1.56E-04	7.32E-06	1.92E+01	1.56E+03	1.14E-04	5.36E-06	1.41E+01	1.14E+03
	500	7.11E-05	3.34E-06	8.78E+00	7.11E+02	5.21E-05	2.45E-06	6.43E+00	5.21E+02
GG-015	10	4.38E-03	7.30E-04	3.86E+01	3.84E+03	3.21E-03	5.35E-04	2.83E+01	2.81E+03
	50	7.11E-04	1.18E-04	6.26E+00	6.23E+02	5.21E-04	8.68E-05	4.59E+00	4.57E+02
	100	3.25E-04	5.41E-05	2.86E+00	2.85E+02	2.38E-04	3.97E-05	2.10E+00	2.09E+02
	250	1.15E-04	1.92E-05	1.02E+00	1.01E+02	8.45E-05	1.41E-05	7.44E-01	7.41E+01
	500	5.27E-05	8.78E-06	4.64E-01	4.62E+01	3.86E-05	6.43E-06	3.40E-01	3.38E+01
GG-016	10	5.51E-03	3.09E-03	1.17E+02	9.22E+03	4.03E-03	2.26E-03	8.54E+01	6.75E+03
	50	8.93E-04	5.01E-04	1.89E+01	1.50E+03	6.54E-04	3.67E-04	1.38E+01	1.10E+03
	100	4.08E-04	2.29E-04	8.64E+00	6.84E+02	2.99E-04	1.68E-04	6.33E+00	5.01E+02
	250	1.45E-04	[illegible].12E-05	3.07E+00	2.43E+02	1.06E-04	5.95E-05	2.25E+00	1.78E+02
	500	6.62E-05	3.71E-05	1.40E+00	1.11E+02	4.85E-05	2.72E-05	1.03E+00	8.12E+01
GG-017	10	5.08E-03	6.38E-03	1.90E+02	5.96E+04	3.72E-03	4.67E-03	1.39E+02	4.37E+04
	50	8.24E-04	1.03E-03	3.08E+01	9.67E+03	6.04E-04	7.58E-04	2.26E+01	7.08E+03
	100	3.77E-04	4.73E-04	1.41E+01	4.42E+03	2.76E-04	3.46E-04	1.03E+01	3.24E+03
	250	1.34E-04	1.68E-04	5.00E+00	1.57E+03	9.79E-05	1.23E-04	3.66E+00	1.15E+03
	500	6.11E-05	7.67E-05	2.29E+00	7.17E+02	4.47E-05	5.62E-05	1.67E+00	5.25E+02
GG-018	10	1.58E-03	7.19E-04	1.23E+01	1.61E+03	1.16E-03	5.26E-04	9.02E+00	1.18E+03
	50	2.56E-04	1.17E-04	2.00E+00	2.61E+02	1.87E-04	8.54E-05	1.46E+00	1.91E+02
	100	1.17E-04	5.33E-05	9.12E-01	1.19E+02	8.57E-05	3.90E-05	6.68E-01	8.73E+01
	250	4.15E-05	1.89E-05	3.24E-01	4.23E+01	3.04E-05	1.39E-05	2.37E-01	3.10E+01
	500	1.90E-05	8.64E-06	1.48E-01	1.93E+01	1.39E-05	6.33E-06	1.08E-01	1.42E+01
GG-019	10	2.16E-03	8.43E-04	1.17E+01	1.61E+03	1.59E-03	6.18E-04	8.57E+00	1.18E+03
	50	3.51E-04	1.37E-04	1.90E+00	2.61E+02	2.57E-04	1.00E-04	1.39E+00	1.91E+02
	100	1.60E-04	6.25E-05	8.67E-01	1.19E+02	1.18E-04	4.58E-05	6.35E-01	8.72E+01
	250	5.70E-05	2.22E-05	3.08E-01	4.23E+01	4.17E-05	1.63E-05	2.26E-01	3.10E+01
	500	2.60E-05	1.01E-05	1.41E-01	1.93E+01	1.91E-05	7.43E-06	1.03E-01	1.42E+01
GG-020	10	3.55E-03	2.07E-03	1.90E+01	3.69E+03	2.60E-03	1.52E-03	1.39E+01	2.70E+03
	50	5.76E-04	3.36E-04	3.08E+00	5.98E+02	4.22E-04	2.46E-04	2.25E+00	4.38E+02
	100	2.63E-04	1.54E-04	1.41E+00	2.73E+02	1.93E-04	1.13E-04	1.03E+00	2.00E+02
	250	9.35E-05	5.45E-05	4.99E-01	9.71E+01	6.85E-05	3.99E-05	3.66E-01	7.11E+01
	500	4.27E-05	2.49E-05	2.28E-01	4.44E+01	3.13E-05	1.83E-05	1.67E-01	3.25E+01
GG-021	10	1.85E-03	1.37E-04	6.71E+00	4.56E+02	1.36E-03	1.00E-04	4.92E+00	3.34E+02
	50	3.01E-04	2.[illegible]3E-05	1.09E+00	7.40E+01	2.20E-04	1.63E-05	7.98E-01	5.42E+01
	100	1.37E-04	1.02E-05	4.98E-01	3.38E+01	1.01E-04	7.45E-06	3.65E-01	2.48E+01
	250	4.88E-05	3.61E-06	1.77E-01	1.20E+01	3.57E-05	2.64E-06	1.29E-01	8.79E+00
	500	2.23E-05	1.65E-06	8.08E-02	5.48E+00	1.63E-05	1.21E-06	5.92E-02	4.02E+00
GG-022	10	1.08E-03	2.49E-04	4.96E+00	7.13E+02	7.88E-04	1.82E-04	3.64E+00	5.22E+02
	50	1.75E-04	4.04E-05	8.05E-01	1.16E+02	1.28E-04	2.96E-05	5.90E-01	8.47E+01
	100	7.98E-05	1.84E-05	3.68E-01	5.28E+01	5.84E-05	1.35E-05	2.69E-01	3.87E+01
	250	2.83E-05	6.55E-06	1.31E-01	1.88E+01	2.08E-05	4.80E-06	9.57E-02	1.37E+01
	500	1.29E-05	2.99E-06	5.97E-02	8.57E+00	9.48E-06	2.19E-06	4.37E-02	6.28E+00
GG-023	10	3.93E-03	9.17E-06	4.21E+00	1.76E+02	2.88E-03	6.71E-06	3.08E+00	1.29E+02
	50	6.37E-04	1.49E-06	6.83E-01	2.86E+01	4.67E-04	1.09E-06	5.00E-01	2.09E+01
	100	2.91E-04	6.79E-07	3.12E-01	1.31E+01	2.13E-04	4.98E-07	2.29E-01	9.57E+00
	250	1.03E-04	2.41E-07	1.11E-01	4.64E+00	7.57E-05	1.77E-07	8.12E-02	3.40E+00
	500	4.72E-05	1.10E-07	5.06E-02	2.12E+00	3.46E-05	8.08E-08	3.71E-02	1.55E+00
GG-024	10	1.20E-03	2.49E-03	1.71E+01	5.29E+03	8.81E-04	1.82E-03	1.25E+01	3.88E+03
	50	1.95E-04	4.04E-04	2.78E+00	8.59E+02	1.43E-04	2.96E-04	2.04E+00	6.29E+02
	100	8.92E-05	1.84E-04	1.27E+00	3.93E+02	6.53E-05	1.35E-04	9.30E-01	2.88E+02
	250	3.17E-05	6.55E-05	4.51E-01	1.39E+02	2.32E-05	4.80E-05	3.30E-01	1.02E+02
	500	1.45E-05	2.99E-05	2.06E-01	6.37E+01	1.06E-05	2.19E-05	1.51E-01	4.66E+01

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TABLE D.3-2. (continued)

SOURCE TERM PARTITION	DISTANCE (M)	A STABILITY, 5 M/S REL1 DOSE RATE (SV/S)	A STABILITY, 5 M/S REL2 DOSE RATE (SV/S)	A STABILITY, 5 M/S 15-MIN DOSE (SV)	A STABILITY, 5 M/S TOTAL DOSE (SV)	F STABILITY, 1 M/S REL1 DOSE RATE (SV/S)	F STABILITY, 1 M/S REL2 DOSE RATE (SV/S)	F STABILITY, 1 M/S 15-MIN DOSE (SV)	F STABILITY, 1 M/S TOTAL DOSE (SV)
GG-025	10	2.23E-03	1.64E-03	2.41E+01	3.84E+03	1.64E-03	1.20E-03	1.77E+01	2.81E+03
	50	3.62E-04	2.66E-04	3.91E+00	6.22E+02	2.65E-04	1.95E-04	2.86E+00	4.56E+02
	100	1.66E-04	1.21E-04	1.79E+00	2.84E+02	1.21E-04	8.89E-05	1.31E+00	2.08E+02
	250	5.88E-05	4.31E-05	6.35E-01	1.01E+02	4.30E-05	3.16E-05	4.65E-01	7.40E+01
	500	2.69E-05	1.97E-05	2.90E-01	4.61E+01	1.97E-05	1.44E-05	2.12E-01	3.38E+01
GG-026	10	1.96E-03	7.00E-05	3.77E+00	2.46E+02	1.44E-03	5.13E-05	2.76E+00	1.80E+02
	50	3.18E-04	1.14E-05	6.11E-01	3.99E+01	2.33E-04	8.32E-06	4.48E-01	2.92E+01
	100	1.45E-04	5.19E-06	2.79E-01	1.82E+01	1.06E-04	3.80E-06	2.05E-01	1.33E+01
	250	5.16E-05	1.84E-06	9.92E-02	6.47E+00	3.78E-05	1.35E-06	7.26E-02	4.74E+00
	500	2.36E-05	8.42E-07	7.53E-02	2.96E+00	1.73E-05	6.17E-07	3.32E-02	2.16E+00
GG-027	10	3.29E-03	5.43E-04	1.03E+01	8.18E+02	2.41E-03	3.98E-04	7.52E+00	5.99E+02
	50	5.34E-04	8.81E-05	1.67E+00	1.33E+02	3.91E-04	6.45E-05	1.22E+00	9.73E+01
	100	2.44E-04	4.02E-05	7.62E-01	6.07E+01	1.79E-04	2.95E-05	5.58E-01	4.44E+01
	250	8.66E-05	1.43E-05	2.70E-01	2.15E+01	6.34E-05	1.05E-05	1.98E-01	1.58E+01
	500	3.96E-05	6.53E-06	1.24E-01	9.84E+00	2.90E-05	4.78E-06	9.05E-02	7.21E+00
GG-028	10	7.51E-03	5.37E-04	7.24E+02	2.76E+04	5.50E-03	3.94E-04	5.31E+02	2.03E+04
	50	1.22E-03	8.72E-05	1.18E+02	4.49E+03	8.93E-04	6.39E-05	8.61E+01	3.29E+03
	100	5.57E-04	3.98E-05	5.37E+01	2.05E+03	4.08E-04	2.92E-05	3.93E+01	1.50E+03
	250	1.98E-04	1.41E-05	1.91E+01	7.28E+02	1.45E-04	1.04E-05	1.40E+01	5.33E+02
	500	9.03E-05	6.46E-06	8.71E+00	3.32E+02	6.62E-05	4.74E-06	6.38E+00	2.44E+02
GG-029	10	2.12E-03	1.02E-03	5.58E+01	7.65E+03	1.56E-03	7.46E-04	4.09E+01	5.60E+03
	50	3.45E-04	1.65E-04	[illegible]	1.24E+03	2.53E-04	1.21E-04	6.63E+00	9.09E+02
	100	1.58E-04	7.55E-05	4.13E+00	5.67E+02	1.15E-04	5.53E-05	3.03E+00	4.15E+02
	250	5.59E-05	2.68E-05	1.47E+00	2.01E+02	4.10E-05	1.96E-05	1.08E+00	1.47E+02
	500	2.56E-05	1.23E-05	6.71E-01	9.20E+01	1.87E-05	8.98E-06	4.91E-01	6.74E+01
GG-030	10	1.12E-02	1.10E-03	4.49E+02	1.60E+04	8.17E-03	8.07E-04	3.29E+02	1.17E+04
	50	1.81E-03	1.79E-04	7.29E+01	2.59E+03	1.33E-03	1.31E-04	5.34E+01	1.90E+03
	100	8.27E-04	8.17E-05	3.33E+01	1.19E+03	6.06E-04	5.99E-05	2.44E+01	8.68E+02
	250	2.94E-04	2.90E-05	1.18E+01	4.21E+02	2.15E-04	2.13E-05	8.66E+00	3.08E+02
	500	1.34E-04	1.33E-05	5.40E+00	1.92E+02	9.83E-05	9.71E-06	3.96E+00	1.41E+02
GG-031	10	5.86E-04	1.30E-03	1.83E+01	5.62E+03	4.29E-04	9.51E-04	1.34E+01	4.12E+03
	50	9.50E-05	2.11E-04	2.97E+00	9.12E+02	6.96E-05	1.54E-04	2.17E+00	6.68E+02
	100	4.34E-05	9.62E-05	1.36E+00	4.17E+02	3.18E-05	7.05E-05	9.94E-01	3.05E+02
	250	1.54E-05	3.42E-05	4.82E-01	1.48E+02	1.13E-05	2.50E-05	3.53E-01	1.08E+02
	500	7.04E-06	1.56E-05	2.20E-01	6.76E+01	5.16E-06	1.14E-05	1.61E-01	4.95E+01
GG-032	10	1.19E-02	2.20E-03	1.23E+03	6.17E+04	8.71E-03	1.61E-03	9.02E+02	4.52E+04
	50	1.93E-03	3.57E-04	2.00E+02	1.00E+04	1.41E-03	2.61E-04	1.46E+02	7.34E+03
	100	8.81E-04	1.63E-04	9.13E+01	4.58E+03	6.45E-04	1.19E-04	6.68E+01	3.35E+03
	250	3.13E-04	5.79E-05	3.24E+01	1.62E+03	2.29E-04	4.24E-05	2.37E+01	1.19E+03
	500	1.43E-04	2.64E-05	1.48E+01	7.42E+02	1.05E-04	1.94E-05	1.08E+01	5.44E+02
GG-033	10	1.64E-03	8.61E-04	5.34E+01	7.19E+03	1.20E-03	6.31E-04	3.91E+01	5.27E+03
	50	2.66E-04	1.40E-04	8.67E+00	1.17E+03	1.95E-04	1.02E-04	6.35E+00	8.55E+02
	100	1.22E-04	6.38E-05	3.96E+00	5.33E+02	8.91E-05	4.68E-05	2.90E+00	3.91E+02
	250	4.32E-05	2.27E-05	1.41E+00	1.89E+02	3.17E-05	1.66E-05	1.03E+00	1.39E+02
	500	1.97E-05	1.04E-05	6.43E-01	8.65E+01	1.45E-05	7.59E-06	4.71E-01	6.34E+01
GG-034	10	2.21E-03	3.40E-04	3.79E+01	3.39E+03	1.62E-03	2.49E-04	2.78E+01	2.48E+03
	50	3.58E-04	5.51E-05	6.15E+00	5.50E+02	2.62E-04	4.04E-05	4.51E+00	4.03E+02
	100	1.64E-04	2.52E-05	2.81E+00	2.51E+02	1.20E-04	1.84E-05	2.06E+00	1.84E+02
	250	5.81E-05	8.94E-06	9.98E-01	8.92E+01	4.25E-05	6.55E-06	7.31E-01	6.54E+01
	500	2.65E-05	4.09E-06	4.56E-01	4.08E+01	1.94E-05	2.99E-06	3.34E-01	2.99E+01
GG-035	10	2.38E-03	7.58E-04	4.15E+02	3.92E+04	1.74E-03	5.55E-04	3.04E+02	2.87E+04
	50	3.86E-04	1.23E-04	6.73E+01	6.36E+03	2.83E-04	9.00E-05	4.93E+01	4.66E+03
	100	1.76E-04	5.62E-05	3.08E+01	2.91E+03	1.29E-04	4.11E-05	2.25E+01	2.13E+03
	250	6.27E-05	1.99E-05	1.09E+01	1.03E+03	4.59E-05	1.46E-05	8.00E+00	7.56E+02
	500	2.86E-05	9.11E-06	4.99E+00	4.72E+02	2.10E-05	6.67E-06	3.66E+00	3.46E+02
GG-036	10	2.07E-03	1.10E-03	2.61E+01	3.40E+03	1.51E-03	8.02E-04	1.91E+01	2.49E+03
	50	3.35E-04	1.78E-04	4.24E+00	5.51E+02	2.46E-04	1.30E-04	3.10E+00	4.04E+02
	100	1.53E-04	8.12E-05	1.94E+00	2.52E+02	1.12E-04	5.95E-05	1.42E+00	1.84E+02
	250	5.44E-05	2.88E-05	6.87E-01	8.94E+01	3.99E-05	2.11E-05	5.03E-01	6.55E+01
	500	2.49E-05	1.32E-05	3.14E-01	4.08E+01	1.82E-05	9.65E-06	[illegible]	2.99E+01

TABLE D.3-2. (continued)

SOURCE TERM PARTITION	DISTANCE (M)	A STABILITY, 5 M/S REL1 DOSE RATE (SV/S)	REL2 DOSE RATE (SV/S)	15-MIN DOSE (SV)	TOTAL DOSE (SV)	F STABILITY, 1 M/S REL1 DOSE RATE (SV/S)	REL2 DOSE RATE (SV/S)	15-MIN DOSE (SV)	TOTAL DOSE (SV)
GG-037	10	1.29E-03	4.01E-04	1.63E+01	1.54E+03	9.46E-04	2.94E-04	1.20E+01	1.13E+03
	50	2.10E-04	6.51E-05	2.65E+00	2.51E+02	1.54E-04	4.77E-05	1.94E+00	1.84E+02
	100	9.58E-05	2.97E-05	1.21E+00	1.15E+02	7.01E-05	2.18E-05	8.87E-01	8.39E+01
	250	3.40E-05	1.06E-05	4.30E-01	4.07E+01	2.49E-05	7.73E-06	3.15E-01	2.98E+01
	500	1.56E-05	4.82E-06	1.96E-01	1.86E+01	1.14E-05	3.53E-06	1.44E-01	1.36E+01
GG-038	10	1.00E-03	1.67E-03	5.17E+01	9.83E+03	7.34E-04	1.22E-03	3.79E+01	7.20E+03
	50	1.63E-04	2.70E-04	8.39E+00	1.60E+03	1.19E-04	1.98E-04	6.15E+00	1.17E+03
	100	7.43E-05	1.24E-04	3.84E+00	7.29E+02	5.44E-05	9.05E-05	2.81E+00	5.34E+02
	250	2.64E-05	4.39E-05	1.36E+00	2.59E+02	1.93E-05	3.21E-05	9.98E-01	1.90E+02
	500	1.21E-05	2.00E-05	6.22E-01	1.18E+02	8.83E-06	1.47E-05	4.56E-01	8.66E+01
GG-039	10	1.06E-03	7.48E-04	9.34E+01	1.09E+04	7.76E-04	5.48E-04	6.84E+01	7.99E+03
	50	1.72E-04	1.21E-04	1.52E+01	1.77E+03	1.26E-04	8.89E-05	1.11E+01	1.30E+03
	100	7.85E-05	5.55E-05	6.93E+00	8.09E+02	5.75E-05	4.06E-05	5.07E+00	5.93E+02
	250	2.79E-05	1.97E-05	2.46E+00	2.87E+02	2.04E-05	1.44E-05	1.80E+00	2.10E+02
	500	1.27E-05	9.00E-06	1.12E+00	1.31E+02	9.33E-06	6.59E-06	8.23E-01	9.61E+01
GG-040	10	1.18E-03	8.59E-04	2.66E+01	4.06E+03	8.61E-04	6.29E-04	1.95E+01	2.98E+03
	50	1.91E-04	1.39E-04	4.31E+00	6.59E+02	1.40E-04	1.02E-04	3.16E+00	4.83E+02
	100	8.72E-05	6.37E-05	1.97E+00	3.01E+02	6.38E-05	4.66E-05	1.44E+00	2.21E+02
	250	3.09E-05	2.26E-05	6.99E-01	1.07E+02	2.27E-05	1.66E-05	5.12E-01	7.83E+01
	500	1.41E-05	1.03E-05	3.20E-01	4.89E+01	1.04E-05	7.56E-06	2.34E-01	3.58E+01
GG-041	10	6.20E-04	6.43E-04	1.35E+01	2.24E+03	4.54E-04	4.71E-04	9.91E+00	1.64E+03
	50	1.01E-04	1.04E-04	2.19E+00	3.63E+02	7.37E-05	7.65E-05	1.61E+00	2.66E+02
	100	4.60E-05	4.77E-05	1.00E+00	1.66E+02	3.37E-05	3.49E-05	7.34E-01	1.22E+02
	250	1.63E-05	1.69E-05	3.56E-01	5.89E+01	1.20E-05	1.24E-05	2.61E-01	4.32E+01
	500	7.46E-06	7.74E-06	1.63E-01	2.69E+01	5.47E-06	5.67E-06	1.19E-01	1.97E+01
GG-042	10	3.49E-04	2.20E-04	3.89E+00	6.69E+02	2.56E-04	1.61E-04	2.85E+00	4.90E+02
	50	5.67E-05	3.56E-05	6.32E-01	1.09E+02	4.15E-05	2.61E-05	4.63E-01	7.95E+01
	100	2.59E-05	1.63E-05	2.89E-01	4.96E+01	1.90E-05	1.19E-05	2.11E-01	3.63E+01
	250	9.20E-06	5.78E-06	1.02E-01	1.76E+01	6.74E-06	4.23E-06	7.51E-02	1.29E+01
	500	4.20E-06	2.64E-06	4.68E-02	8.04E+00	3.08E-06	1.93E-06	3.43E-02	5.89E+00
GG-043	10	3.66E-04	1.86E-03	1.83E+01	5.57E+03	2.68E-04	1.37E-03	1.34E+01	4.08E+03
	50	5.94E-05	3.03E-04	2.97E+00	9.04E+02	4.35E-05	2.22E-04	2.18E+00	6.62E+02
	100	2.72E-05	1.38E-04	1.36E+00	4.13E+02	1.99E-05	1.01E-04	9.94E-01	3.03E+02
	250	9.64E-06	4.91E-05	4.82E-01	1.47E+02	7.06E-06	3.60E-05	3.53E-01	1.07E+02
	500	4.41E-06	2.24E-05	2.20E-01	6.70E+01	3.23E-06	1.64E-05	1.61E-01	4.91E+01
GG-044	10	5.63E-04	1.79E-04	6.72E+00	7.47E+02	4.12E-04	1.31E-04	4.93E+00	5.48E+02
	50	9.13E-05	2.91E-05	1.09E+00	1.21E+02	6.69E-05	2.13E-05	7.99E-01	8.88E+01
	100	4.17E-05	1.33E-05	4.98E-01	5.54E+01	3.06E-05	9.74E-06	3.65E-01	4.06E+01
	250	1.48E-05	4.72E-06	1.77E-01	1.97E+01	1.09E-05	3.46E-06	1.30E-01	1.44E+01
	500	6.77E-06	2.16E-06	8.09E-02	8.99E+00	4.96E-06	1.58E-06	5.92E-02	6.59E+00
GG-045	10	3.45E-04	8.62E-04	8.08E+00	2.15E+03	2.53E-04	6.31E-04	5.92E+00	1.58E+03
	50	5.60E-05	1.40E-04	1.31E+00	3.50E+02	4.10E-05	1.02E-04	9.60E-01	2.56E+02
	100	2.56E-05	6.39E-05	5.99E-01	1.60E+02	1.88E-05	4.68E-05	4.39E-01	1.17E+02
	250	9.09E-06	2.27E-05	2.13E-01	5.67E+01	6.66E-06	1.66E-05	1.56E-01	4.15E+01
	500	4.15E-06	1.04E-05	9.72E-02	2.59E+01	3.04E-06	7.59E-06	7.12E-02	1.90E+01
GG-046	10	2.88E-04	4.31E-05	1.13E+00	1.66E+02	2.11E-04	3.15E-05	8.24E-01	1.21E+02
	50	4.67E-05	6.99E-06	1.83E-01	2.69E+01	3.42E-05	5.12E-06	1.34E-01	1.97E+01
	100	2.13E-05	3.19E-06	8.34E-02	1.23E+01	1.56E-05	2.34E-06	6.11E-02	8.99E+00
	250	7.58E-06	1.13E-06	2.96E-02	4.36E+00	5.55E-06	8.30E-07	2.17E-02	3.19E+00
	500	3.46E-06	5.18E-07	1.35E-02	1.99E+00	2.54E-06	3.79E-07	9.92E-03	1.46E+00
GG-047	10	2.51E-04	1.13E-03	1.22E+01	2.99E+03	1.84E-04	8.29E-04	8.94E+00	2.19E+03
	50	4.07E-05	1.84E-04	1.98E+00	4.84E+02	2.98E-05	1.35E-04	1.45E+00	3.55E+02
	100	1.86E-05	8.39E-05	9.05E-01	2.21E+02	1.36E-05	6.15E-05	6.63E-01	1.62E+02
	250	6.61E-06	2.98E-05	3.21E-01	7.86E+01	4.84E-06	2.18E-05	2.35E-01	5.76E+01
	500	3.02E-06	1.36E-05	1.47E-01	3.59E+01	2.21E-06	9.97E-06	1.08E-01	2.63E+01
GG-048	10	3.80E-04	7.44E-04	8.73E+00	2.60E+03	2.78E-04	5.45E-04	6.40E+00	1.90E+03
	50	6.16E-05	1.21E-04	1.42E+00	4.21E+02	4.51E-05	8.84E-05	1.04E+00	3.08E+02
	100	2.81E-05	5.51E-05	6.47E-01	1.92E+02	2.06E-05	4.04E-05	4.74E-01	1.41E+02
	250	9.99E-06	1.96E-05	[illegible]	6.83E+01	7.32E-06	1.43E-05	1.68E-01	5.00E+01
	500	4.57E-06	8.95E-06	1.05E-01	3.12E+01	3.34E-06	6.55E-06	7.69E-02	2.29E+01

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TABLE D.3-2. (continued)

SOURCE TERM PARTITION	DISTANCE (M)	A STABILITY, 5 M/S REL1 DOSE RATE (SV/S)	A STABILITY, 5 M/S REL2 DOSE RATE (SV/S)	A STABILITY, 5 M/S 15-MIN DOSE (SV)	A STABILITY, 5 M/S TOTAL DOSE (SV)	F STABILITY, 1 M/S REL1 DOSE RATE (SV/S)	F STABILITY, 1 M/S REL2 DOSE RATE (SV/S)	F STABILITY, 1 M/S 15-MIN DOSE (SV)	F STABILITY, 1 M/S TOTAL DOSE (SV)
GG-049	10	1.25E-04	2.08E-05	4.20E+02	3.69E+04	9.18E-05	1.53E-05	3.08E+02	2.70E+04
	50	2.03E-05	3.38E-06	6.82E+01	5.99E+03	1.49E-05	2.47E-06	4.99E+01	4.39E+03
	100	9.29E-06	1.54E-06	3.12E+01	2.74E+03	6.80E-06	1.13E-06	2.28E+01	2.00E+03
	250	3.30E-06	5.48E-07	1.11E+01	9.72E+02	2.42E-06	4.01E-07	8.10E+00	7.12E+02
	500	1.51E-06	2.50E-07	5.05E+00	4.44E+02	1.10E-06	1.83E-07	3.70E+00	3.25E+02
GG-050	10	6.24E-05	1.50E-05	1.48E+02	1.45E+04	4.57E-05	1.10E-05	1.09E+02	1.06E+04
	50	1.01E-05	2.43E-06	2.41E+01	2.36E+03	7.41E-06	1.78E-06	1.76E+01	1.73E+03
	100	4.62E-06	1.11E-06	1.10E+01	1.08E+03	3.39E-06	8.14E-07	8.06E+00	7.89E+02
	250	1.64E-06	3.94E-07	3.91E+00	3.82E+02	1.20E-06	2.89E-07	2.86E+00	2.80E+02
	500	7.50E-07	1.80E-07	1.79E+00	1.75E+02	5.49E-07	1.32E-07	1.31E+00	1.28E+02
GG-051	10	5.32E-05	2.64E-05	5.89E+01	8.58E+03	3.89E-05	1.94E-05	4.31E+01	6.29E+03
	50	8.62E-06	4.29E-06	9.55E+00	1.39E+03	6.32E-06	3.14E-06	7.00E+00	1.02E+03
	100	3.94E-06	1.96E-06	4.37E+00	6.36E+02	2.89E-06	1.44E-06	3.20E+00	4.66E+02
	250	1.40E-06	6.96E-07	1.55E+00	2.26E+02	1.02E-06	5.10E-07	1.14E+00	1.65E+02
	500	6.39E-07	3.18E-07	7.08E-01	1.03E+02	4.68E-07	2.33E-07	5.19E-01	7.56E+01
GG-052	10	1.25E-05	7.70E-06	6.67E+00	1.26E+03	9.16E-06	5.64E-06	6.35E+00	9.24E+02
	50	2.03E-06	1.25E-06	1.41E+00	2.05E+02	1.49E-06	9.15E-07	1.03E+00	1.50E+02
	100	9.27E-07	5.71E-07	6.42E-01	9.35E+01	6.79E-07	4.18E-07	4.71E-01	6.85E+01
	250	3.29E-07	2.03E-07	2.28E-01	3.32E+01	2.41E-07	1.49E-07	1.67E-01	2.43E+01
	500	1.50E-07	9.26E-08	1.04E-01	1.52E+01	1.10E-07	6.79E-08	7.63E-02	1.11E+01
GG-053	10	2.78E-06	1.93E-06	1.63E+00	2.15E+02	2.03E-06	1.41E-06	1.20E+00	1.57E+02
	50	4.50E-07	3.13E-07	2.65E-01	3.49E+01	3.30E-07	2.29E-07	1.94E-01	2.55E+01
	100	2.06E-07	1.43E-07	1.21E-01	1.59E+01	1.51E-07	1.05E-07	8.87E-02	1.17E+01
	250	7.30E-08	5.08E-08	4.30E-02	5.66E+00	5.35E-08	3.72E-08	3.15E-02	4.14E+00
	500	3.34E-08	2.32E-08	1.96E-02	2.58E+00	2.44E-08	1.70E-08	1.44E-02	1.89E+00
GG-054	10	7.08E-07	5.11E-07	2.82E-01	3.09E+01	5.19E-07	3.74E-07	2.07E-01	2.27E+01
	50	1.15E-07	8.29E-08	4.58E-02	5.02E+00	8.42E-08	6.07E-08	3.36E-02	3.68E+00
	100	5.25E-08	3.79E-08	2.09E-02	2.29E+00	3.85E-08	2.77E-08	1.53E-02	1.68E+00
	250	1.86E-08	1.34E-08	7.43E-03	8.14E-01	1.37E-08	9.85E-09	5.44E-03	5.96E-01
	500	8.52E-09	6.14E-09	3.40E-03	3.72E-01	6.24E-09	4.50E-09	2.49E-03	2.72E-01
GG-055	10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	50	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	100	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	250	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	500	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E-00	0.00E+00

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Appendix E Supporting Information for the Risk Analysis

Table E-1contains the descriptive statistics for the risk distributions for each plant damage state The distributions were generated using a LHS sample with 200 observations The risk measures are discribed in Section 8.

Table E-1
Plant Damae State Risk Results

PDS	Consequence Measure	Descriptive Statistics				
		5Th PCT	50TH PCT	95TH PCT	MEAN	STD DEV
Total	Frequency	4.1E-07	1.4E-06	5.6E-06	2.1E-06	2.7E-06
Total	Early Fatality	3.7E-11	2.8E-09	3.9E-08	1.4E-08	5.4E-08
Total	Total Lat. Cancer	4.3E-04	1.9E-03	1.2E-02	3.8E-03	7.6E-03
Total	Pop. Dose within 50 miles	1.3E-03	5.3E-03	3.1E-02	9.9E-03	1.9E-02
Total	Pop. Dose within 1000 miles	9.9E-03	4.4E-02	2.8E-01	8.7E-02	1.7E-01
Total	Indiv. Early Fatalities within 1 mile	4.2E-13	2.6E-11	3.0E-10	9.6E-11	3.4E-10
Total	Indiv. Lat. Cancers within 10 miles	2.5E-10	9.4E-10	4.8E-09	1.6E-09	2.4E-09
PDS1-1	Frequency	1.6E-09	1.4E-08	1.9E-07	4.1E-08	8.3E-08
PDS1-1	Early Fatality	5.3E-13	2.7E-11	5.6E-10	1.2E-10	5.0E-10
PDS1-1	Total Lat. Cancer	1.7E-06	1.9E-05	3.3E-04	6.7E-05	1.4E-04
PDS1-1	Pop. Dose within 50 miles	5.2E-06	5.5E-05	8.3E-04	1.9E-04	4.0E-04
PDS1-1	Pop. Dose within 1000 miles	4.1E-05	4.3E-04	7.4E-03	1.6E-03	3.3E-03
PDS1-1	Indiv. Early Fatalities within 1 mile	6.4E-15	3.0E-13	6.9E-12	1.1E-12	2.6E-12
PDS1-1	Indiv. Lat. Cancers within 10 miles	1.0E-12	1.0E-11	1.5E-10	3.0E-11	6.3E-11
PDS1-2	Frequency	1.4E-10	4.3E-09	1.3E-07	2.3E-08	4.8E-08
PDS1-2	Early Fatality	1.1E-13	2.9E-11	1.3E-09	3.3E-10	1.6E-09
PDS1-2	Total Lat. Cancer	3.4E-07	9.6E-06	3.4E-04	6.5E-05	1.5E-04
PDS1-2	Pop. Dose within 50 miles	9.6E-07	2.5E-05	8.0E-04	1.6E-04	3.6E-04
PDS1-2	Pop. Dose within 1000 miles	7.8E-06	2.2E-04	7.7E-03	1.5E-03	3.5E-03
PDS1-2	Indiv. Early Fatalities within 1 mile	1.4E-15	2.9E-13	1.1E-11	2.3E-12	7.6E-12
PDS1-2	Indiv. Lat. Cancers within 10 miles	1.1E-13	2.9E-12	9.7E-11	1.8E-11	4.2E-11
PDS1-3	Frequency	2.9E-09	1.7E-08	1.6E-07	4.4E-08	1.2E-07
PDS1-3	Early Fatality	1.4E-13	1.2E-10	1.4E-09	3.0E-10	5.7E-10
PDS1-3	Total Lat. Cancer	3.1E-06	3.4E-05	2.9E-04	7.9E-05	1.3E-04
PDS1-3	Pop. Dose within 50 miles	1.0E-05	9.3E-05	7.6E-04	2.0E-04	3.5E-04
PDS1-3	Pop. Dose within 1000 miles	7.3E-05	7.9E-04	6.7E-03	1.8E-03	3.0E-03
PDS1-3	Indiv. Early Fatalities within 1 mile	1.8E-15	1.1E-12	1.0E-11	2.5E-12	4.5E-12
PDS1-3	Indiv. Lat. Cancers within 10 miles	1.7E-12	1.1E-11	9.2E-11	2.7E-11	5.9E-11
PDS1-4	Frequency	4.7E-11	2.0E-09	3.5E-08	9.1E-09	2.9E-08
PDS1-4	Early Fatality	2.9E-14	1.0E-11	3.2E-10	6.5E-11	1.9E-10
PDS1-4	Total Lat. Cancer	5.3E-08	3.7E-06	8.1E-05	1.9E-05	5.4E-05
PDS1-4	Pop. Dose within 50 miles	1.7E-07	1.0E-05	1.8E-04	4.8E-05	1.4E-04
PDS1-4	Pop. Dose within 1000 miles	1.2E-06	8.5E-05	1.8E-03	4.4E-04	1.3E-03
PDS1-4	Indiv. Early Fatalities within 1 mile	3.7E-16	9.8E-14	2.5E-12	5.3E-13	1.4E-12
PDS1-4	Indiv. Lat. Cancers within 10 miles	3.9E-14	1.3E-12	2.4E-11	6.1E-12	1.9E-11

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Table E-1 (continued)
Plant Damae State Risk Results

PDS	Consequence Measure	Descriptive Statistics				
		5Th PCT	50TH PCT	95TH PCT	MEAN	STD DEV
PDS1-5	Frequency	4.4E-10	6.8E-09	4.8E-08	1.4E-08	2.4E-08
PDS1-5	Early Fatality	1.2E-14	5.9E-12	1.4E-10	3.0E-11	5.8E-11
PDS1-5	Total Lat. Cancer	2.8E-07	5.8E-06	5.8E-05	1.4E-05	2.3E-05
PDS1-5	Pop. Dose within 50 miles	1.1E-06	1.9E-05	1.5E-04	4.2E-05	7.1E-05
PDS1-5	Pop. Dose within 1000 miles	6.6E-06	1.4E-04	1.3E-03	3.3E-04	5.4E-04
PDS1-5	Indiv. Early Fatalities within 1 mile	1.6E-16	6.9E-14	1.5E-12	3.1E-13	5.6E-13
PDS1-5	Indiv. Lat. Cancers within 10 miles	1.9E-13	3.3E-12	2.4E-11	7.2E-12	1.3E-11
PDS2-1	Frequency	1.3E-08	1.4E-07	1.5E-06	3.5E-07	6.6E-07
PDS2-1	Early Fatality	2.3E-14	8.6E-11	2.5E-09	6.6E-10	2.6E-09
PDS2-1	Total Lat. Cancer	1.6E-05	1.8E-04	3.0E-03	5.8E-04	1.2E-03
PDS2-1	Pop. Dose within 50 miles	4.8E-05	5.2E-04	7.9E-03	1.6E-03	3.2E-03
PDS2-1	Pop. Dose within 1000 miles	3.8E-04	4.2E-03	6.9E-02	1.4E-02	2.8E-02
PDS2-1	Indiv. Early Fatalities within 1 mile	3.0E-16	1.0E-12	2.9E-11	6.0E-12	1.7E-11
PDS2-1	Indiv. Lat. Cancers within 10 miles	7.8E-12	9.0E-11	1.3E-09	2.6E-10	5.0E-10
PDS2-2	Frequency	2.2E-08	1.5E-07	1.6E-06	5.5E-07	1.9E-06
PDS2-2	Early Fatality	4.6E-14	1.1E-09	2.5E-08	1.0E-08	5.0E-08
PDS2-2	Total Lat. Cancer	4.4E-05	3.5E-04	4.3E-03	1.7E-03	6.4E-03
PDS2-2	Pop. Dose within 50 miles	1.0E-04	9.1E-04	1.1E-02	4.0E-03	1.5E-02
PDS2-2	Pop. Dose within 1000 miles	1.0E-03	8.1E-03	9.8E-02	3.8E-02	1.5E-01
PDS2-2	Indiv. Early Fatalities within 1 mile	5.8E-16	9.9E-12	1.6E-10	6.8E-11	3.0E-10
PDS2-2	Indiv. Lat. Cancers within 10 miles	1.7E-11	1.1E-10	1.7E-09	5.1E-10	1.8E-09
PDS2-3	Frequency	2.7E-09	2.9E-08	4.5E-07	1.1E-07	3.7E-07
PDS2-3	Early Fatality	1.8E-14	2.1E-10	4.9E-09	1.3E-09	4.9E-09
PDS2-3	Total Lat. Cancer	3.7E-06	6.4E-05	1.1E-03	2.8E-04	1.0E-03
PDS2-3	Pop. Dose within 50 miles	1.2E-05	1.5E-04	2.5E-03	6.9E-04	2.5E-03
PDS2-3	Pop. Dose within 1000 miles	8.7E-05	1.5E-03	2.5E-02	6.4E-03	2.3E-02
PDS2-3	Indiv. Early Fatalities within 1 mile	2.3E-16	1.9E-12	3.6E-11	9.9E-12	3.8E-11
PDS2-3	Indiv. Lat. Cancers within 10 miles	1.8E-12	2.2E-11	3.3E-10	8.5E-11	2.6E-10
PDS2-4	Frequency	7.7E-09	8.8E-08	7.5E-07	2.0E-07	3.0E-07
PDS2-4	Early Fatality	2.0E-15	3.2E-11	1.4E-09	2.7E-10	7.4E-10
PDS2-4	Total Lat. Cancer	4.8E-06	9.4E-05	8.3E-04	2.1E-04	3.6E-04
PDS2-4	Pop. Dose within 50 miles	1.8E-05	3.0E-04	2.6E-03	6.8E-04	1.1E-03
PDS2-4	Pop. Dose within 1000 miles	1.1E-04	2.2E-03	2.0E-02	5.0E-03	8.4E-03
PDS2-4	Indiv. Early Fatalities within 1 mile	2.5E-17	3.9E-13	1.6E-11	2.8E-12	6.9E-12
PDS2-4	Indiv. Lat. Cancers within 10 miles	3.4E-12	5.1E-11	4.4E-10	1.1E-10	1.5E-10

Table E-1 (concluded)
Plant Damae State Risk Results

PDS	Consequence Measure	Descriptive Statistics				
		5Th PCT	50TH PCT	95TH PCT	MEAN	STD DEV
PDS2-5	Frequency	7.7E-11	2.7E-09	5.3E-08	1.3E-08	3.4E-08
PDS2-5	Early Fatality	1.2E-14	6.0E-12	2.3E-10	9.4E-11	4.8E-10
PDS2-5	Total Lat. Cancer	1.2E-07	4.4E-06	1.2E-04	2.7E-05	8.3E-05
PDS2-5	Pop. Dose within 50 miles	3.5E-07	1.3E-05	2.9E-04	6.9E-05	2.1E-04
PDS2-5	Pop. Dose within 1000 miles	2.7E-06	1.0E-04	2.8E-03	6.1E-04	1.9E-03
PDS2-5	Indiv. Early Fatalities within 1 mile	1.2E-16	5.2E-14	2.0E-12	6.5E-13	3.1E-12
PDS2-5	Indiv. Lat. Cancers within 10 miles	6.5E-14	1.9E-12	3.9E-11	9.5E-12	2.7E-11
PDS2-6	Frequency	1.7E-11	1.1E-09	2.8E-08	7.4E-09	2.2E-08
PDS2-6	Early Fatality	8.7E-18	9.9E-13	1.0E-10	2.9E-11	1.4E-10
PDS2-6	Total Lat. Cancer	1.2E-08	1.3E-06	5.7E-05	1.2E-05	3.0E-05
PDS2-6	Pop. Dose within 50 miles	4.3E-08	4.0E-06	1.5E-04	3.3E-05	8.2E-05
PDS2-6	Pop. Dose within 1000 miles	2.7E-07	3.1E-05	1.3E-03	2.8E-04	6.8E-04
PDS2-6	Indiv. Early Fatalities within 1 mile	1.1E-19	9.9E-15	6.9E-13	2.1E-13	9.1E-13
PDS2-6	Indiv. Lat. Cancers within 10 miles	1.0E-14	6.8E-13	2.4E-11	5.2E-12	1.4E-11
PDS3-1	Frequency	6.2E-08	3.7E-07	2.4E-06	7.3E-07	1.2E-06
PDS3-1	Early Fatality	0.0E+00	0.0E+00	8.6E-11	1.0E-10	6.5E-10
PDS3-1	Total Lat. Cancer	2.8E-05	3.6E-04	3.5E-03	8.2E-04	1.5E-03
PDS3-1	Pop. Dose within 50 miles	1.2E-04	1.0E-03	8.6E-03	2.2E-03	3.8E-03
PDS3-1	Pop. Dose within 1000 miles	6.3E-04	7.8E-03	7.8E-02	1.8E-02	3.3E-02
PDS3-1	Indiv. Early Fatalities within 1 mile	0.0E+00	0.0E+00	1.1E-12	1.0E-12	6.6E-12
PDS3-1	Indiv. Lat. Cancers within 10 miles	3.4E-11	2.6E-10	2.0E-09	5.6E-10	9.6E-10

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Appendix F Summary Report for Abridged Study of POS 6

For the sake of completeness, this appendix contains a letter report titled "Summary Report: Grand Gulf Low Power and Shutdown Abridged Risk Analysis, POS 6: Early Refueling." This reports documents the abridged study of the early portion of the refueling mode of operation, referred to as POS 6, that was performed under FIN L1679 during the Spring of 1992. The reader is cautioned that there are many differences between the study of POS 5 and the abridged study of POS 6 and the two studies should not be viewed as equal in scope or approach. The abridged study of POS 6 used a much more abbreviated version of the NUREG-1150 methodology. Some of the more important differences include:

- The results from POS 6 are conditional on the occurrence of the plant damage states (PDS) defined in the analysis. Only two PDSs were defined, LOSP and non-LOSP, and these PDSs were based on information from a Level 1 coarse screening study of the Grand Gulf low power and shutdown modes of operation; the screening study did not provide core damage frequencies. Since core damage frequencies for the PDSs were not available, the analysis of POS 6 provides no information on the likelihood of the accidents.

- Estimates of the risk associated with POS 6 were not calculated since core damage frequencies for POS 6 were not available.

- The Accident Progression Event Tree (APET) used in the POS 6 study did not address as many events and the events it did include were treated less detail as compared to the POS 5 APET. Also, the POS 6 analysis included fewer events in its uncertainty analysis, as compared to the POS 5 analysis.

- In the calculation of offsite consequences, the POS 6 analysis used an LHS sample with only 12 observations as opposed to the 200 observations used in the POS 5 analysis. Thus, while the results from the 12 observations can be used to provide an indication of the range of expected results, meaningful statistics cannot be calculated based on this limited sample.

- The onsite consequence analysis for POS 6 was based on a slightly different set of assumptions as compared to the onsite POS 5 analysis. In both studies, the dose was based on exposure from both the immersion and the inhalation pathways. In the POS 6 analysis, the dose rate was also based on both the immersion and the inhalation pathways (very conservative to include inhalation pathway in the calculation of the dose rate) whereas in the POS 5 analysis the dose rate was only based on exposure from the immersion pathway. The dose and dose rates calculated in the two studies were based on different weather scenarios. Also, dose and dose rates in the containment and auxiliary building were calculated in the POS 6 study; similar calculations were not performed in the POS 5 study.

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SUMMARY REPORT OF:
GRAND GULF LOW POWER AND SHUTDOWN
ABRIDGED RISK ANALYSIS

POS 6: Early Refueling

FINAL LETTER REPORT

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ACRONYMS AND INITIALISMS

ADHR	Alternate Decay Heat Removal System
APET	Accident Progression Event Tree
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CCI	Core Concrete Interactions
CDS	Condensate System
CF	Containment Failure
CNT	Containment
CRD	Control Rod Drive System
ECCS	Emergency Core Cooling System
FW	Firewater System
HIS	Hydrogen Ignition System
HPCS	High Pressure Core Spray System
HRA	Human Reliability Analysis
LHS	Latin Hypercube Sample
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
LP&S	Low Power and Shutdown
NRC	Nuclear Regulatory Commission
PDS	Plant Damage State
POS	Plant Operating State
PRA	Probabilistic Risk Assessment
PSW	Plant Service Water
PWR	Pressurized Water Reactor
RFO	Refueling Outage
RHR	Residual Heat Removal System
RPV	Reactor Pressure Vessel
SBGT	Standby Gas Treatment System
SBO	Station Blackout
SDC	Shutdown Cooling System
SPC	Suppression Pool Cooling System
SPMU	Suppression Pool Makeup System
SRV	Safety Relief Valve
SSW	Standby Service Water
TAF	Top of Active Fuel
TBCW	Turbine Building Cooling Water
VB	Vessel Breach

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1.0 INTRODUCTION

1.1 Study Objectives

The Office of Nuclear Regulatory Research at the U. S. Nuclear Regulatory Commission established programs to investigate postulated accidents during low power and shutdown (LP&S) operations of a BWR (Grand Gulf) and a PWR (Surry). One such program is a risk study of accident progressions and consequences.

The objective of this study is to make a preliminary risk determination of the progressions (Level 2 analysis) and the consequences (Level 3 analysis) of accidents during low power and shutdown operations in the Grand Gulf plant. The study was designed to obtain results for regulatory decisions. This letter report documents the methods, findings, and implications of the study done under NRC FIN L1679. A sister study of the Surry plant is reported separately by the staff at Brookhaven National Laboratory (BNL) under NRC FIN L1680.

1.2 Scope of the Study

The abbreviated risk analysis took place from January through April 1992. The study has been referred to as an *abridged risk* analysis. The term *abridged* means that simple event trees (about nine top event questions) were developed and used with assumptions and other approximate methods to compute rough estimates. The term *risk* means conditional consequences (probability of the various events during the accident progressions multiplied by the consequences), given that core damage has occurred. Traditional risk estimates, computed by multiplying the conditional consequences and the frequency of the sequences, could not be made at this time because the core damage frequencies have yet to be determined in companion Level 1 and HRA studies. Uncertainty has been taken into account in a manner consistent with the detail of the abridged study.

This study investigated the possible accident progressions and the associated consequences of a single plant operating state, POS 6, an early stage of refueling, where the reactor vessel head is removed, the steam dryers and separators are removed, the drywell is open, and the containment is open. The sister study at BNL investigated mid-loop operation. The scope of both studies is illustrated in Figure 1-1.

1.3 Methods

The abridged process of computing conditional consequences is shown in Figure 1-2. In general, both the study reported here and the study done at BNL follow this scheme. Some differences in the details of the procedure exist and are noted at the end of Section 1.3. The process used here is an abbreviated form of the NUREG-1150 study [1].

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Accident progressions

The calculations begin with the assumption that core damage has occurred. Given core damage, the reasonable accident progressions are delineated with the accident progression event tree (APET). Much of the delineation is based on information obtained from PRAs of full power operation, knowledge of severe accident phenomena, and deterministic calculations with codes used to compute source terms, such as MELCOR [2]. The likelihood of the various accident progressions is reflected vis-a-vis branch point probabilities.

Branch point probabilities were assigned to reflect the likelihood of various pathways thought to exist. In large scale risk studies, the assignment can be done by groups of *experts* knowledgeable in severe accident issues. Here, because of resource limitations, most of the assignments were done by the project staff. The probabilities are not as rigorous as they could be but this is one of many limitations of the study to be discussed. Some lack of rigor in determining the probabilities is taken into account by repeating the calculations with other possible probabilities; taken together, the repeated calculations constitute an uncertainty analysis.

Through the uncertainty analysis, distributions, instead of point values, were assigned to selected branch points. The distributions are subjective but account for many possible values of the branch points. Point values are selected from the distributions with a form of Monte Carlo sampling known as Latin Hypercube Sampling (LHS) [3]. After making sets of inputs, each set, consisting of point values, is assigned to the branch points and multiplied through to the ends of the APET. The calculations are repeated using the sets of inputs, building a probability distribution a. the end of each pathway.

Source terms

Having delineated accident progressions with the APET, the source terms of the progressions were calculated with a parametric code [4]. The parametric code is a collection of simple mass-balance equations designed to mimic detailed source term codes. The parametric approach is not meant to be a substitute for detailed, mechanistic computer simulations codes. Rather, it is a framework for integrating the results of these codes together with experimental results and expert judgment.

The parametric code determines source terms, given the characteristics of the accident progression and other inputs (e.g., fraction of the inventory a) leaving the reactor vessel; b) involved in core concrete interactions; c) entering the containment). Because these other variables are imprecisely known, many reasonable values can be assigned to the inputs. As in the APET calculations, distributions are assigned to the variables and sampled with LHS to form many sets of input values for repeated calculations. The result is a distribution of source terms for each accident progression pathway.

Because the estimation of the source terms is a critical component of this study, an internal advisory group, call the Source Term Advisory Group, was formed to support this study. The

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members of the advisory group included: John E. Kelly (SNL), Hossein P. Nourbakhsh (BNL), Dana A. Powers (SNL), and Trevor Pratt (BNL). The role of the Source Term Advisory Group was to 1) provide guidance on the identification of phenomena that may be important to the formation of the source term during these modes of operation, and 2) assess the adequacy, relative to the study's objectives and scope, of the assumptions, methods and data used in this study. The results of the accident progression and source term analysis were presented to and discussed with the advisory group in two meetings during the course of this analysis.

Consequences

Three sets of radiological consequences were determined: building dose, onsite dose (so called *parking lot* dose), and offsite consequences.

- o Building dose was determined based on source terms derived from the parametric source term expressions. Doses in the containment and auxiliary building were estimated.

- o Parking lot dose was based on relative concentrations computed with the Ramsdell model [11], in which the release concentration is somewhat proportional to wind speed, and a combination of the Wilson model [12] and the model in Regulatory Guide 1.145 [13], in which the concentration is inversely proportional to wind speed.

- o Offsite consequences were computed using the MACCS code [5,6,7].

Uncertainty was not propagated through the consequence analysis as it was through the APET and the source term calculations. While a sample size of 100 was used in the onsite analysis to propagate accident progression and source term uncertainties, a reduced sample size of 12 was used in the determination of offsite consequences.

Conditional offsite consequences

Conditional risk was computed by multiplying the offsite consequences by their associated accident probability that was determined with the APET. This product of probability and consequences was computed for each accident progression pathway. The products of the pathways were summed. This process was repeated for each of the few samples of the source terms. Then, high, medium, and low results were reported.

Differences

This study differs slightly from its sister program at BNL in three ways. (1) Here, one hundred samples from the uncertainty distributions were propagated through the accident progression and source term analyses whereas, in the BNL study, two hundred samples were taken. (2) Here, twelve samples were propagated through the APET to offsite consequences whereas, in the BNL study, twenty samples from the source term distributions were used in consequence calculations and traced back through the APET for the probabilities needed to compute conditional risk.

(3) Here, uoses in the containment and auxiliary building were calculated whereas, in the BNL study, these calculations were thought unnecessary since the releases were assumed to pass from the containment directly into the environment.

1.4 Limitations and Strengths of the Study

In order to place the calculations in proper context, it is necessary to understand the strengths and limitations of the study.

Limitations

- o The subject of the study is one POS, early refueling. This POS was selected for study because it was identified in a preliminary Level 1 study, known as a coarse screening analysis [8], as potentially occurring at a relatively high frequency. Also, the POS had characteristics (i.e., reactor vessel head removed) of interest to the staff in the Office of Nuclear Reactor Regulations at the NRC.

- o The abridged study is based on the coarse screening analysis where accident sequences potentially having high frequencies were identified. The consequences of these sequences were determined in the Level 2 and 3 abridged study reported here. The frequency is not merged with the Level 2 and 3 calculations to determine risk because the numerical value of the frequency estimate is believed to be too rough for such use.

- o The simple APET accounts for a limited number of factors. The APET consisted of nine top event questions, compared to about one hundred questions in a large scale PRA.

- o The onsite dose estimates stem from simple equations yielding rough estimates.

- o Variables were selected and assigned distributions for the uncertainty analysis by the project staff.

- o Because of gaps in knowledge of the plant configuration and operator actions, assumptions were necessary. The assumptions are documented in the sections to follow.

Strengths

- o Even with the limitations noted above, the abridged study is a systematic evaluation, which includes a limited treatment of the uncertainty in severe accident progressions.

- o The source term analysis was reviewed by an internal advisory group.

- o The project staff and the NRC project staff believe that the APET represents the occurrence of key events during accident progressions.

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- The relationship and timing of accident progression events and factors have been determined to at least a first approximation.

- Estimates of both onsite and offsite conditional consequences were made.

The sections to follow document the abridged study of the Grand Gulf plant. The discussion above is expanded, providing important details and results.

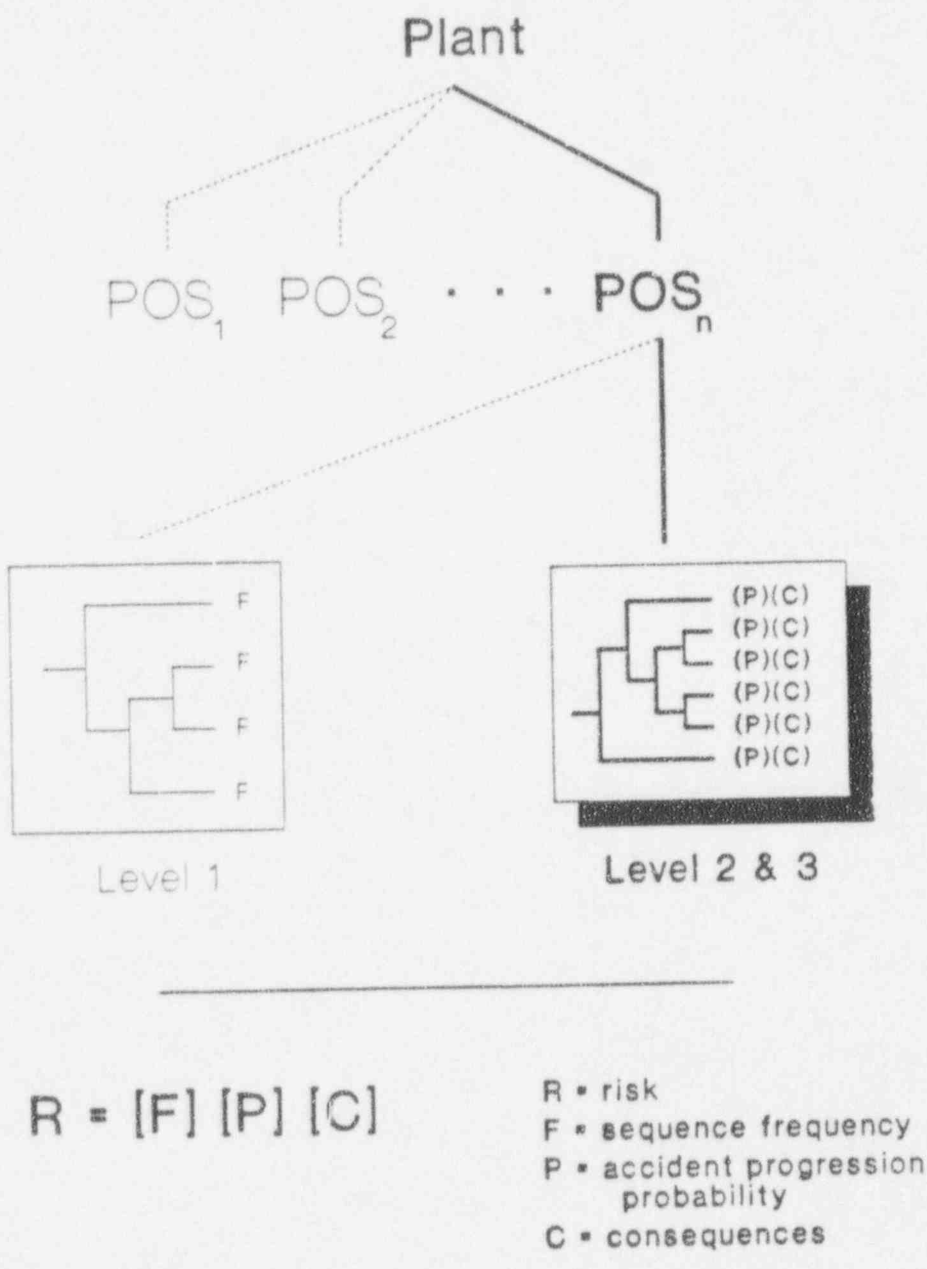

Figure 1-1. Scope of abridged study

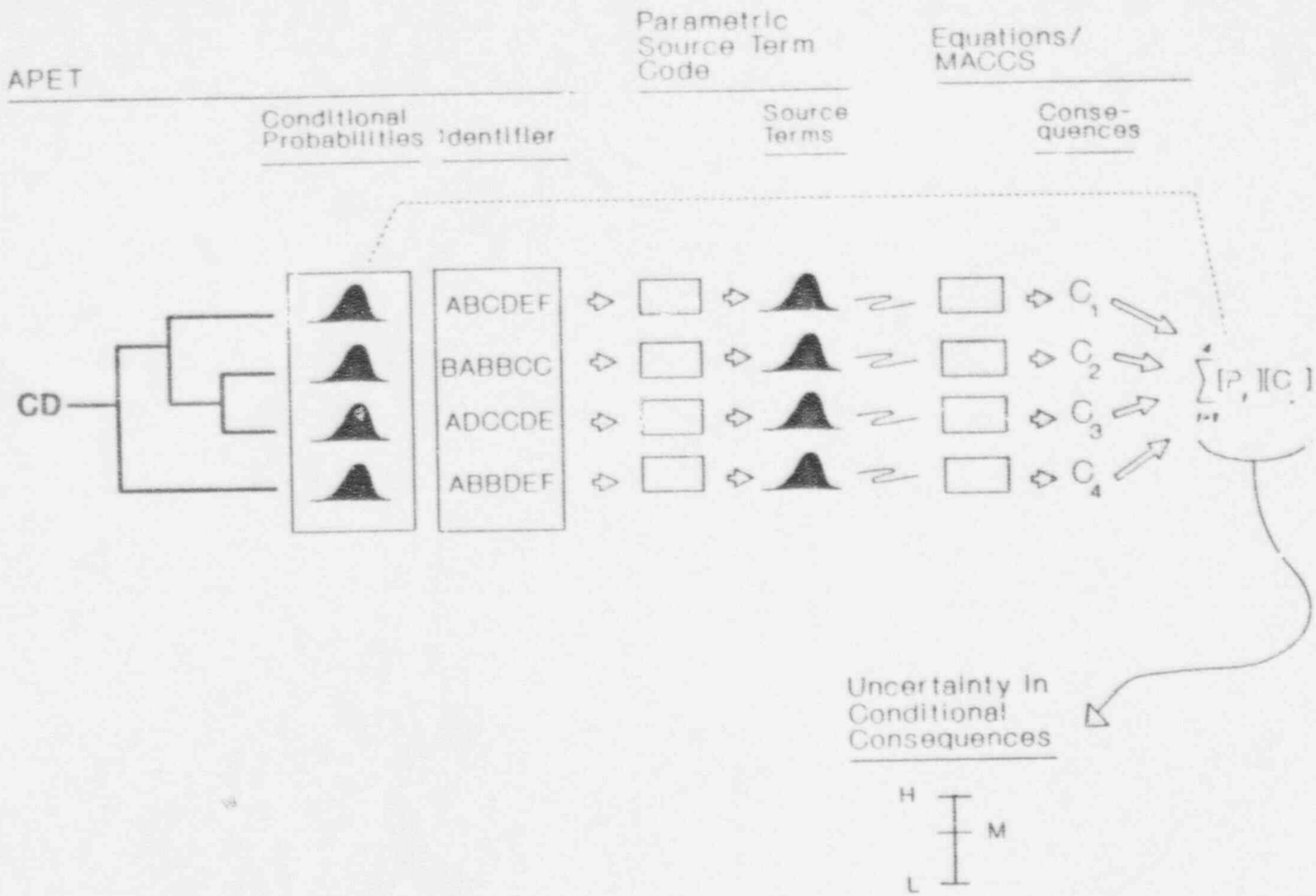

Figure 1-2. Summary of abridged methodology

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2.0 ACCIDENT PROGRESSION ANALYSIS

2.1 Approach

The progression of accidents following core damage are analyzed in the Level 2 portion of the PRA. In this chapter the development and quantification of the accident progression scenarios will be presented. The input to the accident progression analysis is the core damage sequence definitions developed in the Level 1 analysis [8]. The core damage sequences define the successes and failures of equipment and human actions that have resulted in the loss of core cooling and the onset of core damage. The sequence definitions provide information on the status of core cooling systems, containment cooling systems, and containment integrity at the time of core damage. From this information the possible accident progressions, which identify the response of the core and the containment following core damage, are determined. These accident progressions are developed and displayed using an event tree. In this abridged analysis only the most important events that affect the timing and the magnitude of the radionuclide release are addressed. The outputs from the accident progression analysis are the accident progression path definitions and the likelihood, conditional on core damage having occurred, of each path. In the source term analysis, the fission product release associated with each path is estimated. The estimation of the source term is addressed in Chapter 3 and the resulting consequences are presented in Chapter 4.

In the following subsections the configuration of the plant during POS 6 will be presented, the important characteristics of the Level 1 core damage sequences will be identified, and the development of the accident progression paths will be discussed.

2.2 POS 6 Plant Configuration

The configuration of the plant at the onset of core damage is important because it will determine the framework within which the accident will unfold. That is, the plant configuration will define the boundary conditions for the analysis. For example, it will define the mitigative features of the plant that will be available during the accident (e.g., containment, suppression pool, containment sprays).

The abridged risk analysis was performed on the early portion of the refueling mode of operation, referred to as plant operating state 6 (POS 6). During a refueling outage the plant will enter POS 6 prior to loading fresh fuel (i.e., going down) and then following fuel transfer on the way back up to power conditions (i.e., going up). In the Level 1 analysis, the sequence definitions are based on the "going down" phase because (1) more systems are likely to be unavailable (i.e., on the way back up, maintenance and repairs may already have been performed on many systems) and (2) the decay heat levels are higher and, therefore, there is less time to respond to events in the going down phase versus the going up phase. Thus, in this study only the "going down" phase is analyzed. POS 6 begins when the vessel head is detached and ends when the upper reactor cavity has been filled with water. During this POS the following tasks are performed:

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1. Steam dryers are removed,
2. Vessel water level is lowered to the bottom of the steam lines and the steam lines are plugged,
3. Water level is raised and the steam separators are removed, and
4. Vessel water level is raised to flood the upper reactor cavity.

Prior to this mode of operation, the containment equipment hatch and personnel locks have been opened, the drywell head has been removed, and the drywell equipment hatch and personnel locks have been opened. Thus, the suppression pool is effectively bypassed both from the vessel and from the drywell (i.e., steam lines are plugged and the drywell is open).

Timing information for the initiation of the accident in POS 6 is based on Grand Gulf refueling outage (RFO) data. Information was available for the first four RFOs. However, because of the number of special tests that were conducted during the first refueling outage, RFO-1 was considered atypical and, therefore, data from this outage was excluded from the analysis. Thus, only RFO 2,3, and 4 data were used in this study. Based on this data the fastest the plant will enter POS 6 from full power is approximately four days after shutdown and the longest the plant has been in POS 6 (in the "going down" phase) is approximately 12 days (i.e., 16 days from shutdown). In the Level 1 analysis the time window from the initiating event to core damage was based on the decay heat at four days. This assumption is carried through the Level 2/3 analyses.

2.3 Level 1 Sequence Description

2.3.1 Sequence Description

The initial conditions for the accident progression analysis are the core damage sequence descriptions from the Level 1 analysis [8]. That is, a list of attributes that describe the status of systems that can be used to mitigate the accident and the configuration of the plant at the time of core damage. In the Level 1 coarse screening analysis the sequences were placed into three groups: potentially high likelihood group, potentially medium likelihood group, and potentially low likelihood group. Only sequences from the high likelihood group were analyzed in this study. Fourteen different initiating events are associated with these sequences. A list of these 14 initiating events is presented in Table 2.3-1. The initiating events can be divided into four major groups: Loss of Offsite Power (LOSP) Transients, Loss of Support System Transients, Loss of Coolant Accidents (LOCAs), and Decay Heat Removal Challenges. The accident sequences that form the input to this study all progress to core damage in the following manner. The initiating event leads to the loss of the operating shutdown cooling system, subsequent random failures and unavailabilities complete the loss of core cooling and injection. Without a means to keep the core cool, the vessel inventory is lost via boiling and core damage ensues.

In the Level 1 screening analysis both the emergency core cooling system (ECCS) and Makeup (i.e., CRD and CDS) were assumed to be unavailable or unable, due to some postulated failure, to prevent core damage. Thus, only the firewater system (FW) and the standby service water (SSW) cross-tie were considered as potential injection systems.

In POS 6 the suppression pool can be either at its normal level, partially drained, or empty. Furthermore, the suppression pool makeup system (SPMU) is not available. Because a supply of water to the SP is not available, ECCS systems that draw water from the SP could not be used in a continuous mode and, therefore, it was assumed in the Level 1 analysis that these systems were not available to cool the core. Because the containment spray system is one mode of the residual heat removal system (i.e., part of ECCS) and draws water from the SP, it is also unavailable during these postulated accidents.

The CRD system has insufficient capacity to prevent the core inventory from boiling and, therefore, was not considered as a means to cool the core in the Level 1 screening study. (It should be noted, however, that if this system was used, the energy removed from the core via steaming would be sufficient to prevent core damage.) While CDS has more than enough capacity to cool the core, its unavailability due to random failures and maintenance precludes its use as a means to cool the core.

A general description of the core damage sequences for each class of initiators is presented below.

LOSP Transients

The LOSP initiating event leads directly to the loss of the alternate decay heat removal system (ADHR). Subsequent random failures lead to the complete loss of shutdown cooling (SDC), makeup, the standby service water and the firewater system. With ECCS unavailable in this POS, as a result of support system failures, the accident proceeds to core damage because of the lack of core cooling.

Loss of Support System Transients

In these sequences the initiating event leads directly to the loss of ADHR, makeup, and the firewater system. Subsequent random failures lead to the complete loss of SDC and the SSW system.

Decay Heat Removal Challenges

In these sequences the initiating event leads to the loss of the operating shutdown cooling system. In some of these sequences this system is recovered. However, subsequent random failures lead to the complete loss of SDC, the firewater system, and SSW.

LOCAs That Can Be Isolated

In these sequences the isolation of the LOCA also isolates the SDC systems. Subsequent random failures lead to the loss of both the firewater system and the standby service water cross-tie system.

Table 2.3-1 Grand Gulf LP&S POS 6 Initiating Events

Initiating Event Group	Initiating Event Nomenclature	Description
LOSP	T1	Loss of Offsite Power (LOSP) Transient
Loss of Support System Transient	T5B	Loss of all TBCW
	T5C	Loss of all PSW (includes Radial Well)
	TIA	Loss of all Instrument Air
Decay Heat Removal Challenge	E1B	Isolation of SDC Loop B only
	E2B	Loss of SDC Loop B only
	E1D	Isolation of ADHRS
	E2D	Loss of ADHRS only
	E1T	Isolation of SDC Common Suction Line
	E2T	Loss of SDC Common Suction Line
	E1V	Isolation of Common Suction Line for ADHRS
	E2V	Loss of Common Suction Line for ADHRS
Isolated LOCAs	H1	Diversion to Suppression Pool via RHR
	J2	LOCA in Connected System (RHR)

2.3.2 Plant Damage State Description

The Level 1 sequences were divided into two plant damage state (PDS) groups: LOSP and nonLOSP. This distinction is made because of the effect that the LOSP has on injection recovery and containment closure. In the analysis of the nonLOSP PDS it is assumed that if injection is not recovered prior to core damage, it will not be recovered during core damage. The reason for this assumption is that there is a considerable amount of time from the initiating event to core damage for the operators to align and use injection systems to cool the core. If this has not been done by the time of core damage, there is no reason to believe that they will recover core cooling during core damage. Recovery of injection is considered in the LOSP PDS. In these sequences offsite power is unavailable and, therefore, non-emergency systems are unavailable to provide injection to the core. Thus, for the LOSP PDS it is assumed that if offsite power is recovered, injection can be recovered. The availability of ac power also affects the likelihood that the containment is closed prior to core damage. The crane that is used to position the equipment hatch is powered with offsite ac power and, therefore, without offsite ac power the containment cannot be closed. If offsite power is available during the accident, closure of the containment prior to core damage is addressed in the event tree analysis. The key attributes associated with these two PDSs are presented in Table 2.3-2.

Table 2.3-2 Grand Gulf LP&S POS 6 Plant Damage State Attributes

PDS Attributes	Plant Damage States (PDS)	
	LOSP	nonLOSP
Offsite Power	Not Available	Available
Vessel Head	Off	Off
Containment Integrity	Open	Open
Drywell Integrity	Open	Open
Suppression Pool Makeup	Not Available	Not Available
Containment Sprays	Not Available	Not Available
Containment Closure Possible?	No	Yes
Injection Recovery Possible?	Yes	No

From Table 2.3-2 it can be seen that the main differences between the LOSP and nonLOSP PDSs are 1) the containment can be closed only in the nonLOSP PDS and 2) injection can be recovered only in the LOSP PDS. Because sequence frequencies are unavailable from the Level 1 screening analysis, the relative likelihood of the two PDSs is unavailable. The remaining analysis that is presented in this report is conditional on the occurrence of these PDSs.

2.4 Event Tree Analysis

A simplified APET was used in this analysis to delineate and quantify the likelihood of the possible accident progression paths. The selection of events to include in the accident progression analysis was based on (1) insights gained from the NUREG-1150 full power PRAs [1,9], (2) results from MELCOR calculations specifically performed for this analysis, and (3) the plant configuration during POS 6. Events deemed important for inclusion in the APET were events that related to containment performance and the estimation of the radionuclide release.

The APET addresses three general time regimes: prior to core damage, during core damage, and following vessel failure. In the first time regime the issue of containment closure is addressed. Injection recovery, core damage arrest, in-vessel steam explosions and early containment failure are all addressed in the second time regime. The characteristics of the interaction between the core debris release from the vessel and the reactor pedestal are addressed in the last time regime.

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The times associated with these time regimes are based on results from a series of MELCOR calculations that were performed to support this analysis. The timing of key events in the accident progression analysis is presented in Table 2.4-1.

Table 2.4-1 Accident Progression Timing

Calculation	Timing of Key Events from Initiation of Accident (hours)				
	Time to TAF	Core Damage	Vessel Failure	Aux. Bldg Failure	Contain. failure
PRA MODEL INPUT					
PRA Model: Containment Open	13.0	18.3	25.4	21.1	Cnt Open(2)
PRA Model: Containment Fails	13.0	19.4	28.6	No Fail. (3)	30.
MELCOR RESULTS					
Base Case (BC)-No Aux Bldg	12.7	18.3	25.4	(1)	(2)
BC w/ Small Aux. Bldg	13.0	18.8	24.5	21.6	(2)
BC w/ Big Aux. Bldg	13.0	18.8	28.6	28.6	(2)
BC w/ Containment Closed	13.6	19.4	28.6	(1)	22 - 80
BC initiated 15 days after SD	19.7	28.3	39.8	(1)	(2)

Notes:
1. Auxiliary building model not included
2. Containment is open during the accident
3. Containment failure bypasses the auxiliary building
4. MELCOR POS 6 BC Calculation:
   - Accident Initiated 4 days after shutdown
   - Containment is open (i.e. equipment hatch and both personnel locks)
   - Injection, shutdown cooling, and containment sprays are all unavailable
5. Core damage is defined as the first gap release
6. TAF = Collapsed water level at the top of the active fuel

In this table both the times estimated with MELCOR and the times assumed in this PRA are presented. From this table it is apparent that the timings of these accidents are quite different from accidents initiated at full power. For example, it takes approximately 18 hours to progress from the initiation of the accident to the onset of core damage. In comparison, a fast station blackout initiated from full power progresses to a similar point in approximately 1 hour. Another notable entry in this table is the predicted time of auxiliary building failure for cases with the containment open. The building is predicted to overpressurize and fail from the accumulation of steam and noncondensibles during core damage. The exact timing of building failure depends on the volume assumed for the auxiliary building (i.e., various rooms in the building can isolated) and the building failure pressure. For this abridged study, the auxiliary building is estimated to fail approximately half way through the core damage process.

Nine events are used to characterize the accident progression. A graphical depiction of the APET is presented in Figure 2.4-1. The first nine paths are associated with the LOSP PDS and the remaining 7 paths (i.e., paths 10 through 16) are associated with the nonLOSP PDS. The mean probability for each path is also presented in this figure. The path probabilities for each PDS sum to 1.0. The nine events and a brief description of each event are presented below.

1. **Is the containment closed prior to core damage?**

   The containment equipment hatch has been removed prior to entry into POS 6. For the LOSP PDS the lack of offsite ac power precludes containment closure prior to core damage. However, for the nonLOSP PDS it is possible that the plant personnel will close the containment after the initiation of the accident but prior to core damage. The containment can be closed if the operators recognize that a problem exists early in the accident and decide that containment closure would be prudent. Because it takes between 8 to 12 hours to completely close the hatch, it is necessary that the operators begin the closure tasks within the first few hours of the accident. The equipment hatch is a pressure seating hatch which requires the personnel closing the hatch to be in the containment. Thus, the environment in the containment during the boiloff is an important parameter that will affect the personnel's ability to close the containment. MELCOR calculations performed for this analysis indicate that the temperatures in the containment during this phase of the accident will be high (i.e., range from 100 to 140 degrees F) but not so high that it would preclude the personnel from carrying out their tasks. It was also assumed that the radiological environment in the containment will not preclude the closure tasks from being performed. These assumption will have to be verified in future analysis. In this analysis it was assumed that the containment was habitable up until the time of core uncovery (i.e., approximately 13 hours).

2. **If the containment is closed prior to core damage, does it fail prior to vessel failure?**

   The Grand Gulf plant utilizes a Mark III containment to house its BWR-6 reactor. The containment has a volume of 1.6 million cubic feet and a design pressure of 15 psig. The mean estimated failure pressure is 56 psig [9]. Since the containment has a relatively low failure pressure, the pressure rise from the accumulation of steam and noncondensibles can pose a threat to the containment integrity. Actions must also be taken to prevent the combustion of large quantities of hydrogen. Containment venting was not considered in this analysis as a means to control pressure because venting would still result in an open containment. In POS 6 the suppression pool is bypassed and, therefore, the steam and noncondensibles are released directly into the containment atmosphere. Furthermore, the containment sprays are not available. Thus, the containment will pressurize during the core damage process. The peak pressure during this phase of the accident depends on the steam generation rate, the condensation rate in the containment, and the presence and magnitude of hydrogen burns. MELCOR calculations indicate that the containment pressure can exceed the lower range of the containment failure pressure distribution if a burst of steam occurs at the time of vessel failure or if discrete hydrogen burns (not diffusion flames) occur during the core damage phase of the accident. Because steam and hot hydrogen are released directly into the containment in this POS, the effectiveness of the HIS to control the accumulation of hydrogen is uncertain. Thus, it is possible that for some accident scenarios the containment will fail early in the accident. If the containment does not fail early, calculations indicate that it will take several days to reach the mean estimated failure pressure of 56 psig. Therefore, it was assumed that if the containment does not fail early, it will not fail in the time frame of this analysis.

3. **If the containment fails, is the failure in the form of a leak or rupture?**

   The failure size will determine how fast the radionuclides are released from the containment and the amount of radionuclides deposited within the containment.

4. **Is the auxiliary building bypassed?**

   This question distinguishes the accidents in which the releases pass through the auxiliary building from those accidents which result in a release from the containment directly into the environment. The release path is important because it will affect the amount of mitigation that the release experiences before entering the environment. Accidents in which the containment equipment hatch is off will result in a release that passes through the auxiliary building; accidents in which the containment fails bypass the auxiliary building. Based on previous structural analysis of the Grand Gulf containment, it was concluded that the most likely location for failure is the region near the junction of the dome and the cylindrical wall [9]. A failure in this location will result in a release to the enclosure building that surrounds the containment dome. The enclosure building has virtually no pressure retaining capability and is essentially isolated from the auxiliary building. Therefore, it is assumed that following containment failure, the release goes directly from the containment into the environment. The retention in the containment will be fairly small in this case because the containment fails early in the accident. The result will be essentially an unmitigated release. If the containment is open to the auxiliary building, the majority of the radionuclides will quickly enter the auxiliary building and the retention in the containment will be small. For these accidents, the only significant mitigation feature will be the auxiliary building which acts as a large holdup volume allowing time for natural processes to remove radionuclides from the building atmosphere before being released into the environment. The auxiliary building is predicted to overpressurize and fail from the accumulation of steam and noncondensibles during core damage. The exact timing of building failure depends on the volume of the auxiliary building that will pressurize (i.e., some rooms within the building can be isolated and therefore will not pressurize) and the building failure pressure. For this abridged study, the auxiliary building is estimated to fail approximately halfway through the core damage process.

5. **Is injection recovered prior to vessel failure?**

This question is used to identify those accidents in which injection is restored to the vessel during the core damage process. The recovery of injection allows for the possibility that the core damage process will be arrested in the vessel (i.e., prevent vessel failure). Injection can only be recovered for the LOSP PDS. The probability that injection is recovered is based on the probability that offsite ac power is recovered during core damage.

6. **If injection is recovered, when is it recovered?**

The timing of injection recovery during core damage affects the likelihood that the core damage process will be arrested before the vessel fails. For this analysis, the in-vessel phase of the accident (i.e., core damage) has been divided into three time regimes: very early, early, and late. The very early time regime ranges from the initiation of core damage to the onset of autocatalytic oxidation. If injection is recovered during this phase of the accident the core damage process will be arrested in the vessel and the releases will be limited to the inventory in the gap. The early time regime ranges from onset of autocatalytic oxidation to 30% core damage. Based on extrapolation of analysis performed in NUREG-1150, if injection is recovered before 30% of the core has been damaged, it is very likely that the core damage process can be arrested. Because MELCOR calculations indicate that core damage progresses rapidly from 30% to full core damage, the late time regime is defined as 30% core damage to vessel failure. Recovery of injection during this phase of the accident will not prevent vessel failure. The time windows for each of these time regimes is based on results from MELCOR calculations. The possibility of the reactor going critical following the restoration of injection was not addressed in this abridged analysis.

7. **Does an in-vessel steam explosion occur during core damage?**

In-vessel steam explosions are treated in a very limited fashion in this abridged analysis. A primary motivation for including this question in the APET is to highlight the fact that in-vessel steam explosions are possible. The effect of the steam explosion on the accident progression can be quite different from in-vessel steam explosions that occur at full power because the steam and radionuclides that are generated during this event are released directly into the containment atmosphere. In this analysis the treatment of in-vessel steam explosions was limited to the estimation of the source term that is associated with the debris that participates in the steam explosions. Neither the pressure loading from in-vessel steam explosions nor the relocation of intact fuel from the steam explosion was addressed in this study. Both issues were beyond the scope of this abridged study. Ex-vessel steam explosions were not considered in this analysis because the pedestal cavity below the vessel will be essentially dry at the time of vessel failure.

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8. **Is the core damage process arrested in the vessel?**

This question addresses the co[illegible]ity of the core debris following injection recovery. If the core damage process is arre[illegible] before the vessel fails, the core debris will remain in the vessel and core-concrete interactions (CCI) will be prevented. Because only a portion of the core is damaged and CCI is prevented, the source term associated with recovered accidents is typically less than the source term associated with full core damage accidents. If injection is not restored during core damage, the accident always progresses to vessel failure and the core debris relocates to the pedestal cavity below the vessel. The likelihood that the core damage process is arrested before vessel failure depends on when injection is restored during the core damage process (see question 6). If injection is restored during either the very early or early time regimes, analysis indicates that it is very likely the core damage process will be arrested. If, on the other hand, injection is not restored until the late time regime, it is very likely that the vessel will fail and the core debris will relocate to the pedestal cavity.

9. **Do core-concrete interactions occur following vessel failure?**

Core-concrete interactions consist of the thermal and chemical interactions between the core debris and the concrete pedestal. During this process the concrete is eroded and gases and radionuclides are released from the core/concrete mixture. For the accidents analyzed in this study, the vessel will fail and the core debris will enter the cavity if 1) injection is not restored to vessel during core damage or 2) injection is restored during the late time regime. The presence of water can affect CCI in two different ways. First, water can quench the debris and prevent CCI. Second, if the debris is not quenched, the overlying pool of water will retain some of the radionuclides released during CCI and thus, tend to mitigate the release. Thus, for the accidents in which injection is restored but the vessel still fails, there is some probability that the core debris will be quenched and CCI will be prevented. The probability of this occurring is based on information from the NUREG-1150 study [9]. If injection is not restored during core damage, CCI will always proceed in a dry cavity.

From inspection of Figure 2.4-1 it can be seen that there are several important differences between the LOSP PDS and the nonLOSP PDS. In the LOSP PDS injection can be recovered allowing for the possibility to arrest the core damage process in the vessel. If the vessel does fail, it is still possible to quench the core debris in the cavity (i.e., no CCI). Thus, in many of the LOSP accidents the ex-vessel radionuclide release is prevented. The containment, however, cannot be closed in the LOSP PDS and, therefore, the releases always pass into the auxiliary building and then out into the environment. In the nonLOSP PDS, the containment can be closed, however, injection cannot be recovered. Thus, all of the nonLOSP accidents identified in the APET progress to full core damage, vessel failure, and involve CCI. In some of the scenarios the containment is closed. However, because containment cooling (i.e., containment sprays) is unavailable and the suppression pool is bypassed, even if the containment is closed it is possible that it will fail early in the accident from pressure transients associated with events accompanying vessel failure and hydrogen combustion. Based on information from NUREG-1150, it is expected that the containment will fail above the auxiliary building roof. Thus, the releases from the containment will enter the environment without first going through the auxiliary building. Because so many of the mitigative features of the plant are bypassed in this POS (e.g., suppression pool, containment sprays, containment), the auxiliary building plays an important role in reducing the amount of radionuclide material that is released into the environment. Thus, for the nonLOSP accidents there are two extremes: 1) if the containment is closed and remains intact, the releases to the environment are expected to be very small and 2) if the containment fails, the releases to the environment are expected to be quite large because all of the accidents involve full core damage and CCI and the releases bypass the auxiliary building. The scenarios in which the containment is not closed are very similar to the LOSP accidents in which injection is not recovered.

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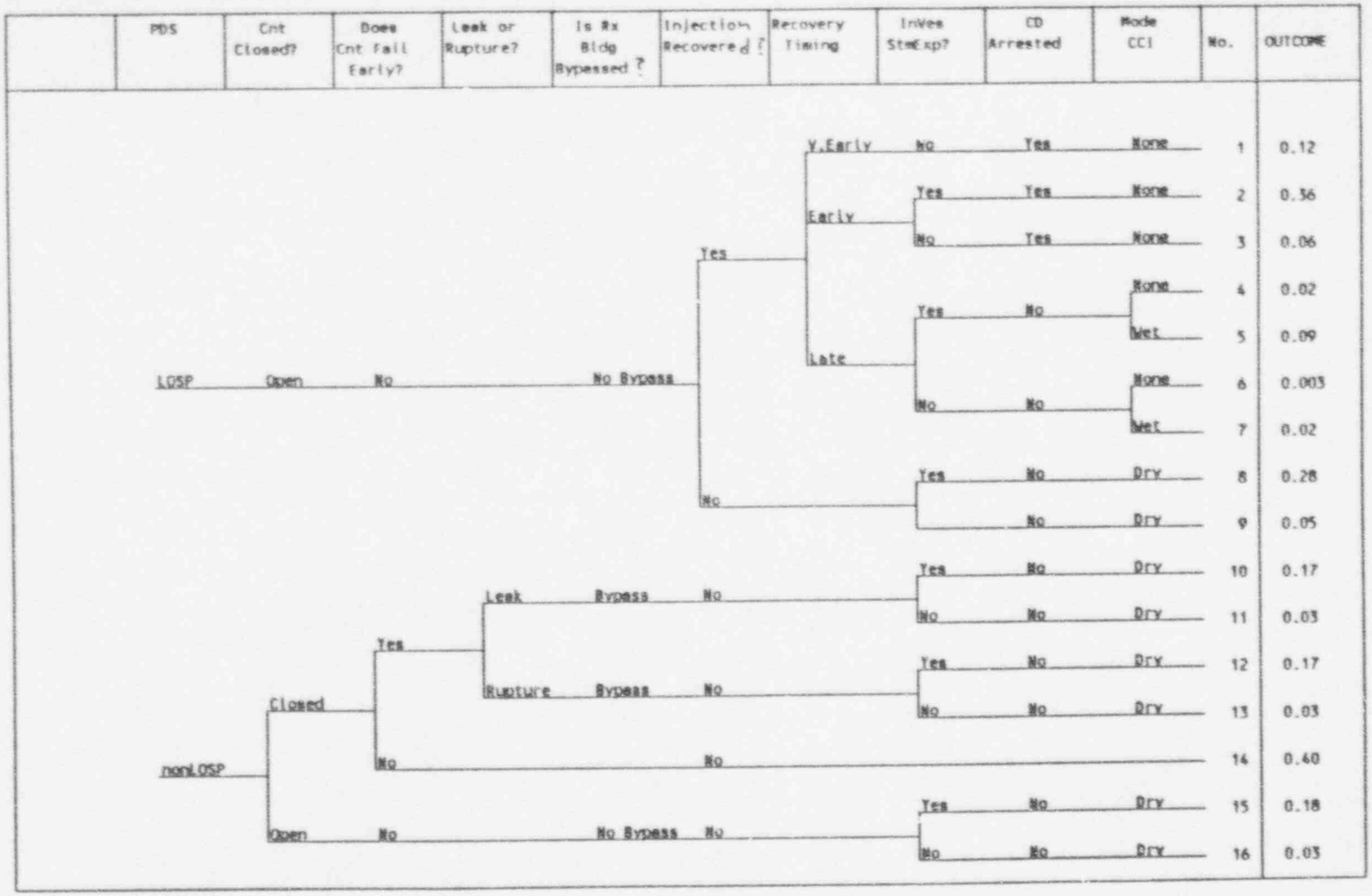

Figure 2.4-1. Grand Gulf POS 6 abridged accident progression event tree (APET)

3.0 SOURCE TERM ANALYSIS

3.1 Approach

A source term is estimated for each accident progression path identified in the APET (see Figure 2.4-1). The simple parametric source term approach that was used in NUREG-1150 to estimate source terms is used in this study. The parametric source approach is used because 1) information from a wide variety of sources can be used in the model, 2) it is easily incorporated into uncertainty analysis, and 3) thousands of source terms can be estimated with this model in a very efficient manner. The parametric source term code GGSOR that was developed in NUREG-1150 [9] was modified for this analysis. The modified parametric code is called GGLPSOR. Modifications were made to the code to incorporate the unique plant configuration associated with accidents initiated in POS 6. Wherever possible, data from NUREG-1150 was used to quantify the model. Results from MELCOR were compared with both the input distributions and the final source terms to verify that distributions developed for full power accidents could be applied to shutdown accidents.

A limited uncertainty analysis was performed in this section of the analysis. For each accident progression path, the model was repeatedly exercised with different combinations of selected input variables. The distributions for these input variables were obtained, when applicable, from NUREG-1150.

3.2 Description of Parametric Model

The parametric source term model GGSOR that was developed in NUREG-1150 was modified to account for unique features of POS 6 that have a strong impact on the source term. In POS 6 both the drywell head and the vessel head have been removed and the steam lines have been plugged. Thus, during the core damage process radionuclides released from the core debris will bypass the suppression pool and directly enter the containment. Furthermore, because most of the internal structures above the core (e.g., steam dryers and separators) have been removed and the steam lines are plugged, there is very little deposition of radionuclides in the vessel. Thus, the mitigative features of both the vessel and the suppression pool, which are present in many full power accident scenarios, are absent in this POS. For scenarios in which the containment hatch is open, the residence time of the radionuclides in the containment atmosphere is fairly short and, thus, there will be limited deposition (i.e., from gravitational settling) of radionuclides in the containment. In this POS the drywell is open to the containment (i.e., the drywell hatch is open) and, therefore, an ex-vessel release will also bypass the suppression pool. For these accidents, the containment sprays are not available and cannot be used to scrub the releases. Thus, the mitigative features of the vessel, suppression pool, containment sprays, and possibly the containment, are bypassed or unavailable.

For scenarios with the containment open, the only major mitigative feature of the plant is the auxiliary building. The auxiliary building encompasses a very large volume and, therefore, acts as a hold up volume for the radionuclides which allows time for the radionuclides to deposit on surfaces within the building. The auxiliary building can play an important role in POS 6 because so many of the other mitigative features of the plant are absent and the characteristics of the radionuclide transport to the auxiliary building are different from the transport associated with full power accidents. In full power accidents the containment pressurizes to the ultimate failure pressure and then blows down into the auxiliary/reactor building (i.e., Peach Bottom analysis in the NUREG-1150 study). Following containment failure the auxiliary building rapidly pressurizes and fails (the failure pressure of the auxiliary building is only a few psi). Thus, the releases are swept through the auxiliary building fairly rapidly. In the POS 6 accident scenarios the steam and radionuclides are released to the auxiliary building much more slowly allowing more time for condensation and deposition. The scenarios that involve containment closure followed by containment failure will not benefit from the auxiliary building because the containment failure location is assumed to be above the roof of the auxiliary building [9]. Thus, the releases will bypass the auxiliary building essentially resulting in an unmitigated release.

Neither the normal ventilation system nor the standby gas treatment system (SBGT) were modeled in this analysis. The filters and charcoal beds in the SBGT system could act to mitigate the release or at least delay the release of radionuclides. Before credit can be given to this system, the capacity of the system and the performance of the filters under severe accident conditions will have to be addressed. The analysis of this system was beyond the scope of this study.

3.3 Results

A source term is estimated for each path through the APET. In addition, because an uncertainty analysis was performed, a distribution of source terms is available for each path. For the sake of brevity, only the mean source terms, expressed as fractions of the core inventory, that enter the environment are presented in Table 3.3-1. When reviewing this table, it must be remembered that the initial inventory of radionuclides four days after shutdown is different from the inventory typical of full power accidents.

Inspection of Table 3.3-1 confirms that many of the releases are essentially unmitigated and, therefore, are quite large. Table 3.3-1 also highlights some of the differences between the various accident scenarios (i.e., paths). Paths 1 through 3 correspond to accidents in which injection is recovered early in the accident and the core damage process is arrested in the vessel. Thus, because only a portion of the core is damaged and there are no ex-vessel releases (i.e., no CCI), the source terms associated with these accidents are relatively small compared to the other source terms presented in this table. The notable exception is Path 14 which corresponds to the scenario in which the containment is closed prior to core damage and remains intact throughout the accident. Because the containment remains intact, only nominal leakage occurs and the resulting source term is quite small. Paths 4 through 9, on the other hand, correspond to full core damage accidents that have the containment open to the auxiliary building. The source terms associated with Paths 4 and 6 tend to be lower than the other full core damage source terms because the core debris is quenched in the pedestal cavity and, therefore, there are

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no releases associated with CCI. This difference is fairly minor, however, and the fact still remains that these are large source terms. Paths 10 through 13 are nonLOSP accidents in which the containment fails around the time of vessel failure. All of these accidents progress to full core damage and CCI. The containment fails via a leak in Paths 10 and 11; the containment ruptures in Paths 12 and 13. In all four of these scenarios the containment fails directly to the environment (i.e., the auxiliary building is bypassed). The source terms associated with the leak failure mode are similar to the source terms when the release passes through the auxiliary building. In the leakage cases, the radionuclides are held up in the containment for a period of time thus allowing a fraction of the radionuclides to settle out of the containment atmosphere. For the rupture cases, however, the containment quickly depressurizes following containment failure and considerably less deposition occurs. Thus, the source terms associated with the rupture cases are quite large. Paths 15 and 16 correspond to the nonLOSP cases where the containment is not closed prior to core damage and the radionuclides pass through the auxiliary building. These source terms are essentially the same as the LOSP full core damage source terms.

Table 3.3-1 Mean Source Terms for Accident Progression Paths (Total Release)

Path	Radionuclide Release Classes									Timing of Release (hr.s)			
	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba	TW	T1	DT1	DT2
	LOSP PDS												
1	0.015	0.002	5.9E-3	1.2E-5	0.0	0.0	0.0	0.0	1.2E-7	16.3	21.1	24.0	0.0
2	0.072	0.012	0.011	6.3E-3	2.1E-3	3.3E-4	1.4E-4	6.7E-4	2.2E-3	16.3	21.1	4.3	0.0
3	0.072	0.012	0.011	6.3E-3	2.1E-3	3.3E-4	1.4E-4	6.7E-4	2.2E-3	16.3	21.1	4.3	0.0
4	0.79	0.17	0.15	0.085	0.027	0.012	3.0E-3	8.7E-3	0.033	16.3	21.1	4.3	10.0
5	1.0	0.25	0.19	0.11	0.042	0.012	4.0E-3	0.011	0.047	16.3	21.1	4.3	10.0
6	0.74	0.15	0.13	0.075	0.022	4.9E-3	1.5E-3	7.1E-3	0.026	16.3	21.1	4.3	10.0
7	1.0	0.25	0.18	0.11	0.041	4.9E-3	2.7E-3	9.6E-3	0.042	16.3	21.1	4.3	10.0
8	1.0	0.25	0.25	0.16	0.08	0.012	6.9E-3	0.012	0.084	16.3	21.1	4.3	10.0
9	1.0	0.25	0.2[illegible]	0.17	0.088	5.4E-3	6.3E-3	0.011	0.089	16.3	21.1	4.3	10.0
	nonLOSP PDS												
10	1.0	0.27	0.27	0.18	0.094	0.013	6.8E-3	0.011	0.083	17.4	30.0	2.0	10.0
11	1.0	0.28	0.28	9.19	0.10	6.[illegible]E-3	6.2E-3	0.011	0.086	17.4	30.0	2.0	10.0
12	1.0	0.62	0.63	0.40	0.22	0.029	0.016	0.027	0.19	17.4	30.0	0.05	10.0
13	1.0	0.62	0.63	0.43	0.24	0.013	0.014	0.025	0.20	17.4	30.0	0.05	10.0
14	5.0E-3	4.1E-7	4.1E-7	2.9E-7	1.4E-7	9.4E-9	9.7E-9	1.9E-8	1.4E-7	16.3	21.1	4.3	10.0
15	1.0	0.25	0.25	0.16	0.08	0.012	6.9E-3	0.012	8.4E-2	16.3	21.1	4.3	10.0
16	1.0	0.25	0.25	0.17	0.088	5.4E-3	6.3E-3	0.011	0.089	16.3	21.1	4.3	10.0

Notes:

1. TW = Warning Time
2. T1 = Timing of first release
3. DT1 = Duration of first release
4. DT2 = Duration of second release (start immediately after first release ends)

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4.0 CONSEQUENCE ANALYSIS

The consequences of a severe accident during POS 6 were calculated as part of the abridged study. As is typically done, the offsite consequences were estimated. The onsite doses were also estimated, which is not typically done.

An important difference between this analysis and those previously performed for full power accidents is that the radionuclides in the fuel have had at least four days to decay resulting in a different inventory than that present at shutdown. ORIGEN2 [10] was used to calculate the inventory in three different fuel assemblies, one which had been irradiated for three fuel cycles, one which had been irradiated for two fuel cycles, and one which had been irradiated for one fuel cycle. All fuel assemblies were then allowed to decay for four days. Based on information from plant personnel, a fuel cycle consisted of 540 days of irradiation and 55 days of decay. The inventory for the whole core four days after shutdown was then summed. This inventory, which was reduced to include only the sixty radionuclides currently available in the MACCS code [5,6,7], was then used as the basis for both the onsite and offsite consequence calculations. This inventory, which does not include short-lived radionuclides, is appropriate for both the onsite and offsite analyses since the reactor has been in shutdown for at least four days at the beginning of the accident thus allowing decay of the short-lived radionuclides.

The following sections detail the methodology and results for the onsite consequences, both in the buildings and in the parking lot, and the offsite consequences.

4.1 Onsite Consequences

Onsite consequences have seldom been considered in the analysis of severe accidents at nuclear power plants. During shutdown there will be hundreds of onsite personnel and, thus, onsite consequences could be large. For this reason a method for estimating the potential doses to onsite personnel had to be developed as part of this study. The primary simplifying assumption of the analysis was that radioactive decay was neglected during the exposure time. This assumption is justified by the fact that the accident under analysis typically occurs no earlier than four days after shutdown by which time the decay heat curve is fairly flat. Other assumptions were employed in the two aspects of the onsite consequences: (1) in building doses and (2) parking lot doses. The method, assumptions, and results of the analyses are discussed in the following two sections.

4.1.1 Building Doses

The onsite consequences for POS 6 were estimated based on the source terms to both the containment and the auxiliary building that were determined with the parametric source term code, GGLPSOR. However, since GGLPSOR calculates integral releases, the time dependence of the two release segments of the source terms was determined from MELCOR calculations. Three different sets of residence times (i.e., estimated time airborne material spends in the building) were used based on the status of the containment. The first set of residence times was used if the containment was open to the auxiliary building at the time of the accident. The residence times through both buildings were based on a MELCOR calculation modeling this scenario. The residence time of the radioactive material in each building was directly proportional to the volume of that building. The second set of residence times was used if the containment ruptured directly to the environment. For this case, the same residence times were used as in the previous scenario, however, the residence time in the auxiliary building was set to zero. In other words, the amount of time the material spent in the containment was the same for both of these scenarios, but in the latter scenario the material did not pass through the auxiliary building. The third set of residence times was used if the containment leaked directly to the environment. In this case, the residence time for the first release was increased by two hours, and again the residence time in the auxiliary building was set to zero. The residence times used under the various conditions are summarized in Table 4.1.1-1.

Table 4.1.1-1. Residence times through the containment and auxiliary building for Grand Gulf POS 6.

Accident Progression Path Number	Containment Residence Time: First Segment (hours)	Containment Residence Time: Second Segment (min)	Auxiliary Building Residence Time: First Segment (hours)	Auxiliary Building Residence Time: Second Segment (hours)
	LOSP PDS			
Paths 1-9	3.4	47	6.1	1.4
	nonLOSP PDS			
Path 10	4.1	47	0.0	0.0
Path 11	4.1	47	0.0	0.0
Path 12	3.4	47	0.0	0.0
Path 13	3.4	47	0.0	0.0
Path 14[1]	NC	NC	NC	NC
Path 15	3.4	47	6.1	1.4
Path 16	3.4	47	6.1	1.4

[1] Building doses were not calculated since the containment is not open.

To estimate the doses in the buildings, the average release fraction of each chemical group was determined for each building. The integrated concentration of each radionuclide in the buildings was then based on the average release fraction of its chemical group and the amount of time spent in that building. Using the integrated concentration for each radionuclide, the immersion and 50 year committed inhalation dose were calculated over the entire exposure time. In addition, the immersion and 50 year committed inhalation dose were calculated for the first 30 minutes of exposure. These doses should be viewed with caution since the integrated concentration in the building was based on an average concentration in the building and therefore the time dependence of the dose is not well represented. The final result estimated in the buildings was a dose rate. These results should also be viewed with caution since they are also based on average concentrations in the building. In addition, the dose rates were calculated by dividing the total dose during a release segment by the transit time through the building. This results in a conservative estimate of the inhalation dose rate. The mean dose due to the entire release, the first 30 minutes of exposure, and the mean dose rates during the first and second release segments in the containment are shown in Table 4.1.1-2 for each of the paths through the APET. Similar estimates are shown in Table 4.1.1-3 for the auxiliary building.

Table 4.1.1-2. Grand Gulf POS 6 mean containment doses and dose rates.

Accident Progression Path Number	Path Conditional Probability	Consequence Measure			
		Total Dose (rem)	30 minute Dose (rem)	Dose Rate First Segment (rem/hr)	Dose Rate Second Segment (rem/hr)
		LOSP PDS			
Path 1	0.10	1.81E+6	2.69E+5	5.38E+5	0.0
Path 2	0.48	4.27E+7	6.35E+6	1.27E+7	0.0
Path 3	0.08	4.27E+7	6.35E+6	1.27E+7	0.0
Path 4	0.02	5.38E+6	7.95E+7	1.59E+8	0.0
Path 5	0.09	5.69E+8	7.95E+7	1.59E+8	4.04E+7
Path 6	0.003	4.27E+8	6.35E+7	1.27E+8	0.0
Path 7	0.01	4.66E+8	6.35E+7	1.27E+8	5.05E+7
Path 8	0.18	6.16E+8	7.95E+7	1.59E+8	1.01E+8
Path 9	0.03	5.25E+8	6.35E+7	1.27E+8	1.26E+8
		nonLOSP PDS			
Path 10	0.16	5.78E+8	6.35E+7	1.27E+8	7.74E+7
Path 11	0.02	4.88E+8	5.05E+7	1.01E+8	9.68E+7
Path 12	0.16	6.16E+8	7.95E+7	1.59E+8	1.01E+8
Path 13	0.02	5.25E+8	6.35E+7	1.27E+8	1.26E+8
Path 14[1]	0.37	NC	NC	NC	NC
Path 15	0.23	6.16E+8	7.95E+7	1.59E+8	1.01E+8
Path 16	0.04	5.25E+8	6.35E+7	1.27E+8	1.26E+8

[1] Building doses were not calculated since the containment is not open.

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Table 4.1.1-3. Grand Gulf POS 6 mean auxiliary building doses and dose rates.

Accident Progression Path Number	Path Conditional Probability	Consequence Measure			
		Total Dose (rem)	30 minute Dose (rem)	Dose Rate First Segment (rem/hr)	Dose Rate Second Segment (rem/hr)
		LOSP PDS			
Path 1	0.10	9.41E+5	7.70E+4	1.54E+5	0.0
Path 2	0.48	2.13E+7	1.74E+6	3.47E+6	0.0
Path 3	0.08	2.13E+7	1.74E+6	3.47E+6	0.0
Path 4	0.02	2.80E+8	2.28E+7	4.56E+7	0.0
Path 5	0.09	2.98E+8	2.28E+7	4.56E+7	1.31E+7
Path 6	0.003	2.20E+8	1.79E+7	3.59E+7	0.0
Path 7	0.01	2.43E+8	1.79E+7	3.59E+7	1.64E+7
Path 8	0.18	3.23E+8	2.28E+7	4.56E+7	3.06E+7
Path 9	0.03	2.74E+8	1.79E+7	3.59E+7	3.83E+7
		nonLOSP PDS			
Path 10	0.16	0.0	0.0	0.0	0.0
Path 11	0.02	0.0	0.0	0.0	0.0
Path 12	0.16	0.0	0.0	0.0	0.0
Path 13	0.02	0.0	0.0	0.0	0.0
Path 14[1]	0.37	NC	NC	NC	NC
Path 15	0.23	3.23E+8	2.28E+7	4.56E+7	3.06E+7
Path 16	0.04	2.74E+8	1.79E+7	3.59E+7	3.83E+7

[1] Building doses were not calculated since the containment is not open.

To illustrate the uncertainty in the dose rate in the containment and the auxiliary building due to the uncertainty in the source term, the 5th, 50th, and 95th percentile dose rates as well as the mean dose rate for two pathways through the APET are shown in Figure 4.1.1-1. The first of these paths represents a scenario in which injection is recovered very early in the accident, thus arresting core damage. Note that in the recovered accident, CCI does not occur therefore the source term consists of only one segment and only one dose rate was calculated. The second path represents a scenario in which full core damage occurs.

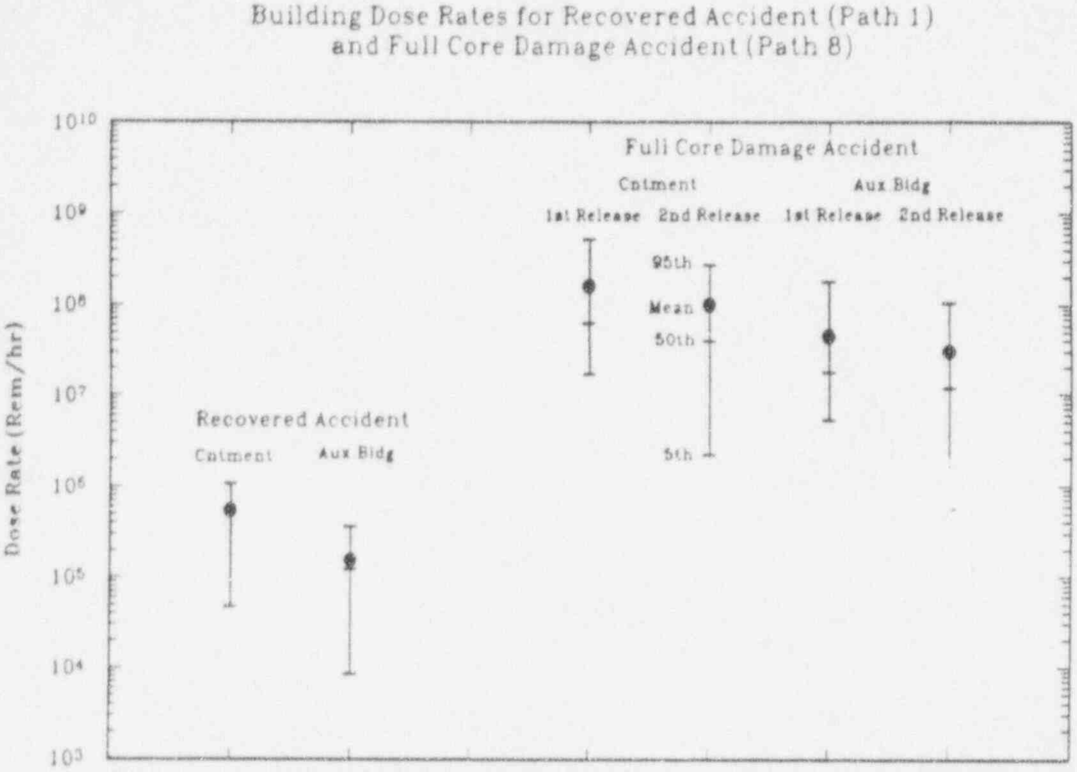

Figure 4.1.1-1. Containment and auxiliary building dose rates for selected paths

4.1.2 Parking Lot Doses

The dose due to immersion and inhalation was also estimated for several distances from the reactor. The source terms were obtained from the parametric source term code, GGLPSOR. In contrast to the building doses, the timing of the source terms was taken directly from GGLPSOR. For comparative purposes, three different wake effect models were used to estimate the relative concentrations downwind of the reactor. These models were developed by Ramsdell [11], Wilson [12], and the NRC [13]. For simplicity, the directional dependence of the weather was ignored and doses were calculated for several distances from the reactor. The weather used in each of the wake effect models was chosen to represent conservative values for the model. In the case of the Ramsdell model the relative concentration is somewhat proportional to the wind speed and the stability class. For this reason the highest wind speed and the corresponding stability class in a year of weather data at Grand Gulf was chosen as input to this model. In addition, the relative concentration is predicted to be somewhat inversely proportional to the area of the building, therefore, the minimum area was utilized. In the case of the Wilson and NRC models the relative concentration is predicted to be inversely proportional to the wind speed.

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Therefore, a wind speed of 1 m/s and a stability class of F (i.e., moderately stable meteorological conditions) were used in these models. Using the integrated air concentrations for each building wake effect model, the dose and dose rate due to immersion and inhalation for the entire source term was determined for each of the unique accident progression paths. As with the building dose rates, the dose rates in the parking lot are very conservative since the inhalation dose rate was determined by dividing the 50 year committed dose by the exposure time. The dose due to 30 minutes of exposure was also estimated. Table 4.1.2-1 contains the mean total dose, 30 minute dose, and dose rates for each segment of the release based on the Ramsdell building wake effect model at 100 meters from the reactor. Similar estimates of the mean doses and dose rates at 100 meters based on the Wilson model is shown in Table 4.1.2-2.

Table 4.1.2-1. Grand Gulf POS 6 mean doses and dose rates at 100 m based on the Ramsdell building wake effect model.

Accident Progression Path Number	Path Conditional Probability	Consequence Measure			
		Total Dose (rem)	30 minute Dose (rem)	Dose Rate First Segment (rem/hr)	Dose Rate Second Segment (rem/hr)
		LOSP PDS			
Path 1	0.10	4.23E+2	8.80	17.6	0.0
Path 2	0.48	9.42E+3	1.09E+3	2.19E+3	0.0
Path 3	0.08	9.42E+3	1.09E+3	2.19E+3	0.0
Path 4	0.02	1.32E+5	1.53E+4	3.06E+4	0.0
Path 5	0.09	1.73E+5	1.53E+4	3.06E+4	4.08E+3
Path 6	0.003	1.04E+5	1.21E+4	2.43E+4	0.0
Path 7	0.01	1.55E+5	1.21E+4	2.43E+4	5.10E+3
Path 8	0.18	2.16E+5	1.53E+4	3.06E+4	8.44E+3
Path 9	0.03	2.10E+5	1.21E+4	2.43E+4	1.05E+4
		nonLOSP PDS			
Path 10	0.16	2.26E+5	3.41E+4	6.83E+4	8.94E+3
Path 11	0.02	2.19E+5	2.68E+4	5.37E+4	1.12E+4
Path 12	0.16	5.24E+5	3.20E+5	6.22E+6	2.13E+4
Path 13	0.02	5.10E+5	2.56E+5	4.88E+6	2.66E+4
Path 14	0.37	0.899	5.25E-2	0.105	4.50E-2
Path 15	0.23	2.16E+5	1.53E+4	3.06E+4	8.44E+3
Path 16	0.04	2.10E+5	1.21E+4	2.43E+4	1.05E+4

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IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

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Table 4.1.2-2. Grand Gulf POS 6 mean doses and dose rates at 100 m based on the Wilson building wake effect model.

Accident Progression Path Number	Path Conditional Probability	Consequence Measure			
		Total Dose (rem)	30 minute Dose (rem)	Dose Rate First Segment (rem/hr)	Dose Rate Second Segment (rem/hr)
		LOSP PDS			
Path 1	0.10	9.48E+3	1.97E+2	3.95E+2	0.0
Path 2	0.48	2.11E+5	2.45E+4	4.91E+4	0.0
Path 3	0.08	2.11E+5	2.45E+4	4.91E+4	0.0
Path 4	0.02	2.95E+6	3.43E+5	6.86E+5	0.0
Path 5	0.09	3.56E+6	3.43E+5	6.86E+5	9.14E+4
Path 6	0.003	2.34E+6	2.72E+5	5.44E+5	0.0
Path 7	0.01	3.48E+6	2.72E+5	5.44E+5	1.14E+5
Path 8	0.18	4.84E+6	3.43E+5	6.86E+5	1.89E+5
Path 9	0.03	4.70E+6	2.72E+5	5.44E+5	2.36E+5
		nonLOSP PDS			
Path 10	0.16	5.06E+6	7.65E+5	1.53E+6	2.00E+5
Path 11	0.02	4.91E+6	6.00E+5	1.20E+6	2.50E+5
Path 12	0.16	1.17E+7	7.16E+6	1.39E+8	4.77E+5
Path 13	0.02	1.14E+7	5.72E+6	1.09E+8	5.96E+5
Path 14	0.37	20.1	1.17	2.34	1.01
Path 15	0.23	4.84E+6	3.43E+5	6.86E+5	1.89E+5
Path 16	0.04	4.70E+6	2.72E+5	5.44E+5	2.36E+5

Figure 4.1.2-1 contains the 5th, 50th, 95th percentile as well as the mean parking lot dose rates for the first release for both the Ramsdell and Wilson/Regulatory Guide models for distances of 10 - 500 meters from the reactor. A similar plot for the second release is shown in Figure 4.1.2-2. The uncertainty in both the building wake effect models and the source term is shown by the wide range of dose rates at each distance.

4.2 Offsite Consequences

The MACCS code [5,6,7] was used to estimate the consequences to the general public. MACCS models the transport and dispersion of plumes of radioactive material released from the plant. As the plumes travel through the atmosphere, material is deposited on the ground. Several of the pathways through which the general population can be exposed are considered. Emergency

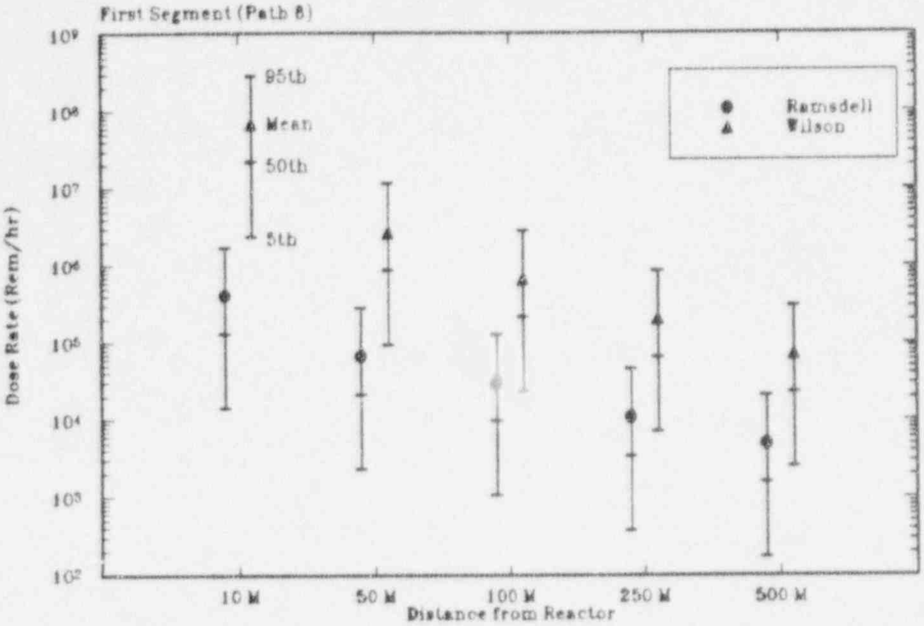

Figure 4.1.2-1. Parking Lot dose rates for Ramsdell and Wilson/Regulatory Guide models for distances from 10 - 500 meters from the reactor: First Release Segment

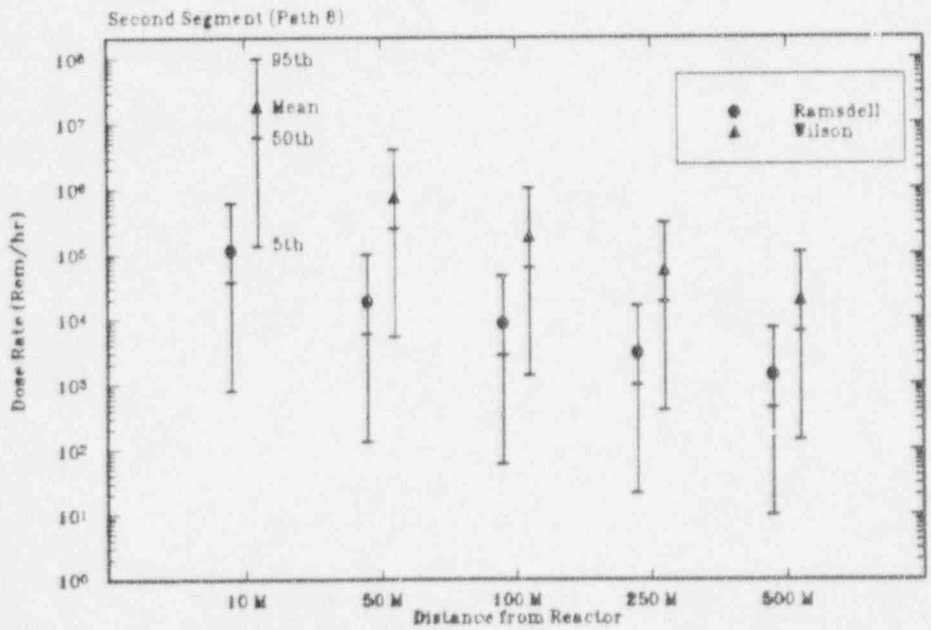

Figure 4.1.2-2. Parking Lot dose rates for Ramsdell and Wilson/Regulatory Guide models for distances from 10 - 500 meters from the reactor: Second Release Segment

response and protective action guides are also considered as means to mitigate the extent of the public exposure.

The input used in this study is identical to that used for Grand Gulf in the NUREG-1150 study [9] with the exception of the core inventory for which the inventory four days after shutdown was used and the source terms which resulted from GGLPSOR. The emergency response assumptions were not changed for this analysis.

Table 4.2-1 contains the estimated mean number of early fatalities, latent cancers, 50 mile population dose, and 1000 mile population dose for the sixteen paths through the APET along with the conditional probability of that path. The mean number of early fatalities ranged from 0 to 3.9 x 10^{-2} while the mean number of latent cancers ranged from 0 to 1940.

Table 4.2-1 Grand Gulf POS 6 Offsite Mean Consequences

Accident Progression Path Number	Path Conditional Probability	Consequence Measure			
		Early Fatalities	Total Latent Cancers	50 mile Population Dose[1]	1000 Mile Pop. Dose[1]
	LOSP PDS				
Path 1[2]	0.10	NC	NC	NC	NC
Path 2	0.48	1.3E-5	102	77,000	591,000
Path 3	0.08	1.3E-5	102	77,000	591,000
Path 4	0.02	4.8E-3	684	330,000	4,010,000
Path 5	0.09	4.8E-3	984	496,000	5,800,000
Path 6	0.003	4.0E-3	588	293,000	3,4[illegible],000
Path 7	0.01	4.0E-3	940	479,000	5,560,000
Path 8	0.18	5.2E-3	1270	652,000	7,480,000
Path 9	0.03	4.7E-3	1260	662,000	7,460,000
	nonLOSP PDS				
Path 10	0.16	8.9E-3	1190	624,000	7,090,000
Path 11	0.02	9.3E-3	1200	640,000	7,130,000
Path 12	0.16	3.7E-2	1920	939,000	11,300,000
Path 13	0.02	3.9E-2	1940	966,000	11,500,000
Path 14[2]	0.37	NC	NC	NC	NC
Path 15	0.23	5.2E-3	1270	652,000	7,480,000
Path 16	0.04	4.7E-3	1260	662,000	7,460,000

Table Notes:
[1] Dose is in Person Rem
[2] Offsite consequences were not evaluated for these paths because the offsite consequences associated with these paths were assessed to be negligible.

5.0 INTEGRATED RESULTS CONDITIONAL ON CORE DAMAGE

In the previous section the consequences associated with individual accident progression paths were presented. In this section the offsite consequences conditional on the occurrence of the LOSP PDS and the nonLOSP PDS are presented and are compared to full power PRA results extracted from the Grand Gulf analysis presented in NUREG-1150. Onsite consequences were not evaluated in NUREG-1150 and, therefore, an analogous comparison is not provided.

The consequences for a given PDS are calculated by taking a weighted average of the consequences for the individual paths. The weighted average is based on the conditional probability of each path. The PDS consequence is the sum of all of the "weighted" path consequences for the given PDS.

The offsite consequence distributions associated with the LOSP and nonLOSP PDSs are presented in Figure 5.1. Because a relatively small LHS sample was used in the evaluation of offsite consequences, the presentation of exact quantiles (i.e., 95^{th}) is inappropriate. Instead of quantiles, the high, low, median, and mean values are presented in this figure. From this figure it can be seen that the consequences associated with the nonLOSP PDS tend to be higher than the consequences associated with the LOSP PDS. This stems from the assumption that injection cannot be recovered in the nonLOSP PDS and, therefore, all of these accidents proceed to full core damage and CCI. Although the probability that the containment is closed during this PDS is significant, the lack of a means to control the containment pressure results in a significant probability of early containment failure. Containment failure bypasses the auxiliary building and results in, essentially, an unmitigated release.

Also presented in Figure 5.1 are the conditional consequences from the Grand Gulf full power PRA. The full power results are for internal events and are "averaged" over all of the accidents analyzed in the study. In addition to the global consequences, the mean consequences associated with a selected full power accident are also presented (i.e., triangle on the full power distribution). This selected accident is a fast station blackout that progresses to full core damage. The containment is ruptured during core damage; the containment sprays are unavailable throughout the accident. Thus, this accident is similar to the accidents analyzed in this abridged study in that many of the mitigative features of the plant (i.e., the containment and sprays) are unavailable. In this full power accident, however, the in-vessel releases are typically scrubbed by the suppression pool. From Figure 5.1 it can be seen that the number of early fatalities associated with POS 6 are very similar to the number of early fatalities associated with full power accidents. This may seem somewhat surprising at first because the inventory of radionuclides important to early fatalities during POS 6 is less than the inventory at full power. However, this difference is compensated by the lack of mitigative features in POS 6. In POS 6 the inventory has been reduced by decay but because of the lack of mitigative features, a significant amount of the radionuclides are released to the environment. In the full power accidents, on the other hand, there is a large inventory, however, mitigative features of the plant limited the size of the release. In full power accidents, for example, a considerable fraction of these radionuclides are retained in the suppression pool. The net effect is that the number of

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early fatalities is roughly the same. The number of latent cancers associated with POS 6 accidents is greater than the number of latent cancers associated with full power accidents. The radionuclides that are important to latent health effects are long lived isotopes and, therefore, four days of decay will not have a significant impact on the radiological potential of the release to cause latent cancer fatalities. Thus, the magnitude of the release is the driving factor for latent cancer fatalities. Because in POS 6 the releases tend to be higher than the full power accidents, the number of latent cancers associated with POS 6 are greater than the number of latent cancers associated with full power accidents. The factors that influence latent cancers also affect the population dose.

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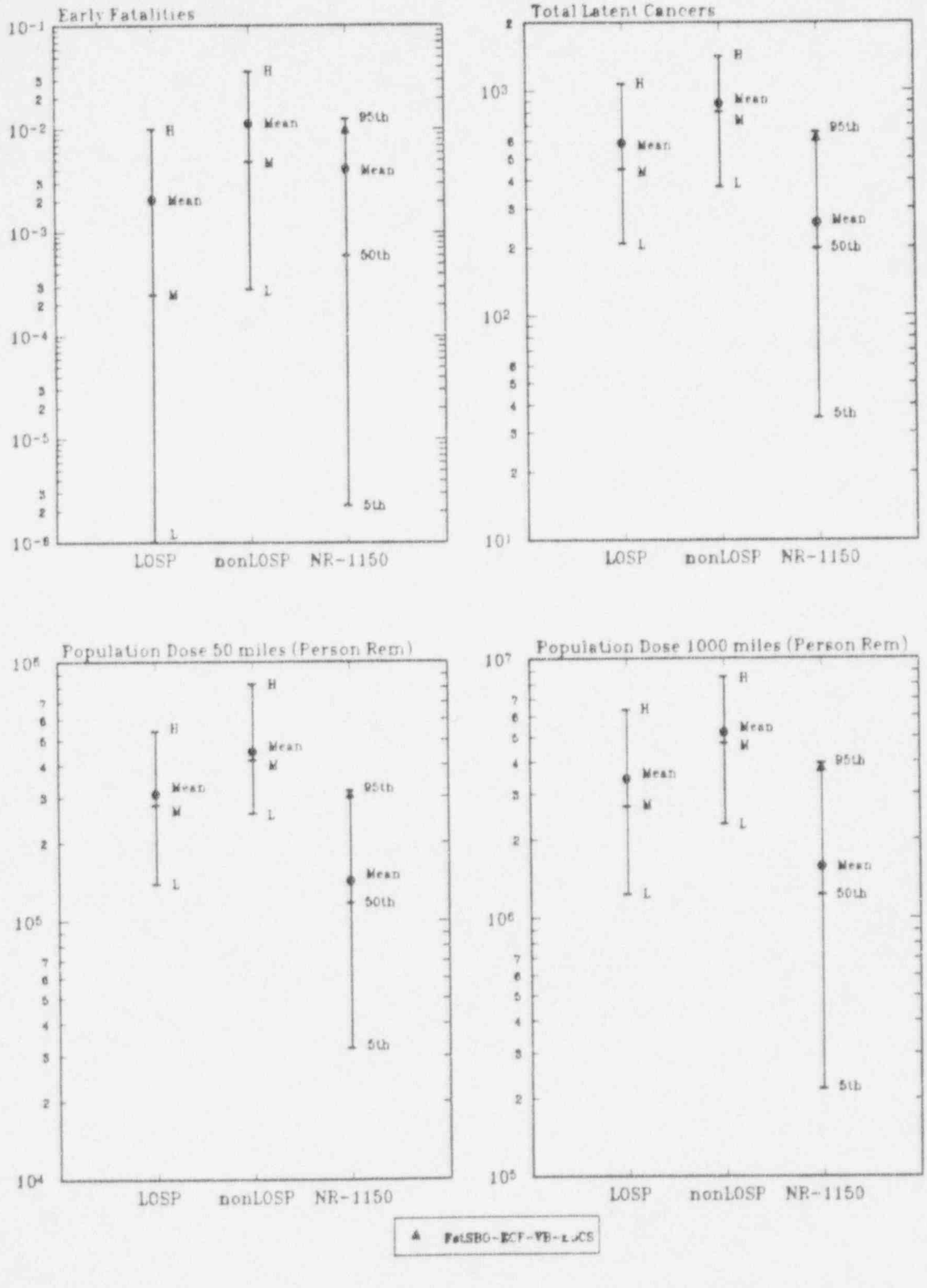

Figure 5-1. Grand Gulf POS 6 offsite consequences for LOSP and nonLOSP PDSs

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6.0 INSIGHTS AND CONCLUSIONS

The results and insights presented in this study are ***conditional on the occurrence of core damage***. Thus, this study gives no indication about the likelihood of these postulated accidents, but rather what could be expected given that core damage does occur. The input to this analysis is the core damage sequence definitions from the Level 1 coarse screening analysis. In this Level 1 scoping analysis conservative assumptions were made with regard to the availability of certain systems and the performance of the plant operators. These assumptions provided the necessary simplifications such that the dominant sequences could be identified and still keep the scope of the study manageable. While the calculated frequencies from the Level 1 study are used to rank the sequences, the absolute values of these frequencies were not reported due to the conservative nature of many of the necessary simplifications. Thus, frequencies were not propagated through to the Level 2 and 3 analyses. It is within this framework that the abridged study was performed. Therefore, when interpreting these results it must be remembered that frequency information is not available to indicate the likelihood of accidents and simplifying assumptions were made in both the Level 1 and the Level 2/3 studies.

The following is a list of insights obtained from this study:

o During POS 6 the majority of the mitigative features of the plant are bypassed or are unavailable. The vessel and drywell are open to the containment and, thus, the suppression pool is effectively bypassed. Furthermore, the containment spray system is unavailable during these accidents. Thus, steam and radionuclides are released directly into the containment atmosphere without being scrubbed by either the suppression pool or the containment sprays. For the accidents in which the containment hatch is removed, the only significant plant mitigative feature is the deposition that occurs in the auxiliary building. If the containment is closed but then fails during core damage, the auxiliary building is also bypassed.

o Because of the lack of mitigative features associated with these accidents, the source terms tend to be quite large.

o The consequences associated with these accidents are also significant. Offsite consequences are comparable with consequences associated with full power accidents. Onsite consequences are large.

o The time from the accident initiation to the onset of core damage is significant (i.e., from 18 to 28 hours). Thus, there is a considerable amount of time to restore core cooling and to close the containment. If offsite ac power is available, it is likely that the operators will close the containment prior to core damage.

o The pressure suppression features (i.e., suppression pool and containment sprays) of the Mark III design are bypassed during POS 6. Since the ultimate pressure capacity of the containment is fairly low, the plant is vulnerable to pressurization events accompanying vessel failure and associated with hydrogen burns. Failure to avoid or mitigate pressure excursions from these events could result in early containment failure.

o Because of the large recovery potential associated with these accidents (i.e., which were not fully accounted for in either the Level 1 analysis or this abridged analysis because of simplifying assumptions), POS 6 offsite risk *could* be significantly lower than the risk associated with full power accidents.

o Recovered accidents can pose a significant threat to onsite personnel.

o Because of the lack of mitigation features associated with accidents initiated in this POS, the auxiliary building and the SBGTs could play a significant role in the mitigation of the release, especially for recovered accidents.

There were many issues that were identified in this study that could affect the possible accident progressions and consequences. The resolution of many of these issues was beyond the scope of this abridged analysis and will have to be addressed in any more detailed analysis that is performed in the future. The following is a list of potentially significant issues:

o **Containment Closure.** The effects that the temperature, humidity, and radiation have on the plant personnel's ability to close the containment needs to be addressed in more detail. Containment closure is a critical issue that will affect the consequences associated with these accidents.

o **Containment Loading.** Hydrogen combustion phenomena associated with this plant configuration need to be investigated. In this plant configuration steam and hot hydrogen are released directly into the containment atmosphere. The amount of steam blanketing and air ingression and the availability of ignition sources will all affect the likelihood and magnitude of hydrogen burns. The effectiveness of the hydrogen ignition system in this plant configuration also needs to be investigated. The loading from in-vessel steam explosions is another issue that needs to be addressed. With the vessel head off in this POS and the relatively low failure pressure of this containment, in-vessel steam explosions could be a significant mechanism for early containment failure.

o **Source Term.** There are several events that can enhance the source term that were not included in the PRA model. First, the role that air ingression plays during core damage needs to be investigated. If significant air ingression does occur, the in-vessel phase of the core damage process could be significantly altered and the release of certain radionuclides enhanced. Second, the relocation of intact fuel from an in-vessel steam explosion could also result in the enhancement of an early source term. This issue was not addressed in this analysis. Third, for recovered accidents the embrittlement and failure of the clad could lead to a release earlier than what is currently modeled. This could be particularly important for onsite consequences.

- o **Auxiliary Building.** For accidents in which the containment is open during core damage, the auxiliary building could play a major role in mitigating the release. The radionuclide retention capabilities of this building need to be assessed in more detail than what was done in this abridged analysis. Furthermore, the effectiveness of the SBGT system to mitigate the release, especially for recovered accidents, also needs to be assessed.

- o **Onsite Consequences.** Only a scoping type analysis of onsite consequences was performed in this study. In the calculation of doses in the building, the integrated concentrations were based on average concentrations from GGLPSOR and on crude estimates of the residence times in the buildings. More detailed information on the concentration as a function of time and on the residence time would produce more realistic dose estimates.

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Vol. 6, Part 2

Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1

Evaluation of Severe Accident Risks for Plant Operational State 5 During a Refueling Outage

Supporting MELCOR Calculations

Draft Completed: June 1994
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Abstract

To gain a better understanding of the risk significance of low power and shutdown modes of operation, the Office of Nuclear Regulatory Research at the NRC established programs to investigate the likelihood and severity of postulated accidents that could occur during low power and shutdown (LP&S) modes of operation at commercial nuclear power plants. To investigate the likelihood of severe core damage accidents during off power conditions, probabilistic risk assessments (PRAs) were performed for two nuclear plants: Unit 1 of the Grand Gulf Nuclear Station which is a BWR-6 Mark III boiling water reactor (BWR) and Unit 1 of the Surry Power Station which is three loop, subatmospheric, pressurized water reactor (PWR). The analysis of the BWR was conducted at Sandia National Laboratories while the analysis of the PWR was performed at Brookhaven National Laboratory.

This multi-volume report presents and discusses the results of the BWR analysis. The subject of this part presents the deterministic code calculations, performed with the MELCOR code, that were used to support the development and quantification of the PRA models. The background for the work documented in this report is summarized, including how deterministic codes are used in PRAs, why the MELCOR code is used, what the capabilities and features of MELCOR are, and how the code has been used by others in the past. Brief descriptions of the Grand Gulf plant and its configuration during LP&S operation, and of the MELCOR input model developed for the Grand Gulf plant in its LP&S configuration are given. The results of MELCOR analyses of various accident sequences for the POS 5 plant configuration are presented, for accidents initiated at several different times after scram and shutdown, including shortened thermal/hydraulic and core damage calculations done in support of the Level 1 analysis and full plant analyses, including containment response and source terms, supporting the Level 2 analysis. MELCOR calculations of various accident scenarios for POS 6 (*i.e.*, a selected regime of refueling mode of operation) also are given; these include a reference calculation and sensitivity studies on both plant configuration assumed and on code input options used.

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Executive Summary

The safety of commercial nuclear plants during full power operation has been previously assessed in many probabilistic safety assessment studies. Recent events at several nuclear power generating stations, recent safety studies, and operational experience, however, have all highlighted the need to assess the safety of plants during low power and shutdown modes of operation. In contrast to full power operation, there is very little information on the safety of plants during low power and shutdown modes of operation. In the past, the assumption has been that power operation is the risk dominant mode of operation because the decay energy is greatest at the time of shutdown and then decays as a function of time. Thus, the rationale was that during shutdown modes of operation the decay heat would be sufficiently low that there would be plenty of time to respond to any abnormal event that may threaten the core cooling function. Furthermore, given the unlikely event that a release did occur, radioactive decay would lessen the radiological potential of the release. This argument's Achilles' heel is that the technical specifications allow for more equipment to be inoperable in off power conditions. Thus, while there may be more time to respond to an accident during shutdown, many of the systems that are relied on to mitigate an accident during power operation may not be available during shutdown.

To gain a better understanding of the risk significance of low power and shutdown modes of operation, the Office of Nuclear Regulatory Research at the NRC established programs to investigate the likelihood and severity of postulated accidents that could occur during low power and shutdown (LP&S) modes of operation at commercial nuclear power plants. To investigate the likelihood of severe core damage accidents during off power conditions, probabilistic risk assessments (PRAs) were performed for two nuclear plants: Unit 1 of the Grand Gulf Nuclear Station which is a BWR-6 Mark III boiling water reactor (BWR) and Unit 1 of the Surry Power Station which is three loop, subatmospheric, pressurized water reactor (PWR). These studies consist of the following five analysis components: accident frequency analysis, accident progression analysis, analysis of the release and transport of radioactive material (*i.e.*, source term analysis), consequence analysis, and a risk integration analysis. A principle product of such a Level 3 PRA is an expression for risk.

The analysis of the BWR was conducted at Sandia National Laboratories while the analysis of the PWR was performed at Brookhaven National Laboratory. This multivolume report presents and discusses the results of the BWR analysis. Volume 1 summarizes the BWR study. Volumes 2-5 present the accident frequency analysis (*i.e.*, Level 1). Volume 6 presents the Level 2/3 analysis performed under FIN L1679. Part 1 of Volume 6 presents the accident progression, radionuclide release and transport, consequence and risk analyses. The subject of this part, *i.e.*, Part 2 of Volume 6, presents the deterministic code calculations, performed with the MELCOR code, that were used to support the development and quantification of the PRA models.

In this report, the background for the work documented in this report is first summarized, including how deterministic codes are used in PRAs, why the MELCOR code is used, what the capabilities and features of MELCOR are, and how the code has been used by others in the past. Brief descriptions of the Grand Gulf plant and its configuration during LP&S operation, and of the MELCOR input model developed for the Grand Gulf plant in its LP&S configuration are given. The results of MELCOR analyses of various accident sequences for the POS 5 plant configuration are presented, for accidents initiated at several different times after scram and shutdown, including shortened thermal/hydraulic and core damage calculations done in support of the Level 1 analysis and full plant analyses, including containment response and source terms, supporting the Level 2 analysis. MELCOR calculations of various accident scenarios for POS 6 (*i.e.*, a selected regime of refueling mode of operation) also are given; these include a reference calculation and sensitivity studies on both plant configuration assumed and on code input options used.

MELCOR is an integrated, relatively fast-running, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants, being developed at Sandia National Laboratories for the NRC and the U. S. Department of Energy (USDOE). An entire spectrum of severe accident phenomena is modelled in MELCOR in a unified framework for both boiling water reactors and pressurized water reactors. Characteristics of severe accident progression that can be treated with MELCOR include the thermal/hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heatup, degradation and relocation; fission product release and transport; hydrogen production, transport and combustion; core-concrete attack; heat structure response; and the impact of engineered safety features on thermal/hydraulic and radionuclide behavior. The MELCOR computer code has been developed to the point that it is now being successfully applied in both experiment analyses, intended for code validation, and in plant analyses, in support of PRAs and accident management studies.

A series of MELCOR calculations were done to support the quantification of the Level 1 PRA models for POS 5. POS 5 is rigorously defined as: "Cold Shutdown (Operating Condition 4) and Refueling (Operating Condition 5) only to the point where the vessel head is off." For these calculations, the parameters of interest include the times to reach various pressure and/or level setpoints, the time to top-of-active-fuel (TAF) uncovery, the times to core heatup and clad failure and the time to vessel failure. Several general scenarios when the plant is in POS 5 have been considered:

open MSIVs,
low pressure boiloff,
high pressure boiloff with closed RPV head vent,
high pressure boiloff with open RPV head vent,
large break LOCA,
station blackout with failure to isolate SDC,
station blackout with firewater addition,
station blackout with 10 hr firewater addition followed by high pressure boiloff,
station blackout with 10 hr firewater addition followed by failure to isolate SDC.

In all these Level 1 cases, the drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open.

Calculations were performed for several different times from shutdown for each of these accident scenarios: 7 hr, 24 hr, 59 hr, 12 days, and 40 days. The first two times correspond to the times used to determine the decay heats for the first and second time windows; the third time corresponds to the midpoint of the second time window; the last time corresponds to the time corresponding to the decay heat level in the third time window. Because the primary interest was in time to core damage, these Level 1 support calculations were run until any of the following: vessel failure, code abort or 24 hr of transient. If any sequence produced no significant core damage within 24 hr for a given decay heat level, no further calculations were done with longer shutdown time s (*i.e.*, lower decay heat levels).

Based partly on the results of the MELCOR calculations done in support of the POS 5 Level 1 analysis, a number of accident sequences were eliminated from consideration as not resulting in core damage within the first 24 hr from the start of the accident. The remaining sequences, those leading to core damage within 1 day and with a frequency greater than the Level 1 truncation frequency, were grouped into plant damage states or PDSs. The plant damage states are ranked by their relative contribution to core damage frequency as:

MELCOR Level 2 Support Calculations – Sequences and Relative Contribution of Plant Damage States to Core Damage Frequency

Plant Damage State	Time After Shutdown	Fraction Contributed	Sequence Description
PDS 3-1	40 day	0.338	LBLOCA with flooded containment
PDS 2-2	24 hr	0.242	SBO w/o firewater, break in SDC
PDS 2-1	24 hr	0.170	LBLOCA with flooded containment
PDS 2-4	24 hr	0.104	Low-P Boiloff with flooded containment
PDS 1-3	7 hr	0.032	SBO w/10 hr-firewater, High-P Boiloff
PDS 1-1	7 hr	0.019	LBLOCA with flooded containment
PDS 1-2	7 hr	0.015	SBO w/o firewater, break in SDC
PDS 1-5	7 hr	0.008	Low-P Boiloff with flooded containment
PDS 2-5	24 hr	0.007	High-P Boiloff with closed containment
PDS 2-6	24 hr	0.006	Open MSIVs with closed containment
PDS 2-3	24 hr	0.054	Same as PDS 2-2, but with potential to recover AC power
PDS 1-4	7 hr	0.005	Same as PDS 1-2, but with potential to recover AC power

Complete MELCOR accident analyses have been done for these sequences in support of the Level 2 PRA, with results described in detail. (The last two sequences in the table are identical to other sequences in the table with regard to MELCOR calculations, but with different recovery assumptions in the Level 2 PRA.)

An abridged risk analysis was also performed on the early portion of the refueling mode of operation. In the Level 1 coarse screening analysis this mode of operation is referred to as plant operating state 6 (POS 6). During a refueling outage, the plant will enter POS 6 prior to loading fresh fuel (*i.e.*, going down) and then following fuel transfer on the way back up to power conditions (*i.e.*, going up). In this POS 6 study, only the going-down phase is analyzed. POS 6 begins when the vessel head is detached and ends when the upper reactor cavity has been filled with water. Prior to this mode of operation, the containment equipment hatch and personnel locks have been opened, the drywell head has been removed and the drywell equipment hatch and personnel locks have been opened. Thus the suppression pool is effectively bypassed both from the vessel and from the drywell (*i.e.*, steam lines are plugged and the drywell is open).

A series of MELCOR calculations also were performed to support the abridged risk analysis of POS 6. All the MELCOR POS 6 calculations were done assuming that, at the start of the accident, shutdown cooling, suppression pool cooling and containment sprays are all unavailable and remain unavailable during the accident; coolant injection is not provided to the vessel during the accident, and suppression pool makeup is not dumped into the suppression pool. The MELCOR POS 6 calculations done included a number of variations on the exact plant configuration assumed. In addition, a few sensitivity studies were done on various code options and/or parameters. As part of these sensitivity studies, a set of calculations were performed to investigate the effect that possible ingression of air into the core region would have on the core damage process.

1 Introduction

1.1 Background

The safety of commercial nuclear plants during power operation has been previously assessed in many probabilistic safety assessment studies. The U.S. Nuclear Regulatory Commission (NRC) has been an active participant in these studies including the landmark Reactor Safety Study [1], the five plant studies performed as part of the NUREG-1150 study [2] and the LaSalle plant analysis performed under RMIEP/PRUEP programs [3, 4]. Furthermore, all licenses are required to perform an individual plant examination (IPE) that assesses the safety of the plant during full power operation.

Recent events at several nuclear power generating stations, recent safety studies, and operational experience, however, have all highlighted the need to assess the safety of plants during low power and shutdown modes of operation. In contrast to full power operation, there is very little information on the safety of plants during low power and shutdown modes of operation. In the past, the assumption has been that power operation is the risk dominant mode of operation because the decay energy is greatest at the time of shutdown and then decays as a function of time. Thus, the rationale was that during shutdown modes of operation the decay heat would be sufficiently low that there would be plenty of time to respond to any abnormal event that may threaten the core cooling function. Furthermore, given the unlikely event that a release did occur, radioactive decay would lessen the radiological potential of the release. This argument's Achilles' heel is that the technical specifications allow for more equipment to be inoperable in off power conditions. Thus, while there may be more time to respond to an accident during shutdown, many of the systems that are relied on to mitigate an accident during power operation may not be available during shutdown.

To gain a better understanding of the risk significance of low power and shutdown modes of operation, the Office of Nuclear Regulatory Research at the NRC established programs to investigate the likelihood and severity of postulated accidents that could occur during low power and shutdown (LP&S) modes of operation at commercial nuclear power plants. To investigate the likelihood of severe core damage accidents during off power conditions, probabilistic risk assessments (PRAs) were performed for two nuclear plants: Unit 1 of the Grand Gulf Nuclear Station which is a BWR-6 Mark III boiling water reactor (BWR) and Unit 1 of the Surry Power Station which is three loop, subatmospheric, pressurized water reactor (PWR). These studies are Level 3[1] PRAs and, as such, consist of the following five analysis components: accident frequency analysis, accident progression analysis, analysis of the release and transport of radioactive material (*i.e.*, source term analysis), consequence analysis, and a risk integration analysis. A principle product of a Level 3 PRA is an expression for risk.

[1]The Level 1 analysis consists of the accident frequency analysis; the Level 2 analysis consists of the accident progression and radionuclide release and transport analyses; and the Level 3 analysis consists of the consequence analysis. A Level 3PRA combines the results from each of the constituent analyses and develops an expression for risk.

The analysis of the BWR was conducted at Sandia National Laboratories while the analysis of the PWR was performed at Brookhaven National Laboratory. The LP&S PWR analysis is reported in NUREG/CR-6144 [5] and will not be discussed any further in this report. This multi-volume report presents and discusses the results of the BWR analysis. Volumes 2-5 present the accident frequency analysis (*i.e.*, Level 1). Volume 6 presents the Level 2/3 analysis performed under FIN L1679. Part 1 of Volume 6 presents the accident progression, radionuclide release and transport, consequence and risk analyses. The subject of this part, *i.e.*, Part 2 of Volume 6, presents the deterministic code calculations, performed with the MELCOR code [6], that were used to support the development and quantification of the PRA models.

1.2 Use of Deterministic Codes in Level 3 PRA

Deterministic calculations are vital analysis that are used to support the development and quantification of the PRA models used in the Level 1 and 2 analyses. Deterministic calculations are used to define success criteria and timing characteristics for the Level 1 analysis. For example, these calculations are used to: (1) define the regimes under which certain injection system can be used to cool the core, (2) determined the amount of time the operators have to respond to an initiating event and perform appropriate actions to terminate or mitigate the accident, and (3) determine when the onset of core damage occurs. In the Level 2 analysis, deterministic calculations are used to estimate the timing of key events in the accident (*e.g.*, the onset of core damage, the time at which the vessel fails, and the time when the containment fails), characteristics of the core degradation process, the conditions in the containment as a function of time (*e.g.*, temperature, pressure, composition of the atmosphere), the occurrence and impact of certain phenomena (*e.g.*, hydrogen combustion), and the release and transport of radioactive material in the containment. Wherever possible, a consistent set of calculations are use to support both the Level 1 and Level 2 analysis to ensure that a consistent set of assumptions are being used and to maintain continuity in the timing of events.

The results from deterministic analyses are incorporated in the Level 2 analysis in the following manner:

- Calculations are performed for the important accident sequences (*i.e.*, typically Plant Damage States) that lead to core damage; sensitivity calculations are performed to investigate important facets of the accident,
- Following a general understanding of the possible accident progressions from the deterministic calculations and other source of information (*e.g.*, results from experiments), major events that can affect the progression of the accident and the release and transport of radioactive material are identified. These events form some of the top events of the Level 2 Accident Progression Event Tree.
- Results from these calculations supplemented by other information serve as the basis for quantifying the PRA models. Since uncertainty is unavoidable in these

calculations (*e.g.*, in the initial conditions, phenomenological models, and the model of the plant), judgement techniques are often used to translate results from deterministic analyses into a form suitable for probabilistic analysis. For example, a deterministic calculation may indicate that based on the prescribed initial and boundary conditions, a combustible mixture of hydrogen will form in the containment and combustion of this mixture will result in a peak pressure P. However, the initial and boundary conditions are uncertain and there are many uncertainties associated with the phenomena involved in this process, for example, the amount of hydrogen produced, the likelihood that the mixture will ignite, and once ignited, the rate of combustion. Thus, the results from the calculations are assessed in light of the uncertainties involved in the process to yield expressions for the likelihood that the burn occurs and the likelihood that various pressures are realized.

In this PRA, the MELCOR code was used to perform the deterministic calculations because:

- it addresses all major aspects of a severe core damage accident,
- its input structure allows the user to modify the plant model such that the many possible plant configurations during shutdown can be modelled,
- runs quickly enough that integral calculations (*i.e.*, from accident initiation to the release of radioactive material from the plant into the environment) and supporting sensitivity calculations can be performed for the dominant accident scenarios, and
- it allows parametric studies to be performed on parameters that may be important to the progression to of the accident and the release of radioactive material.

1.3 Description of MELCOR

MELCOR [6] is a fully integrated, relatively fast-running, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants, being developed at Sandia National Laboratories for the NRC and the U. S. Department of Energy (USDOE). An entire spectrum of severe accident phenomena is modelled in MELCOR in a unified framework for both boiling water reactors and pressurized water reactors. Characteristics of severe accident progression that can be treated with MELCOR include the thermal/hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heatup, degradation and relocation; fission product release and transport; hydrogen production, transport and combustion; core-concrete attack; heat structure response; and the impact of engineered safety features on thermal/hydraulic and radionuclide behavior.

MELCOR is composed of a number of different packages, each of which models a different portion of the accident phenomenology or program control. For example, the Control Volume Hydrodynamics (CVH) package calculates the thermal/hydraulics of control volumes, and the Core (COR) package evaluates the core behavior. Each of the packages presently in MELCOR is listed:

BH Bottom Head: Models the bottom head in BWR systems. (This model was developed by Oak Ridge National Laboratory.)

BUR Combustion of Gases: Compares conditions within control volumes against criteria for deflagrations and detonations. Initiates and propagates deflagrations involving hydrogen and carbon monoxide. Calculates burn completeness and flame speed.

CAV Core-concrete Interactions: CORCON-MOD2 with enhanced sensitivity analysis and multi-cavity capabilities.

CF Control Functions: Evaluates user-specified "control functions" and applies them to define or control various aspects of the computation such as opening and closing of valves; controlling plot, edit, and restart frequencies; defining new plot variables, *etc.*

COR Core Behavior: Evaluates the behavior of the fuel and other core and lower plenum structures including heatup, candling, flow blockages, debris formation and relocation, bottom head failure, and release of core material to containment.

CVH Control Volume Hydrodynamics: In conjunction with the FL package, evaluates mass and energy flows between control volumes.

CVT Control Volume Thermodynamics: Evaluates the thermodynamic state within each control volume for the CVH package.

DCH Decay Heat: Used by other packages to evaluate decay heat power associated with radionuclide decay.

EDF External Data Files: Controls the reading and writing of large external data files, in close interface to the Control Function and Transfer Process packages.

EOS Equation of State: The CVT, H2O, and NCG packages are stored as one block of code under this name.

ESF Engineered Safety Features: Models the thermal/hydraulics of fan coolers, storage tanks, injection and recirculation pumps and heat exchangers, and ice condensers. Currently, only the fan cooler model is included. The containment sprays are a separate package.

EXEC Executive Package: Controls execution of MELGEN and MELCOR.

FDI Fuel Dispersal Interactions: Models ex-vessel debris relocation, heat transfer, and oxidation due to fuel-coolant interactions and direct heating.

FL Flow Paths: Models, in conjunction with the CVH package, the flow rates of gases and liquid water through the flow paths that connect control volumes.

H2O Water Properties: Evaluates the water properties based on the Keenan and Keyes equation of state extended to high temperatures using the JANAF data.

HS Heat Structures: Models the thermal response of heat structures and mass and heat transfer between heat structures and control volume pools and atmospheres. Treats conduction, condensation, convection, and radiation, as well as degassing of unlined concrete.

MP Material Properties: Evaluates the physical properties of materials for other packages except for common steam and non-condensible gas properties (see H2O and NCG)

NCG Non-Condensable Gas Equation of State: Evaluates the properties of noncondensable gas mixtures using an equation of state based on the JANAF data.

PROG Part of MELGEN/MELCOR executive package separated for computer library and link purposes.

RN Radionuclide Behavior: Models radionuclide releases, aerosol and fission product vapor behavior, transport through flow paths, and removal due to ESFs. Allows for simplified chemistry.

SPR Containment Sprays: Models the mass and heat transfer rates between containment spray droplets and control volumes.

TF Tabular Functions: Evaluates user-selected "tabular functions" to define or control various aspects of the computation such as mass and energy sources; integral decay heat; plot, edit, and restart frequencies, *etc.*

TP Transfer Process: Controls the transfer of core debris between various packages and the associated transfer of radionuclides within the RN package.

UTIL Utility Package: Contains various utilities employed by the rest of the code.

Only a brief summary of the phenomenological modelling in the major packages can be included here; for more detailed information, see [6].

Thermal/hydraulic processes are modelled in MELCOR by the CVH/FL packages, while the thermodynamic calculations are performed within the CVT package. The CVH package is concerned with control volumes and their contents, and the FL package represents the connections which allow transfer of these contents between control volumes.

No formal distinction is made between the reactor coolant system and containment; the same models and solution algorithms are used for both and the resulting equations solved simultaneously. Within the basic control volume formulation, the treatment is quite general; unlike the MAAP code [7], no specific nodalization is built in, and there are no predefined models for reactor components such as steam generators. All systems and components are built up from general control volumes, flow paths, and other elements (such as heat structures and control functions). In some cases, the control volumes may correspond to physical tanks, with the flow paths representing pipes connecting them; in other cases, the volumes may be geometrical regions such as portions of larger physical rooms, with the flow paths representing the geometrical surfaces separating them.

Hydrodynamic materials in control volumes (*i.e.*, coolant and noncondensables) are assumed to separate under gravity within a control volume to form a pool beneath an atmosphere. The separation need not be complete; the pool may contain vapor bubbles and the atmosphere may contain liquid droplets. The shape of the volume is defined though a user-input volume/altitude table to allow the elevation of the pool surface to be determined. The mass exchange models include both an optional thermal and mechanical equilibrium model which assumes the same pressure and temperature for both pool and atmosphere, and the default thermal nonequilibrium model which assumes the same pressure but different temperatures for pool and atmosphere.

The control volumes are connected by flow paths through which hydrodynamic materials move without residence time, driven by a momentum equation. Each control volume may be connected to an arbitrary number of others, and parallel flow paths (connecting the same pair of control volumes) are permitted; there are no restrictions on the connectivity of the network built up in this way. The flow path area can be modified by input to model valves, obstructions, *etc.* Appropriate hydrostatic head terms are included in the momentum equation for the flow paths, allowing calculation of natural circulation.

The HS package in MELCOR calculates one-dimensional heat conduction within an intact, solid structure and energy transfer across its boundary surfaces into control volumes. The modelling capabilities of heat structures are general and can include pressure vessel internals and walls, fuel rods with nuclear or electrical heating, steam generator tubes, piping walls, *etc.*

Convective heat transfer is calculated using an extensive set of heat transfer coefficient correlations for natural or forced convection to both the pool and atmosphere; pool boiling heat transfer utilizes correlations for nucleate boiling, critical heat flux, film boiling and transition boiling. Radiation heat transfer can be specified between a heat structure surface and the boundary volume atmosphere, with two options (an equivalent band model and a gray gas model) available.

Mass transfer models for heat structure surfaces include condensation and evaporation in the presence of noncondensables with an appropriate limit for pure steam, and flashing in any environment. Liquid films on heat structure surfaces are also modelled. A user-input degassing model is provided for the release of gases from materials which are contained in heat structures, for example, to represent the release of water vapor or carbon dioxide from concrete as its temperature increases.

The MELCOR COR package calculates the thermal response of the core and lower plenum structure, including the portion of the lower head directly beneath the core, and models the relocation of core materials during melting, slumping and debris formation. The core and lower plenum are divided into a number of user-specified axial levels and concentric radial rings. A number of component types and materials are modelled. Fuel pellets, cladding, grid spacers, canister walls (for BWRs), other structure (*e.g.*, support plates, control rods, guide tubes) and particulate debris are modelled separately within individual COR cells. Either PWR or BWR systems may be modelled.

A number of heat transfer processes are modelled in each COR cell. Thermal radiation within a cell and between cells in both the axial and radial directions is calculated, as well as radiation to boundary heat structures (*e.g.*, the core shroud or upper plenum) from the outer and upper cells; radiation to a liquid pool (or the lower head if no pool is present) and to steam is also included. Conduction radially across the fuel-clad gap and axially between cells, and optionally between the core and radial boundary heat structures, is modelled; an analytical model for axial conduction is applied within structures that are partially covered with liquid pool. Convection to the control volume fluids is modelled for a wide variety of fluid conditions and structure surface temperatures, including nucleate and film boiling.

Oxidation of zircaloy and steel is modelled for both the limiting cases of solid-state diffusion of oxygen through the oxide layer and gaseous diffusion of steam or oxygen through the mixture. The core degradation model treats eutectic liquefaction and dissolution reactions, candling of molten core materials (*i.e.*, downward flow and refreezing), and the formation of liquid and particulate debris. Geometric variables (*e.g.*, cell surface areas and volumes) are updated for changing core geometry. A lower head penetration failure model is also included.

The interaction of the core debris released from the vessel with the concrete basemat in the cavity is modelled by the CAV package in MELCOR using the CORCON-Mod2 code [8]. The molten debris may contain large amounts of unoxidized metals such as zirconium and chromium as well as oxidic species such as ZrO_2 and UO_2. These materials are assumed to stratify in the cavity because they have different densities. CORCON calculates the rate of erosion in the concrete basemat; the temperature and composition of the molten layers; and the temperature, flow rate and composition of gases (such as CO_2, CO, H_2 and water vapor) evolving from the concrete. Heat generation in the molten pool is due both to decay heat and to the heat of reactions.

The molten core debris in the cavity is assumed to be stratified into a dense bottom layer and a lighter top layer. Initially, the oxide layer is calculated to be less dense than the metallic layer, but after the molten concrete slag dilutes the heavy oxide layer, the oxide layer becomes less dense than the metallic layer and rises to the top. Each layer is assumed to be isothermal. Heat is exchanged between the melt and the concrete, the layers in the melt, and the top surface of the melt and the atmosphere and structures above it. The melt-concrete heat transfer is modelled by a gas film model which assumes the occurrence of Taylor-instability bubbling on the pool bottom and a flowing gas film vertically along the melt pool. Inter-layer heat transfer in the presence of gas bubbling is modelled. If a coolant layer is present over the melt pool, boiling heat transfer to the overlying coolant layer is also modelled.

The RN package models the behavior of fission product aerosols and vapors and other trace species, including release from fuel and debris, aerosol dynamics with vapor condensation and revaporization, deposition on structure surfaces, transport through flow paths and removal by engineered safety features. The package also allows for simplified chemistry controlled by the user.

Rather than tracking all fission product isotopes, the masses of all the isotopes of an element are modelled as a sum; furthermore, elements are combined into classes, groups of elements with similar chemical characteristics. Fifteen material classes are used by default, twelve containing fission products plus boron, water and concrete oxides. User-specified combination of classes to form new classes upon release (*e.g.*, Cs + I to CsI) is permitted.

The release of fission products from the fuel within the vessel is modelled using either the CORSOR, CORSOR-M or CORSOR-Booth representations of radiological release data for irradiated fuel. The CORSOR model is a simple correlational relationship based on data from early experiments [63]. Release of volatiles is assumed to be limited by diffusion, and all volatiles share the same release parameters, obtained by averaging experimental results; release of nonvolatiles is assumed to be limited by vaporization, and vapor pressures are scaled for consistency with experimental observations. The fractional release coefficients in CORSOR are simple exponentials, with constants selected for each species in specific temperature ranges based upon fitting experimental data; the fractional release coefficients used in CORSOR-M utilize an Arrhenius-type equation with constants representing empirical fits to experimental data. Other parameters possibly affecting release rates (such as pressure, atmospheric composition, fuel characteristics, chemistry, radiation environment, flow rates and the extent of fuel degradation) are not considered explicitly in either the CORSOR or CORSOR-M correlations. Time-dependent Cs release data from the expanded experiment data base currently available were used to fit parameters describing an effective diffusion coefficient in the new diffusion- and mass-transfer-based CORSOR-Booth model [10]; release rates of other species are then scaled to the Cs release rate. This model includes high- and low-burnup expressions, and also is a function of fuel grain size.

Releases of radionuclides occurring during core-concrete interactions in the reactor cavity are calculated using the VANESA [11] release model, which is designed to accept melt temperatures and gas generation rates from CORCON.

Aerosol dynamic processes and the condensation and evaporation of fission product vapors after release from fuel are considered by codes included within the RN package. The aerosol dynamics models are based upon MAEROS [12], a multisection, multicomponent aerosol dynamics code, but without calculation of condensation. Aerosols can deposit directly onto surfaces such as heat structures and water pools, or can agglomerate and eventually settle out. The condensation and evaporation of radionuclide vapors at aerosol surfaces, pool surfaces and heat structure surfaces are evaluated by rate equations from the TRAP-MELT2 code [13], which are based on the surface area, mass transfer coefficients, and the differences between the present surface concentration and the saturation surface concentration.

Models are available for the removal of radionuclides by pool scrubbing, filter trapping and containment spray scrubbing. The pool scrubbing model is based on the SPARC code

[14], and treats both spherical and elliptical bubbles; the model includes condensation at the pool entrance, Brownian diffusion, gravitational settling, inertial impaction and evaporative forces for the rising bubble. The filter model can remove aerosols and fission products vapors with a specified maximum mass loading. The containment spray model is based on the model in HECTR 1.5 [15] and removes both vapors and aerosols from the atmosphere.

1.4 Related MELCOR Applications

The MELCOR computer code has been developed to the point that it is now being successfully applied in both experiment analyses, intended for code validation, and in plant analyses, in support of PRAs and accident management studies. A review of MELCOR verification, validation and assessment to date reveals that most of the severe accident phenomena modelled by MELCOR have received or are receiving some evaluation.

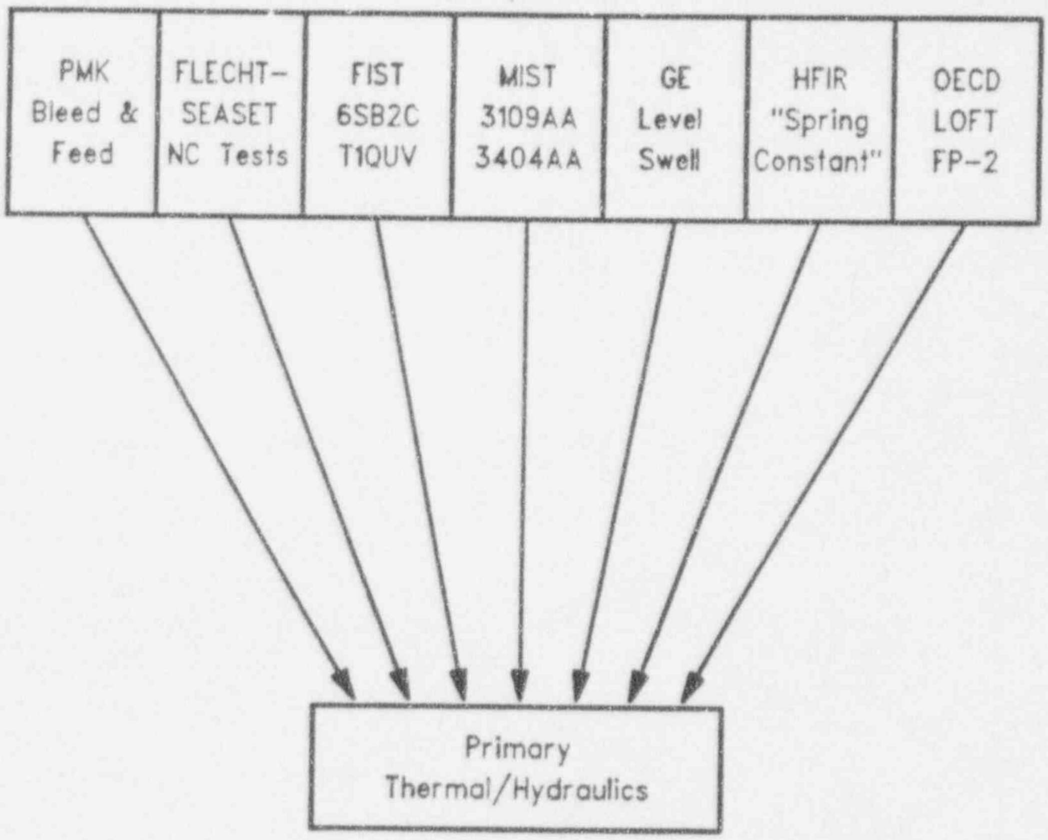
Figures 1.4.1, 1.4.2 and 1.4.3 summarize the available MELCOR assessment against experimental test data for primary system thermal/hydraulics, in-vessel core damage and fission product release and transport, and ex-vessel and containment phenomenology, respectively. Only analyses that are completed or already underway are included; analyses scheduled but not yet begun are not included.

Reactor coolant system thermal/hydraulic response, core heatup and degradation, and fission product and aerosol release and transport in a PWR geometry all were studied at full plant scale in the TMI-2 accident analysis [16], and are important in LOFT LP-FP-2 [17]. However, there is no experiment (not even the TMI accident) which represents all features of a severe accident (*i.e.*, primary system thermal/hydraulics; in-vessel core damage; fission product and aerosol release, transport and deposition; ex-vessel core-concrete interaction; and containment thermal/hydraulics, and hydrogen transport and combustion), and only the TMI accident is at full, plant scale. It is therefore necessary for severe accident codes to supplement standard assessment against experiment (and against simple problems with analytic or otherwise obvious solutions) with plant calculations that cannot be fully verified, but that can be judged against expert opinion for reasonableness and internal self-consistency (particularly using sensitivity studies) and also can be compared to other code calculations for consistency. Table 1.4.1 summarizes the plant analyses done with MELCOR to date, many with sensitivity studies and/or code-to-code comparisons. Only analyses that are completed are included; analyses in progress or scheduled but not yet begun are not included.

In the NUREG-1150 study [2] reassessing risk at five plants, MELCOR was used to perform containment response calculations [18]. In the phenomenology and risk uncertainty evaluation program (PRUEP), MELCOR calculations were performed as part of an integrated risk assessment for the LaSalle plant [4]. MELCOR calculations have been done updating the source term for three accident sequences (AG, S2D and S3D) in the Surry plant [19]. A TMLB' station blackout analysis for Surry, comparing results from

MELCOR
ASSESSMENT AGAINST EXPERIMENTS

EXPERIMENTS (IN-VESSEL THERMAL/HYDRAULICS)



MELCOR (IN-VESSEL THERMAL/HYDRAULICS)

Figure 1.4.1. MELCOR Primary System Thermal/Hydraulic Phenomena Assessment to Date

MELCOR
ASSESSMENT AGAINST EXPERIMENTS

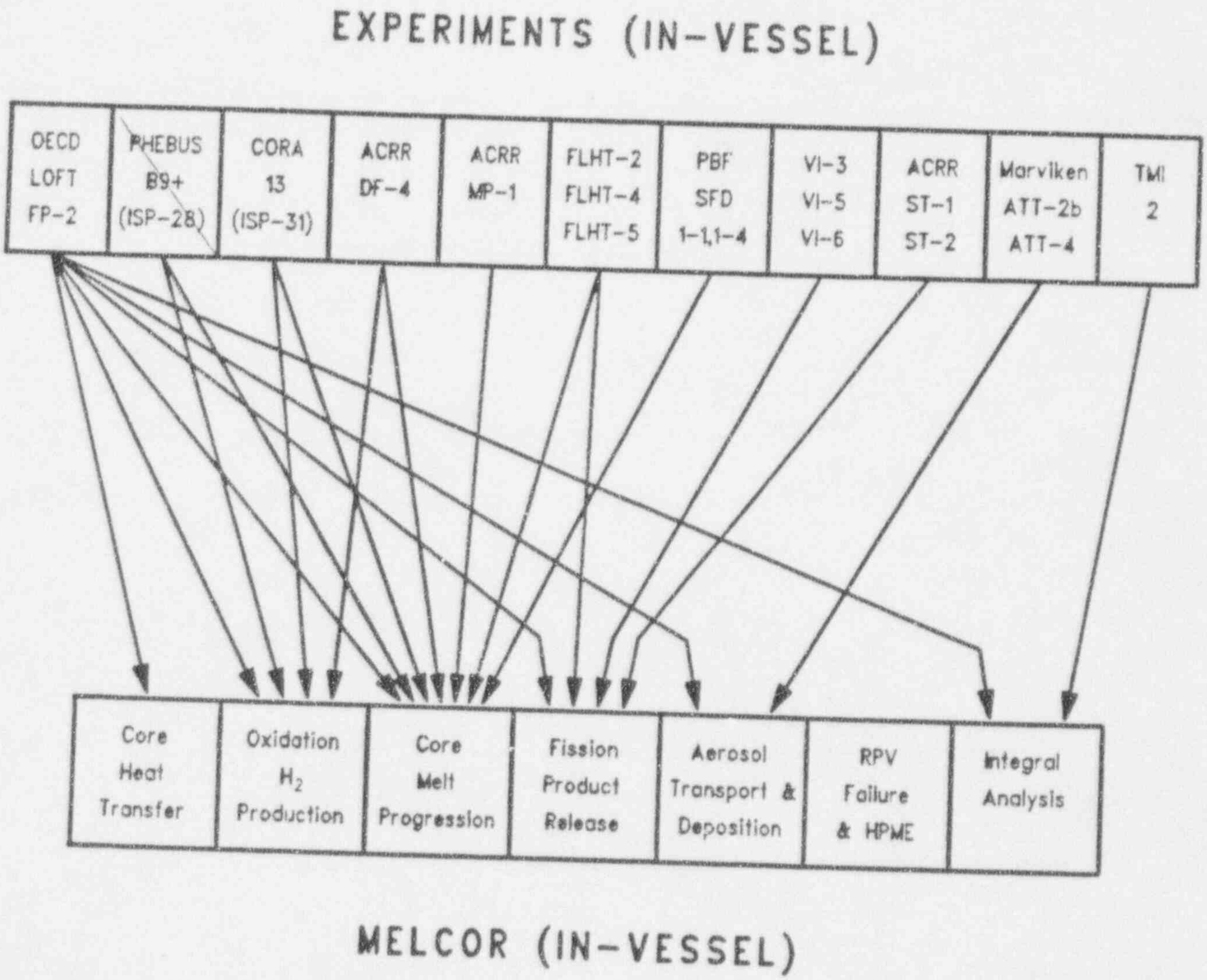
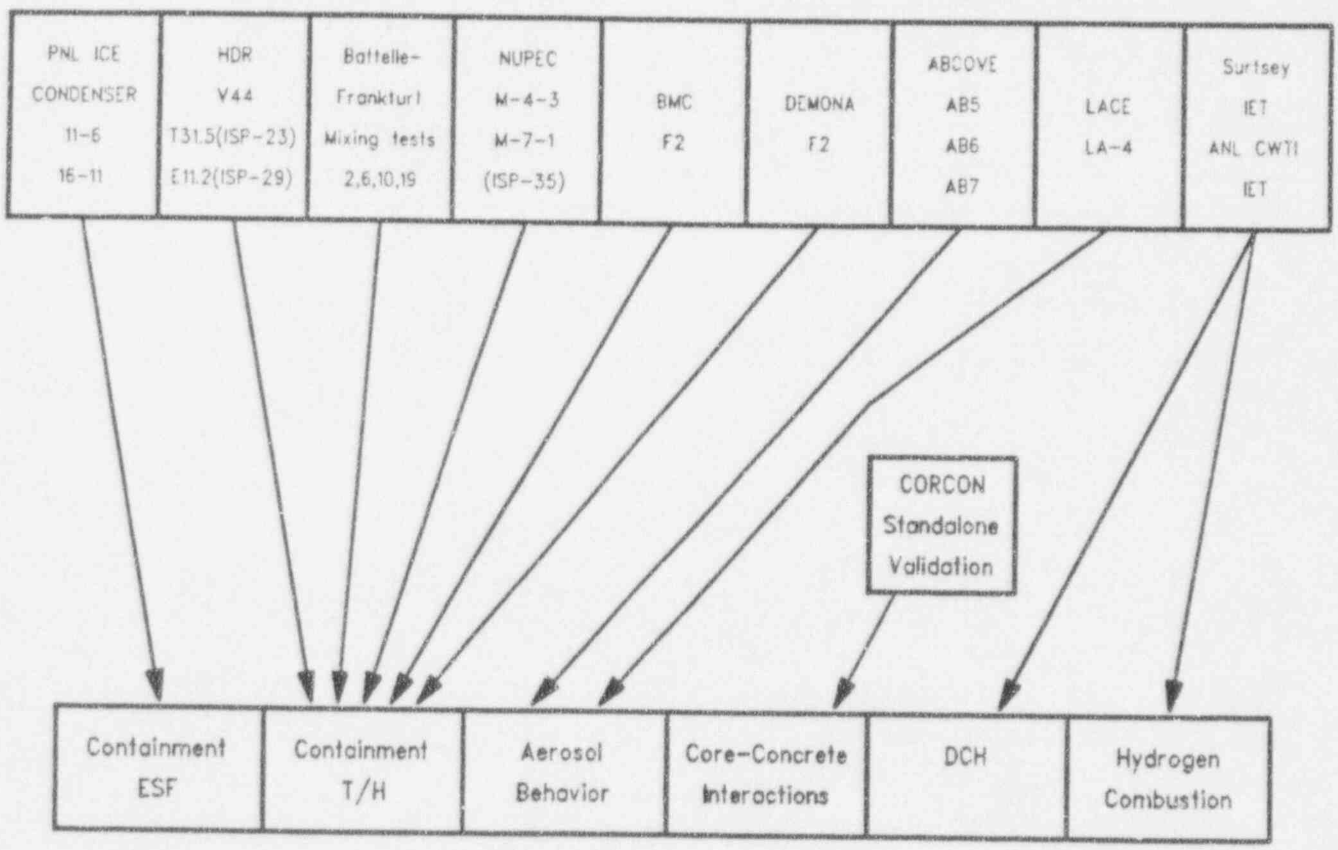


Figure 1.4.2. MELCOR In-Vessel Severe Accident Phenomena Assessment to Date

MELCOR
ASSESSMENT AGAINST EXPERIMENTS

EXPERIMENTS (EX-VESSEL)



MELCOR (EX-VESSEL)

Figure 1.4.3. MELCOR Ex-Vessel and Containment Phenomena Assessment to Date

Table 1.4.1. MELCOR Plant Calculations

Plant	Plant Type	Scenarios Analyzed
TMI-2	B&W PWR	
LaSalle	BWR/5, Mark II	Station blackout (SBO)
Surry	3-loop PWR	TMLB' w/ and w/o DCH AG, S_2D, S_3D TMLB' w/surge-line-break LBLOCA
Peach Bottom	BWR/4, Mark I	Station blackout
Oconee	B&W PWR	LOCA, TMLB'
Calvert Cliffs	CE 3-loop PWR	
Zion	4-loop PWR	
Peach Bottom	BWR/4, Mark I	Station blackout LBLOCA
Point Beach	2-loop PWR	SBO
Peach Bottom	BWR/4, Mark I	TQUX, AE
Browns Ferry	BWR/?, Mark I	S_2E SBLOCA
TVO	BWR	TB, MSLBreak 10% SBLOCA
Mühleberg	BWR/4, Mark I	SBO w/ and w/o ADS, V-sequence, SBLOCA
Beznau	2 loop PWR	SBO, V-sequence, SGTR, HL SBLOCA, IBLOCA, LBLOCA
Gösgen	3-loop PWR	SBO
Leibstadt	BWR/6, Mark III	
Ascó II	3-loop PWR	AB and V-sequence, SGTR
Garoña	BWR/3, Mark I	SBO

MELCOR 1.8.2 with results from MELCOR 1.8.1 for the same transient [20], was done as a task in the Sandia MELCOR development project. SCDAP/RELAP5 calculations of natural circulation in the Surry TMLB' accident scenario [21] were independently reviewed and assessed by Sandia [22]; a number of identified uncertainties were examined by building a corresponding MELCOR model of the Surry plant and performing sensitivity studies with MELCOR on several modelling parameters. MELCOR calculations [23] have been done to study the effects of air ingression on the consequences of various severe accident scenarios; one set of calculations analyzed a station blackout with surge line failure prior to vessel breach, starting from nominal operating conditions, while the other set of calculations analyzed a station blackout occurring during shutdown (refueling) conditions, both for the Surry plant. MELCOR calculations have been done at Sandia recently for severe accident sequences in the ABWR and the results compared with MAAP calculations for the same sequences [24].

The BNL MELCOR assessment effort includes plant analyses for the Peach Bottom BWR [25, 26]; Zion, a 4-loop Westinghouse PWR [27], as part of a MAAP/MELCOR comparison exercise; Oconee, a B&W PWR plant [28, 29, 30]; and Calvert Cliffs, a CE PWR plant [31], including comparison to other code calculations. ORNL has completed a MELCOR analysis characterizing the severe accident source term for a low-pressure, short-term station blackout sequence, with flooded and dry cavities, and a LBLOCA, in the Peach Bottom BWR-4 [32]. MELCOR has been used as a severe accident analysis tool for several of the Oak Ridge test reactor programs. MELCOR has been validated by ORNL [33] as part of the High Flux Isotope Reactor (HFIR) Safety Analysis Report (SAR) quality assurance program, before using MELCOR as the primary analysis tool for their Chapter-15 design-basis accident analyses. As part of a severe accident study for the Advanced Neutron Source (ANS) Conceptual Safety Analysis Report (CSAR), MELCOR has been used at Oak Ridge to predict the transport of fission product nuclides and their release from containment [34]. A MAAP/MELCOR comparison study for the Point Beach plant was done as a master's thesis at the University of Wisconsin [35].

AEA Technology at Winfrith Technology Centre have examined the performance of the code in plant calculations, in particular for the TMLB' sequence in Surry with and without surge line failure [36]. Three accident sequences (AB, V, and SGTR) for the Ascó II plant [37, 38, 39, 40], and two station blackout sequences in the Garoña plant, have been done [41] by the Catedra de Tecnologia Nuclear, Universidad Politecnica de Madrid. MELCOR has been used by the Netherlands Energy Research Foundation, Energieonderzoek Centrum Nederland (ECN) mainly to analyze severe accidents for the General Electric ABWR and SBWR designs.

MELCOR calculations have been done for two plant scenarios, a station blackout and a main steam line break, in the Teollisuuden Voima Oy (TVO Power Company) nuclear power plant [42, 43, 44], including a MAAP/MELCOR comparison study with the MAAP runs done by TVO and the MELCOR runs done by Valtion Teknillinen Tutkimuskeskus (VTT), the Technical Research Centre of Finland. More recently, an initial station blackout with a 10% break in the main steam line with recovery of power and reflooding of the overheated reactor core with auxiliary feedwater has been analyzed for the TVO plant using the MAAP, MELCOR and SCDAP/RELAP5/MOD3 computer codes [45].

There is substantial MELCOR use and experience at HSK (Hauptabteilung für die Sicherheit der Kernanlagen, the Swiss Federal Nuclear Safety Inspectorate) [46]. The extensive set of plant analyses done for four plants includes a number of accident sequences, sensitivity studies and a MELCOR/MAAP comparison.

MELCOR is being used in the Nuclear Power Engineering Center of the Japan Institute of Nuclear Safety (NUPEC/JINS) as a second generation code for once-through analysis of light water reactor severe accidents, to improve the accuracy of containment event tree analysis and source term analysis in level 2 PSAs for Japanese light water reactors. Preliminary calculations performed using MELCOR 1.8.0 included calculations of two Peach Bottom BWR plant severe accident sequences [47]. More recent calculations done with MELCOR 1.8.1 [48] include PWR [49] and BWR [50] plant sequence analyses in support of PSA studies. The Japanese Atomic Energy Research Institute (JAERI) has done a comparative study of source terms in a BWR severe accident as predicted by THALES-2, the Source Term Code Package (STCP), and MELCOR [51, 52].

MELCOR is being used by a number of groups to model VVER nuclear power plants, even though the code models are not all readily applicable to the VVER design and even though there has been no development of MELCOR for VVER phenomenology. MELCOR is being used in Hungary and in Russia to model a VVER-440/213 reactor and plant [53].

There have been other innovative applications of MELCOR, beyond its original planned uses. A Level 3 PRA was done for N Reactor, a USDOE production reactor, with phenomenological supporting calculations performed with HECTR and MELCOR [54]. MELCOR was used to perform independent safety calculations for two proposed SP-100 space reactors designs [55]; it proved possible to model and analyze simple pressure and temperature excursions for lithium coolant with the existing code. (This successful application to space reactors helps demonstrate the code's worth as a flexible analysis tool.)

1.5 Report Outline

Section 1 summarizes the background for the work documented in this report, including how deterministic codes are used in PRAs, why the MELCOR code is used, what the capabilities and features of MELCOR are, and how the code has been used by others in the past. Section 2 provides a brief description of the Grand Gulf plant and its configuration during LP&S operation. The MELCOR input model developed for the Grand Gulf plant in its LP&S configuration is described in Section 3. Section 4 presents the results of MELCOR analyses of various accident sequences for the POS 5 plant configuration, initiated at several different times after scram and shutdown, including shortened thermal/hydraulic and core damage calculations done in support of the Level 1 analysis and full plant analyses, including containment response and source terms, supporting the Level 2 analysis. MELCOR calculations of various accident scenarios for POS 6 are given in Section 5; these include a reference calculation and sensitivity studies on both plant configuration assumed and on code input options used. Section 6 contains a brief summary of this work.

2 Plant Description

The Grand Gulf Nuclear Station located in southwestern Mississippi is a BWR/6 boiling water reactor with an 800-fuel-assembly core and a Mark III containment.

The containment is divided into two main regions, the drywell and the outer containment, as shown in Figure 2.1. The drywell is a cylindrical region that surrounds the reactor pressure vessel. The outer containment surrounds the drywell and is separated from it by the drywell wall. The two regions are further isolated by an annular suppression pool which is located at the base of containment. The suppression pool is contained between the outer containment wall and a shorter wall in the drywell called the weir wall. Besides leakage through the drywell wall, the only flow path between the two containment regions in a station blackout accident is through horizontal vents that are submerged in the suppression pool.

The weir wall containing the suppression pool is not high enough to prevent backflow of suppression pool water onto the drywell floor when the outer containment pressure exceeds the drywell pressure by a large enough margin to overcome the water head in the weir annulus. The water overflowing onto the drywell floor would then drain into the pedestal region beneath the vessel through floor drains.

The reactor pressure vessel normally vents through safety relief valves (SRVs) into the suppression pool. There are vacuum breakers in the piping between the SRVs and the suppression pool which open to avoid condensation-induced problems in the tailpipes following SRV reclosure. If these vacuum breakers fail to reclose, a portion of the subsequent flow through the SRVs would enter the drywell directly, and the remainder would continue to be discharged to the suppression pool.

The outer containment can be cooled by a spray system, with injection nozzles located in the upper dome. Because this system is ac-powered, it would not be available during a station blackout. However, if ac power was restored during a station blackout, the sprays would become available.

Grand Gulf is equipped with igniters in both the drywell and outer containment to provide controlled burning of hydrogen and carbon monoxide during accidents. Because of this, threats from containment burning are not important for many sequences. However, these igniters are ac-powered, so they would not be available during station blackout sequences.

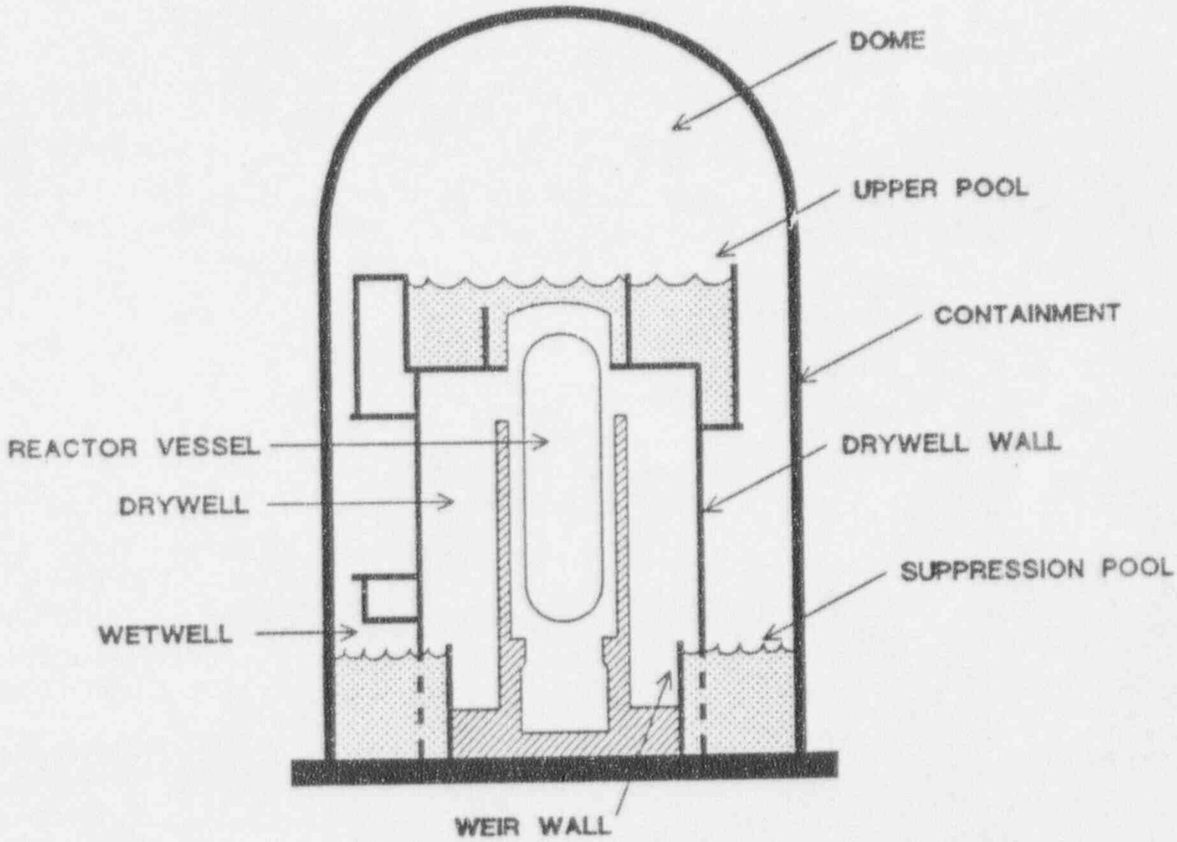


Figure 2.1. Grand Gulf Containment

3 MELCOR Computer Model

The basecase MELCOR input model used for these Grand Gulf shutdown analyses is shown in Figure 3.1. There are a total of 19 control volumes, 36 flow paths, and 57 heat structures in this basecase model; a few control volumes, flow paths and/or heat structures were added to or removed from this model for various analyses, as required. All control volumes were specified to use nonequilibrium thermodynamics and were specified to be vertical volumes; all heat structures used the steady-state temperature-gradient self-initialization option. Detailed volume-altitude tables and junction flow segments were used to correctly represent subcomponents in and between the major components modelled.

The primary system (*i.e.*, the reactor pressure vessel) was represented by six control volumes: one each for the downcomer, lower plenum, upper-plenum/steam-separators, steam dome and the core and bypass channels. The vessel model [56] is depicted in more detail in Figure 3.2, with flow paths and heat structures shown. (The core model is discussed separately later in this section.) The recirculation loop piping was not modelled for these calculations, because it was assumed that circulation within the recirculation piping would not significantly affect the boiloff results.

Previous Grand Gulf calculations [18] used a modified LaSalle core and reactor cooling system. These models, particularly the core model, have been improved to better represent Grand Gulf [4]; these models still contain LaSalle-specific data but the parameters of importance have been converted to or verified as Grand Gulf data to the extent possible given the limited available plant data. For instance, the core model has the proper fuel assembly and control rod masses, and the primary system volumes are in reasonable agreement with the volumes stated in the FSAR [57], but certain flow loss coefficients were not known specifically for Grand Gulf.

For the POS 5 analyses discussed in Section 4, a flow path was added representing the RPV head vent, a piping line extending from the upper head to the pedestal cavity; depending on the sequence being simulated, the RPV head vent flow path was open or closed, and the SRV flow path was locked open, locked closed, or cycled in the relief mode, as required. Flow paths were added for the open MSIV line and for the SDC break as needed for individual POS 5 scenarios. For the POS 6 analyses discussed in Section 5, a flow path was added representing the vessel upper head open to the drywell and the flow path representing the SRVs was set to a zero area. In all cases, a flow path representing the vessel breach provided the thermal/hydraulic outflow when penetrations in the lower head failed, because the COR package only handles ejection of core debris.

Figure 3.3 highlights the MELCOR input model for the containment, taken directly from the MELCOR model used for the NUREG-1150 supporting analyses [18]. The outer containment was represented by five control volumes (dome, equipment hatch, upper annulus and lower annulus, and wetwell) and the inner containment by three (upper drywell, pedestal cavity and weirwall). Flow paths representing the drywell personnel lock and the containment personnel locks and the containment equipment hatch were

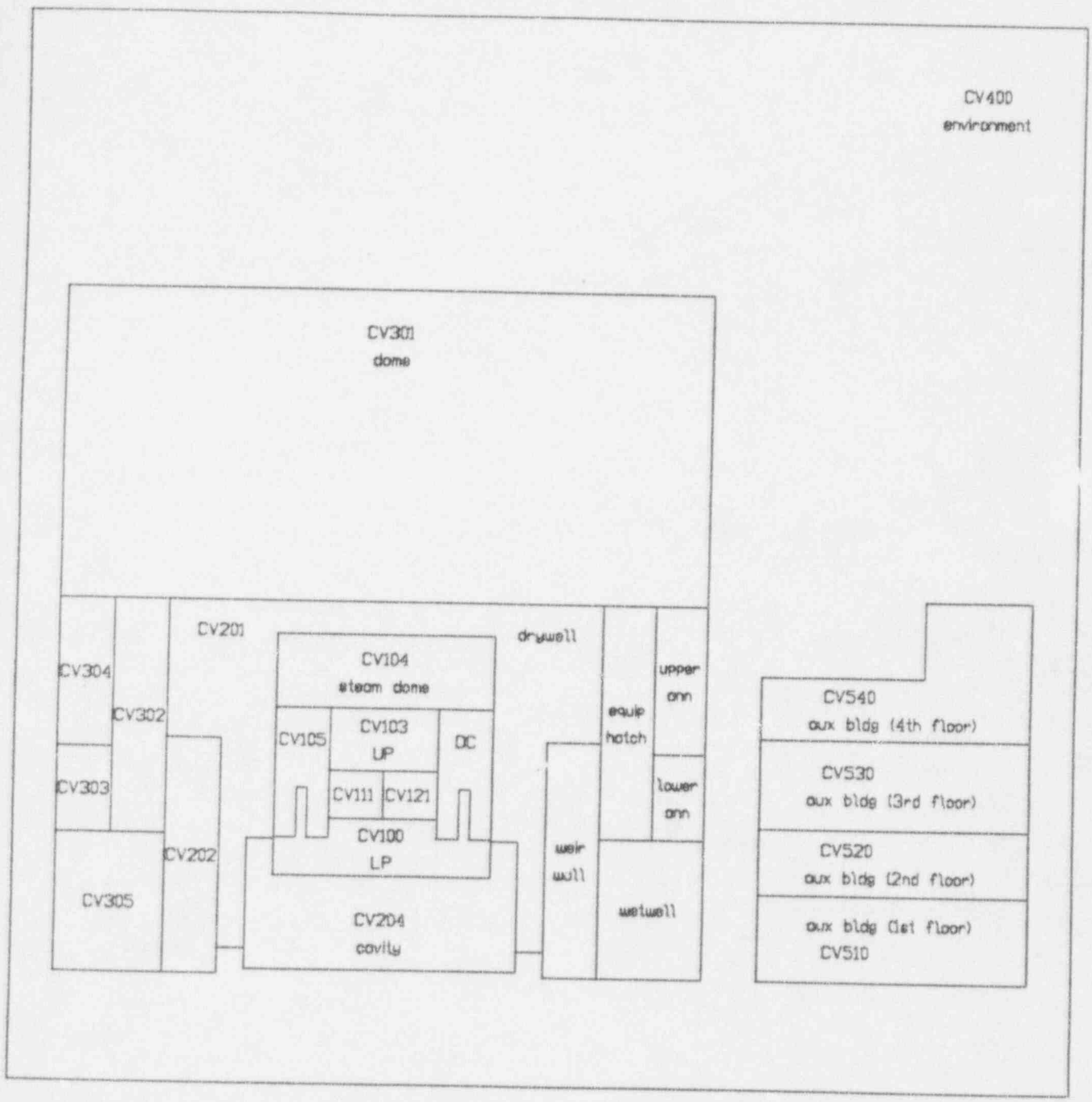


Figure 3.1. Basecase MELCOR Model for Grand Gulf Analyses

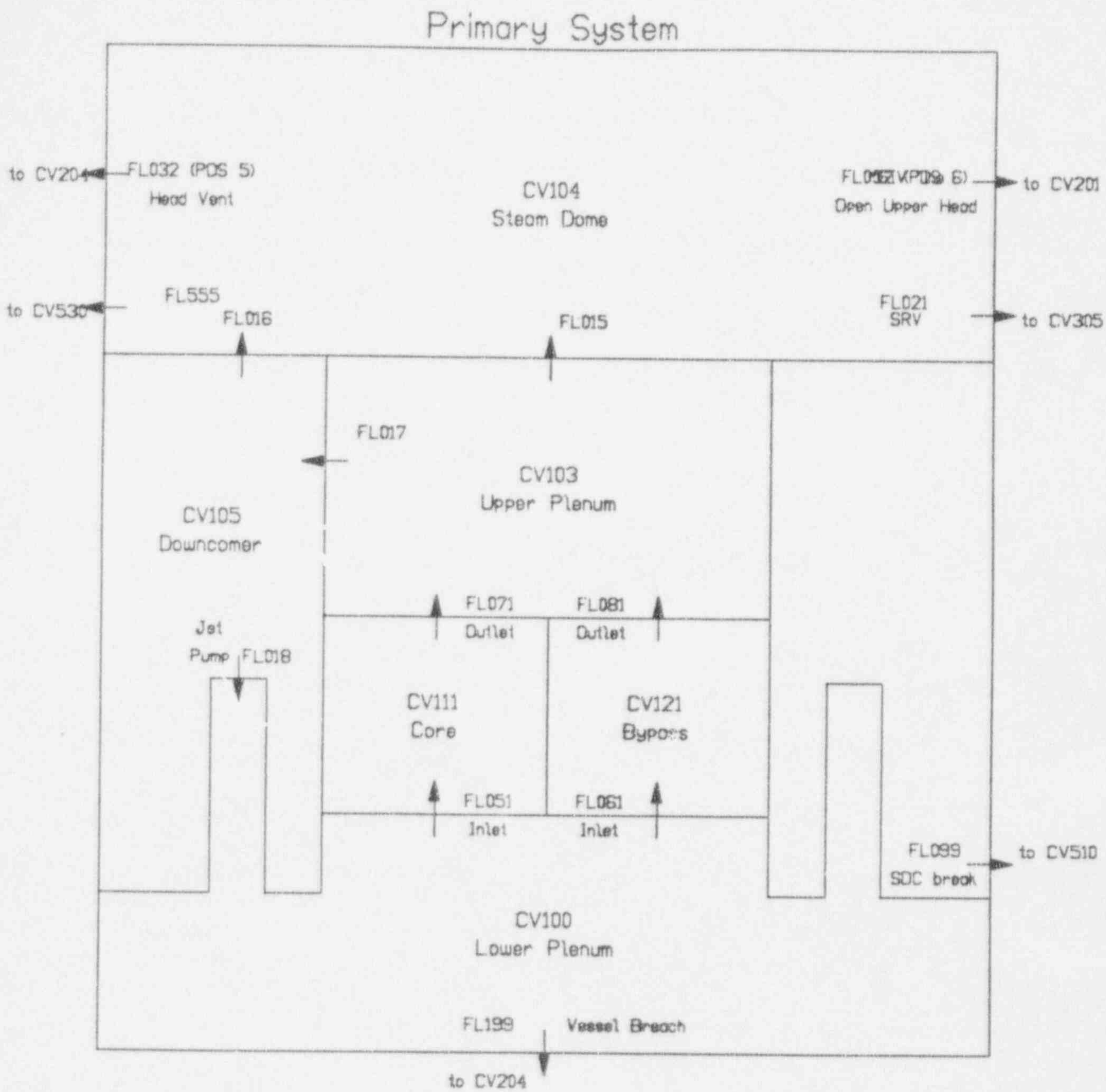


Figure 3.2. MELCOR Model for Grand Gulf Primary System

added. In the POS 5 analyses described in Section 4, the flow path modelling the drywell personnel lock was always fully open, while the flow paths for the containment upper and lower personnel locks and equipment hatch were open or closed as required in particular accident sequences; in the POS 6 analyses described in Section 5, the drywell head was modelled as open, the flow path modelling the containment equipment hatch was always open, while the flow paths for the upper and lower containment personnel locks were sometimes open, and closed in other calculations. Several of the flow paths between volumes in the containment were divided into higher-elevation and lower-elevation flow-path pairs to allow better representation of gas and liquid flows. In some calculations the containment was assumed open to the auxiliary building or directly to the environment; in others, a 489.5 kPa (71 psia) containment failure pressure was used.

The cavity was specified to be a flat-bottomed cylinder with an internal depth and radius of 3.921 m and 3.226 m, respectively; the concrete is 1.752 m thick on the sides and 2.0 m thick below the cavity. The cavity consists of limestone/common sand concrete with 0.135 kg/kg rebar; the ablation temperature is set to 1503 K.

A model for the auxiliary building, depicted in Figure 3.4, was developed specifically for these analyses, primarily from the limited information in the FSAR [57]. Two variations were considered: in both, the auxiliary building model consists of four control volumes (one for each floor), a number of flow paths (three between floors, one from the stairwell to the environment and various inflow paths from containment) and heat structures (five for floors and/or ceilings, four for external walls and four for internal walls), but the volumes and surface areas are changed. The open auxiliary building model represented open interior doors, resulting in larger open volumes and heat structure surface areas for flow-through and potential retention and/or deposition of aerosols before the stairwell door to the environment is blown open at 135.85 kPa (5 psig overpressure). The closed auxiliary building model represented the interior doors remaining closed while the stairwell door to the environment is blown open.

The containment equipment hatch and upper personnel lock open to the fourth floor in the auxiliary building, while the containment lower personnel lock opens to the second floor. For one POS 5 sequence the flow path representing the MSIV line was open, and goes from the upper vessel to the third floor of the auxiliary building. For several other POS 5 scenarios a break in the SDC line is represented, which goes from the vessel downcomer to the first floor of the auxiliary building. The auxiliary building can vent to the environment through a stairway door, taken as coming from the second floor of the auxiliary building.

The basecase core model [56] consists of six radial rings and 13 axial levels, for a total of 78 core cells, as illustrated in Figure 3.5. Axially, five levels are used in the lower plenum, one of which corresponds to the core support plate, and eight levels are used in the core itself. The active fuel region of the core was subdivided into six axial levels of equal height (25in); the lowest and highest levels in the core region contain only support structures, not fuel.

The 800-assembly Grand Gulf core contains a total of 179,760lb of Zr, 98.7lbm in each assembly canister and 126lbm in the fuel rods. In addition, the FSAR gives the total fuel

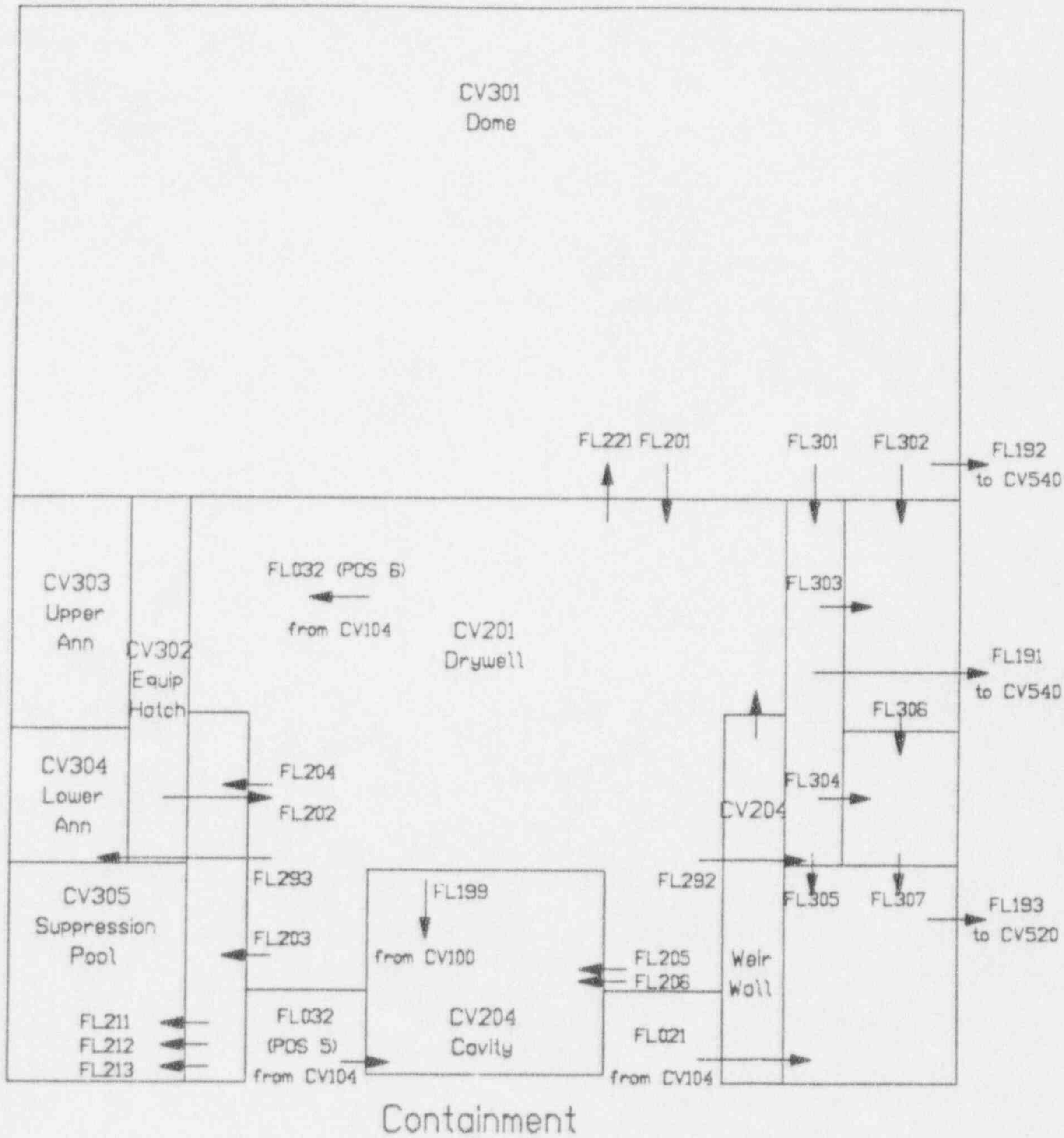


Figure 3.3. MELCOR Model for Grand Gulf Containment

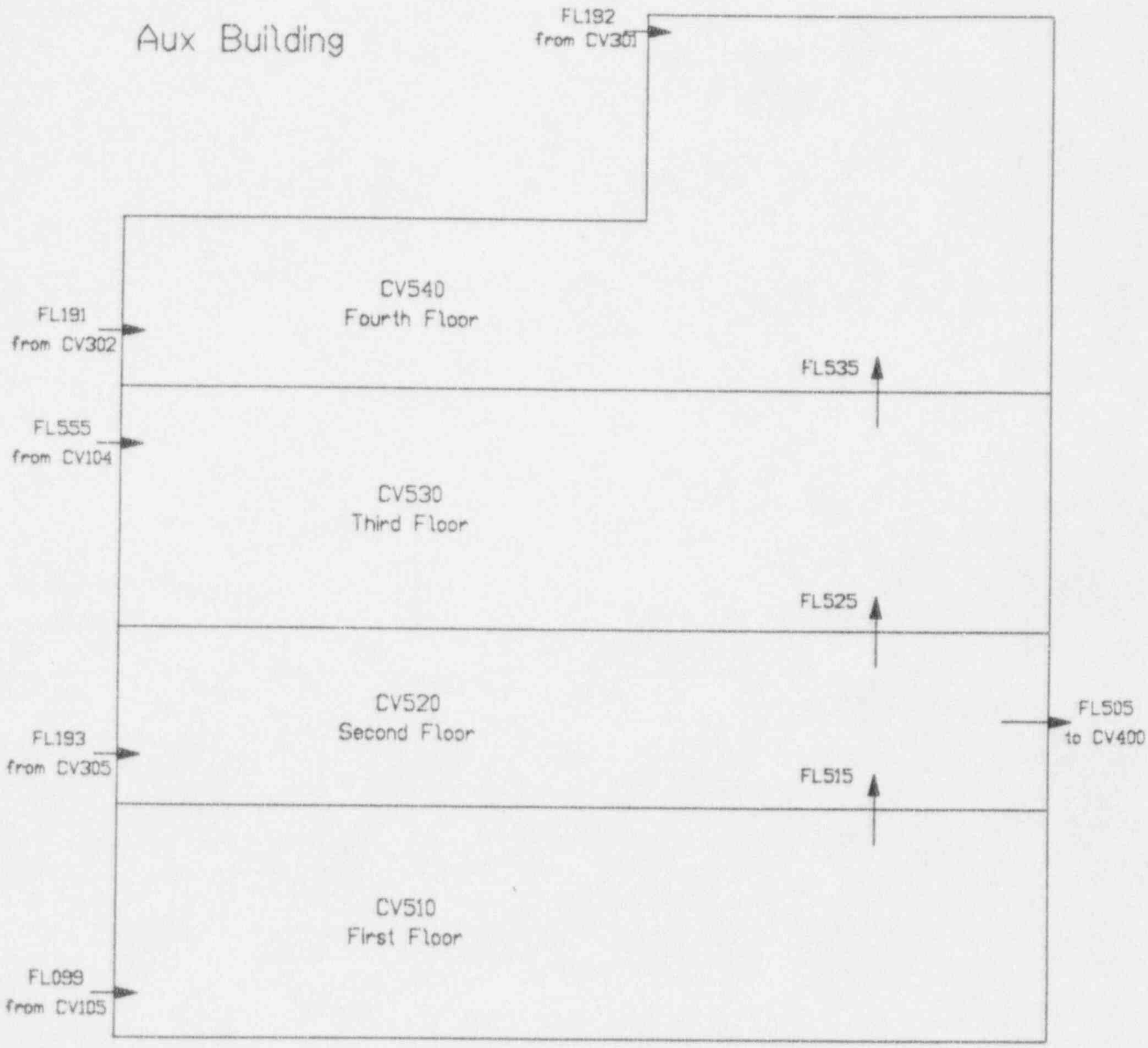


Figure 3.4. MELCOR Model for Grand Gulf Auxiliary Building

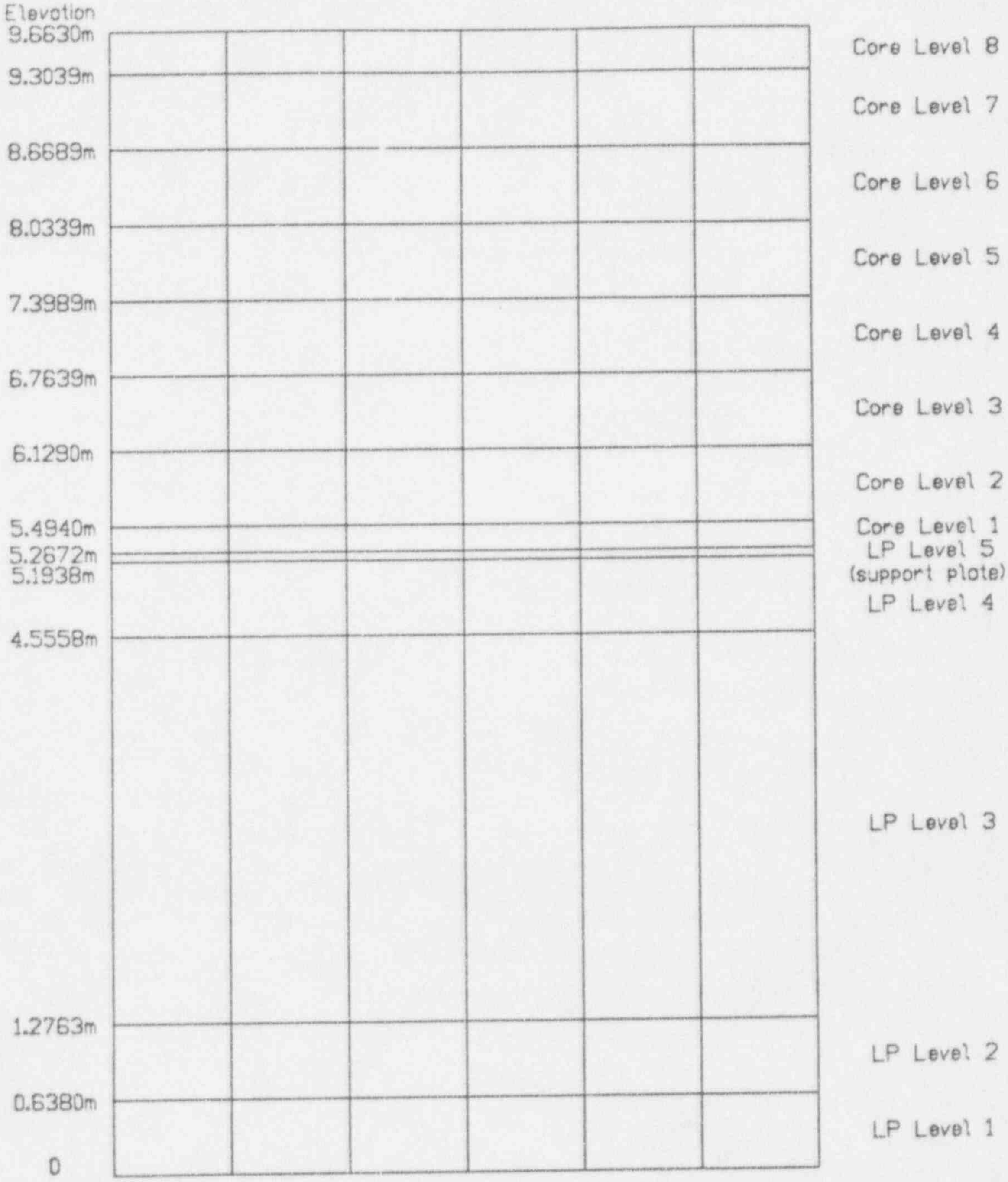


Figure 3.5. Basecase MELCOR COR Model for Grand Gulf

mass as 458lbm/assembly for a total UO_2 mass of 366,400lbm. The total fuel assembly and control masses are given as 699 and 218lb, respectively; there are 193 control rods in the core. The Grand Gulf fuel rods appear to be identical to the LaSalle rods and both have an 8×8 matrix. Grand Gulf, however, has a thicker canister than LaSalle, in addition to 36 more fuel assemblies and 8 more control rods than LaSalle.

LaSalle data was used for the top guide, core plate, fuel supports, control rod tubes and housings masses. These were subdivided into radial and axial cells corresponding to the cells for the power distribution. The subdivided masses are reasonably accurate for the active fueled core region and the correct total masses are maintained. The mass distribution outside of the fueled region (*i.e.*, the handles, the lower tie plate, the fuel support pieces, control rod velocity limiters, *etc.*) were estimated from the available data and drawings.

Other core model input were computed in a similar manner as the masses; these include the component surface areas, the flow areas, cross-sectional areas, and equivalent diameters. Inputs for the vessel lower head and penetrations still reflect the LaSalle data.

The core decay power distribution was developed from FSAR EOC data. Since the radial power distribution dips at the core center, the inner portion of the core was subdivided to focus on the region with the highest power density (the second ring). The time-dependent decay power was calculated using the normalized time-dependent power distribution developed for the LaSalle plant (which is the same power curve used in previous Grand Gulf calculations). The operating power level was 3833 Mw when the reactor was tripped.

The default classes in the MELCOR RN and DCH packages were used. The default classes and initial inventories are presented in Tablern-inv-t0; as shown in this table, a small fraction of these were specified to be in the gap rather than in the fuel. Most of our calculations were done using the MELCOR default fission product release model (*i.e.*, CORSOR-M); section 5.4.1 presents the results of using a POS 6 analysis using the alternative CORSOR release model option. These Grand Gulf shutdown analyses also were done specifying two MAEROS components, one for the noble gases (Class 1) and another for all other aerosols, and five aerosol distribution size bins (the MELCOR default), with the minimum diameter reduced by an order of magnitude from the default value, to $0.1\mu m$.

MELCOR gives radionuclide inventories in terms of both "total" mass and "radioactive" mass. Only the radioactive masses are given in this report. The total and radioactive values can be different for the Cs, Ba, Te, Ru, Mo, Ce, U and Sn classes. For several of these, the difference is due only to the use of a different compound molecular weight for the total than the elemental weight used for the radioactive mass, *i.e.*, CsOH *vs* Cs, TeO *vs* Te, and UO_2 *vs* U. There is no difference in the default elemental and compound molecular weights for the other classes with unequal total and radioactive masses; instead, the differences between total and radioactive masses are due to the inclusion of degraded core structural materials and clad. The platinoids class (Class 6, represented by ruthenium) includes nickel, found in stainless steel; the other major components of

Table 3.1. Initial Radionuclide Class Inventories

Class	Initial Mass (kg)	Initial Gap Inventory (%)
1 (Xe)	463.71	3.0
2 (Cs)	268.35	5.0
3 (Ba)	207.52	0.0001
4 (I)	20.931	1.7
5 (Te)	40.789	0.01
6 (Ru)	306.99	0.0
7 (Mo)	350.64	0.0
8 (Ce)	593.95	0.0
9 (La)	571.05	0.0
10 (U)	132,386	0.0
11 (Cd)	1.4065	0.0
12 (Sn)	8.5872	0.0

stainless steel, iron and chromium, are included in the Mo class. Zircaloy and released as the clad melts. The tetravalent class (Class 8, represented by cerium) includes zirconium, a major clad component; the Sn class includes the tin found normally in Zircaloy and released as the clad melts.

Also note that, while there are 15 default RN classes in MELCOR and those default classes were used for the POS 6 analyses (with CsI added as Class 16 in the POS 5 analyses), no values are given in this report for Class 13 (boron), Class 14 (water) or Class 15 (nonradioactive aerosols generated during core-concrete interaction), all of which have identically zero radioactive masses.

A large number of control functions were used to track the total and radioactive masses of each class released from the intact fuel and/or debris in the vessel (either in the core, the bypass or in the lower plenum); released from the debris in the cavity; remaining in the primary system (*i.e.*, the reactor vessel); in the inner containment (in the drywell and cavity, and the weirwall atmosphere and walls); in the outer containment (in the dome, annulus, equipment hatch, and suppression pool atmosphere and walls); in the water in the suppression pool and weirwall; in the auxiliary building; and in the environment. Those control functions provided time-dependent source term release and distribution data for subsequent postprocessing. Control functions were used also to force edit and restart dumps when specified events occurred (*e.g.*, when the clad first failed, when specified amounts of hydrogen had been generated, when each lower head penetration failed, when the containment and/or auxiliary building failed).

Most of the MELCOR calculations done for the POS 5 Level 1 study (described in Section 4.2 were run with MELCOR 1.8.2 (version 1.8OC) on an IBM/RISC-6000 Model 550 workstation; most of the MELCOR calculations done for the POS 5 Level 2/3 study (described in Section 4.3 were run with MELCOR 1.8.2 (version 1.8OM) on a HP/9000 Model 755 workstation. All MELCOR calculations for the POS 6 study were run with MELCOR 1.8.1 (version 1.8IV) on the IBM/RISC-6000 Model 550 workstation.

4 POS 5 Calculations

4.1 Description of POS 5

POS 5 is rigorously defined as: "Cold Shutdown (Operating Condition 4) and Refueling (Operating Condition 5) only to the point where the vessel head is off." POS 5 can be entered into coming down from power or in going back up to power. During a refueling outage the plant can be in POS 5 for an extended period of time; the event that initiates the accident can occur anytime during this time period. Since the decay heat load from the core decreases with time, the amount of time that is available to the operators to respond to an accident will depend on when the event that initiates the accident occurs during POS 5. inventory also changes with time and, therefore, the radiological potential of the accident will also change with time. dependency on time, the time the plant is in POS 5 is divided into segments or "time windows"; a unique decay heat level is then assigned to each window. To keep the calculations manageable, only three time windows were defined for POS 5. The selection of the time windows was based on the availability of systems used to mitigate the accident and the time required to perform actions necessary to restore systems designed to mitigate the accident. In POS 5, there are two natural time segments, the time the plant is in POS 5 before refueling (*i.e.*, coming down from power) and the time the plant is in POS 5 following refueling (*i.e.*, going backup to power). The decay heats for these two segments will be significantly different. The first segment was further subdivided to account for the availability of an alternate source of decay heat removal. The Alternate Decay Heat Removal System (ADHRS) can be used to remove decay heat from the core once the reactor has been shutdown for at least 24 hours. Thus, the first segment was divided to distinguish the time in POS 5 prior to 24 hours after shutdown from the time in POS 5 after 24 hours after shutdown.

Based on reviews of the refueling outage critiques for RFOs 2, 3, and 4, on average, the plant enters POS 5 14 hours after shutdown and remains in POS 5 for 80 hours before entering POS 6. On the way back up to power, the plant again enters POS 5 40 days after shutdown and remains in POS 5 for 10.4 days. Based on this information, the three time windows were defined as:

Time Window 1: Starts 14 hours after shutdown and has a duration of 10 hours

Time Window 2: Starts 24 hours after shutdown and has a duration of 70 hours

Time Window 3: Starts 40 days after shutdown and has a duration of 10.4 days

Although the plant can enter POS 5 during a refueling outage (RFO) as fast as 7 hours after shutdown, 7 hours was not used as the start time for Window 1 because review of the refueling outage critiques indicated that 14 hours was a more typical value. However, to account for the fact that the plant could enter POS 5 as soon as 7 hours after shutdown, the decay heat load used to represent Window 1 was the decay heat load 7 hours after shutdown. The decay heat used to represent Window 2 is the decay heat load 24 hours after shutdown. Similarly, the decay heat used to represent Window 3 is the decay heat load 40 days after shutdown.

The configuration of the plant during POS 5, as modelled in the Level 2/3 analysis, was determined from requirements imposed by the technical specifications [60] and from plant procedures and practices during a refueling outage (*i.e.*, information was received in the form of critiques of refueling outages and interviews with plant personnel). The technical specifications were used to define the minimum set of requirements. If a system was not required by the technical specifications to be operable, then the plant procedures and practices were reviewed to obtain the status of the system. In actual practice, the configuration of the plant continues to change during POS 5. For example, the containment equipment hatch is removed during this POS. Thus, when the POS is initially entered, the hatch is attached and then it is subsequently removed during the POS changing the configuration of the plant in the process. To keep the analysis manageable, it was often necessary to make simplifying assumptions with regard to the configuration of the plant when the accident was initiated. The configuration of the plant at the start of the accident, as modelled in the Level 2/3 analysis, is defined below.

- Containment: The technical specifications do not require the primary or the secondary containments during POS 5. Review of the Grand Gulf refueling critiques indicated that the containment equipment hatch is typically removed shortly after entering POS 5. In this analysis, it was assumed that the equipment hatch and both personnel locks are open when the accident is initiated. Given that the necessary support systems are available, it was assumed that the containment could be vented in the event that the containment was closed prior to the onset of core damage.

- Drywell Integrity: The technical specifications do not require that the drywell integrity be maintained during POS 5. Review of the Grand Gulf refueling critiques indicated that the drywell personnel lock is open and equipment hatch is typically removed early in POS 5. Furthermore, during POS 5 a portion of the upper reactor pool is drained and the drywell head is removed It was assumed that either the drywell equipment hatch or the drywell personnel locks were open and remained open throughout the accident.

- Reactor Pressure Vessel: In cold shutdown the reactor pressure vessel head is on. While the technical specifications do not require any SRVs to be available, Grand Gulf administrative procedures require at least two SRVs to be available. Therefore, in this analysis it was assumed that two SRVs were available. The temperature of the vessel water is required by the technical specifications to be less than 200°F. The water level can either be at the normal level or the natural circulation level. For the purposes of this analysis, it was assumed that at the start of the accident the reactor water was at the normal level and its temperature was 200°F. The RPV head vent was assumed to be open at the start of the accident. The status of the MSIVs (*i.e.*, open or closed) is accident specific.

- Suppression Pool: The suppression pool inventory is accident specific. Three levels were considered: (1) Low water level (18 ft -4 1/2 in), (2) Drained level 12 ft 8 in. and (3) empty with 170,000 gal available to HPCS from the condensate storage tank.

- Hydrogen Ignition System: The technical specifications do not require the HIS to be available during POS 5. However, since it is the practice at the plant to perform train based maintenance during a refueling outage, and half of the igniters are on Train A and the other half are on Train B, it was assumed in this analysis that at least one train of HIS will always be available (Note, however, the HIS will not operate without ac power).

4.2 Level 1 Thermal/Hydraulic Support Calculations

A series of MELCOR calculations were done to support the quantification of the Level 1 PRA models. For these calculations, the parameters of interest include the times to reach various pressure and/or level setpoints, the time to top-of-active-fuel (TAF) uncovery, the times to core heatup and clad failure (at 1173 K) and the time to vessel failure.

Several general scenarios when the plant is in POS 5 have been considered:

1. Open MSIVs: At the initiation of the accident, the MSIVs on all four steam lines are open. The initiating event then results in a loss of all core cooling and coolant makeup. The SRVs and the reactor pressure vessel head vent are closed at the beginning of the transient.

2. Low Pressure Boiloff: At the initiation of the accident, two SRVs are open. The initiating event then results in a loss of all core cooling and coolant makeup. The reactor pressure vessel head vent is closed at the beginning of the transient.

3. High Pressure Boiloff with Closed RPV Head Vent: At the initiation of the accident, the SRVs are closed. The SRVs remain closed during the accident and only open to relieve pressure at the safety setpoint. The initiating event then results in a loss of all core cooling and coolant makeup. The reactor pressure vessel head vent is closed at the beginning of the transient.

4. High Pressure Boiloff with Open RPV Head Vent: This scenario is identical to case 3, except that the reactor pressure vessel head vent is open.

5. Large Break LOCA: This accident is initiated by a large break LOCA in a 24 in-OD recirculation line. At the start of the accident, the SRVs are closed. The break drains the vessel to 2/3 core height. The initiating event then results in a loss of all core cooling and coolant makeup. The reactor pressure vessel head vent is closed at the beginning of the transient.

6. Station Blackout with Failure to Isolate SDC: The accident is initiated by a loss of offsite power. Following the initiating event, onsite power is lost leading to a SBO and loss of all core cooling and coolant makeup. The operator fails to open the SRVs and steam the core at low pressure (*i.e.*, the SRVs operate in the relief mode). Since the SRVs are closed, the RPV will pressurize. The SBO precludes the isolation of the low pressure piping in the SDC system. This low-pressure SDC system piping fails when the RPV pressure reaches 3.135 MPa (440 psig) resulting in an interfacing systems LOCA.

7. Station Blackout with Firewater Addition: The accident is initiated by a loss of offsite power. Following the initiating event, onsite power is lost leading to a SBO and loss of all core cooling and coolant makeup. The operator opens two SRVs at 2 hr and steams the core at low pressure while adding coolant from the firewater system to the core bypass region. Firewater addition can be maintained indefinitely.

8. Station Blackout with 10 hr Firewater Addition Followed by High Pressure Boiloff: The accident is initiated by a loss of offsite power. Following the initiating event, onsite power is lost leading to a SBO and loss of all core cooling and coolant makeup. The operator opens two SRVs at 2 hr and steams the core at low pressure while adding coolant from the firewater system to the core bypass region. The SRVs are shut at 12 hr after accident initiation, after which they operate in the relief mode. Since the SRVs are now closed, the RPV can pressurize.

9. Station Blackout with 10 hr Firewater Addition Followed by Failure to Isolate SDC: The accident is initiated by a loss of offsite power. Following the initiating event, onsite power is lost leading to a SBO and loss of all core cooling and coolant makeup. The operator opens two SRVs at 2 hr and steams the core at low pressure while adding coolant from the firewater system to the core bypass region. The SRVs are shut at 12 hr after accident initiation, after which they will operate in the relief mode. Since the SRVs are now closed, the RPV will pressurize. The SBO precludes the isolation of the low pressure piping in the SDC system. This low-pressure SDC system piping fails when the RPV pressure reaches 3.135 MPa (440 psig) resulting in an interfacing systems LOCA.

In all cases, at the initiation of the accident, the reactor vessel is depressurized, and the coolant is at the normal level (*i.e.*, 554.7 in actual level or 569.7 in measured level). Also, in all these cases, the drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open (*i.e.*, "open containment").

Calculations were performed for several different times from shutdown for each of these accident scenarios: 7 hr, 24 hr, 59 hr, 12 days, and 40 days. The first two times correspond to the times used to determine the decay heats for the first and second time windows; the third time corresponds to the midpoint of the second time window; the last time corresponds to the time corresponding to the decay heat level in the third time window. (Some calculations were done for 12 days after shutdown while the decay heat table in the MELCOR deck only extended to 1.0×10^6 s after scram; after the decay heat table was extended to $\geq$50 days, calculations were done starting 40 days after shutdown.)

Because the primary interest was in time to core damage, these Level 1 support calculations were run until any of the following: vessel failure, code abort or 24 hr of transient. If any sequence produced no significant core damage within 24 hr for a given decay heat level, no further calculations were done with longer shutdown time s (*i.e.*, lower decay heat levels).

4.2.1 Open MSIVs

At the initiation of the accident, the reactor vessel is depressurized, the coolant is at the normal level and the MSIVs on all four steam lines are open. The vessel water inventory is at 366.5 K (200°F), which corresponds to the maximum temperature allowed by the Grand Gulf technical specifications for operation in POS 5. The initiating event then results in a loss of all core cooling and coolant makeup. The SRVs and the reactor pressure vessel head vent are closed at the beginning of the transient. The drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open (*i.e.*, "open containment").

Figure 4.2.1.1 gives the vessel pressures calculated starting this accident scenario at several different times after scram. In all cases, the system begins pressurizing as all core cooling is lost but only pressurizes to ~150kPa before the steam flow out the open MSIVs is sufficient to remove all the decay heat. The steam flow out the MSIVs in turn pressurizes the auxiliary building and, through the open equipment hatch and personnel locks, pressurizes the containment, as shown in Figure 4.2.1.2. The auxiliary building is assumed to fail on a 0.345 kPa (5 psig) overpressure. The longer after shutdown and scram that this accident sequence begins, the lower the decay heat and the longer it takes to fail the auxiliary building.

The coolant inventory in the vessel drops as the decay heat boils water to steam which is lost out the open MSIVs, faster for higher decay heat levels than for lower decay heat levels, as presented in Figure 4.2.1.3. Figure 4.2.1.4 gives the predicted upper plenum liquid level drop due to this inventory loss, for different decay heat levels and highlighting when a Level 3 trip (544.4 in) would be generated; this is the autoisolation signal for SDC. Figure 4.2.1.5 gives the upper plenum and corresponding core liquid level drops due to this inventory loss, for different decay heat levels and highlighting when TAF uncovery is calculated to occur; horizontal lines indicate both the boundary between the upper plenum and the core at 9.6 m and the top-of-active-fuel elevation at 9.3 m. The core uncovery begins when the upper plenum still has substantial liquid left, with liquid downflow restricted by countercurrent flow limiting by upflow of the steam being generated in the core, but the two-phase level in the core does not drop substantially below the top of the active fuel until after the upper plenum is mostly drained. We take TAF uncovery as the drop of the collapsed level in the core below the TAF elevation.

The early core heatup is illustrated in Figures 4.2.1.6 through 4.2.1.8, as calculated for accident sequences initiated by stuck-open MSIVs at 7 hr, 24 hr and 59 hr after shutdown.

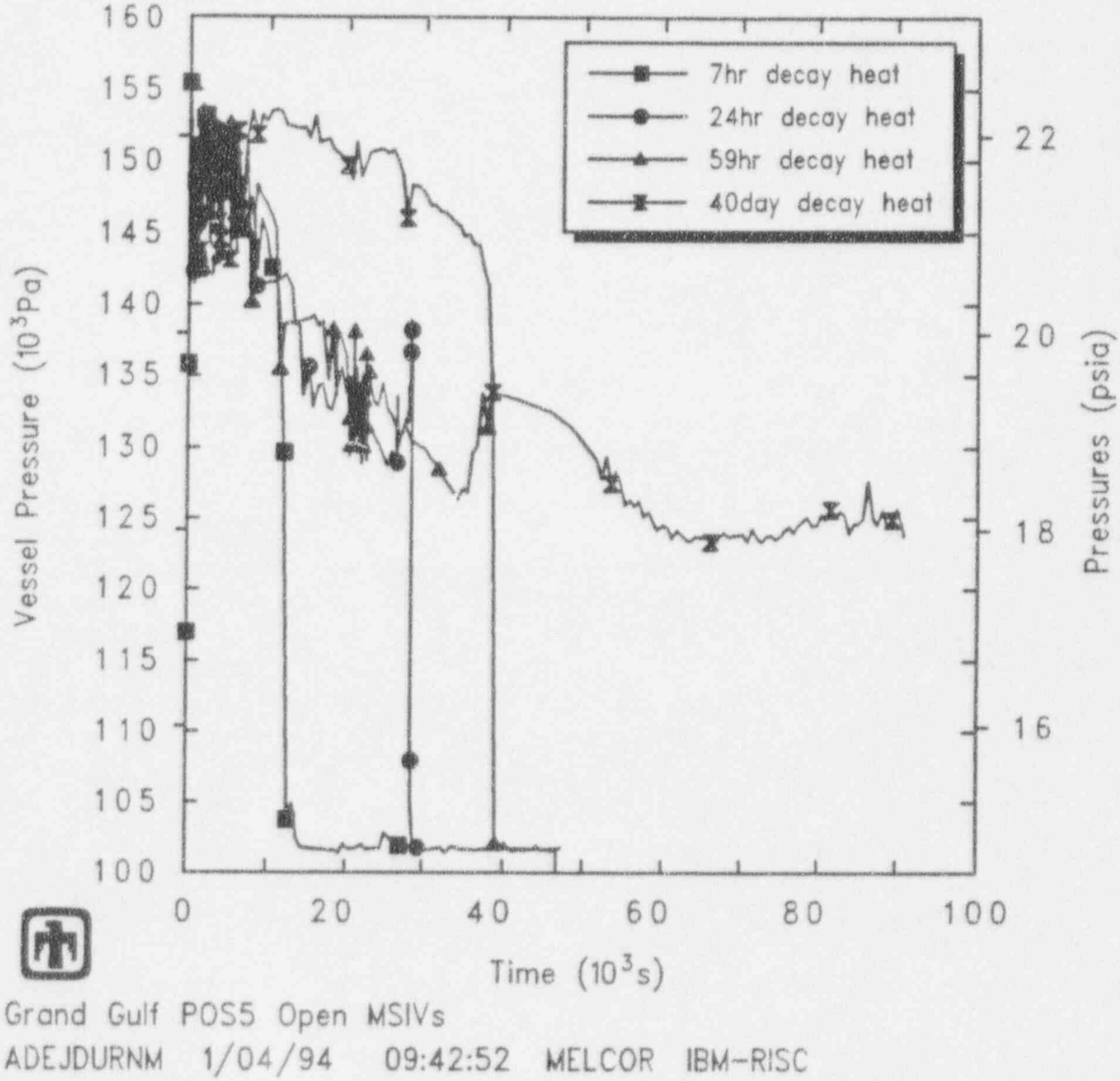


Figure 4.2.1.1. Reactor Vessel Pressures for Grand Gulf POS 5 - Open MSIVs, Initiated at Various Times After Shutdown

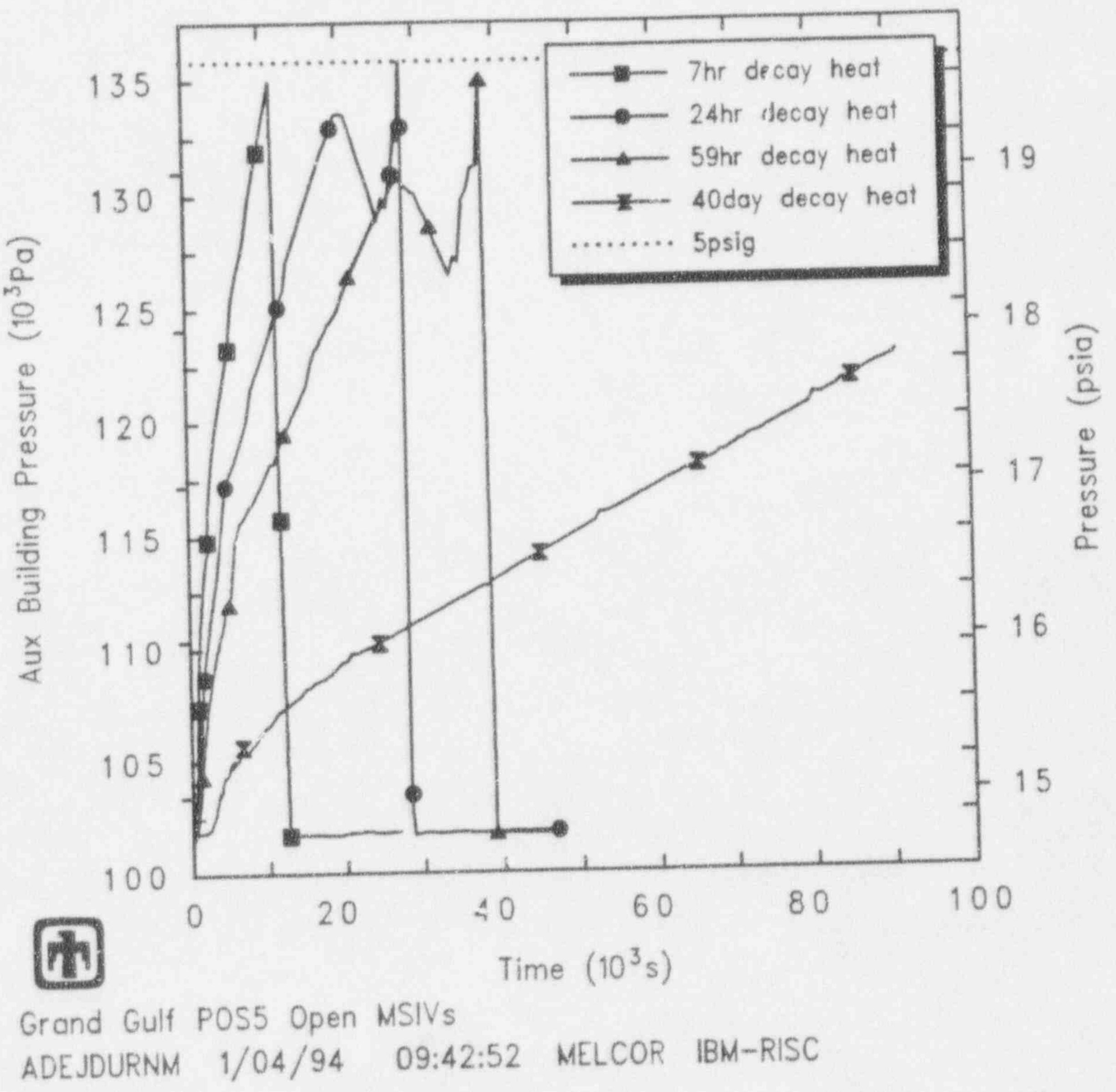


Figure 4.2.1.2. Auxiliary Building Pressures for Grand Gulf POS 5 - Open MSIVs, Initiated at Various Times After Shutdown

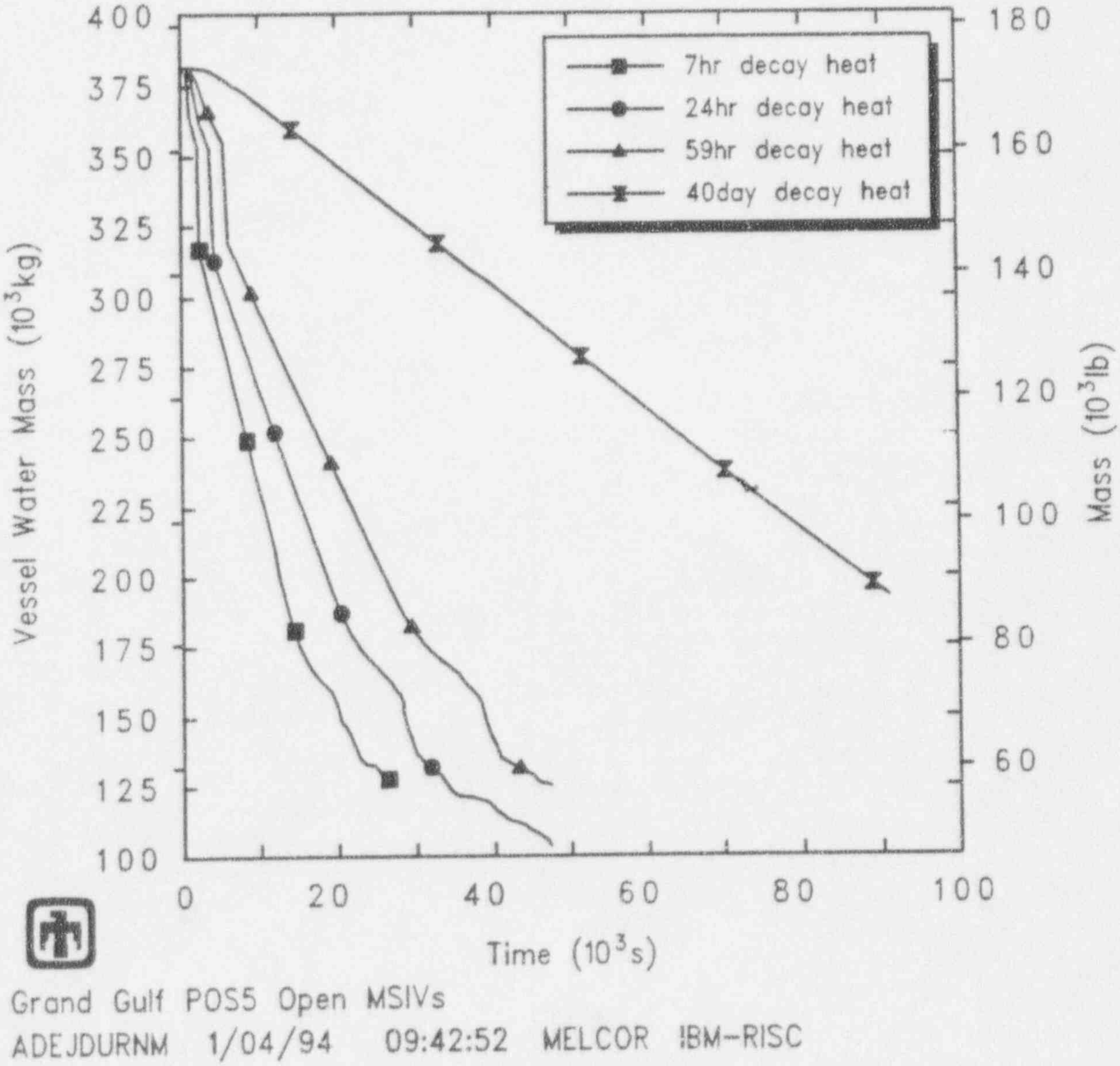


Figure 4.2.1.3. Reactor Vessel Water Masses for Grand Gulf POS 5 - Open MSIVs, Initiated at Various Times After Shutdown

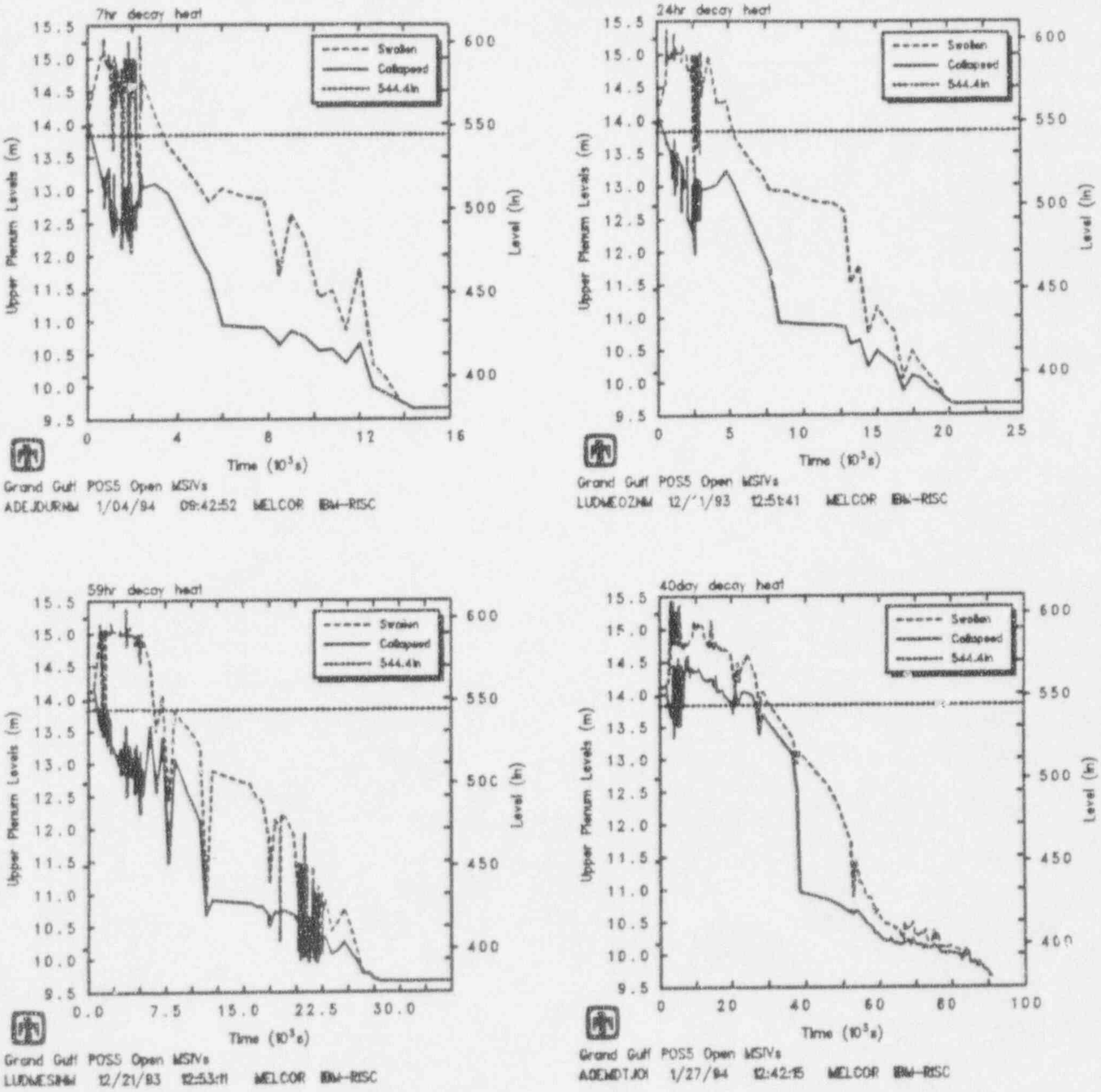


Figure 4.2.1.4. Upper Plenum Liquid Levels for Grand Gulf POS 5 - Open MSIVs, Initiated 7 hr (upper left), 24 hr (upper right), 59 hr (lower left) and 40 day (lower right) After Shutdown

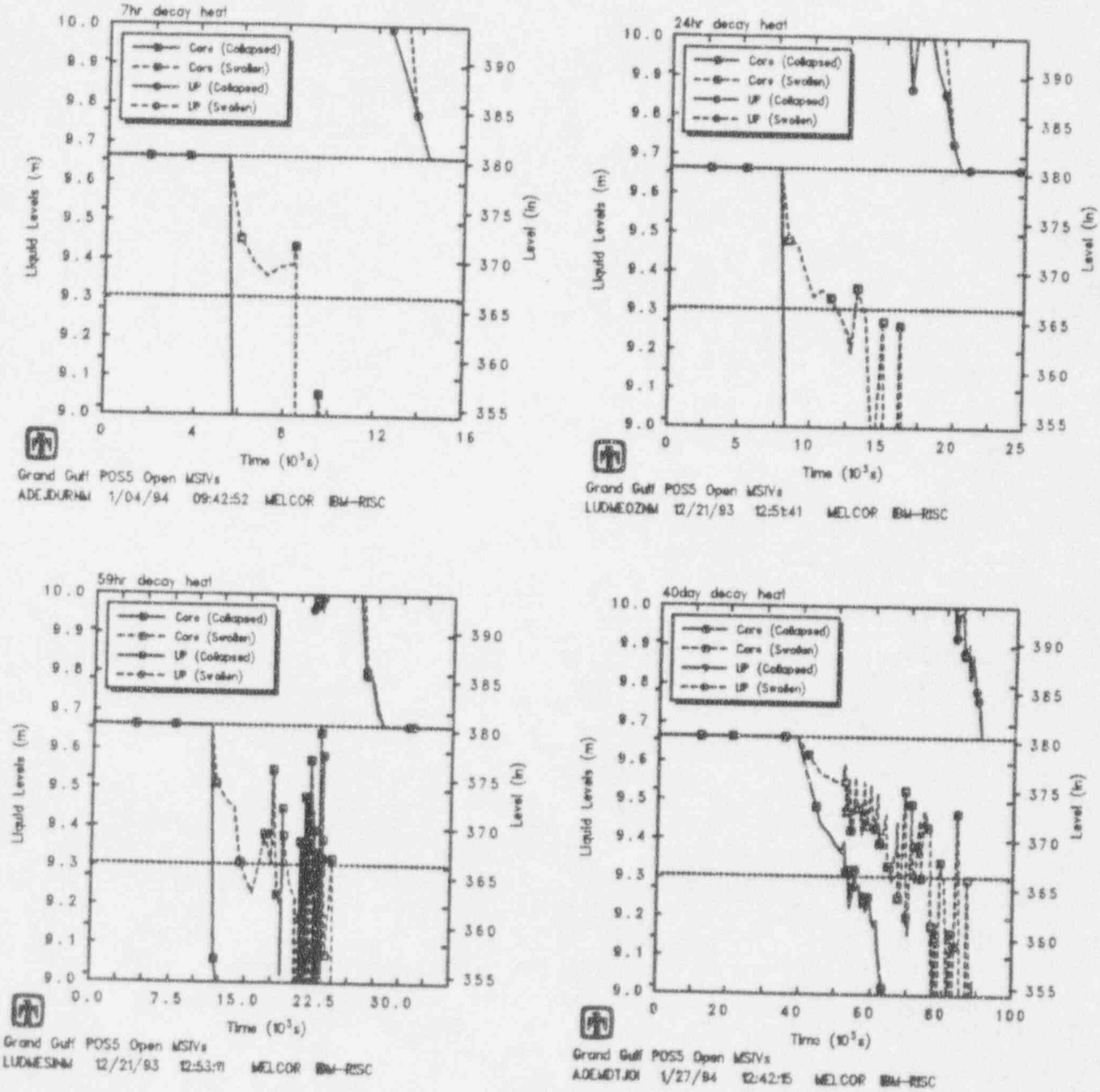


Figure 4.2.1.5. Core Liquid Levels for Grand Gulf POS 5 - Open MSIVs, Initiated 7 hr (upper left), 24 hr (upper right), 59 hr (lower left) and 40 day (lower right) After Shutdown

As with TAF uncovery, core uncovery begins sooner and proceeds more rapidly at higher decay heat levels. (The calculation begun 40 days after shutdown showed no core heatup by about 90,000 s, when stopped.)

Tables 4.2.1.1 and 4.2.1.2 summarize the timings of various key events predicted using MELCOR for this sequence assuming various times after shutdown and associated decay heat levels.

4.2.2 Low Pressure Boiloff

At the initiation of the accident, the reactor vessel is depressurized, the coolant is at the normal level and two SRVs are open. The vessel water inventory is at 366.5 K (200°F), which corresponds to the maximum temperature allowed by the Grand Gulf technical specifications for operation in POS 5. The initiating event then results in a loss of all core cooling and coolant makeup. The reactor pressure vessel head vent is closed at the beginning of the transient. The drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open (*i.e.*, "open containment").

Figure 4.2.2.1 gives the vessel pressures calculated starting this accident scenario at several different times after scram. In all cases, the system begins pressurizing as all core cooling is lost but only pressurizes slightly before the steam flow out the two open SRVs is sufficient to remove all the decay heat; the higher the decay heat (*i.e.*, the sooner after shutdown), the higher the early-time pressure peak before the flow out the open SRVs can fully remove the decay heat.

The steam flow out the two open SRVs in turn pressurizes the containment and, through the open equipment hatch and personnel locks, pressurizes the auxiliary building, as shown in Figure 4.2.2.2. The longer after shutdown and scram that this accident sequence begins, the lower the decay heat and the longer it takes to fail the auxiliary building.

The coolant inventory in the vessel drops as the decay heat boils water to steam which is lost out the open SRVs, faster for higher decay heat levels than for lower decay heat levels, as presented in Figure 4.2.2.3. Figure 4.2.2.4 gives the predicted upper plenum liquid level drop due to this inventory loss, for different decay heat levels and highlighting when a Level 3 trip (544.4 in) would be generated. Both collapsed and swollen (two-phase) liquid levels in the upper plenum are quite oscillatory, and we chose the first time the collapsed level crossed the 544.4 in level setpoint as the signal generation.

Figure 4.2.2.5 gives the corresponding core liquid level drop due to this inventory loss, for different decay heat levels and highlighting when TAF uncovery is calculated to occur; horizontal lines are included both at the top of the core (9.6 m) and at the TAF elevation (9.3 m). The collapsed liquid level in the upper core generally drops rapidly and smoothly; the swollen liquid level in the upper core in contrast oscillates substantially. The core uncovery begins when the upper plenum still has substantial liquid left, with liquid downflow restricted by countercurrent flow limiting by upflow of the steam being

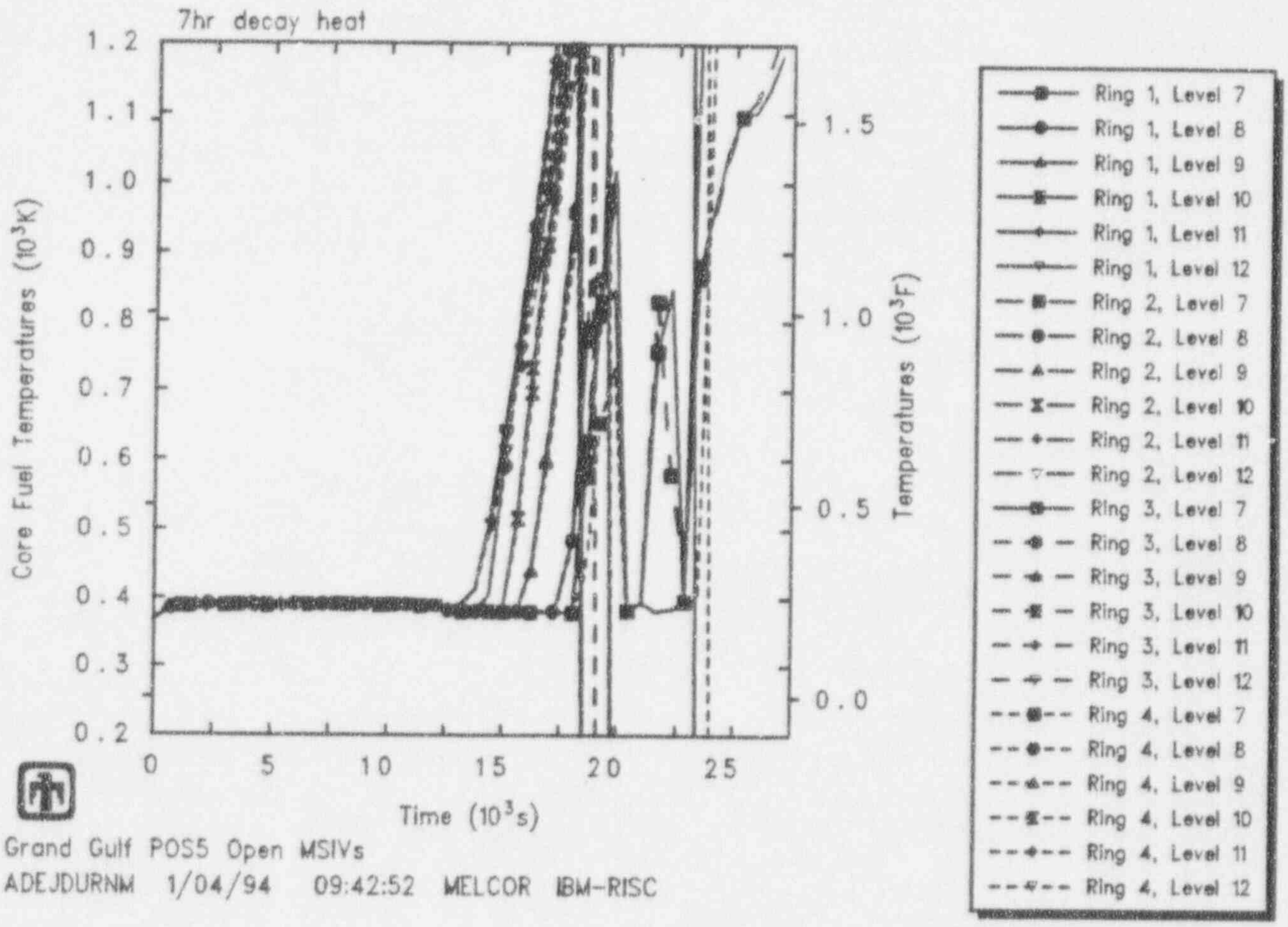


Figure 4.2.1.6. Core Fuel Temperatures for Grand Gulf POS 5 - Open MSIVs, Initiated 7 hr After Shutdown

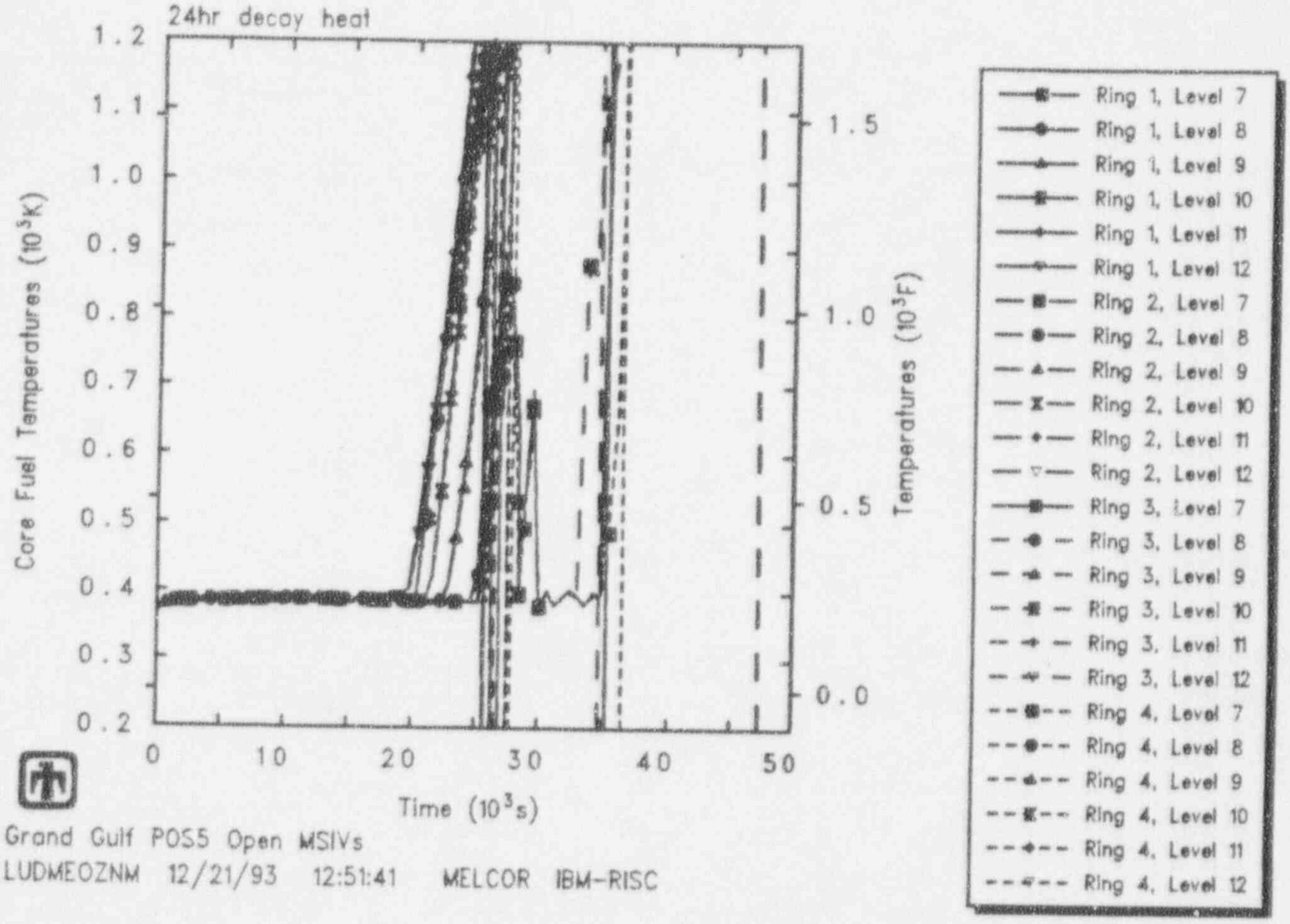


Figure 4.2.1.7. Core Fuel Temperatures for Grand Gulf POS 5 - Open MSIVs, Initiated 24 hr After Shutdown

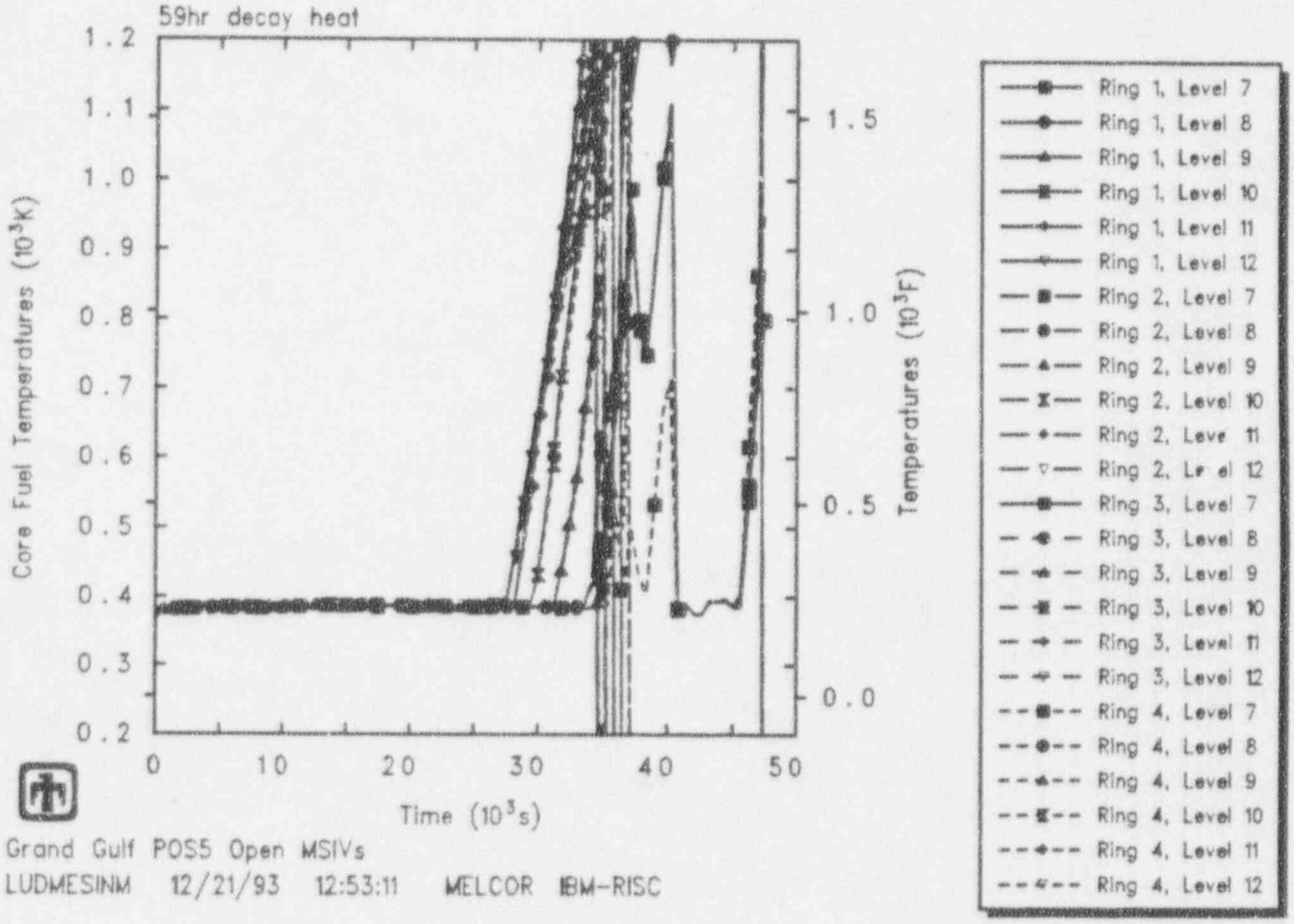


Figure 4.2.1.8. Core Fuel Temperatures for Grand Gulf POS 5 - Open MSIVs, Initiated 59 hr After Shutdown

Table 4.2.1.1. Key Event Times for Grand Gulf POS 5 – Open MSIVs, Initiated at Various Times After Shutdown

	Time to (s)			
Initiation Time After Shutdown	TAF Uncovery†	Core Heatup	First Gap Release	Vessel Failure
7 hr	5,500	13,000	17,000	–‡
24 hr	8,000	20,000	24,100	–‡
59 hr	10,500	27,000	33,200	–‡
12 day				
40 day	54,000	90,000	–‡	–‡

†Collapsed liquid level
‡Calculation stopped before event occurred

Table 4.2.1.2. Key Signal Times for Grand Gulf POS 5 – Open MSIVs, Initiated at Various Times After Shutdown

	Time to (s)	
Initiation Time After Shutdown	Collapsed Level <544.4 in	Swollen Level <544.4 in
7 hr	200	3,500
24 hr	500	5,250
59 hr	1000	6,500
12 day	2500	
40 day	3600	

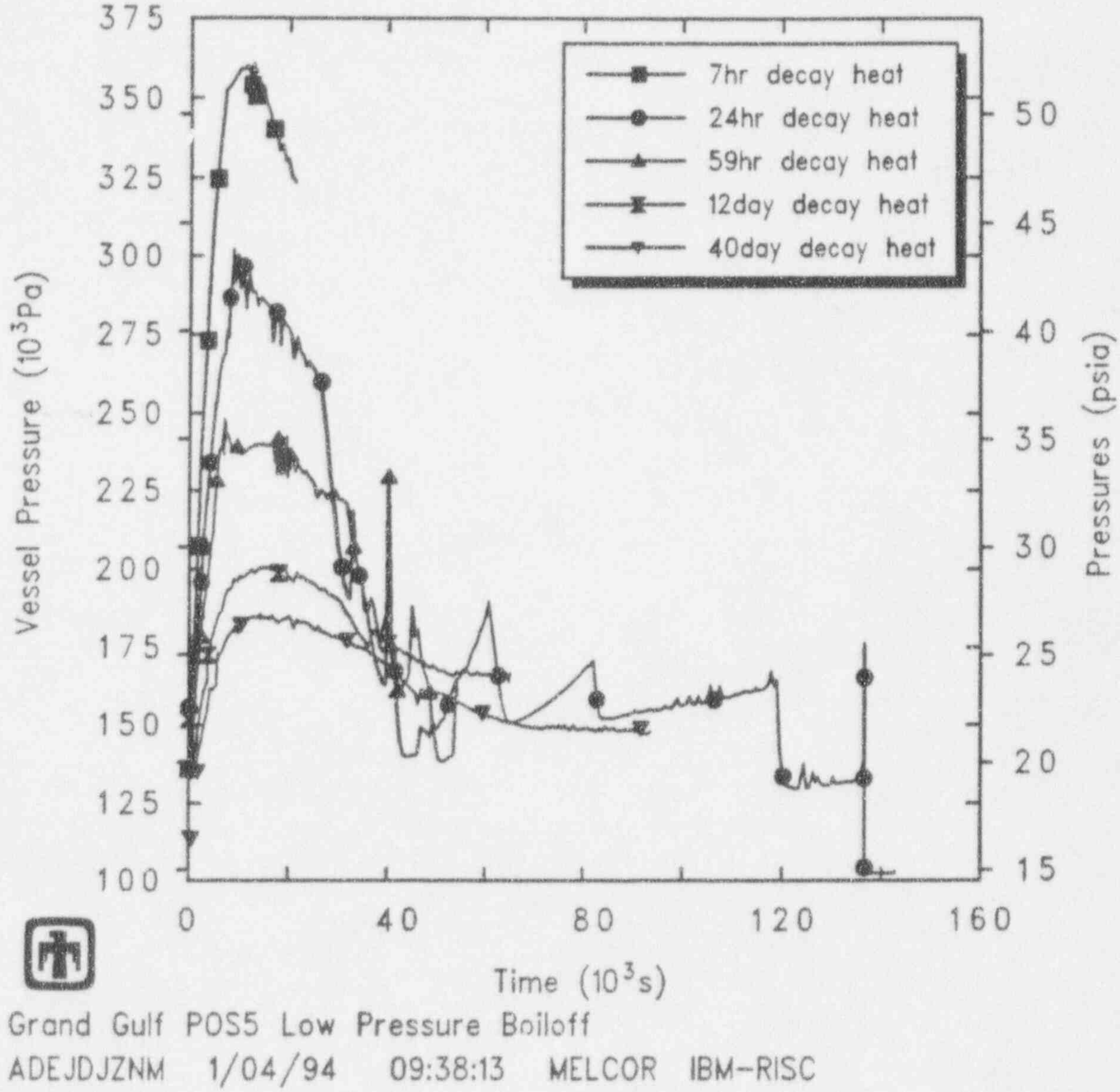


Figure 4.2.2.1. Reactor Vessel Pressures for Grand Gulf POS 5 - Low Pressure Boiloff, Initiated at Various Times After Shutdown

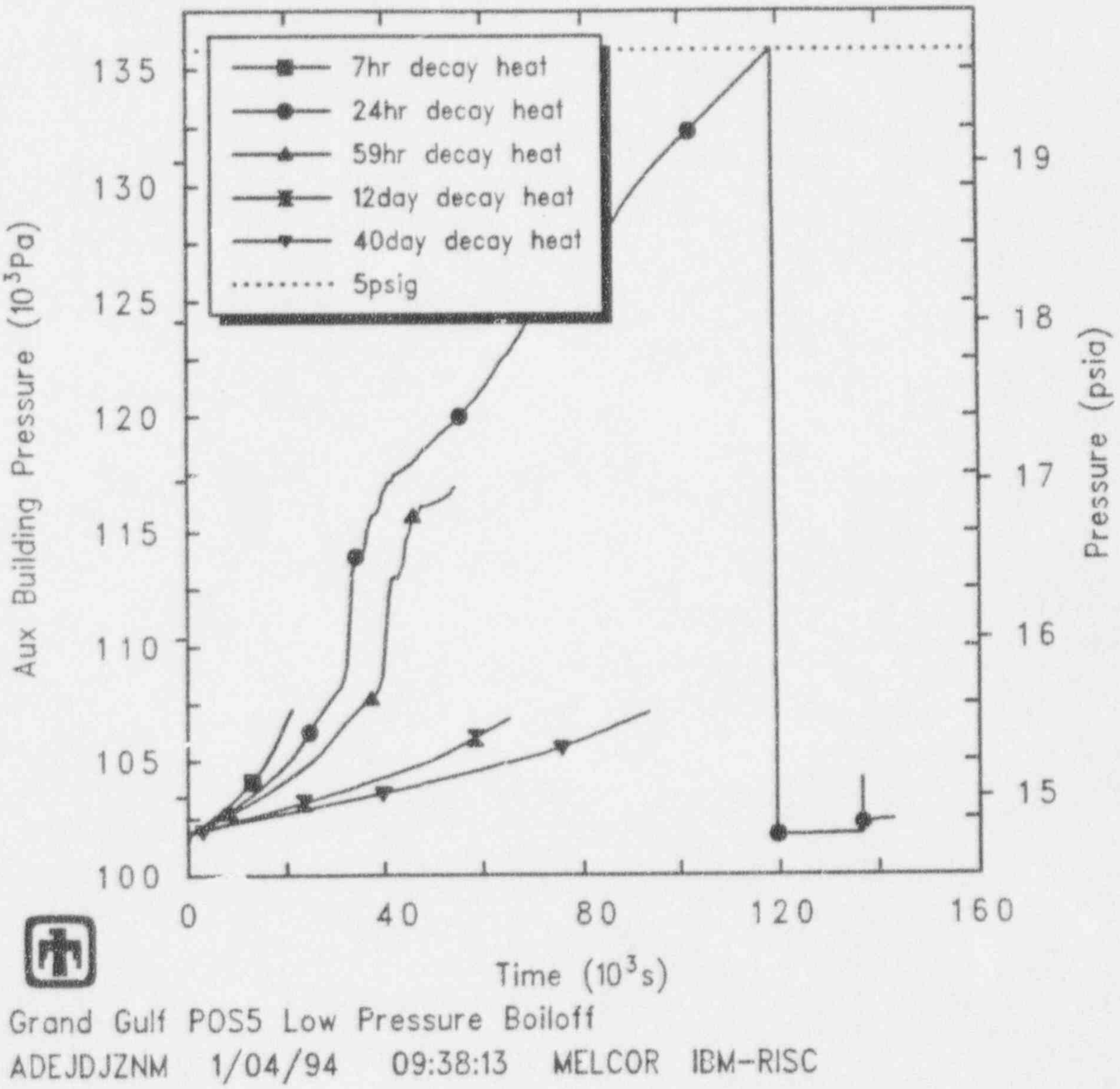


Figure 4.2.2.2 Auxiliary Building Pressures for Grand Gulf POS 5 - Low Pressure Boiloff, Initiated at Various Times After Shutdown

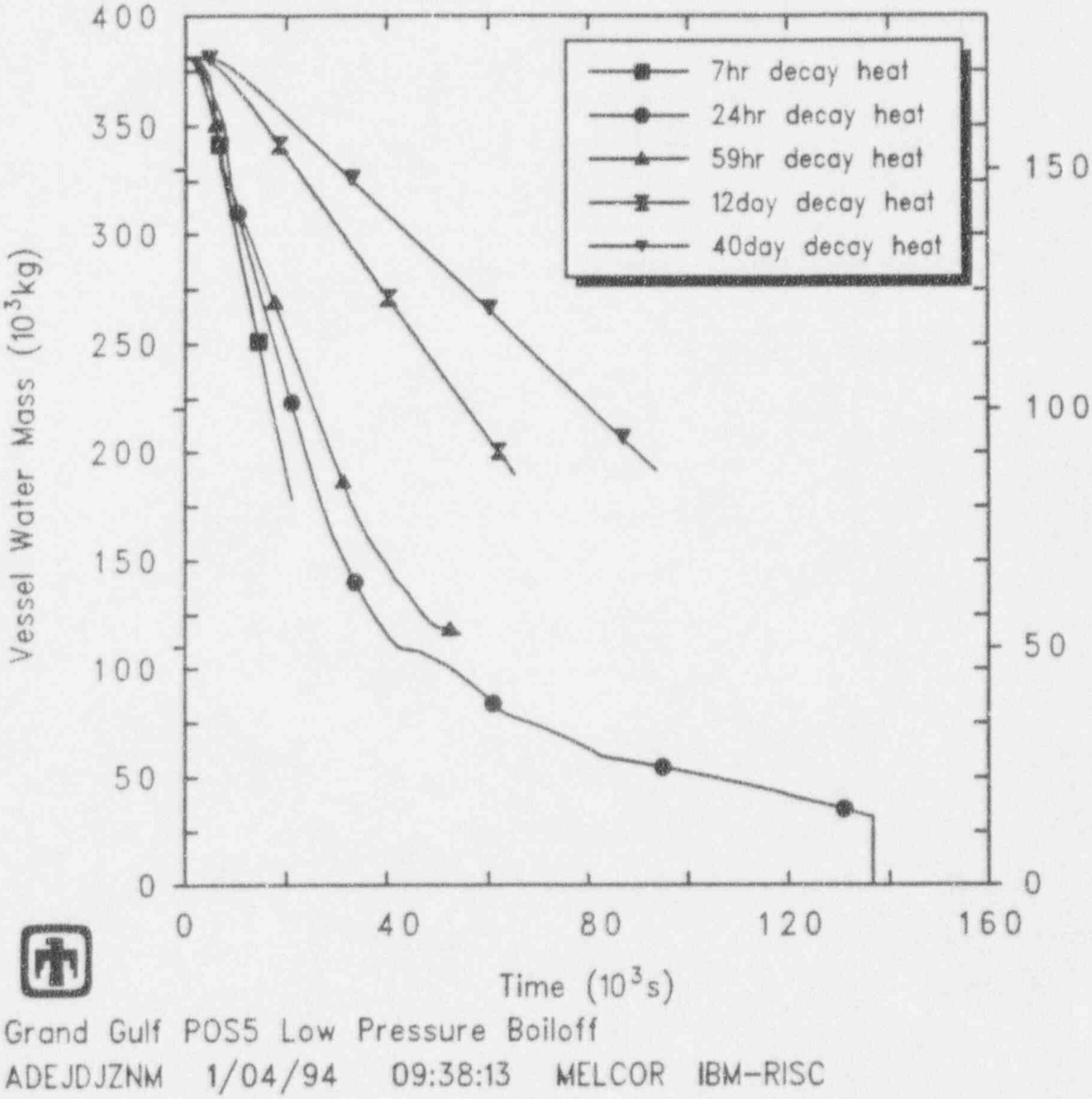


Figure 4.2.2.3. Reactor Vessel Water Masses for Grand Gulf POS 5 - Low Pressure Boiloff, Initiated at Various Times After Shutdown

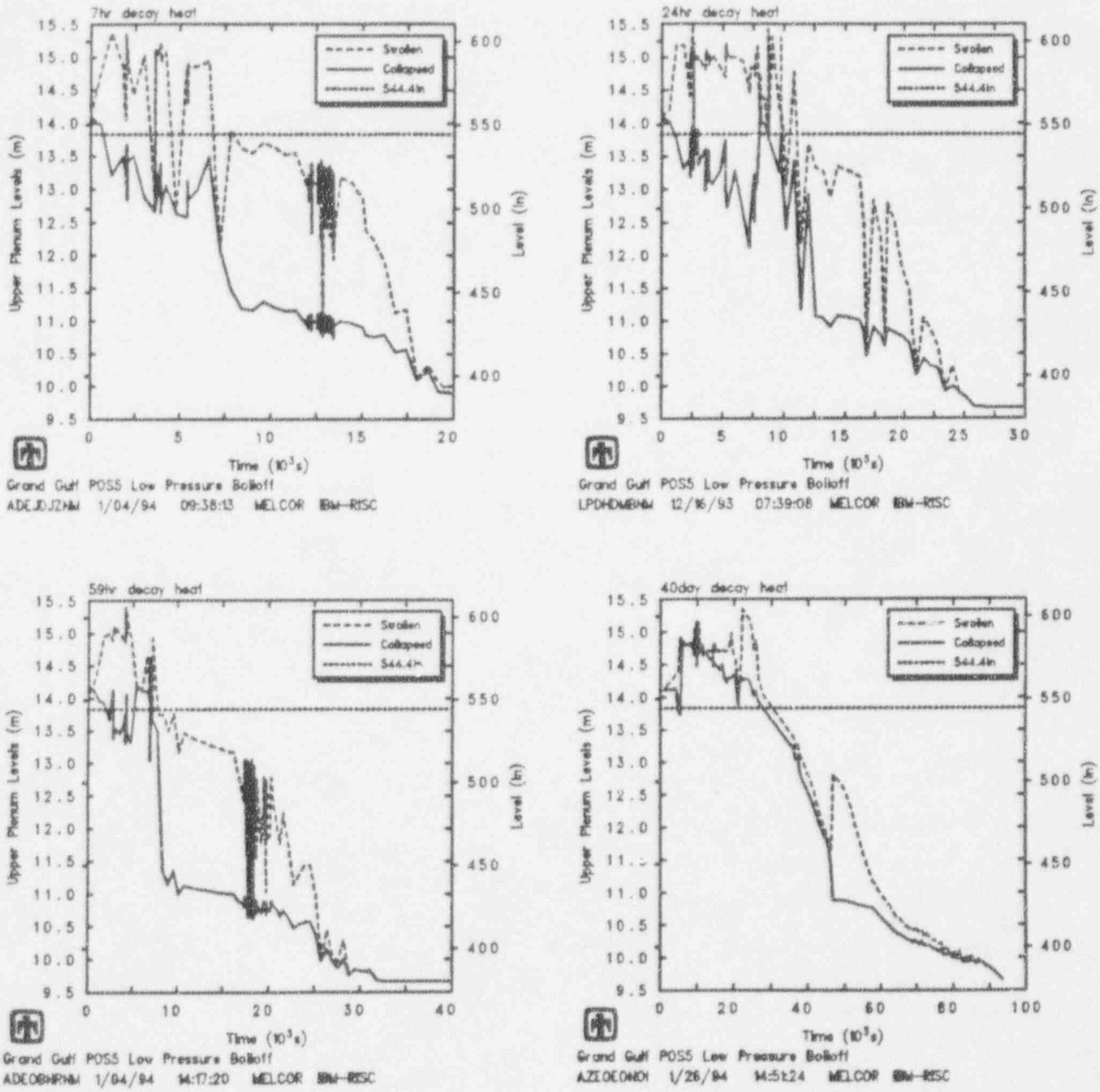


Figure 4.2.2.4. Upper Plenum Liquid Levels for Grand Gulf POS 5 – Low Pressure Boiloff, Initiated 7 hr (upper left), 24 hr (upper right), 59 hr (lower left) and 40 day (lower right) After Shutdown

generated in the core, but the two-phase level in the core does not drop substantially below the top of the active fuel until after the upper plenum is mostly drained. We take TAF uncovery as the drop of the collapsed level in the core below the TAF elevation.

The early core heatup is illustrated in Figures 4.2.2.6 through 4.2.2.8, as calculated for low-pressure boiloffs starting at 7 hr, 24 hr and 59 hr after shutdown. As with TAF uncovery, core uncovery begins sooner and proceeds more rapidly at higher decay heat levels. The calculation begun 12 days after shutdown showed core heatup just beginning by about 63,000 s, when stopped; the calculation begun 40 days after shutdown showed no core heatup by 90,000 s, when stopped. (Recall that the period of interest for all these Level 1 analyses is either from accident initiation to core heatup, or 1 day after accident start.)

Tables 4.2.2.1 and 4.2.2.2 summarize the timings of various key events predicted using MELCOR for this sequence assuming various times after shutdown and associated decay heat levels.

4.2.3 High Pressure Boiloff with Closed RPV Head Vent

At the initiation of the accident, the reactor vessel is depressurized, the coolant is at the normal level and the SRVs are closed. The SRVs remain closed during the accident and only open to relieve pressure at the safety setpoint. The vessel water inventory is at 366.5 K (200°F), which corresponds to the maximum temperature allowed by the Grand Gulf technical specifications for operation in POS 5. The initiating event then results in a loss of all core cooling and coolant makeup. The reactor pressure vessel head vent is closed at the beginning of the transient. The drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open.

(A calculation beginning 40 days after shutdown was not done for this sequence because the results of the analysis beginning 12 days after shutdown showed no significant core uncovery or damage within the 1 day maximum time window of interest.)

Figure 4.2.3.1 gives the vessel pressures calculated starting this accident scenario at several different times after scram. In all cases, the system begins pressurizing as all core cooling is lost and continues pressurizing, with no relief, until reaching the SRV setpoint. The SRVs then cycle around the valve setpoints, intermittently opening and allowing the steam flow out the SRVs to remove the decay heat. The higher the decay heat (*i.e.*, the sooner after shutdown), the faster the initial pressurization and associated inventory loss, and the earlier the vessel fails.

The steam flow out the SRVs in turn pressurizes the containment and, through the open equipment hatch and personnel locks, pressurizes the auxiliary building, as shown in Figure 4.2.3.2. The longer after shutdown and scram that this accident sequence begins, the lower the decay heat and the slower the auxiliary building pressurizes. In all these cases, the auxiliary building does not reach its 5 psig overpressure failure setpoint before vessel failure; the auxiliary building fails on a sudden pressure spike corresponding to vessel failure and debris ejection.

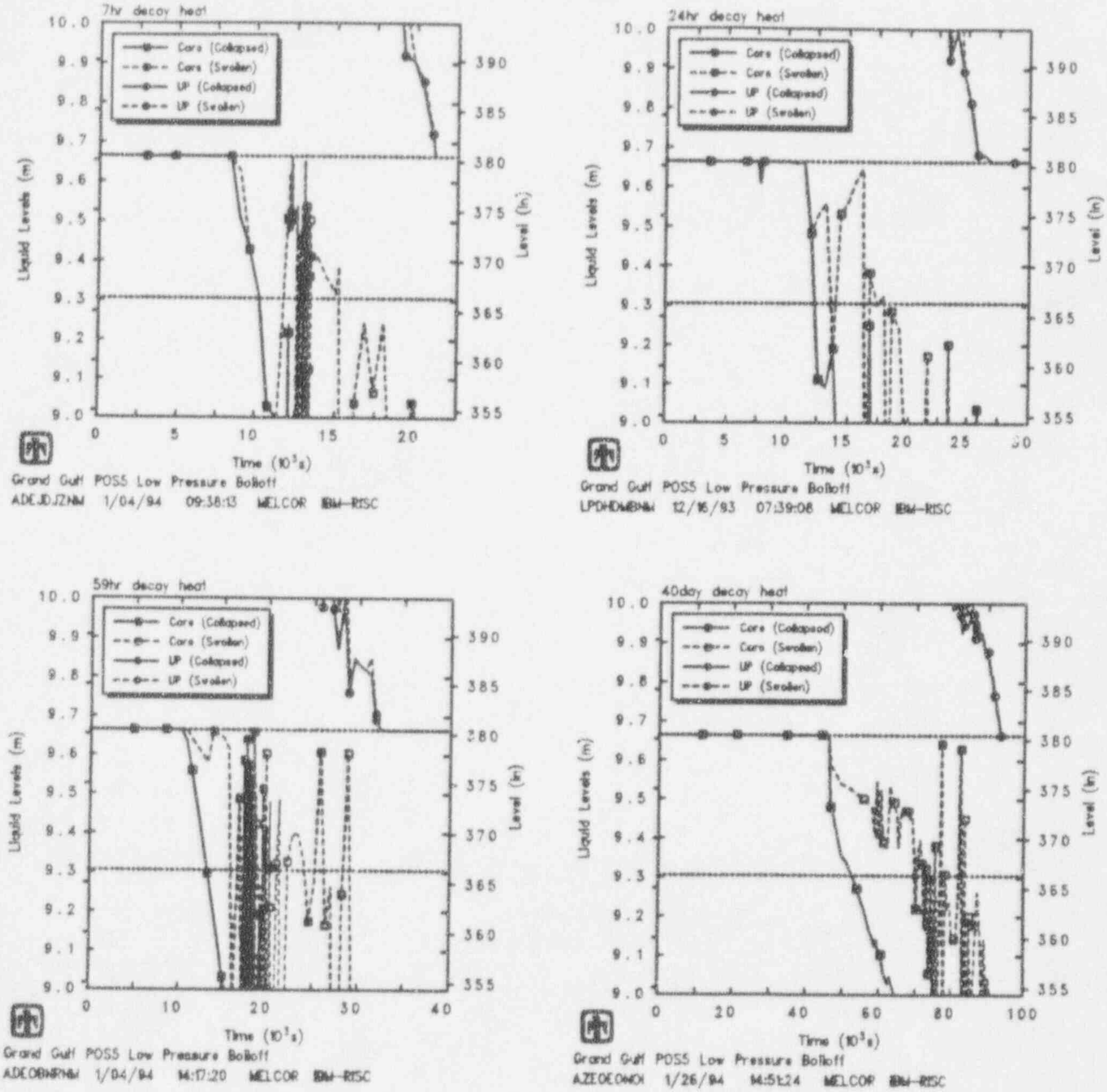


Figure 4.2.2.5. Core Liquid Levels for Grand Gulf POS 5 - Low Pressure Boiloff, Initiated 7 hr (upper left), 24 hr (upper right), 59 hr (lower left) and 40 day (lower right) After Shutdown

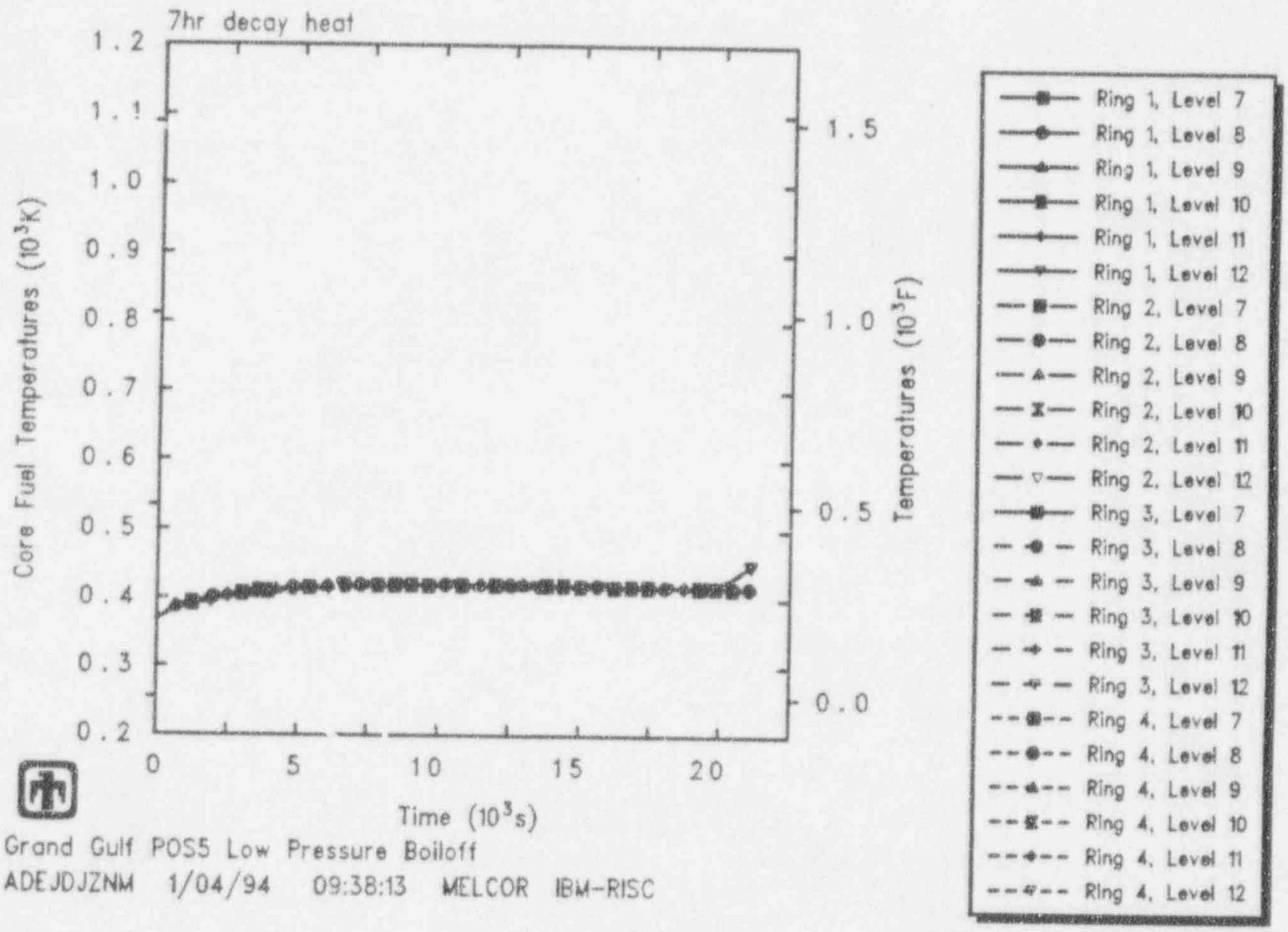


Figure 4.2.2.6. Core Fuel Temperatures for Grand Gulf POS 5 - Low Pressure Boiloff, Initiated 7 hr After Shutdown

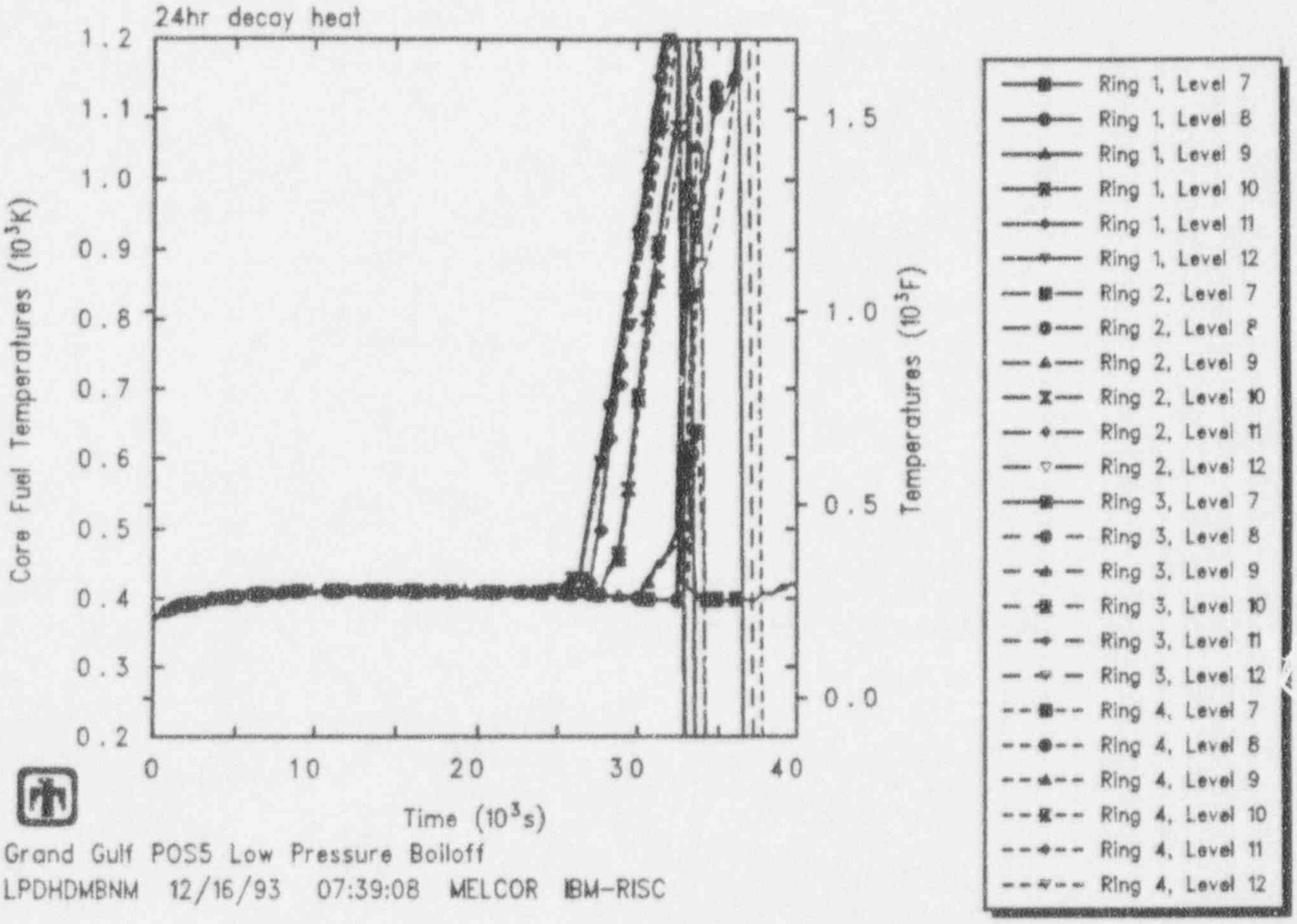


Figure 4.2.2.7. Core Fuel Temperatures for Grand Gulf POS 5 - Low Pressure Boiloff, Initiated 24 hr After Shutdown

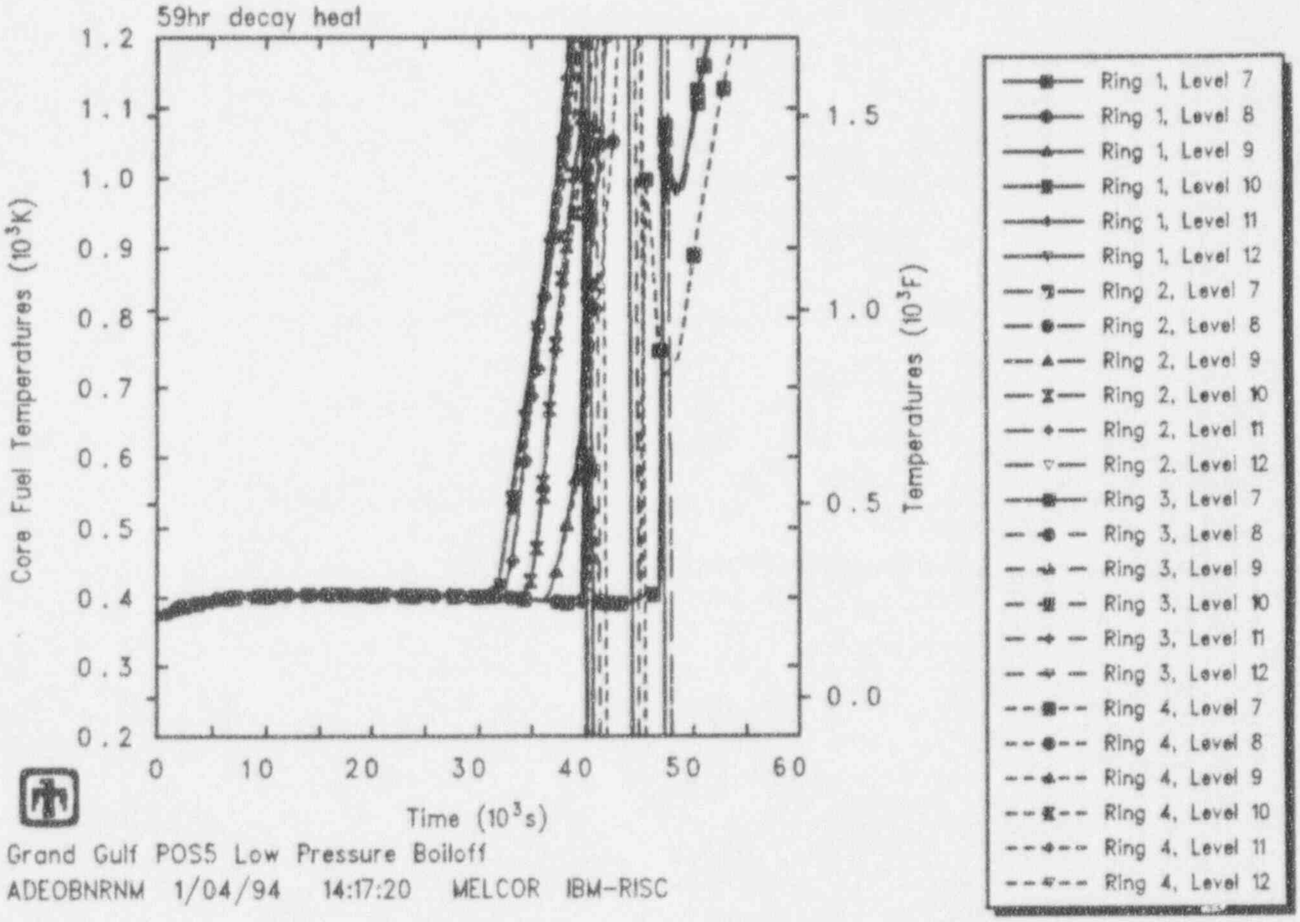


Figure 4.2.2.8. Core Fuel Temperatures for Grand Gulf POS 5 – Low Pressure Boiloff, Initiated 59 hr After Shutdown

Table 4.2.2.1. Key Event Times for Grand Gulf POS 5 - Low Pressure Boiloff, Initiated at Various Times After Shutdown

Initiation Time After Shutdown	Time to (s) TAF Uncovery†	Core Heatup	First Gap Release	Vessel Failure
7 hr	10,250	20,000	-	-‡
24 hr	12,250	25,400	31,600	136,386
59 hr	13,200	31,600	32,500	-‡
12 day	30,400	63,000	-	-‡
40 day	52,000	>90,000	-	-‡

†Collapsed liquid level

‡Calculation stopped before event occurred

Table 4.2.2.2. Key Signal Times for Grand Gulf POS 5 - Low Pressure Boiloff, Initiated at Various Times After Shutdown

Initiation Time After Shutdown	Time to (s) Collapsed Level <544.4 in	Swollen Level <544.4 in
7 hr	750	
24 hr	1,000	
59 hr	2,000	
12 day	3,600	
40 day	(1.4 hr)	

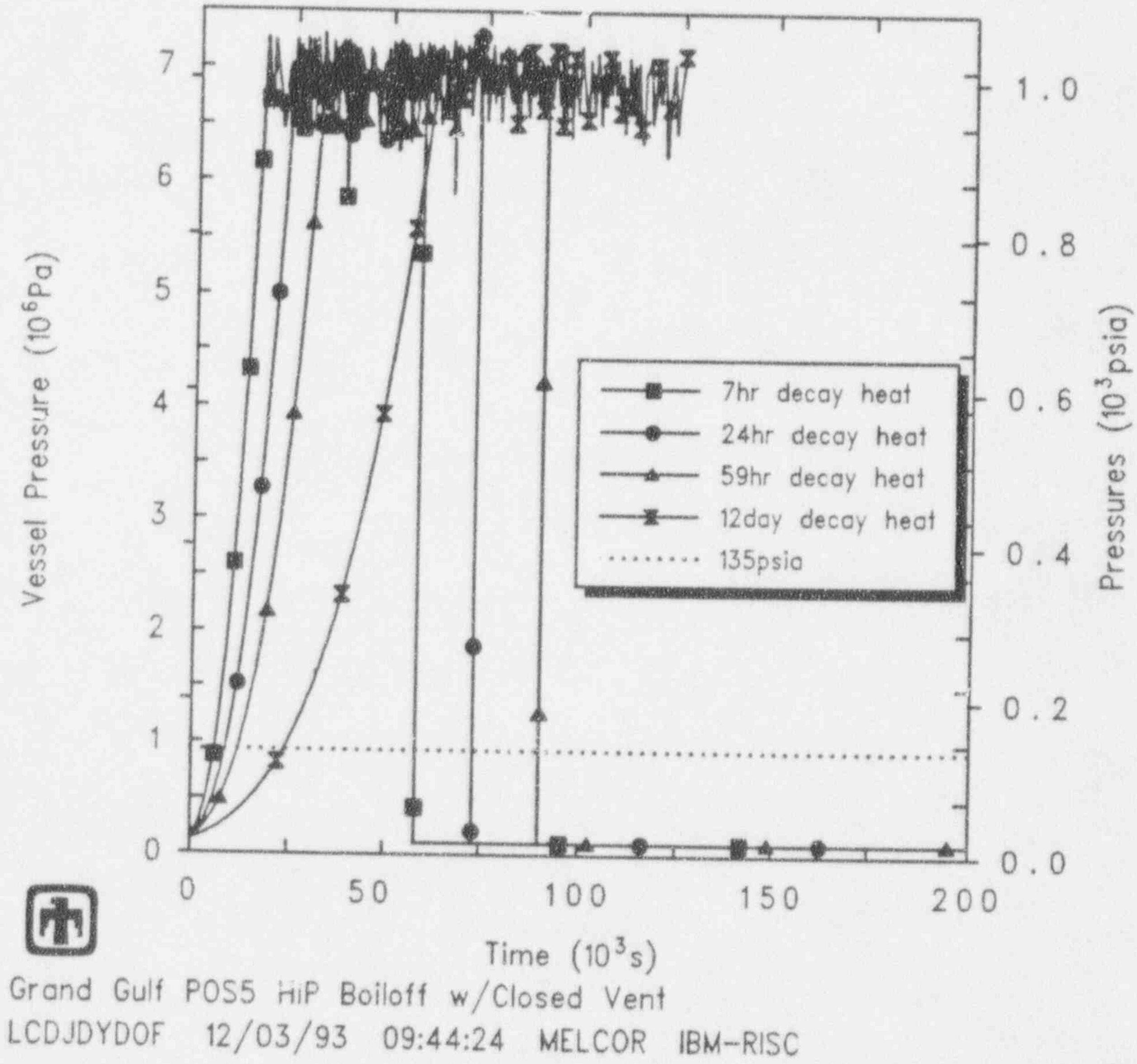


Figure 4.2.3.1. Reactor Vessel Pressures for Grand Gulf POS 5 - High Pressure Boiloff with Closed RPV Vent, Initiated at Various Times After Shutdown

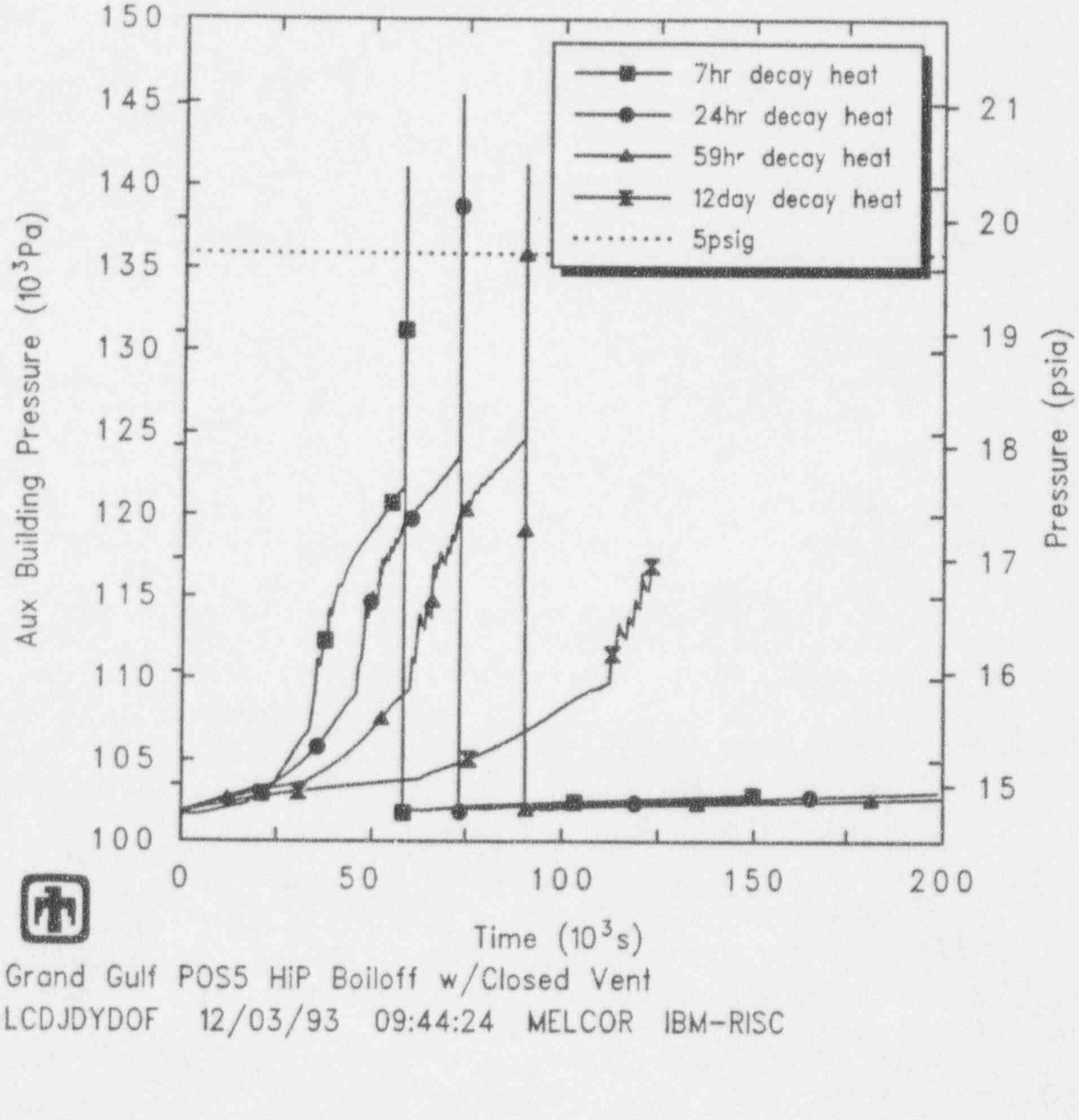


Figure 4.2.3.2. Auxiliary Building Pressures for Grand Gulf POS 5 - High Pressure Boiloff with Closed RPV Vent, Initiated at Various Times After Shutdown

Initially, the vessel water mass remains constant while the system pressurizes due to the loss of core cooling. After the SRV setpoint is reached, the coolant inventory in the vessel drops as the decay heat boils water to steam which is lost out the open SRVs, faster for higher decay heat levels, as presented in Figure 4.2.3.3.

Figure 4.2.3.4 gives the predicted upper plenum liquid level drop due to this inventory loss, for different decay heat levels and highlighting when a Level 3 trip (544.4 in) would be generated. The level initially rises as the vessel pressurizes, faster for higher decay heat levels, until the SRV begins cycling. The level then appears to remain constant for a brief time, and then drops as inventory continues to be lost out the SRV. The plateau in liquid level is an artifact of the MELCOR nodalization, in which the upper plenum volume extends up to just over 15.43 m; during the apparent level plateau, the liquid level in the vessel rises into the dryer/steam-dome control volume just above the upper-plenum/steam-separators control volume.

Figure 4.2.3.5 gives the corresponding upper core liquid level drop due to this inventory loss, for different decay heat levels and highlighting when TAF uncovery is calculated to occur; horizontal lines are included both at the top of the core (9.6 m) and at the TAF elevation (9.3 m). The swollen and collapsed liquid levels in the upper plenum generally drop rapidly and smoothly; the swollen and collapsed liquid levels in the upper core in contrast oscillate substantially. The core uncovery begins when the upper plenum still has substantial liquid left, with liquid downflow restricted by countercurrent flow limiting by upflow of the steam being generated in the core, but the two-phase level in the core does not drop substantially below the top of the active fuel until after the upper plenum is mostly drained. We take TAF uncovery as the final, substantive drop of the collapsed level below the TAF elevation, rather than as any of the earlier, intermittent oscillations.

The core heatup is illustrated in Figures 4.2.3.6 through 4.2.3.8, as calculated for this high-pressure boiloff with closed RPV vent starting at 7 hr, 24 hr and 59 hr after shutdown. As with TAF uncovery, core uncovery begins sooner and proceeds more rapidly at higher decay heat levels. The calculation begun 12 days after shutdown showed core heatup beginning after about 90,000 s, and is not shown because the period of interest for all these Level 1 analyses is the shorter of either accident initiation to core damage or 1 day after accident start.

Tables 4.2.3.1 and 4.2.3.2 summarize the timings of various key events predicted using MELCOR for this sequence assuming various times after shutdown and associated decay heat levels. A calculation beginning 40 days after shutdown was not done for this sequence because the results of the analysis beginning 12 days after shutdown showed no significant core uncovery or damage within the 1 day maximum time window of interest.

4.2.4 High Pressure Boiloff with Open RPV Head Vent

At the initiation of the accident, the reactor vessel is depressurized, the coolant is at the normal level and the SRVs are closed. The SRVs remain closed during the accident and only open to relieve pressure at the safety setpoint. The vessel water inventory is at

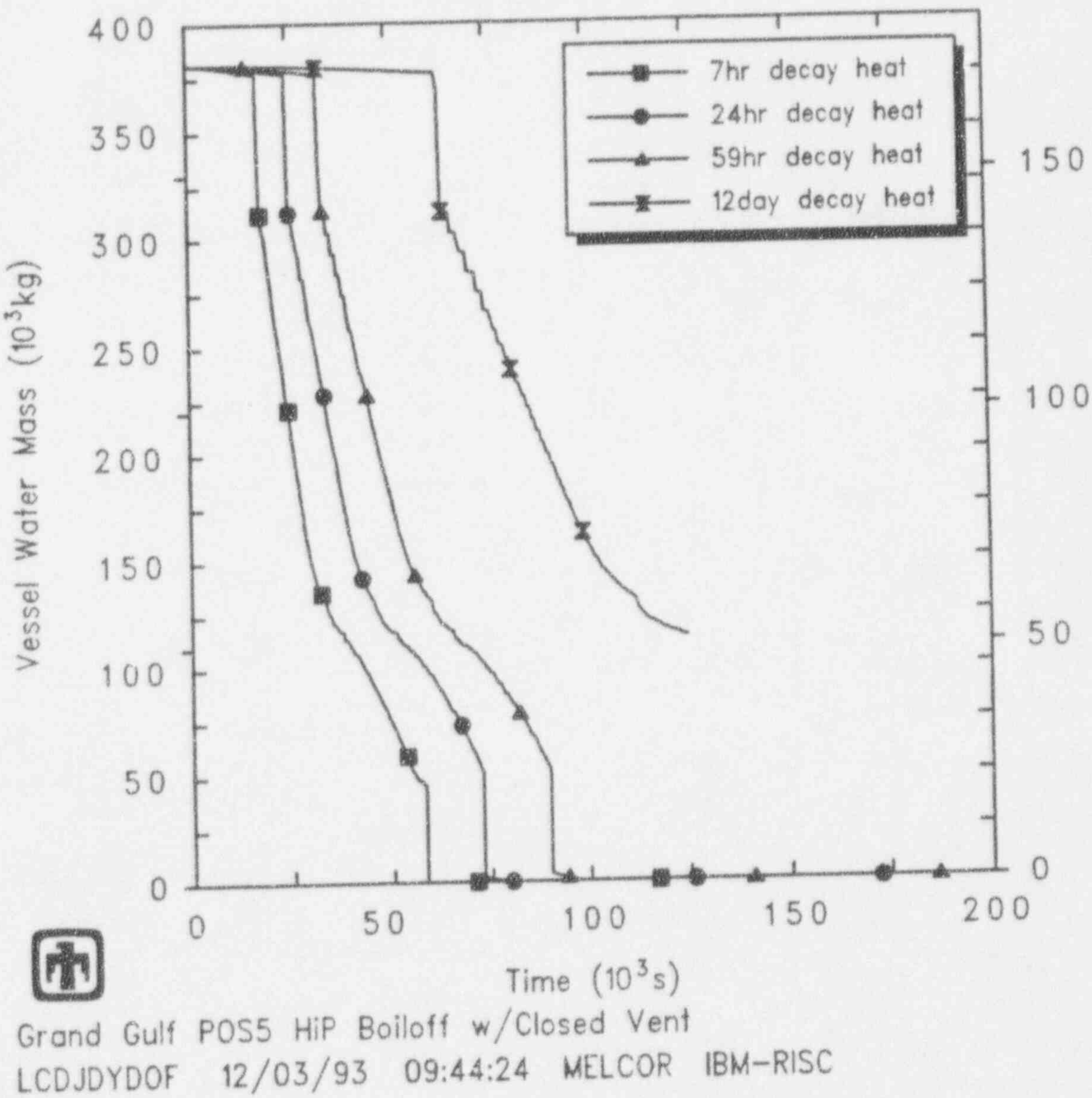


Figure 4.2.3.3. Reactor Vessel Water Masses for Grand Gulf POS 5 - High Pressure Boiloff with Closed RPV Vent, Initiated at Various Times After Shutdown

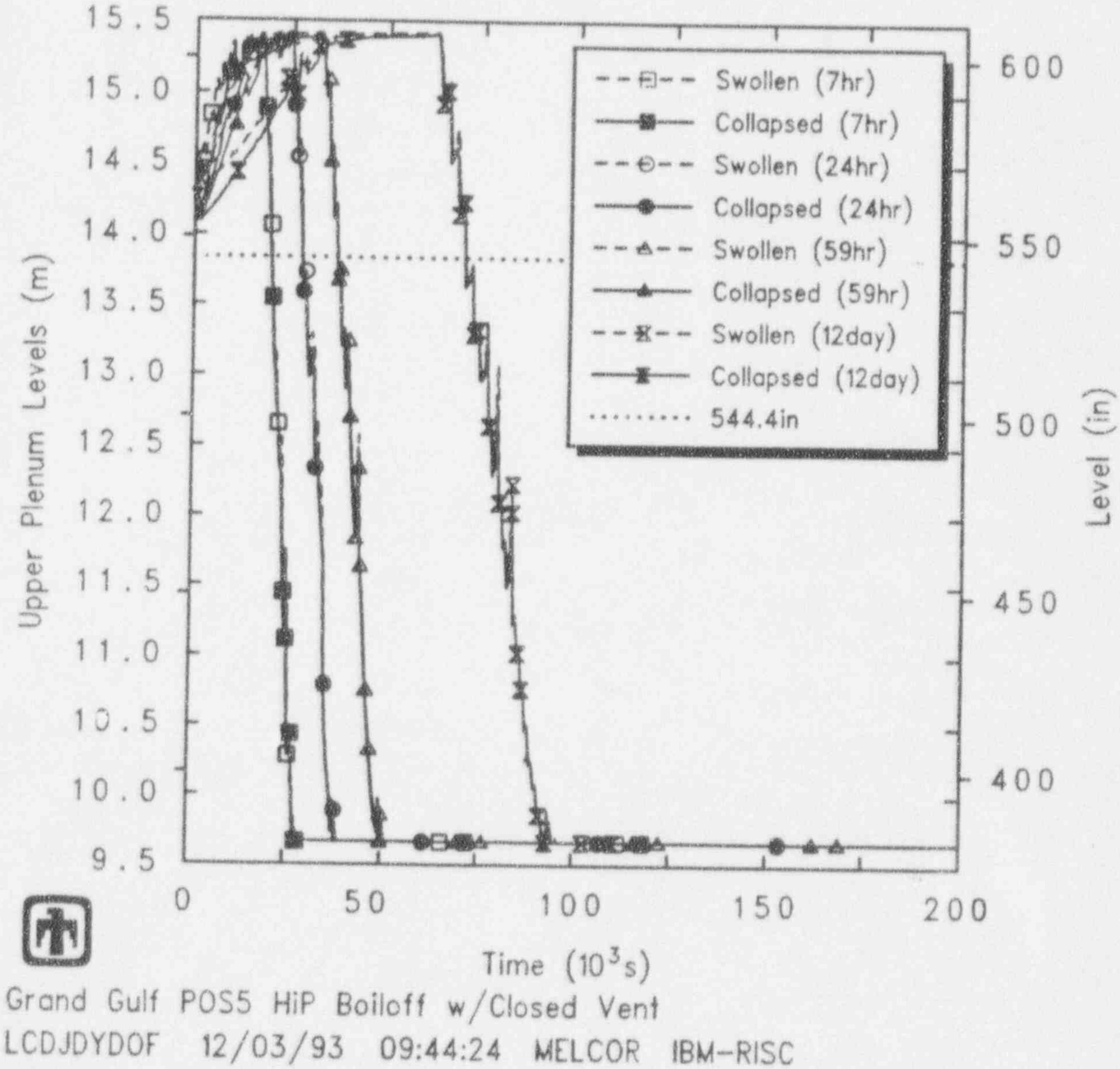


Figure 4.2.3.4. Upper Plenum Liquid Levels for Grand Gulf POS 5 - High Pressure Boiloff with Closed RPV Vent, Initiated at Various Times After Shutdown

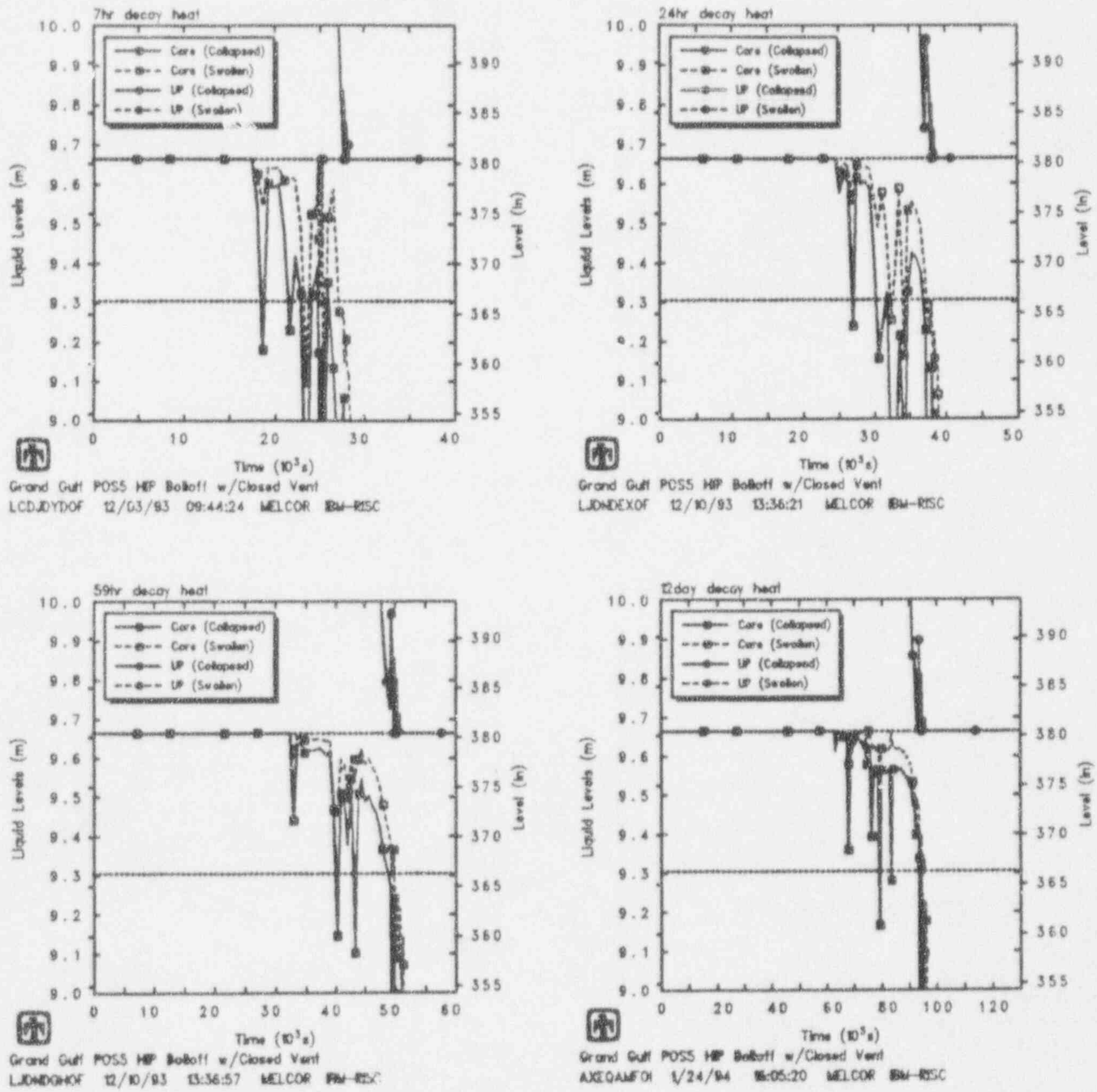


Figure 4.2.3.5. Core Liquid Levels for Grand Gulf POS 5 - High Pressure Boiloff with Closed RPV Vent, Initiated 7 hr (upper left), 24 hr (upper right), 59 hr (lower left) and 12 day (lower right) After Shutdown

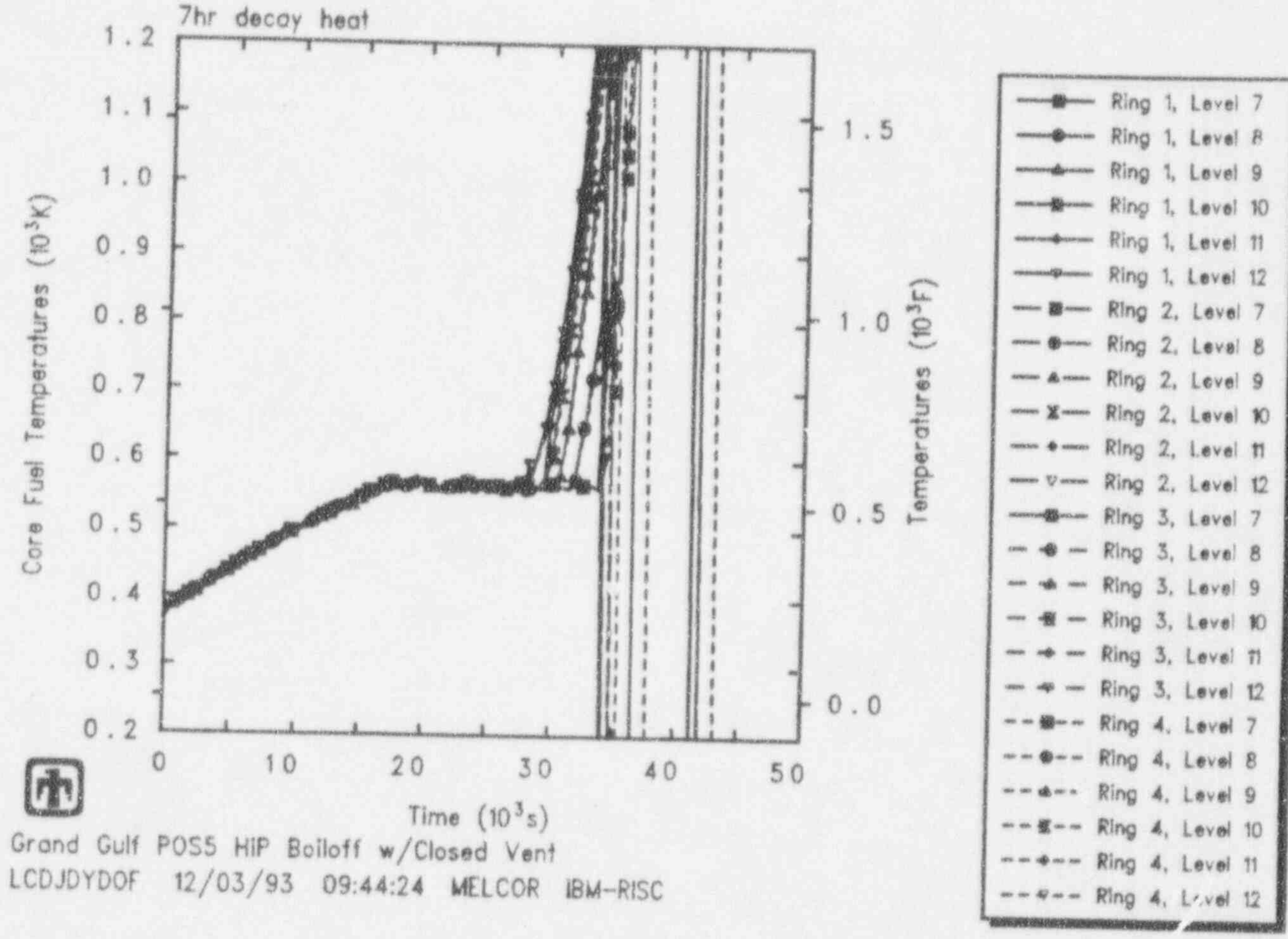


Figure 4.2.3.6. Core Fuel Temperatures for Grand Gulf POS 5 - High Pressure Boiloff with Closed RPV Vent, Initiated 7 hr After Shutdown

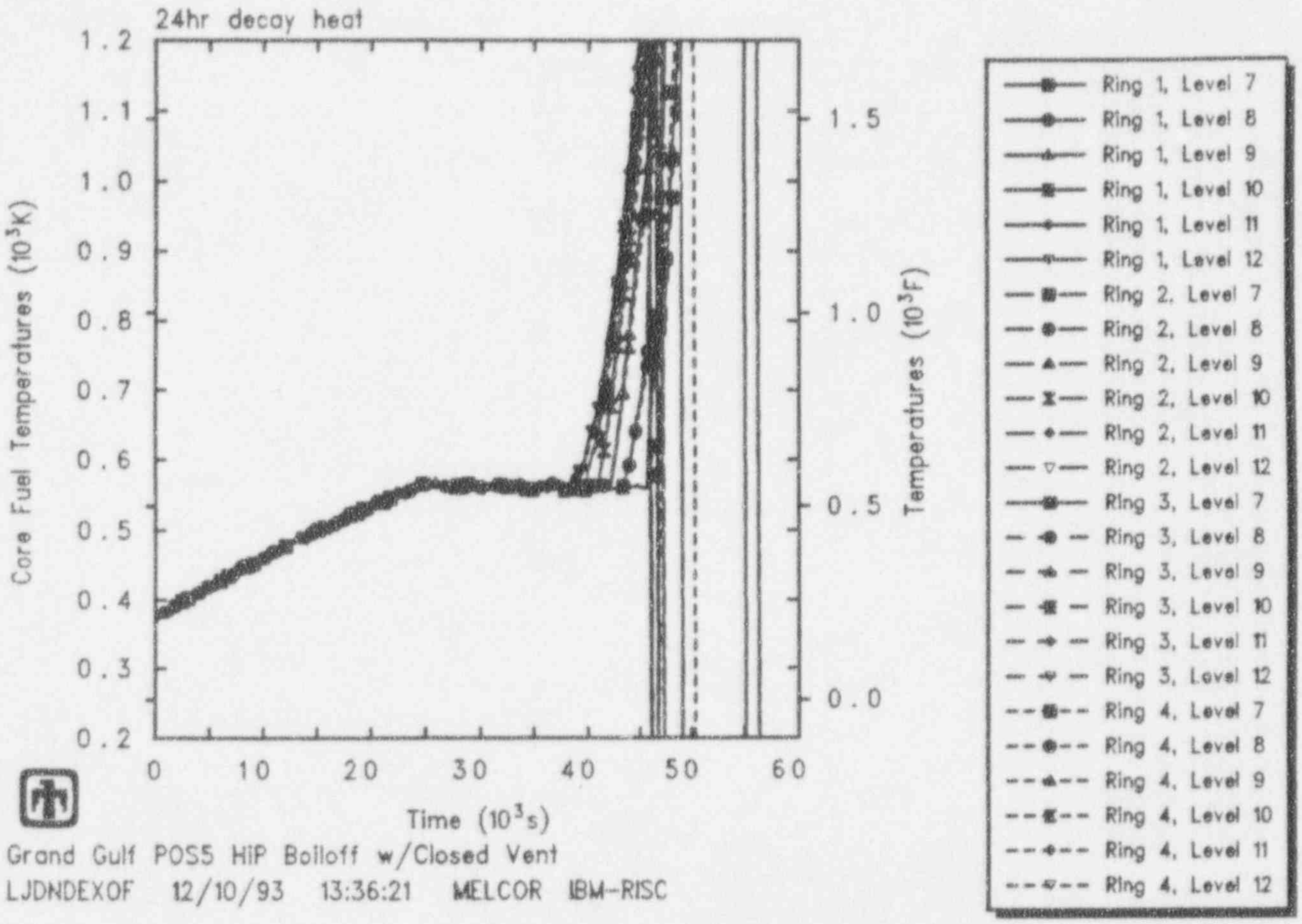


Figure 4.2.3.7. Core Fuel Temperatures for Grand Gulf POS 5 – High Pressure Boiloff with Closed RPV Vent, Initiated 24 hr After Shutdown

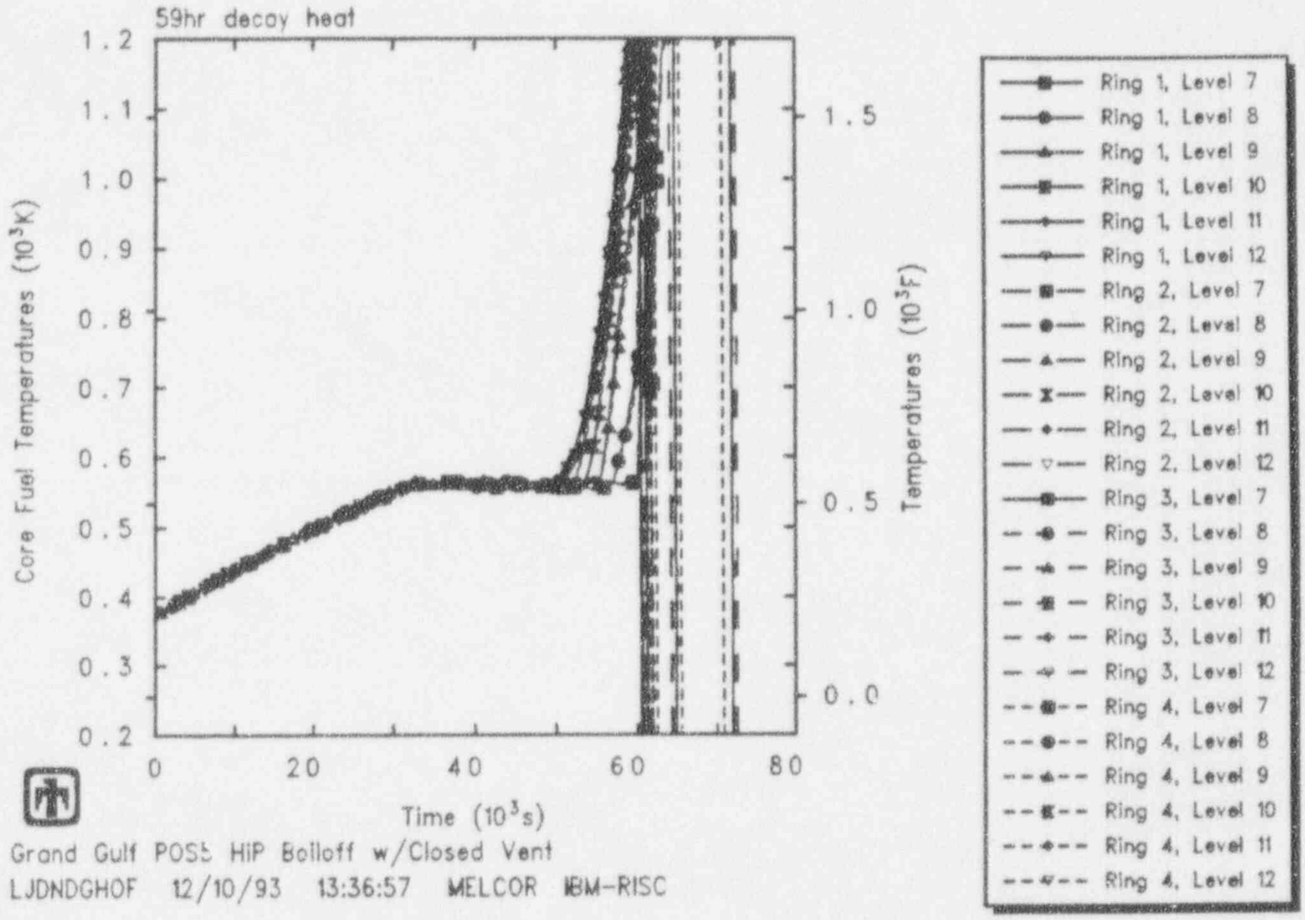


Figure 4.2.3.8. Core Fuel Temperatures for Grand Gulf POS 5 – High Pressure Boiloff with Closed RPV Vent, Initiated 59 hr After Shutdown

Table 4.2.3.1. Key Event Times for Grand Gulf POS 5 - High Pressure Boiloff with Closed RPV Head Vent, Initiated at Various Times After Shutdown

	Time to (s)			
Initiation Time After Shutdown	TAF Uncovery†	Core Heatup	First Gap Release	Vessel Failure
7 hr	26,000	28,400	32,638	58,043
24 hr	36,650	37,800	44,451	72,784
59 hr	48,800	50,400	58,624	89,888
12 day	93,000	96,200	110,500	–‡

†Collapsed liquid level

‡Calculation stopped before event occurred

Table 4.2.3.2. Key Signal Times for Grand Gulf POS 5 - High Pressure Boiloff with Closed RPV Head Vent, Initiated at Various Times After Shutdown

	Time to (s)			
Initiation Time After Shutdown	Collapsed Level <544.4 in	Swollen Level <544.4 in	P> 135 psia	P> 160 psig
7 hr	20,000		6,200	
24 hr	25,500		9,000	
59 hr	37,200		12,200	
12 day	(19.44 hr)	(6.63 hr)	23,500 (7.67 hr)	

366.5 K (200°F), which corresponds to the maximum temperature allowed by the Grand Gulf technical specifications for operation in POS 5. The drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open (*i.e.*, "open containment"). This scenario is identical to case 3, except that the reactor pressure vessel head vent is open.

(As for the high pressure boiloff with closed RPV head vent in the previous section, a calculation beginning 40 days after shutdown was not done for this sequence because the results of the analysis beginning 12 days after shutdown showed no significant core uncovery or damage within the 1 day time window of interest.)

Figure 4.2.4.1 gives the vessel pressures calculated starting this accident scenario at several different times after scram. In all cases, the system begins pressurizing as all core cooling is lost and continues pressurizing until reaching the SRV setpoint. As in the sequence with a closed RPV vent, the SRVs then cycle around the valve setpoints, intermittently opening. However, with the RPV vent line open, there is continual, limited relief out the vent line throughout the entire period. This increases inventory loss. The system does not remain at the SRV cycling setpoints until vessel failure, but instead remains at the SRV cycling setpoints for only a few valve cycles before dropping due to continual inventory loss out the open RPV vent line. However, whether the RPV vent is open or closed, the higher the decay heat (*i.e.*, the sooner after shutdown), the faster the initial pressurization and associated inventory loss, and the earlier the eventual vessel fails.

The steam flow out both the SRVs and the RPV vent pressurizes the containment and the auxiliary building, as shown in Figure 4.2.4.2. The longer after shutdown and scram that this accident sequence begins, the lower the decay heat and the slower the auxiliary building pressurizes. Unlike the results with the RPV vent closed, the auxiliary building reaches its 5 psig overpressure failure setpoint before vessel failure, due to the continued inventory loss through the open RPV vent for the higher decay heat level cases (*i.e.*, 7 hr, 24 hr and 59 hr after shutdown). Only for lower decay heat level s (*i.e.*, 12 days after shutdown) is the behavior the same with the RPV vent open or closed: the auxiliary building does not reach its 5 psig overpressure failure setpoint before vessel failure, but instead fails on a containment pressure spike caused by vessel failure and debris ejection.

Figure 4.2.4.3 illustrates that the vessel water mass drops more continuously with the RPV vent open than for the same accident scenario but with the RPV vent closed (Figure 4.2.4.3), in both cases dropping faster for higher decay heat levels.

Figure 4.2.4.4 gives the predicted upper plenum swollen and collapsed liquid levels for different decay heat levels and highlighting when a Level 3 trip (544.4 in) would be generated. The level initially rises as the vessel pressurizes, faster for higher decay heat levels, and then drops as inventory continues to be lost out the RPV vent and the SRV. The levels rise more slowly and later drop more slowly with the RPV vent open than with it closed (Figure 4.2.3.4), reflecting the difference between a more gradual, continual loss of inventory out the RPV vent in addition to flow out the cycling SRVs in the case with the RPV head vent open, compared to an inventory loss out the SRVs beginning

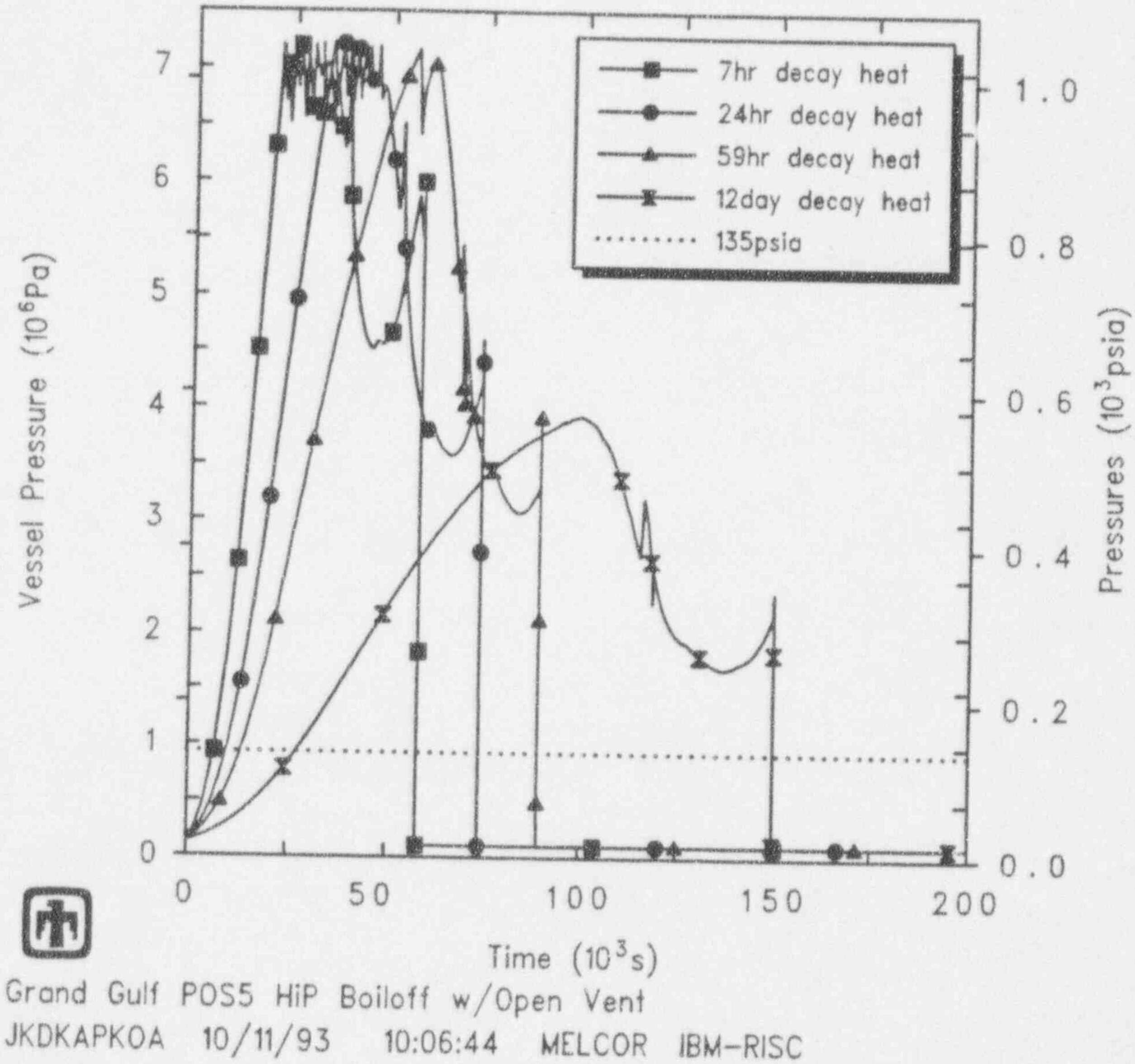


Figure 4.2.4.1. Reactor Vessel Pressures for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Vent, Initiated at Various Times After Shutdown

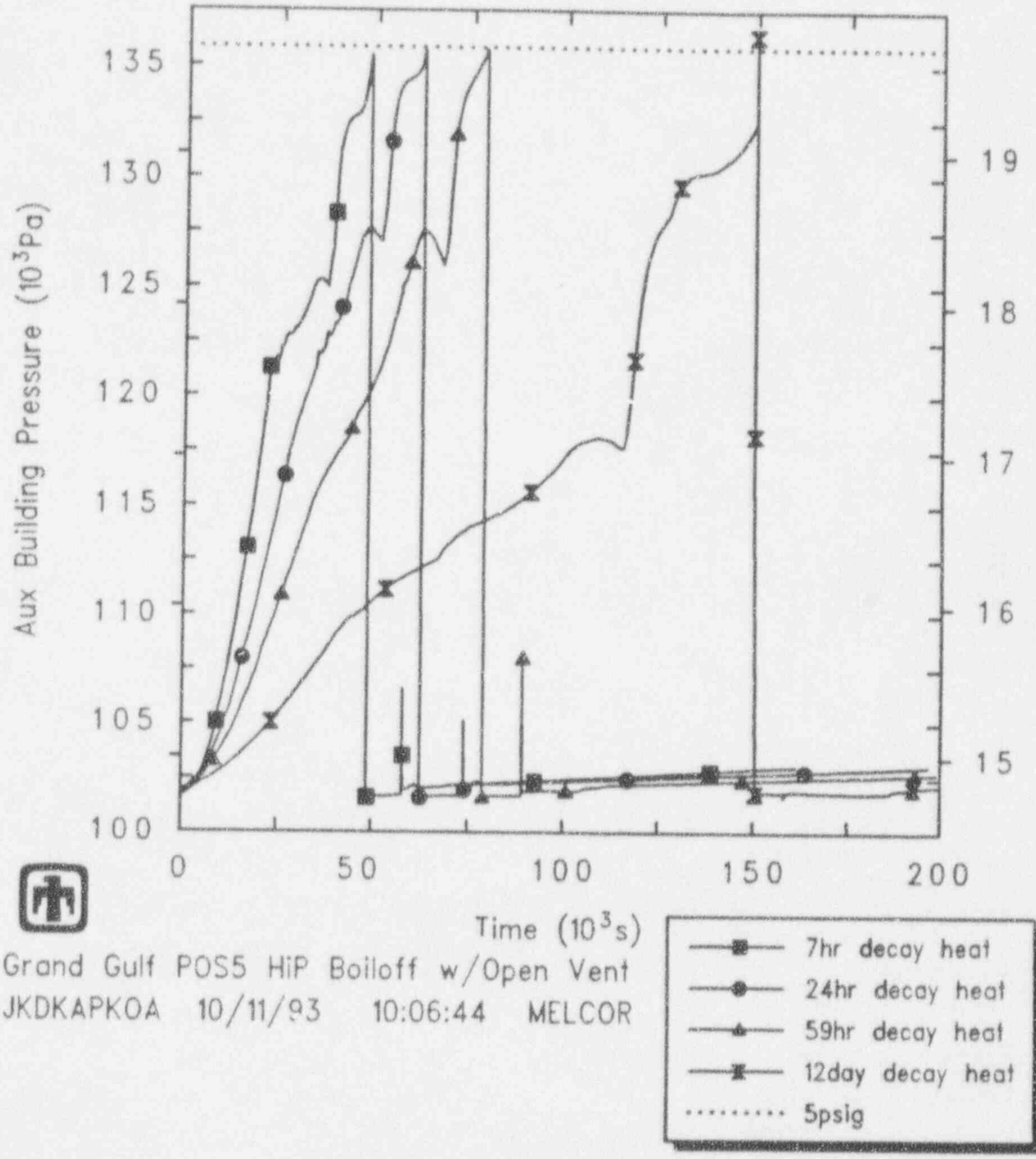


Figure 4.2.4.2. Auxiliary Building Pressures for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Vent, Initiated at Various Times After Shutdown

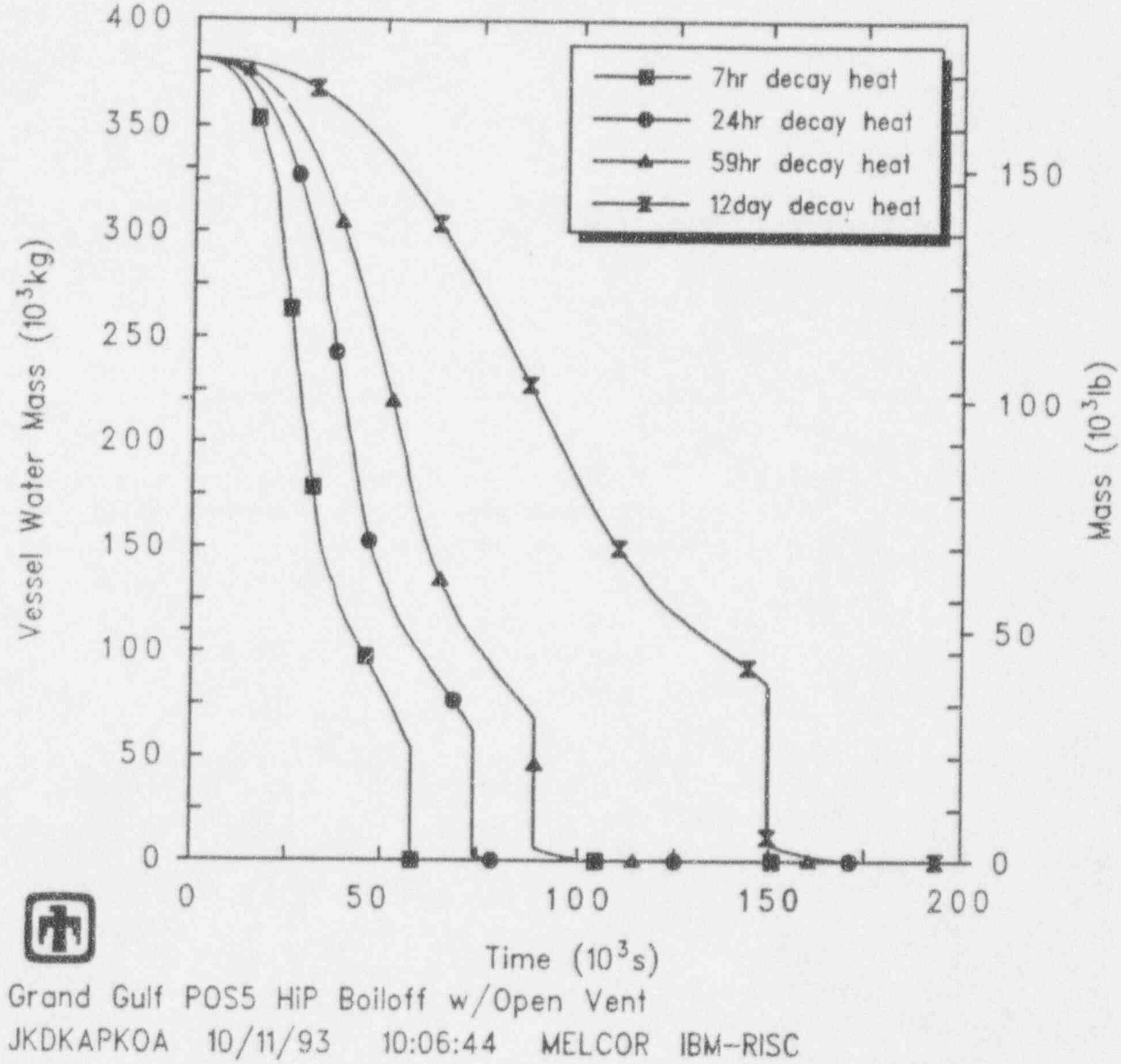


Figure 4.2.4.3. Reactor Vessel Water Masses for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Vent, Initiated at Various Times After Shutdown

later but progressing more rapidly as the system remains at pressure at the SRV setpoint longer with the RPV vent closed.

Figure 4.2.4.5 gives the corresponding upper core liquid level drop due to this inventory loss, for different decay heat levels and highlighting when TAF uncovery is calculated to occur; horizontal lines are included both at the top of the core (9.6 m) and at the TAF elevation (9.3 m). With the RPV vent open, the swollen and collapsed liquid levels in the upper core generally drop more smoothly than corresponding analyses with the RPV vent closed (Figure 4.2.4.5).

The core heatup is illustrated in Figures 4.2.4.6 through 4.2.4.8, as calculated for this high-pressure boiloff with the RPV vent open starting at 7 hr, 24 hr and 59 hr after shutdown. The results with the RPV vent open and closed are generally quite similar. As with TAF uncovery, core uncovery begins sooner and proceeds more rapidly at higher decay heat levels. As with the RPV vent closed, the calculation with the RPV vent open and initial decay heat corresponding to 12 days after shutdown showed core heatup beginning only after about 90,000 s, and is not shown because the period of interest for all these Level 1 analyses is the first 24 hr after accident initiation.

Tables 4.2.4.1 and 4.2.4.2 summarize the timings of various key events predicted using MELCOR for this sequence assuming various times after shutdown and associated decay heat levels. (A calculation beginning 40 days after shutdown was not done for this sequence because the results of the analysis beginning 12 days after shutdown showed no significant core uncovery or damage within the 1 day time window of interest.)

4.2.5 Large Break LOCA

This accident is initiated by a large break LOCA in the recirculation line. At the start of the accident, the reactor vessel is depressurized, the coolant is at the normal level and the SRVs are closed. The vessel water inventory is at 366.5 K (200°F), which corresponds to the maximum temperature allowed by the Grand Gulf technical specifications for operation in POS 5. The break drains the vessel to 2/3 core height. The initiating event then results in a loss of all core cooling and coolant makeup. The reactor pressure vessel head vent is closed at the beginning of the transient. The drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open (*i.e.*, "open containment").

Figure 4.2.5.1 gives the vessel pressures calculated for this accident scenario initiated at several different times after scram. In all cases, the primary system remains near atmospheric as the large break maintains pressure near-equilibrium between the primary and the containment, while the open personnel locks and equipment hatch vent the containment to the auxiliary building. For any given decay heat level, the smaller pressure spikes seen in Figure 4.2.5.1 generally correspond to core heatup and damage, while the largest pressure spikes seen in Figure 4.2.5.1 correspond to vessel failure.

The water and steam coolant flowing out through the break pressurizes the containment and, through the open equipment hatch and personnel locks, pressurizes the

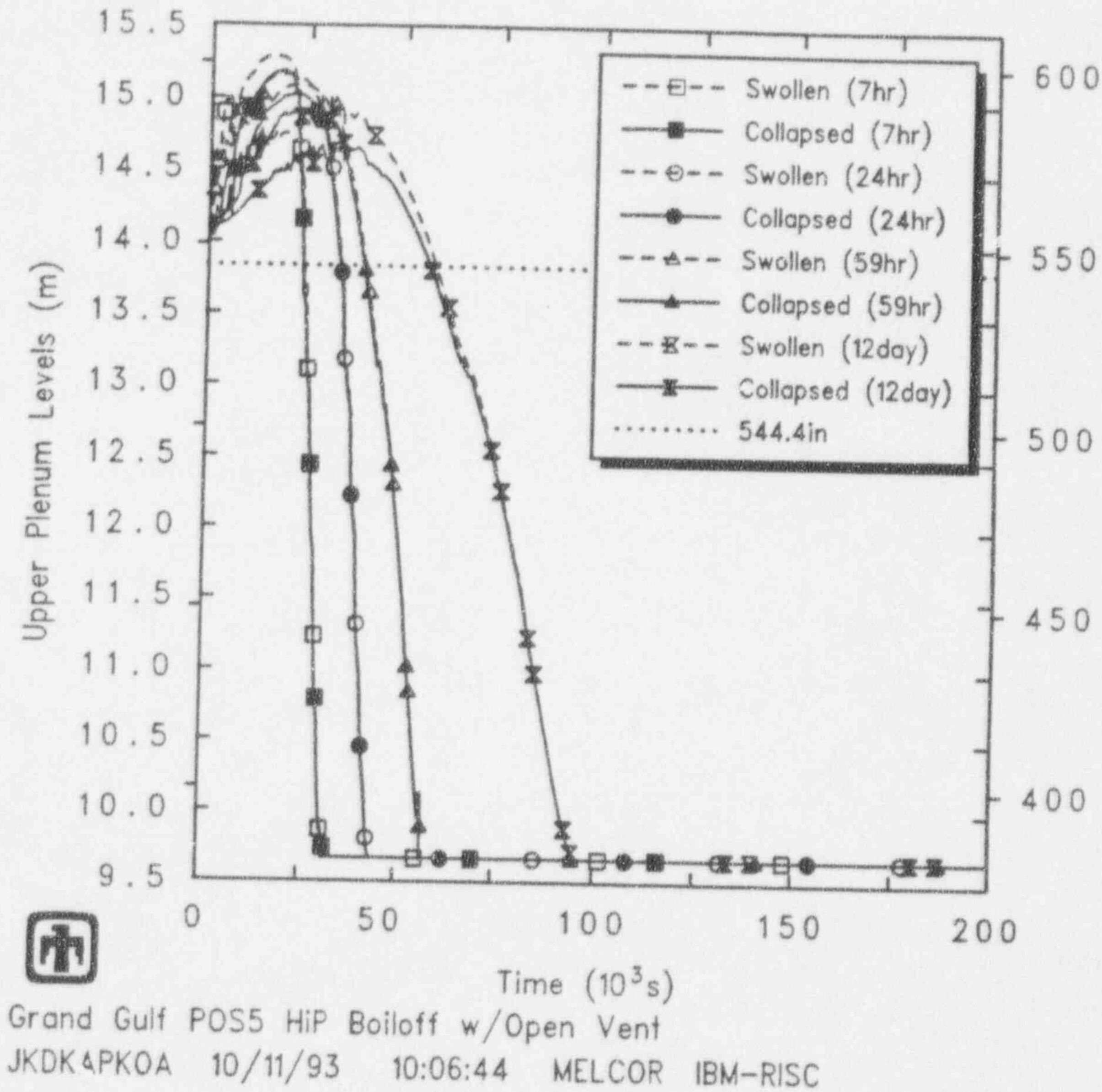


Figure 4.2.4.4. Upper Plenum Liquid Levels for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Vent, Initiated at Various Times After Shutdown

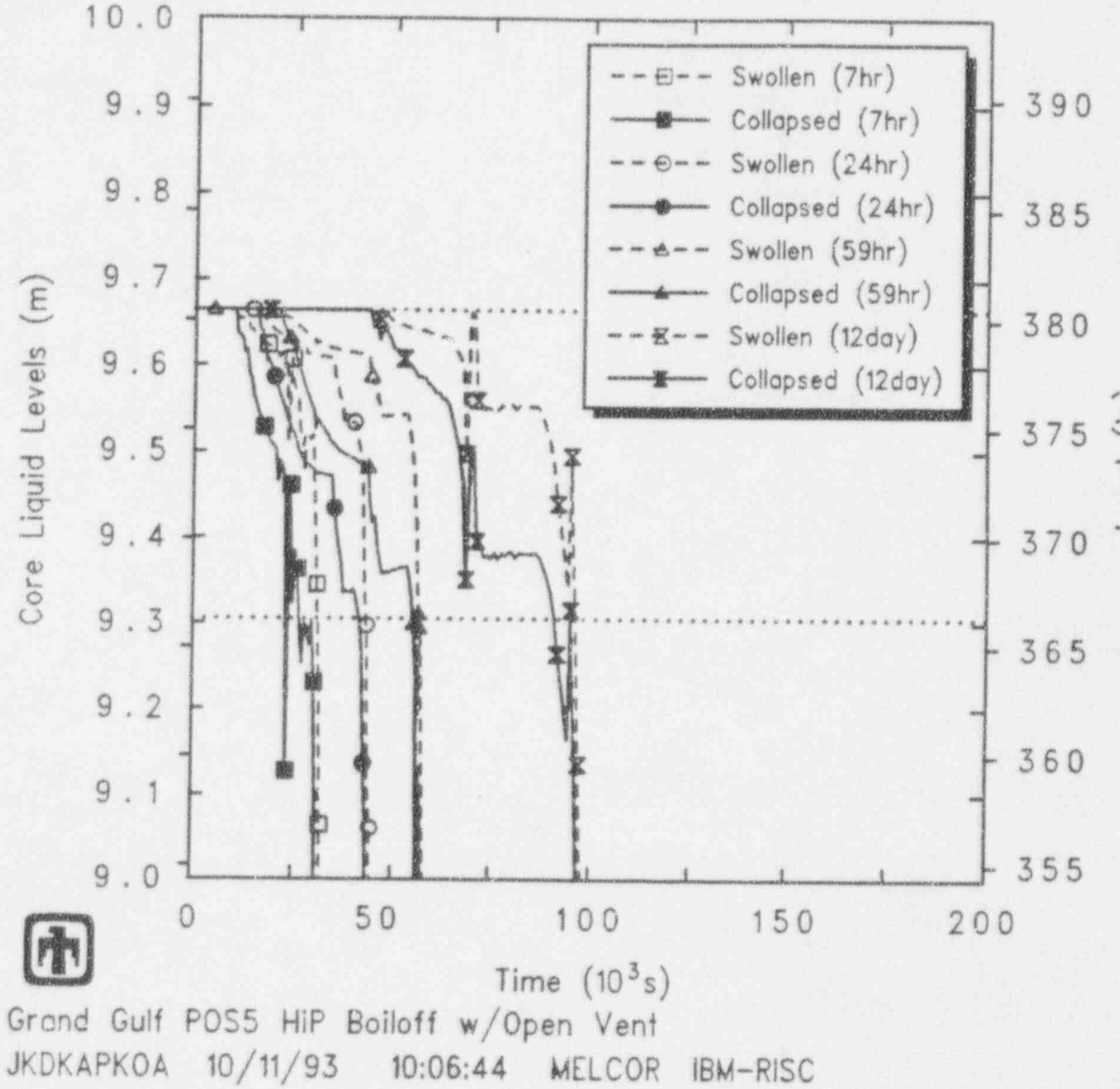


Figure 4.2.4.5. Core Liquid Levels for Grand Gulf POS 5 – High Pressure Boiloff with Open RPV Vent, Initiated at Various Times After Shutdown

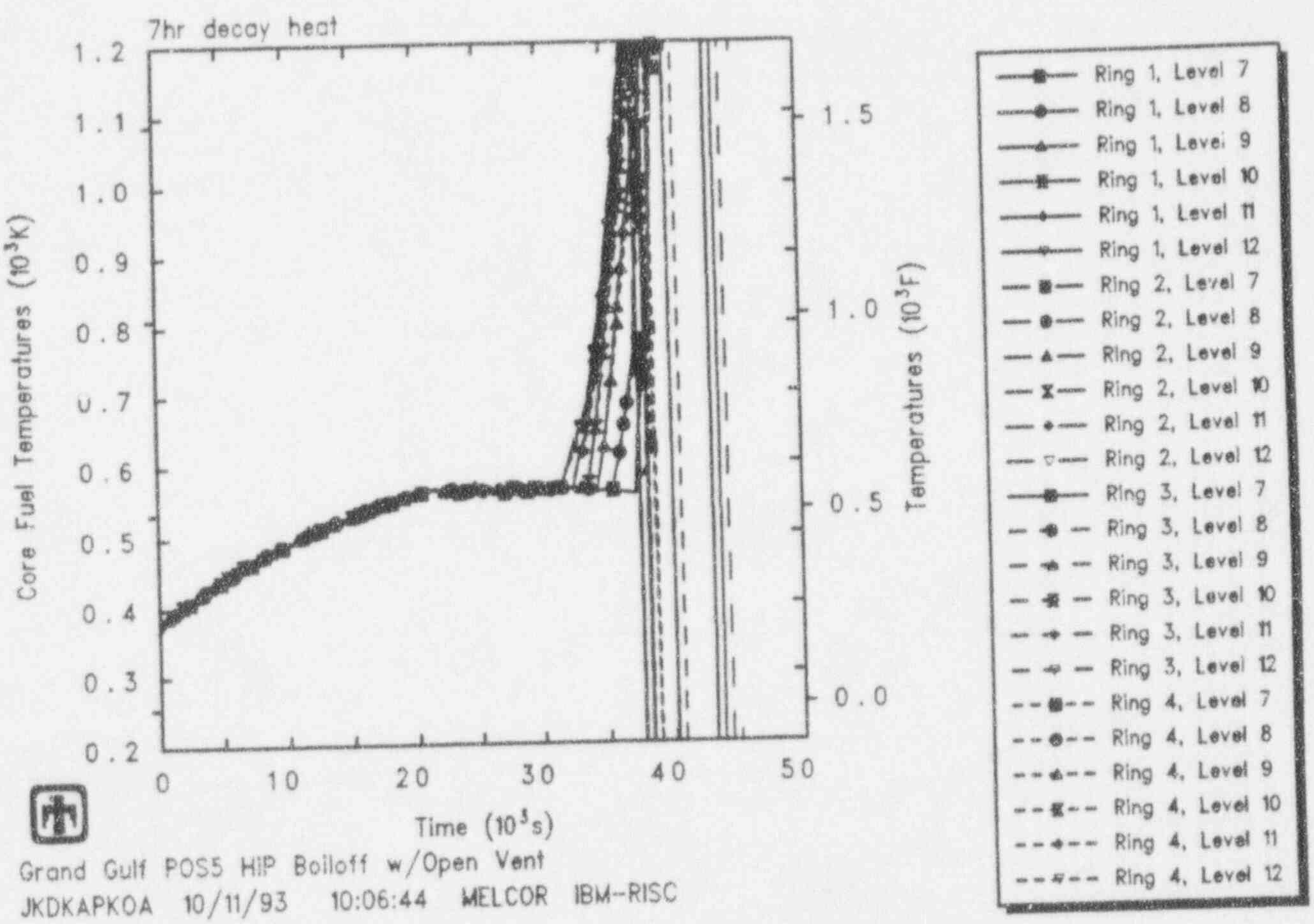


Figure 4.2.4.6. Core Fuel Temperatures for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Vent, Initiated 7 hr After Shutdown

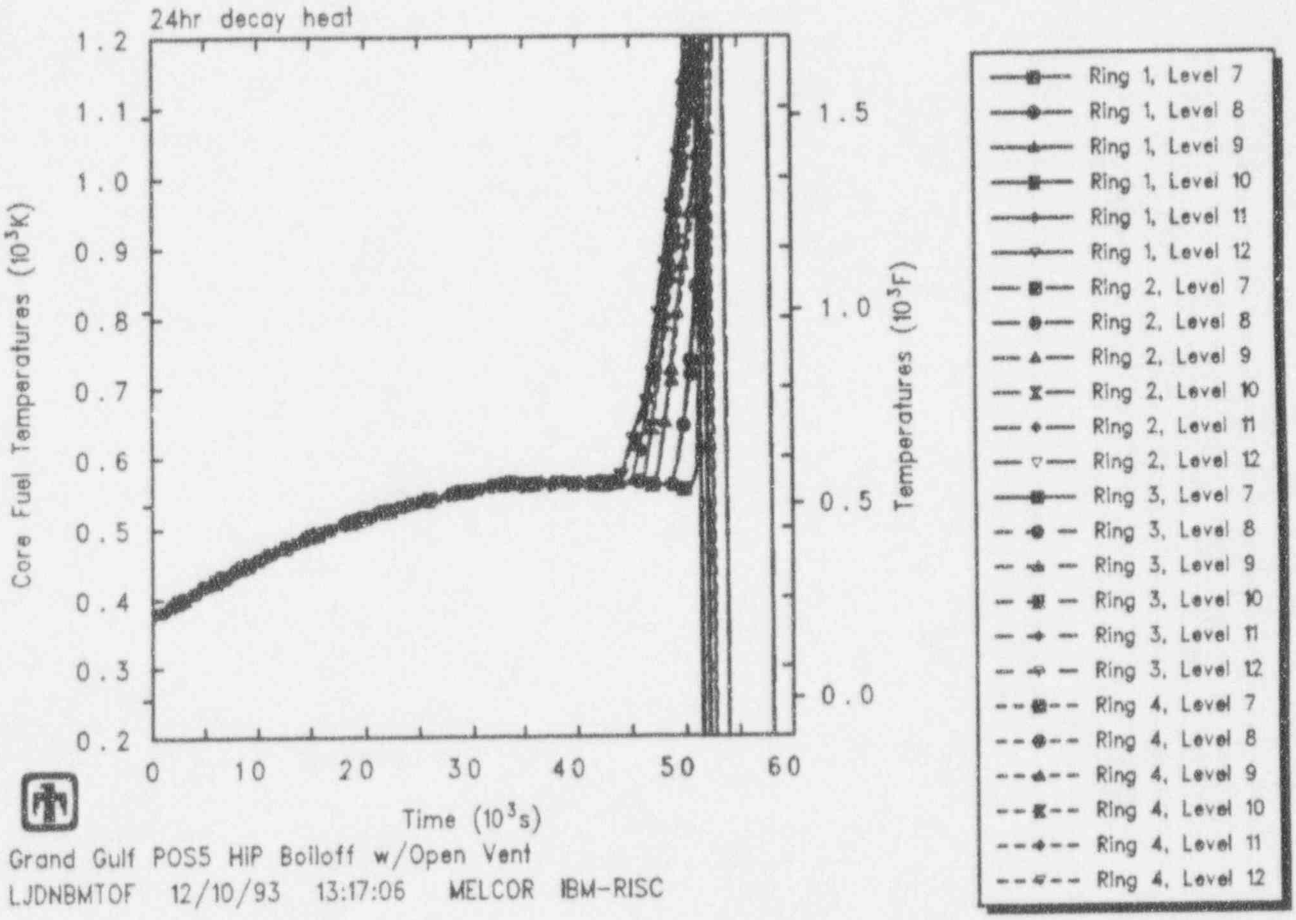


Figure 4.2.4.7. Core Fuel Temperatures for Grand Gulf POS 5 – High Pressure Boiloff with Open RPV Vent, Initiated 24 hr After Shutdown

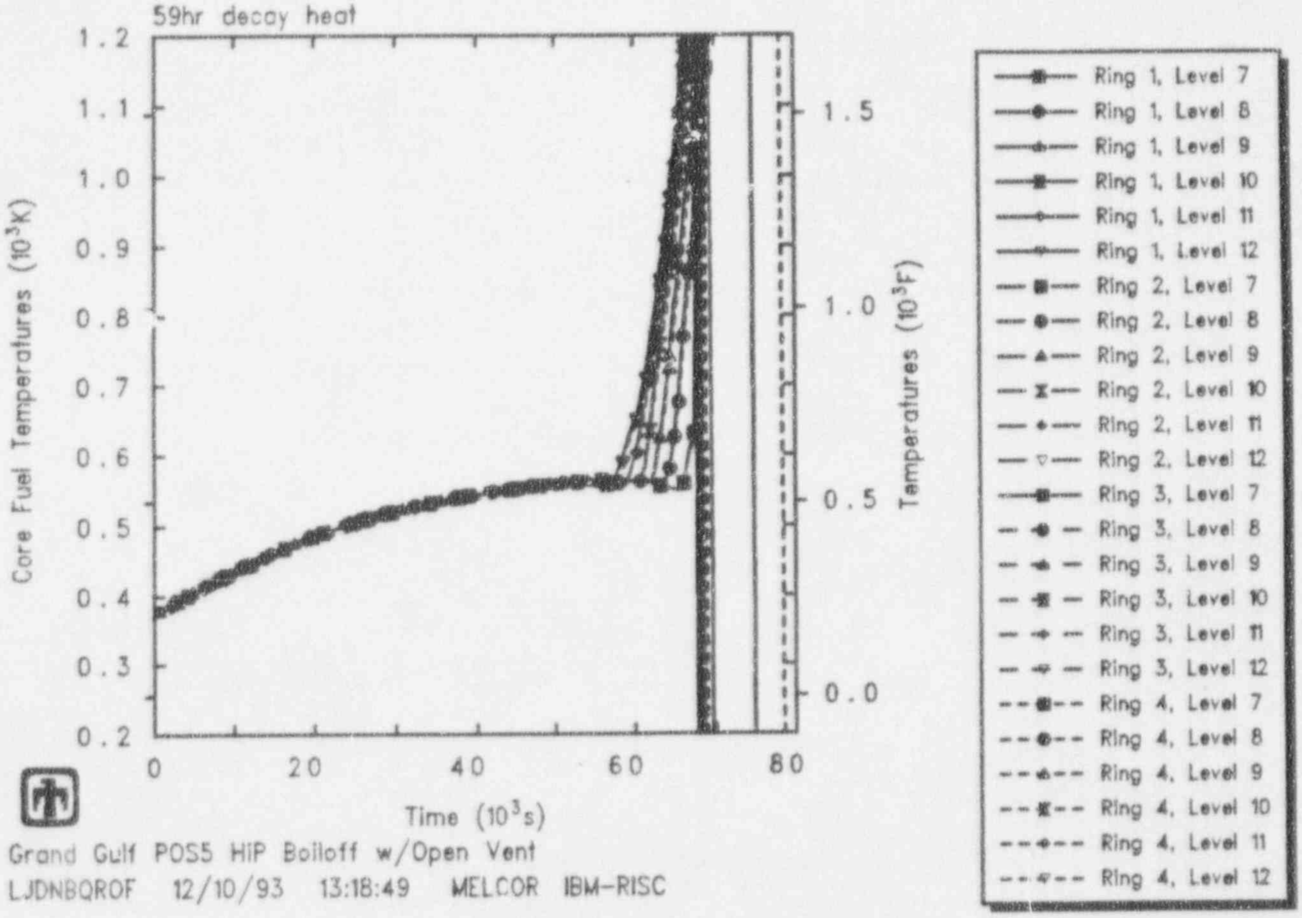


Figure 4.2.4.8. Core Fuel Temperatures for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Vent, Initiated 59 hr After Shutdown

Table 4.2.4.1. Key Event Times for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Head Vent, Initiated at Various Times After Shutdown

	Time to (s)			
Initiation Time After Shutdown	TAF Uncovery†	Core Heatup	First Gap Release	Vessel Failure
7 hr	30,000	31,600	36,470	57,780
24 hr	40,850	43,800	49,930	73,550
59 hr	55,200	58,400	65,890	88,970
12 day	91,000	97,500	113,000	-‡

†Collapsed liquid level

‡Calculation stopped before event occurred

Table 4.2.4.2. Key Signal Times for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Head Vent, Initiated at Various Times After Shutdown

	Time to (s)			
Initiation Time After Shutdown	Collapsed Level <544.4 in	Swollen Level <544.4 in	P> 135 psia	P> 160 psig
7 hr	24,000		6,700	
24 hr	33,500		9,900	
59 hr	39,600		13,200	
12 day	(15.56 hr)	(7.58 hr)	27,300 (9 hr)	

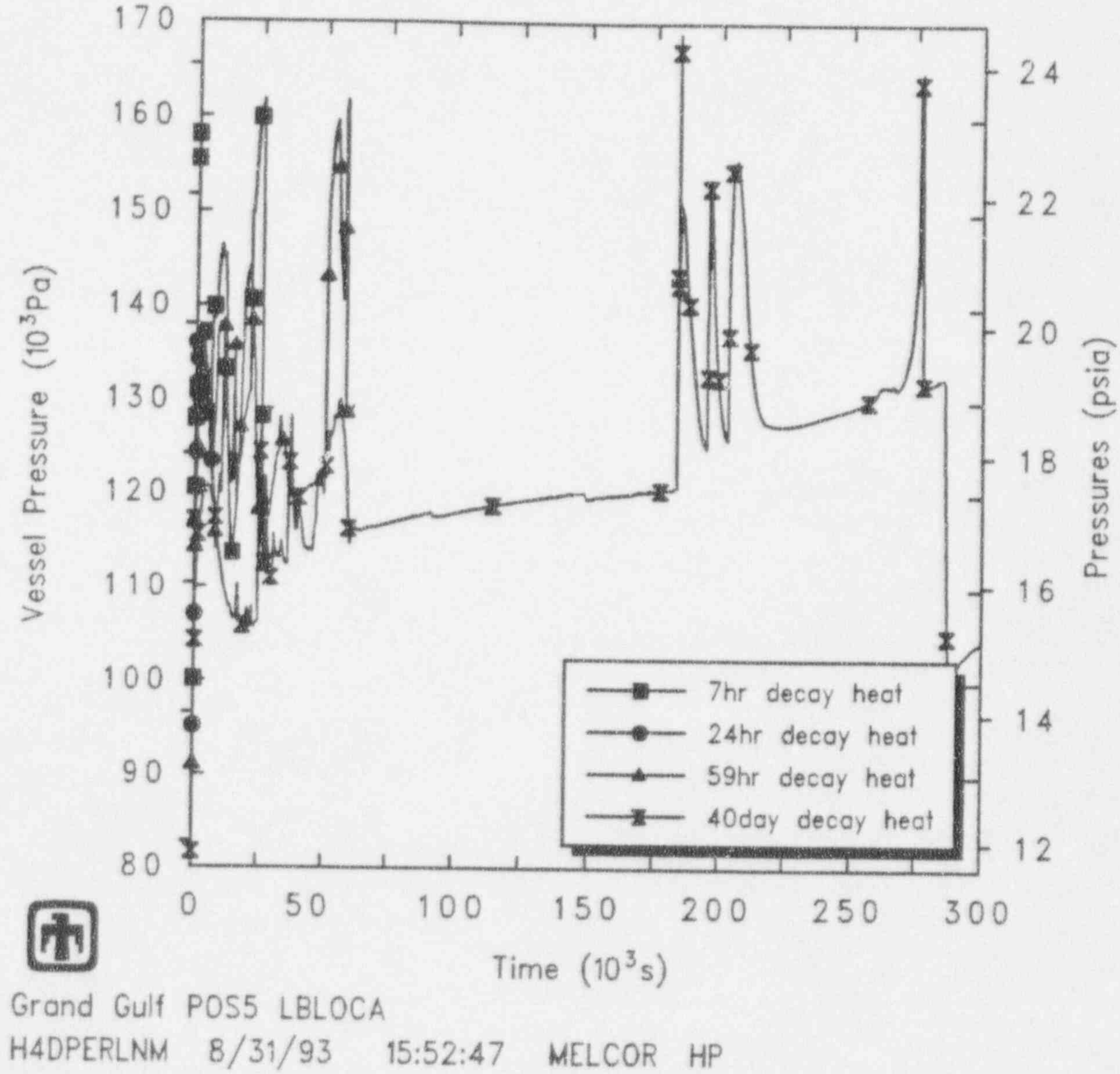


Figure 4.2.5.1. Reactor Vessel Pressures for Grand Gulf POS 5 – Large Break LOCA, Initiated at Various Times After Shutdown

Table 4.2.5.1. Key Event Times for Grand Gulf POS 5 - Large Break LOCA, Initiated at Various Times After Shutdown

	Time to (s)			
Initiation Time After Shutdown	TAF Uncovery†	Core Heatup	First Gap Release	Vessel Failure
7 hr	61	500	3,875	21,030
24 hr	62	1,000	5,445	33,850
59 hr	65	1,500	7,125	50,475
40 day	71	4,500	22,200	183,500

†Collapsed liquid level

auxiliary building, as shown in Figure 4.2.5.2. The longer after shutdown and scram that this accident sequence begins, the lower the decay heat and the longer it takes to fail the auxiliary building. The auxiliary building pressure rises somewhat more slowly during the early stages of core uncovery, heatup and damage, then spikes up to the failure point at vessel failure.

The coolant inventory in the vessel drops due to coolant and steam loss out the break, with a very rapid loss of about 60-70% of the inventory as liquid followed by a more gradual loss of the remaining inventory due to boiling and steam outflow, as presented in Figure 4.2.5.3. The amount of liquid inventory lost in the initial liquid blowdown is determined by the elevation of the break and is therefore about the same regardless of the decay heat level; later, as would be expected, the gradual inventory loss due to continued steaming is faster for higher decay heat levels.

The upper plenum and core liquid levels drop very quickly as the break drains the vessel to 2/3 core height, within seconds or minutes, and are not shown for this accident scenario.

The early core heatup is illustrated in Figures 4.2.5.4 through 4.2.5.7, as calculated for LBLOCA accidents initiated at 7 hr, 24 hr, 59 hr and 40 days after shutdown. Core uncovery and heatup begins sooner and proceeds more rapidly at higher decay heat levels.

Table 4.2.5.1 summarizes the timings of various key events predicted using MELCOR for this sequence assuming various times after shutdown and associated decay heat levels.

4.2.6 Station Blackout with Failure to Isolate SDC

The accident is initiated by a loss of offsite power with the reactor vessel depressurized and the coolant at the normal level. The vessel water inventory is at 366.5 K (200°F), which corresponds to the maximum temperature allowed by the Grand Gulf technical

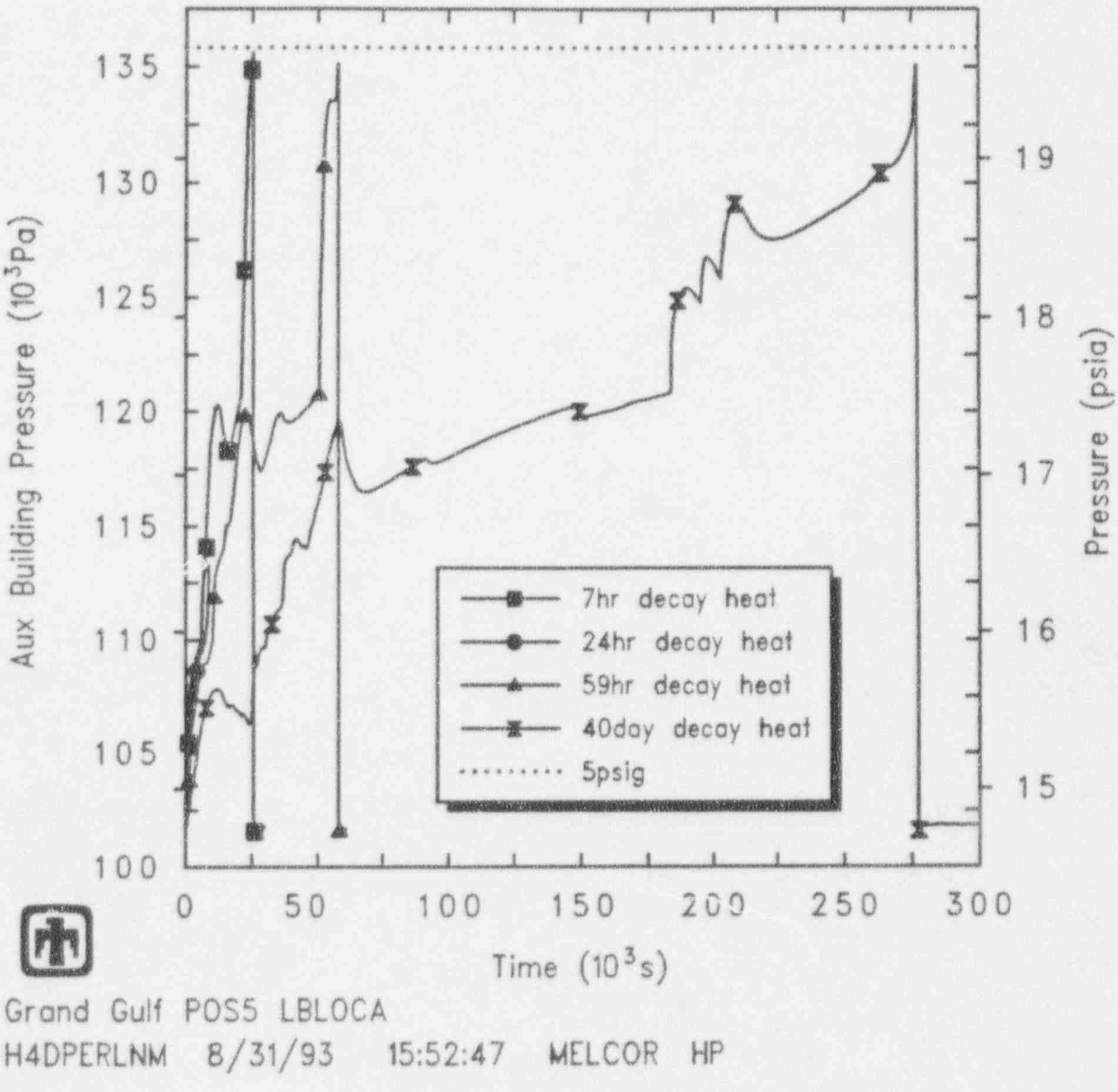


Figure 4.2.5.2. Auxiliary Building Pressures for Grand Gulf POS 5 – Large Break LOCA, Initiated at Various Times After Shutdown

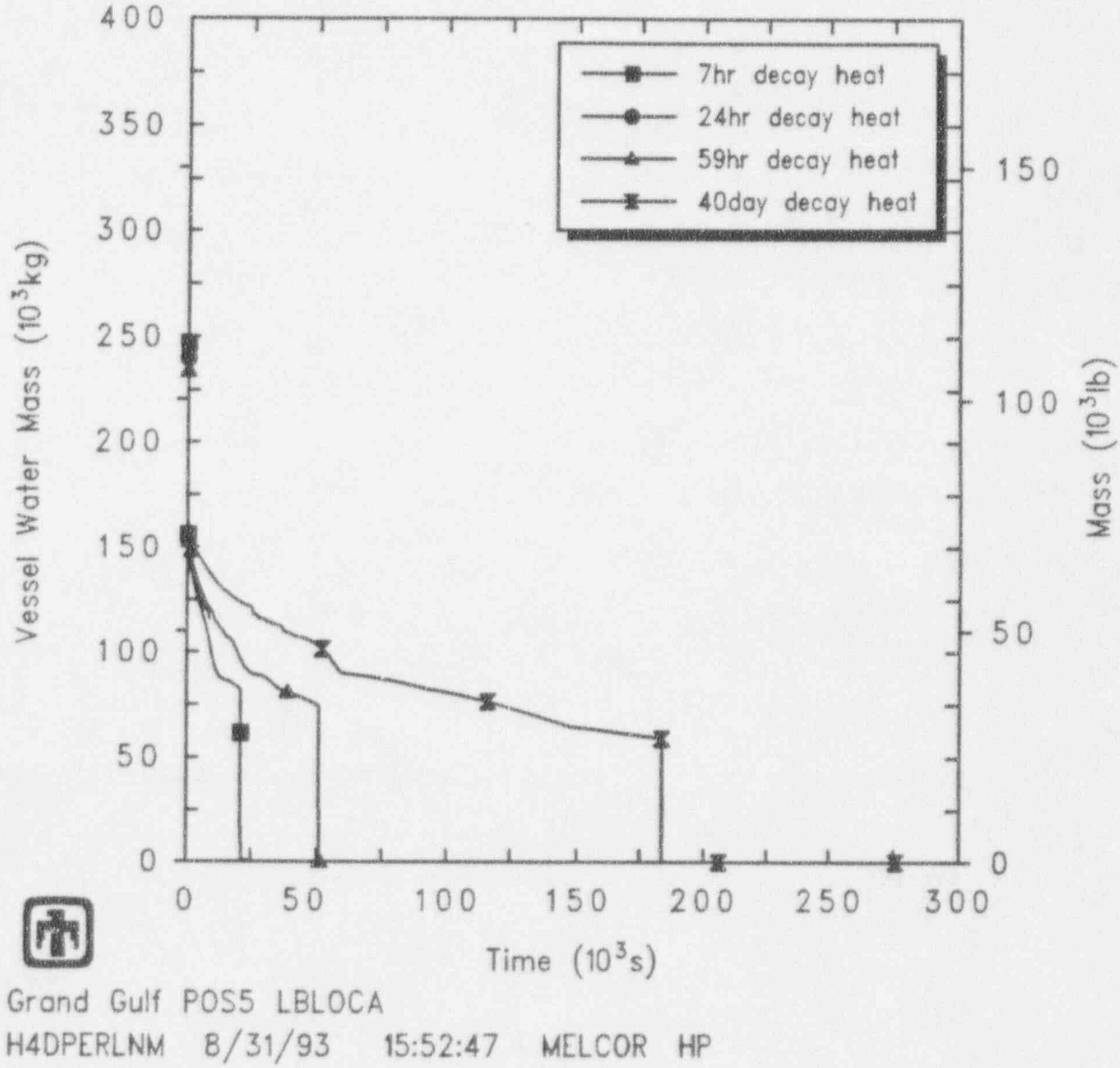


Figure 4.2.5.3. Reactor Vessel Water Masses for Grand Gulf POS 5 – Large Break LOCA, Initiated at Various Times After Shutdown

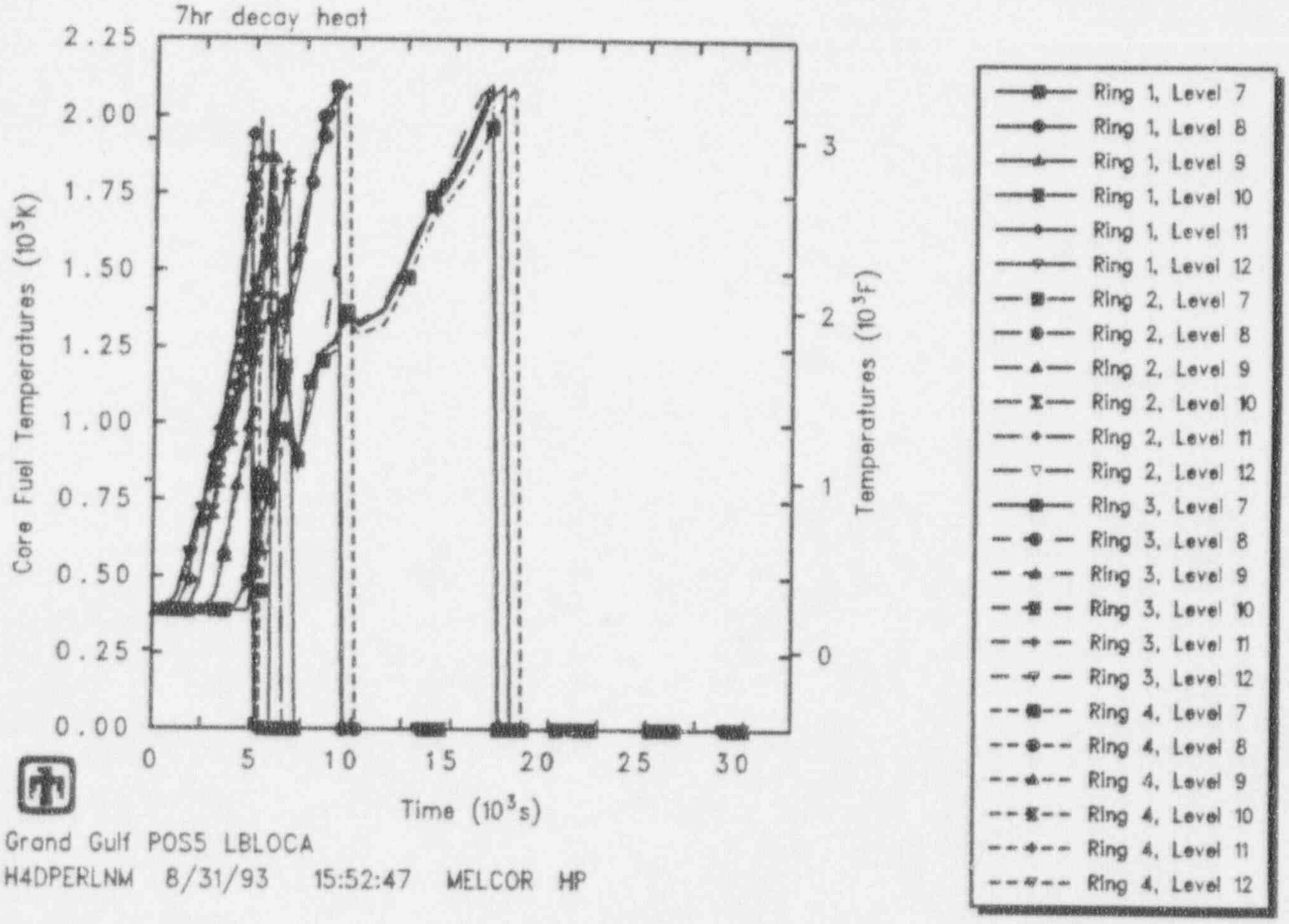


Figure 4.2.5.4. Core Fuel Temperatures for Grand Gulf POS 5 – Large Break LOCA, Initiated 7 hr After Shutdown

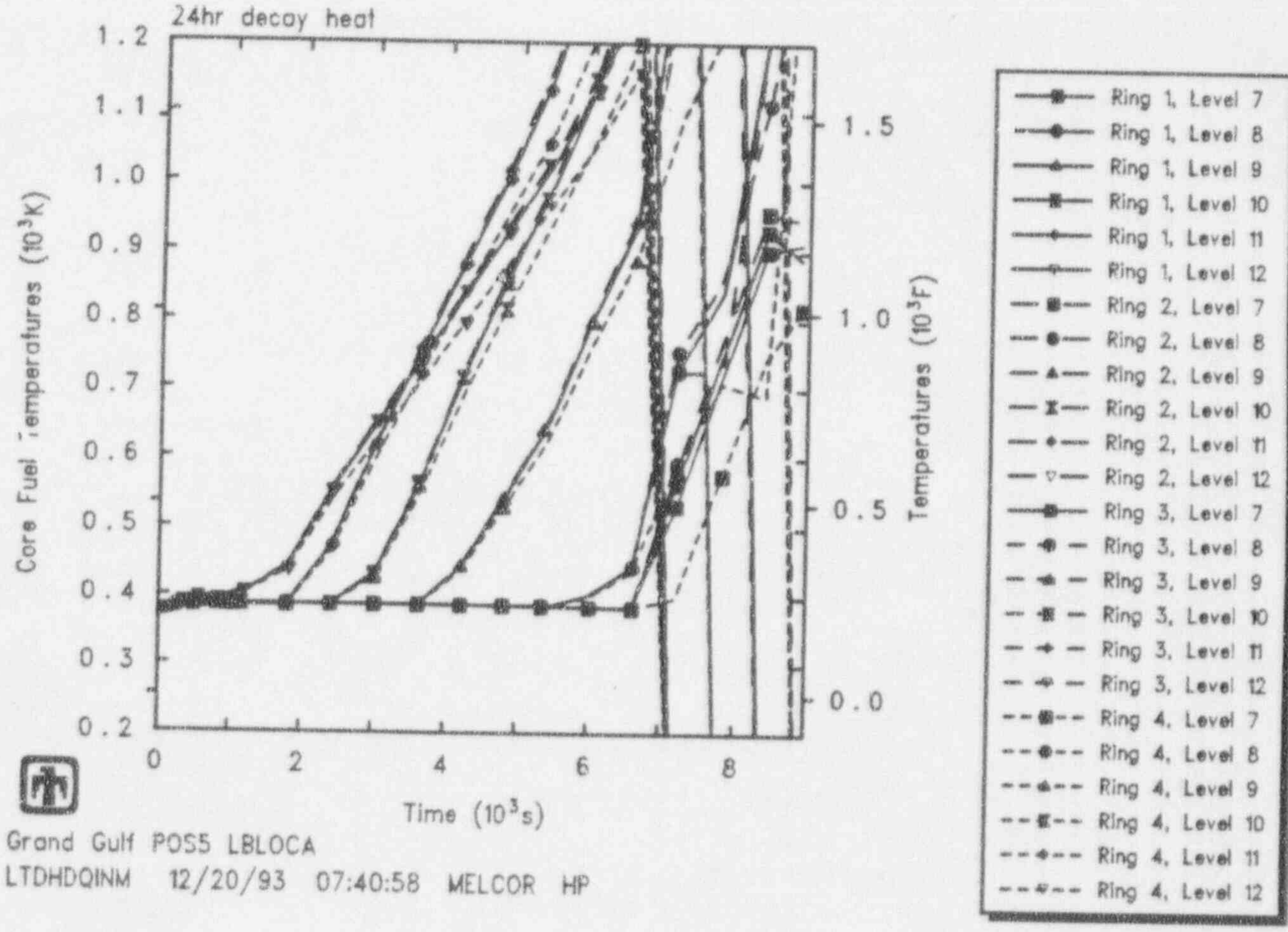


Figure 4.2.5.5. Core Fuel Temperatures for Grand Gulf POS 5 – Large Break LOCA, Initiated 24 hr After Shutdown

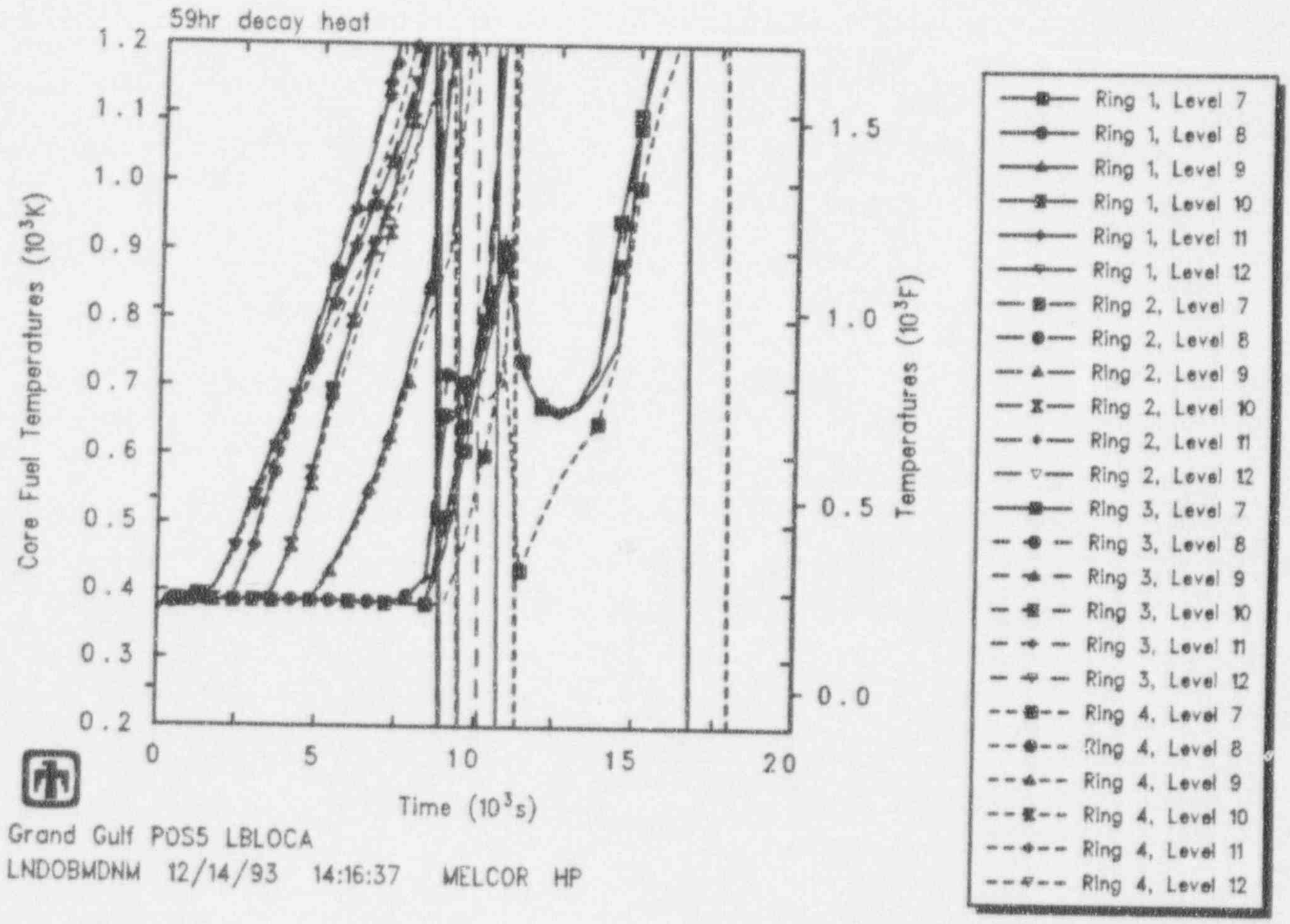


Figure 4.2.5.6. Core Fuel Temperatures for Grand Gulf POS 5 – Large Break LOCA, Initiated 59 hr After Shutdown

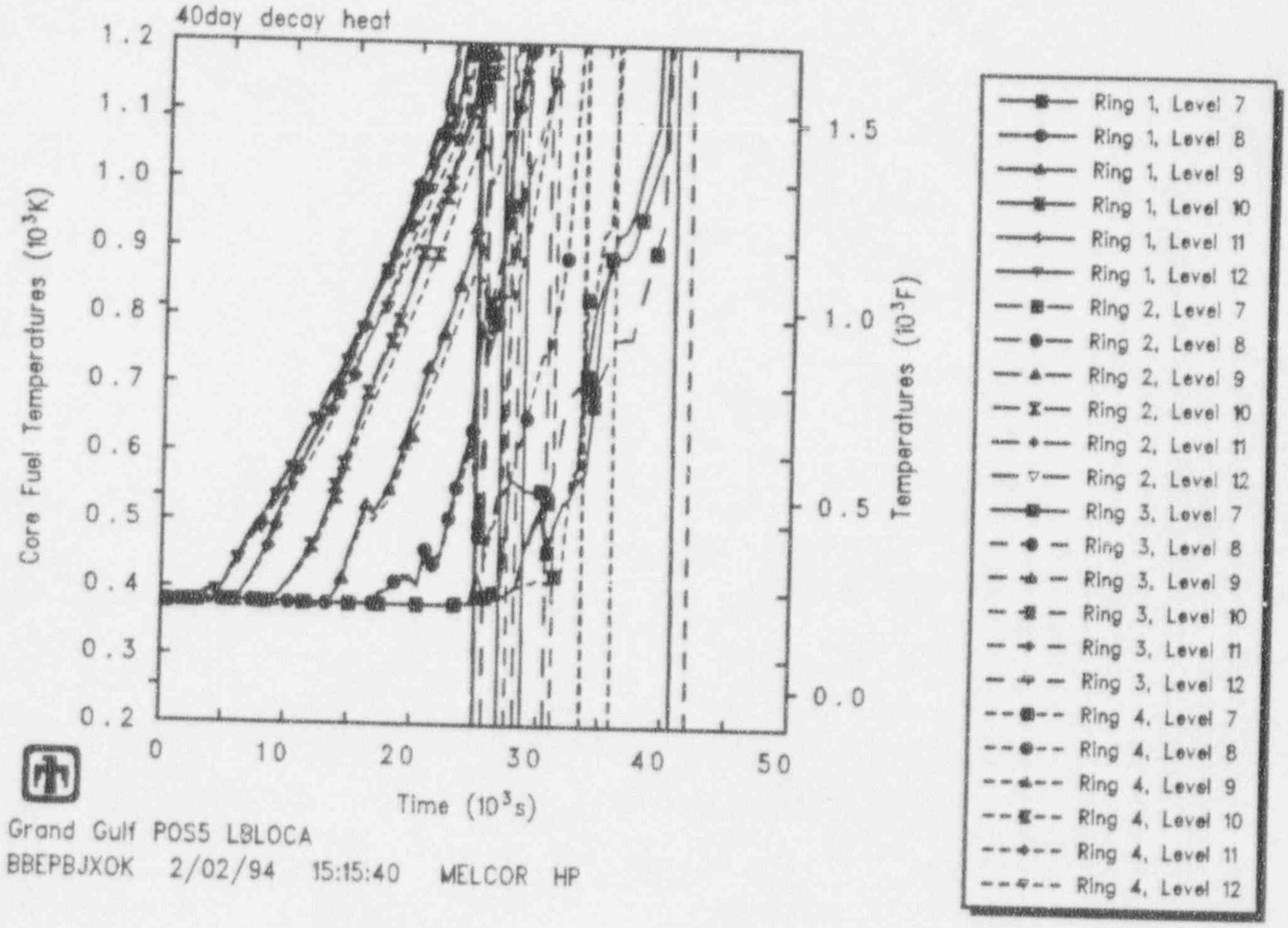


Figure 4.2.5.7. Core Fuel Temperatures for Grand Gulf POS 5 – Large Break LOCA, Initiated 40 day After Shutdown

specifications for operation in POS 5. Following the initiating event, onsite power is lost leading to a SBO and loss of all core cooling and coolant makeup. The operator fails to open the SRVs and steam the core at low pressure (*i.e.*, the SRVs operate in the relief mode). Since the SRVs are closed, the RPV will pressurize. The SBO precludes the isolation of the low pressure piping in the SDC system. This low-pressure SDC system piping fails when the RPV pressure reaches 3.135 MPa (440 psig), resulting in an interfacing systems LOCA with outflow from the vessel downcomer to the first floor of the auxiliary building. The drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open (*i.e.*, "open containment").

Figure 4.2.6.1 presents the vessel pressures calculated starting this accident scenario at several different times after scram; Figure 4.2.6.1 also includes lines at 440 psig, the postulated SDC break setpoint, and at 160 psig, a pressure signal of interest because it is the failure pressure for any shutdown cooling provided by the ADHRS. In all cases, the system begins pressurizing as all core cooling is lost. For most decay heat levels the primary system pressurizes to 3.135 MPa (440 psig), which actuates the postulated SDC break; however, for a decay heat level corresponding to 40 days after shutdown, relief through the open RPV vent line is sufficient to cause the primary system pressure to begin dropping before reaching the SDC break setpoint. The flow out the SDC line break goes directly to the auxiliary building first floor and pressurizes the auxiliary building, as indicated in Figure 4.2.6.2. Even with the SDC break remaining closed for the sequence initiated 40 days after shutdown, the flow out the open RPV vent line pressurizes the containment and, through the open equipment hatch and personnel locks, pressurizes the auxiliary building. As expected, the lower the decay heat the slower the auxiliary building pressurizes and the longer it takes to fail the auxiliary building.

The coolant inventory in the vessel drops as the decay heat boils water to steam which is lost out the SDC break and the open RPV vent, faster for higher decay heat levels, as presented in Figure 4.2.6.3. The opening of the SDC break is reflected in the extremely rapid loss of about 75% of the vessel inventory seen at various times; that inventory loss then slows down when the break uncovers, until subsequent vessel failure.

Figure 4.2.6.4 gives the predicted upper plenum liquid level drop due to this inventory loss, for different decay heat levels and highlighting when a Level 3 trip (544.4 in) would be generated. In all cases, the upper plenum level initially rises as the primary system pressurizes and then falls rapidly when the SDC break is opened. For lower decay heat level s (*i.e.*, longer after shutdown), the upper plenum level peaks and begins dropping steadily before the SDC break opens, due to flow out the open RPV vent.

Figure 4.2.6.5 gives the corresponding core liquid level drop due to this inventory loss, for different decay heat levels and highlighting when TAF uncovery is calculated to occur; horizontal lines indicate both the boundary between the upper plenum and the core at 9.6 m and the top-of-active-fuel elevation at 9.3 m. Note that, for decay heat levels such that the primary system pressurizes sufficiently to open the postulated SDC break, the core liquid levels drop precipitously when the SDC break opens, as did the upper plenum liquid levels also. The behavior is qualitatively different for a decay heat level low enough that relief through the open RPV vent line is sufficient to cause

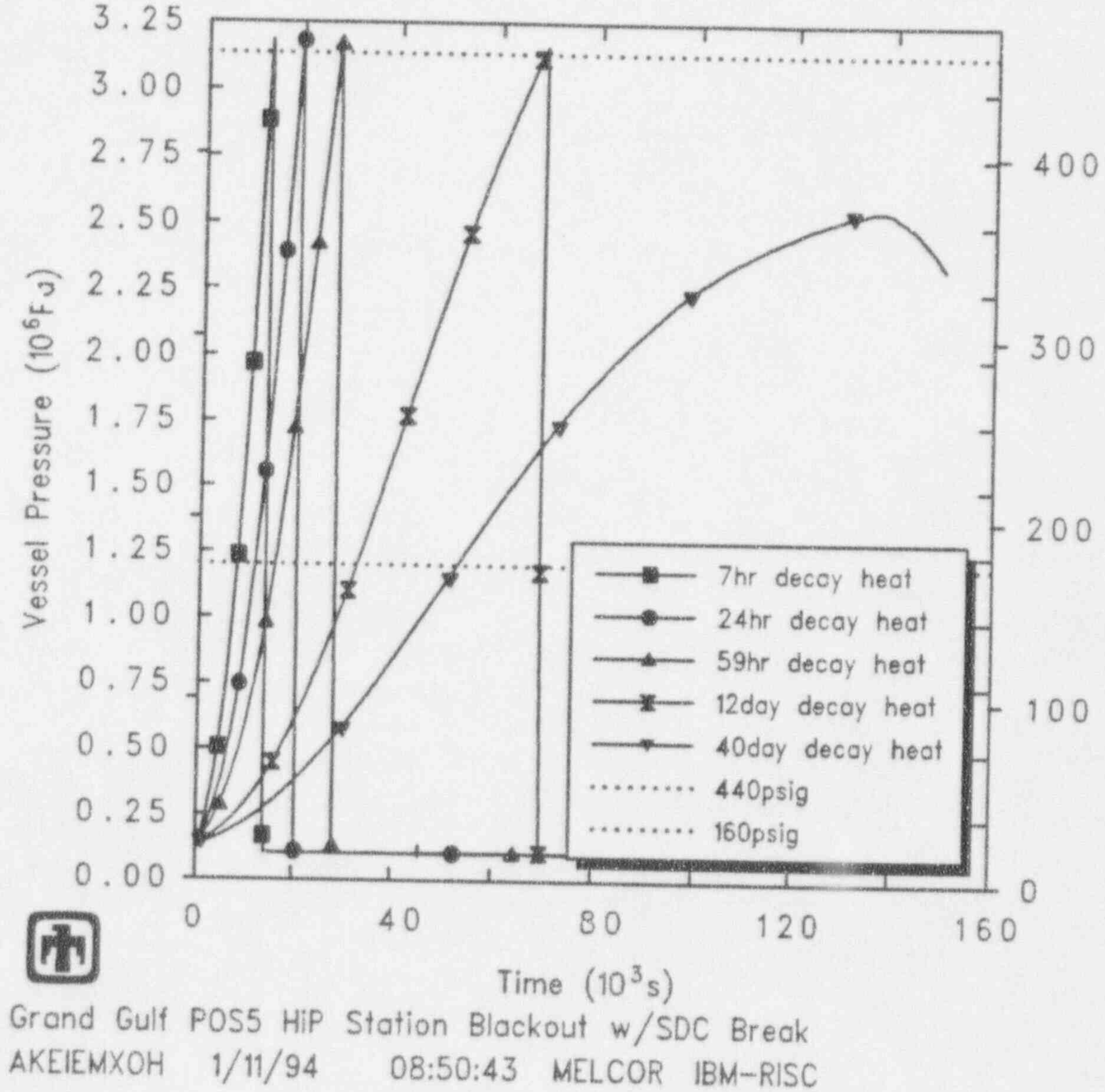


Figure 4.2.6.1. Reactor Vessel Pressures for Grand Gulf POS 5 - Station Blackout with SDC Break, Initiated at Various Times After Shutdown

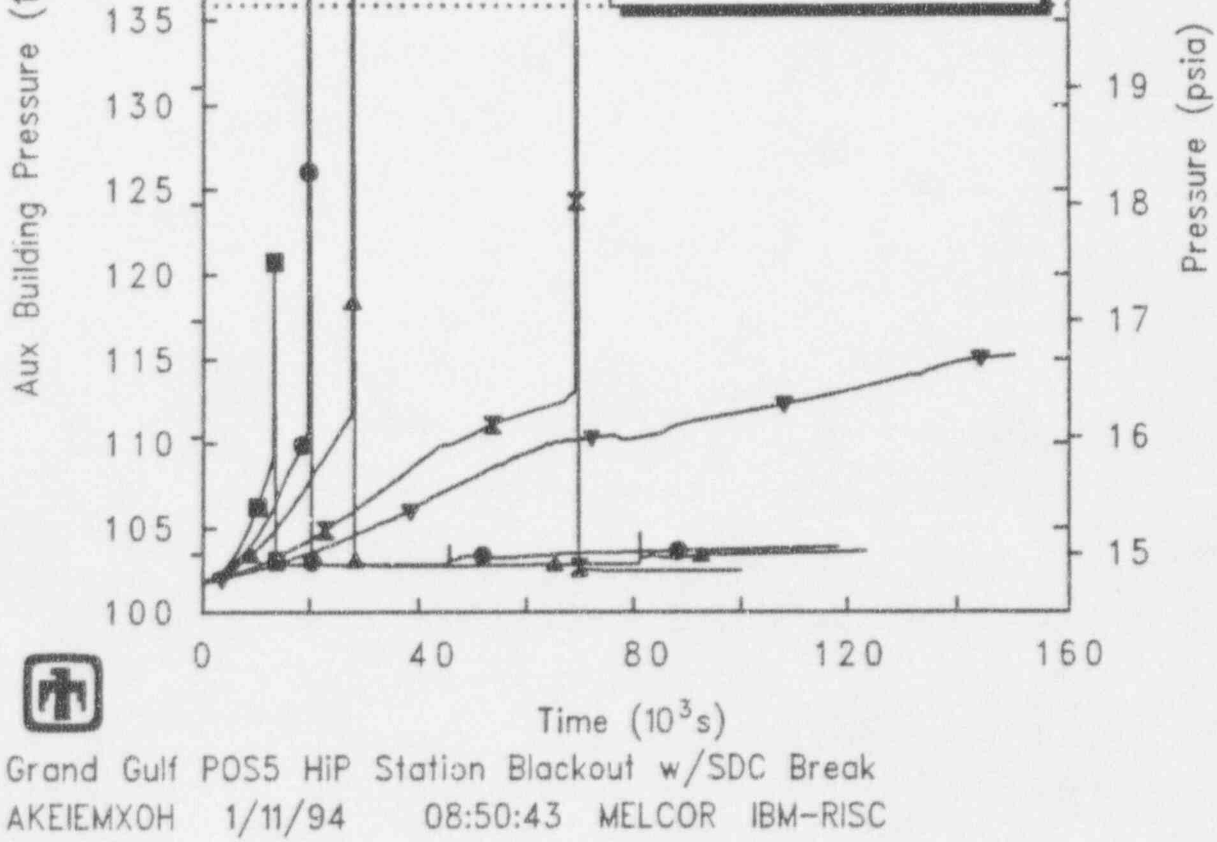


Figure 4.2.6.2. Auxiliary Building Pressures for Grand Gulf POS 5 - Station Blackout with SDC Break, Initiated at Various Times After Shutdown

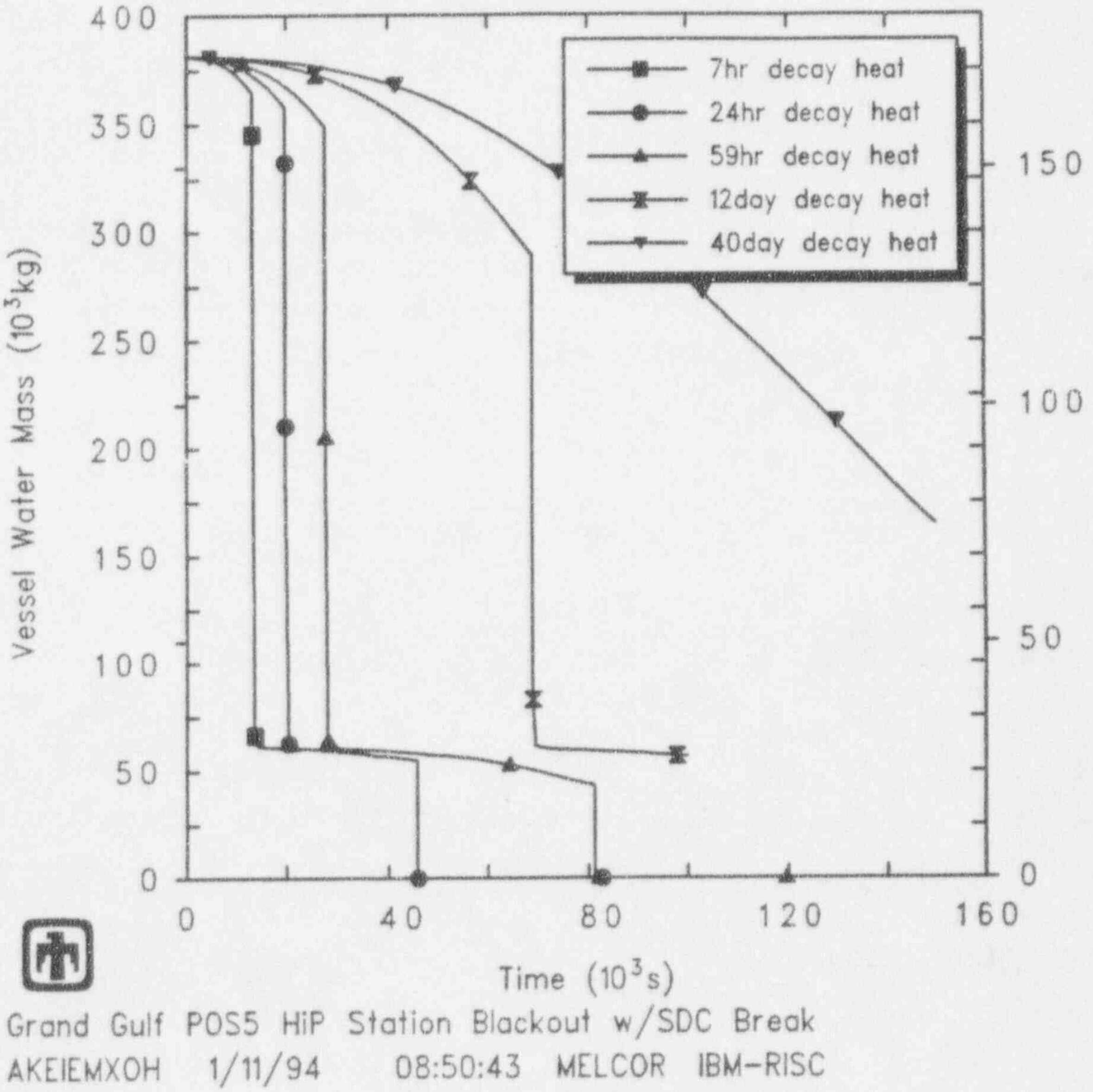


Figure 4.2.6.3. Reactor Vessel Water Masses for Grand Gulf POS 5 - Station Blackout with SDC Break, Initiated at Various Times After Shutdown

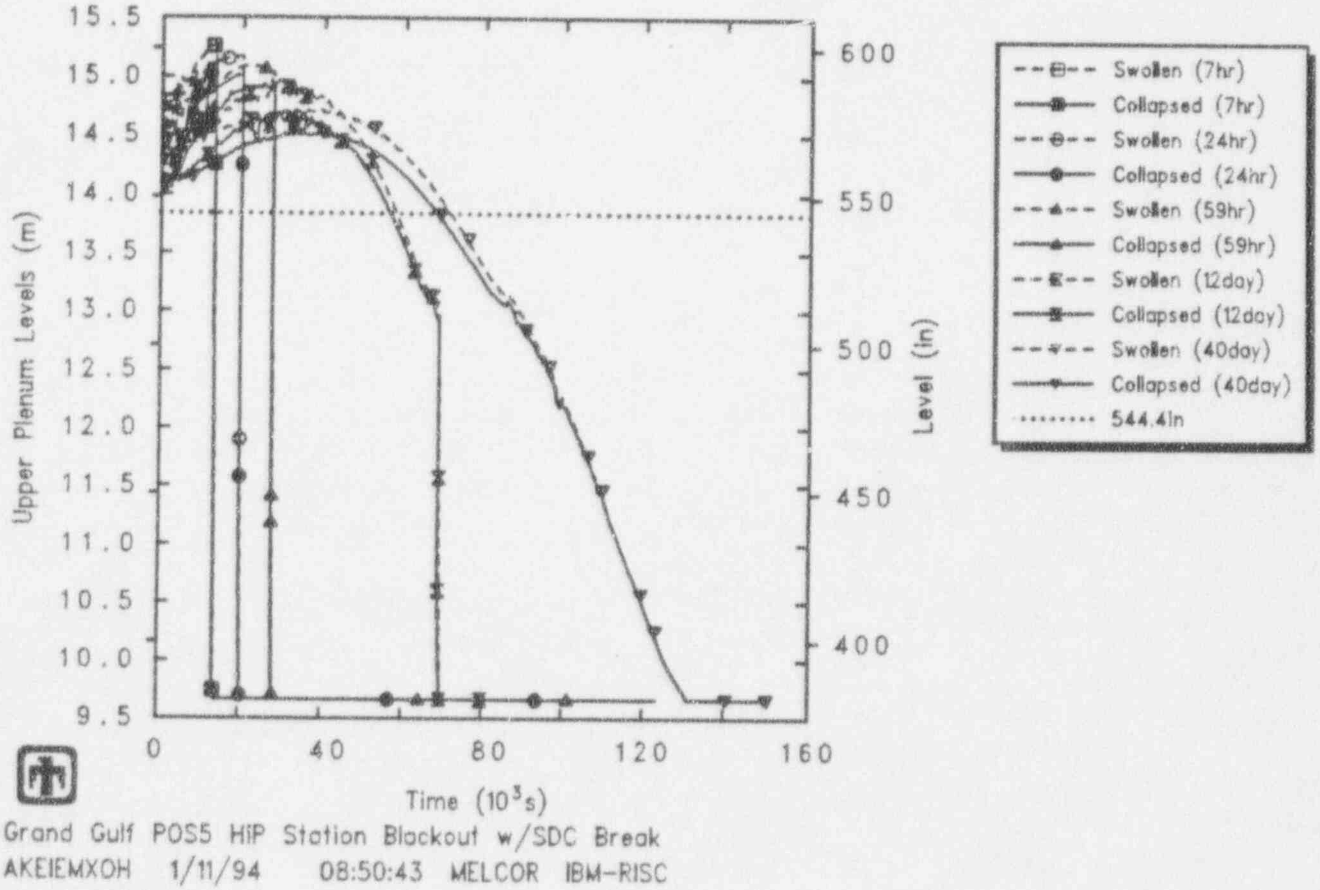


Figure 4.2.6.4. Upper Plenum Liquid Levels for Grand Gulf POS 5 – Station Blackout with SDC Break, Initiated at Various Times After Shutdown

the primary system pressure to begin dropping before reaching the SDC break setpoint. While the upper plenum levels are dropping gradually, as illustrated in Figure 4.2.6.4, the uppermost core is being uncovered slowly and intermittently; after the upper plenum has uncovered completely the core then begins sustained uncovery.

The early core heatup is illustrated in Figures 4.2.6.6 through 4.2.6.10, as calculated for accident sequences initiated by station blackouts at 7 hr, 24 hr and 59 hr, and 12 day and 40 day, after shutdown. As with TAF uncovery, core uncovery begins sooner and proceeds more rapidly at higher decay heat levels. The calculation begun 40 days after shutdown showed core heatup only beginning when the calculation was stopped at ~150,000 s; the calculation was stopped because this was long after the 1 day (86,400 s) maximum time period of interest for these Level 1 analyses.

Tables 4.2.6.1 and 4.2.6.2 summarize the timings of various key events predicted using MELCOR for this sequence assuming various times after shutdown and associated decay heat levels.

4.2.7 Station Blackout with Firewater Addition

The accident is initiated by a loss of offsite power with the reactor vessel depressurized and the coolant at the normal level. The vessel water inventory is at 366.5 K (200°F), which corresponds to the maximum temperature allowed by the Grand Gulf technical specifications for operation in POS 5. Following the initiating event, onsite power is lost leading to a SBO and loss of all core cooling and coolant makeup. The operator opens two SRVs at 2 hr and steams the core at low pressure while adding coolant from the firewater system. The drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open (*i.e.*, "open containment").

Figure 4.2.7.1 presents the vessel pressures calculated starting this accident scenario at two different times after scram. Initially, the system begins pressurizing as all core cooling is lost, more quickly for higher decay heat; the pressure then begins dropping after two SRVs are opened 2 hr after the start of the accident. The flow out the open RPV vent line and later out the SRVs also pressurizes the containment and the auxiliary building, as indicated in Figure 4.2.7.2, more rapidly for higher decay heat.

Although the operator aligns the firewater system to inject coolant into the vessel starting at 2 hr after accident initiation, injection does not begin until the vessel has depressurized sufficiently (as determined by the pump characteristics). Figure 4.2.7.3 shows that firewater can be injected as soon as desired if the accident is assumed to start 24 hr after shutdown, but firewater injection can not begin until the vessel is depressurized for about 4 hr if the accident is assumed to start 7 hr after shutdown (a higher decay heat level). At the lower decay heat the firewater injection quickly rises to its maximum level after beginning, while at higher decay heat levels the firewater injection rises to its maximum level more slowly as the vessel continues to depressurize through the open SRVs.

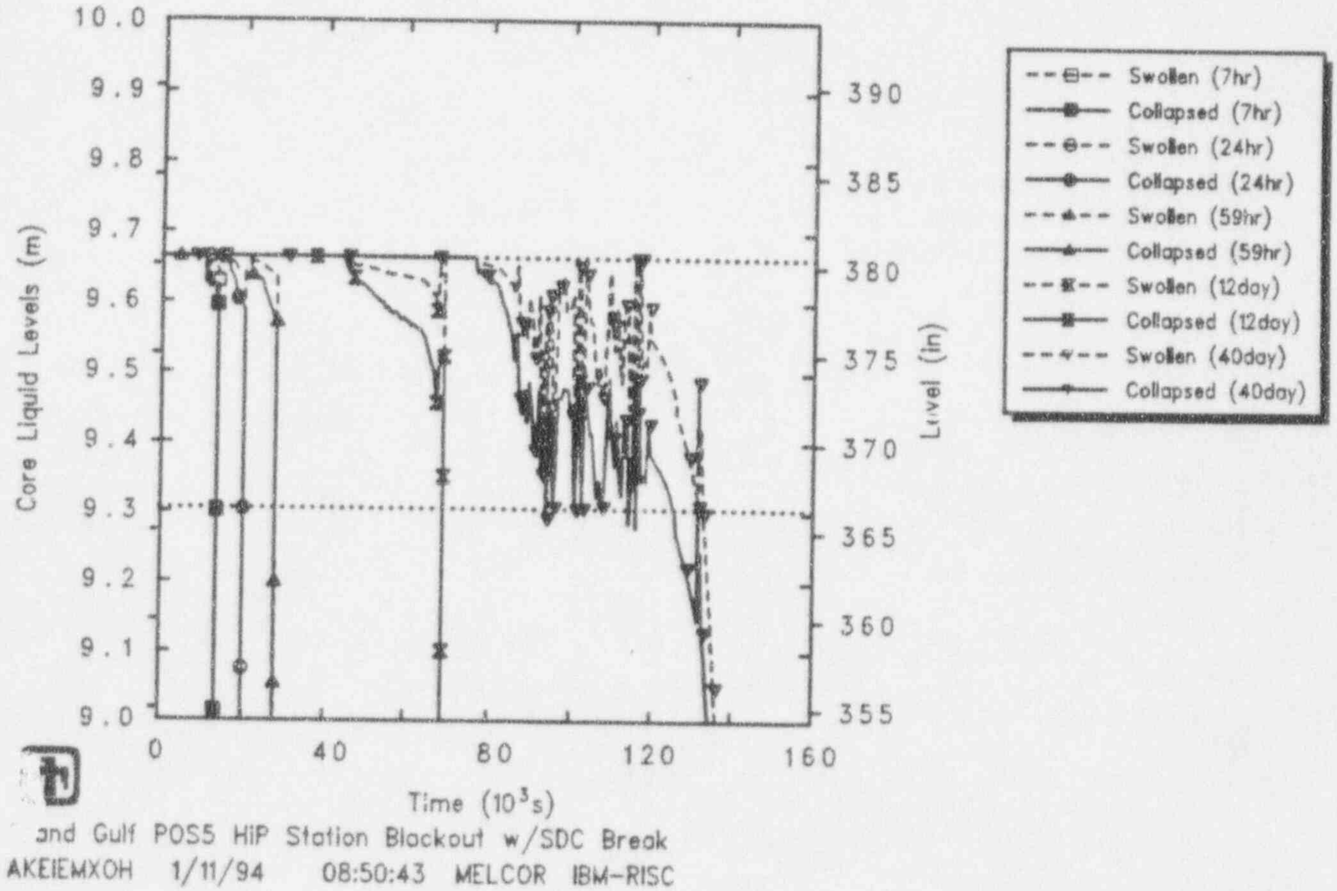


Figure 4.2.6.5. Core Liquid Levels for Grand Gulf POS 5 - Station Blackout with SDC Break, Initiated at Various Times After Shutdown

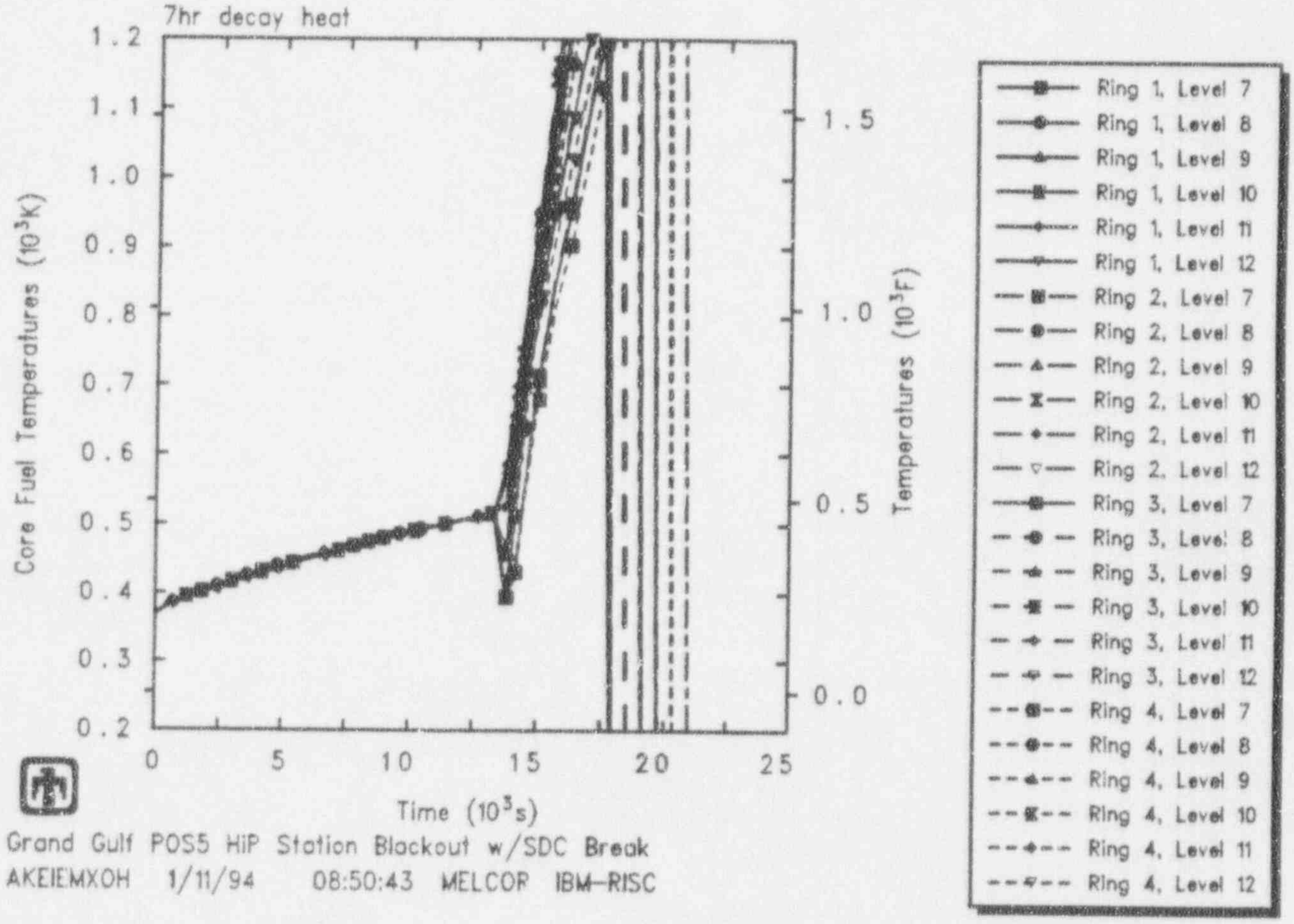


Figure 4.2.6.6. Core Fuel Temperatures for Grand Gulf POS 5 – Station Blackout with SDC Break, Initiated 7 hr After Shutdown

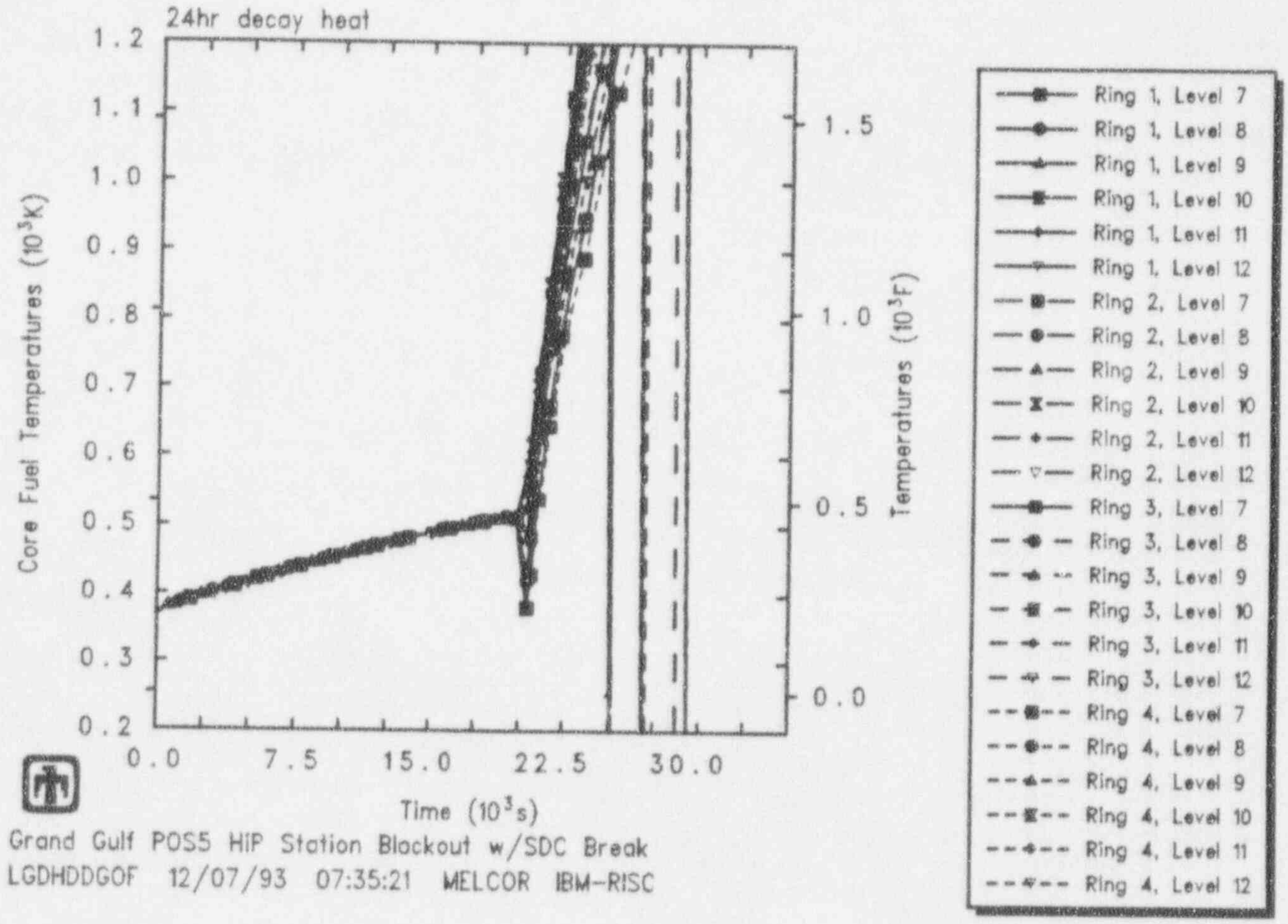


Figure 4.2.6.7. Core Fuel Temperatures for Grand Gulf POS 5 - Station Blackout with SDC Break, Initiated 24 hr After Shutdown

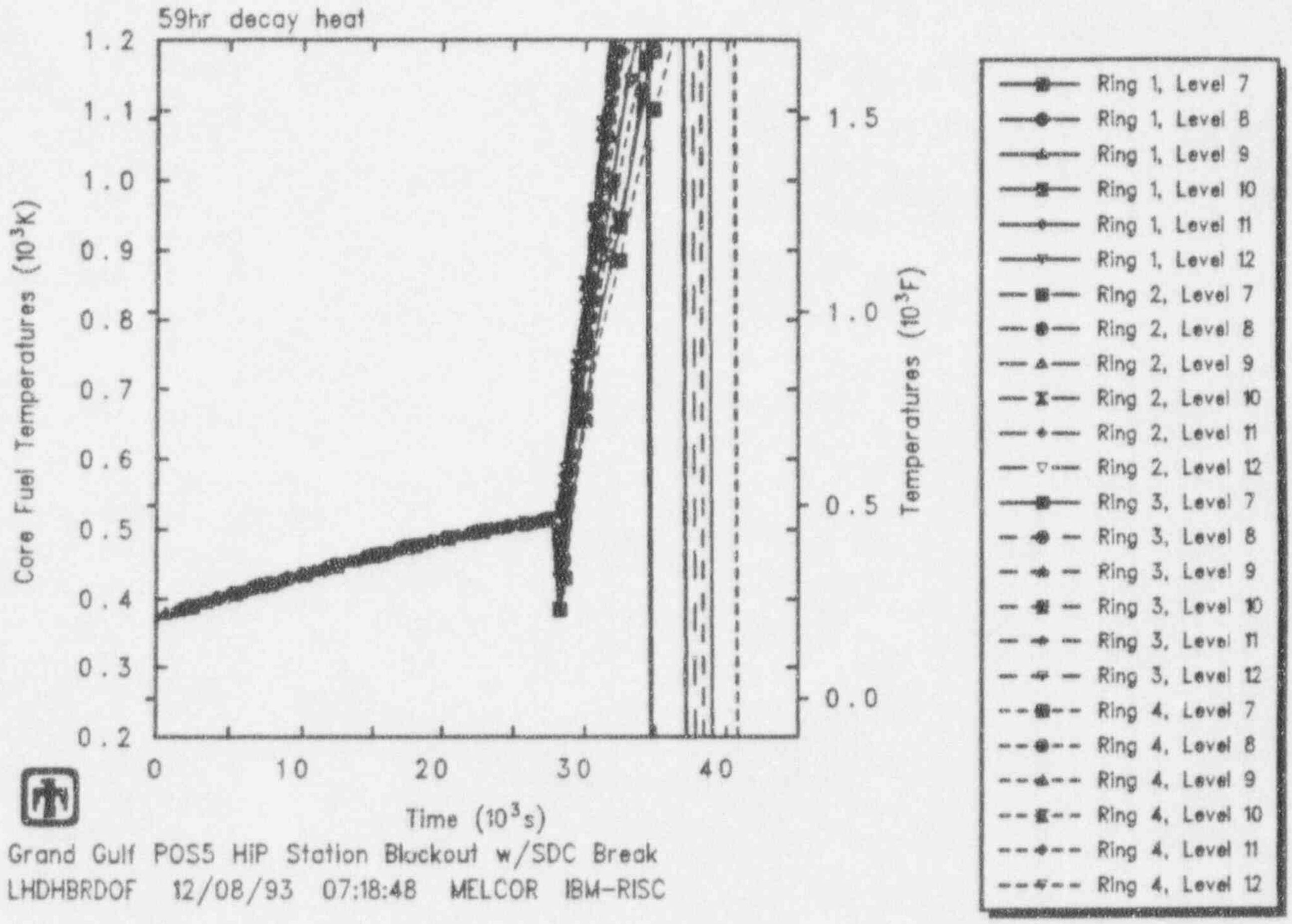


Figure 4.2.6.8. Core Fuel Temperatures for Grand Gulf POS 5 - Station Blackout with SDC Break, Initiated 59 hr After Shutdown

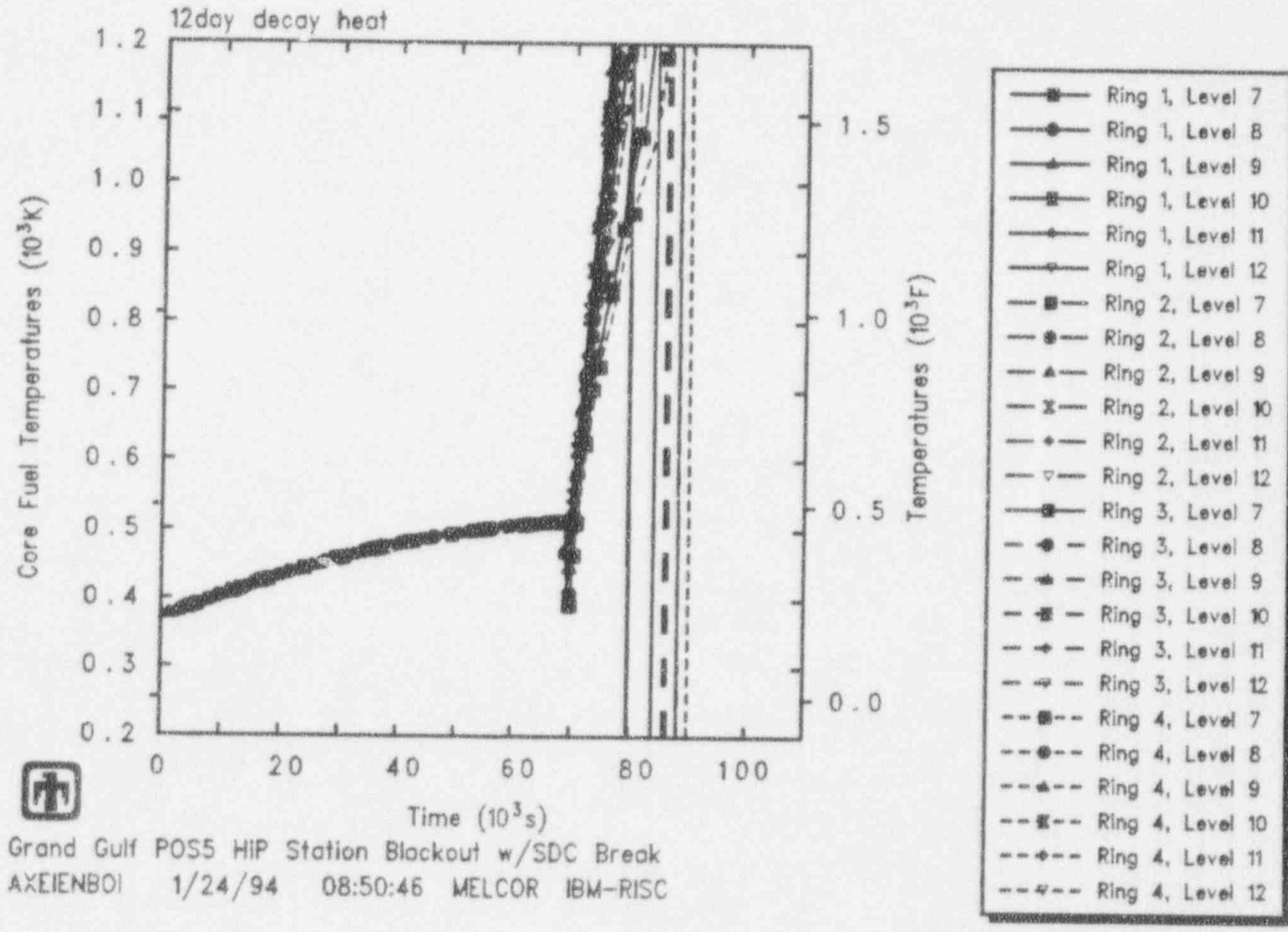


Figure 4.2.6.9. Core Fuel Temperatures for Grand Gulf POS 5 - Station Blackout with SDC Break, Initiated 12 day After Shutdown

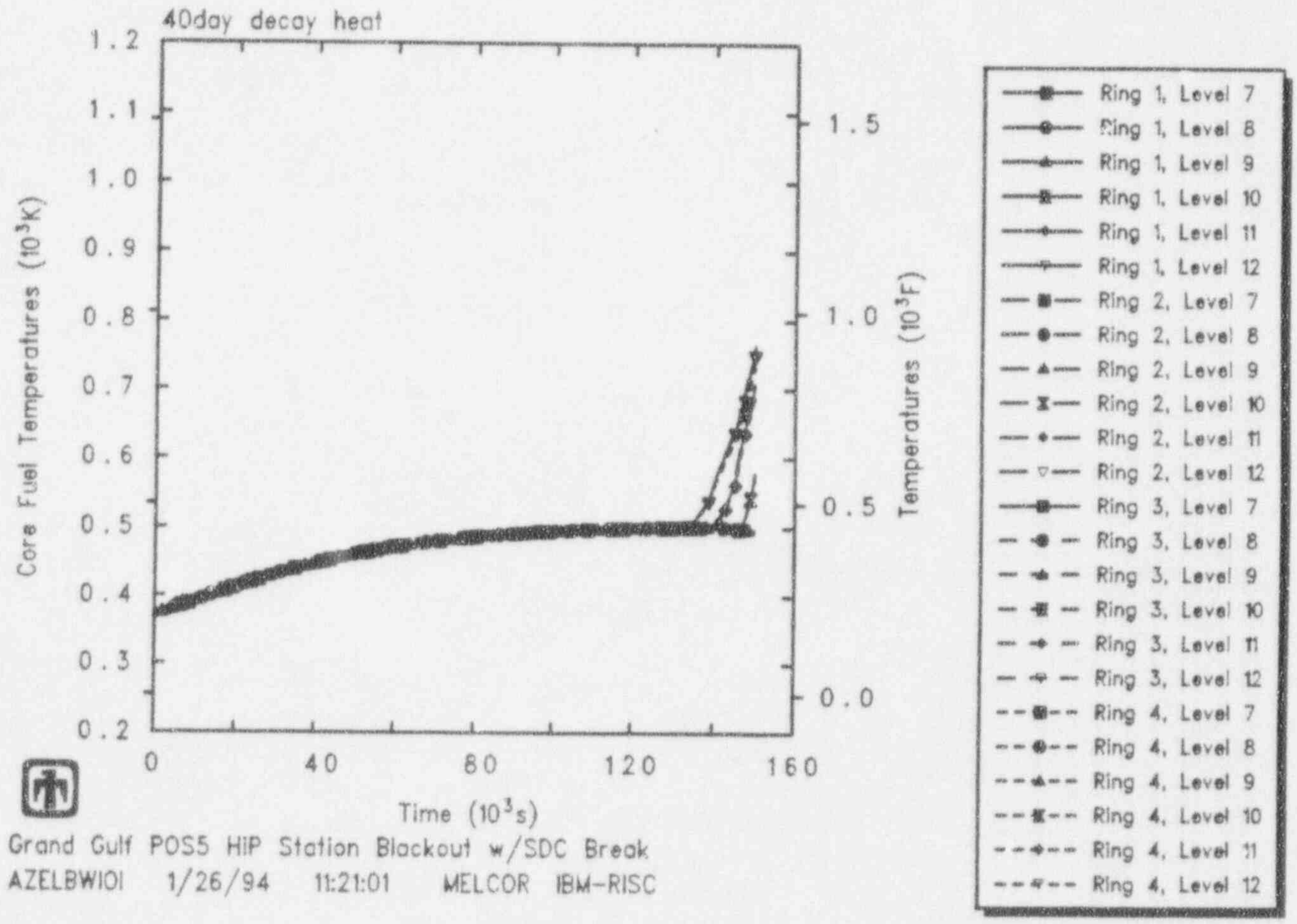


Figure 4.2.6.10. Core Fuel Temperatures for Grand Gulf POS 5 - Station Blackout with SDC Break, Initiated 40 day After Shutdown

Table 4.2.6.1. Key Event Times for Grand Gulf POS 5 - Station Blackout with SDC Break, Initiated at Various Times After Shutdown

	Time to (s)			
Initiation Time After Shutdown	TAF Uncovery	Core Heatup	First Gap Release	Vessel Failure
7 hr	13,300	31,600	15,685	51,770
24 hr	19,750	43,800	22,840	45,390
59 hr	26,200	58,400	31,570	81,135
12 day	75,600	75,600	82,800	-†
40 day	124,800	132,800	-	-†

†Calculation stopped before event occurred

Table 4.2.6.2. Key Signal Times for Grand Gulf POS 5 - Station Blackout with SDC Break, Initiated at Various Times After Shutdown

	Time to (s)			
Initiation Time After Shutdown	Collapsed Level <544.4 in	Swollen Level <544.4 in	P> 135 psia	P> 160 psig
7 hr				7,600
24 hr				11,400
59 hr				15,600
12 day				
40 day				52,200

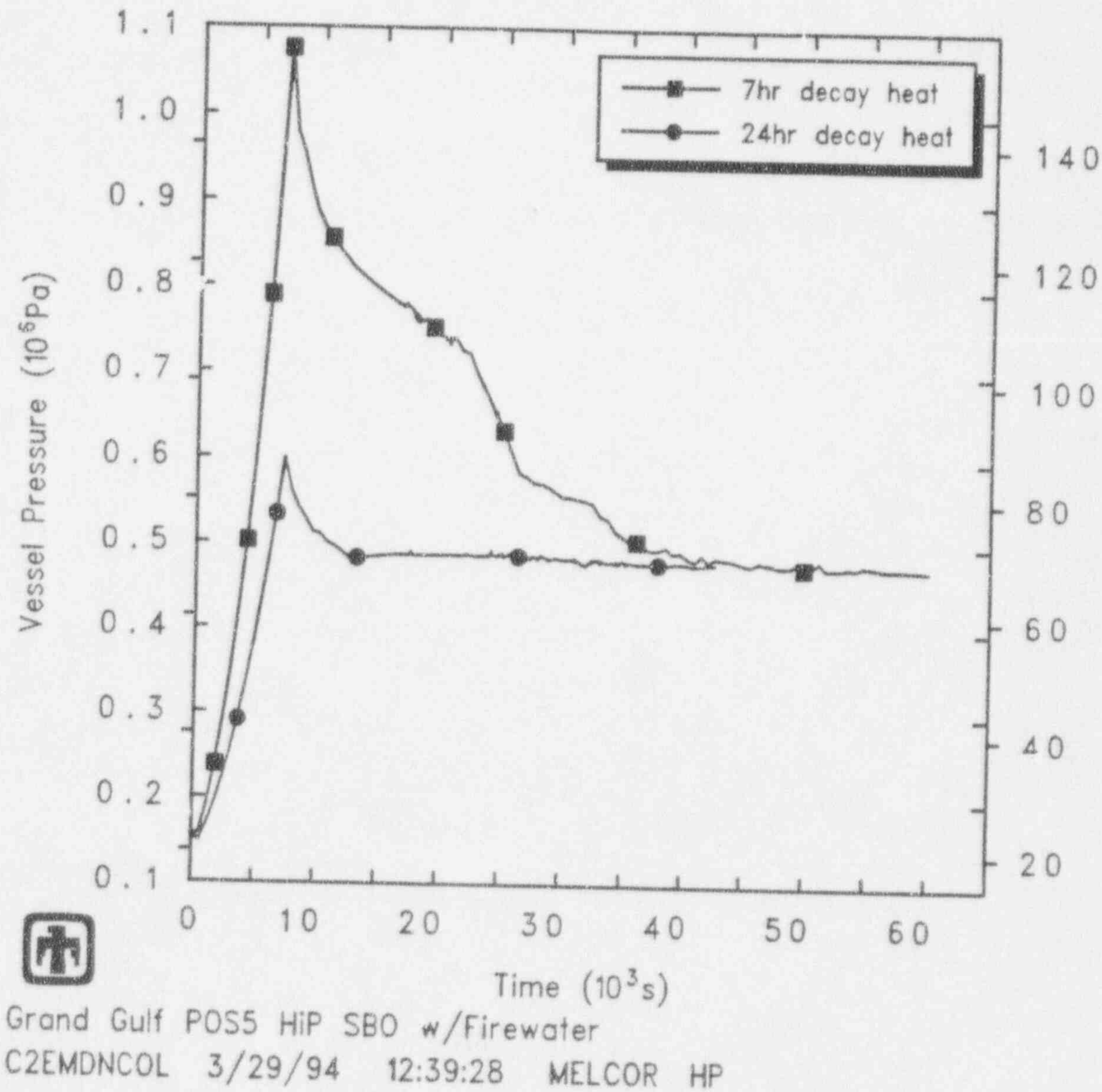


Figure 4.2.7.1. Reactor Vessel Pressures for Grand Gulf POS 5 – Station Blackout with Firewater, Initiated at Various Times After Shutdown

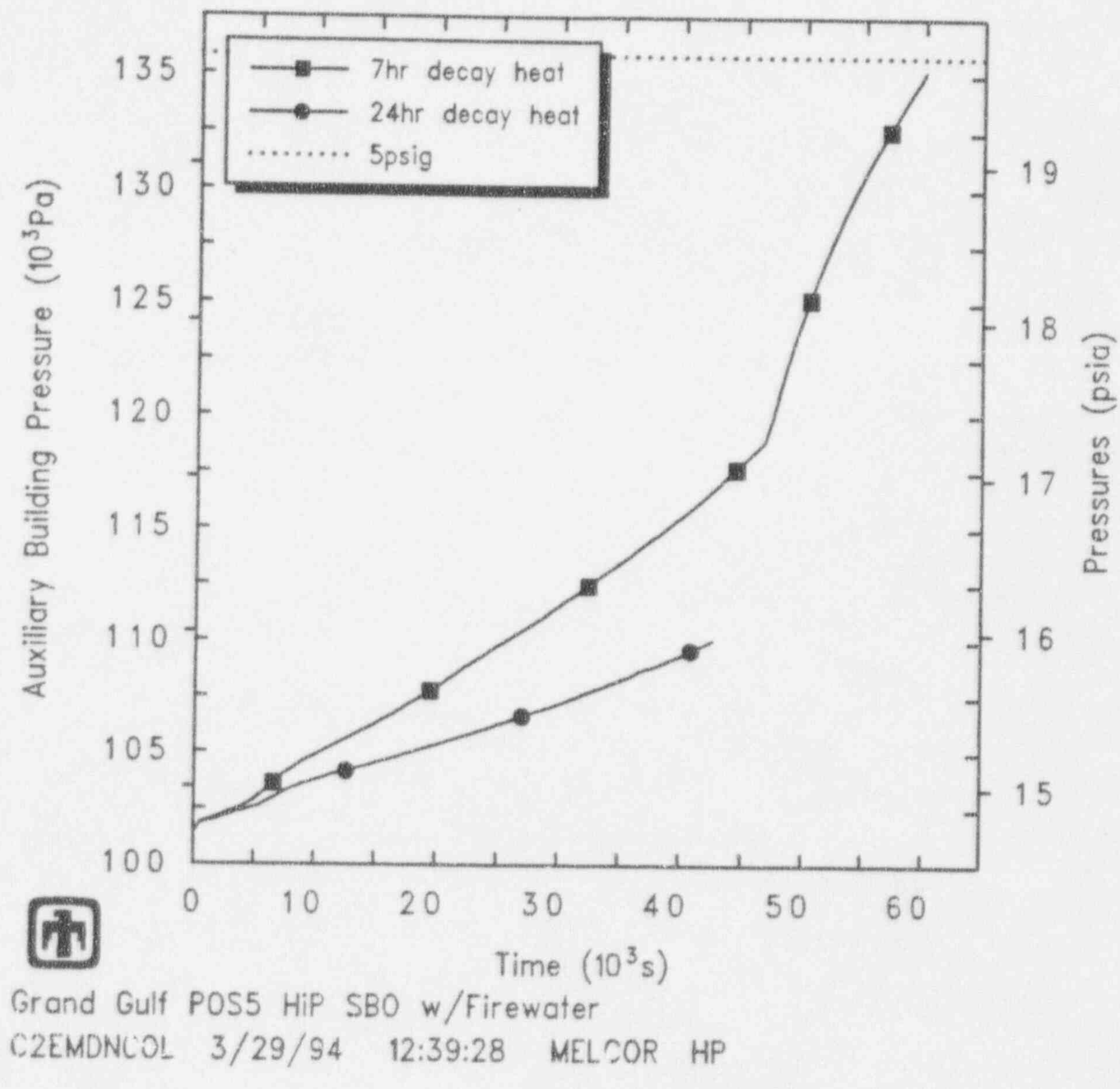


Figure 4.2.7.2. Auxiliary Building Pressures for Grand Gulf POS 5 – Station Blackout with Firewater, Initiated at Various Times After Shutdown

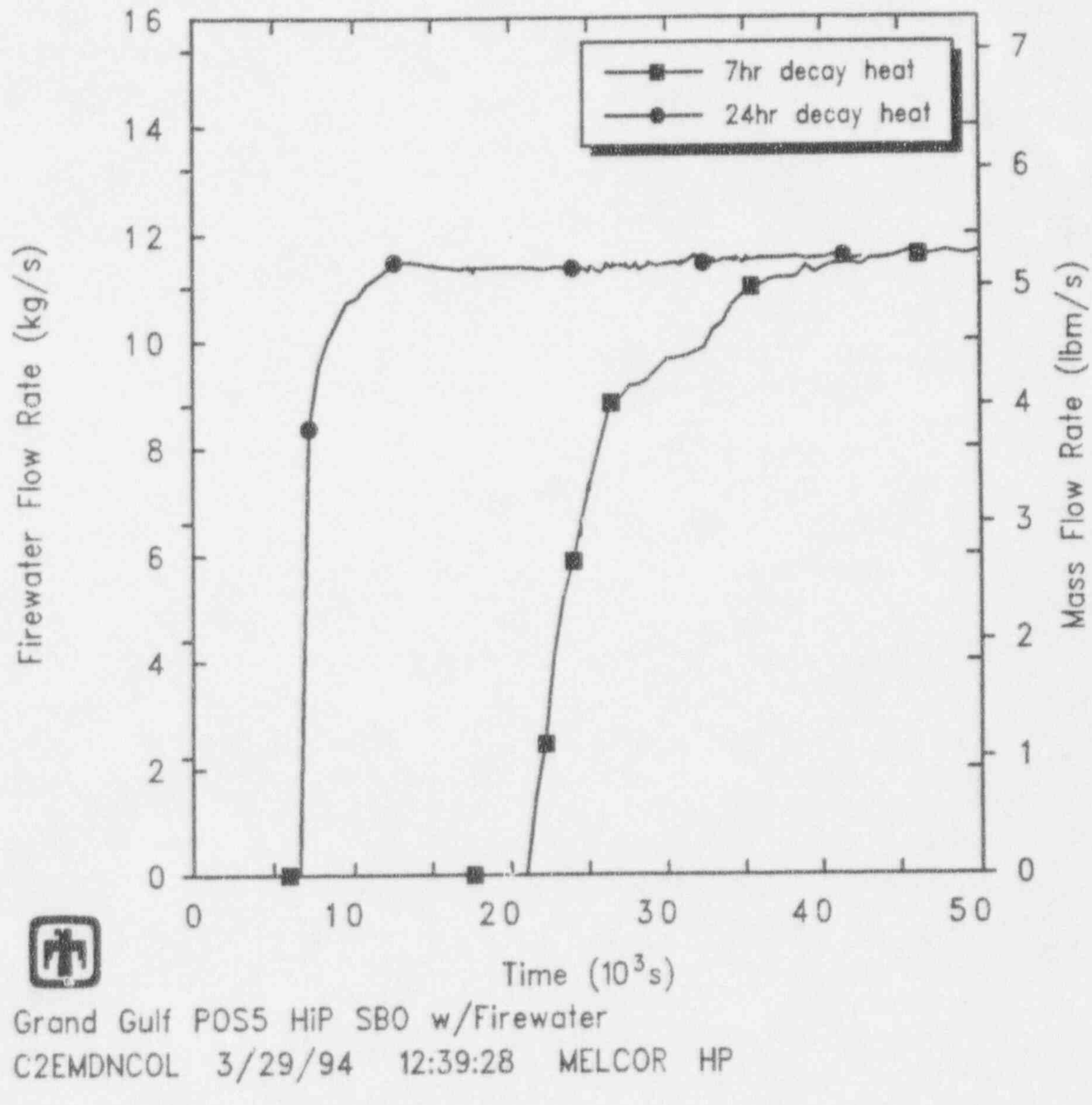


Figure 4.2.7.3. Firewater Injection Flow Rates for Grand Gulf POS 5 - Station Blackout with Firewater, Initiated at Various Times After Shutdown

Coolant addition from firewater is partially countered by increased steaming in the core and steam flow out the open SRVs. Figure 4.2.7.4 indicates that, at lower decay heats the firewater injection causes a net increase in vessel inventory, while at higher decay heat levels firewater injection does not equal and reverse inventory loss for about 5hr.

Figure 4.2.7.5 gives the predicted upper plenum liquid level drop due to this inventory loss, for different decay heat levels and highlighting when a Level 3 trip (544.4 in) would be generated. The upper plenum liquid levels reflect the overall vessel coolant inventory response presented in Figure 4.2.7.4 – at lower decay heats the upper plenum levels remain nearly constant, while at higher decay heat levels the upper plenum levels drop for about 5hr after the SRVs are opened before the firewater addition is sufficient to begin raising the liquid levels back up.

The same general response is found in the core also, as illustrated in Figure 4.2.7.6. (Horizontal lines are included in the figure to indicate both the boundary between the upper plenum and the core at 9.6 m and the top-of-active-fuel elevation at 9.3 m.) The collapsed level in the core drops below the core midplane before stabilizing and rising again for the case initiated at 7 hr after shutdown, but the swollen level drops only about a foot into the active fuel region before the firewater addition is sufficient to begin raising the vessel inventory and liquid levels back up. At lower decay heat levels, there is no core uncovery at all.

The small core uncovery at the higher decay heat level does not result in significant core heatup before the firewater addition is sufficient to begin raising the vessel inventory and liquid levels back up, as demonstrated in Figure 4.2.7.7. At lower decay heat level s (*i.e.*, for 24 hr after shutdown), there is no core heatup at all because there is no uncovery at all (while firewater injection continues). Because firewater injection was sufficient to prevent core uncovery and heatup at decay heats 1 day after shutdown, calculations were not done for lower decay heat levels.

4.2.8 Station Blackout with 10 hr Firewater Addition Followed by High Pressure Boiloff

The accident is initiated by a loss of offsite power. The vessel water inventory is at 366.5 K (200°F), which corresponds to the maximum temperature allowed by the Grand Gulf technical specifications for operation in POS 5. Following the initiating event, onsite power is lost leading to a SBO and loss of all core cooling and coolant makeup. The operator opens two SRVs at 2 hr and steams the core at low pressure while adding coolant from the firewater system to the core bypass region. The depletion of the station batteries 12 hr after the start of the accident cause the SRVs to close (*i.e.*, the SRVs require DC power to remain open), after which they operate in the relief mode. Since the SRVs are now closed, the RPV can pressurize. The reactor pressure vessel head vent is open. The drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open (*i.e.*, "open containment").

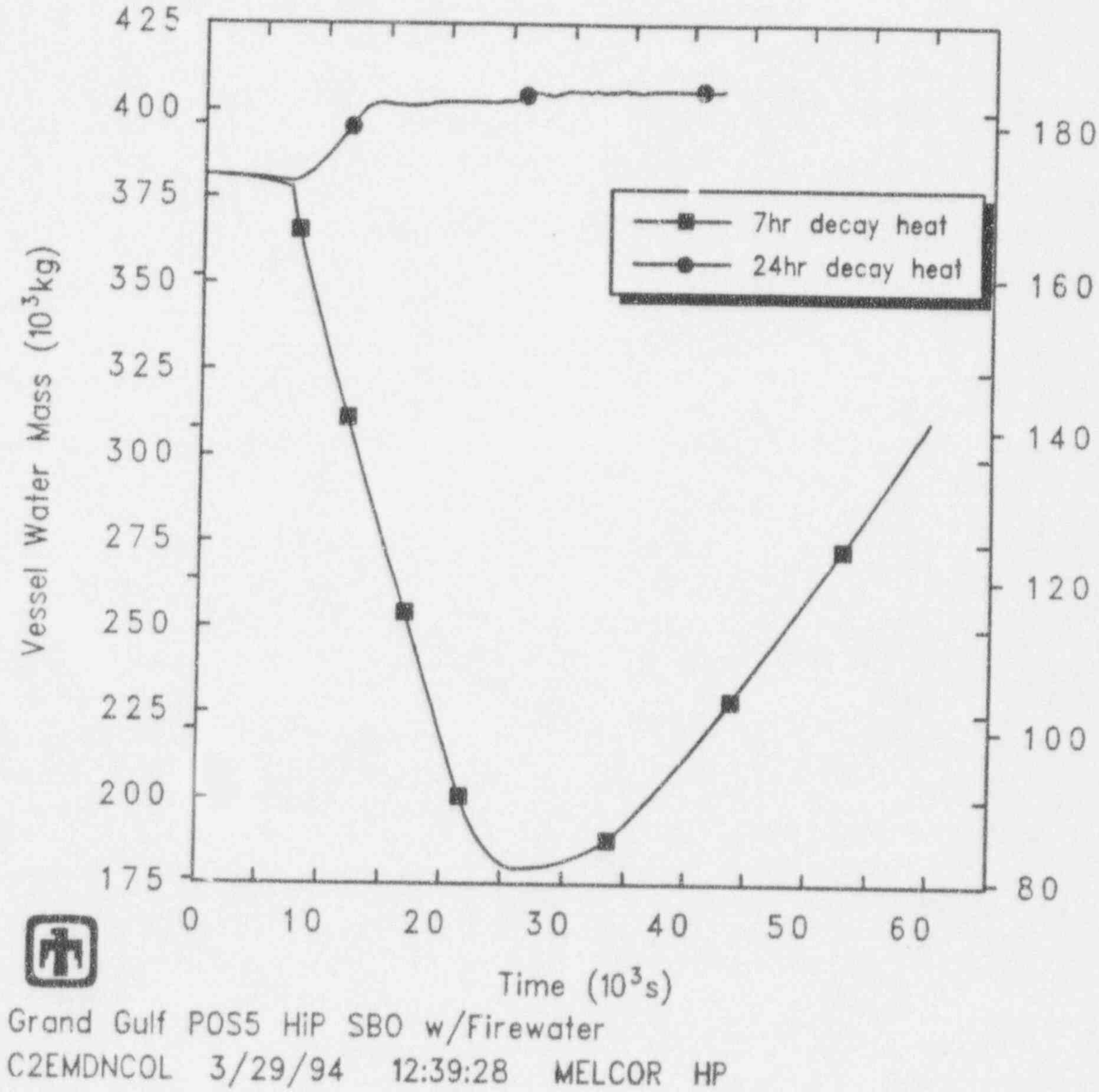


Figure 4.2.7.4. Reactor Vessel Water Masses for Grand Gulf POS 5 – Station Blackout with Firewater, Initiated at Various Times After Shutdown

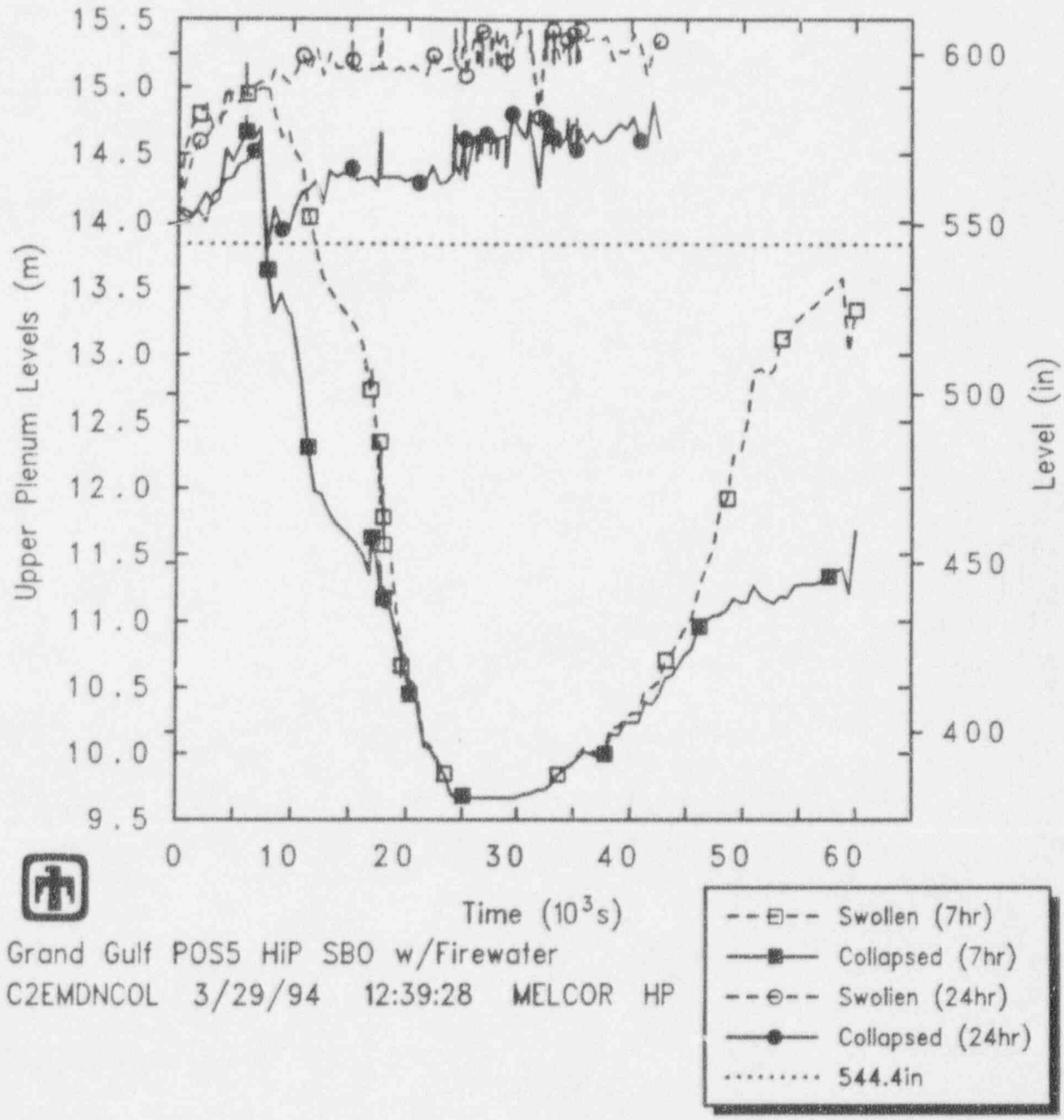


Figure 4.2.7.5. Upper Plenum Liquid Levels for Grand Gulf POS 5 – Station Blackout with Firewater, Initiated at Various Times After Shutdown

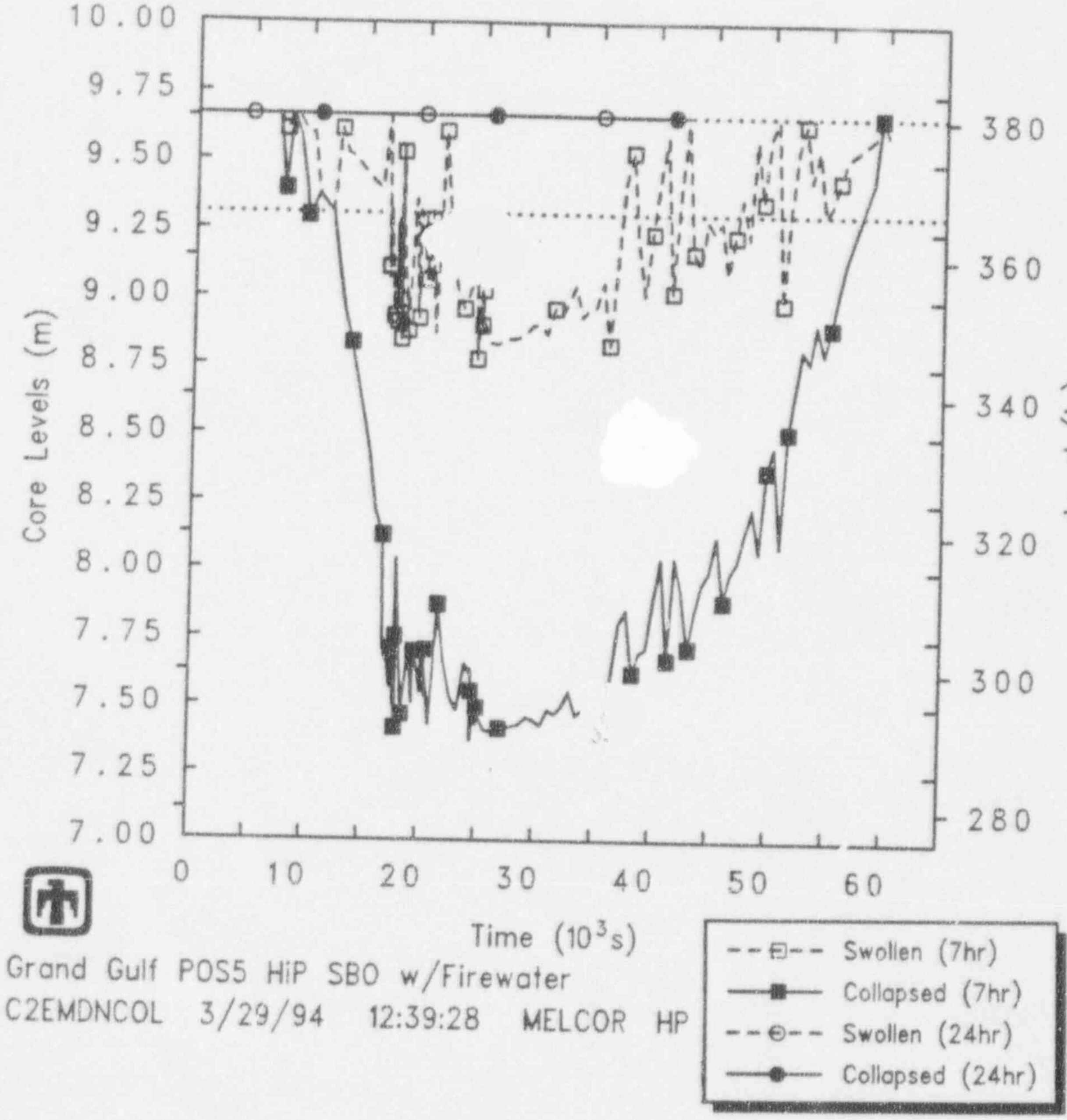


Figure 4.2.7.6. Core Liquid Levels for Grand Gulf POS 5 – Station Blackout with Firewater, Initiated at Various Times After Shutdown

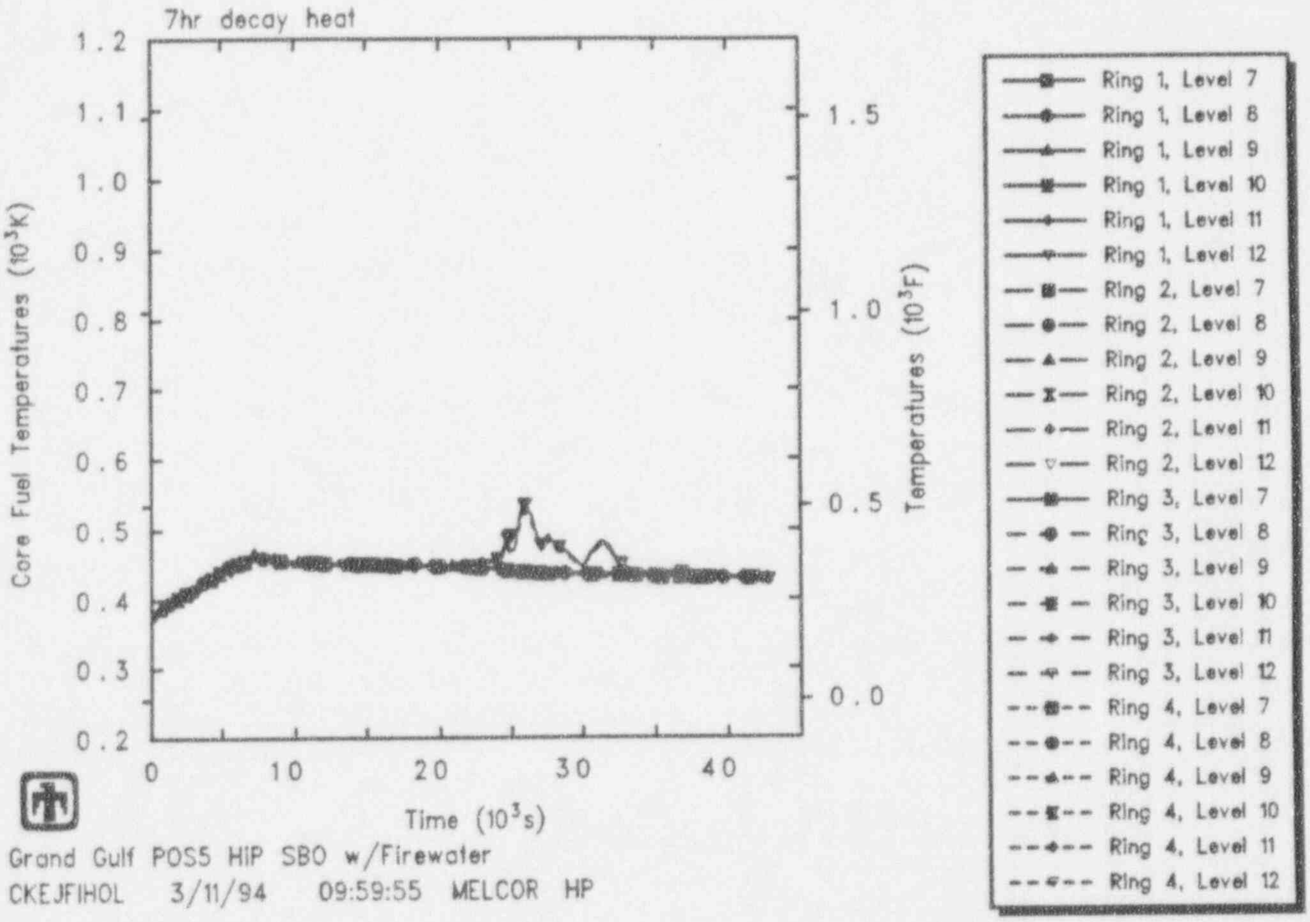


Figure 4.2.7.7. Core Fuel Temperatures for Grand Gulf POS 5 – Station Blackout with Firewater, Initiated 7 hr After Shutdown

Although the operator aligns the firewater system to inject coolant into the vessel starting at 2 hr after accident initiation, injection does not begin until the vessel has depressurized sufficiently (as determined by the pump characteristics). Figure 4.2.8.1 shows that firewater can be injected as soon as desired if the accident is assumed to start 24 hr after shutdown, but firewater injection can not begin until the vessel is depressurized for about 4 hr if the accident is assumed to start 7 hr after shutdown (a higher decay heat level). At the lower decay heat the firewater injection quickly rises to its maximum level after beginning, while at higher decay heat levels the firewater injection rises to its maximum level more slowly as the vessel continues to depressurize through the open SRVs. Firewater injection stops soon after 12 hr because after the SRVs close the system quickly repressurizes.

Figure 4.2.8.2 presents the vessel pressures calculated starting this accident scenario at two different times after scram. Initially, the system begins pressurizing as all core cooling is lost, more quickly for higher decay heat; the pressure then begins dropping after two SRVs are opened 2 hr after the start of the accident. Firewater cooling and steaming out the SRVs keep the vessel pressure down until 12 hr, when depletion of the station batteries cause the SRVs to close. Since the SRVs are now closed, the RPV pressurizes until the SRVs begin operating in the relief mode. After some time, the continued inventory loss out the open RPV vent is sufficient to relieve the steaming in the core and the SRVs close. The pressure continues to drop until core heatup and damage begins; there is then a brief repressurization, followed very quickly by a final, sharp depressurization due to vessel failure.

The flow out the open RPV vent line and later out the SRVs also pressurizes the containment and the auxiliary building, as indicated in Figure 4.2.8.3, more rapidly for higher decay heat than for lower decay heats. At both decay heat levels, for this scenario the auxiliary building fails when the SRVs begin cycling at their safety setpoint. The auxiliary building pressure briefly spikes later when the vessel fails.

As in the results presented in the previous section for a station blackout with continual firewater injection, Figure 4.2.8.4 indicates that, at lower decay heats the firewater injection causes a net increase in vessel inventory, while at higher decay heat levels firewater injection does not equal and reverse inventory loss for about 5hr. After the SRVs close at 12 hr, the system pressurizes until the SRV setpoint is reached; coolant inventory is then lost as the SRVs cycle at the safety setpoint until vessel failure, when all the remaining coolant in the vessel drains to the cavity abruptly.

Figure 4.2.8.5 gives the predicted upper plenum liquid level drop due to this inventory loss, for different decay heat levels and highlighting when a Level 3 trip (544.4 in) would be generated. The upper plenum liquid levels reflect the overall vessel coolant inventory response presented in Figure 4.2.8.4 – at lower decay heats the upper plenum levels remain nearly constant, while at higher decay heat levels the upper plenum levels drop for about 5hr after the SRVs are opened before the firewater addition is sufficient to raise the liquid levels back up briefly. The liquid level in the upper plenum resumes dropping soon after firewater injection is stopped after 12 hr for the accident initiated 7 hr after shutdown. For the same scenario initiated 24 hr after shutdown the liquid level in the

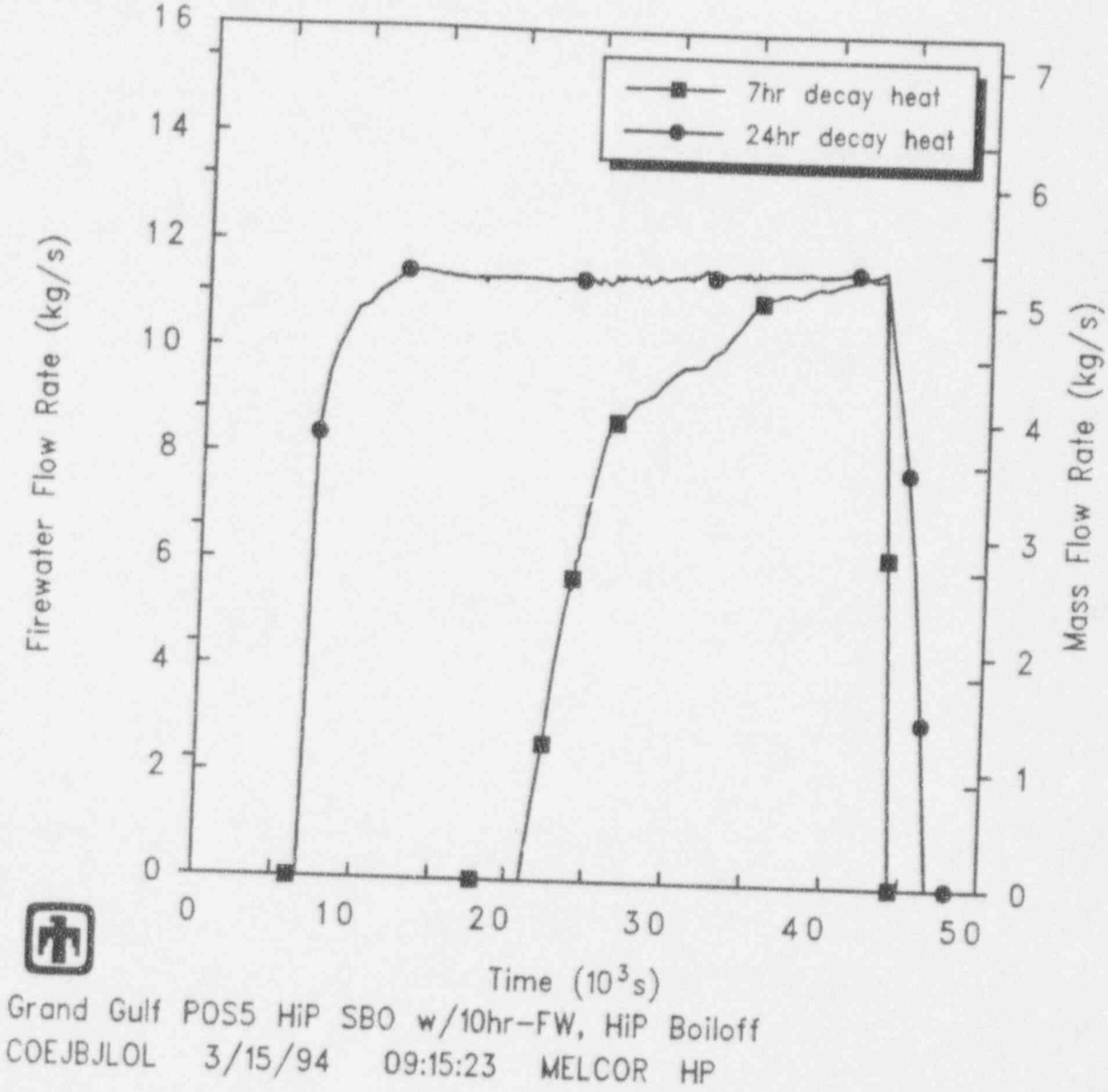


Figure 4.2.8.1. Firewater Injection Flow Rates for Grand Gulf POS 5 - Station Blackout with 10 hr Firewater Addition Followed by High Pressure Boiloff, Initiated at Various Times After Shutdown

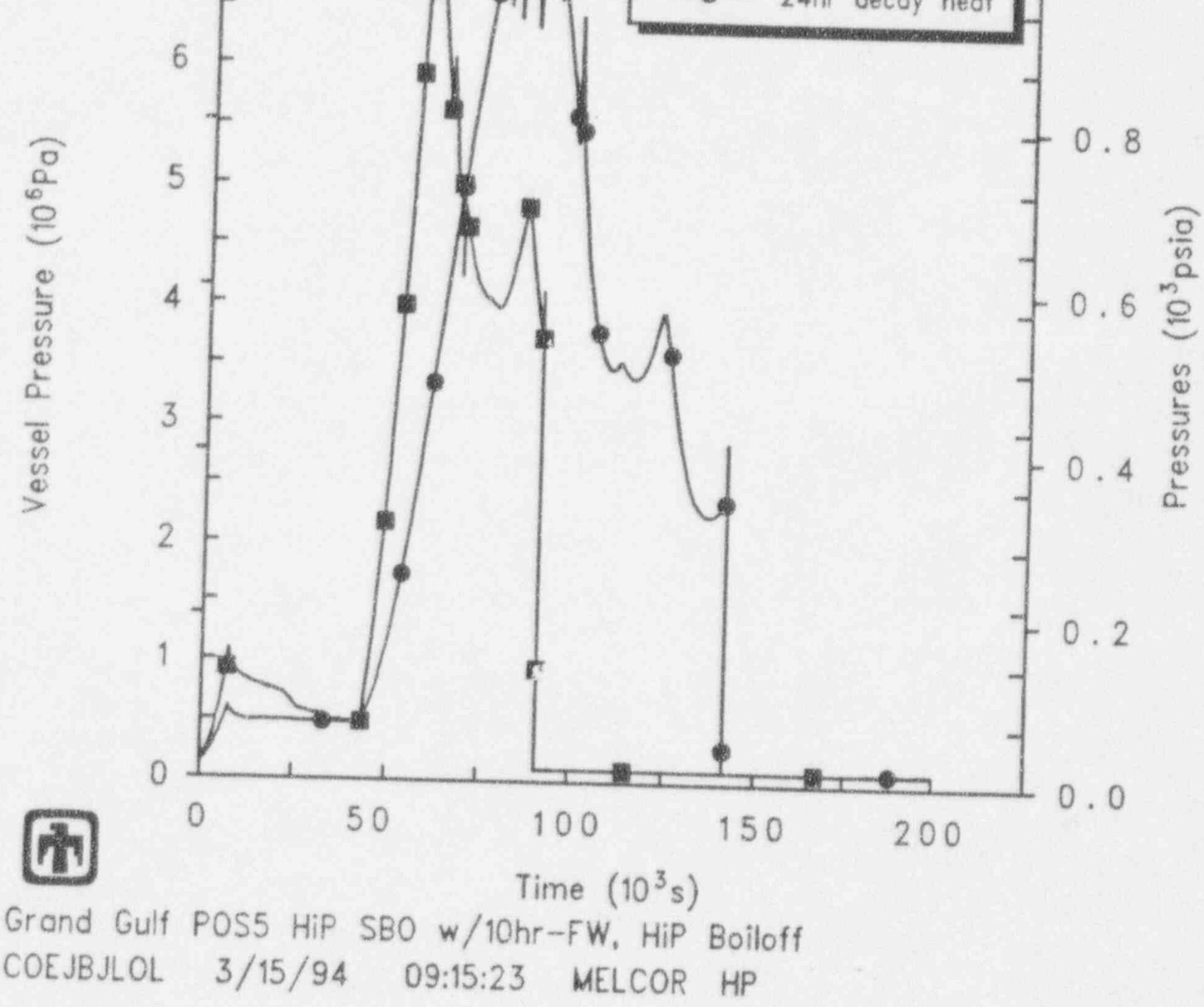


Figure 4.2.8.2. Reactor Vessel Pressures for Grand Gulf POS 5 - Station Blackout with 10 hr Firewater Addition Followed by High Pressure Boiloff, Initiated at Various Times After Shutdown

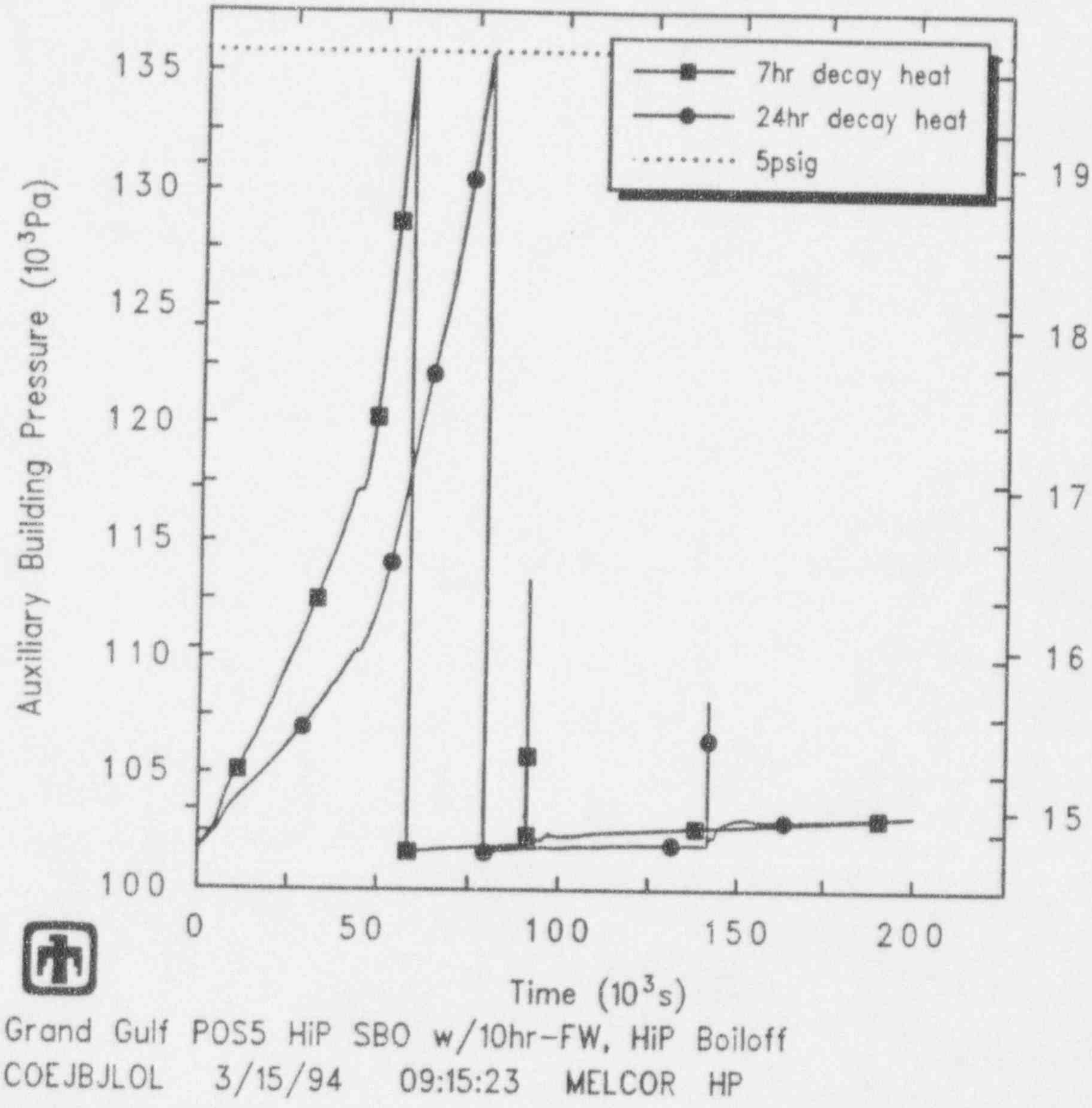


Figure 4.2.8.3. Auxiliary Building Pressures for Grand Gulf POS 5 - Station Blackout with 10 hr Firewater Addition Followed by High Pressure Boiloff, Initiated at Various Times After Shutdown

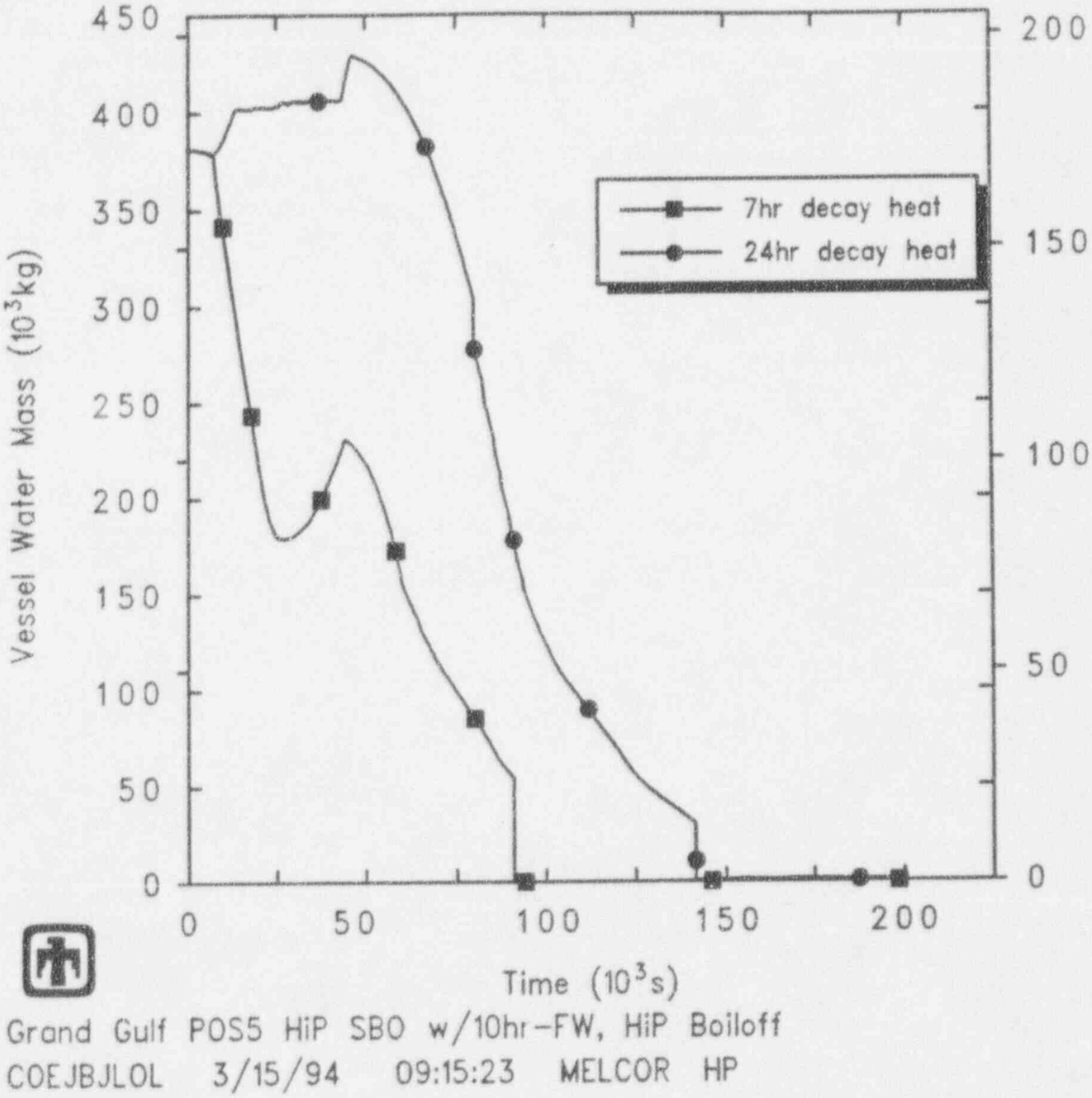


Figure 4.2.8.4. Reactor Vessel Water Masses for Grand Gulf POS 5 - Station Blackout with 10 hr Firewater Addition Followed by High Pressure Boiloff, Initiated at Various Times After Shutdown

upper plenum drops later, reflecting the higher vessel inventory when the SRVs are closed and firewater injection stops and the longer period to pressurize to the SRV setpoint at the lower decay heat level; the upper plenum levels in both cases drop when the SRVs begin cycling in the relief mode.

The same general response is found in the core also, as illustrated in Figure 4.2.8.6. (Horizontal lines are included in the figure to indicate both the boundary between the upper plenum and the core at 9.6 m and the top-of-active-fuel elevation at 9.3 m.) The collapsed level in the core drops below the core midplane before stabilizing and rising again for the case initiated at 7 hr after shutdown, but the swollen level drops only about a foot into the active fuel region before the firewater addition is sufficient to begin raising the vessel inventory and liquid levels back up. At lower decay heat levels, there is no core uncovery at all while firewater injection continues. The liquid level in the core resumes dropping soon after firewater injection is stopped after 12 hr for the accident initiated 7 hr after shutdown. For the same scenario initiated 24 hr after shutdown the liquid levels in the core also begin dropping when firewater injection stops. However, the liquid levels in the core do not drop below the TAF elevation until later, when the upper plenum is empty. The core levels in both cases drop sharply when the SRVs begin cycling in the relief mode.

The small core uncovery at the higher decay heat level does not result in significant core heatup before the firewater addition is sufficient to begin raising the vessel inventory and liquid levels back up, as demonstrated in Figure 4.2.8.7. At lower decay heat level s (*i.e.*, for 24 hr after shutdown), there is no core heatup at all because there is no uncovery at all while firewater injection continues. In both cases, after firewater injection ends at 12 hr there is a slow temperature increase, reflecting the rise in saturation temperature as the system pressurizes to the SRV setpoint. Later, after TAF uncovery, core heatup and damage begins. Because core heatup and damage did not begin until more than 1 day after accident initiation for the case initiated 24 hr after shutdown, calculations were not done for lower decay heat levels.

Table 4.2.8.1 summarizes the timings of various key events predicted using MELCOR for this sequence assuming various times after shutdown and associated decay heat levels.

4.2.9 Station Blackout with 10 hr Firewater Addition Followed by Failure to Isolate SDC

The accident is initiated by a loss of offsite power. The vessel water inventory is at 366.5 K (200°F), which corresponds to the maximum temperature allowed by the Grand Gulf technical specifications for operation in POS 5. Following the initiating event, onsite power is lost leading to a SBO and loss of all core cooling and coolant makeup. The operator opens two SRVs at 2 hr and steams the core at low pressure while adding coolant from the firewater system to the core bypass region. The depletion of the station batteries 12 hr after the start of the accident cause the SRVs to close (*i.e.*, the SRVs require DC power to remain open), after which they operate in the relief mode. Since the SRVs are now closed, the RPV will pressurize. The SBO precludes the isolation of

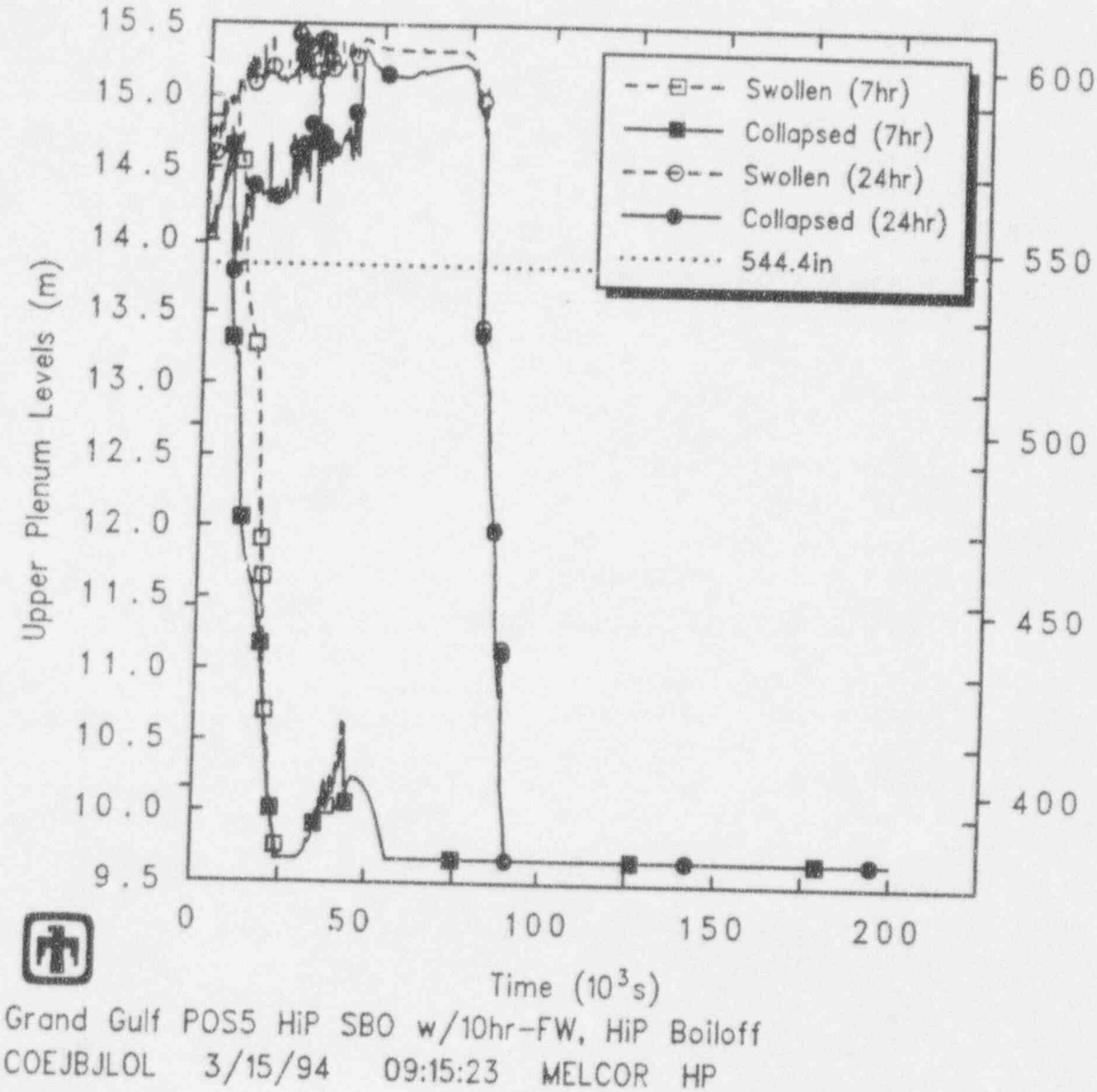


Figure 4.2.8.5. Upper Plenum Liquid Levels for Grand Gulf POS 5 - Station Blackout with 10 hr Firewater Addition Followed by High Pressure Boiloff, Initiated at Various Times After Shutdown

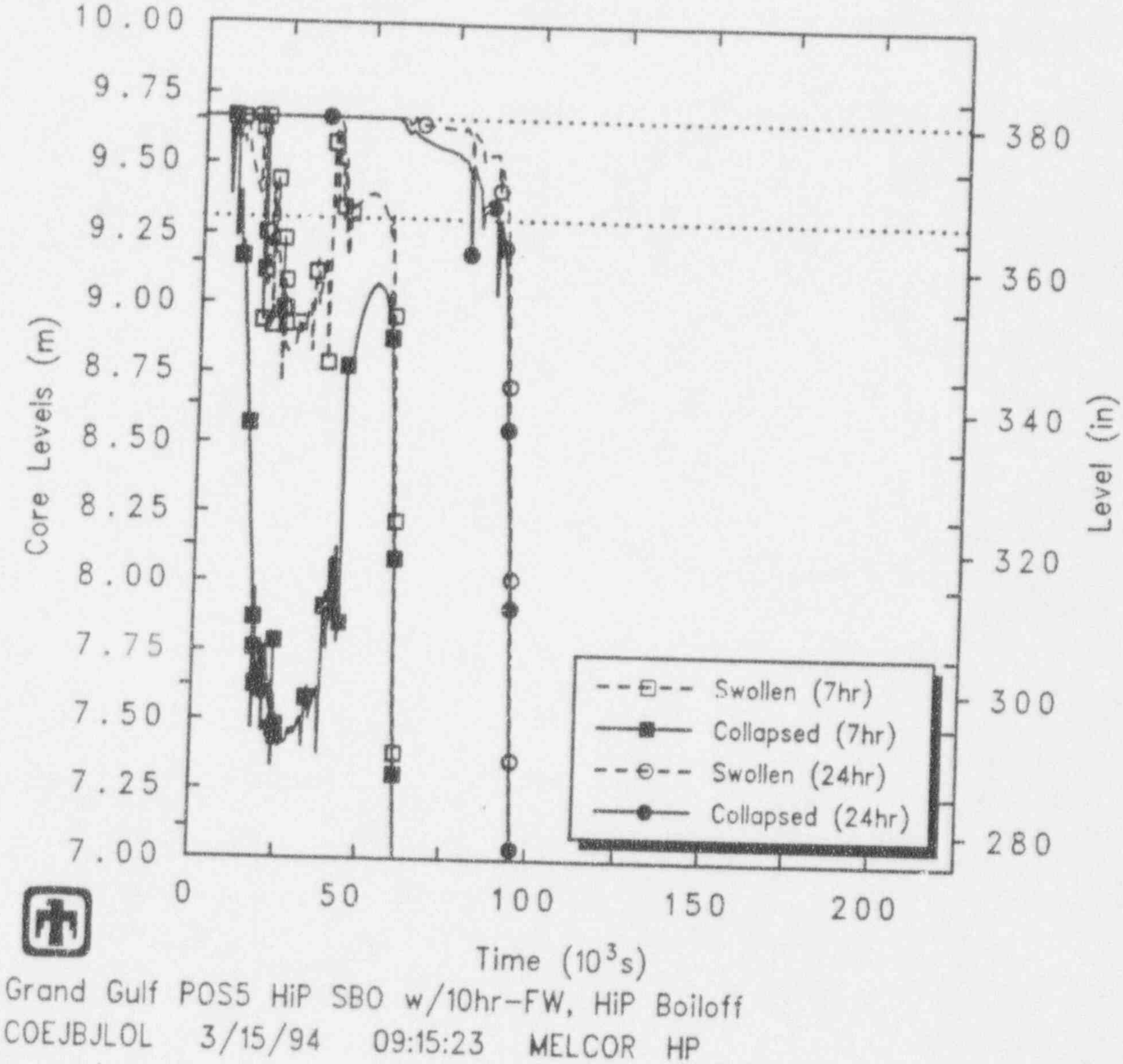


Figure 4.2.8.6. Core Liquid Levels for Grand Gulf POS 5 - Station Blackout with 10 hr Firewater Addition Followed by High Pressure Boiloff, Initiated at Various Times After Shutdown

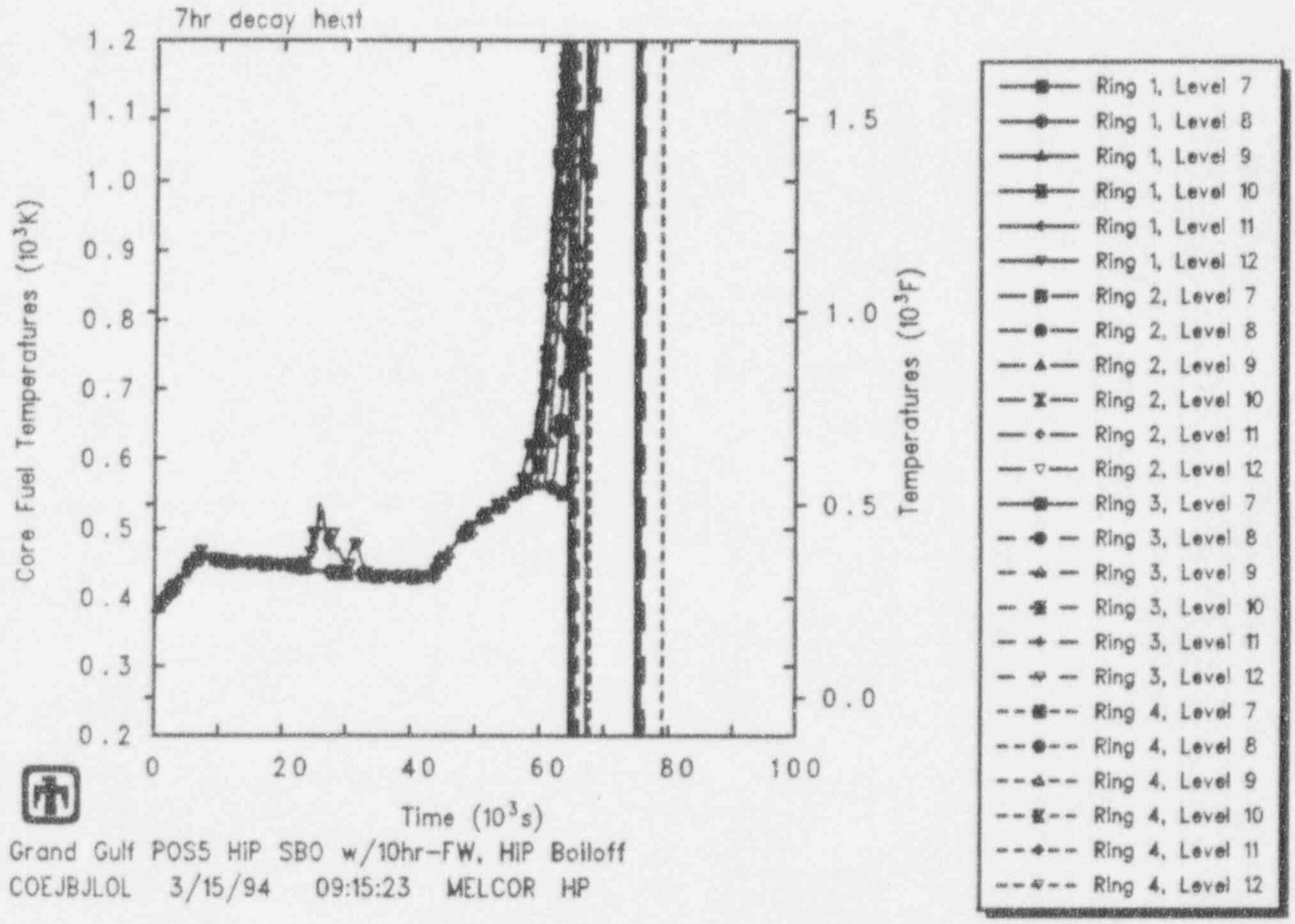


Figure 4.2.8.7. Core Fuel Temperatures for Grand Gulf POS 5 - Station Blackout with 10 hr Firewater Addition Followed by High Pressure Boiloff, Initiated 7 hr After Shutdown

Table 4.2.8.1. Key Event Times for Grand Gulf POS 5 - Station Blackout with 10 hr Firewater Addition Followed by High Pressure Boiloff, Initiated at Various Times After Shutdown

	Time to (s)			
Initiation Time After Shutdown	TAF Uncovery†	Core Heatup	First Gap Release	Vessel Failure
7 hr	9,780	56,500	63,038	90,582
24 hr	79,530	90,000	97,950	141,447

†Collapsed liquid level

the low pressure piping in the SDC system. This low-pressure SDC system piping fails when the RPV pressure reaches 3.135 MPa (440 psig) resulting in an interfacing systems LOCA. The break in the SDC line is opened when the vessel pressure reaches 3.135 MPa (440 psig). The SDC break runs from the vessel downcomer, 4.38 m above the bottom of the vessel to the first floor of the auxiliary building, 8.18 m below the bottom of the vessel. The reactor pressure vessel head vent is open. The drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open (*i.e.*, "open containment").

The thermal/hydraulic and core damage behavior for this scenario are quite similar to those in the station blackout with 10 hr firewater addition followed by high pressure boiloff, described in the previous section; they are completely identical for the first ≥12 hr, until the system pressurizing is interrupted by the failure to isolate SDC at 3.135 MPa (440 psig) in this case. Figure 4.2.9.1 presents the vessel pressures calculated starting this accident scenario at two different times after shutdown.

Figure 4.2.9.2 gives the predicted upper plenum and core liquid levels, highlighting when a Level 3 trip (544.4 in) would be generated and when TAF (at 9.3 m) is uncovered. There is a temporary core uncovery for this cenario initiated 7 hr after scram but no core uncovery while firewater injection continues for this scenario initiated at 24 hr decay heat, as noted in the previous two sections. The upper plenum and core liquid levels both drop very quickly after the SDC break opens.

Figures 4.2.9.3 and 4.2.9.4 present the core clad temperatures during the firewater addition period and the subsequent core heatup for this scenario initiated 7 hr and 24 hr after shutdown, respectively. There is a brief core heatup during the early, temporary core uncovery in this sequence initiated 7 hr after shutdown; At decay heat levels corresponding to accident initiation 24 hr after shutdown, there is no core heatup at all while firewater injection continues, because there is no uncovery at all. In both cases, after firewater injection ends at 12 hr there is a slow temperature increase, reflecting the rise in saturation temperature as the system pressurizes to the SRV setpoint. Later, after

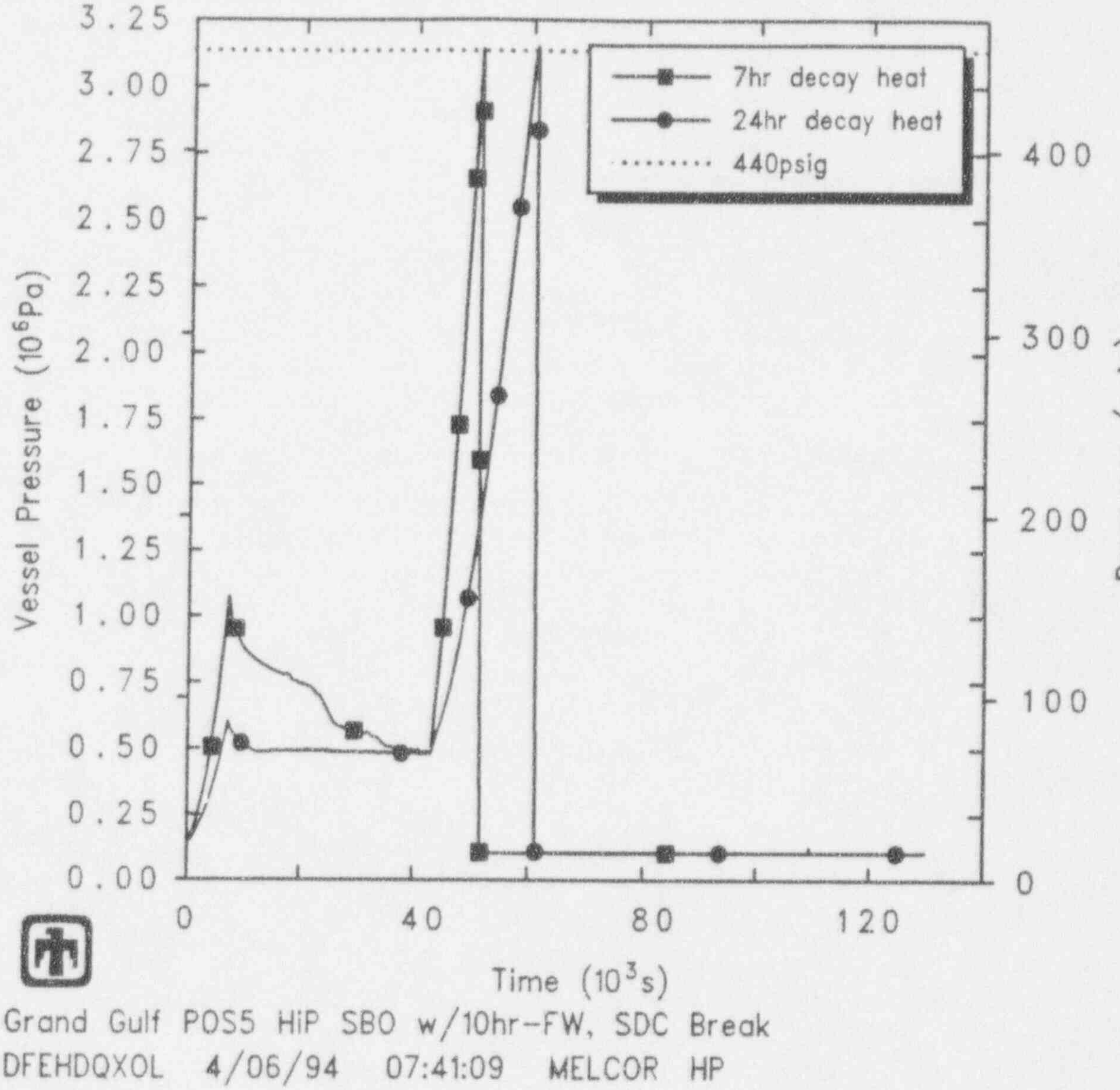


Figure 4.2.9.1. Reactor Vessel Pressures for Grand Gulf POS 5 - Station Blackout with 10 hr Firewater Addition Followed by Failure to Isolate SDC, Initiated at Various Times After Shutdown

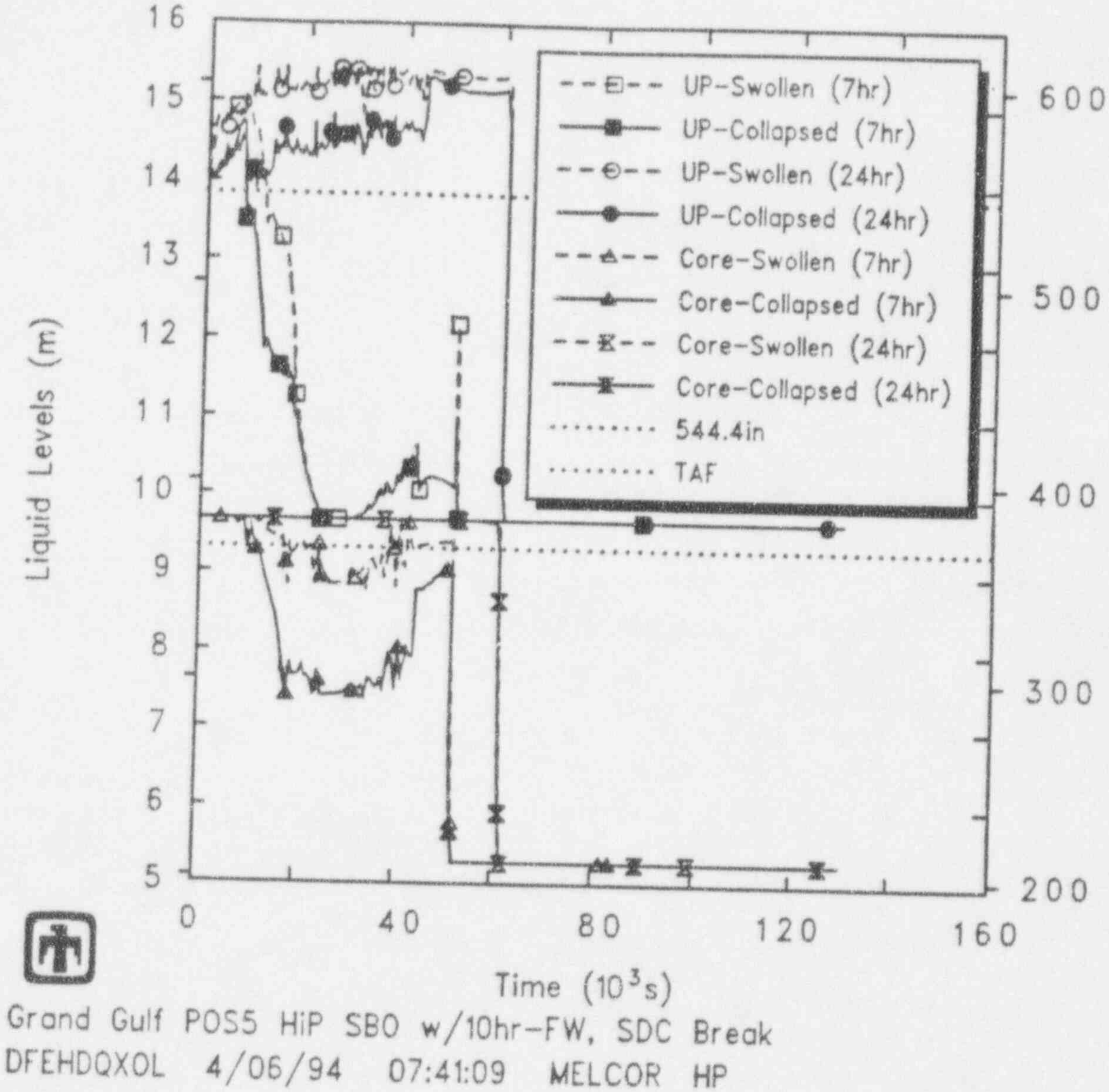


Figure 4.2.9.2. Upper Plenum and Core Liquid Levels for Grand Gulf POS 5 – Station Blackout with 10 hr Firewater Addition Followed by Failure to Isolate SDC, Initiated at Various Times After Shutdown

Table 4.2.9.1. Key Event Times for Grand Gulf POS 5 - Station Blackout with 10 hr Firewater Addition Followed by Failure to Isolate SDC, Initiated 24 hr After Shutdown

	Time to (s)			
Initiation Time After Shutdown	TAF Uncovery†	Core Heatup	First Gap Release	Vessel Failure
7 hr	9,924	53,000	53,720	93,800
24 hr	60,520	63,000	63,940	109,527

†Collapsed liquid level

TAF uncovery, core heatup and damage begins.

Table 4.2.9.1 summarizes the timings of various key events predicted using MELCOR for this sequence initiated 24 hr after shutdown.

4.3 Level 2 Support Calculations

Based partly on the results of the MELCOR calculations done in support of the Level 1 analysis, a number of accident sequences were eliminated from consideration as not resulting in core damage within the first 24 hr from the start of the accident. The remaining sequences, those leading to core damage within 1 day and with a frequency greater than the Level 1 truncation frequency, were grouped into plant damage states or PDSs (see Section XX of Volume YY). The plant damage states are ranked by their relative contribution to core damage frequency in Table 4.3.1. Complete MELCOR accident analyses have been done for these sequences in support of the Level 2 PRA, with results described in the following subsections. (The last two sequences in the table are identical to other sequences in the table with regard to MELCOR calculations, but with different recovery assumptions in the Level 2 PRA.)

4.3.1 Large Break LOCA with Flooded Containment, Initiated 7 hr, 24 hr and 40 day After Shutdown

This accident is initiated by a large break LOCA in the recirculation line. At the start of the accident, the reactor vessel is depressurized, the coolant is at the normal level and the SRVs are closed. The vessel water inventory is at 366.5 K (200°F), which corresponds to the maximum temperature allowed by the Grand Gulf technical specifications for operation in POS 5. The break drains the vessel to 2/3 core height. The initiating event then results in a loss of all core cooling and coolant makeup. The reactor pressure vessel

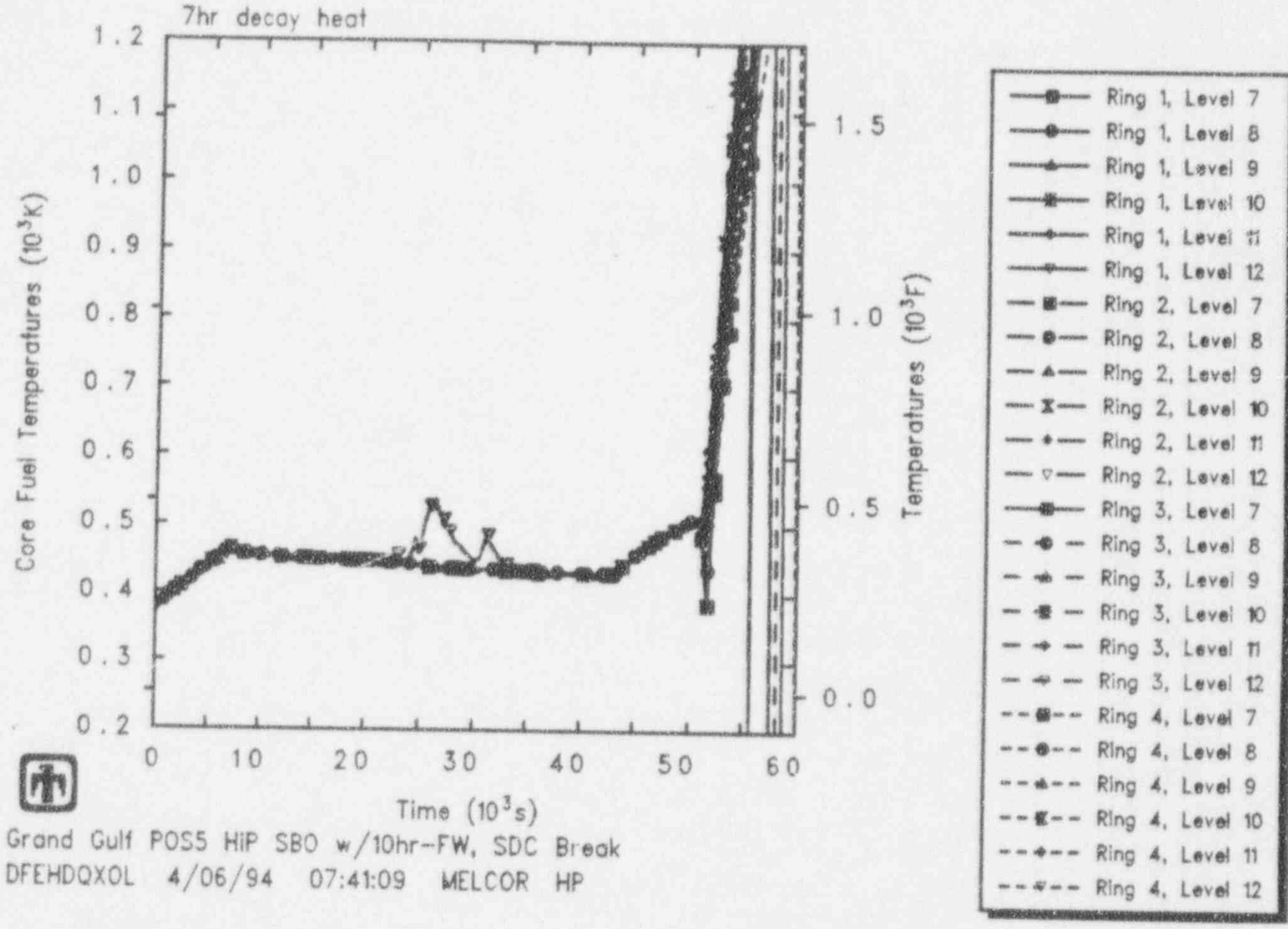


Figure 4.2.9.3. Core Fuel Temperatures for Grand Gulf POS 5 - Station Blackout with 10 hr Firewater Addition Followed by Failure to Isolate SDC, Initiated 7 hr After Shutdown

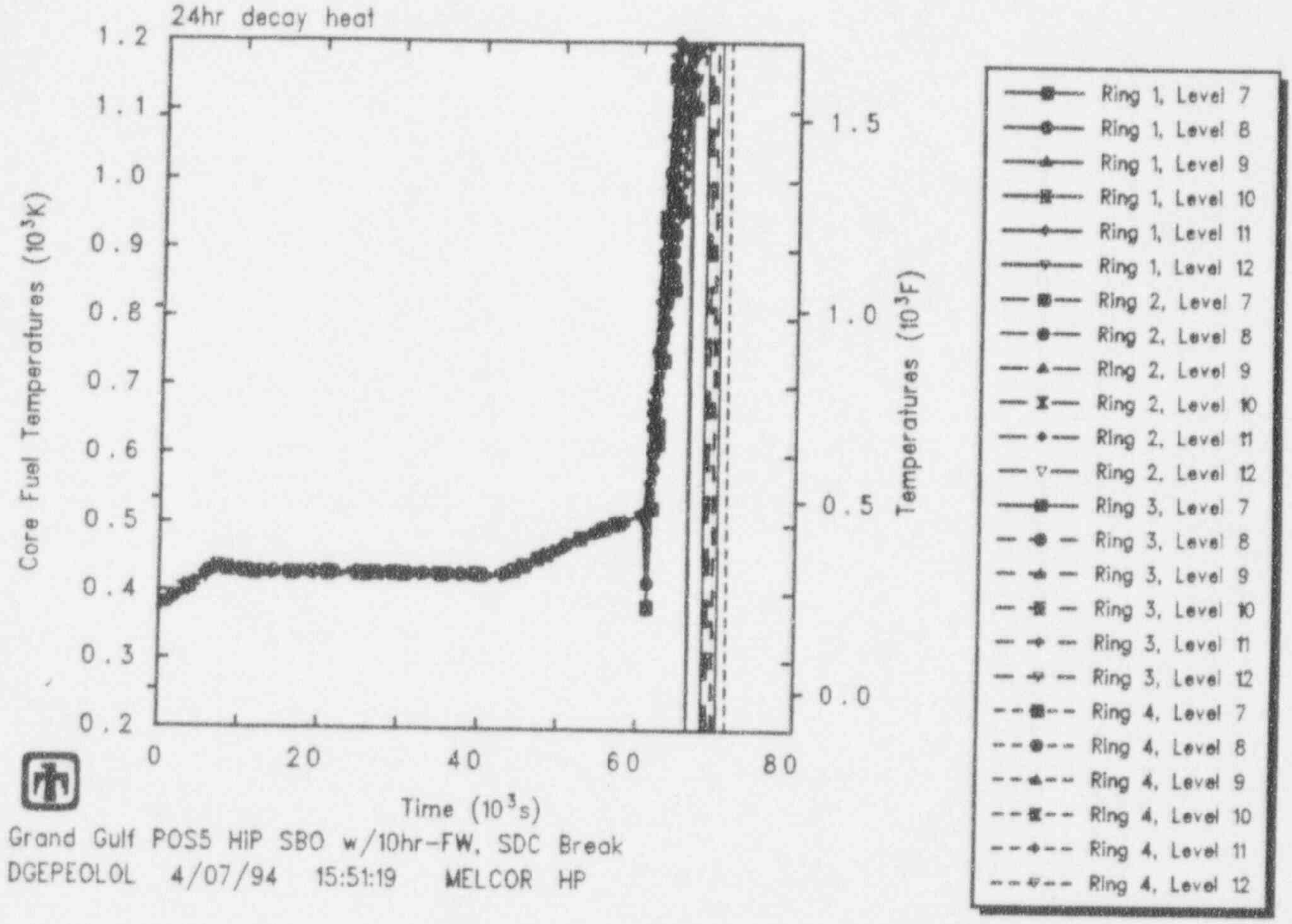


Figure 4.2.9.4. Core Fuel Temperatures for Grand Gulf POS 5 - Station Blackout with 10 hr Firewater Addition Followed by Failure to Isolate SDC, Initiated 24 hr After Shutdown

Table 4.3.1. MELCOR Level 2 Support Calculations – Sequences and Relative Contribution of Plant Damage States to Core Damage Frequency

Plant Damage State	Time After Shutdown	Fraction Contributed	Sequence Description
PDS 3-1	40 day	0.338	LBLOCA with flooded containment
PDS 2-2	24 hr	0.242	SBO w/o firewater, break in SDC
PDS 2-1	24 hr	0.170	LBLOCA with flooded containment
PDS 2-4	24 hr	0.104	Low-P Boiloff with flooded containment
PDS 1-3	7 hr	0.032	SBO w/10 hr-firewater, High-P Boiloff
PDS 1-1	7 hr	0.019	LBLOCA with flooded containment
PDS 1-2	7 hr	0.015	SBO w/o firewater, break in SDC
PDS 1-5	7 hr	0.008	Low-P Boiloff with flooded containment
PDS 2-5	24 hr	0.007	High-P Boiloff with closed containment
PDS 2-6	24 hr	0.006	Open MSIVs with closed containment
PDS 2-3	24 hr	0.054	Same as PDS 2-2, but with potential to recover AC power
PDS 1-4	7 hr	0.005	Same as PDS 1-2, but with potential to recover AC power

head vent is open at the beginning of the transient. The containment has been flooded to the elevation of the lower personnel lock, 9.65 m or 31.67 ft above the suppression pool floor. The containment (suppression pool, pedestal cavity and drywell) water inventory is at 300.5 K (80°F); the containment is at 305.4 K (90°F). The drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open (*i.e.*, "open containment").

This sequence is almost identical to the large break LOCA scenario discussed in Section 4.2.5, except that in those Level 1 analyses the containment was dry while in these Level 2 analyses the containment was assumed to be flooded.

The sequence of events predicted by MELCOR for this accident with different initiation times is given in Table 4.3.1.1.

Figure 4.3.1.1 gives the vessel pressures calculated for this same accident scenario initiated at three different times after scram. In all cases, the primary system remains near atmospheric as the large break maintains pressure near-equilibrium between the primary and the containment, while the open personnel locks and equipment hatch vent the containment to the auxiliary building. For any given decay heat level, the smaller pressure spikes seen in Figure 4.3.1.1 generally correspond to core heatup and damage, while the largest pressure spikes seen in Figure 4.3.1.1 correspond to vessel failure to auxiliary building failure.

The water and steam coolant flowing out through the break pressurizes the containment and, through the open equipment hatch and personnel locks, pressurizes the auxiliary building, as shown in Figure 4.3.1.2. The longer after shutdown and scram that this accident sequence begins, the lower the decay heat and the longer it takes to fail the auxiliary building. The auxiliary building pressure rises somewhat more slowly during the early stages of core uncovery, heatup and damage, then spikes up to the failure point after vessel failure. Because of the rapid decrease in the exponentially-dropping decay heat soon after scram and the much more gradual decline in decay heat much later after scram, the time to vessel and auxiliary building failure for this accident initiated 40 days after scram is not proportionally greater than the time to vessel and auxiliary building failure for this accident initiated 24 hr after scram.

The pressure histories in all the control volumes modelling the vessel are virtually identical to the results shown in Figure 4.3.1.1 for the core control volume; the pressure histories in the four control volumes modelling different floors in the auxiliary building are all virtually identical to the results shown in Figure 4.3.1.2 for the second floor. In each case, the pressure response in the drywell and cavity generally tracks the vessel pressure, while the pressure response in the outer containment (*i.e.*, dome, equipment hatch, *etc.*) is very similar to that shown for the auxiliary building.

The coolant inventory in the vessel drops due to coolant and steam loss out the break, with a very rapid loss of about 60-70% of the inventory as liquid followed by a more gradual loss of the remaining inventory due to boiling and steam outflow, as presented in Figure 4.3.1.3. The amount of liquid inventory lost in the initial liquid blowdown is determined by the elevation of the break and is therefore about the same

Table 4.3.1.1. Sequence of Events Predicted by MELCOR for Large Break LOCA with Flooded Containment, Initiated 7 hr, 24 hr and 40 day After Shutdown

Event	Time After Shutdown 7 hr	24 hr	40 day
Accident initiation	0.0	0.0	0.0
Core uncovery (TAF) begins	69 s	70 s	70 s
Core heatup begins	2,000 s (0.56 hr)	3,000 s (0.83 hr)	4,500 s (1.25 hr)
Clad failure/Gap release			
(Ring 1)	9,393 s (2.61 hr)	14,766 s (4.10 hr)	22,264 s (6.18 hr)
(Ring 2)	9,296 s (2.58 hr)	14,590 s (4.05 hr)	22,102 s (6.14 hr)
(Ring 3)	9,409 s (2.61 hr)	14,832 s (4.12 hr)	22,465 s (6.24 hr)
(Ring 4)	10,007 s (2.78 hr)	15,754 s (4.38 hr)	23,773 s (6.60 hr)
(Ring 5)	12,563 s (3.49 hr)	19,612 s (5.45 hr)	28,391 s (7.89 hr)
(Ring 6)	16,461 s (4.57 hr)	25,602 s (7.11 hr)	34,570 s (9.60 hr)
Core plate failed			
(Ring 1)	98,755 s (27.43 hr)	146,396 s (40.67 hr)	218,961 s (60.82 hr)
(Ring 2)	95,954 s (26.65 hr)	145,749 s (40.49 hr)	218,100 s (60.58 hr)
(Ring 3)	98,940 s (27.48 hr)	141,858 s (39.41 hr)	218,090 s (60.58 hr)
(Ring 4)	101,503 s (28.20 hr)	141,478 s (39.30 hr)	217,619 s (60.45 hr)
(Ring 5)	94,884 s (26.36 hr)	140,514 s (39.03 hr)	216,292 s (60.08 hr)
(Ring 6)	92,455 s (25.68 hr)	139,997 s (38.89 hr)	213,691 s (59.36 hr)
Vessel LH penetration failed			
(Ring 1)	92,646 s (25.74 hr)	141,280 s (39.24 hr)	218,100 s (60.58 hr)
(Ring 2)	92,603 s (25.72 hr)	140,621 s (39.06 hr)	214,252 s (59.51 hr)
(Ring 3)	92,574 s (25.72 hr)	140,257 s (38.96 hr)	213,956 s (59.43 hr)
(Ring 4)	92,559 s (25.71 hr)	140,146 s (38.93 hr)	213,868 s (59.41 hr)
(Ring 5)	92,544 s (25.71 hr)	140,100 s (38.92 hr)	213,823 s (59.40 hr)
(Ring 6)	92,571 s (25.71 hr)	140,100 s (38.92 hr)	213,801 s (59.39 hr)
Commence debris ejection	92,544 s (25.71 hr)	140,100 s (28.92 hr)	213,823 s (59.40 hr)
Auxiliary building failed	117,500 s (32.6 hr)	205,000 s (57.0 hr)	315,000 s (87.5 hr)
Cavity rupture			
End of calculation	500,000 s (138.9 hr)	662,916 s (184.1 hr)	787,100 s (218.6 hr)

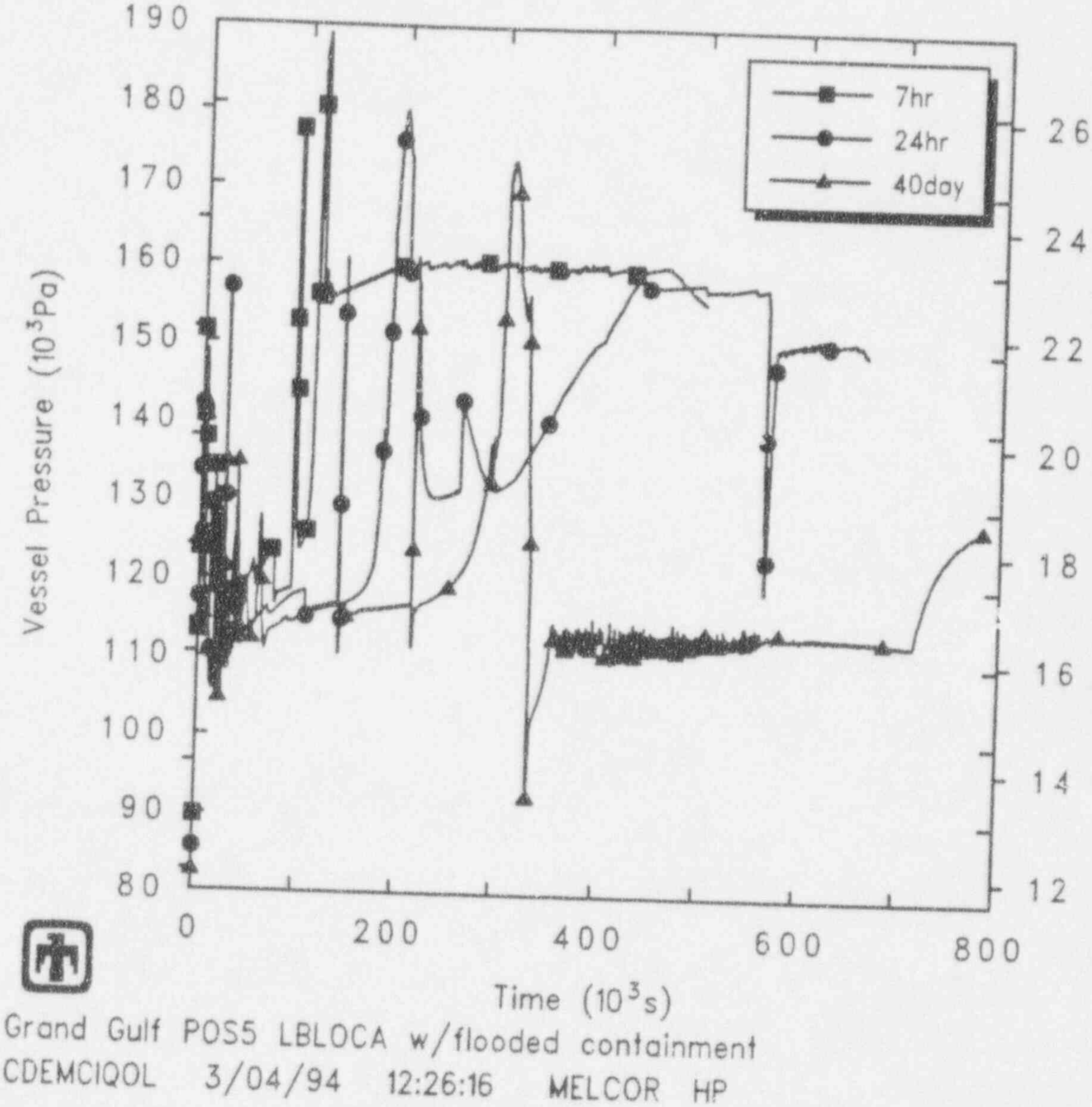


Figure 4.3.1.1. Reactor Vessel Pressures for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated at Various Times After Shutdown

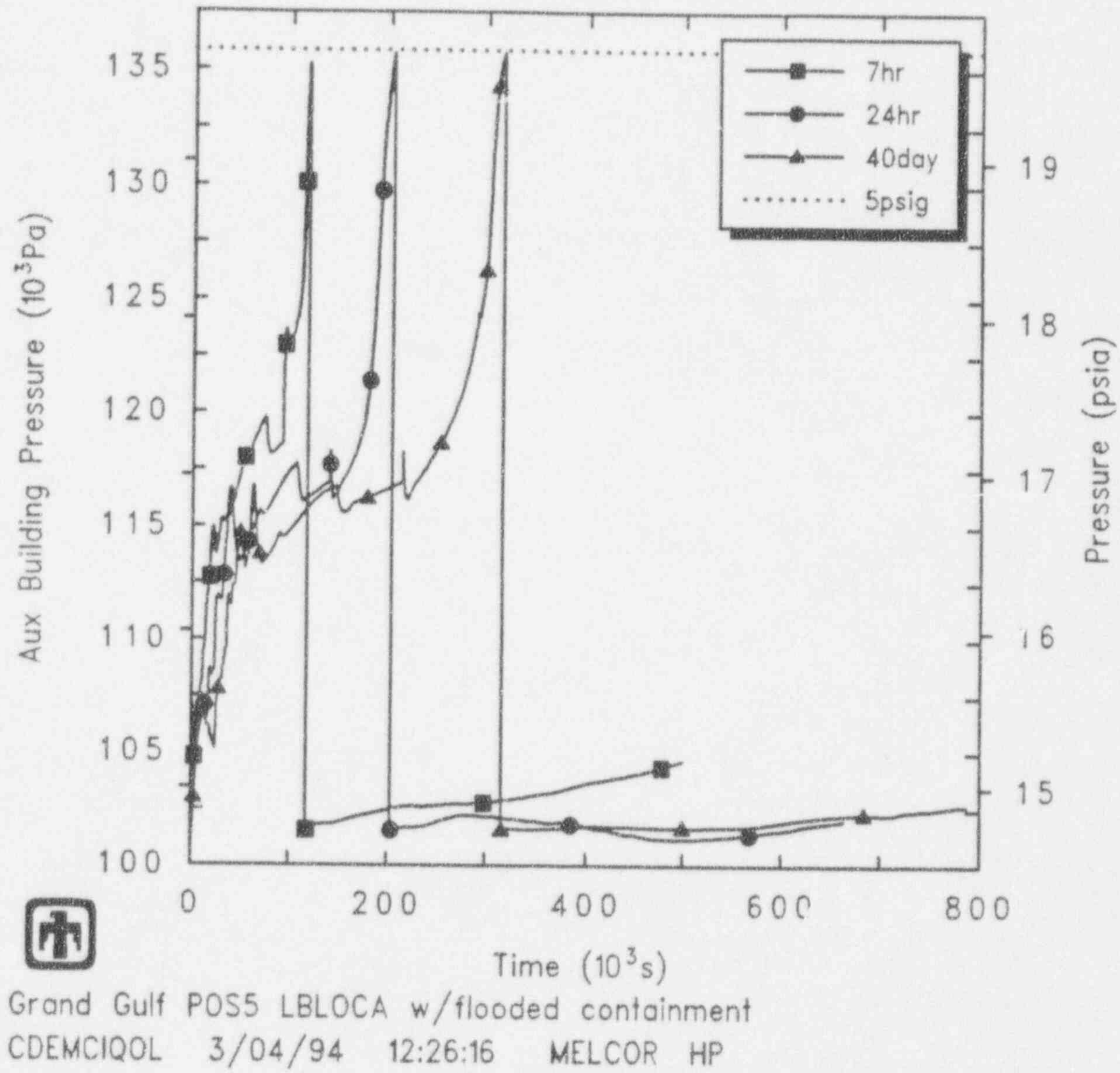


Figure 4.3.1.2. Auxiliary Building Pressures for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated at Various Times After Shutdown

regardless of the decay heat level; later, as would be expected, the gradual inventory loss due to continued boiloff is faster for higher decay heat levels than for lower decay heat levels. The vessel inventory then drops to zero very quickly upon vessel failure.

Figures 4.3.1.4 and 4.3.1.5 give the core and lower plenum swollen and collapsed liquid levels for this accident sequence initiated at three different times after scram. (Note the change in time scale on the abcissa in these two figures.) The upper plenum liquid levels drop very quickly as the break drains the vessel to 2/3 core height, within seconds or minutes, and are not shown. As with the vessel total inventory comparison, the core levels initially drop rapidly to 2/3 core height as liquid inventory is lost out the break, followed by a more gradual loss of the remaining inventory due to boiling and steam outflow, as presented in Figure 4.3.1.4. The swollen (*i.e.*, two-phase, frothy) liquid levels in the core remain substantially above the collapsed liquid levels during most of core uncovery. The level drop continues from the core region down into the lower plenum, shown in Figure 4.3.1.5, with the levels dropping more slowly once the core is uncovered and less swelling predicted in the lower plenum region than in the core. The lower plenum is still mostly full when vessel failure occurs and any remaining liquid inventory is lost out the vessel break to the cavity.

The heatup of the intact fuel and clad is illustrated in Figures 4.3.1.6 through 4.3.1.8, as calculated for scenarios initiated at 7 hr, 24 hr and 40 days after shutdown, respectively. Core uncovery and heatup begins sooner and proceeds more rapidly at higher decay heat levels than for the same accident initiated longer after scram. The fuel/clad component temperatures in MELCOR are set to zero in a cell when that component fails, so these figures show both the overall heatup rate and the time to failure.

Figures 4.3.1.9 through 4.3.1.11 present corresponding core debris temperatures in the active fuel region calculated for scenarios initiated at 7 hr, 24 hr and 40 days after shutdown, respectively; these are the temperatures of the debris bed formed by the failure of the intact fuel/clad component in MELCOR in a core cell, whose (intact) temperatures were given in Figures 4.3.1.6 through 4.3.1.8. The intact fuel/clad component temperatures reach a peak of $\geq$2000 K ($\geq$3140°F) since the component generally fails at the zircaloy clad melt temperature, taken as 2098 K (3317°F) in MELCOR. The debris bed in the active fuel region in contrast reaches peak temperatures $\geq$3250 K (5390°F), just above the UO_2 melt temperature of 3113 K (5144°F). The debris bed temperatures reached in the active fuel region are slightly higher for accidents initiated at higher decay heat levels than for lower decay heat levels, as would be expected.

The temperatures of the active fuel region debris bed drop to zero when the core plate fails and the debris relocates to the lower plenum. This occurs much later than the collapse of the intact fuel and clad into a debris bed. The core support plate is assumed to fail at 1273 K (1832°F) and, with the new debris radial relocation model added in MELCOR 1.8.2, the core support plate needs to fail in only one ring before debris from cells in the active fuel region in all radial rings can potentially flow sideways and down, fall through the failed plate, and then spread sideways into cells in the lower plenum in all radial rings. (Thus a lower head penetration can now fail in a ring before the core plate in that ring fails.)

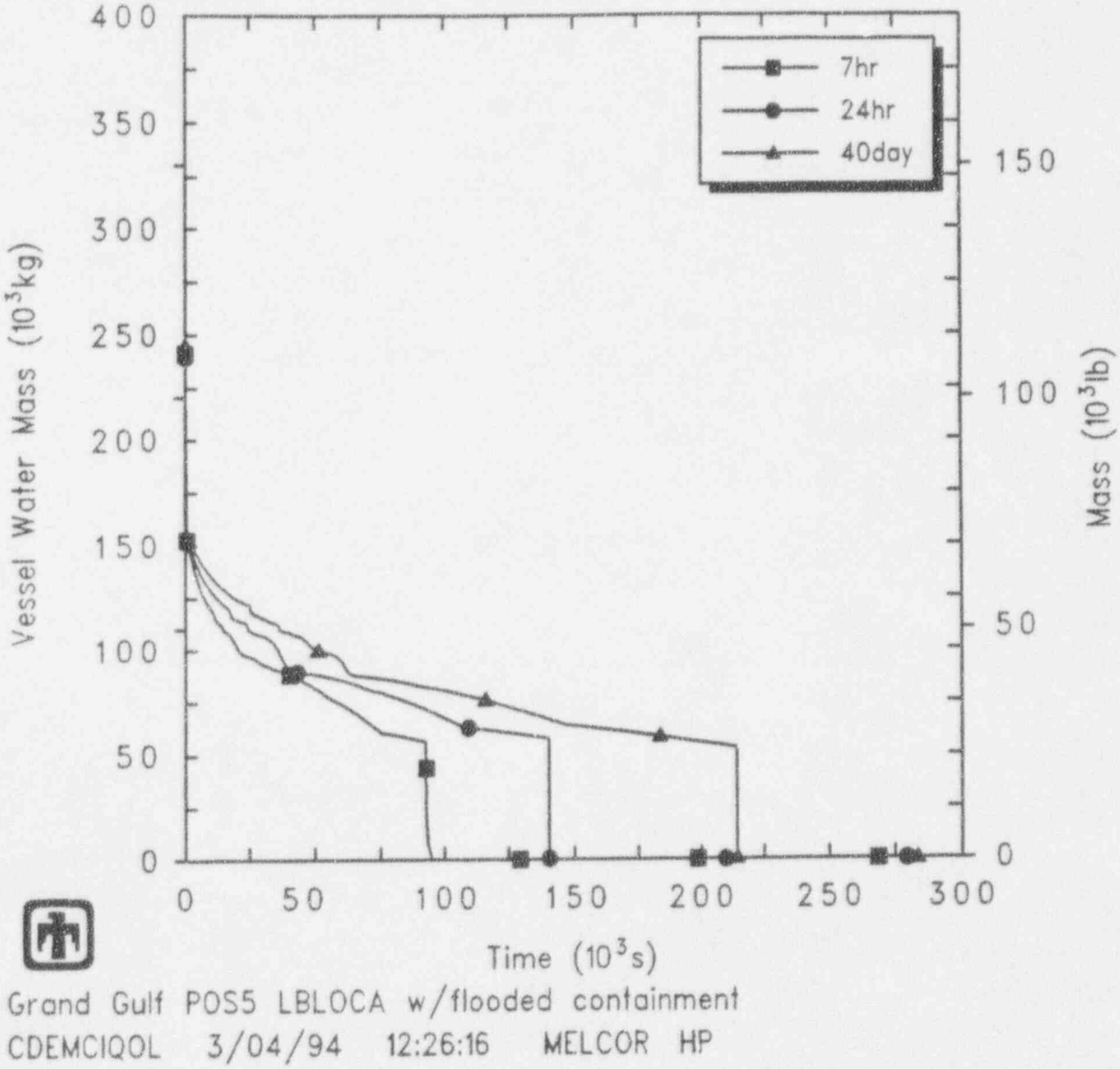


Figure 4.3.1.3. Reactor Vessel Water Masses for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated at Various Times After Shutdown

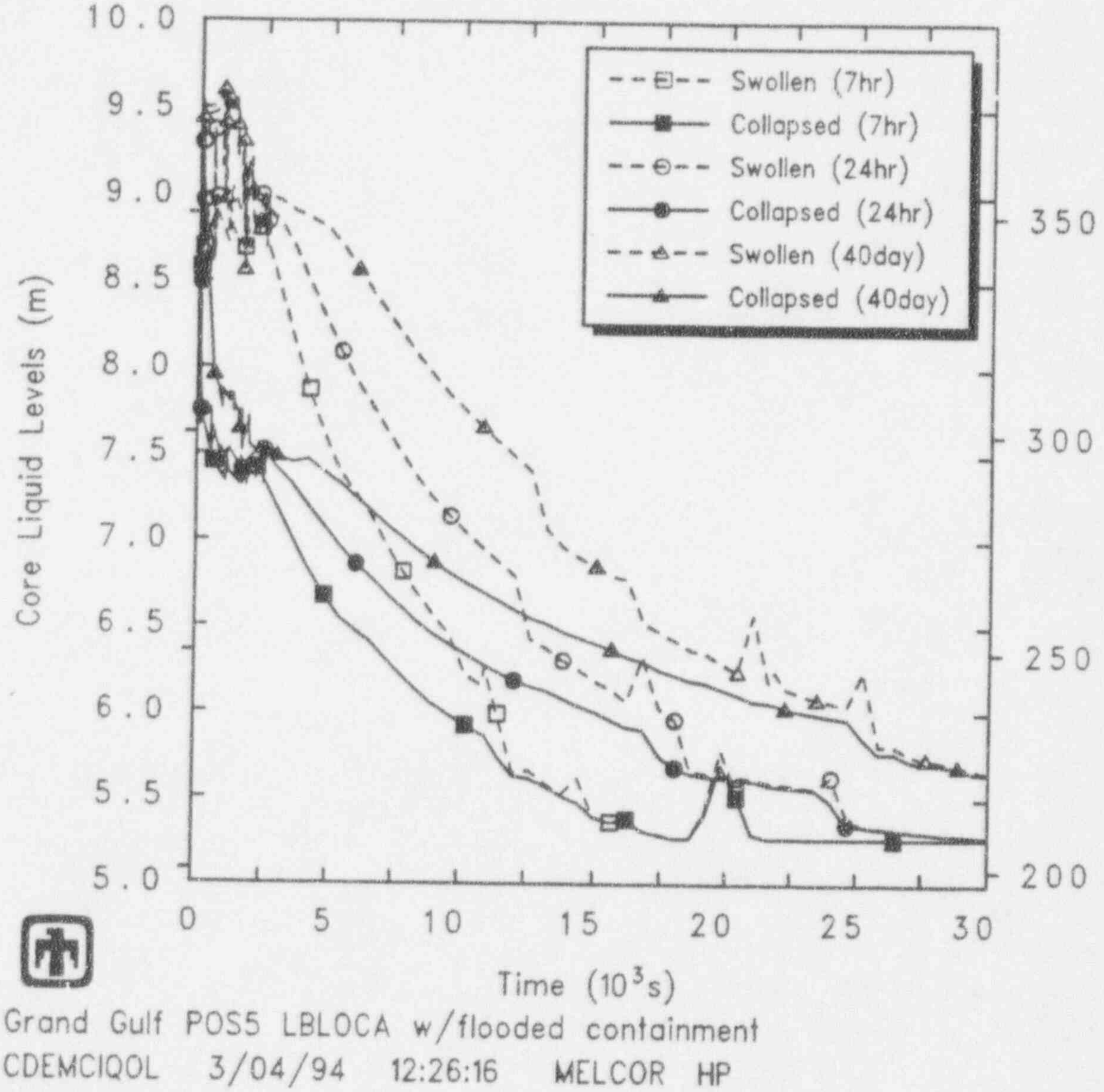


Figure 4.3.1.4. Core Liquid Levels for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated at Various Times After Shutdown

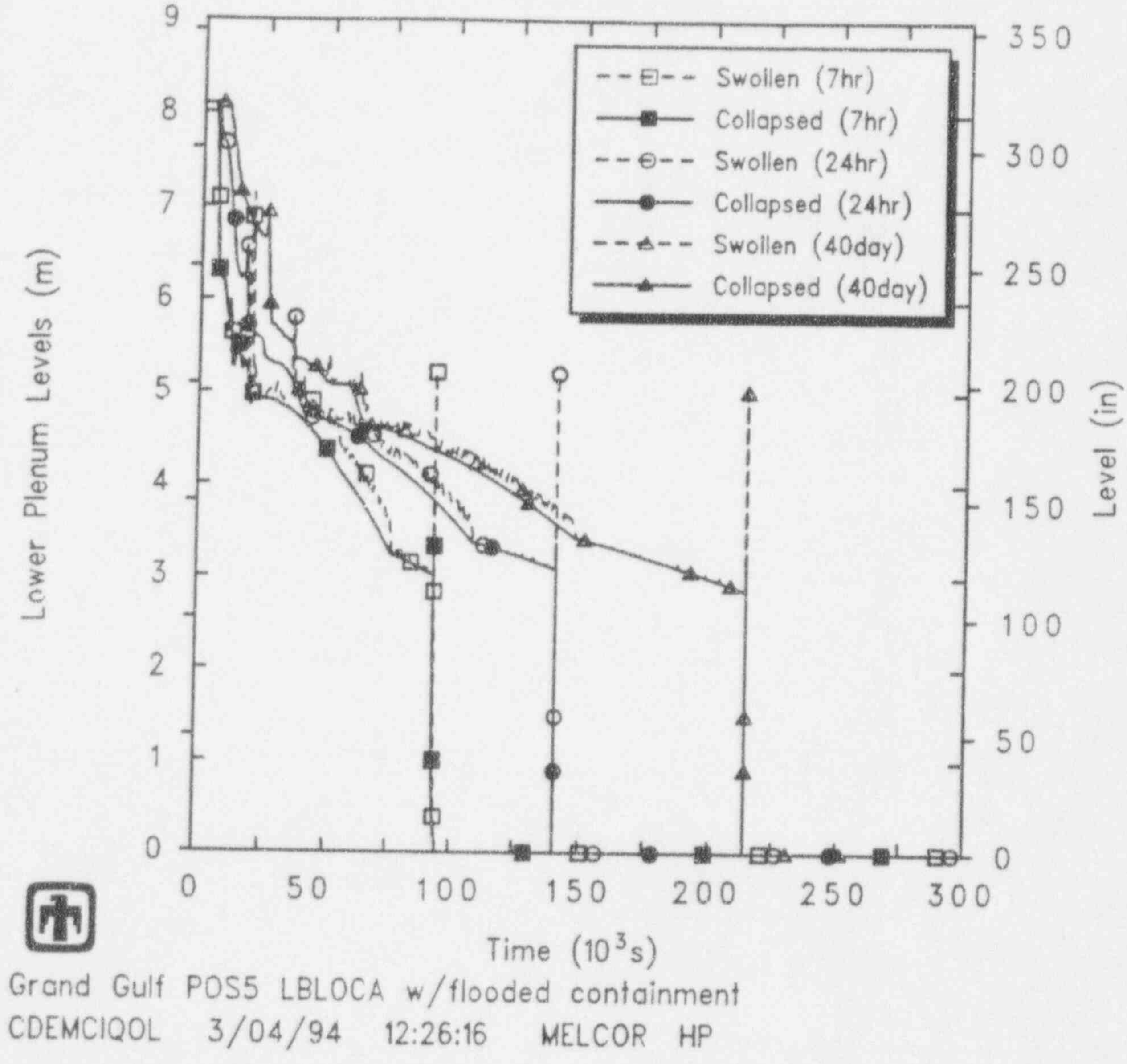


Figure 4 3.1.5. Lower Plenum Liquid Levels for Grand Gulf POS 5 – Large Break LOCA with Flooded Containment, Initiated at Various Times After Shutdown

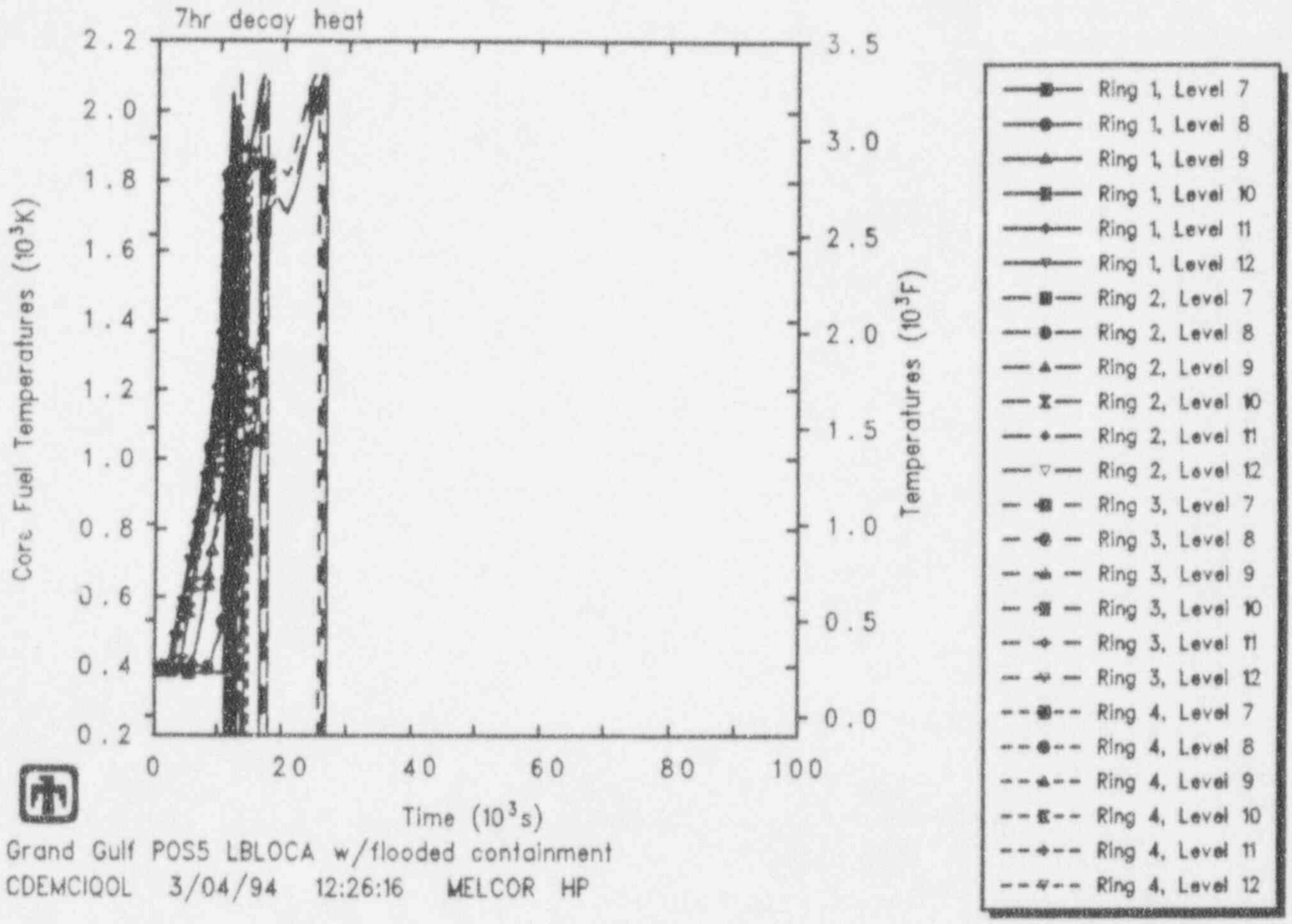


Figure 4.3.1.6. Core Intact Fuel/Clad Temperatures for Grand Gulf POS 5 – Large Break LOCA with Flooded Containment, Initiated 7 hr After Shutdown

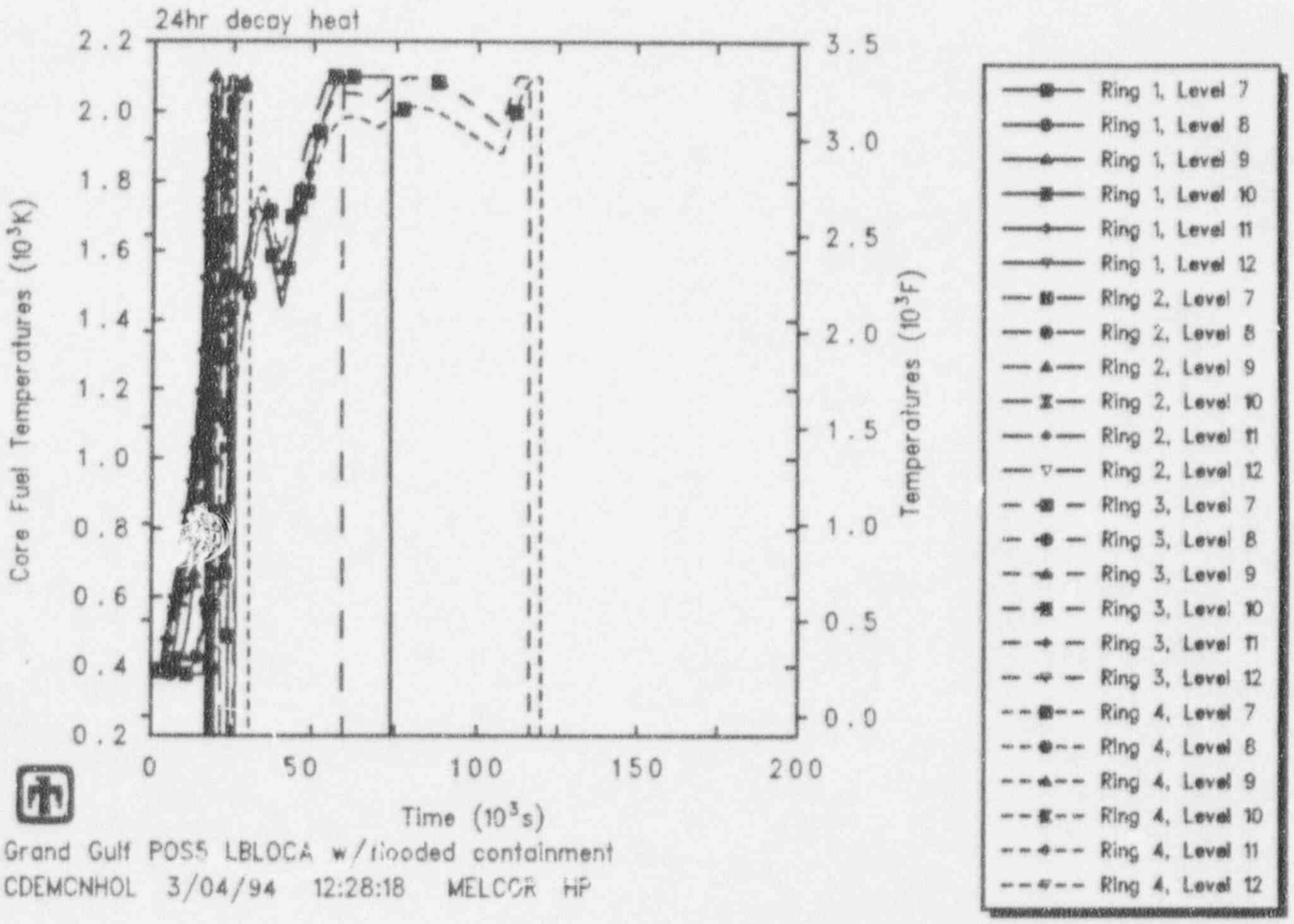


Figure 4.3.1.7. Core Intact Fuel/Clad Temperatures for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 24 hr After Shutdown

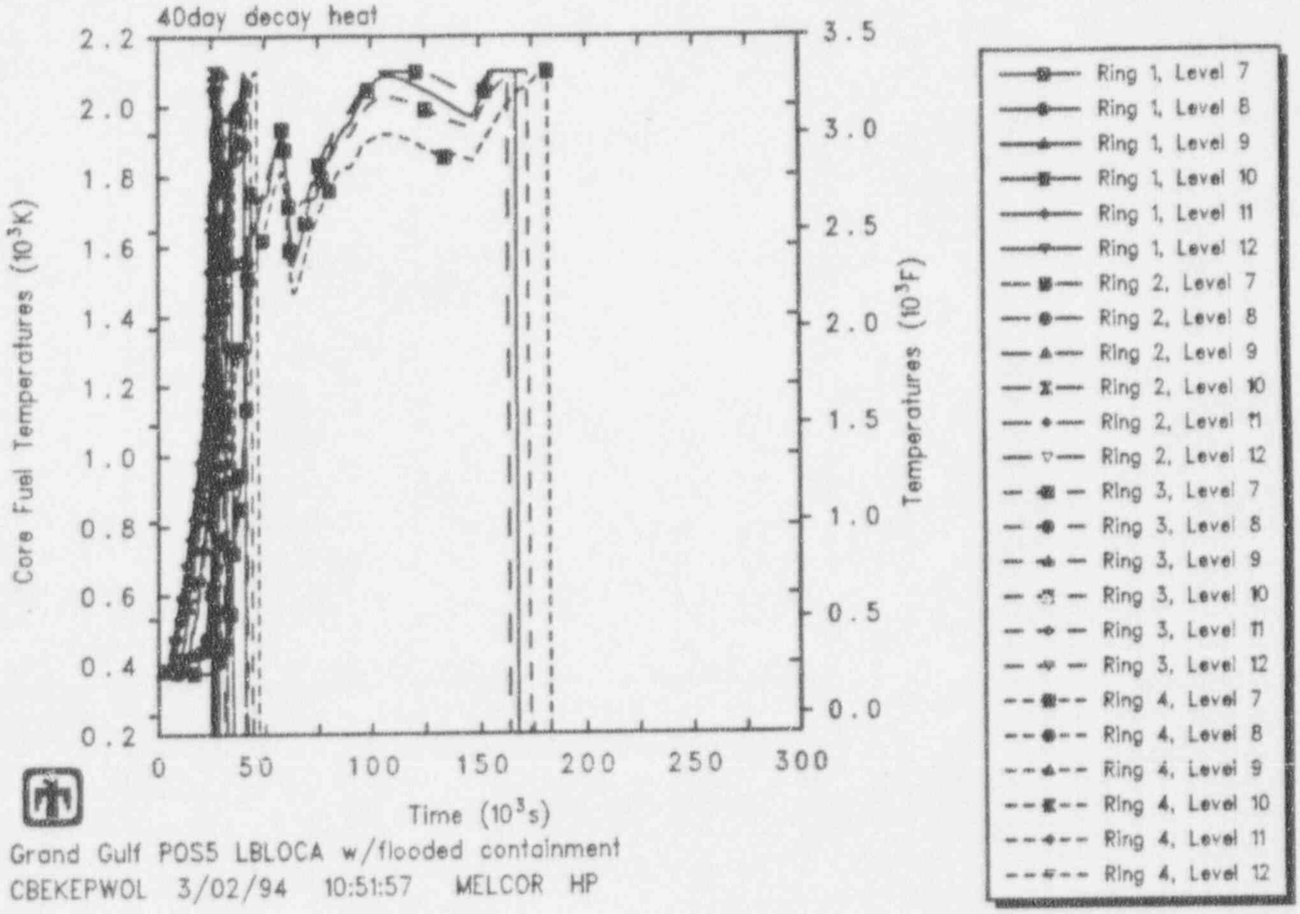


Figure 4.3.1.8. Core Intact Fuel/Clad Temperatures for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 40 day After Shutdown

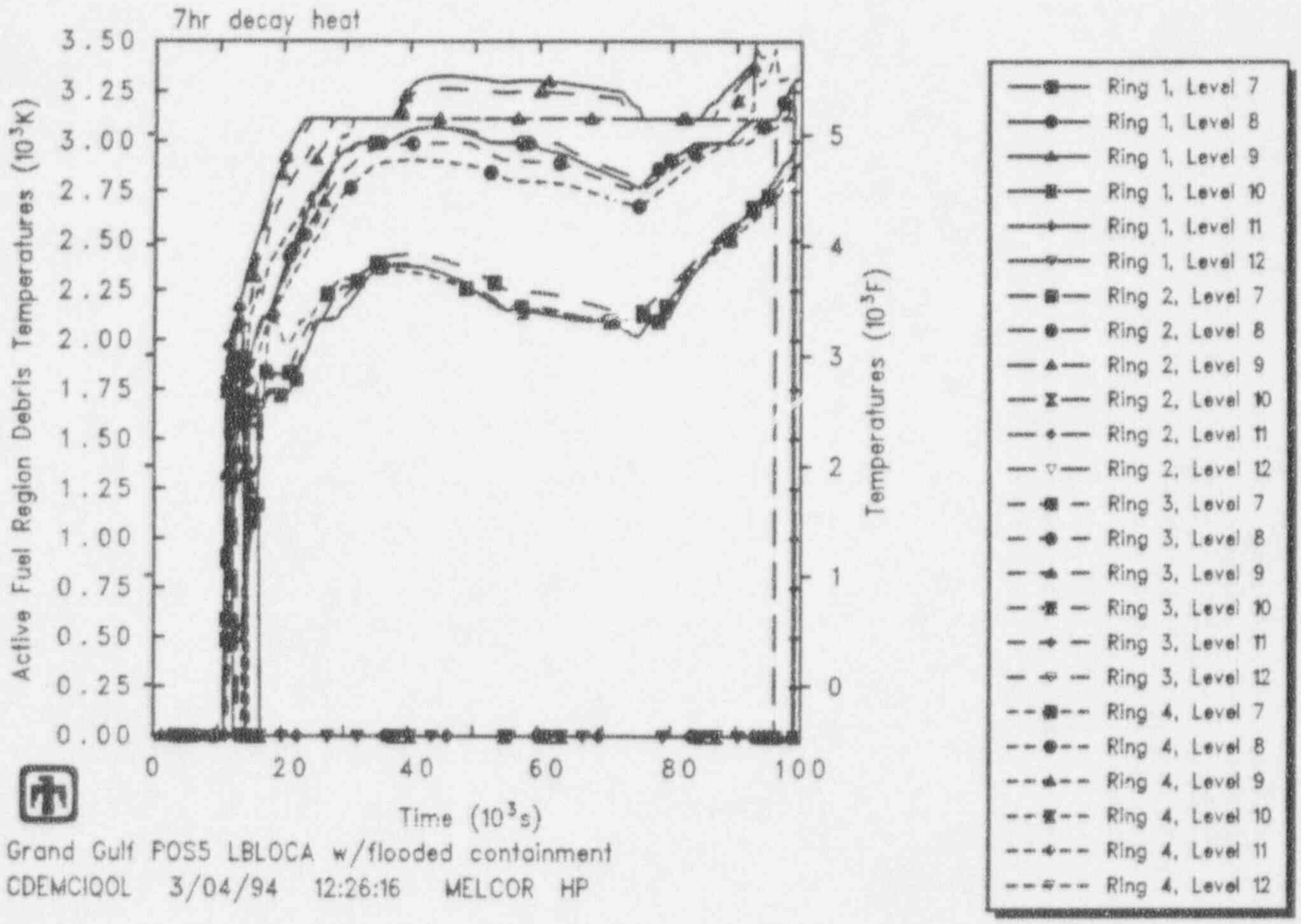


Figure 4.3.1.9. Core Active Fuel Region Debris Bed Temperatures for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 7 hr After Shutdown

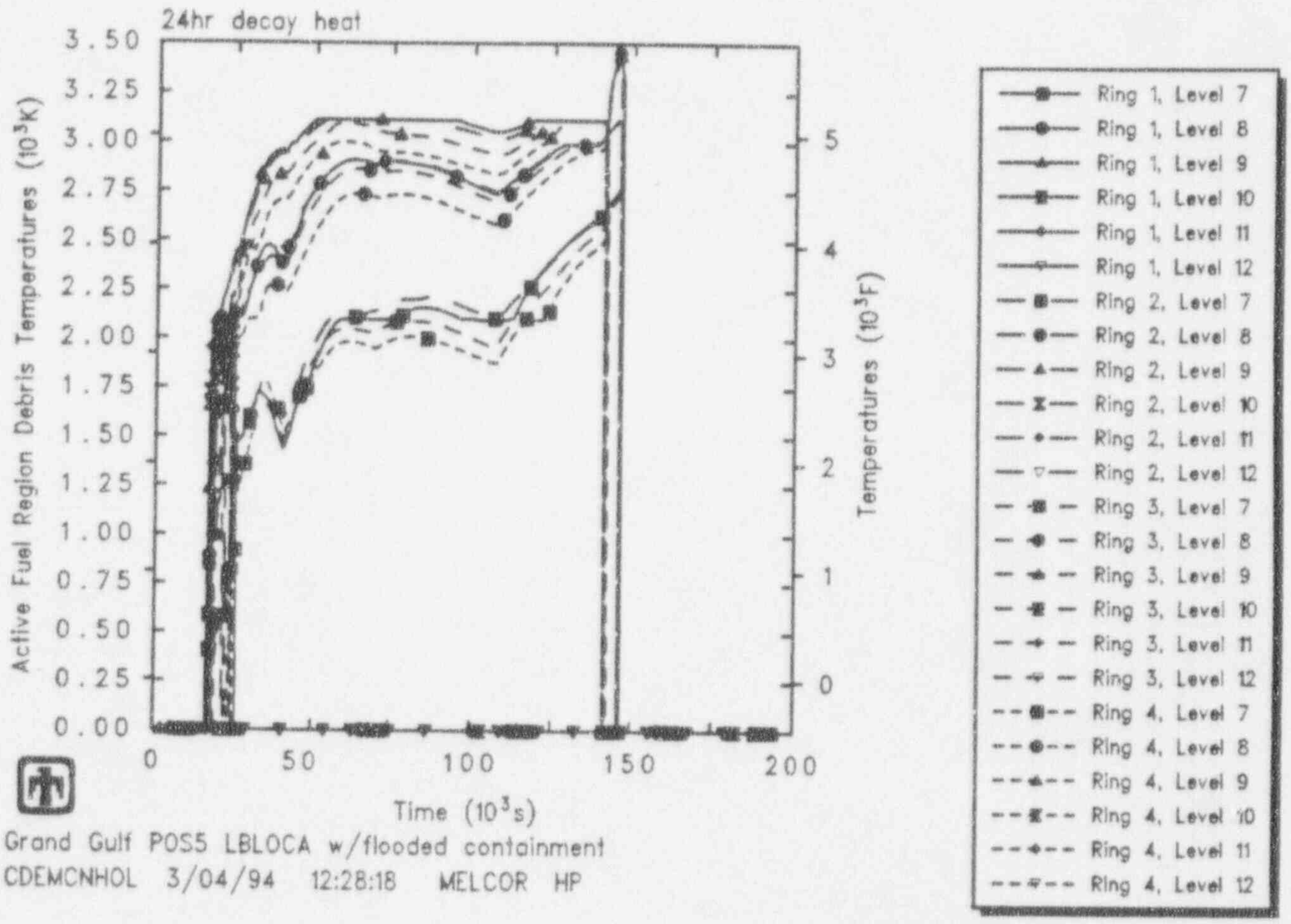


Figure 4.3.1.10. Core Active Fuel Region Debris Bed Temperatures for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 24 hr After Shutdown

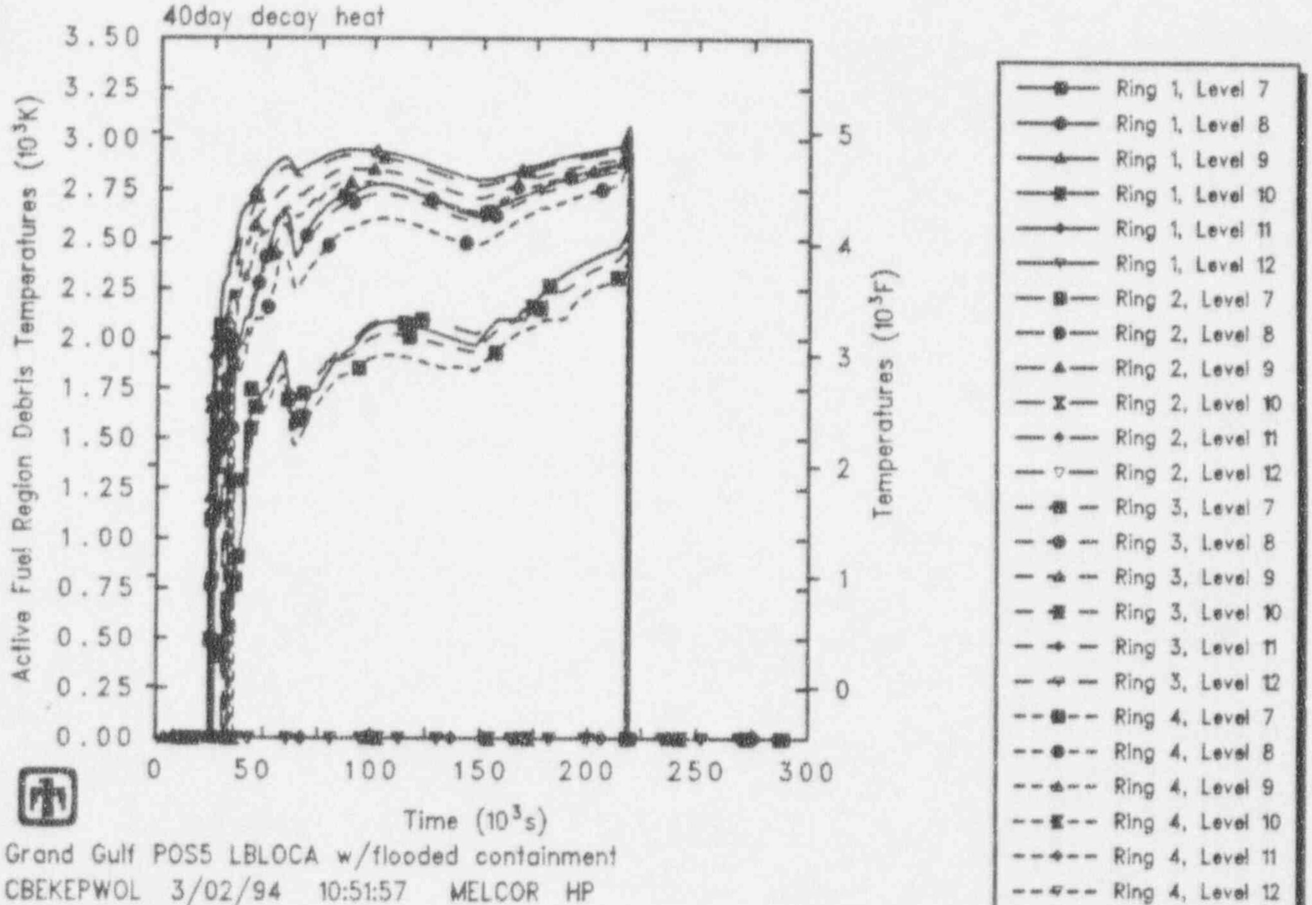


Figure 4.3.1.11. Core Active Fuel Region Debris Bed Temperatures for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 40 day After Shutdown

The predicted temperatures in the debris bed in the lower plenum and core plate are given in Figures 4.3.1.12 through 4.3.1.14, for scenarios initiated at 7 hr, 24 hr and 40 days after shutdown, respectively. In all cases, prior to core support plate failure there is some cold, refrozen debris both on the core support plate (level 5) and on the lower core structural material just above the core support plate (level 6); the cooling and refreezing of this debris is the cause of the continued gradual drop in lower plenum liquid level due to steaming seen in Figure 4.3.1.5. The debris temperature rises gradually to the core support plate failure temperature of 1273 K (1832°F). After core support plate failure, hot high-temperature debris begins appearing the the lower plenum as debris falls from the active fuel region into the lower plenum. The lower head penetrations begin failing almost immediately, and the lower plenum debris temperatures begin dropping to zero as debris is ejected from the vessel to the cavity.

Figures 4.3.1.15 through 4.3.1.17 indicate what fraction of each material in the active fuel region has collapsed into a debris rubble bed held up by the core support plate, prior to core plate failure and subsequent lower head failure and debris ejection, for this large break LOCA scenario initiated at 7 hr, 24 hr and 40 days after shutdown, respectively. The debris bed forms relatively quickly, taking 10,000-20,000 s to reach its final configuration. The fraction of material in the debris bed then remains nearly constant for 50,000-100,000 s as the debris material continues to heat up.

Figure 4.3.1.18 shows the total masses of core materials (UO_2, Zircaloy and ZrO_2, stainless steel and steel oxide, and control rod poison) remaining in the vessel. This includes both material in the active fuel region and in the lower plenum. Debris ejection began very soon after lower head failure. This figure illustrates that most of the core material was lost from the vessel to the cavity quickly, in step-like stages. In all cases, all of the UO_2 was transferred to the cavity within ~1 hr after initial lower head penetration failure, as was the unoxidized zircaloy, the associated zirc oxide and the control rod poison. A small fraction (1-10%) of the structural steel in the lower plenum, and some associated steel oxide, was predicted to remain unmelted and in place throughout the entire transient period (most noticably for the sequence initiated 40 day after scram).

The debris material lost from the vessel is ejected to the reactor pedestal cavity. Since almost all the material in the core active fuel region and lower plenum is lost within a very short time period after vessel failure, the core debris mass in the cavity is about the same for this sequence initiated at three different times after scram. Figure 4.3.1.19 indicates that the amount of concrete ablated and the total cavity debris mass (*i.e.*, core debris combined with concrete ablation products) is also similar for this sequence initiated at three different times after scram, except for a shift in timing (with debris ejection occurring and core-concrete interaction beginning later at lower decay heat levels than for higher decay heat levels). In all cases, concrete ablation is quite rapid soon after debris ejection (while the core debris is hot, >2000 K, and consists of a layer of metallic debris above a heavy oxide layer), and concrete ablation slows significantly after a short time (after enough concrete has been ablated for the debris bed configuration to invert to a light oxide layer above a layer of metallic debris, mixed to a lower average temperature of ~1500 K).

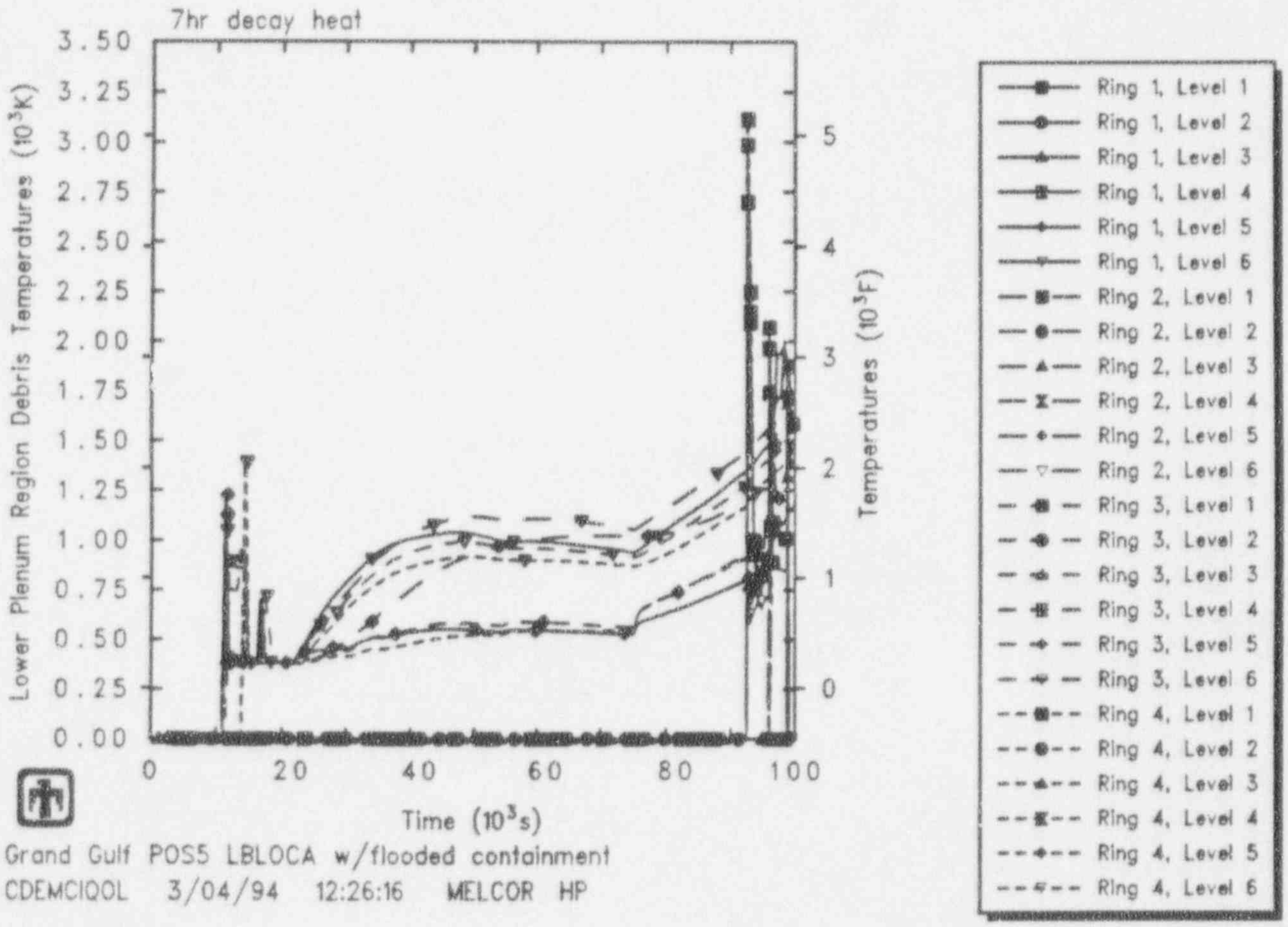


Figure 4.3.1.12. Core Lower Plenum and Core Support Plate Debris Bed Temperatures for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 7 hr After Shutdown

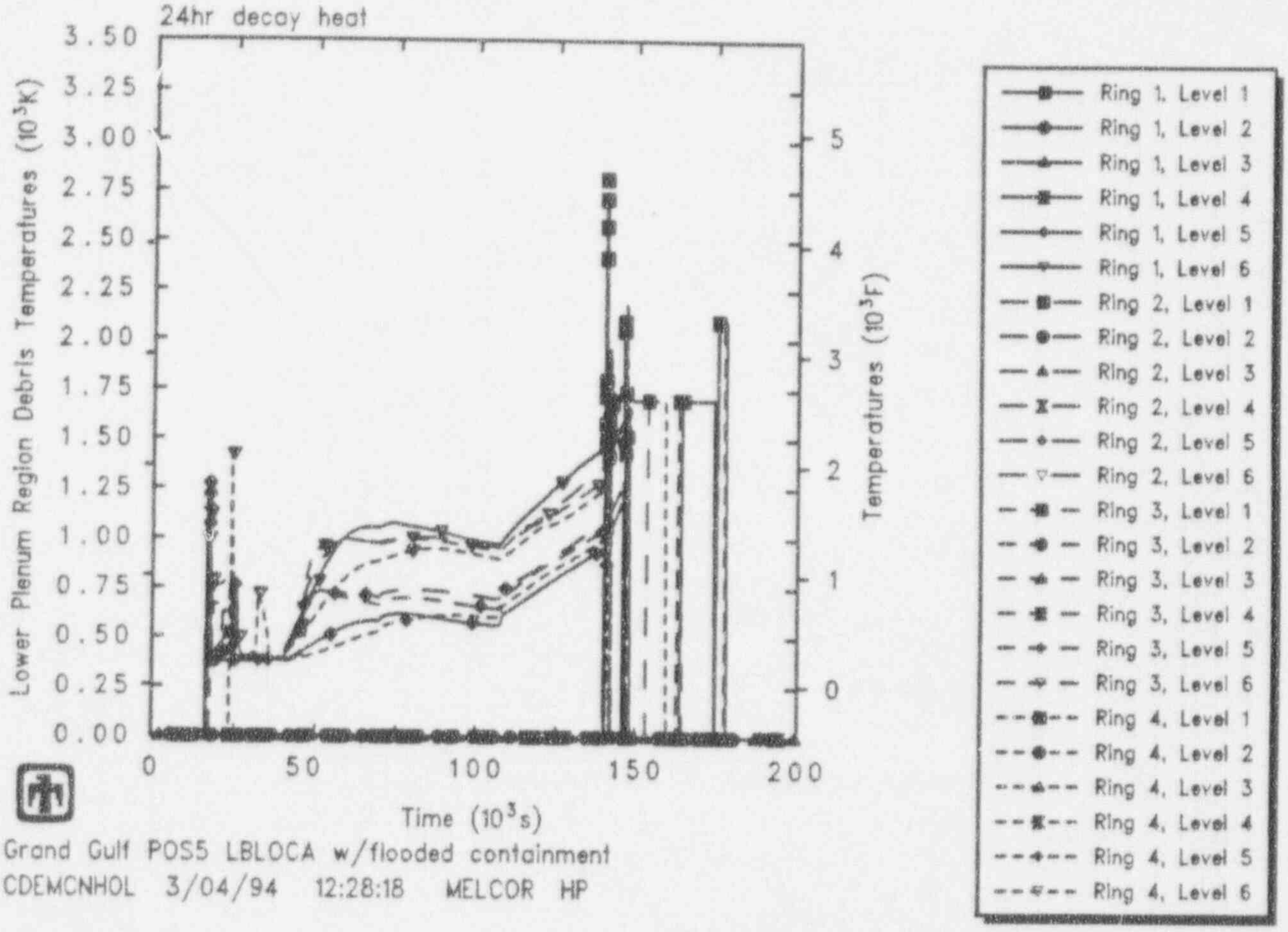


Figure 4.3.1.13. Core Lower Plenum and Core Support Plate Debris Bed Temperatures for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 24 hr After Shutdown

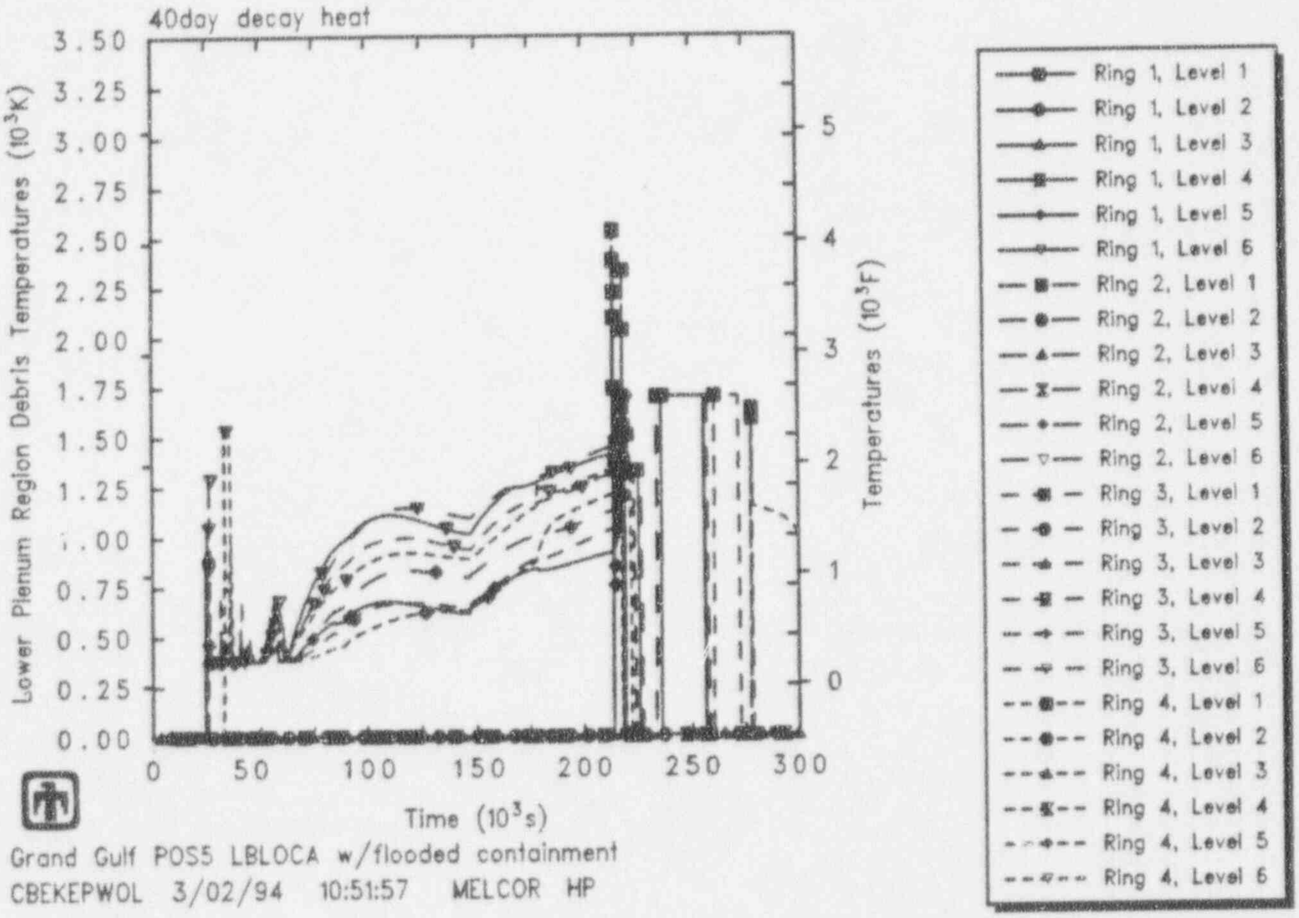


Figure 4.3.1.14. Core Lower Plenum and Core Support Plate Debris Bed Temperatures for Grand Gulf POS 5 – Large Break LOCA with Flooded Containment, Initiated 40 day After Shutdown

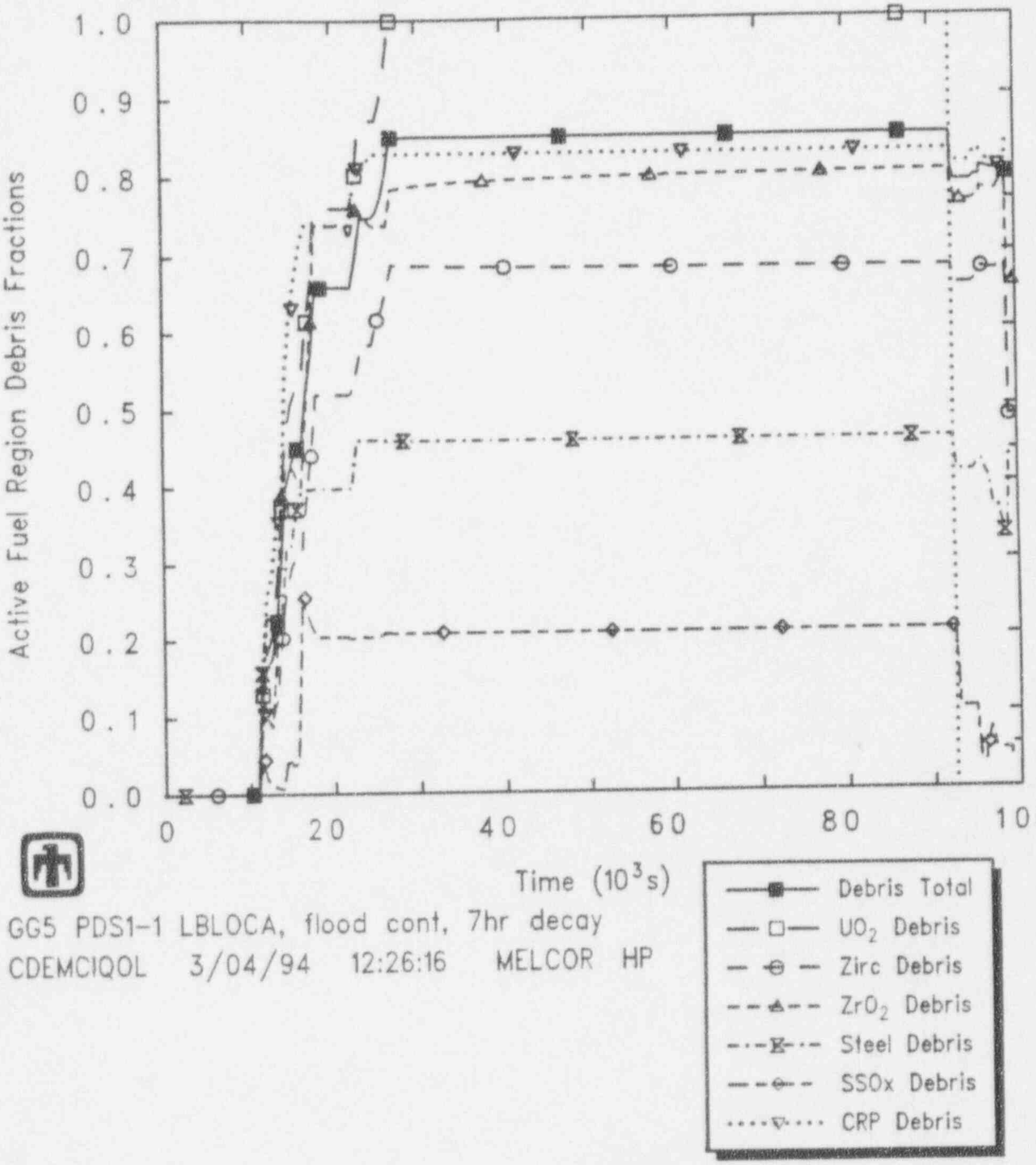


Figure 4.3.1.15. Core Active Fuel Region Degraded Material Fractions for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 7 hr After Shutdown

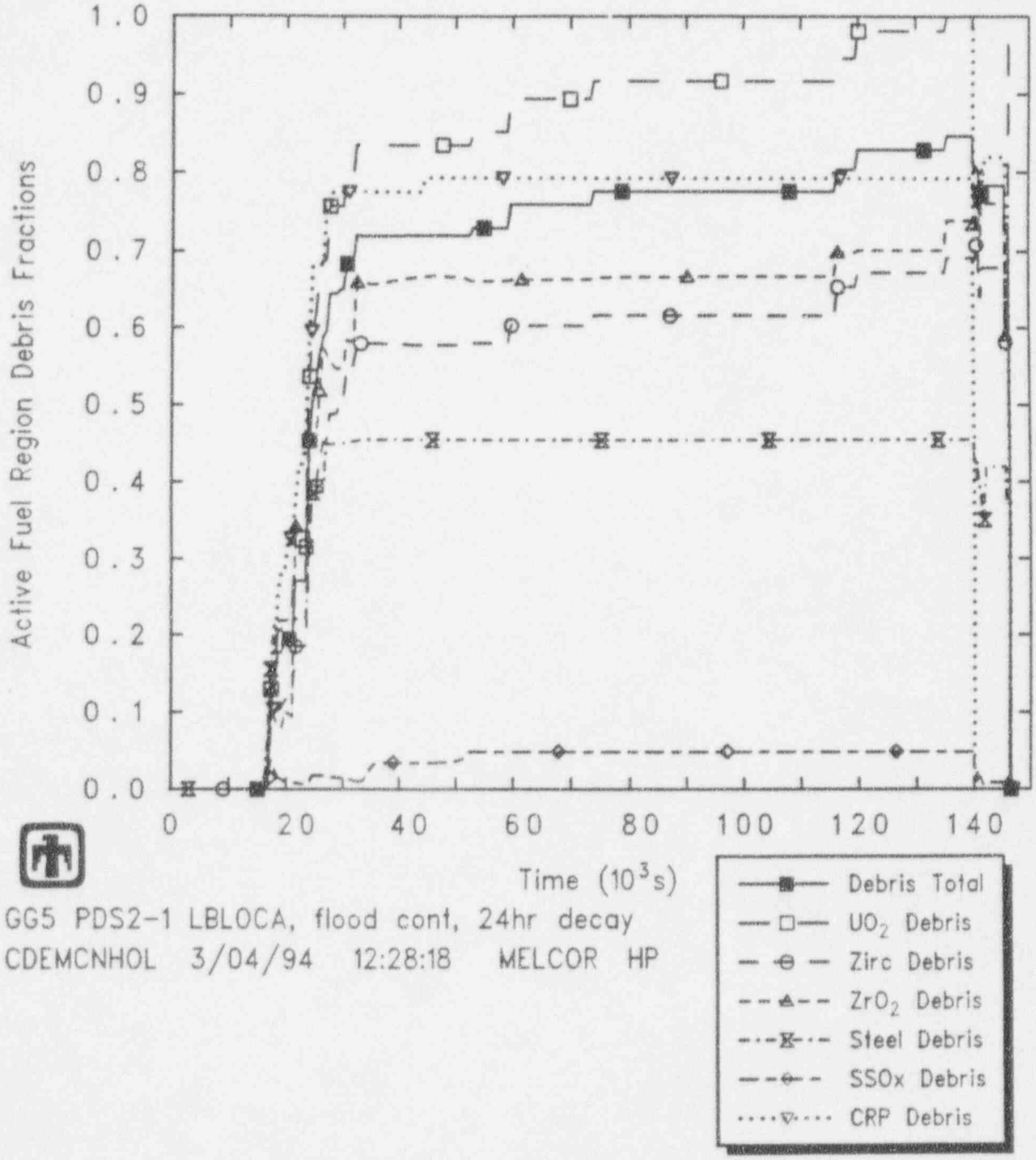


Figure 4.3.1.16. Core Active Fuel Region Degraded Material Fractions for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 24 hr After Shutdown

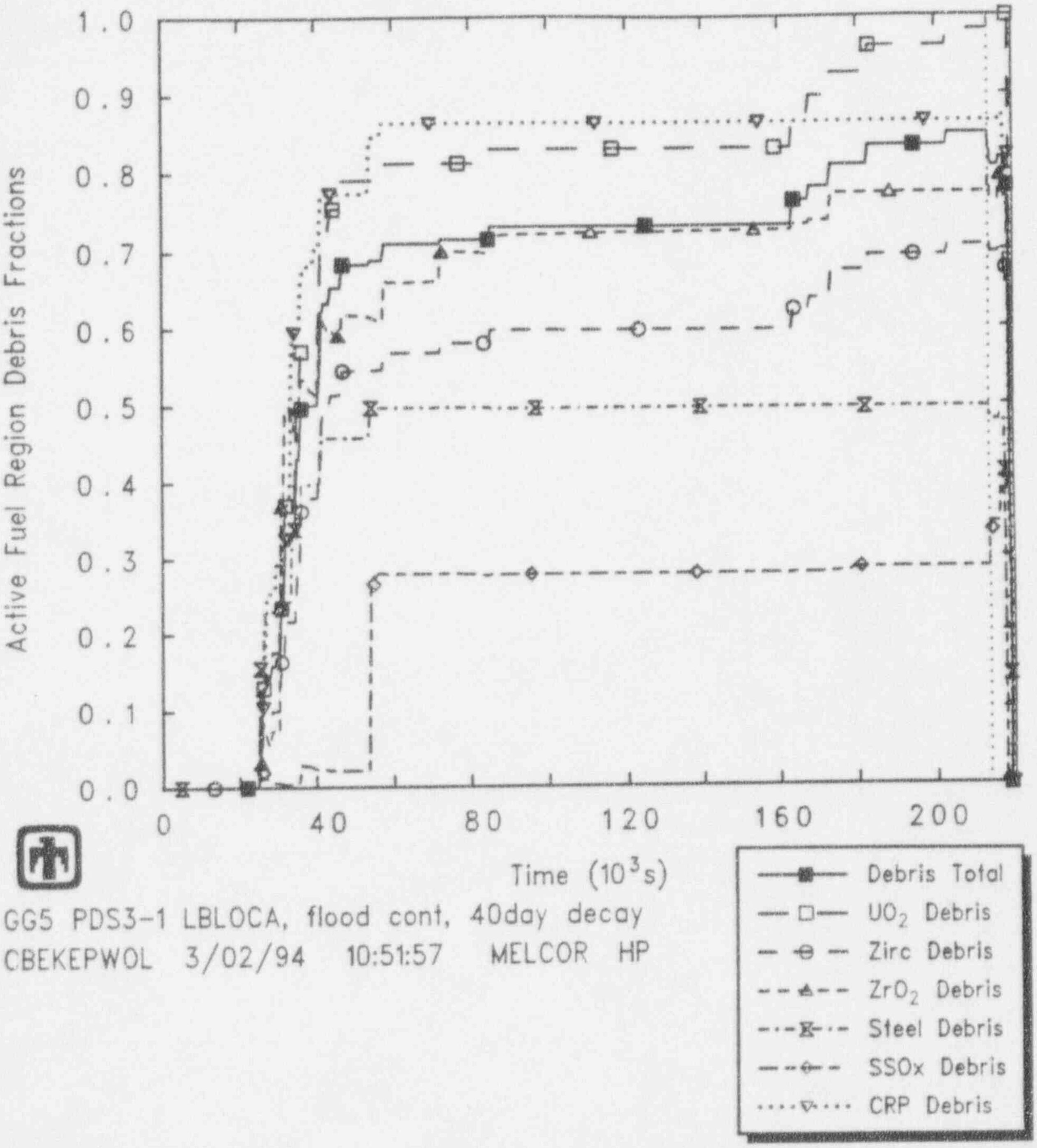


Figure 4.3.1.17. Core Active Fuel Region Degraded Material Fractions for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 40 day After Shutdown

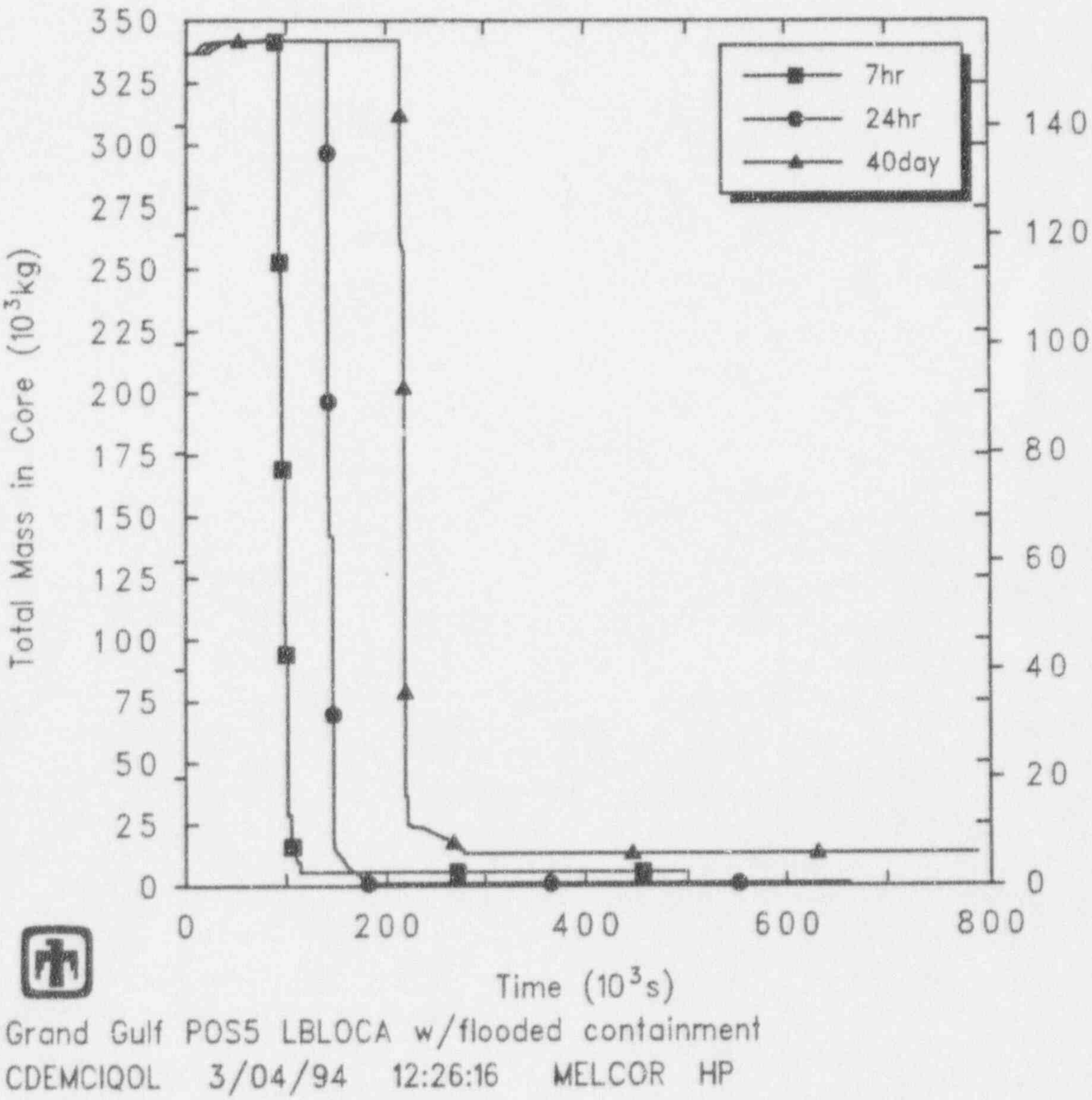


Figure 4.3.1.18. Total Core Material Masses for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated at Various Times After Shutdown

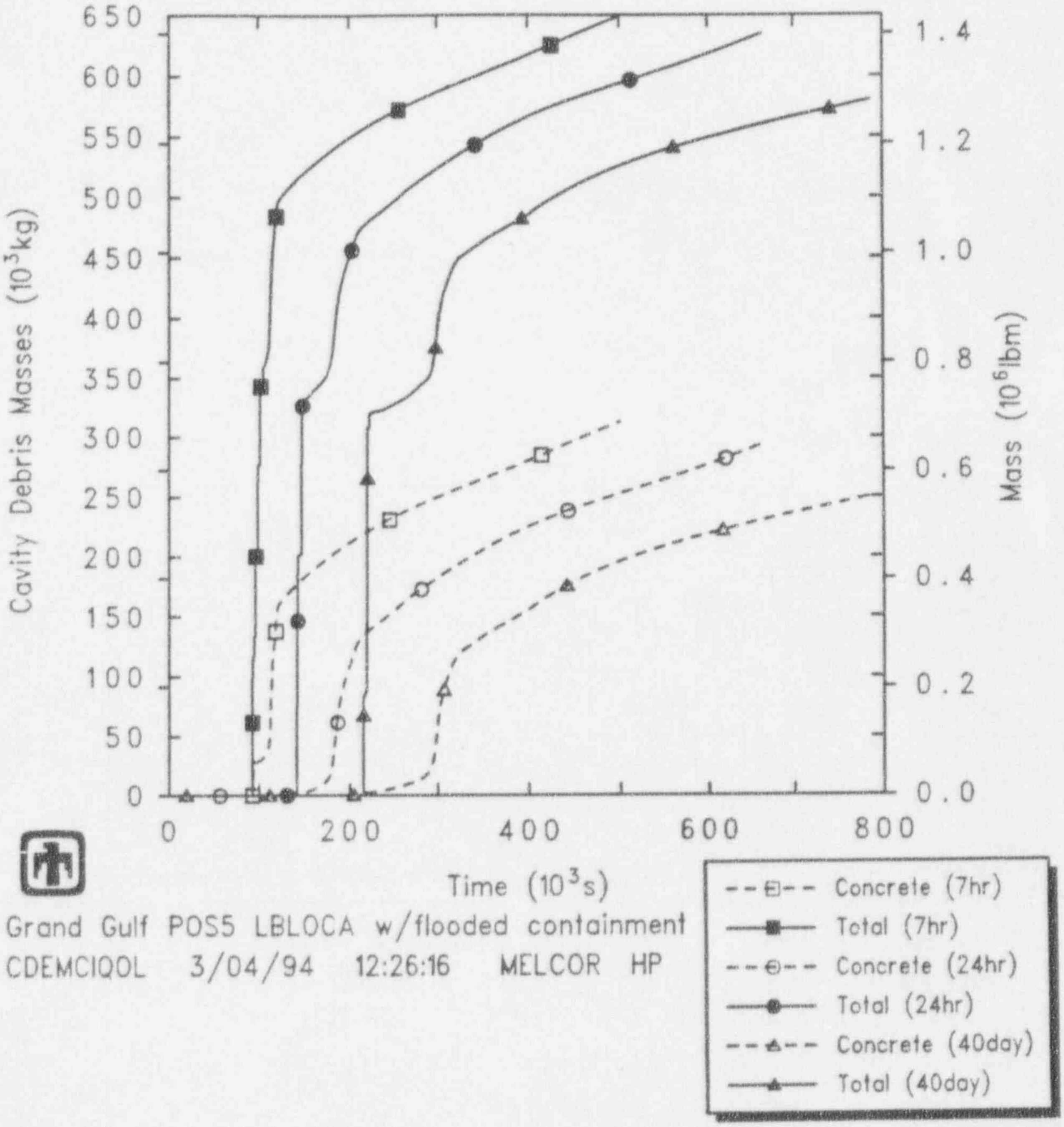


Figure 4.3.1.19. Cavity Total and Concrete Debris Masses for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated at Various Times After Shutdown

The calculated production of noncondensable gases (H_2, CO, CO_2 and H_2O) is summarized in Figure 4.3.1.20. The hydrogen production shown includes both in-vessel production (the initial step increase) and ex-vessel production in the cavity (the later-time increase). The in-vessel hydrogen generation corresponds to the oxidation of about 15-20% of the zircaloy and about 1-2% of the steel in the core and lower plenum, prior to vessel failure and debris ejection. As soon as the core debris enters the cavity, core-concrete interaction begins, resulting in the production of carbon dioxide and hydrogen; reduction of these gases by the molten metal in the core debris also gives rise to carbon monoxide and hydrogen.

This generation of noncondensables changes the composition of the atmosphere in the containment and in the auxiliary building. The mole fractions in the drywell, containment dome, containment equipment hatch and auxiliary building (second floor) are presented in Figures 4.3.1.21 through 4.3.1.23 for this sequence initiated at various times after shutdown, including a vertical dotted line at vessel failure for reference. The drywell control volume atmosphere consists mostly of steam both before and after vessel and auxiliary building failure. The atmosphere composition in the outer containment volumes and in most of the auxiliary building is generally similar, with little steam or hydrogen (about 5% each) present before vessel failure but a steadily increasing steam concentration and potentially flammable amounts of hydrogen and CO building up late in time. The behavior is qualitatively the same in all three cases, just stretched out in time more at the lower decay heat levels compared to higher decay heats.

Figures 4.3.1.24 through 4.3.1.26 illustrate the time-dependent release of radionuclides from the fuel debris both within the vessel and in the cavity, for cases initiated 7 hr, 24 hr and 40 day after scram, respectively. The vertical dotted lines within the plots mark the time of vessel failure, indicating that most of the in-vessel release occurs prior to vessel failure, from the hot debris bed in the active fuel region, while most of the ex-vessel release occurs within a short time period after vessel failure and debris ejection to the cavity, while the core debris is still hot, >2000 K, and consists of a layer of metallic debris above a heavy oxide layer, before enough concrete has been ablated for the debris bed configuration to invert to a light oxide layer above a layer of metallic debris, mixed to a lower average temperature of ~1500 K. Table 4.3.1.2 summarizes the in-vessel, ex-vessel and total amounts of each radionuclide class released, all normalized to the initial inventories of each class. (Note that these amounts generally consider only the release of radioactive forms of these classes, and not additional releases of nonradioactive aerosols from structural materials.)

The release behavior predicted by MELCOR can be grouped into several subdivisions. Almost all (≃100%) of the volatile Class 1 (noble gases), Class 2 (CsOH), Class 4 (I_2) and Class 5 (Te) radionuclide species are released, primarily in-vessel, as are most (~75-85%) of the Class 3 (Ba) and Class 12 (Sn) inventories. The next major release fraction, dropping rapidly with lower decay heat levels and cooler debris (as shown in Figures 4.3.1.9 through 4.3.1.11) is for uranium. Around 1% of the total inventories of Ru and Mo, Ce and La, are released. Finally, a total ≤0.01% of the initial inventory of Class 11 (Cd) is predicted to be released. Note that the CORSOR-M fission product

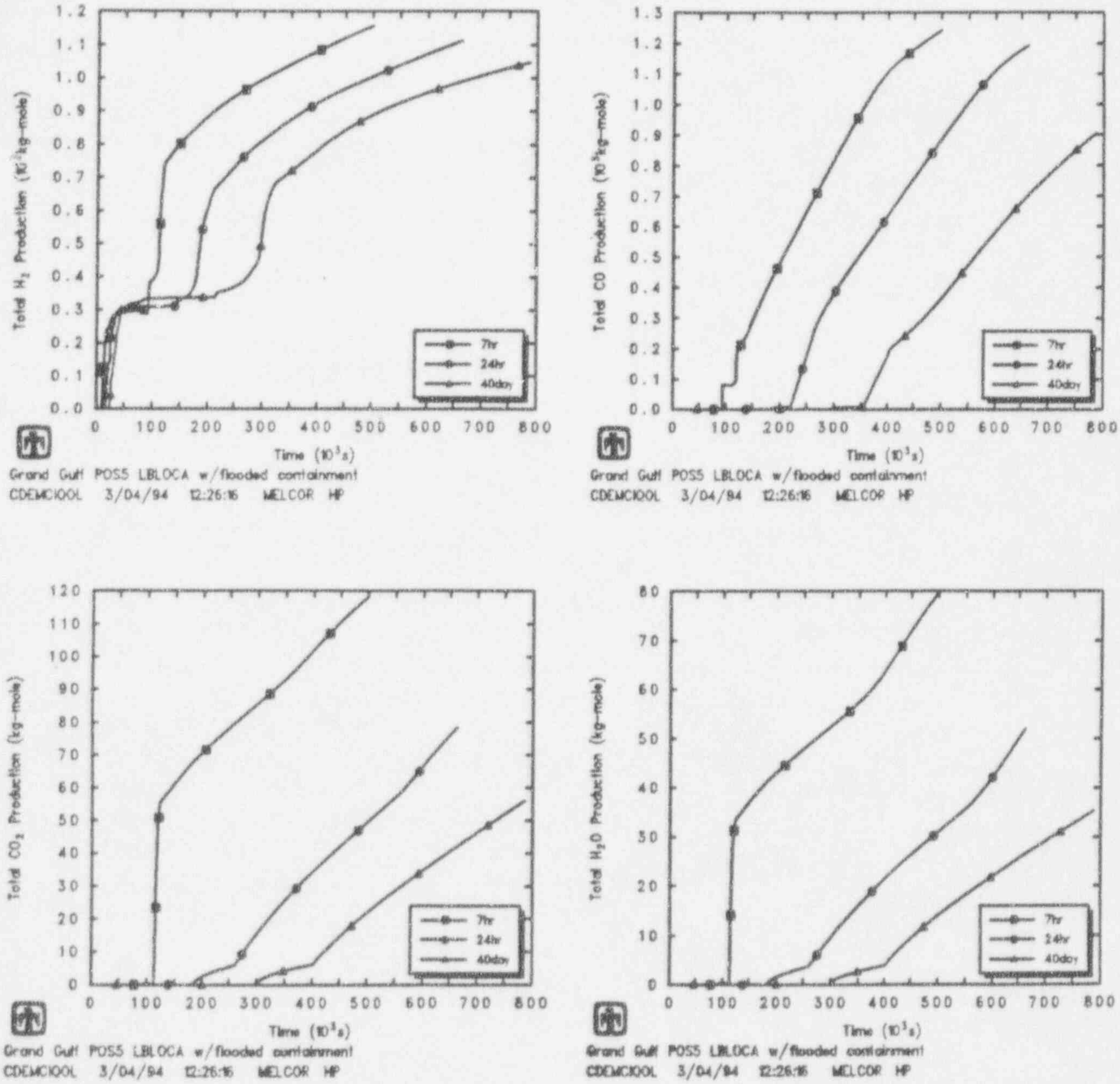


Figure 4.3.1.20. Hydrogen (upper left), Carbon Monoxide (upper right), Carbon Dioxide (lower left) and Steam (lower right) Generation for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated at Various Times After Shutdown

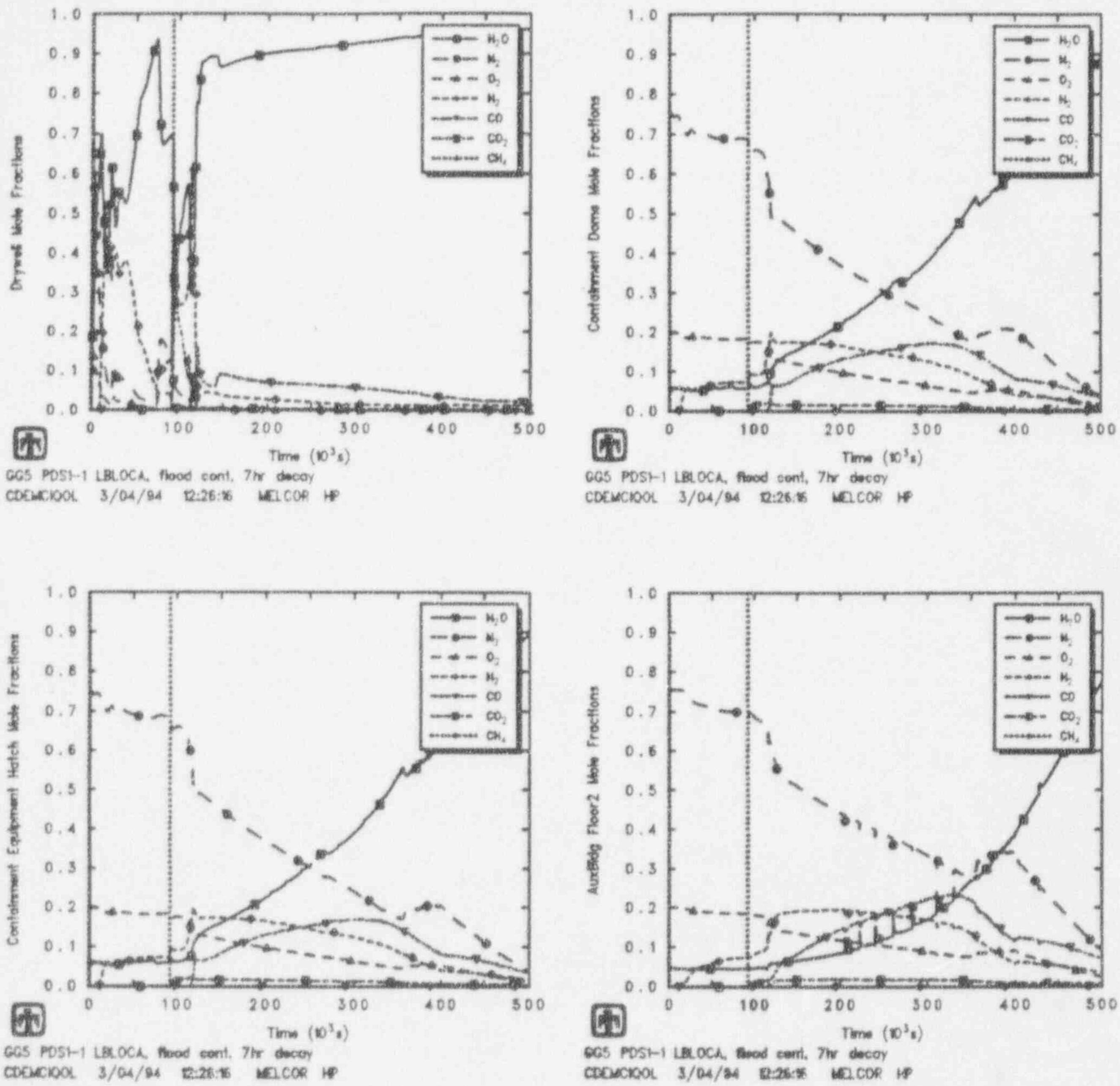


Figure 4.3.1.21. Mole Fractions in Drywell (upper left), Containment Dome (upper right), Containment Equipment Hatch (lower left) and Auxiliary Building (lower right) for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 7 hr After Shutdown

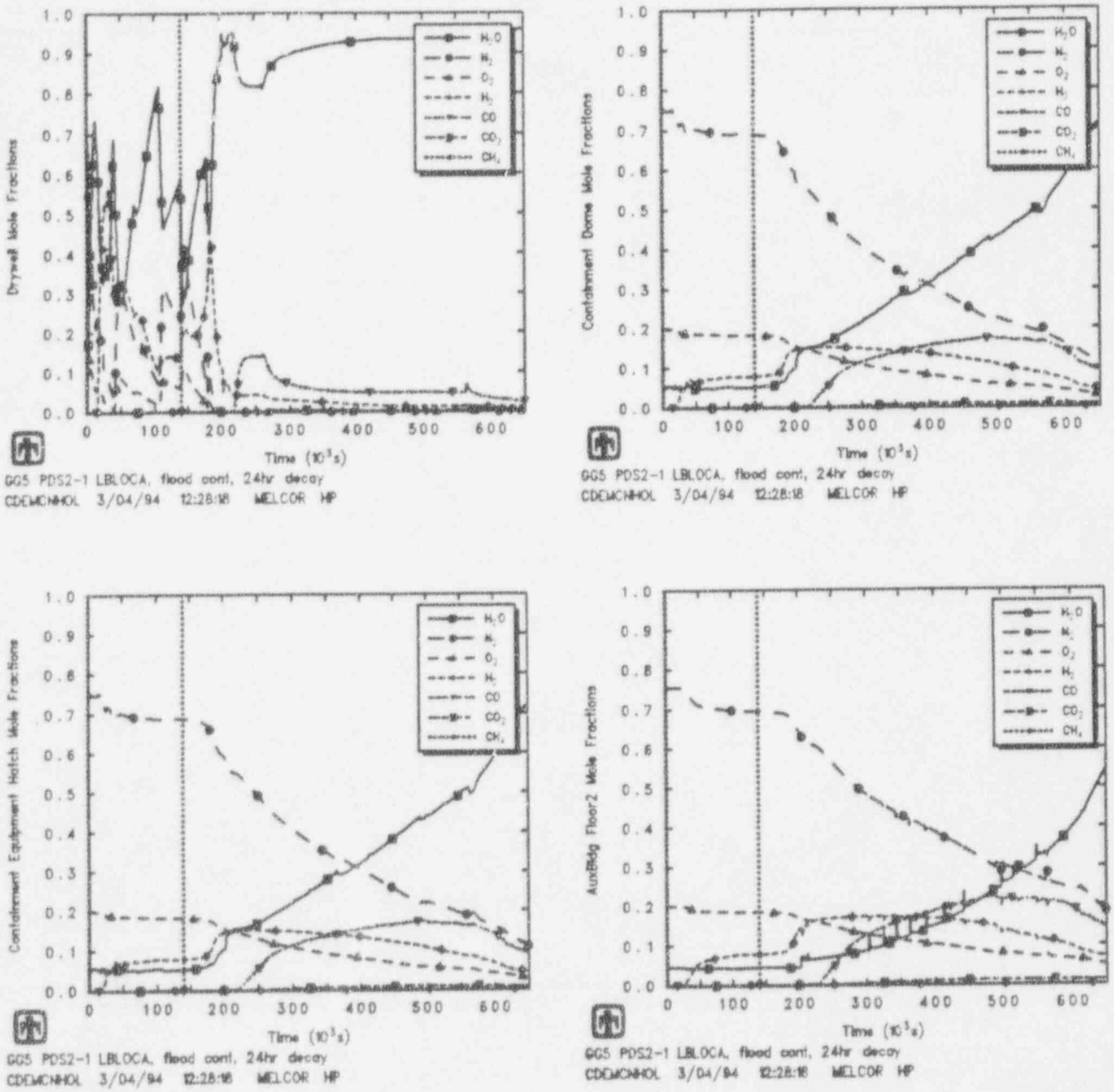


Figure 4.3.1.22. Mole Fractions in Drywell (upper left), Containment Dome (upper right), Containment Equipment Hatch (lower left) and Auxiliary Building (lower right) for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 24 hr After Shutdown

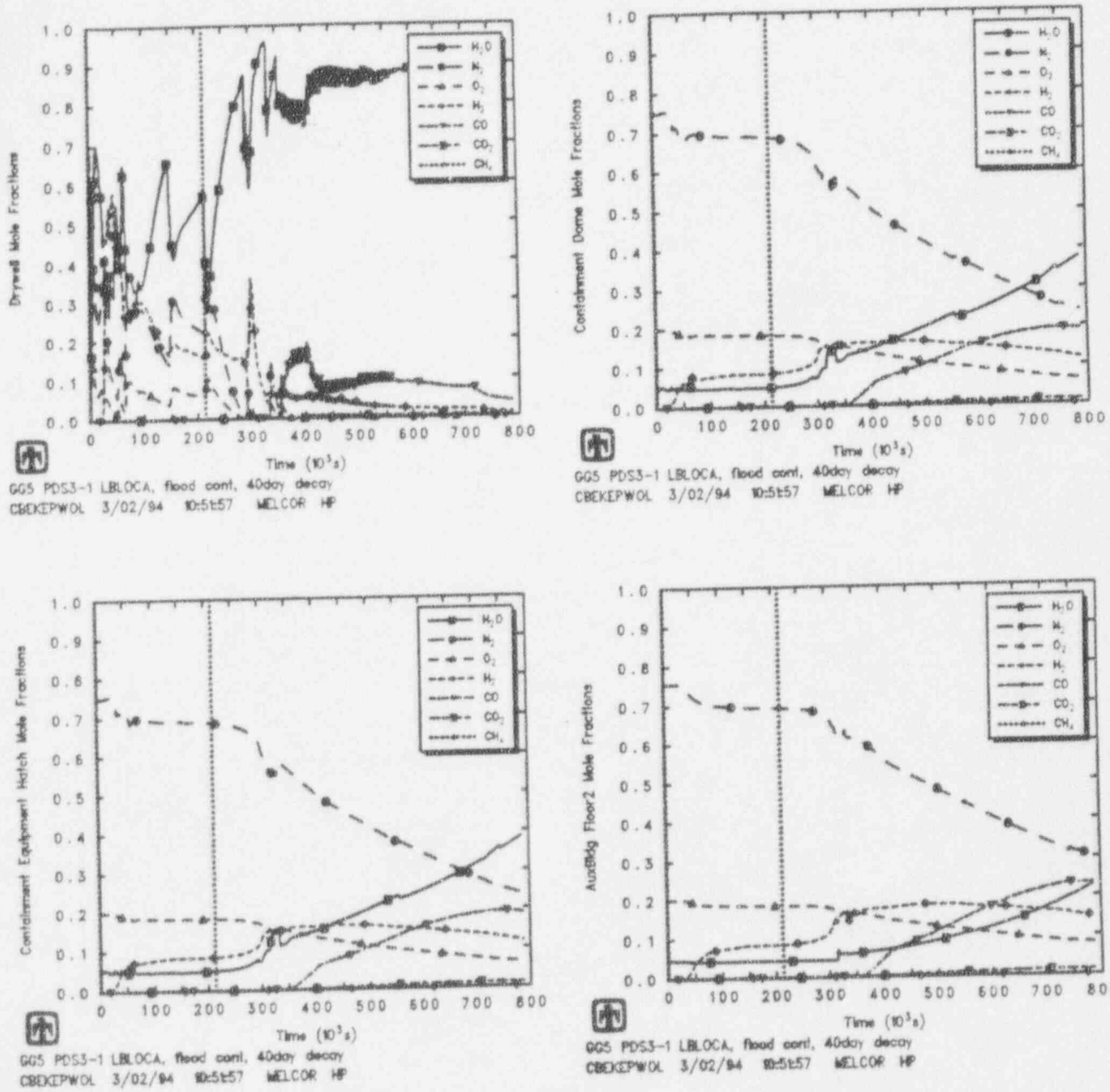


Figure 4.3.1.23. Mole Fractions in Drywell (upper left), Containment Dome (upper right), Containment Equipment Hatch (lower left) and Auxiliary Building (lower right) for Grand Gulf POS 5 – Large Break LOCA with Flooded Containment, Initiated 40 day After Shutdown

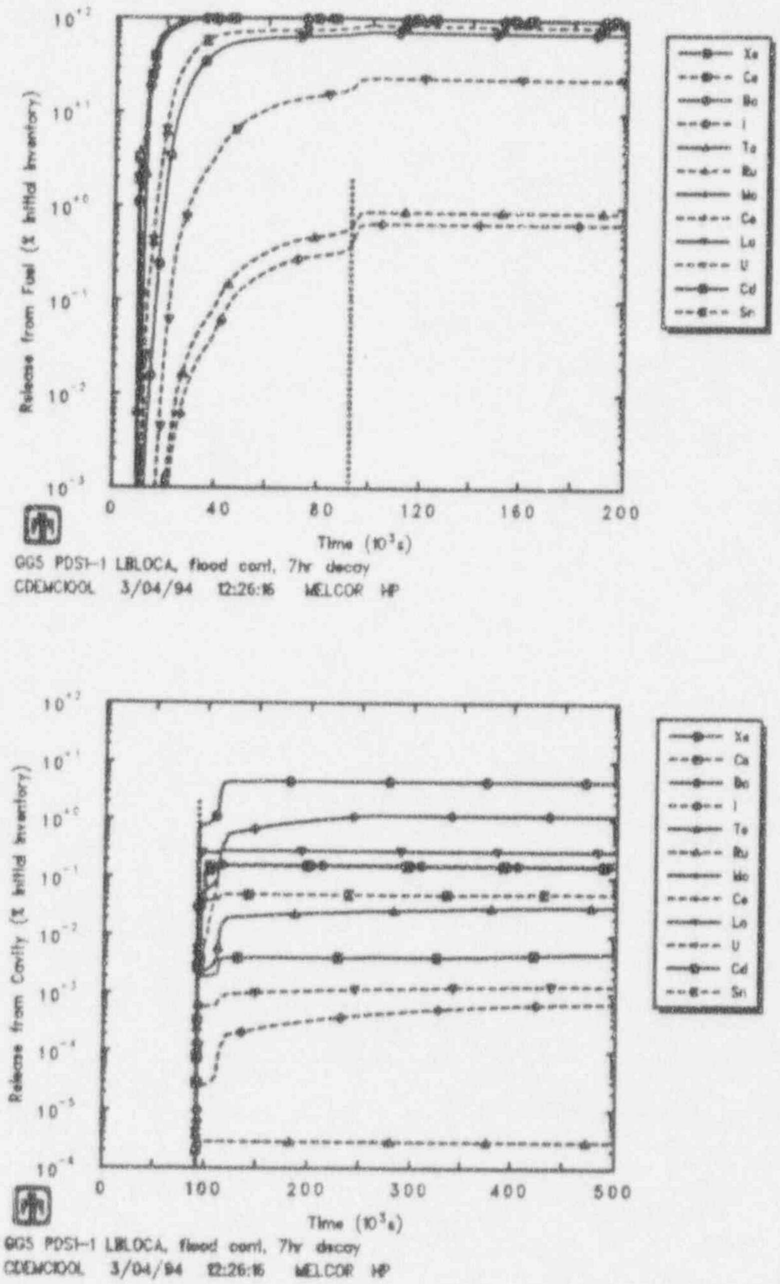


Figure 4.3.1.24. In-Vessel (top) and Ex-Vessel (bottom) Radionuclide Release Fractions for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 7 hr After Shutdown

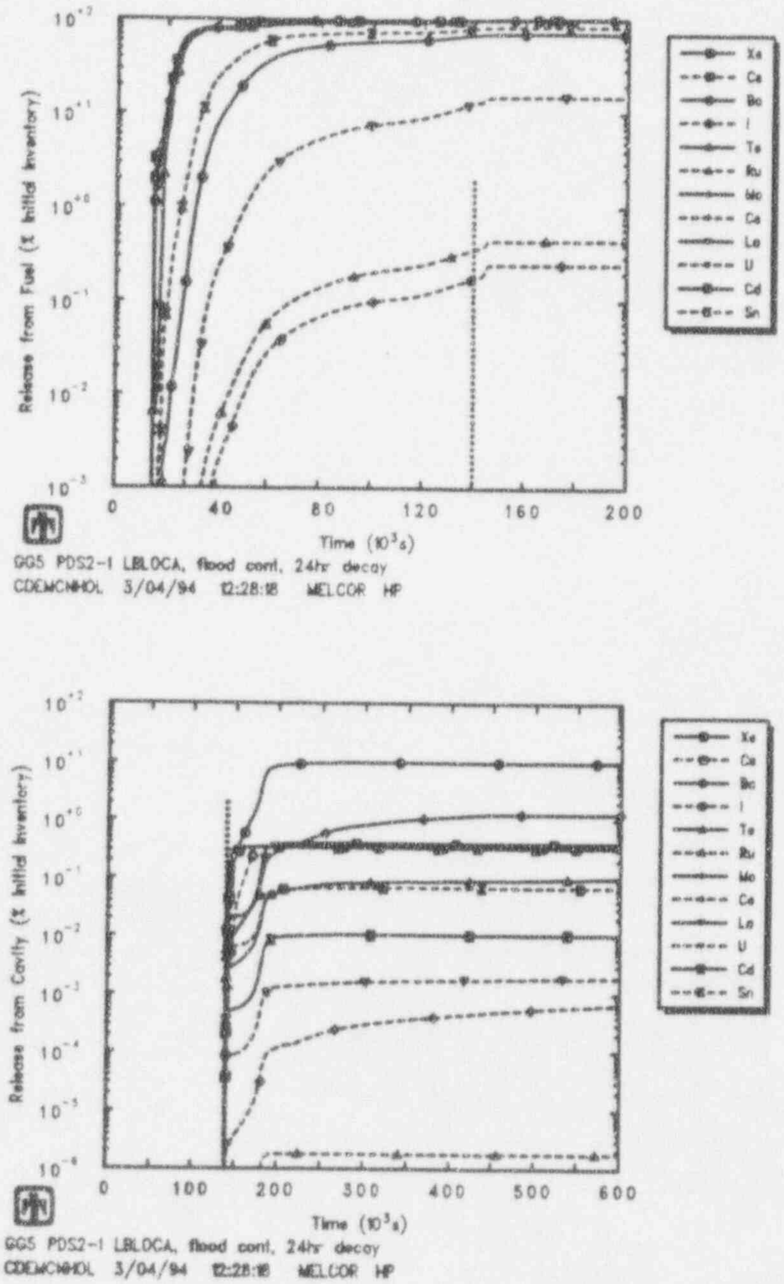
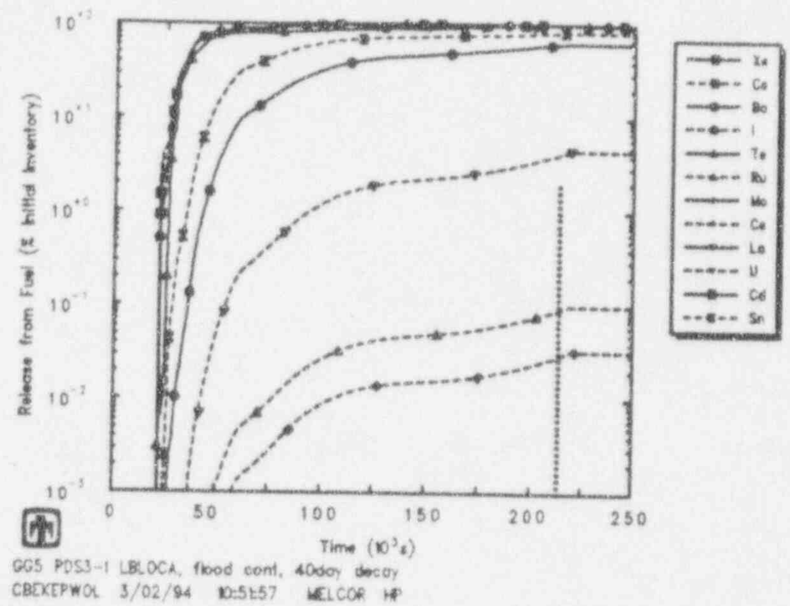


Figure 4.3.1.25. In-Vessel (top) and Ex-Vessel (bottom) Radionuclide Release Fractions for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 24 hr After Shutdown



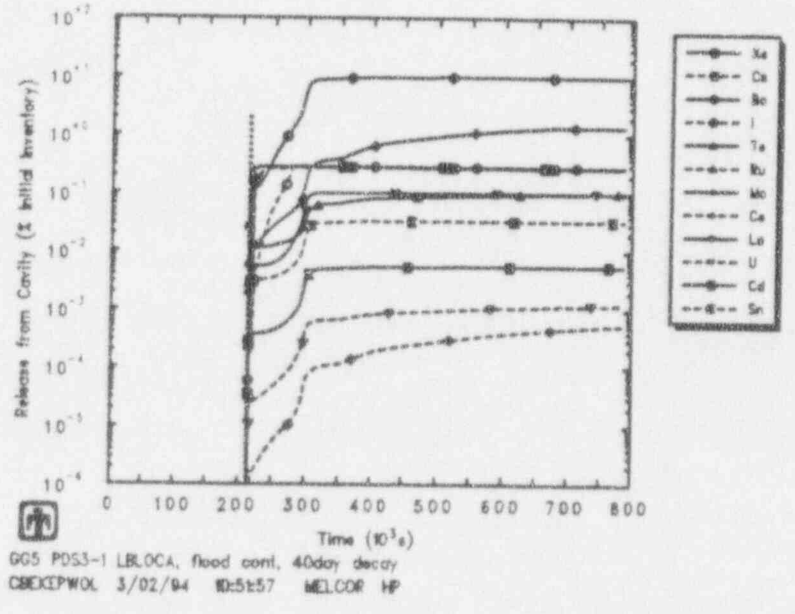


Figure 4.3.1.26. In-Vessel (top) and Ex-Vessel (bottom) Radionuclide Release Fractions for Grand Gulf POS 5 – Large Break LOCA with Flooded Containment, Initiated 40 day After Shutdown

Table 4.3.1.2. Final Radionuclide Release Fractions for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated at Various Times After Shutdown

Class	Fission Products Released from Fuel (% Initial Inventory)								
	7 hr			24 hr			40 day		
	In-Vessel	Ex-Vessel	Total	In-Vessel	Ex-Vessel	Total	In-Vessel	Ex-Vessel	Total
Xe	99.81	0.16	99.96	99.61	0.33	99.94	99.70	0.28	99.97
Cs	99.75	0.15	99.91	99.60	0.32	99.92	99.74	0.26	100.0
Ba	71.54	4.47	76.02	70.51	9.59	80.10	63.66	9.74	73.40
I	99.77	0.16	99.93	99.59	0.34	99.93	99.71	0.28	99.99
Te	99.77	0.03	99.81	99.58	0.10	99.68	99.52	0.10	99.63
Ru	0.89	3×10^{-6}	0.89	0.44	2×10^{-6}	0.44	0.10	3×10^{-7}	0.10
Mo	0	1.15	1.15	0	1.23	1.23	0	1.35	1.35
Ce	0.67	0.0007	0.67	0.25	0.0007	0.25	0.03	0.0005	0.03
La	0	0.28	0.28	0	0.39	0.39	0	0.10	0.10
U	23.11	0.0014	23.11	14.92	0.0019	14.92	4.50	0.0012	4.51
Cd	0	0.005	0.005	0	0.011	0.011	0	0.006	0.006
Sn	84.70	0.05	84.75	83.05	0.068	83.12	82.81	3.29	86.10

release model option used in these analyses has identically zero release in-vessel of Class 7 (Mo), Class 9 (La) and Class 11 (Cd). These are higher release fractions of Ba, Te, Ru, Ce, La and Sn than seen in MELCOR analyses of severe accidents at full power operation in LWR plants [19, 20, 32], reflecting the high debris temperatures calculated during in-vessel core degradation (shown in Figures 4.3.1.9 through 4.3.1.11).

Figure 4.3.1.27 gives the total radioactive release to the environment in these three cases. The releases are similar in magnitude for accidents begun at different times after shutdown, but shifted in time reflecting the slower accident progression at lower decay heat levels than at higher decay heat levels. These environmental releases do not correspond to immediate release of all radionuclides released from the fuel; there is considerable retention of most radionuclide species within the containment and auxiliary building (as discussed below). Only the noble gases and halogens (*i.e* , iodine) have substantial releases to the environment by the end of the transient periods simulated, because gaseous forms are not scrubbed, filtered, deposited or otherwise retained. There is a total of 484.63 kg of noble gases and halogens released from the fuel; the release to the environment is >90% of this by the end of the simulations begun at 7 hr and 24 hr after shutdown, and is about 75% of this when the calculation begun 40 days after scram was stopped. The temperatures are low enough in these shutdown sequences with flooded containment that the other volatile species released from the fuel (*i.e.*, Cs and Te) are found mostly in aerosol form and are retained in the primary system, containment and auxiliary building.

Tables 4.3.1.3 through 4.3.1.5 summarize the distribution of the initial radionuclide inventory at the end of the three calculations initiated at various times after shutdown; they provide an overview of how much of the radionuclides remain bound up in fuel debris in either the core or the cavity, and of how much of the released radionuclides are retained in the primary system *vs* how much of the released radionuclides are released to, or released in, either the containment or the auxiliary building and the environment, all normalized to the initial inventories of each class. Table 4.3.1.6 presents a slightly different breakdown of the released radionuclide final distribution, giving the fractions of released inventory for each class in control volume atmospheres (including the environment), in pools, or deposited or settled onto heat structures at the end of the calculations. (As in Table 4.3.1.2, these amounts consider only the release of radioactive forms of these classes, and not additional releases of nonradioactive aerosols from structural materials.)

These fission product distribution tables show that, of the radionuclides with significant (≥80% of initial inventory) release from fuel, most of the noble gases released are in the environment, in the atmosphere. While most of the volatile species (Cs and Te) releases occurred in-vessel, the largest part (about 90%) of those releases are retained in the containment, in water pools; most of the remaining volatiles release are retained in the auxiliary building, very small fractions of these volatiles are released to the environment for this large break LOCA scenario with flooded containment. (Only the low-pressure boiloff sequence discussed in Section 4.3.4, also with flooded containment, shows similarly high retention and small environmental releases of volatiles.) Two classes of radionuclides forming aerosols only had substantial releases (also occurring mostly in-vessel); for those

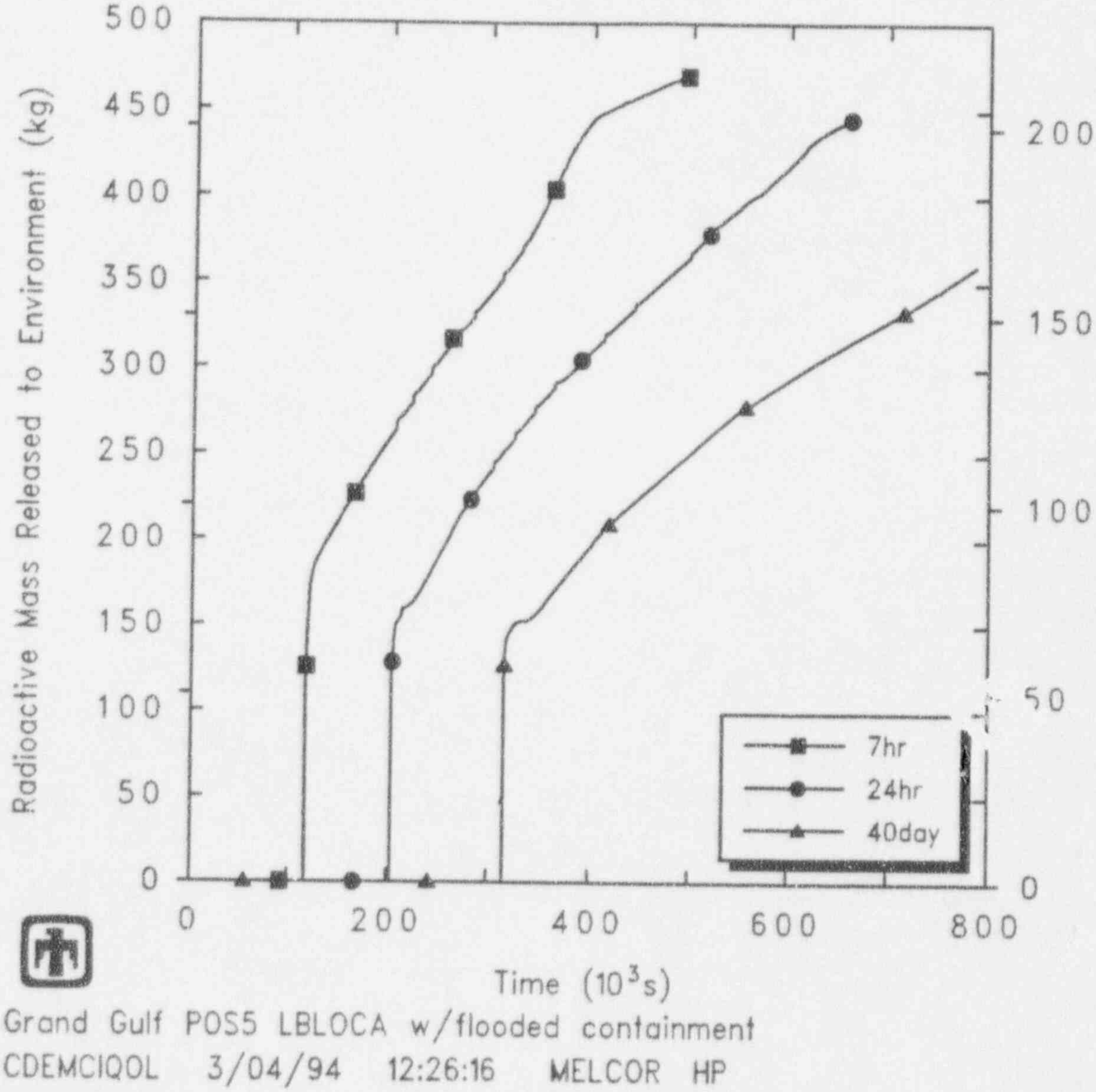


Figure 4.3.1.27. Total Environmental Radionuclide Releases for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated at Various Times After Shutdown

Table 4.3.1.3. Final Radionuclide Distribution for Grand Gulf POS 5 – Large Break LOCA with Flooded Containment, Initiated 7 hr After Shutdown

Class	Fission Product Distribution (% Initial Inventory)				
	Fuel Debris	Primary System	Containment	Auxiliary Building	Environment
Xe	~0	~0	0.426	3.413	96.1
CsOH	~0	0.658	88.76	10.56	0.0044
Ba	24.0	39.1	35.22	1.60	0.0042
Te	0.137	0.657	89.07	10.25	0.0026
Ru	99.1	0.375	0.496	0.021	0.0002
Mo	98.9	0.001	1.063	0.077	0.006
Ce	99.3	0.242	0.406	0.016	0.0002
La	99.7	0.0008	0.272	0.007	0.00007
U	78.7	10.2	10.6	0.47	0.0033
Cd	~100	0.00002	0.004	0.0004	0.0001
Sn	15.3	39.3	43.0	2.43	0.004

Table 4.3.1.4. Final Radionuclide Distribution for Grand Gulf POS 5 – Large Break LOCA with Flooded Containment, Initiated 24 hr After Shutdown

Class	Fission Product Distribution (% Initial Inventory)				
	Fuel Debris	Primary System	Containment	Auxiliary Building	Environment
Xe	~0	~0	1.80	6.46	91.7
CsOH	~0	1.17	85.36	13.20	0.14
Ba	19.9	45.4	31.68	3.06	0.0101
Te	0.292	1.07	85.31	13.18	0.19
Ru	99.6	0.237	0.191	0.016	0.00002
Mo	98.8	0.149	0.902	0.17	0.0103
Ce	99.8	0.119	0.117	0.010	0.000033
La	99.6	0.044	0.327	0.018	0.000086
U	86.3	7.79	5.59	0.383	0.00062
Cd	~100	0.0012	0.009	0.0007	0.000044
Sn	16.9	46.8	32.15	4.17	0.0015

Table 4.3.1.5. Final Radionuclide Distribution for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated 40 day After Shutdown

Class	Fission Product Distribution (% Initial Inventory)				
	Fuel Debris	Primary System	Containment	Auxiliary Building	Environment
Xe	~0	~0	9.29	16.41	74.3
CsOH	~0	0.881	92.76	6.4266	0.00065
Ba	26.6	38.6	34.06	0.75	0.0037
Te	0.358	0.745	92.90	6.03	0.0019
Ru	99.9	0.0584	0.0406	0.00073	0.0000035
Mo	98.7	0.0003	1.34	0.008	0.00033
Ce	~100	0.018	0.014	0.00024	0.0000013
La	99.9	0.000056	0.098	0.0005	0.000016
U	95.8	2.44	1.68	0.34	0.00014
Cd	~100	0.0000004	0.0054	0.00005	0.0000018
Sn	17.2	45.1	37.5	1.54	0.0006

Table 4.3.1.6. Final Radionuclide State for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, Initiated at Various Times After Shutdown

Class	Fission Products Released from Fuel (% Released Inventory)								
	7 hr			24 hr			40 day		
	Atmosphere	Pool	Deposited	Atmosphere	Pool	Deposited	Atmosphere	Pool	Deposited
Xe	~100	0	0	~100	0	0	~100	0	0
CsOH	0.004	84.6	15.3	0.15	74.65	25.22	0	79.48	20.52
Ba	0.006	40.9	59.1	0.013	29.24	70.74	0.005	39.03	60.97
I	~100	0	0	~100	0	0	~100	0	0
Te	0.003	84.0	16.0	0.005	74.67	25.75	0.086	78.73	21.27
Ru	0.02	49.14	50.86	0.005	31.69	68.33	0.00034	32.34	67.66
Mo	0.5	97.96	1.49	0.84	52.00	47.17	0.025	99.89	0.089
Ce	0.03	55.33	44.65	0.018	35.35	64.63	0.0001	34.78	65.26
La	0.03	97.3	2.64	0.025	56.13	43.87	0.0015	99.86	0.13
U	0.02	42.9	57.1	0.005	29.14	70.74	0.0034	31.77	68.10
Cd	3.60	93.98	2.42	1.10	55.28	43.65	0.033	99.75	0.21
Sn	0.005	45.48	54.52	0.002	31.90	68.09	0.00074	36.23	63.76

classes (Ba and Sn), about half the releases are retained in the vessel, primarily deposited on structures, while the other half of the releases are retained in the containment, mostly in water pools and a small fraction deposited on structure surfaces.

4.3.2 Station Blackout with Failure to Isolate SDC, Initiated 7 hr and 24 hr After Shutdown

At the initiation of the accident, the reactor vessel is depressurized and the coolant is at the normal level. The vessel water inventory is at 366.5 K (200°F), which corresponds to the maximum temperature allowed by the Grand Gulf technical specifications for operation in POS 5. The reactor pressure vessel head vent is open. At the start of the accident all core cooling and injection is lost and the SRVs are closed. Before the SRVs can cycle at their pressure relief setpoint, the break in the SDC line is opened when the vessel pressure reaches 3.135 MPa (440 psig). The SDC break runs from the vessel downcomer, 4.38 m above the bottom of the vessel to the first floor of the auxiliary building, 8.18 m below the bottom of the vessel. The suppression pool level is 3.86 m (12.67 ft) from the suppression pool floor. The containment is at 305.4 K (90°F) and the suppression pool is at 308.2 K (95°F). The drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open.

This sequence is identical to the Level 1 analysis of a station blackout sequence with failure to isolate SDC discussed in Section 4.2.6, initiated at 7 hr and 24 hr after shutdown.

The sequence of events predicted by MELCOR for this accident with different initiation times is given in Table 4.3.2.1.

Figure 4.3.2.1 gives the vessel pressures calculated for this same accident scenario initiated at two different times after scram. In both cases, the primary system pressure rises to the SDC failure pressure at 3.135 MPa (440 psig), which actuates the postulated SDC break. The flow out the SDC line break goes directly to the auxiliary building first floor and pressurizes the auxiliary building, as indicated in Figure 4.3.2.2. As expected, the lower the decay heat the slower the auxiliary building pressurizes and the longer it takes to fail the auxiliary building. The open personnel locks and equipment hatch keep the containment equilibrated to the auxiliary building in this sequence.

The coolant inventory in the vessel drops as the decay heat boils water to steam which is lost out the SDC break and the open RPV vent, faster for higher decay heat levels, as presented in Figure 4.3.2.3. The opening of the SDC break is reflected in the extremely rapid loss of about 75% of the vessel inventory seen at various times; that inventory loss then slows down when the break uncovers, and is followed by a more gradual loss of the remaining inventory due to boiling and steam outflow until vessel failure. The amount of liquid inventory lost in the initial liquid blowdown is determined by the elevation of the break and is therefore about the same regardless of the decay heat level; later, as would be expected, the gradual inventory loss due to continued boiloff is faster for higher decay heat levels than for lower decay heat levels. The vessel inventory then drops to zero very quickly upon vessel failure.

Table 4.3.2.1. Sequence of Events Predicted by MELCOR for Station Blackout with Failure to Isolate SDC, Initiated 7 hr and 24 hr After Shutdown

Event	Time After Shutdown 7 hr	24 hr
Accident initiation	0.0	0.0
Core uncovery (TAF) begins	13,375 s (3.72 hr)	19.717 s (5.48 hr)
Core heatup begins	13,500 s (3.75 hr)	20.000 s (5.56 hr)
SDC break at 440 psig	13,750 s (3.82 hr)	20.250 s (5.63 hr)
Auxiliary building failed	13,750 s (3.82 hr)	20.250 s (5.63 hr)
Clad failure/Gap release		
(Ring 1)	15,714 s (4.36 hr)	22.876 s (6.35 hr)
(Ring 2)	15,670 s (4.35 hr)	22.817 s (6.34 hr)
(Ring 3)	15,708 s (4.36 hr)	22.869 s (6.35 hr)
(Ring 4)	15,941 s (4.43 hr)	23.180 s (6.44 hr)
(Ring 5)	16,959 s (4.71 hr)	24.520 s (6.81 hr)
(Ring 6)	19,279 s (5.36 hr)	27.389 s (7.61 hr)
Core plate failed		
(Ring 1)	55,519 s (15.42 hr)	56.345 s (15.65 hr)
(Ring 2)	55,477 s (15.41 hr)	44.848 s (12.46 hr)
(Ring 3)	55,399 s (15.39 hr)	55.630 s (15.45 hr)
(Ring 4)	56,138 s (15.59 hr)	55.875 s (15.52 hr)
(Ring 5)	54,003 s (15.00 hr)	58,377 s (16.22 hr)
(Ring 6)	52,994 s (14.72 hr)	59,495 s (16.53 hr)
Vessel LH penetration failed		
(Ring 1)	53,123 s (14.76 hr)	44.930 s (12.48 hr)
(Ring 2)	53,105 s (14.75 hr)	44,941 s (12.48 hr)
(Ring 3)	53,079 s (14.74 hr)	44,931 s (12.48 hr)
(Ring 4)	53,074 s (14.74 hr)	44.934 s (12.48 hr)
(Ring 5)	53,074 s (14.74 hr)	44,938 s (12.48 hr)
(Ring 6)	53,139 s (14.76 hr)	44,939 s (12.48 hr)
Commence debris ejection	53,074 s (14.74 hr)	44,930 s (12.48 hr)
Cavity rupture	218,431 s (60.68 hr)	
End of calculation	218,431 s (60.68 hr)	200,000 s (55.56 hr)

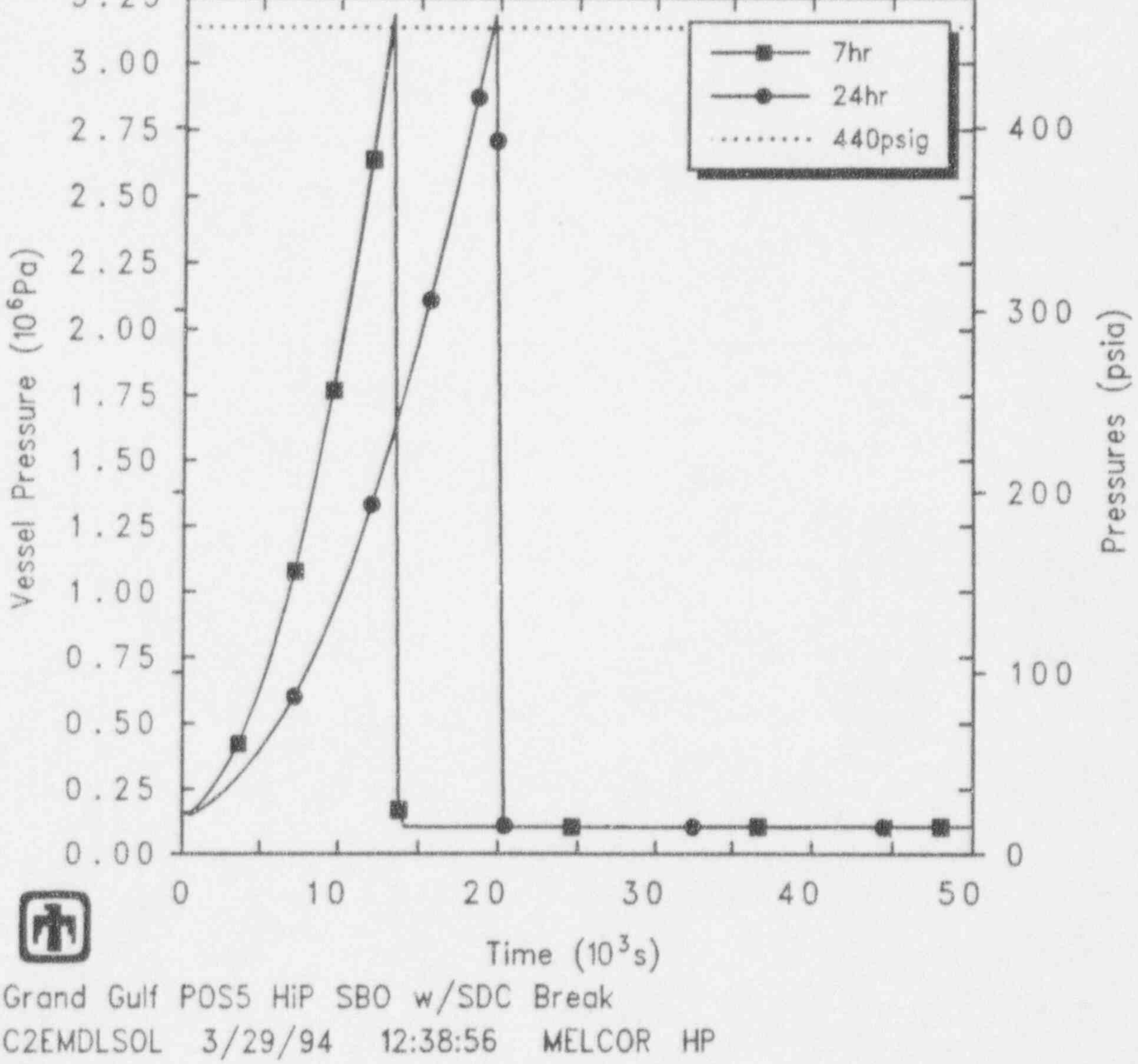


Figure 4.3.2.1. Reactor Vessel Pressures for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated at Various Times After Shutdown

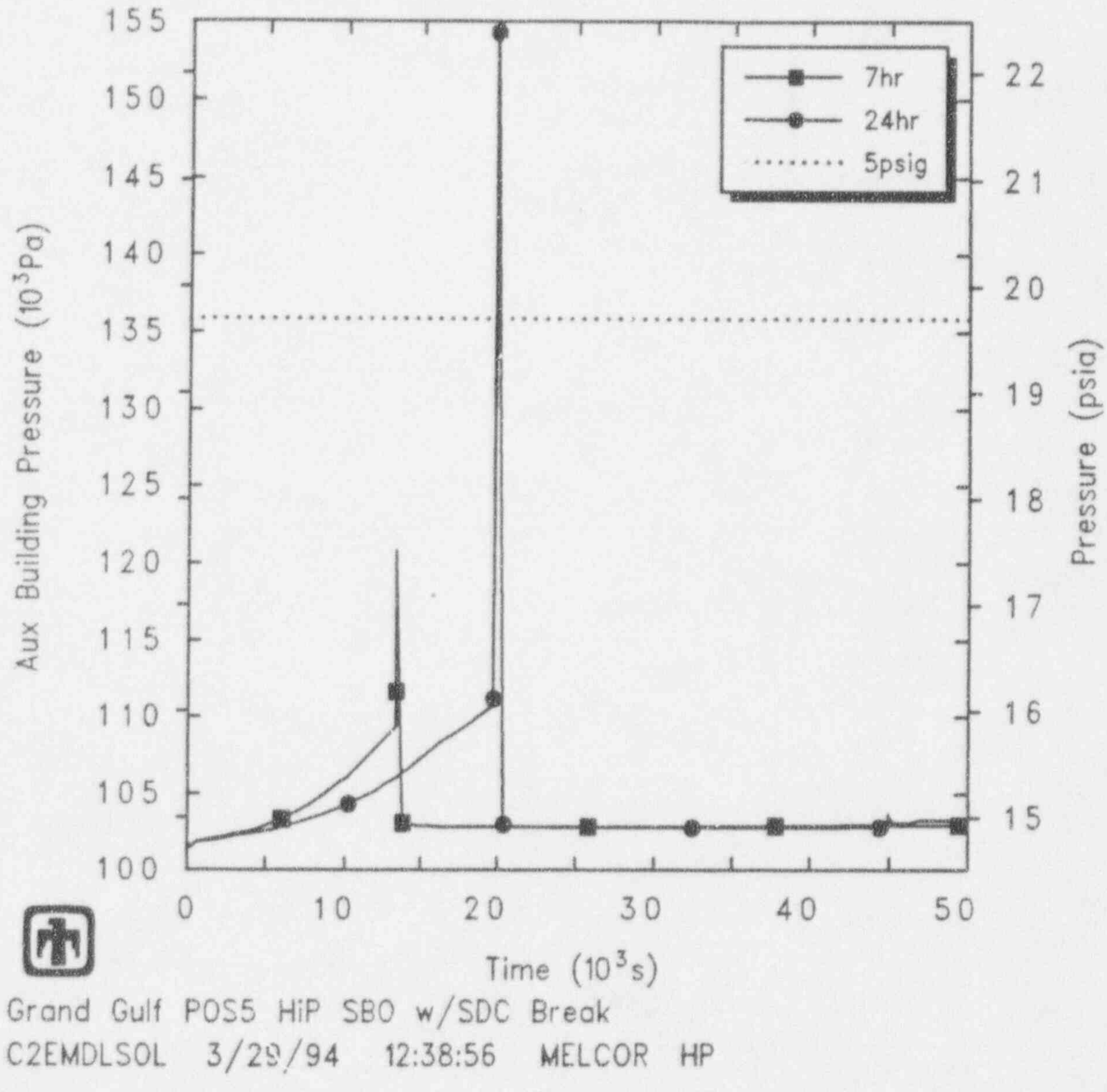


Figure 4.3.2.2. Auxiliary Building Pressures for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated at Various Times After Shutdown

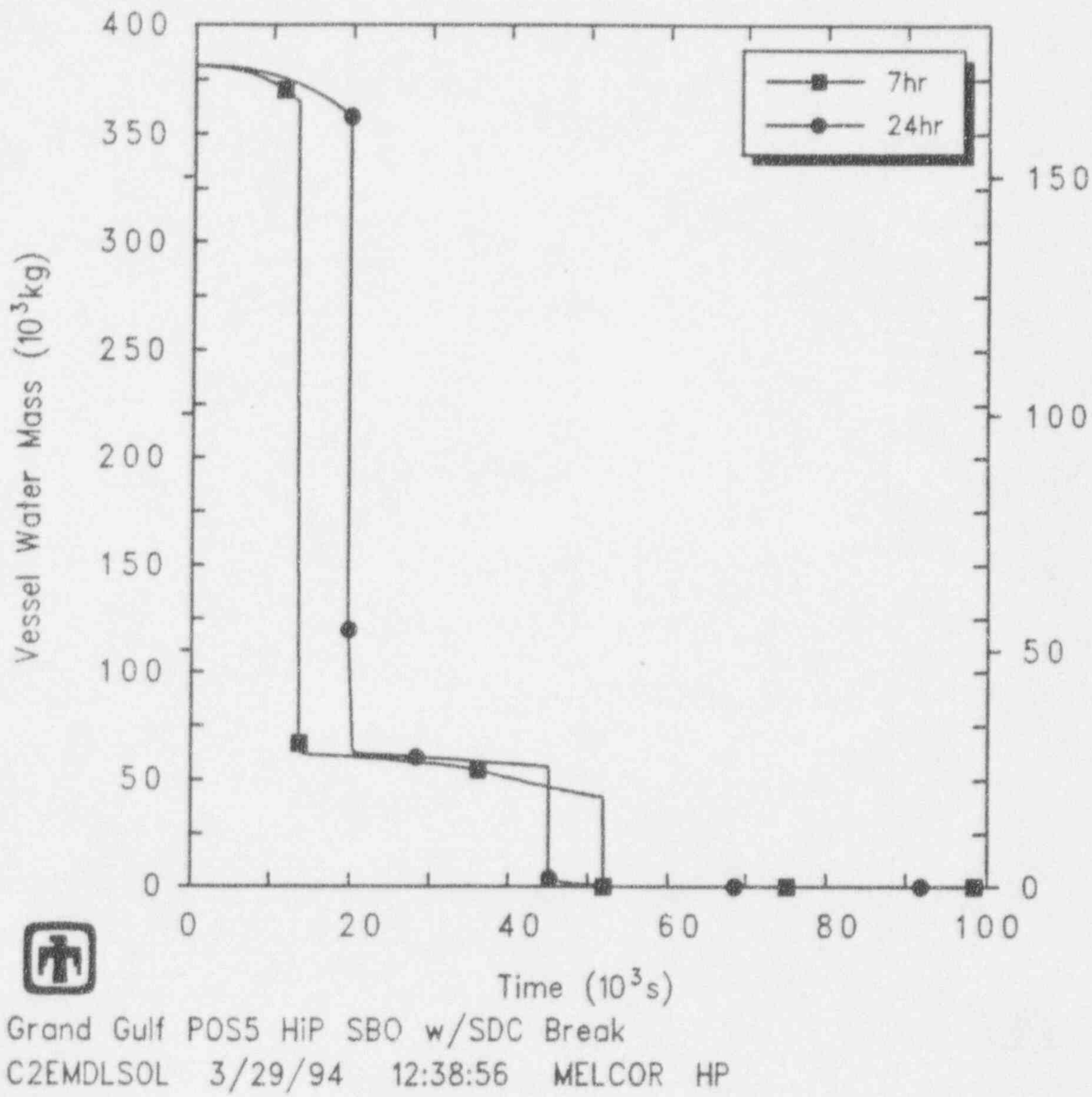


Figure 4.3.2.3. Reactor Vessel Water Masses for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated at Various Times After Shutdown

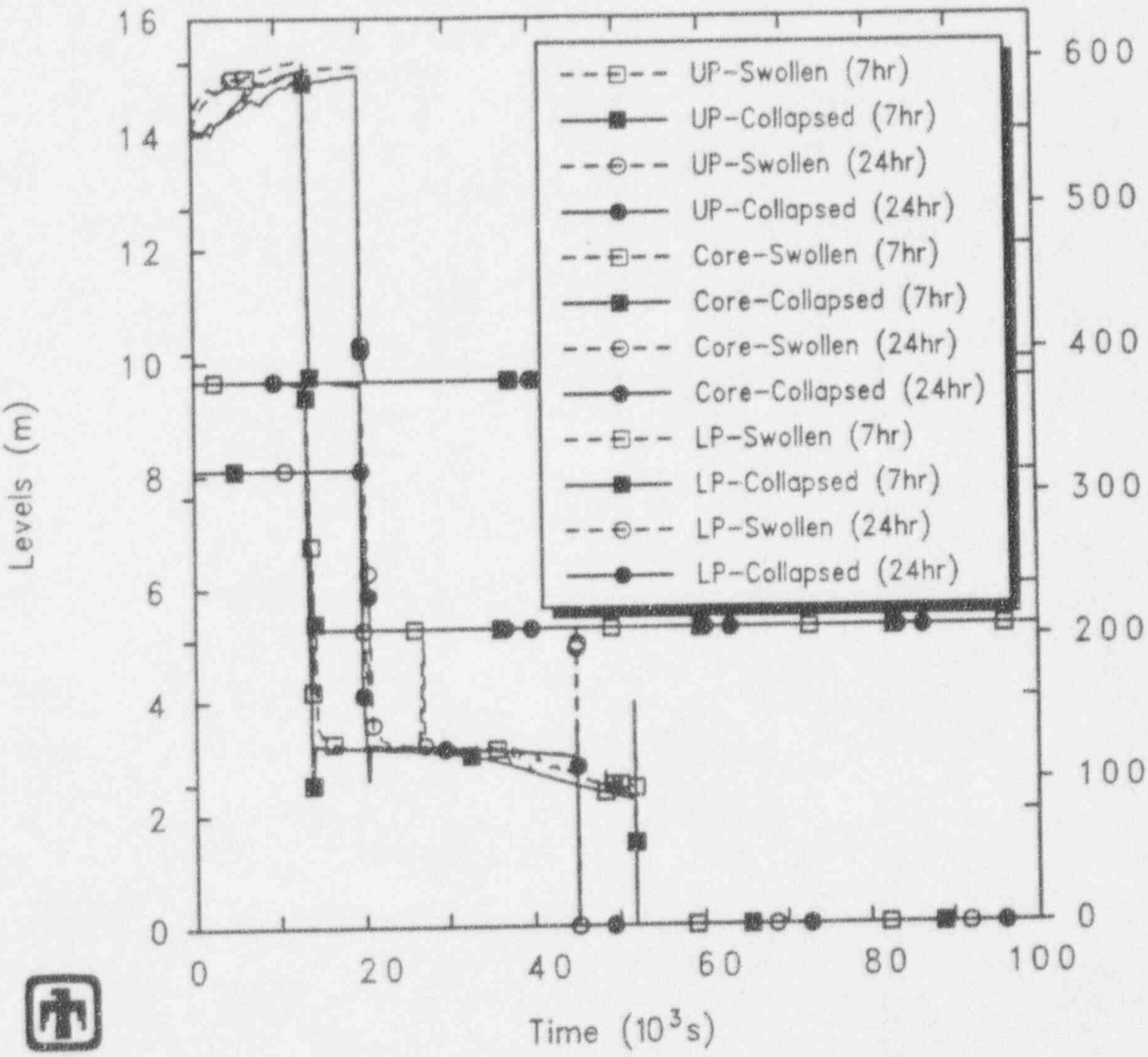
Figure 4.3.2.4 presents the upper plenum, core and lower plenum swollen and collapsed liquid levels for this accident sequence initiated at two different times after scram. The upper plenum level initially rises as the primary system pressurizes and then falls rapidly when the SDC break is opened. The vessel liquid level drops smoothly through the upper plenum into the core and continue dropping smoothly partway into the lower plenum, followed by a more gradual loss of the remaining inventory due to boiling and steam outflow. The amount of liquid inventory lost in the blowdown out the SDC break is determined by the elevation of the break and is therefore about the same regardless of the decay heat level; later, as would be expected, the gradual core uncovery due to continued boiloff is faster for higher decay heat levels than for lower decay heat levels. There is very little pool frothing or swelling in any of the vessel volumes.

The heatup of the intact fuel and clad is illustrated in Figures 4.3.2.5 and 4.3.2.6, as calculated for scenarios initiated at 7 hr and 24 hr after shutdown, respectively. Core uncovery and heatup begins sooner and proceeds more rapidly at the higher decay heat level resulting from beginning this accident 7 hr after scram than for a lower decay heat in the same accident initiated 24 hr after scram. The fuel/clad component temperatures in MELCOR are set to zero in a cell when that component fails, so these figures show both the overall heatup rate and the time that the intact fuel/clad component fails through melting of the clad.

Figures 4.3.2.7 and 4.3.2.8 present corresponding core debris temperatures in the active fuel region calculated for scenarios initiated at 7 hr and 24 hr after shutdown, respectively; these are the temperatures of the debris bed formed by the failure of the intact fuel/clad component in MELCOR in a core cell, whose (intact) temperatures were given in Figures 4.3.2.5 and 4.3.2.6. The intact fuel/clad component temperatures reach a peak of ≥2000 K (≥3140°F) since the component generally fails at the zircaloy clad melt temperature, taken as 2098 K (3317°F) in MELCOR. The debris bed in the active fuel region in contrast reaches peak temperatures ≥4250 K (7190°F), significantly above the UO_2 melt temperature of 3113 K (5144°F), except in the lowermost active fuel level where the debris bed temperature remains near the UO_2 melt temperature. The debris bed temperatures reached in the active fuel region are slightly higher for the accident initiated at a higher decay heat level than at the lower decay heat level, as would be expected. (Notice that the debris bed temperatures predicted in these station blackout sequences with failure to isolate SDC are substantially higher than those predicted in the large break LOCA analyses presented in the previous section.)

The temperatures of the active fuel region debris bed drop to zero when the core plate fails and the debris relocates to the lower plenum. This occurs much later than the collapse of the intact fuel and clad into a debris bed. An unexpected result in these station blackout sequences with failure to isolate SDC is the failure of the core plate (and subsequently the vessel) earlier in the case initiated 24 hr after shutdown than in the case initiated 7 hr after shutdown.

Figures 4.3.2.9 and 4.3.2.10 depict the structure temperatures for the core support plate ("level 5") and for the lower core support structure in the level just above the core support plate and below the first active fuel level ("level 6", with active fuel beginning



Grand Gulf POS5 HiP SBO w/SDC Break
C2EMDLSOL 3/29/94 12:38:56 MELCOR HP

Figure 4.3.2.4. Upper Plenum, Core and Lower Plenum Liquid Levels for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated at Various Times After Shutdown

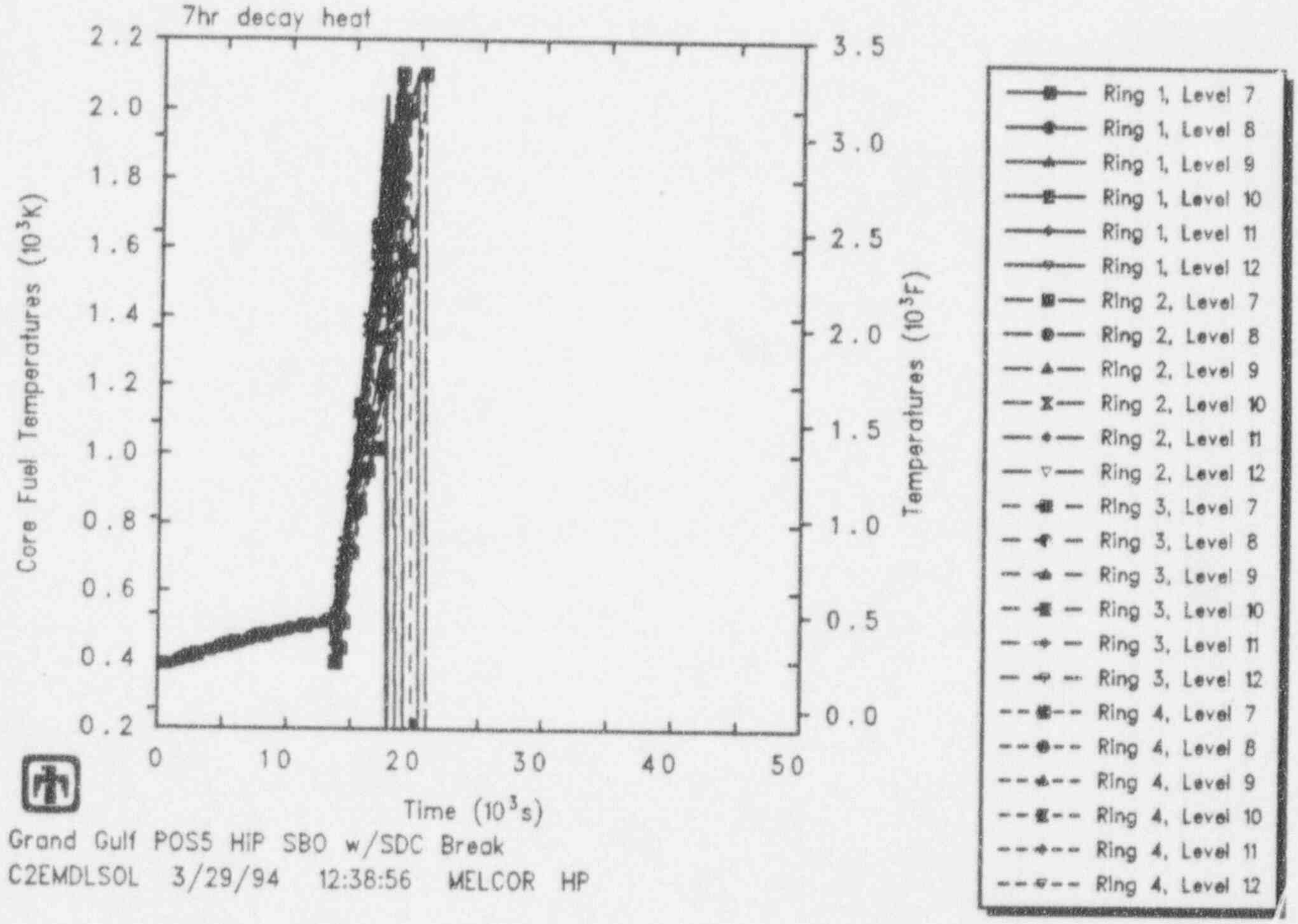


Figure 4.3.2.5. Core Intact Fuel/Clad Temperatures for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated 7 hr After Shutdown

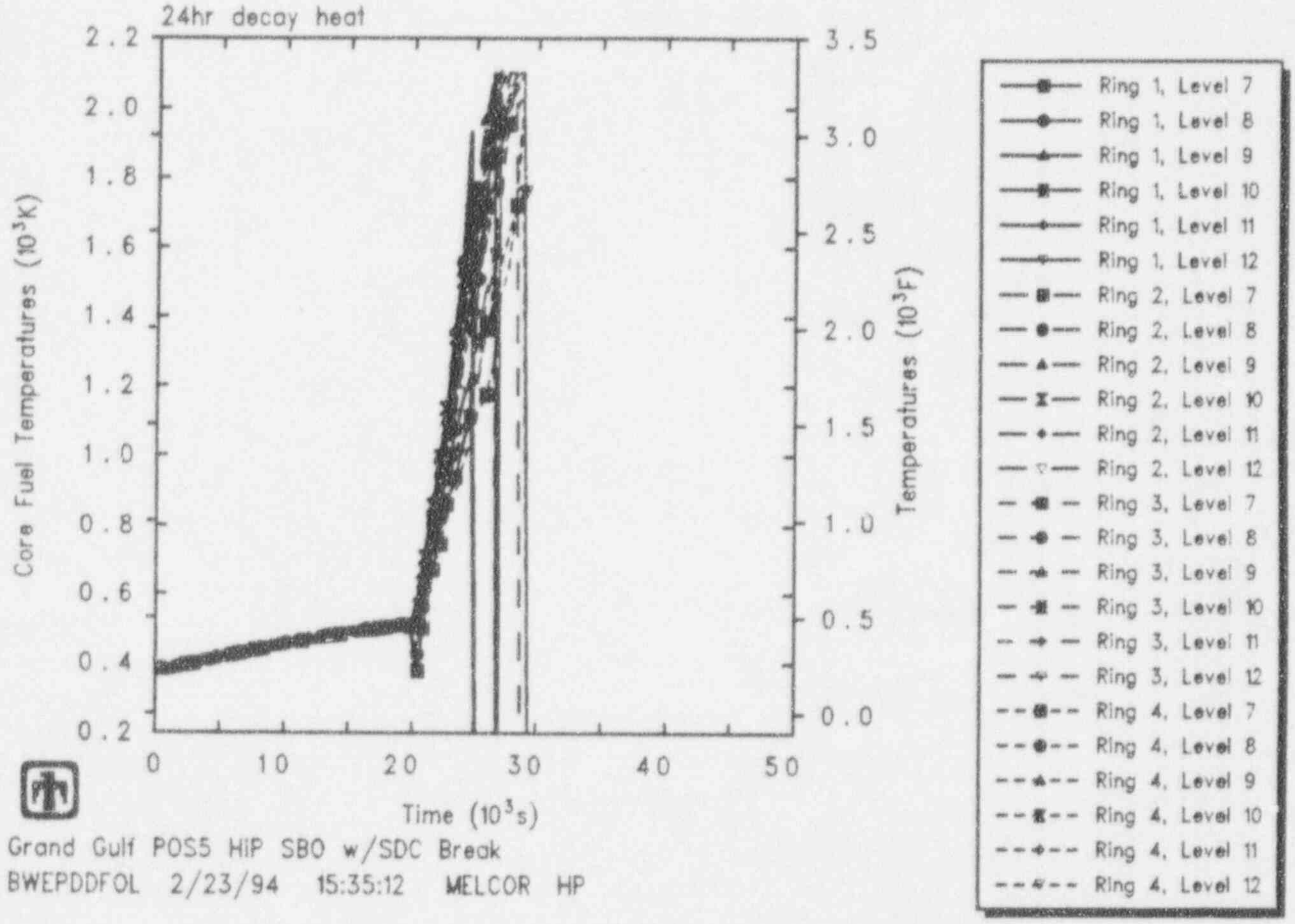


Figure 4.3.2.6. Core Intact Fuel/Clad Temperatures for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated 24 hr After Shutdown

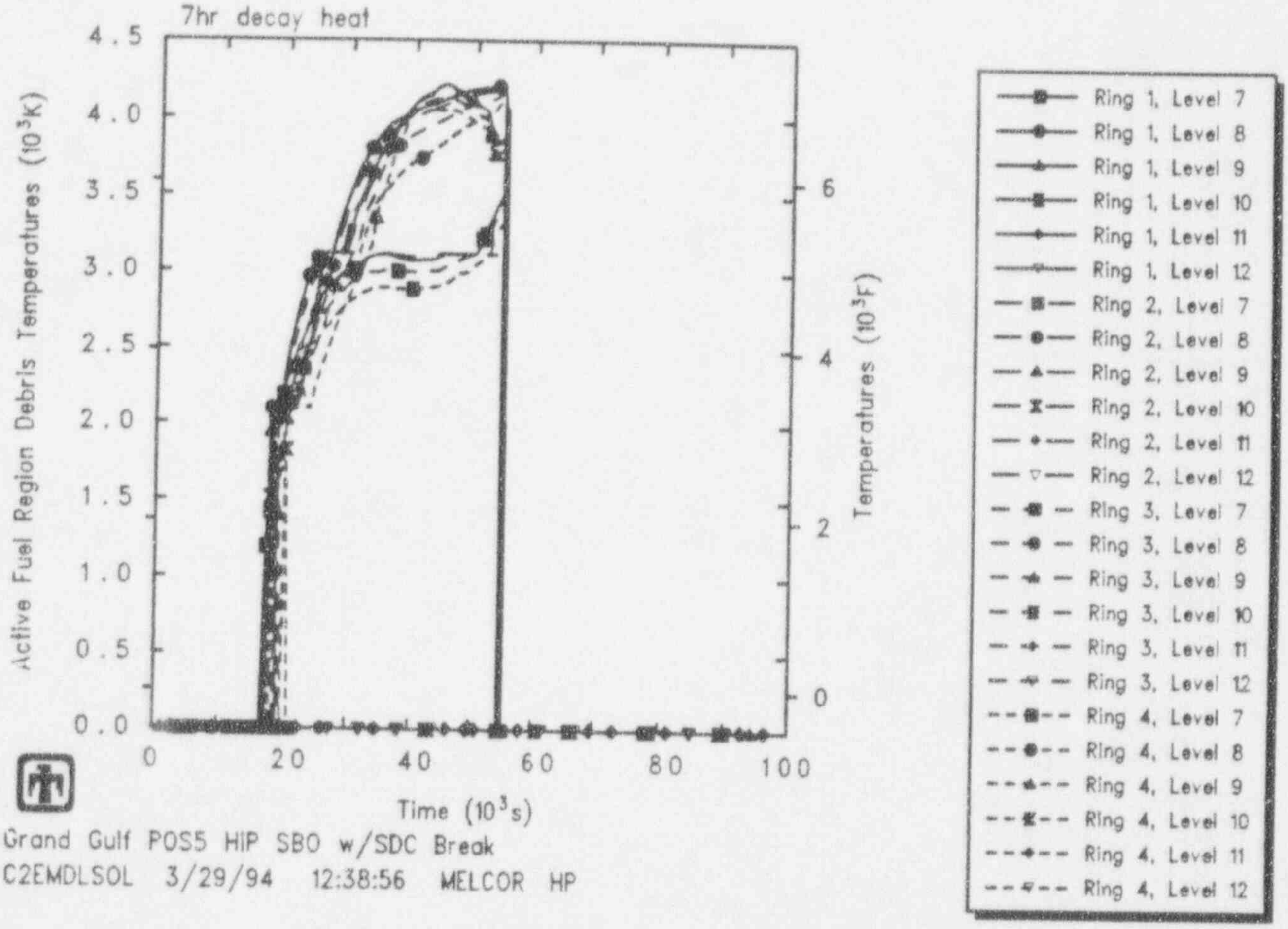


Figure 4.3.2.7. Core Active Fuel Region Debris Bed Temperatures for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated 7 hr After Shutdown

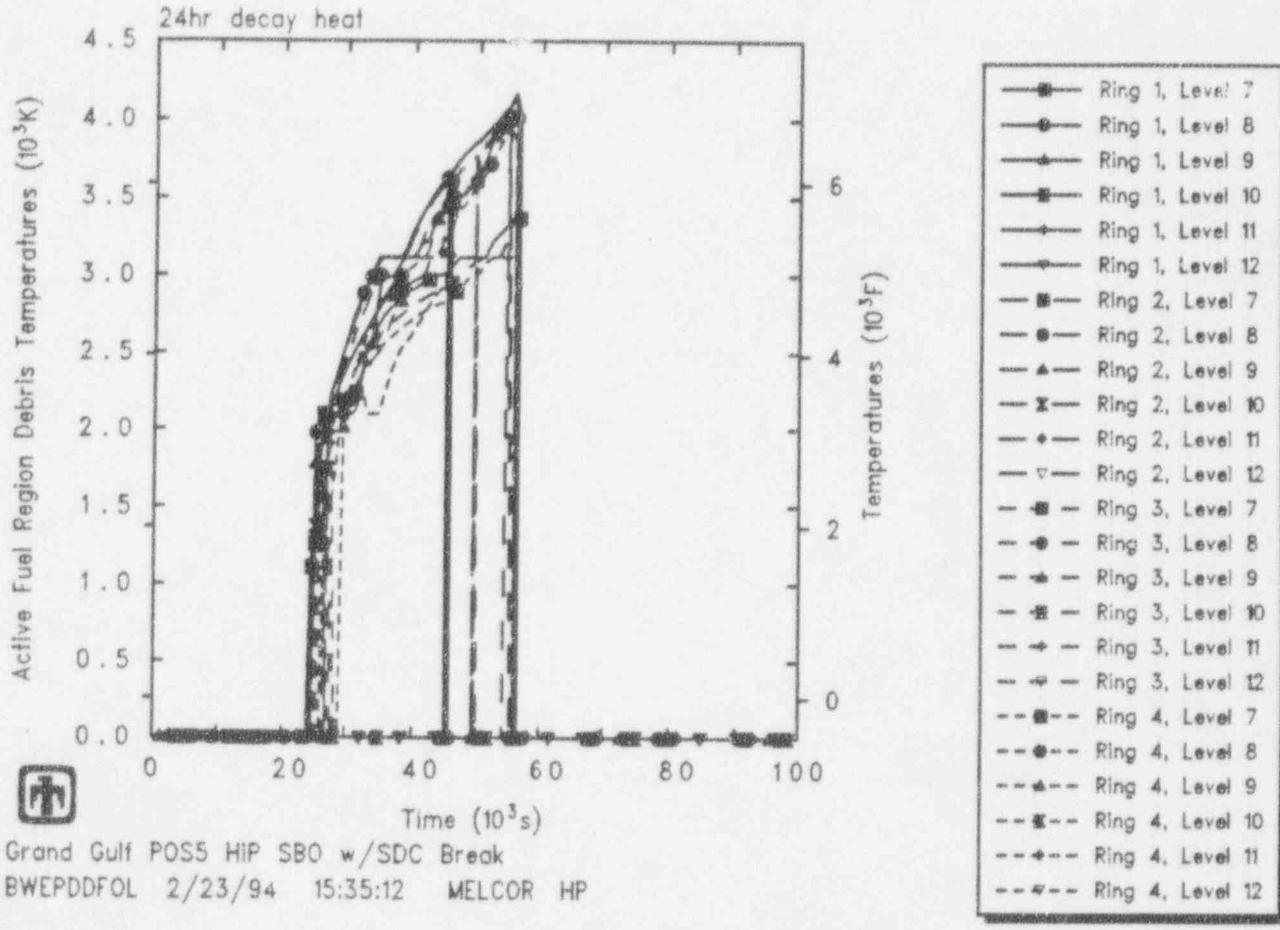


Figure 4.3.2.8. Core Active Fuel Region Debris Bed Temperatures for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated 24 hr After Shutdown

in "level 7"). The core support plate is assumed to fail at 1273 K (1832°F), a criterion also shown in these figures. The support structure above the core plate reaches this temperature at about the time the debris bed forms in the active fuel region, but the temperature of the support structure above the core plate then remains nearly constant and increases only gradually as the temperature of the debris bed in the active fuel region reaches values of 3100-4200 K; this growing temperature gradient is probably due to the neglect of axial conduction in the particulate debris component in the MELCOR COR package. The core support plate itself remains substantially cooler than the support structure above the core plate, increasing only slowly. In the calculation initiated 7 hr after shutdown, the core support plate temperatures in all radial rings remain nearly equal as the core plate is heated, while in the calculation initiated 24 hr after shutdown, the lower core support structure and the core support plate temperatures in the second ring increase much more quickly than for the other three rings. On physical grounds, given most of the active fuel material forming a relatively uniform debris bed, the core plate temperatures in the various radial rings should remain nearly equal; if this had happened in the calculation initiated at 24 hr after scram, Figure 4.3.2.10 indicates that the core plate should have failed at ~56,000 s, later than in the calculation initiated 7 hr after shutdown.

The predicted temperatures in the debris bed in the lower plenum and core plate are given in Figures 4.3.2.11 and 4.3.2.12, for scenarios initiated at 7 hr and 24 hr after shutdown, respectively. In both cases, prior to core plate failure there is some cold, refrozen debris both on the core support plate and on the lower core structural material just above the core support plate; the cooling and refreezing of this debris is the cause of the continued gradual drop in lower plenum liquid level due to steaming seen in Figure 4.3.2.4. The debris temperature rises gradually to the core support plate failure temperature of 1273 K (1832°F). After core plate failure hot, high-temperature debris begins appearing the the lower plenum as debris falls from the active fuel region into the lower plenum. With the new debris radial relocation model added in MELCOR 1.8.2, the core plate needs to fail in only one ring before debris from cells in the active fuel region in all radial rings can potentially flow sideways and down, fall through the failed plate, and then spread sideways into cells in the lower plenum in all radial rings. (Thus a lower head penetration can now fail in a ring before the core plate in that ring fails.) The lower head penetrations begin failing almost immediately, and the lower plenum debris temperatures begin dropping to zero as debris is ejected from the vessel to the cavity. (Notice that the calculation initiated 24 hr after shutdown shows some quenched debris fallen into the lower plenum in the second ring prior to core plate failure, not seen in the other rings or in any ring in the calculation initiated 7 hr after shutdown; this is probably related to the anomalous core plate heatup and failure behavior discussed above.)

Figures 4.3.2.13 and 4.3.2.14 indicate what fraction of each material in the active fuel region has collapsed into a debris rubble bed held up by the core support plate, prior to core plate failure, debris relocation, lower head failure and debris ejection, for this station blackout scenario with failure to isolate SDC initiated at 7 hr and 24 hr after shutdown, respectively. The fractions of each material and the overall fraction of total

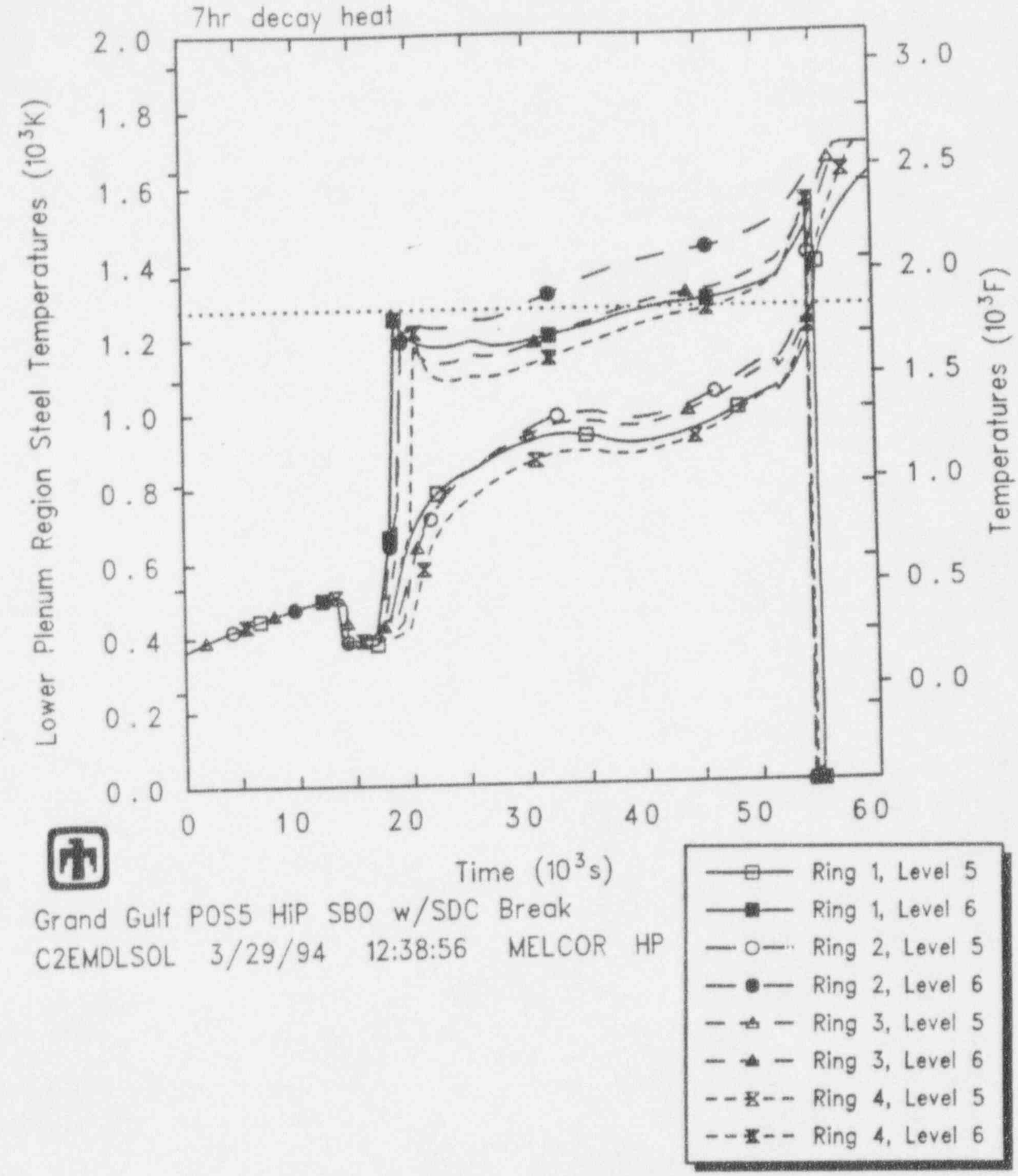


Figure 4.3.2.9. Core Support Plate and Lower Core Support Structure Temperatures for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated 7 hr After Shutdown

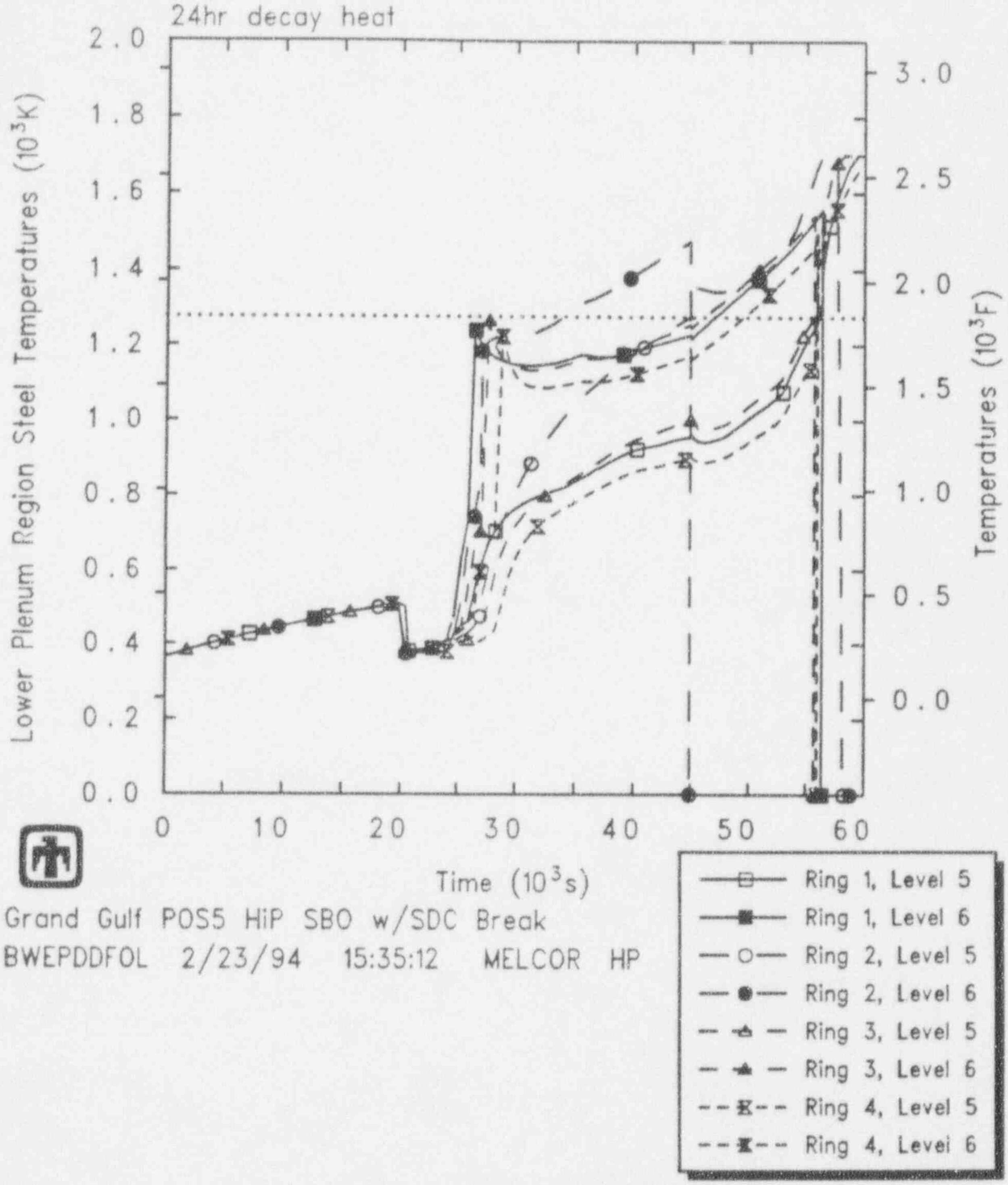


Figure 4.3.2.10. Core Support Plate and Lower Core Support Structure Temperatures for Grand Gulf POS 5 – Station Blackout with Failure to Isolate SDC, Initiated 24 hr After Shutdown

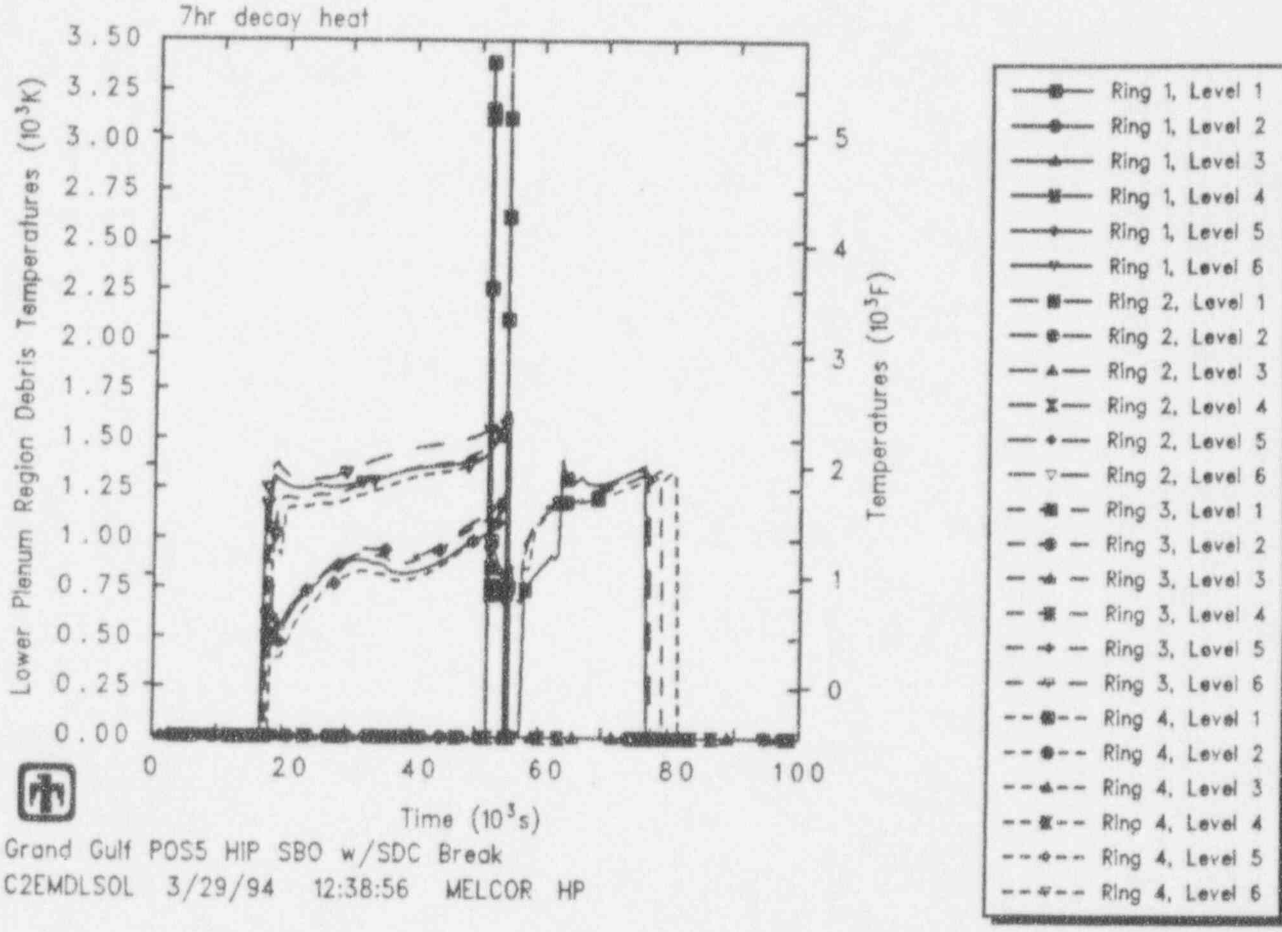


Figure 4.3.2.11. Core Lower Plenum and Core Support Plate Debris Bed Temperatures for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated 7 hr After Shutdown

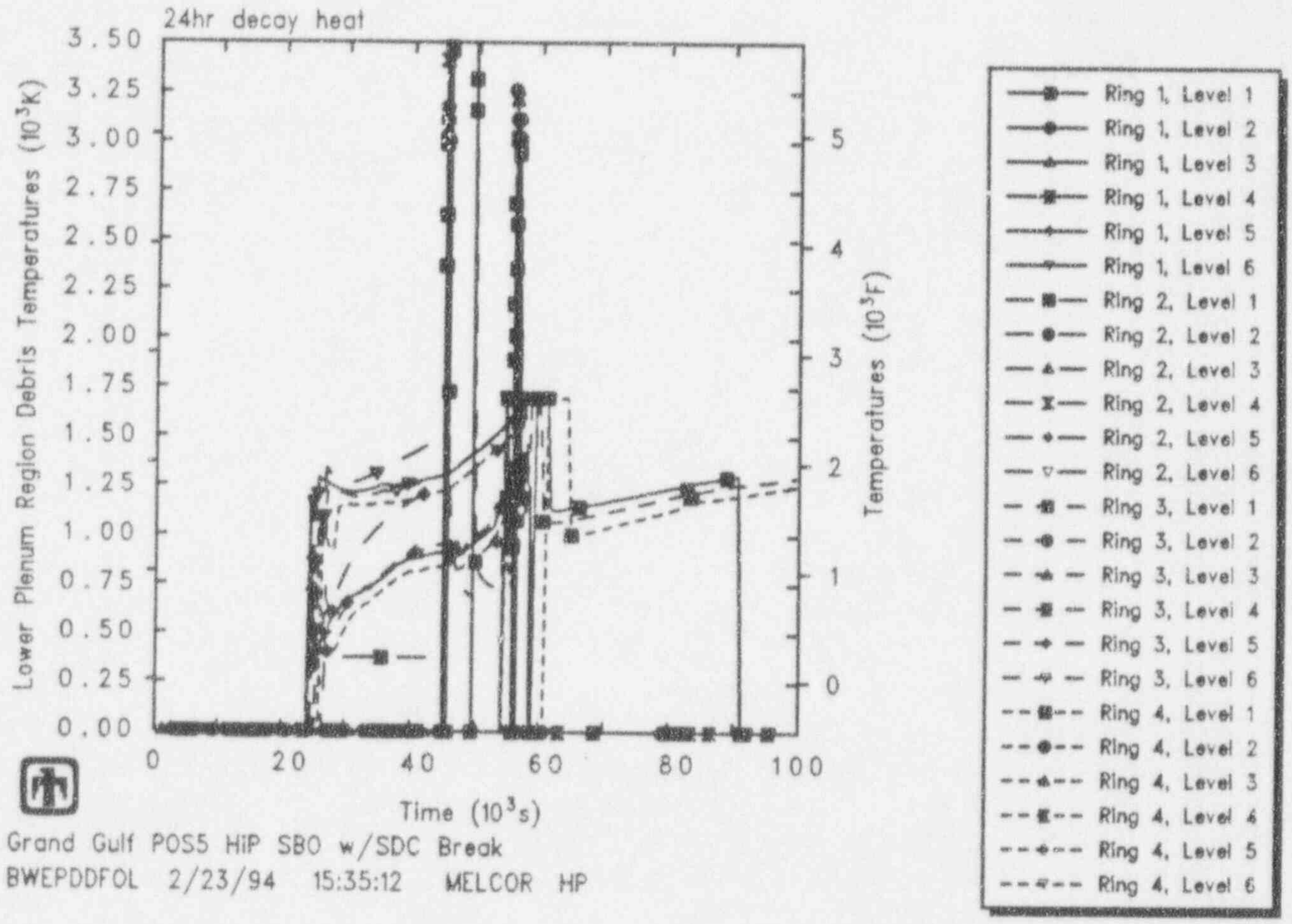


Figure 4.3.2.12. Core Lower Plenum and Core Support Plate Debris Bed Temperatures for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated 24 hr After Shutdown

material in the active fuel region degraded into particulate debris and are similar in the two calculations. The majority of the debris bed is formed within about 8,000 s at the higher decay heat level and within about 9.000 s at the lower decay heat level.

Figure 4.3.2.15 shows the total masses of core materials (UO_2, Zircaloy and ZrO_2, stainless steel and steel oxide, and control rod poison) remaining in the vessel. This includes both material in the active fuel region and in the lower plenum. Debris ejection began very soon after lower head failure. This figure illustrates that most of the core material was lost from the vessel to the cavity quickly, in step-like stages. In all cases, all of the UO_2 was transferred to the cavity within ~1 hr after vessel failure, as was the unoxidized zircaloy, the associated zirc oxide and the control rod poison. A small fraction (1-5%) of the structural steel in the lower plenum, and some associated steel oxide, was predicted to remain unmelted and in place.

The debris material lost from the vessel is ejected to the drywell pedestal cavity. Since almost all the material in the core active fuel region and lower plenum is lost within a very short time period after vessel failure, the core debris mass in the cavity is about the same for these two calculations initiated at different times after scram. Figure 4.3.2.16 indicates that the amount of concrete ablated and the total cavity debris mass (*i.e.*, core debris combined with concrete ablation products) are also very similar for this sequence initiated at different times after scram. In both cases, concrete ablation is quite rapid soon after debris ejection (while the core debris is hot, >2000 K, and consists of a layer of metallic debris above a heavy oxide layer), and concrete ablation slows significantly after a short time (after enough concrete has been ablated for the debris bed configuration to invert to a light oxide layer above a layer of metallic debris, mixed to a lower average temperature of ~1500 K).

The calculated production of steam and noncondensable gases (H_2, CO, CO_2 and H_2O) is summarized in Figure 4.3.2.17. The hydrogen production shown includes both in-vessel production (the initial step increase) and ex-vessel production in the cavity (the later-time increase). The in-vessel hydrogen generation corresponds to the oxidation of about 10-20% of the zircaloy and about 1% of the steel in the core and lower plenum, prior to vessel failure and debris ejection. As soon as the core debris enters the cavity, core-concrete interaction begins, resulting in the production of carbon dioxide and hydrogen; reduction of these gases by the molten metal in the core debris also gives rise to carbon monoxide and hydrogen. The production rate of noncondensables from core-concrete interaction resembles the concrete ablation rate: quite rapid soon after debris ejection, later slowing after a CORCON "layer flip" has occurred. On a molar basis, similar amounts are produced of all these gases.

This generation of noncondensables changes the composition of the atmosphere in the containment and in the auxiliary building. The mole fractions in the drywell, containment dome and auxiliary building (first and second floors) are presented in Figures 4.3.2.18 and 4.3.2.19 for this sequence initiated at two different times after shutdown, including vertical dotted lines at auxiliary building failure and at vessel failure for reference. The mole fractions in the cavity resemble the behavior shown for the drywell; the mole fractions in the containment equipment hatch are very similar to those shown for the

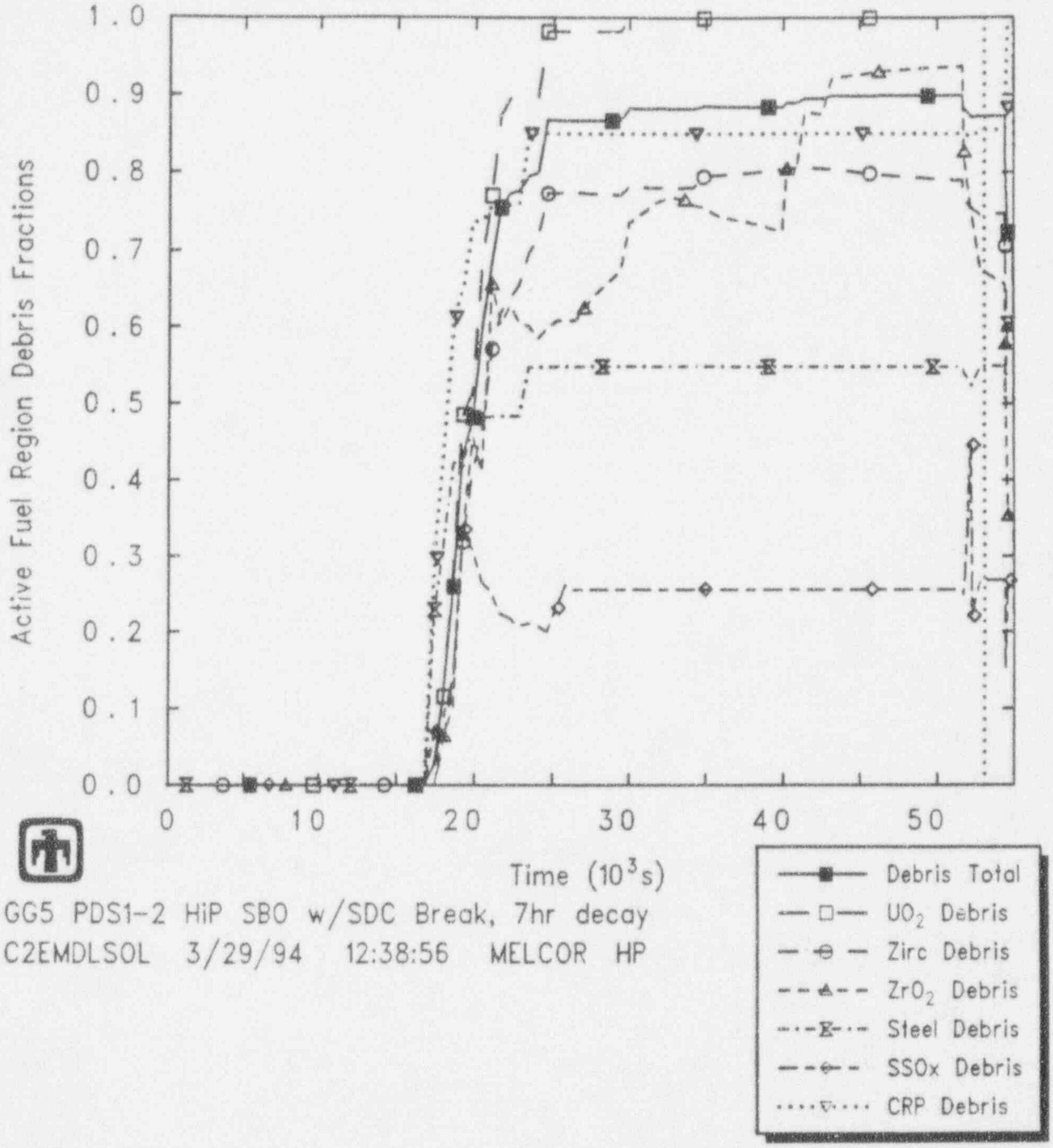


Figure 4.3.2.13. Core Active Fuel Region Degraded Material Fractions for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated 7 hr After Shutdown

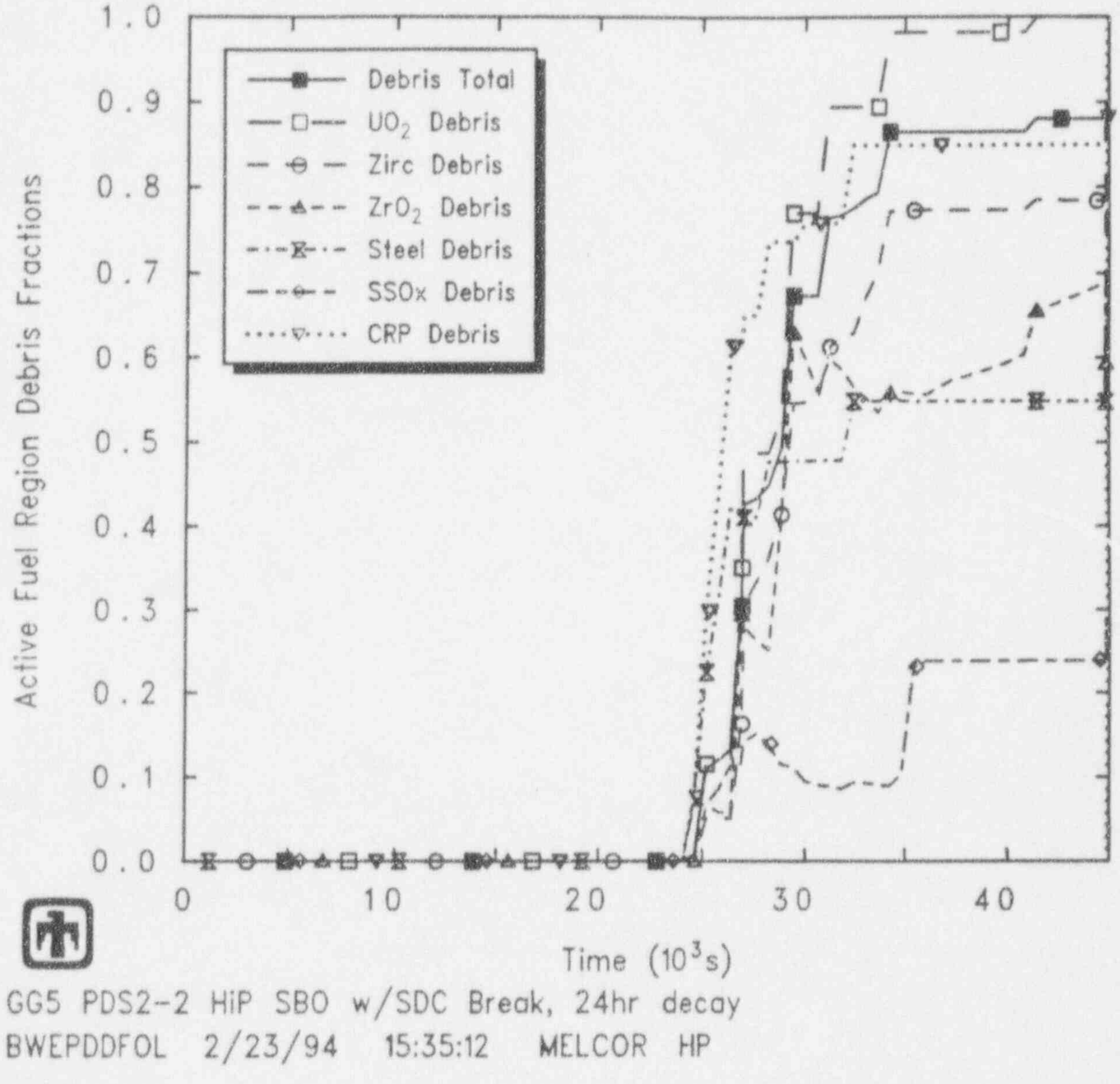


Figure 4.3.2.14. Core Active Fuel Region Degraded Material Fractions for Grand Gulf POS 5 – Station Blackout with Failure to Isolate SDC, Initiated 24 hr After Shutdown

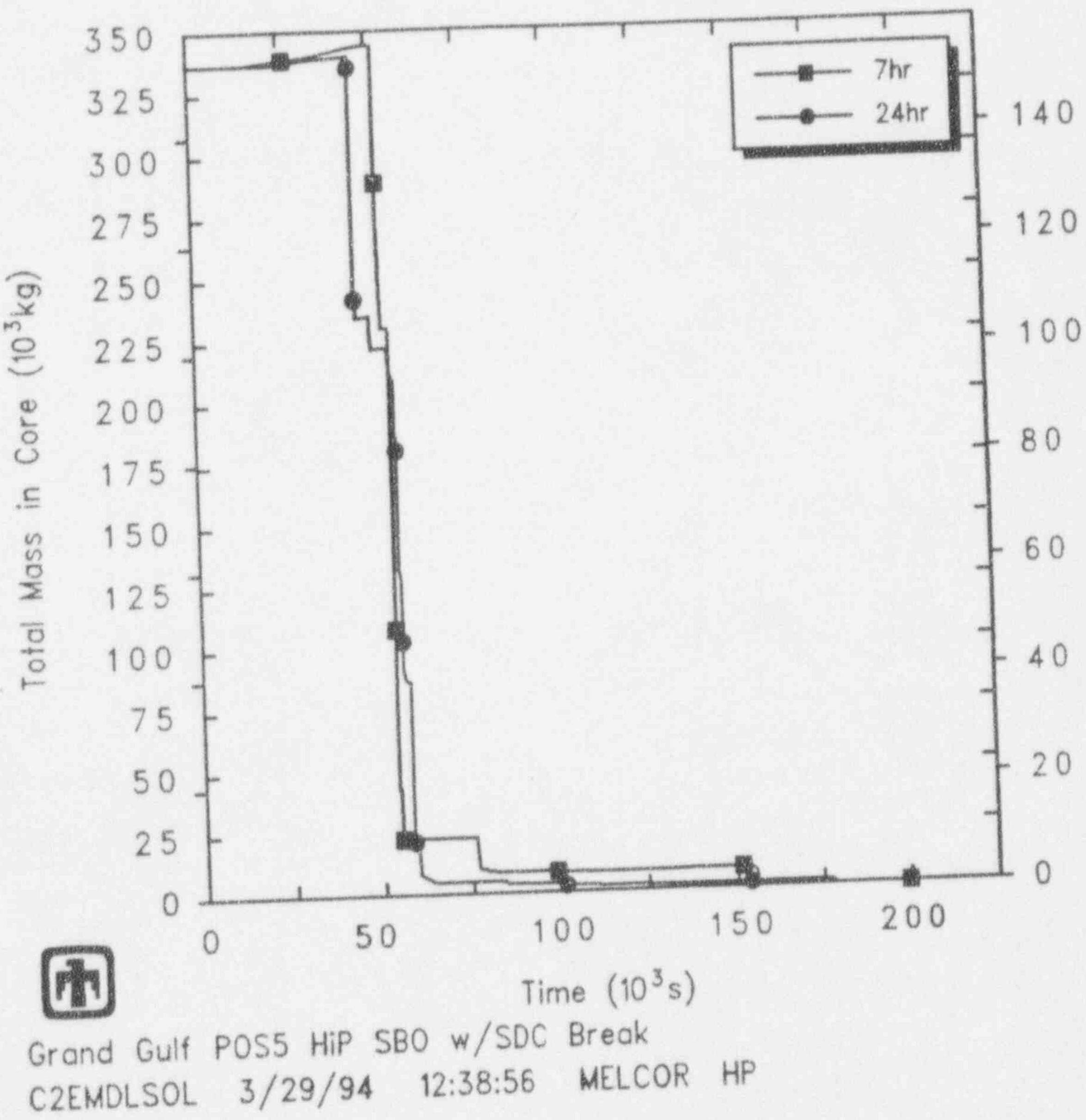


Figure 4.3.2.15. Total Core Material Masses for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated at Various Times After Shutdown

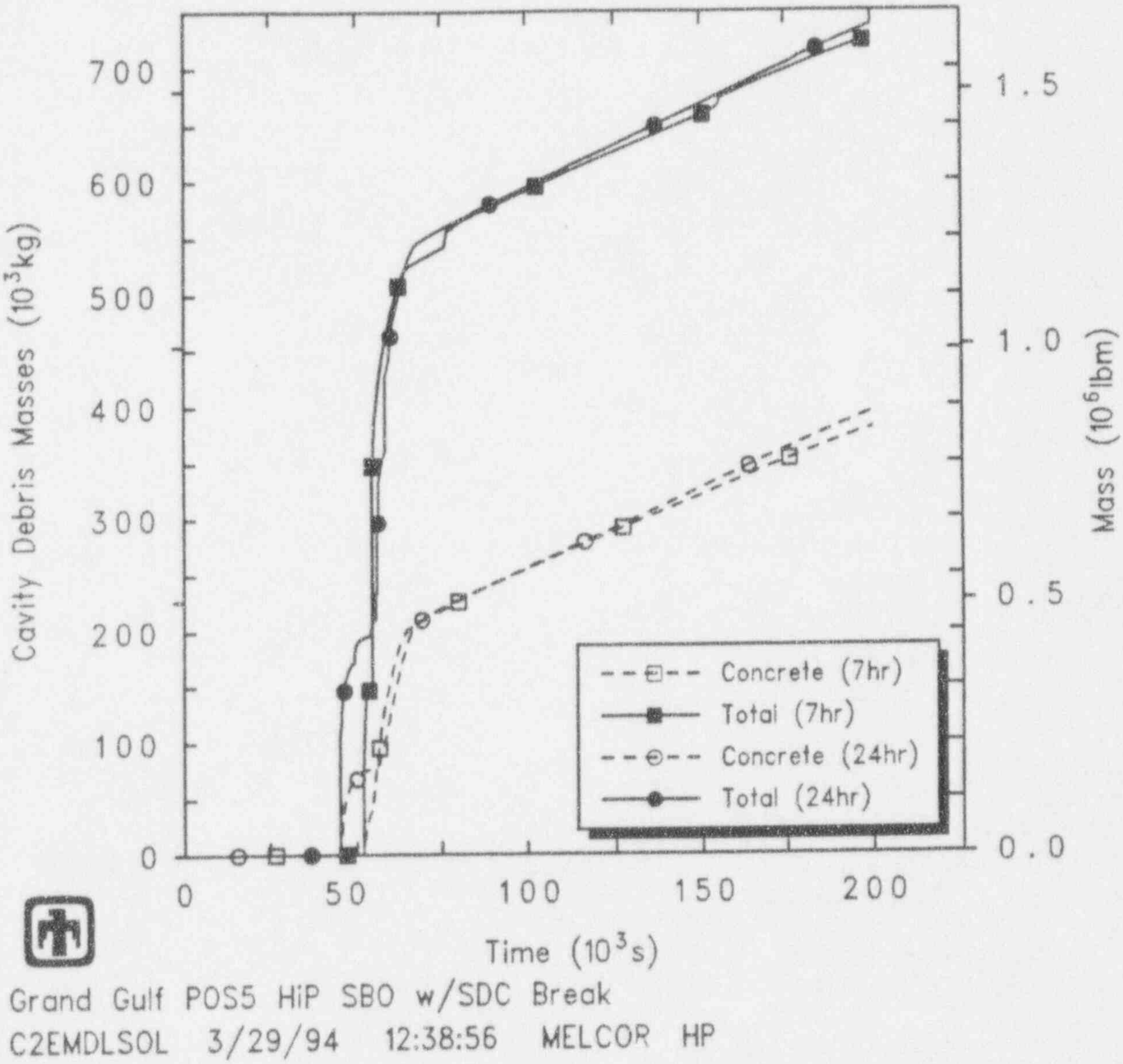


Figure 4.3.2.16. Cavity Total and Concrete Debris Masses for Grand Gulf POS 5 – Station Blackout with Failure to Isolate SDC, Initiated at Various Times After Shutdown

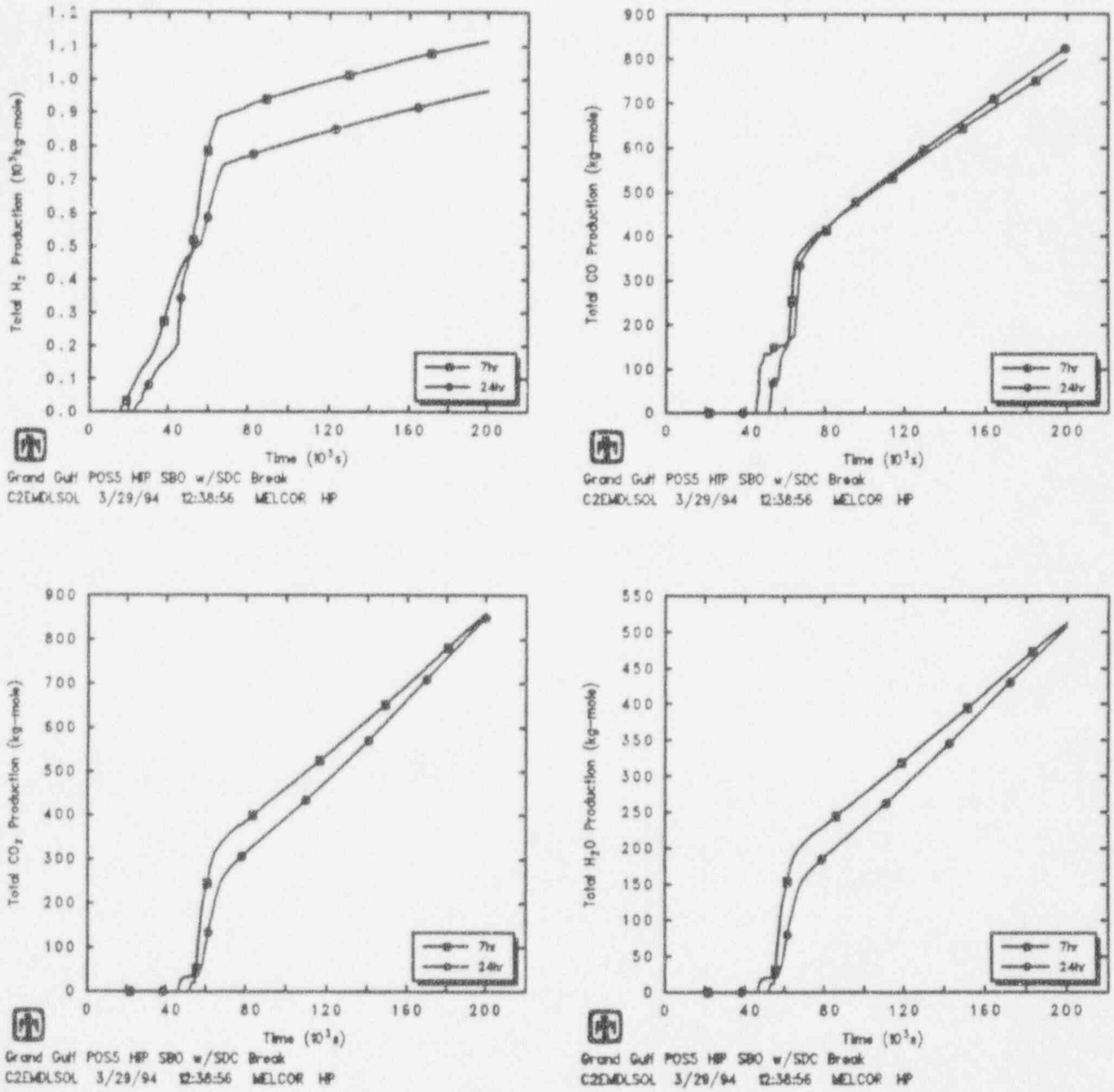


Figure 4.3.2.17. Hydrogen (upper left), Carbon Monoxide (upper right), Carbon Dioxide (lower left) and Steam (lower right) Generation for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated at Various Times After Shutdown

containment dome; and the mole fractions in the upper floors of the auxiliary building generally resemble the behavior shown for the second floor of the auxiliary building (with the behavior in the first floor different because of the SDC break outlet located there).

The drywell control volume atmosphere consists mostly of steam for relatively short times just before and after auxiliary building failure and vessel failure, and late in the accident, and there is a substantial CO concentration spike a short time after vessel failure. The atmosphere composition in the outer containment volumes remains mostly air (nitrogen and oxygen), with little steam or hydrogen (about 10% each) present. The SDC break vents to the first floor of the auxiliary building, resulting in a very high steam concentration in that volume; higher in the auxiliary building the atmosphere composition closely resembles that in the outer containment (because the containment equipment hatch and both of the containment personnel locks are open). The behavior is qualitatively the same in both cases, just stretched out in time more at the lower decay heat levels compared to higher decay heats.

Figures 4.3.2.20 and 4.3.2.21 illustrate the time-dependent release of radionuclides from the fuel debris both within the vessel and in the cavity, for cases initiated 7 hr and 24 hr after scram, respectively. The vertical dotted lines within the plots mark the time of vessel failure, indicating that most of the in-vessel release occurs prior to vessel failure, from the hot debris bed in the active fuel region, while most of the ex-vessel release occurs within a short time period after vessel failure and debris ejection to the cavity, while the core debris is still hot, >2000 K, and consists of a layer of metallic debris above a heavy oxide layer, before enough concrete has been ablated for the debris bed configuration to invert to a light oxide layer above a layer of metallic debris, mixed to a lower average temperature of ~1500 K. Table 4.3.2.2 summarizes the in-vessel, ex-vessel and total amounts of each radionuclide class released, all normalized to the initial inventories of each class. (Note that these amounts generally consider only the release of radioactive forms of these classes, and not additional releases of nonradioactive aerosols from structural materials.)

Unlike the results for the large break LOCA accident simulations described in the previous section, in this station blackout scenario (and the remainder of the Level 2 MELCOR analyses done) the MELCOR model included the formation of CsI from Cs and I_2 released from the fuel, and its subsequent transport, deposition and release. The initial radionuclide inventories are such that all the I_2 released reacts to form CsI while most of the Cs remains unreacted and forms CsOH (the default Cs form).

Almost all (~100%) of the volatile Class 1 (noble gases), Class 2 (CsOH), Class 5 (Te) and Class 16 (CsI) radionuclide species are released from the fuel, primarily in-vessel, as are most (~90-100%) of the Class 3 (Ba) and Class 12 (Sn) inventories. The next major release fraction, dropping rapidly with lower decay heat levels and cooler debris is for uranium. Around 1-10% of the total inventories of Ru and Mo, Ce and La, are released. Finally, a total $\leq$0.1% of the initial inventory of Class 11 (Cd) is predicted to be released. Note that the CORSOR-M fission product release model option used in these analyses has identically zero release in-vessel of Class 7 (Mo), Class 9 (La) and Class 11 (Cd). These are higher release fractions of Ba, Te, Ru, Ce, La and Sn than

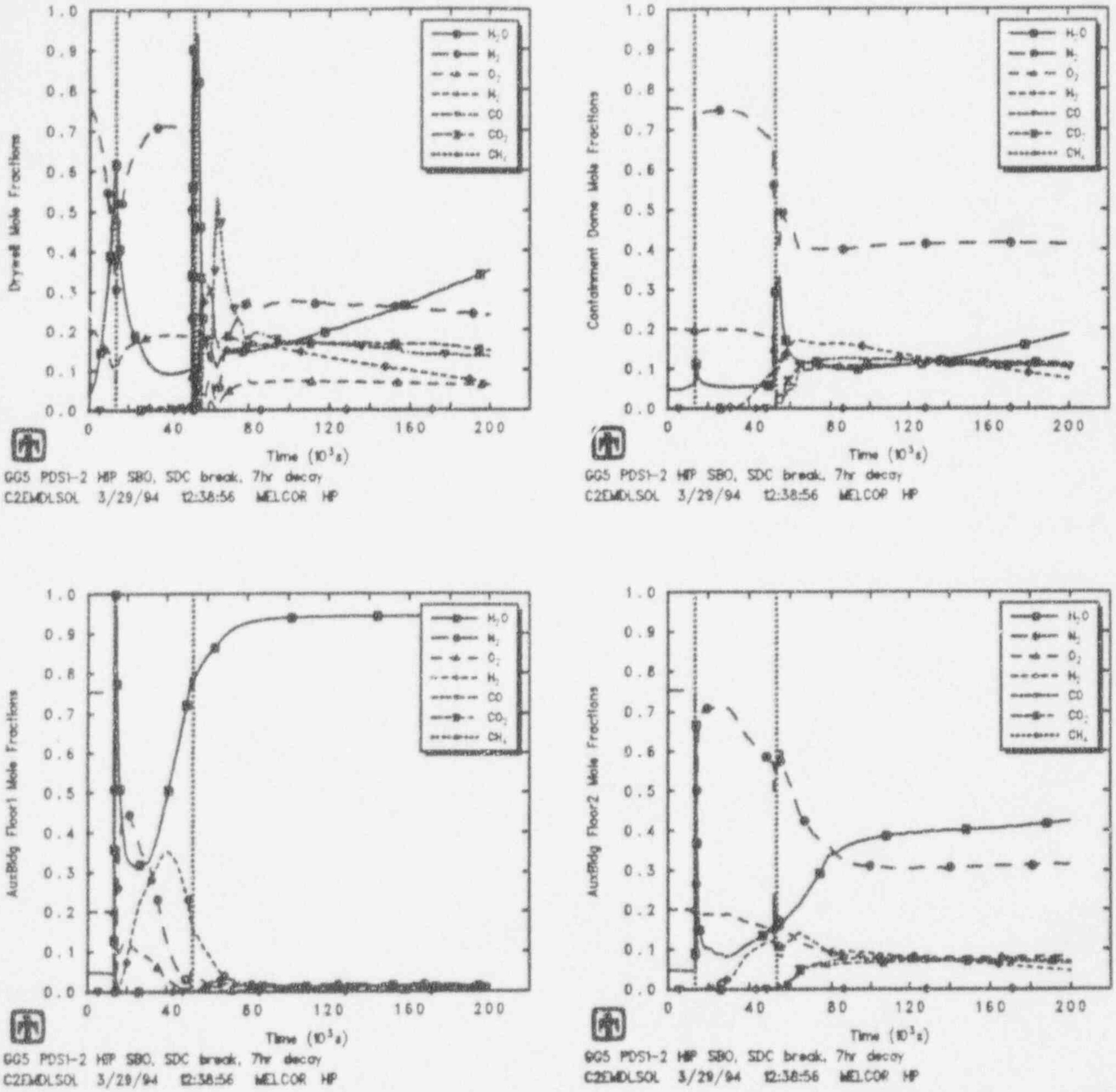


Figure 4.3.2.18. Mole Fractions in Drywell (upper left), Containment Dome (upper right), and Auxiliary Building First Floor (lower left) and Second Floor (lower right) for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated 7 hr After Shutdown

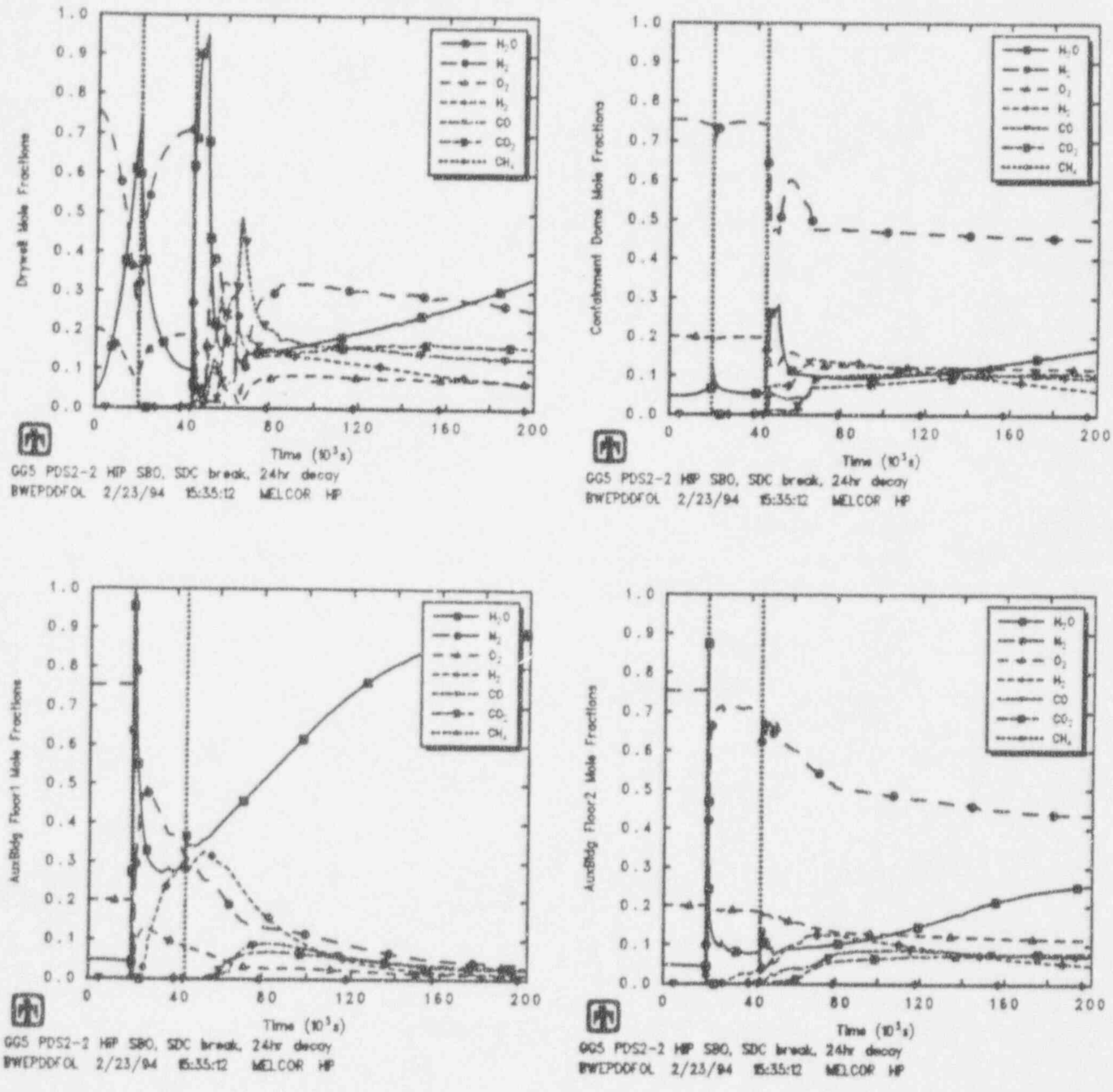
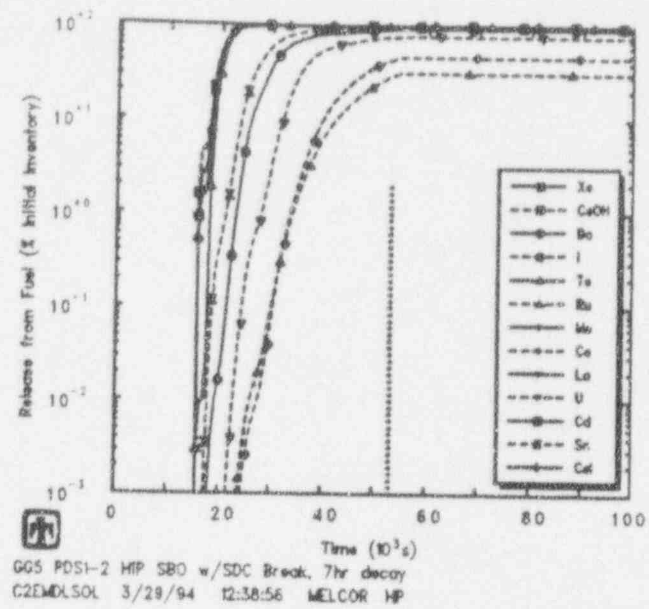


Figure 4.3.2.19. Mole Fractions in Drywell (upper left), Containment Dome (upper right), and Auxiliary Building First Floor (lower left) and Second Floor (lower right) for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated 24 hr After Shutdown



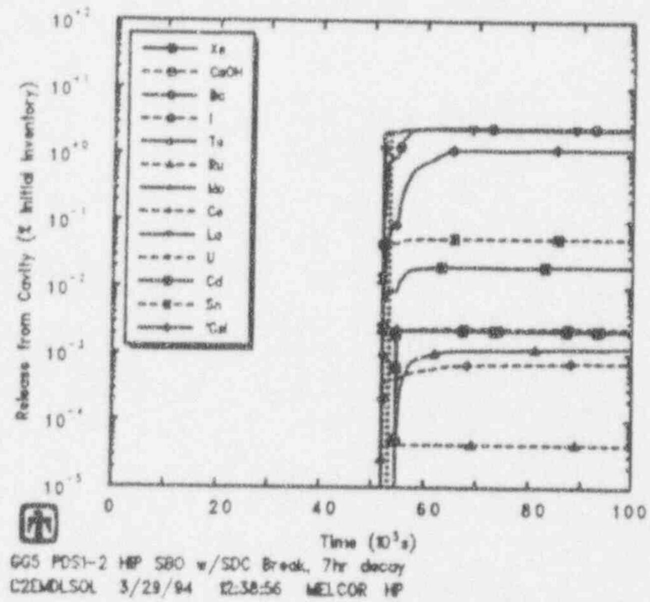
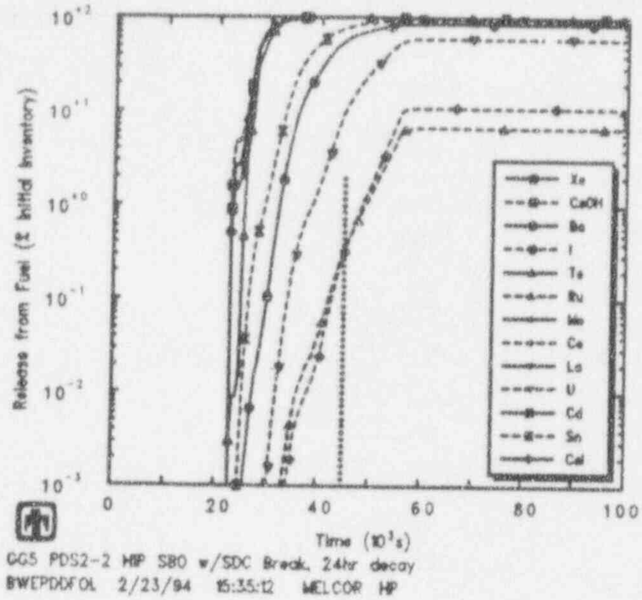


Figure 4.3.2.20. In-Vessel (top) and Ex-Vessel (bottom) Radionuclide Release Fractions for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated 7 hr After Shutdown



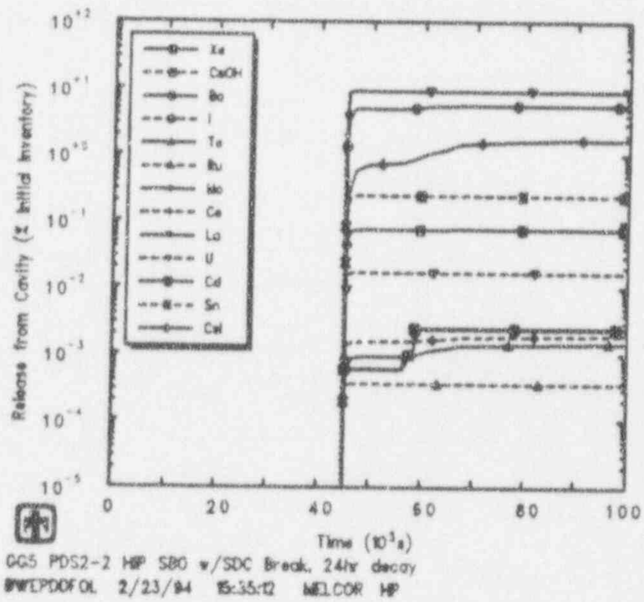


Figure 4.3.2.21. In-Vessel (top) and Ex-Vessel (bottom) Radionuclide Release Fractions for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated 24 hr After Shutdown

Table 4.3.2.2. Final Radionuclide Release Fractions for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated at Various Times After Shutdown

Class	Fission Products Released from Fuel (% Initial Inventory)					
	7 hr			24 hr		
	In-Vessel	Ex-Vessel	Total	In-Vessel	Ex-Vessel	Total
Xe	99.98	0.0022	99.98	99.99	0.0024	99.99
CsOH	~100	0.0020	~100	~100	0.0020	~100
Ba	93.16	2.48	95.64	86.01	5.524	91.534
I	~0	~0	~0	~0	~0	~0
Te	99.97	0.0015	99.97	99.99	0.002	99.99
Ru	31.47	0.00004	31.47	6.704	0.0004	6.70
Mo	0	1.20	1.20	0	1.664	1.664
Ce	46.33	0.0009	46.33	10.88	0.0022	10.88
La	0	2.37	2.37	0	8.99	8.99
U	76.64	0.0025	76.64	59.64	0.017	59.66
Cd	0	0.025	0.025	0	0.079	0.079
Sn	98.05	0.056	99.11	96.03	0.25	96.28
CsI	99.99	0.0023	99.99	~100	0.0024	~100

seen in MELCOR analyses of the large break LOCA sequences described in the previous subsection, reflecting the very high debris temperatures calculated during in-vessel core degradation (shown in Figures 4.3.2.7 and 4.3.2.8).

Figure 4.3.2.22 gives the total radioactive release to the environment in these two cases. The total releases and time history of the release for this accident initiated at two different decay heat levels are nearly identical. The release fractions of individual classes to the environment are shown in Figures 4.3.2.23 and 4.3.2.24. With the break in the SDC system and the failure of the auxiliary building early in this scenario, fission products released during in-vessel core heatup and degradation can immediately escape to the environment (although the only significant release fraction is for the noble gases). There is an increased release of all radionuclide classes at vessel failure, as the core debris falling into and flashing the lower plenum water pool (either immediately in the lower plenum or subsequently in the cavity) generates a substantial steam spike which is vented out the containment and auxiliary building. There is later a continued low-level release of some radionuclide classes, in particular for the volatiles CsOH, CsI and Te.

These environmental releases do not correspond to immediate release of all radionuclides released from the fuel; there is considerable retention of most radionuclide species within the containment and auxiliary building (as discussed below). The noble gases have the greatest releases (>90%) to the environment by the end of the transient periods simulated, because gaseous forms are not scrubbed, filtered, deposited or otherwise retained. There is some release to the environment of the other volatile species (*i.e.*, CsOH, CsI and Te) also, although these are found mostly in aerosol form (and are generally retained in the auxiliary building); the temperatures are higher enough in this station blackout sequence than in the large break LOCA for the volatiles' vapor form to persist, primarily because the containment was flooded in the large break LOCA scenario and dry in the station blackout scenario.

Tables 4.3.2.3 and 4.3.2.4 summarize the distribution of the initial radionuclide inventory at the end of the two calculations initiated at different times after shutdown; they provide an overview of how much of the radionuclides remain bound up in fuel debris in either the core or the cavity, and of how much of the released radionuclides are retained in the primary system *vs* how much of the released radionuclides are released to, or released in, either the containment or the auxiliary building and the environment, all normalized to the initial inventories of each class. Table 4.3.2.5 presents a slightly different breakdown of the released radionuclide final distribution, giving the fractions of released inventory for each class in control volume atmospheres (including the environment), in pools, or deposited or settled onto heat structures at the end of the calculations. (As in Table 4.3.2.2, these amounts consider only the release of radioactive forms of these classes, and not additional releases of nonradioactive aerosols from structural materials.)

These tables show fission product distributions generally similar to those found for the large break LOCA sequences (discussed in the previous section) for the radionuclides with significant ($\geq$80% of initial inventory) release from fuel. In both accident scenarios, most of the noble gases released are in the environment, in the atmosphere. Most of the volatile species (CsOH, CsI and Te) releases occurred in-vessel in both scenarios. However, in this

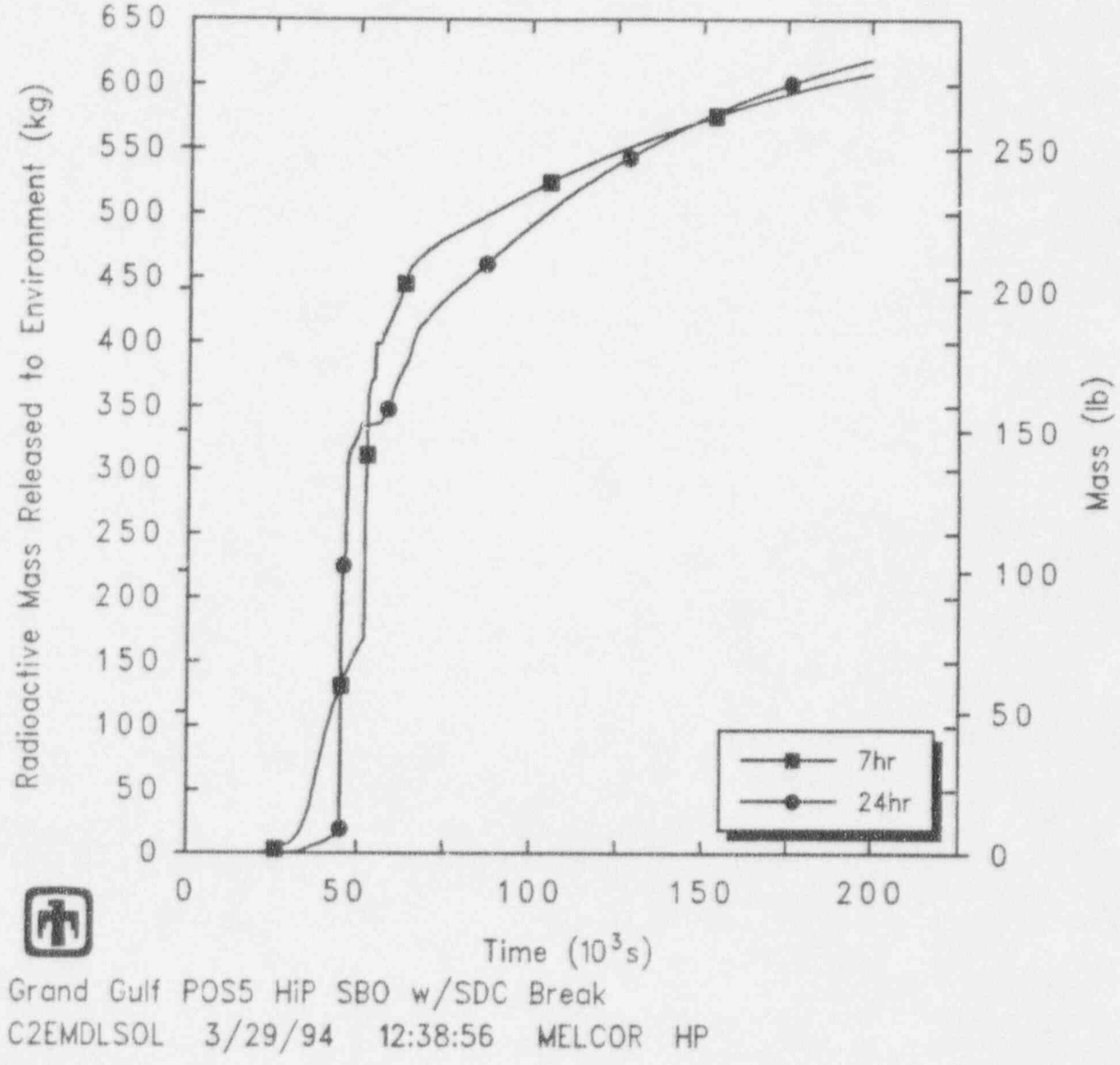


Figure 4.3.2.22. Total Environmental Radionuclide Releases for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated at Various Times After Shutdown

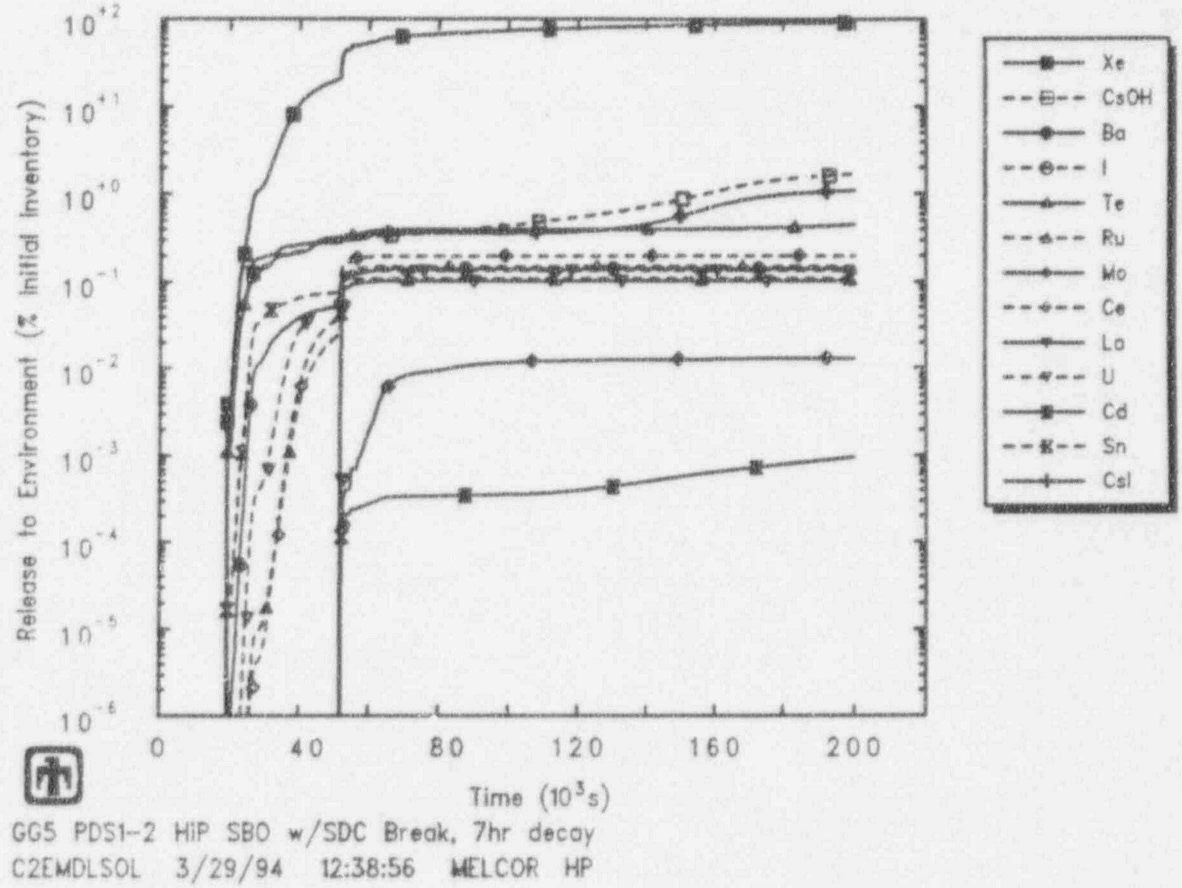


Figure 4.3.2.23. Environmental Radionuclide Release Fractions for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated 7 hr After Shutdown

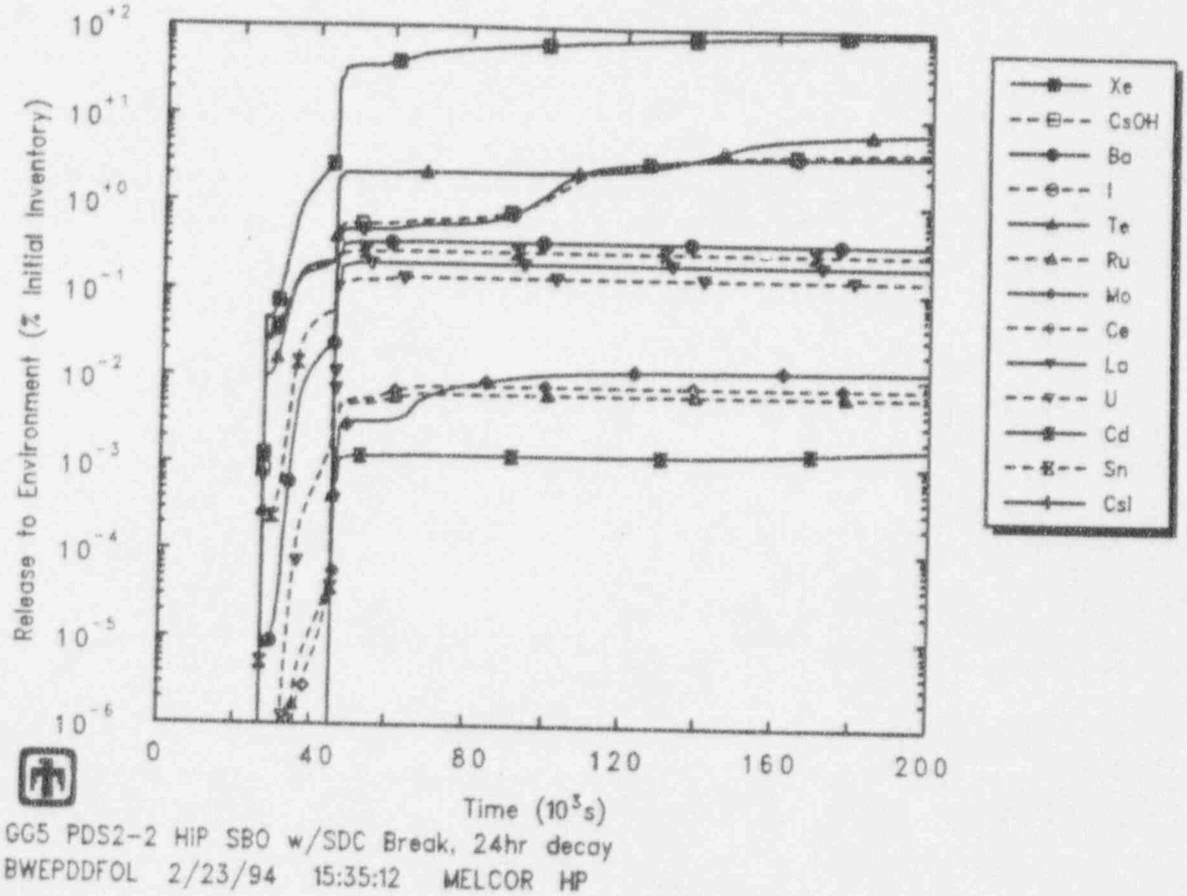


Figure 4.3.2.24. Environmental Radionuclide Release Fractions for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated 24 hr After Shutdown

Table 4.3.2.3. Final Radionuclide Distribution for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated at 7 hr After Shutdown

Class	Fission Product Distribution (% Initial Inventory)				
	Fuel Debris	Primary System	Containment	Auxiliary Building	Environment
Xe	~0	0.0113	4.36	3.29	92.3
CsOH	~0	1.19	5.43	91.74	1.68
Ba	4.37	49.5	20.52	25.42	0.138
Te	0.289	0.293	5.89	93.39	0.442
Ru	68.5	15.7	4.72	10.95	0.147
Mo	98.8	0.087	0.95	0.147	0.013
Ce	53.7	23.6	6.09	16.44	0.192
La	97.6	0.276	1.79	0.204	0.10
U	29.4	38.2	11.74	20.48	0.12
Cd	~100	0.0023	0.019	0.0027	0.0009
Sn	1.89	50.7	14.9	32.39	0.107
CsI	~0	0.0113	4.81	93.94	1.09

Table 4.3.2.4. Final Radionuclide Distribution for Grand Gulf POS 5 – Station Blackout with Failure to Isolate SDC, Initiated at 24 hr After Shutdown

Class	Fission Product Distribution (% Initial Inventory)				
	Fuel Debris	Primary System	Containment	Auxiliary Building	Environment
Xe	~0	0.011	4.45	5.04	90.5
CsOH	~0	0.206	7.42	88.14	4.19
Ba	8.46	41.7	32.30	17.23	0.352
Te	0.003	0.332	8.11	84.62	6.90
Ru	93.3	1.78	4.64	0.28	0.0063
Mo	98.3	0.225	1.29	0.13	0.0124
Ce	89.1	2.80	7.68	0.39	0.008
La	91.0	1.35	7.04	0.39	0.199
U	45.0	16.9	34.67	3.22	0.127
Cd	99.0	0.012	0.062	0.0037	0.0016
Sn	3.72	48.2	20.95	26.81	0.274
CsI	~0	0.151	6.91	89.21	3.77

Table 4.3.2.5. Final Radionuclide State for Grand Gulf POS 5 - Station Blackout with Failure to Isolate SDC, Initiated at Various Times After Shutdown

Class	Fission Products Released from Fuel (% Released Inventory)					
	7 hr			24 hr		
	Atmosphere	Pool	Deposited	Atmosphere	Pool	Deposited
Xe	~100	0	0	~100	0	0
CsOH	2.27	91.1	6.65	5.19	88.2	6.65
Ba	0.145	33.9	66.0	0.39	31.2	68.4
I	~100	0	0	~100	0	0
Te	0.64	92.2	7.11	8.46	84.8	6.75
Ru	0.47	39.8	59.8	0.0094	25.8	74.1
Mo	1.09	40.0	58.9	0.77	38.0	61.3
Ce	0.41	39.7	59.8	0.073	25.4	74.5
La	4.22	40.3	55.5	2.21	35.7	62.0
U	1.71	34.5	65.3	0.23	27.7	72.0
Cd	6.30	34.9	58.8	3.00	34.1	62.9
Sn	0.11	37.7	62.1	0.29	35.0	64.7
CsI	1.44	93.5	5.06	4.50	89.3	6.15

station blackout with the SDC break venting directly to the auxiliary building most of those releases are retained in the auxiliary building, while in the large break LOCA most of those releases are retained in the containment (but primarily in water pools in both cases). About 1-7% of the volatile species are released to the environment in this accident scenario, an order of magnitude or more than in the large break LOCA sequence. The two classes of radionuclides forming aerosols which had substantial releases (Ba and Sn, also occurring mostly in-vessel) were predicted to have about half those releases retained in the vessel, primarily deposited on structures, in both accident scenarios; for this station blackout with failure to isolate SDC the other half of the releases are retained about equally in the containment and in the auxiliary building, about equally in water pools and deposited on structure surfaces, while for the large break LOCA the other half of the releases are retained in the containment, mostly in water pools and a small fraction deposited on structure surfaces.

4.3.3 Station Blackout with Firewater Addition Followed by High Pressure Boiloff, Initiated 7 hr After Shutdown

At the initiation of the accident, the reactor vessel is depressurized and the coolant is at the normal level. The vessel water inventory is at 366.5 K (200°F), which corresponds to the maximum temperature allowed by the Grand Gulf technical specifications for operation in POS 5. The reactor pressure vessel head vent is open. At the start of the accident all core cooling and injection is lost and the SRVs are closed. Two hours after the start of the accident two SRVs are opened and firewater is injected into the core bypass region at a flow rate determined by the pump head curve. Twelve hours after the start of the accident the SRVs close due to depletion of the station batteries, and subsequently the SRVs cycle at their pressure relief setpoint. The suppression pool level is 3.86 m (12.67 ft) from the suppression pool floor. The containment is at 305.4 K (90°F) and the suppression pool is at 308.2 K (95°F). The drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open.

This sequence is identical to the Level 1 station blackout sequence with firewater addition followed by a high pressure boiloff discussed in Section 4.2.8, initiated 7 hr after shutdown.

The sequence of events predicted by MELCOR for this accident with different initiation times is given in Table 4.3.3.1.

The pressure response is identical to that presented in Figures 4.2.8.2 and 4.2.8.3 for the vessel and auxiliary building, respectively, in Section 4.2.8 for this sequence initiated 7 hr after shutdown. Initially, the system begins pressurizing as all core cooling is lost, more quickly for higher decay heat; the pressure then begins dropping after two SRVs are opened 2 hr after the start of the accident. Firewater cooling and steaming out the SRVs keep the vessel pressure down until 12 hr, when depletion of the station batteries cause the SRVs to close. Since the SRVs are now closed, the RPV pressurizes until the SRVs begin operating in the relief mode. After some time, the continued inventory loss out the open RPV vent is sufficient to relieve the steaming in the core and the SRVs close. The

Table 4.3.3.1. Sequence of Events Predicted by MELCOR for Station Blackout with Firewater Addition Followed by High Pressure Boiloff, Initiated 7 hr After Shutdown

Event	Time After Shutdown 7 hr
Accident initiation	0.0
Firewater injection enabled	7,200 s (2 hr)
Core uncovery (TAF) begins	9,787 s (2.72 hr)
Firewater injection stopped	43,200 s (12 hr)
Auxiliary building failed	56,000 s (15.56 hr)
Core heatup begins	56,000 s (15.56 hr)
Clad failure/Gap release	
(Ring 1)	63,097 s (17.53 hr)
(Ring 2)	63,032 s (17.51 hr)
(Ring 3)	63,086 s (17.52 hr)
(Ring 4)	63,427 s (17.62 hr)
(Ring 5)	64,862 s (18.02 hr)
(Ring 6)	79,190 s (22.00 hr)
Core plate failed	
(Ring 1)	90,492 s (25.14 hr)
(Ring 2)	95,165 s (26.43 hr)
(Ring 3)	94,525 s (26.26 hr)
(Ring 4)	94,502 s (26.25 hr)
(Ring 5)	102,598 s (28.50 hr)
(Ring 6)	112,341 s (31.21 hr)
Vessel LH penetration failed	
(Ring 1)	90,582 s (25.16 hr)
(Ring 2)	90,598 s (25.17 hr)
(Ring 3)	90,603 s (25.17 hr)
(Ring 4)	90,653 s (25.18 hr)
(Ring 5)	102,741 s (28.54 hr)
(Ring 6)	112,898 s (31.36 hr)
Commence debris ejection	90,582 s (25.16 hr)
Cavity rupture	199,146 s (55.32 hr)
End of calculation	199,146 s (55.32 hr)

pressure continues to drop until core heatup and damage begins; there is then a brief repressurization, followed very quickly by a final, sharp depressurization due to vessel failure. The flow out the open RPV vent line and later out the SRV's also pressurizes the containment and the auxiliary building. The auxiliary building fails when the SRV's begin cycling at their safety setpoint. The auxiliary building pressure briefly spikes later when the vessel fails.

The firewater injection rate and the vessel inventory response are also identical to the results discussed for the corresponding Level 1 analysis presented in Section 4.2.8 (shown in Figures 4.2.7.3 and 4.2.8.4, respectively). Firewater injection does not equal and reverse inventory loss for about 5hr. After the SRV's close at 12 hr, coolant inventory is lost as the SRV's cycle at the safety setpoint until vessel failure, when all the remaining coolant in the vessel drains to the cavity abruptly.

Figure 4.3.3.1 presents the upper plenum, core and lower plenum swollen and collapsed liquid levels for this accident sequence. The upper plenum levels drop for about 5hr after the SRV's are opened before the firewater addition is sufficient to raise the liquid levels back up briefly. The liquid level in the upper plenum resumes dropping soon after firewater injection is stopped after 12 hr when the SRV's begin cycling in the relief mode. The collapsed level in the core drops below the core midplane before stabilizing and rising again during the 10hr of firewater injection, but the swollen level drops only about a foot into the active fuel region before the firewater addition is sufficient to begin raising the vessel inventory and liquid levels back up. After firewater injection is stopped at 12 hr and the SRV's begin cycling in the relief mode, the vessel liquid level drops smoothly through the upper plenum into the core and continue dropping smoothly partway into the lower plenum, followed by a more gradual loss of the remaining inventory due to boiling and steam outflow. There is very little pool frothing or swelling in any of the vessel volumes in this sequence.

The heatup of the intact fuel and clad is illustrated in Figure 4.3.3.2. The small core uncovery early in the accident progression does not result in significant core heatup before the firewater addition raises the vessel inventory and liquid levels back up After firewater injection ends at 12 hr there is a slow temperature increase, reflecting the rise in saturation temperature as the system pressurizes to the SRV setpoint. Later, after TAF uncovery, core heatup and damage begins. Because the fuel/clad component temperatures in MELCOR are set to zero in a cell when that component fails, this figure shows both the overall heatup rate and the time that the intact fuel/clad component fails through melting of the clad at 2100 K (3320°F).

Figure 4.3.3.3 presents corresponding core debris temperatures in the active fuel region; these are the temperatures of the debris bed formed by the failure of the intact fuel/clad component in MELCOR in a core cell, whose (intact) temperatures were given in Figure 4.3.3.2. The debris bed in the active fuel region reaches peak temperatures $\geq$3500 K (5840°F), significantly above the UO_2 melt temperature of 3113 K (5144°F), except in the lowermost active fuel level where the debris bed temperature remains below the UO_2 melt temperature. The debris bed temperatures predicted in this station blackout sequence with 10hr of firewater addition are somewhat lower than those predicted

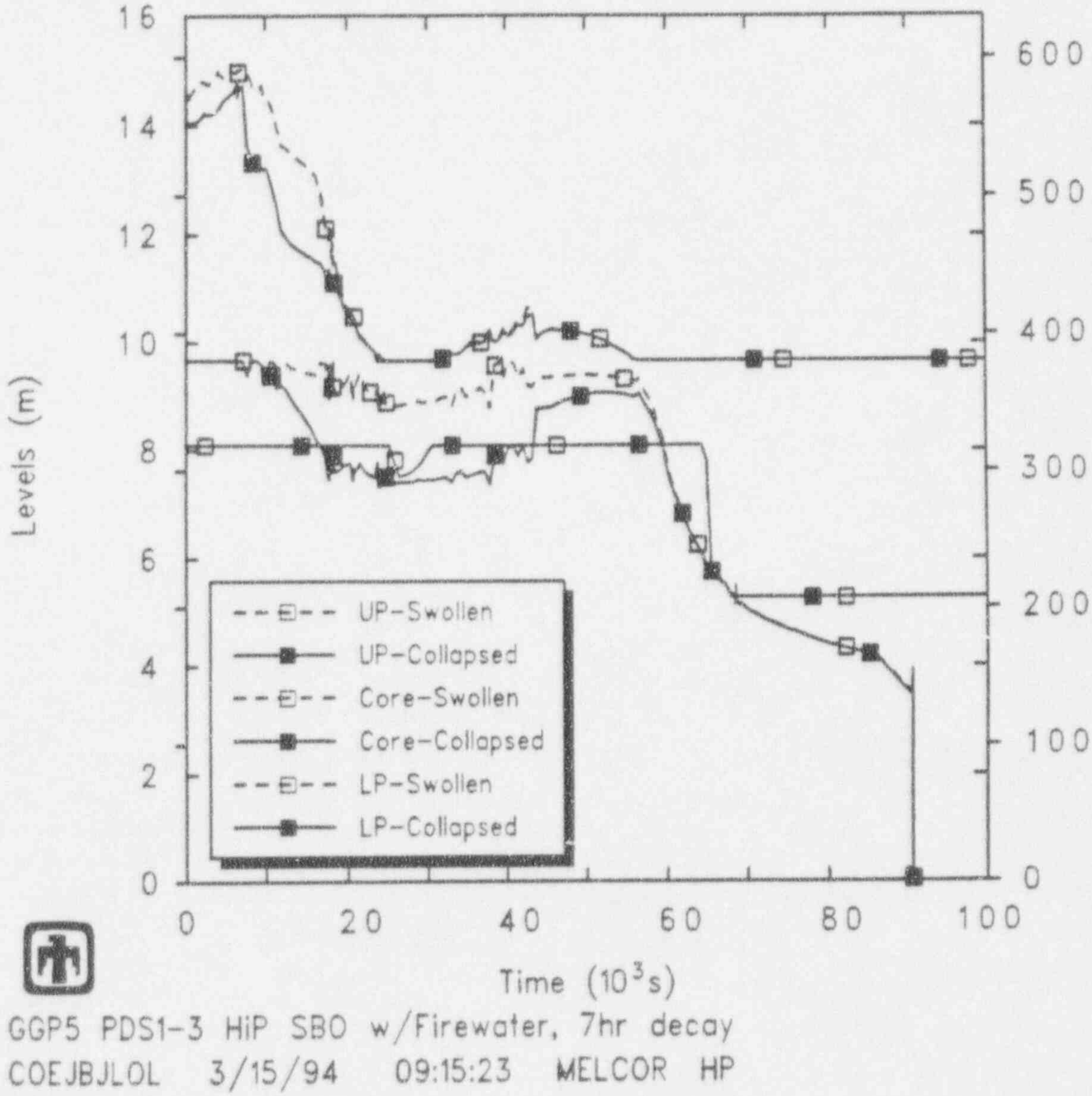


Figure 4.3.3.1. Upper Plenum, Core and Lower Plenum Liquid Levels for Grand Gulf POS 5 - Station Blackout with Firewater Addition Followed by High Pressure Boiloff, Initiated 7 hr After Shutdown

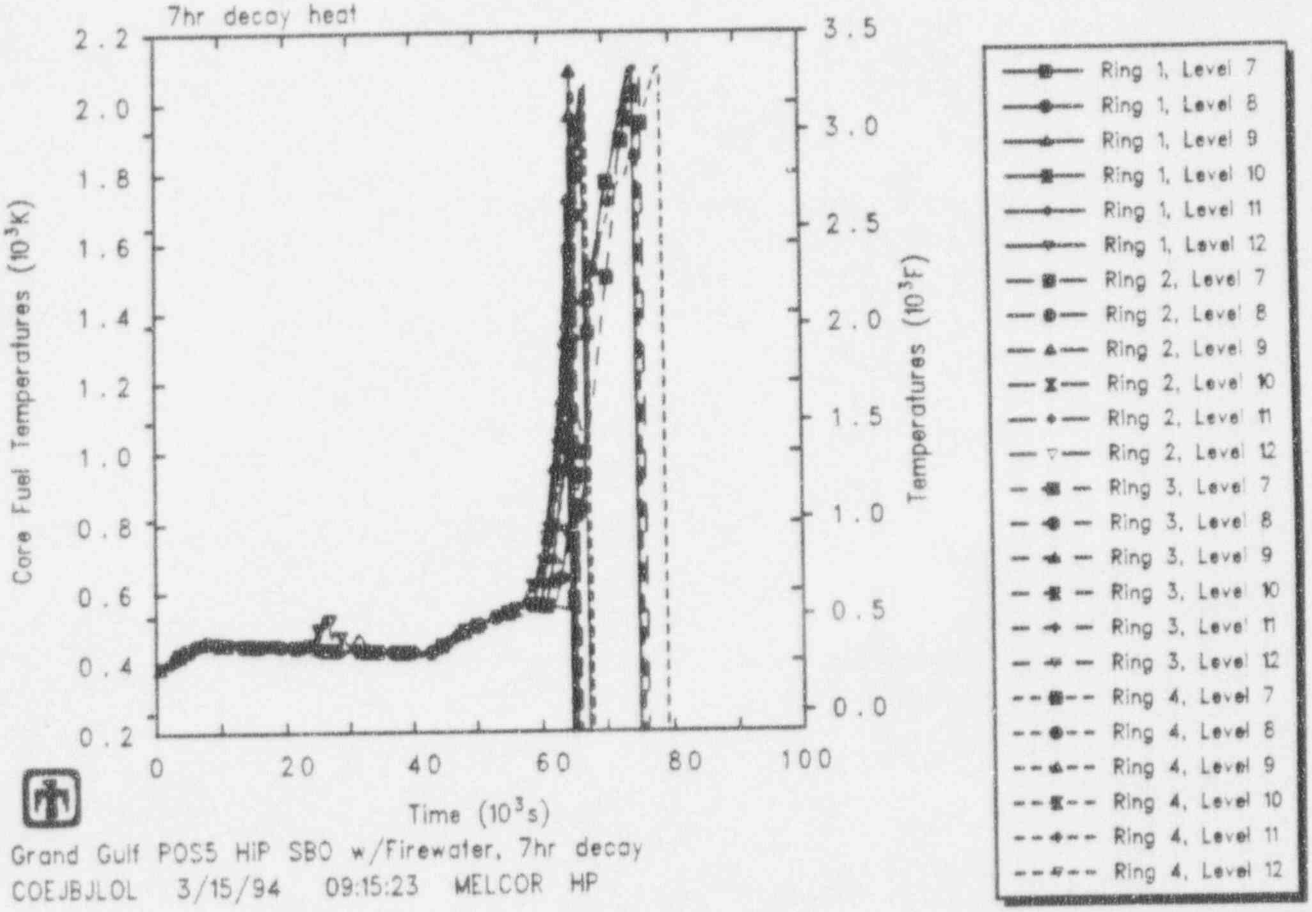


Figure 4.3.3.2. Core Intact Fuel/Clad Temperatures for Grand Gulf POS 5 – Station Blackout with Firewater Addition Followed by High Pressure Boiloff, Initiated 7 hr After Shutdown

in the station blackout sequences with failure to isolate SDC (and no firewater addition) presented in the previous section.

The temperatures of the active fuel region debris bed drop to zero when the core plate fails and the debris relocates to the lower plenum. The predicted temperatures in the debris bed in the lower plenum and core plate are given in Figure 4.3.3.4. Prior to core plate failure there is some cold, refrozen debris both on the core support plate and on the lower core structural material just above the core support plate: the cooling and refreezing of this debris is the cause of the continued gradual drop in lower plenum liquid level due to steaming seen in Figure 4.3.3.1. The debris temperature rises gradually to the core support plate failure temperature of 1273 K (1832°F). After core plate failure hot, high-temperature debris begins appearing the the lower plenum as debris falls from the active fuel region into the lower plenum. With the new debris radial relocation model added in MELCOR 1.8.2, the core plate needs to fail in only one ring before debris from cells in the active fuel region in all radial rings can potentially flow sideways and down, fall through the failed plate, and then spread sideways into cells in the lower plenum in all radial rings. (Thus a lower head penetration can now fail in a ring before the core plate in that ring fails.) The lower head penetrations begin failing almost immediately, and the lower plenum debris temperatures begin dropping to zero as debris is ejected from the vessel to the cavity. Some cool, quenched debris remains present in the lower plenum for a significant period of time, however, as indicated by the 1000-1250 K debris temperatures in the lowest level after vessel failure.

Figure 4.3.3.5 illustrates what fraction of each material in the active fuel region has collapsed into a debris rubble bed held up by the core support plate, prior to core plate failure, debris relocation, lower head failure and debris ejection, for this station blackout scenario with firewater. The fractions of each material and the overall fraction of total material in the active fuel region degraded into particulate debris in this sequence are visibly lower than the corresponding fractions predicted for the station blackout scenarios without firewater addition and with failure to isolate SDC, due to the relatively lower debris temperatures calculated for this sequence. The debris bed forms later in time, due to the delay in core heatup until after firewater injection is stopped, and remains in the active fuel region for a shorter time than predicted for the station blackout scenarios without firewater addition and with failure to isolate SDC.

Figure 4.3.3.6 shows both the total and the individual masses of core materials (UO_2, Zircaloy and ZrO_2, stainless steel and steel oxide, and control rod poison) remaining in the vessel. This includes both material in the active fuel region and in the lower plenum. Debris ejection began very soon after lower head failure. This figure illustrates that most of the core material was lost from the vessel to the cavity quickly, in step-like stages. In all cases, all of the UO_2 was transferred to the cavity within ~1 hr after the initial vessel lower head penetration failure, as was the unoxidized zircaloy, the associated zirc oxide and the control rod poison. A small fraction (10-15%) of the structural steel in the lower plenum, and some associated steel oxide, was predicted to remain unmelted and in place, more than in the station blackout scenarios without firewater addition and with failure to isolate SDC.

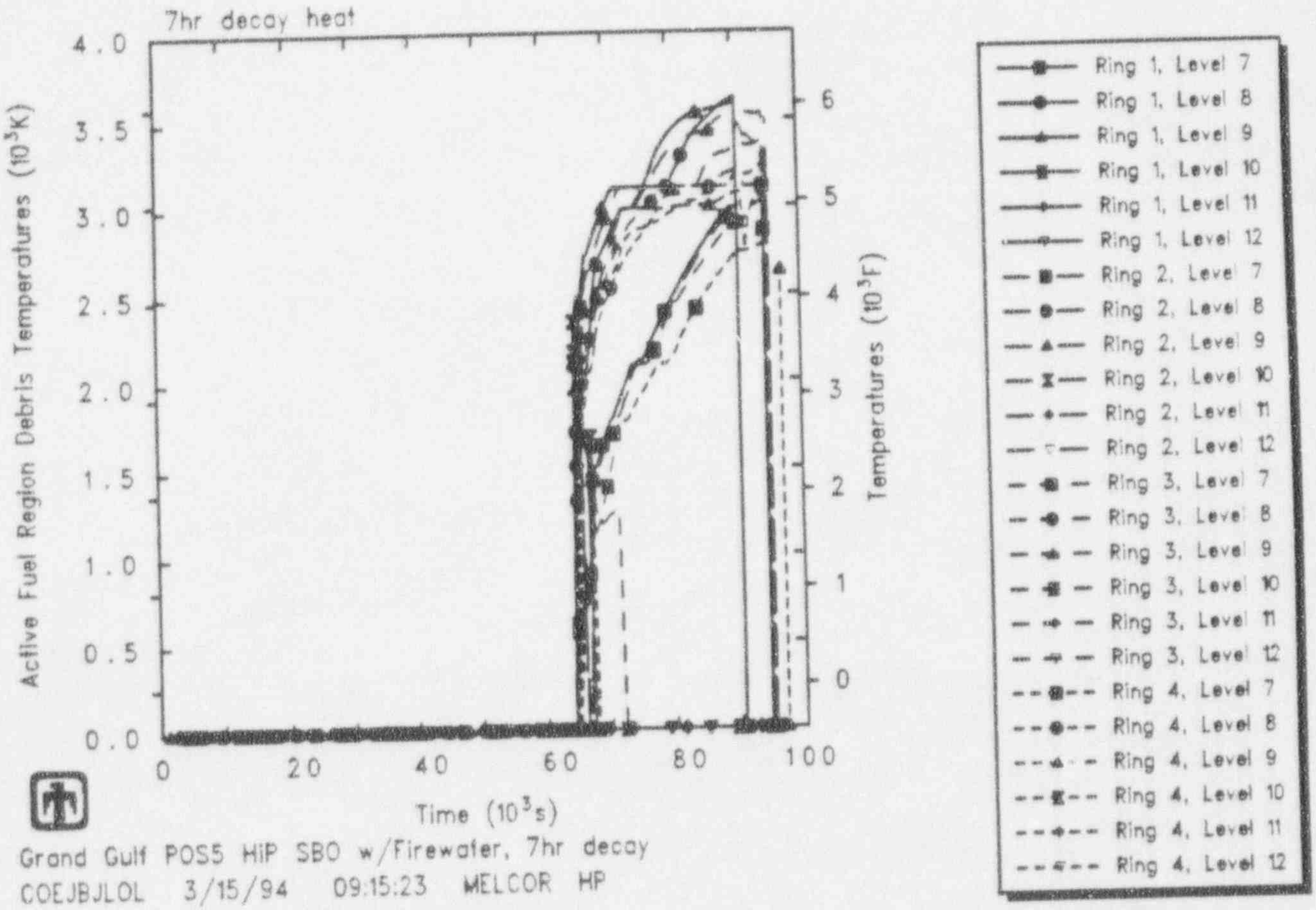


Figure 4.3.3.3. Core Active Fuel Region Debris Bed Temperatures for Grand Gulf POS 5 - Station Blackout with Firewater Addition Followed by High Pressure Boiloff, Initiated 7 hr After Shutdown

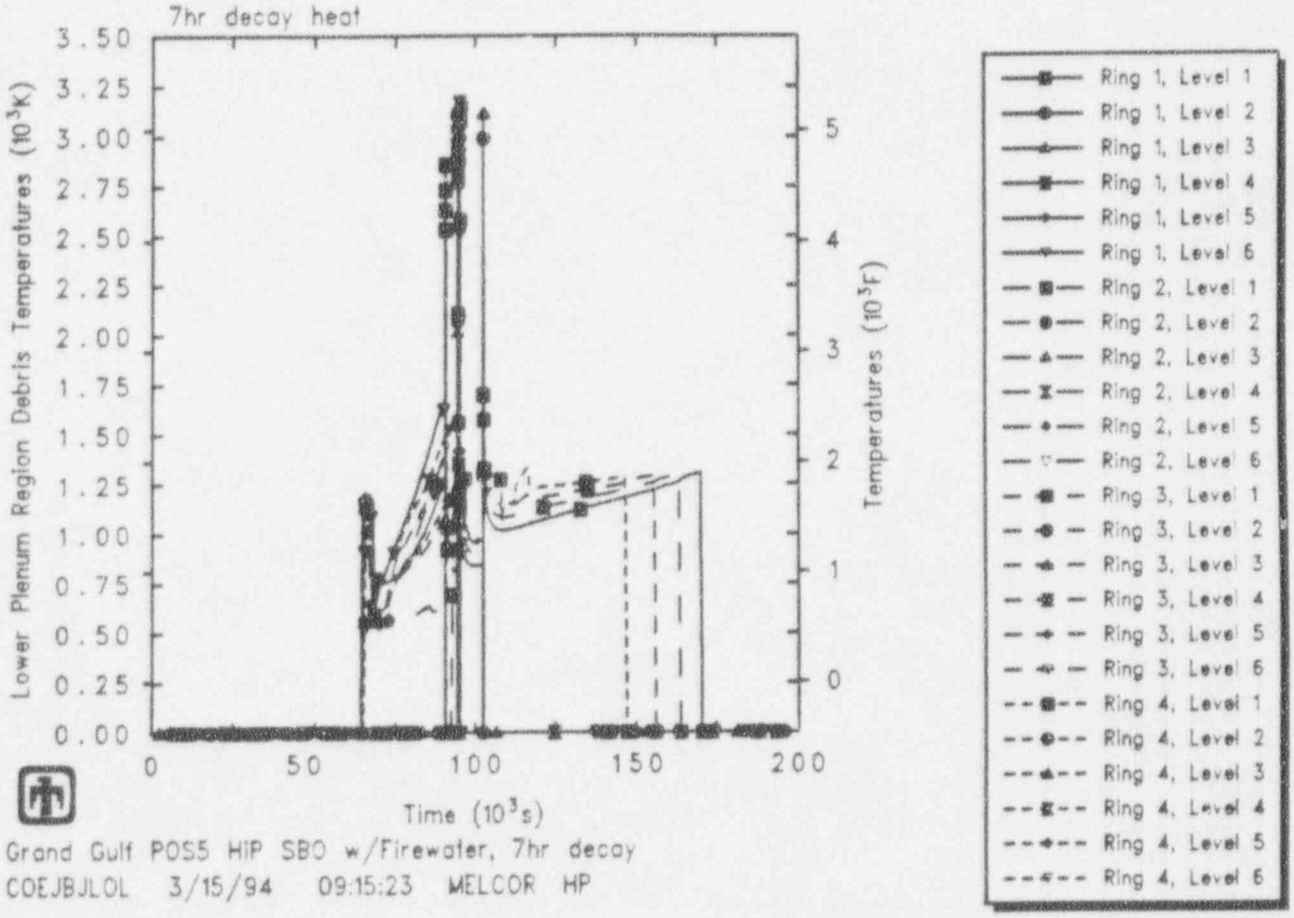


Figure 4.3.3.4. Core Lower Plenum and Core Support Plate Debris Bed Temperatures for Grand Gulf POS 5 – Station Blackout with Firewater Addition Followed by High Pressure Boiloff, Initiated 7 hr After Shutdown

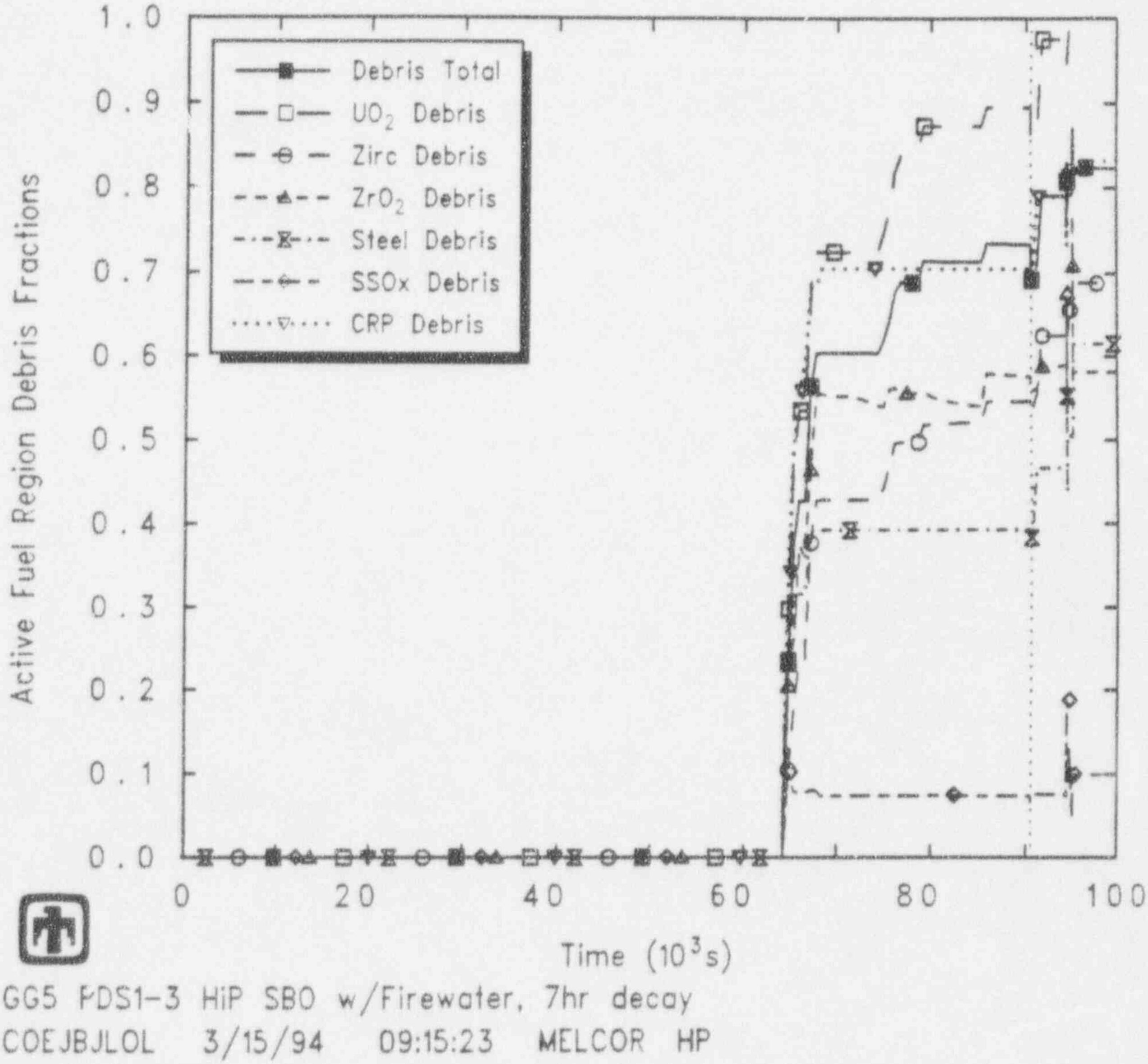


Figure 4.3.3.5. Core Active Fuel Region Degraded Material Fractions for Grand Gulf POS 5 - Station Blackout with Firewater Addition Followed by High Pressure Boiloff, Initiated 7 hr After Shutdown

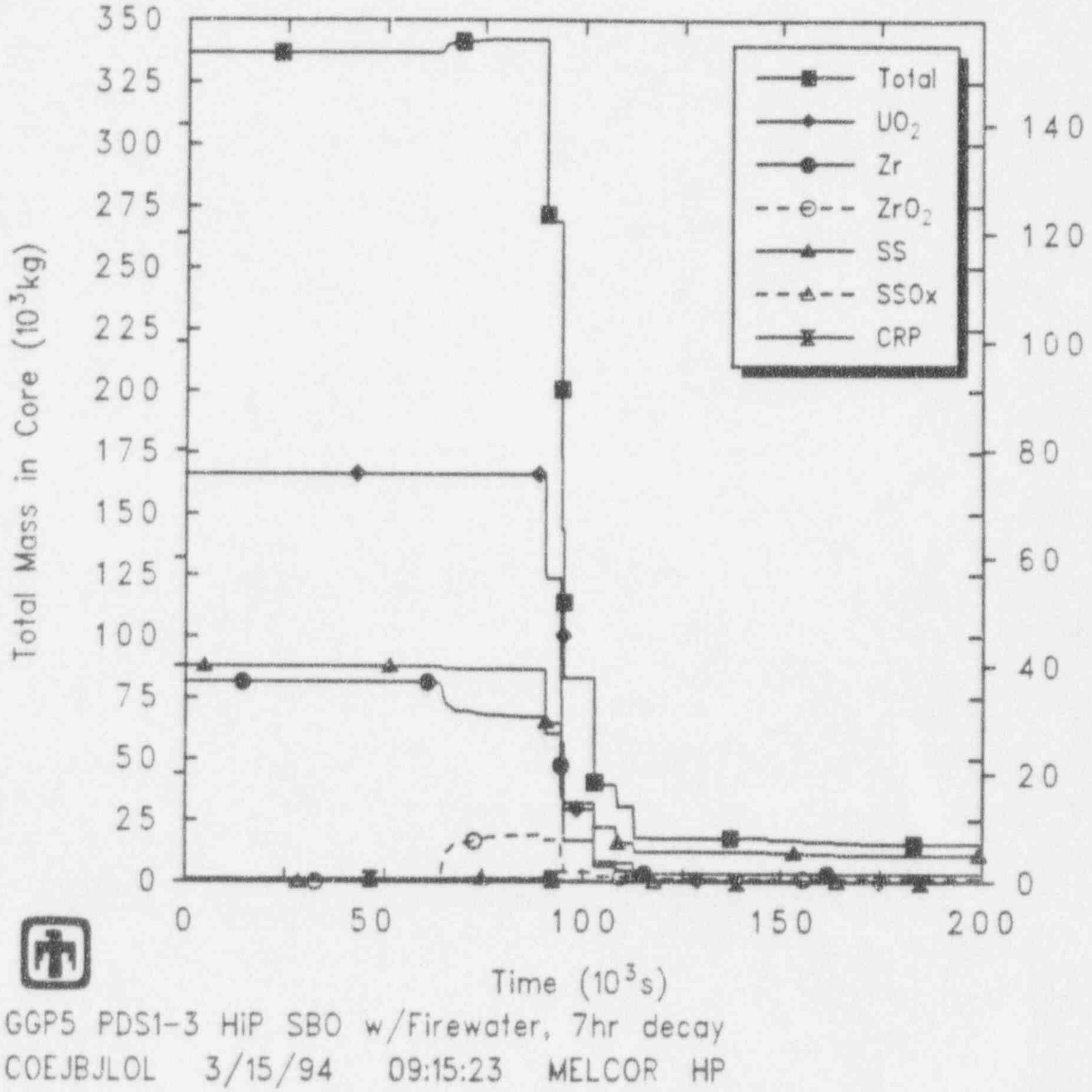


Figure 4.3.3.6. Total and Individual Core Material Masses for Grand Gulf POS 5 - Station Blackout with Firewater Addition Followed by High Pressure Boiloff, Initiated 7 hr After Shutdown

The debris material lost from the vessel is ejected to the drywell pedestal cavity. Figure 4.3.3.7 presents the amounts of ejected core debris, concrete ablated and the total cavity debris mass (*i.e.*, core debris combined with concrete ablation products). As in the other sequences analyzed, concrete ablation is quite rapid soon after debris ejection while the core debris is hot (>2000 K) and consists of a layer of metallic debris above a heavy oxide layer, and then slows noticably after enough concrete has been ablated for the debris bed configuration to invert to a light oxide layer above a layer of metallic debris, mixed to a lower average temperature of ~1500 K.

The calculated production of steam and noncondensable gases (H_2, CO, CO_2 and H_2O) is depicted in Figure 4.3.3.8. The hydrogen production shown includes both in-vessel production (the initial step increase) and ex-vessel production in the cavity (the later-time increase). The in-vessel hydrogen generation corresponds to the oxidation of about 15% of the zircaloy and about 1% of the steel in the core and lower plenum, prior to vessel failure and debris ejection. As soon as the core debris enters the cavity, core-concrete interaction begins, resulting in the production of carbon dioxide and hydrogen; reduction of these gases by the molten metal in the core debris also gives rise to carbon monoxide and hydrogen. The generation rates and amounts of these gases produced, and the amount of concrete ablated, are generally similar in this station blackout sequence with 10hr of firewater addition followed by a high pressure boiloff to the corresponding rates and amounts calculated in the station blackout scenarios with failure to isolate SDC and no firewater addition, described in the previous section.

The mole fractions in the drywell, containment dome and auxiliary building (first and second floors) are shown in Figure 4.3.3.9, including vertical dotted lines at auxiliary building failure and at vessel failure for reference. The mole fractions in the cavity resemble the behavior shown for the drywell; the mole fractions in the containment equipment hatch are very similar to those shown for the containment dome, and the mole fractions in the upper floors of the auxiliary building generally resemble the behavior shown for the second floor of the auxiliary building (but with more steam higher in the auxiliary building late in time and correspondingly less nitrogen). The inner containment atmosphere consists mostly of steam, building up rapidly after the SRVs are first locked open and later cycle in the relief mode, decreasing somewhat after vessel failure and noncondensable gas generation due to core-concrete interaction, but remaining more than half steam throughout the transient period simulated. The outer containment steam concentration begins rising slowly when the SRVs are locked open and later increases rapidly to almost 50% steam after the SRVs begin cycling in the relief mode. The containment is open to the auxiliary building in the second and fourth floors. The atmosphere in the dead-end first floor of the auxiliary building remains near ambient with small fractions of steam and noncondensables added from the upper floors; higher in the auxiliary building the atmosphere composition closely resembles that in the outer containment (because the containment equipment hatch and both of the containment personnel locks are open), but with more steam and core-concrete interaction noncondensables higher in the auxiliary building late in time and correspondingly less nitrogen and oxygen.

Figure 4.3.3.10 illustrate the time-dependent release of radionuclides from the fuel

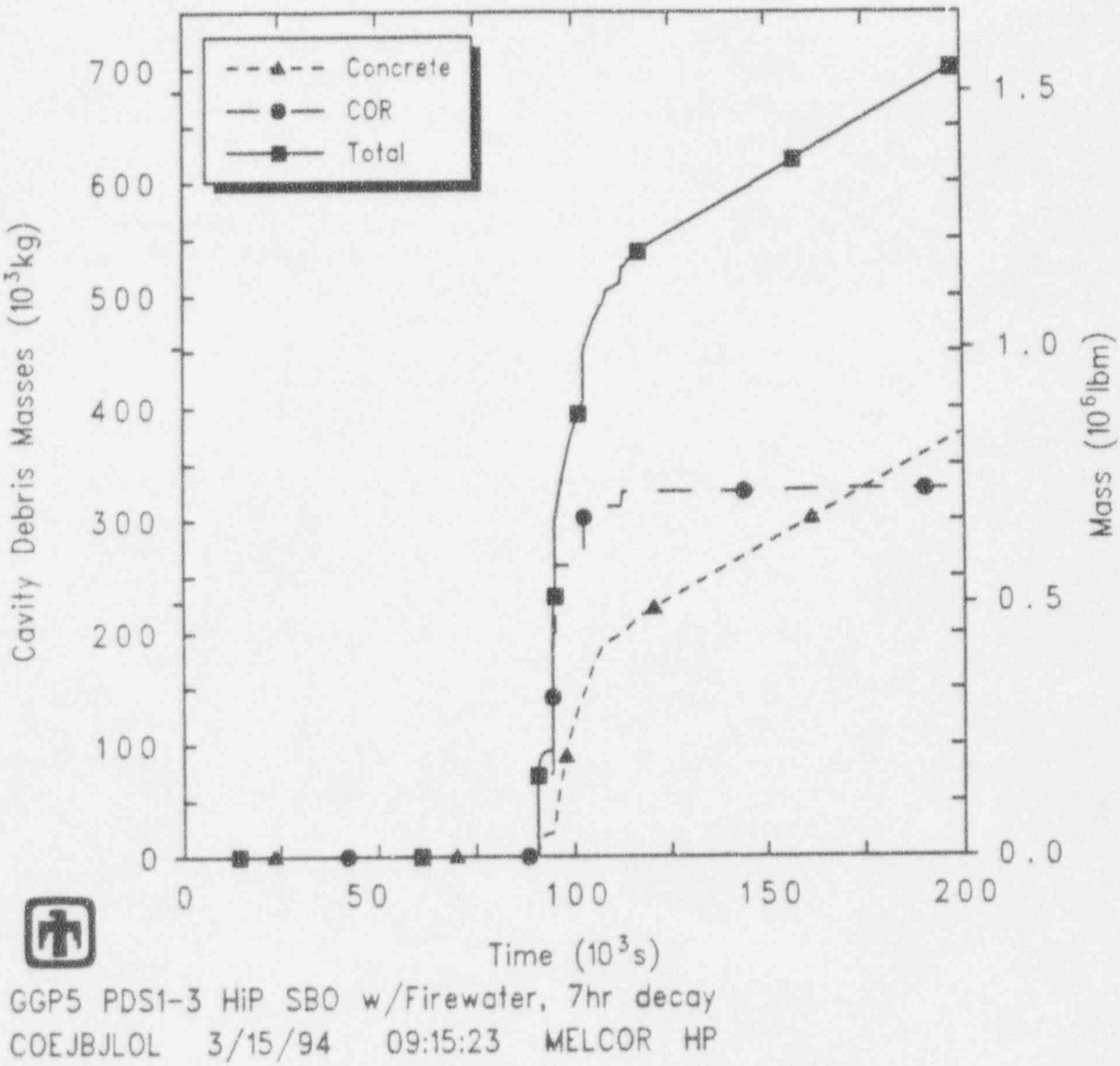


Figure 4.3.3.7. Cavity Total and Core and Concrete Debris Masses for Grand Gulf POS 5 – Station Blackout with Firewater Addition Followed by High Pressure Boiloff, Initiated 7 hr After Shutdown

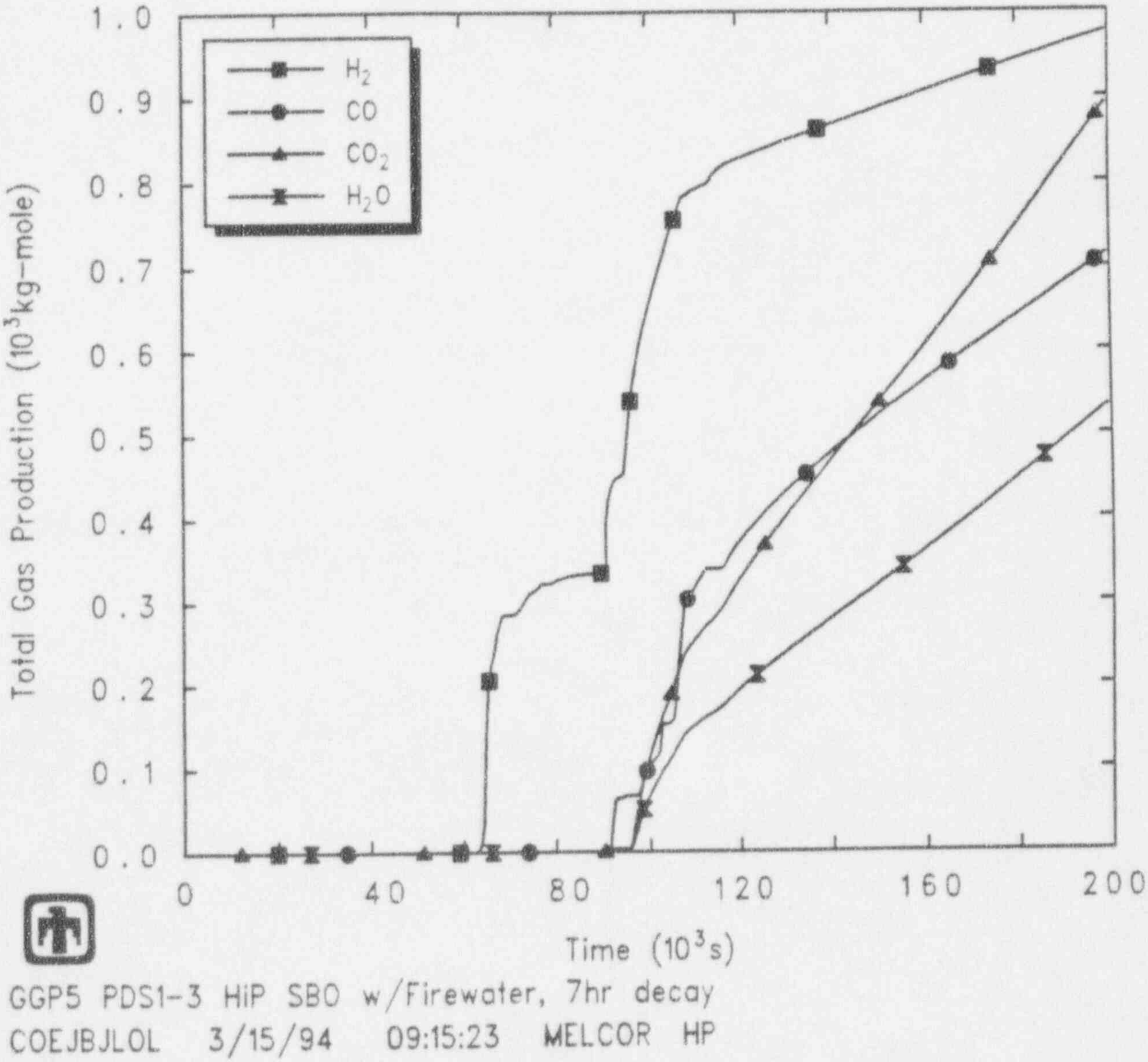


Figure 4.3.3.8. Hydrogen, Carbon Monoxide, Carbon Dioxide and Steam Generation for Grand Gulf POS 5 – Station Blackout with Firewater Addition Followed by High Pressure Boiloff, Initiated 7 hr After Shutdown

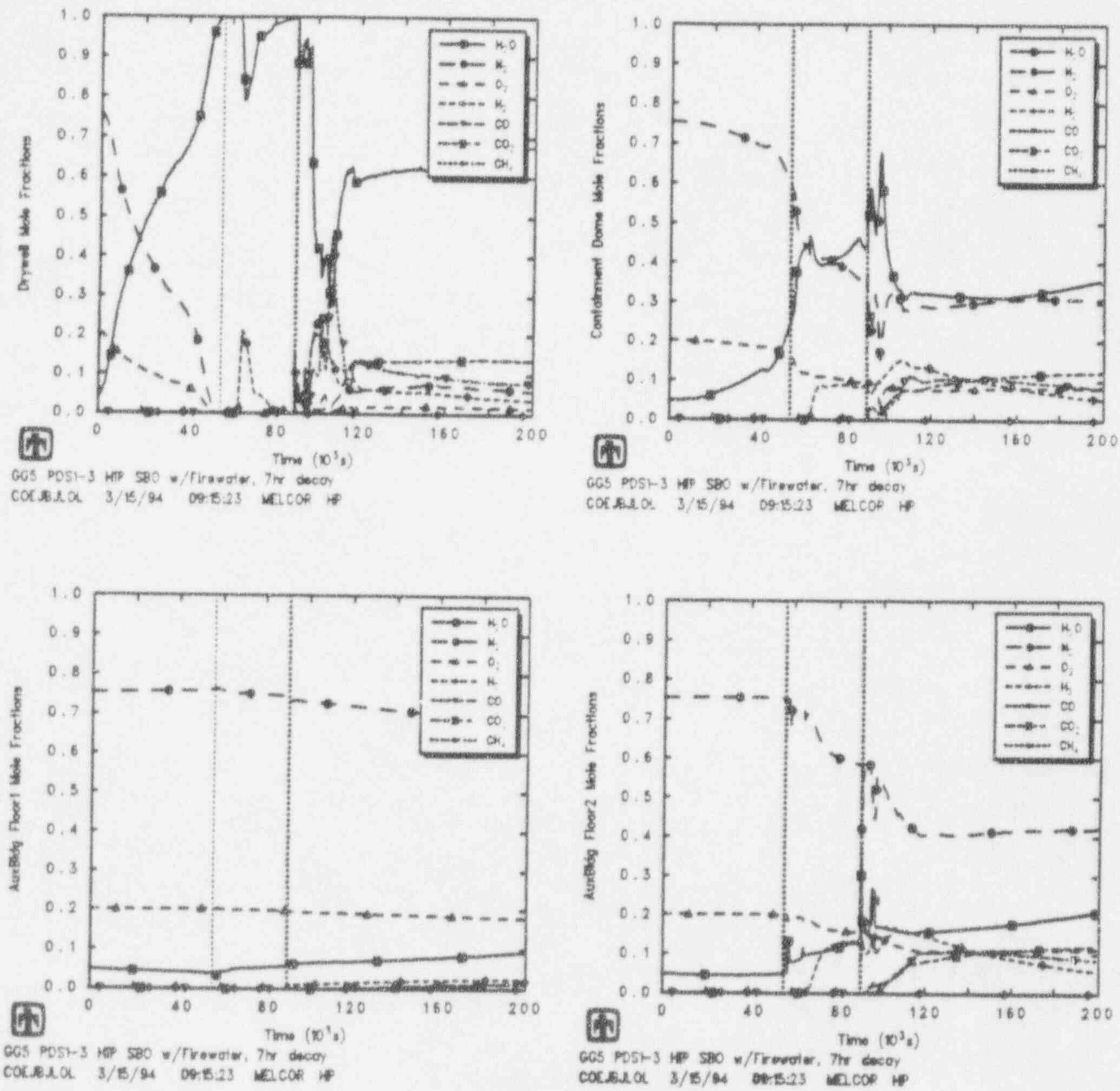


Figure 4.3.3.3. Mole Fractions in Drywell (upper left), Containment Dome (upper right), and Auxiliary Building First Floor (lower left) and Second Floor (lower right) for Grand Gulf POS 5 – Station Blackout with Firewater Addition Followed by High Pressure Boiloff, Initiated 7 hr After Shutdown

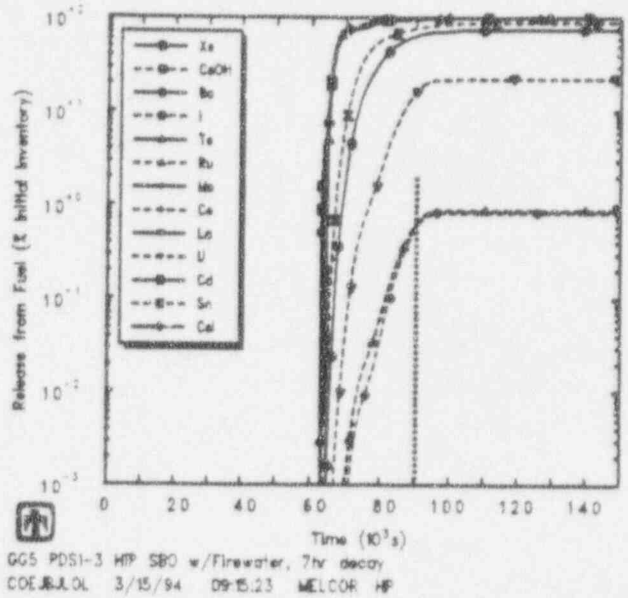
debris both within the vessel and in the cavity. The vertical dotted lines within the plots mark the time of vessel failure, indicating that most of the in-vessel release occurs prior to vessel failure, from the hot debris bed in the active fuel region, while most of the ex-vessel release occurs within a short time period after vessel failure and debris ejection to the cavity, while the core debris is still hot, before enough concrete has been ablated for the debris bed configuration to cool and invert; this behavior is seen in most of our MELCOR analyses. Table 4.3.3.2 summarizes the in-vessel, ex-vessel and total amounts of each radionuclide class released, all normalized to the initial inventories of each class. (Note that these amounts generally consider only the release of radioactive forms of these classes, and not additional releases of nonradioactive aerosols from structural materials.)

The release behavior predicted by MELCOR can be grouped into several subdivisions. Almost all (~100%) of the volatile Class 1 (noble gases), Class 2 (CsOH), Class 5 (Te) and Class 16 (CsI) radionuclide species are released, primarily in-vessel, as are most (80-90%) of the Class 3 (Ba) and Class 12 (Sn) inventories. The next major release fraction, dropping rapidly with lower decay heat levels and cooler debris is for uranium. Around 0.1-2% of the total inventories of Ru and Mo, Ce and La, are released. Finally, a total ≤0.01% of the initial inventory of Class 11 (Cd) is predicted to be released. Note that the CORSOR-M fission product release model option used in these analyses has identically zero release in-vessel of Class 7 (Mo), Class 9 (La) and Class 11 (Cd).

Figure 4.3.3.11 gives the total radioactive release to the environment in these two cases. The release fractions of individual classes to the environment are shown in Figure 4.3.3.12. The release to the environment begins before vessel failure in this sequence. Fission products released during the in-vessel core heatup and degradation process are transported to the containment through the cycling SRVs and the open RPV vent line; they then move from the containment to the auxiliary building through the open containment equipment hatch and personnel locks, and can escape to the environment as soon as the auxiliary building fails (at about 56,000 s or 15-16 hr).

These environmental releases do not correspond to immediate release of all radionuclides released from the fuel; there is considerable retention of most radionuclide species within the containment and auxiliary building (as discussed below). The noble gases have the greatest releases (>90%) to the environment by the end of the transient period simulated, because gaseous forms are not scrubbed, filtered, deposited or otherwise retained; in addition, there is some release to the environment of the other volatile species (*i.e.*, CsOH, CsI and Te) also, although these are found mostly in aerosol form and are largely retained in the containment. (Note that most of the retention was in the auxiliary building in the station blackout sequences with failure to isolate SDC because that was where the outlet of the SDC break was located; most of the retention is in the containment in this station blackout scenario with firewater addition followed by a high pressure boiloff because in this case the outflow is primarily through the SRVs, the open RPV head vent and the vessel lower head penetration failures, which all go to the containment.)

Table 4.3.3.3 summarizes the distribution of the initial radionuclide inventory at the end of the two calculations initiated at different times after shutdown; they provide an overview of how much of the radionuclides remain bound up in fuel debris in either the



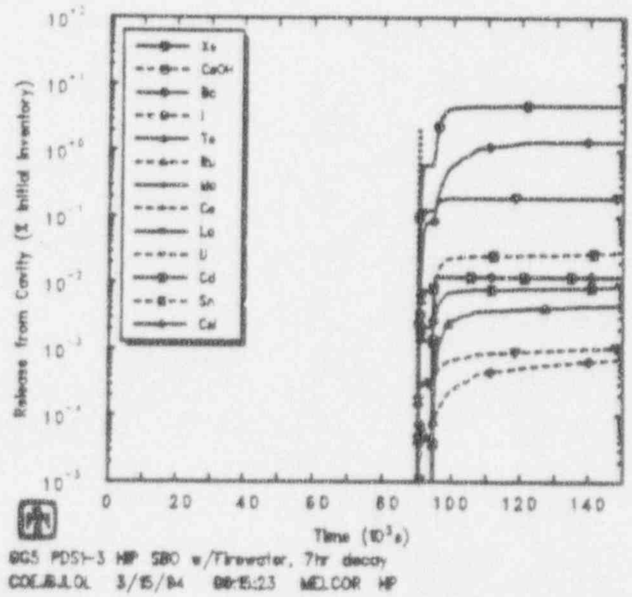


Figure 4.3.3.10. In-Vessel (top) and Ex-Vessel (bottom) Radionuclide Release Fractions for Grand Gulf POS 5 – Station Blackout with Firewater Addition Followed by High Pressure Boiloff, Initiated 7 hr After Shutdown

Table 4.3.3.2. Final Radionuclide Release Fractions for Grand Gulf POS 5 – Station Blackout with Firewater Addition Followed by High Pressure Boiloff. Initiated 7 hr After Shutdown

Class	Fission Products Released from Fuel (% Initial Inventory)		
	In-Vessel	Ex-Vessel	Total
Xe	99.99	0.0122	~100
CsOH	~100	0.0120	~100
Ba	74.83	4.77	79.60
I	~0	~0	~0
Te	99.98	0.0055	99.99
Ru	0.894	0.000003	0.894
Mo	0	1.35	1.35
Ce	0.834	0.0009	0.834
La	0	0.192	0.192
U	22.19	0.00126	22.19
Cd	0	0.014	0.014
Sn	88.85	0.049	88.90
CsI	99.99	0.0124	~100

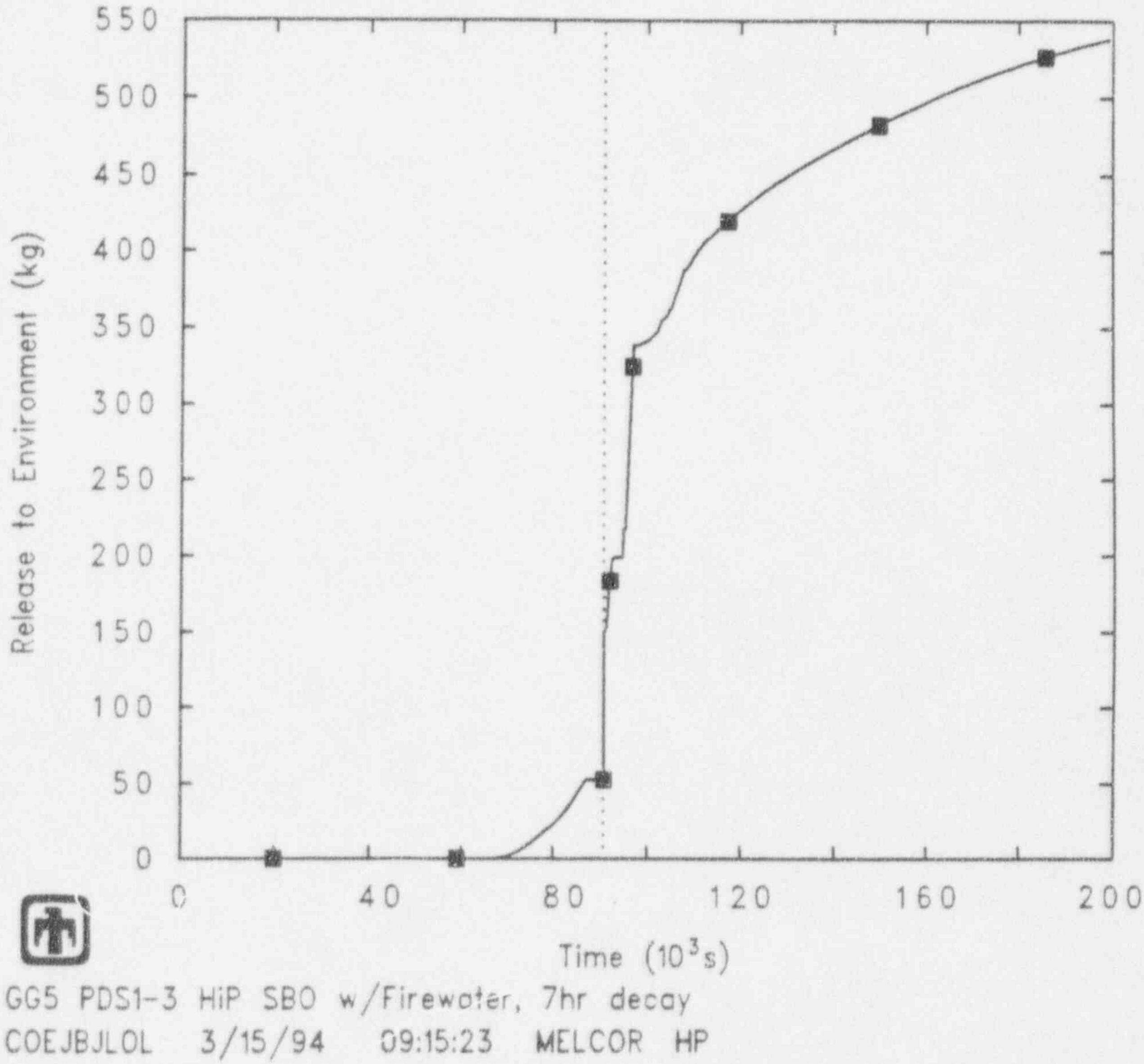


Figure 4.3.3.11. Total Environmental Radionuclide Releases for Grand Gulf POS 5 - Station Blackout with Firewater Addition Followed by High Pressure Boiloff, Initiated 7 hr After Shutdown

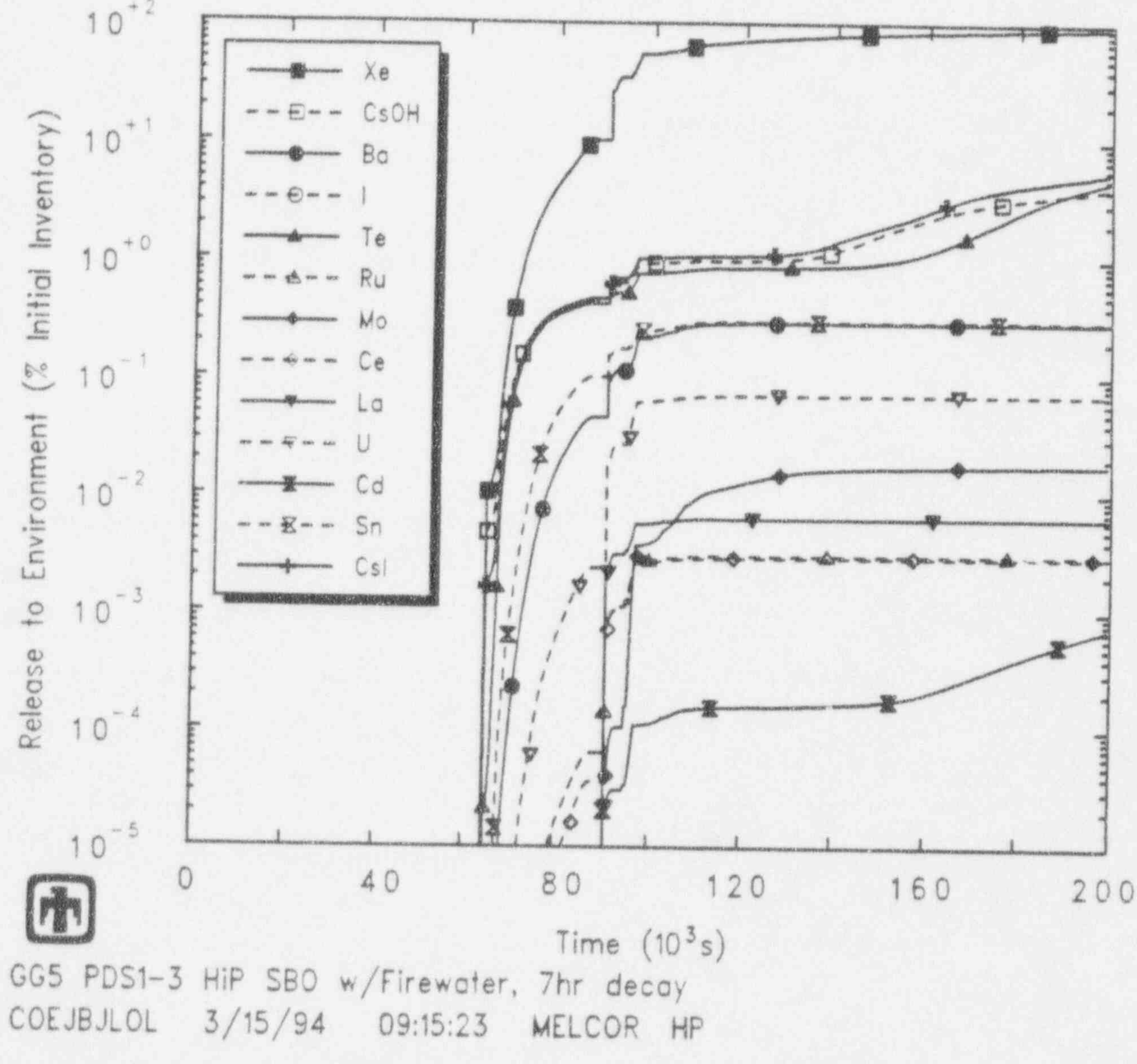


Figure 4.3.3.12. Environmental Radionuclide Release Fractions for Grand Gulf POS 5 - Station Blackout with Firewater Addition Followed by High Pressure Boiloff, Initiated at 7 hr After Shutdown

core or the cavity, and of how much of the released radionuclides are retained in the primary system *vs* how much of the released radionuclides are released to, or released in, either the containment or the auxiliary building and the environment, all normalized to the initial inventories of each class. Table 4.3.3.4 presents a different breakdown of the released radionuclide final distribution, giving the fractions of released inventory for each class in control volume atmospheres (including the environment), in pools, or deposited or settled onto heat structures at the end of the calculations. (As in Table 4.3.3.2, these amounts consider only the release of radioactive forms of these classes, and not additional releases of nonradioactive aerosols from structural materials.)

These tables show fission product distributions somewhat different than those found for any of the other sequences analyzed, for the radionuclides with significant ($\geq$80% of initial inventory) release from fuel. As in all the accident scenarios analyzed, most of the noble gases released are in the environment, in the atmosphere. Significant fractions of the volatile species (CsOH, CsI and Te) released are retained everywhere, in the primary system (15-35%), containment (40-50%), and auxiliary building (20-25%); about 5% of the total initial inventories of these volatiles is released to the environment in this case, an environmental release similar to that for the other station blackout sequence analyzed, with failure to isolate SDC and no firewater addition (discussed in the previous section). The two classes of radionuclides forming aerosols which had substantial in-vessel releases (Ba and Sn) also were predicted to have substantial fractions retained everywhere, slightly more in the primary system (35-45%), about the same in containment (40%), and significantly less in the auxiliary building (2-2.5%).

4.3.4 Low Pressure Boiloff with Flooded Containment, Initiated 7 hr and 24 hr After Shutdown

At the initiation of the accident, the reactor vessel is depressurized. Following the initiating event, two SRVs are opened. For this scenario, the vessel and containment are flooded, *i.e.*, the vessel water level is at the steam lines, 16.46 m or 648 in, and the containment (suppression pool, pedestal cavity and drywell) is flooded up to the lower personnel lock, 9.65 m or 31.67 ft above the suppression pool floor. The vessel water inventory is at 300.5 K (80°F), as is the suppression and containment water; the containment is at 305.4 K (90°F). Since the lower personnel lock is open, the auxiliary building is flooded which results in the loss of all core cooling. The reactor pressure vessel head vent is closed at the beginning of the transient. Since both the drywell and the containment hatches are open, the drywell is open to the containment and the containment is open to the auxiliary building (*i.e.*, "open containment").

This sequence is almost identical to the low-pressure boiloff scenario discussed in Section 4.2.2, except that in those Level 1 analyses the containment was dry while in these Level 2 analyses the containment was assumed to be flooded.

The sequence of events predicted by MELCOR for this accident with different initiation times is given in Table 4.3.4.1.

Table 4.3.3.3. Final Radionuclide Distribution for Grand Gulf POS 5 – Station Blackout with Firewater Addition Followed by High Pressure Boiloff. Initiated at 7 hr After Shutdown

Class	Fission Product Distribution (% Initial Inventory)				
	Fuel Debris	Primary System	Containment	Auxiliary Building	Environment
Xe	~0	0.007	2.49	5.00	92.5
CsOH	~0	33.6	42.6	19.68	4.11
Ba	20.4	33.1	43.9	2.34	0.294
Te	0.009	34.1	42.1	19.08	4.70
Ru	99.1	0.307	0.57	0.0125	0.00305
Mo	98.6	0.095	1.04	0.19	0.018
Ce	99.2	0.282	0.54	0.0121	0.0029
La	99.8	0.02	0.144	0.022	0.0063
U	79.6	7.13	12.95	0.303	0.0654
Cd	~100	0.0095	0.062	0.003	0.0007
Sn	11.1	45.0	41.02	2.67	0.303
CsI	~0	16.3	53.42	24.72	5.54

Table 4.3.3.4. Final Radionuclide State for Grand Gulf POS 5 - Station Blackout with Firewater Addition Followed by High Pressure Boiloff. Initiated 7 hr After Shutdown

Class	Fission Products Released from Fuel (% Released Inventory)		
	Atmosphere	Pool	Deposited
Xe	~100	0	0
CsOH	13.26	42.9	43.9
Ba	0.37	24.8	74.8
I	~100	0	0
Te	17.04	39.6	43.3
Ru	0.34	25.7	74.1
Mo	1.35	46.4	52.3
Ce	0.36	25.7	73.9
La	0.33	49.7	47.0
U	0.32	25.5	74.2
Cd	15.66	37.8	46.5
Sn	0.35	22.1	77.6
CsI	21.78	48.9	29.3

Table 4.3.4.1. Sequence of Events Predicted by MELCOR for Low-Pressure Boiloff with Flooded Containment, Initiated 7 hr and 24 hr After Shutdown

Event	Time After Shutdown 7 hr	24 hr
Accident initiation	0.0	0.0
Core uncovery (TAF) begins	10,262 s (2.85 hr)	14,339 s (3.98 hr)
Core heatup begins	22,000 s (6.11 hr)	28,500 s (7.92 hr)
Clad failure/Gap release		
(Ring 1)	27,154 s (7.54 hr)	36,361 s (10.10 hr)
(Ring 2)	27,055 s (7.52 hr)	36,260 s (10.07 hr)
(Ring 3)	27,167 s (7.55 hr)	36,383 s (10.11 hr)
(Ring 4)	27,723 s (7.70 hr)	36,963 s (10.27 hr)
(Ring 5)	29,374 s (8.16 hr)	38,565 s (10.71 hr)
(Ring 6)	32,139 s (9.48 hr)	42,863 s (11.91 hr)
Core plate failed		
(Ring 1)	89,990 s (25.00 hr)	112,516 s (31.25 hr)
(Ring 2)	89,164 s (24.77 hr)	111,475 s (30.97 hr)
(Ring 3)	88,949 s (24.71 hr)	112,350 s (31.21 hr)
(Ring 4)	88,000 s (24.44 hr)	112,785 s (31.33 hr)
(Ring 5)	83,548 s (23.21 hr)	110,645 s (30.73 hr)
(Ring 6)	82,308 s (22.86 hr)	109,936 s (30.54 hr)
Vessel LH penetration failed		
(Ring 1)	82,534 s (22.93 hr)	110,098 s (30.58 hr)
(Ring 2)	82,446 s (22.90 hr)	110,065 s (30.57 hr)
(Ring 3)	82,421 s (22.89 hr)	110,047 s (30.57 hr)
(Ring 4)	82,406 s (22.89 hr)	110,034 s (30.57 hr)
(Ring 5)	82,397 s (22.89 hr)	110,025 s (30.56 hr)
(Ring 6)	82,410 s (22.89 hr)	110,302 s (30.64 hr)
Commence debris ejection	82,397 s (22.89 hr)	110,025 s (30.56 hr)
Auxiliary building failed	99,000 s (27.50 hr)	120,000 s (33.33 hr)
Cavity rupture		
End of calculation	400,000 s (111.1 hr)	400,000 s (111.1 hr)

Figure 4.3.4.1 gives the vessel pressures calculated starting this accident scenario at two different times after scram. In both cases, the system begins pressurizing as all core cooling is lost but only pressurizes slightly before the steam flow out the two open SRV's is sufficient to remove all the decay heat; the higher the decay heat (*i.e.*, the sooner after shutdown), the higher the early-time pressure peak before the flow out the open SRV's can fully remove the decay heat. The steam flow out the two open SRV's in turn pressurizes the containment and, through the open equipment hatch and personnel locks, pressurizes the auxiliary building, as shown in Figure 4.3.4.2. The longer after shutdown and scram that this accident sequence begins, the lower the decay heat and the longer it takes to fail the auxiliary building.

The coolant inventory in the vessel drops as the decay heat boils water to steam which is lost out the open SRVs, faster for the higher decay heat level than for the lower decay heat, as presented in Figure 4.3.4.3.

Figure 4.3.4.4 presents the upper plenum, core and lower plenum swollen and collapsed liquid levels for this accident sequence initiated at two different times after scram. The upper plenum collapsed level initially falls but the two-phase level rises as the primary system pressurizes. There is considerable pool frothing and swelling in both the upper plenum and core volumes and the vessel inventory is boiled away. Both the initial, more rapid level drop in the core and upper plenum and the later, gradual lower plenum uncovery due to continued boiloff is faster for higher decay heat levels than for lower decay heat levels. The lower plenum levels still show substantial amounts of liquid remaining at vessel failure, when that water is either flashed to steam by the falling core debris or drains into the cavity through the failed lower head penetrations.

The heatup of the intact fuel and clad is illustrated in Figures 4.3.4.5 and 4.3.4.6, as calculated for scenarios initiated at 7 hr and 24 hr after shutdown, respectively. Core uncovery and heatup begins sooner and proceeds more rapidly at the higher decay heat level resulting from beginning this accident 7 hr after scram than for a lower decay heat in the same accident initiated 24 hr after scram, as would be expected. The fuel/clad component temperatures in MELCOR are set to zero in a cell when that component fails, so these figures show both the overall heatup rate and the time that the intact fuel/clad component fails through melting of the clad. The intact fuel/clad component temperatures reach a peak of $\geq$2000 K ($\geq$3140°F) since the component generally fails at the zircaloy clad melt temperature, taken as 2098 K (3317°F) in MELCOR.

Figures 4.3.4.7 and 4.3.4.8 present corresponding core debris temperatures in the active fuel region calculated for scenarios initiated at 7 hr and 24 hr after shutdown, respectively; these are t[illegible]mperatures of the debris bed formed by the failure of the intact fuel/clad componen[illegible] MELCOR in a core cell, whose (intact) temperatures were given in Figures 4.3.4.5 and 4.3.4.6. The debris bed in the active fuel region reaches peak temperatures $\geq$3500 K (5840°F), significantly above the UO_2 melt temperature of 3113 K (5144°F), in the middle and upper active fuel regions; in the lower active fuel levels the debris bed temperatures remain below the UO_2 melt temperature. The debris bed temperatures reached in the active fuel region are visibly higher for the accident initiated at a higher decay heat level than at the lower decay heat level.

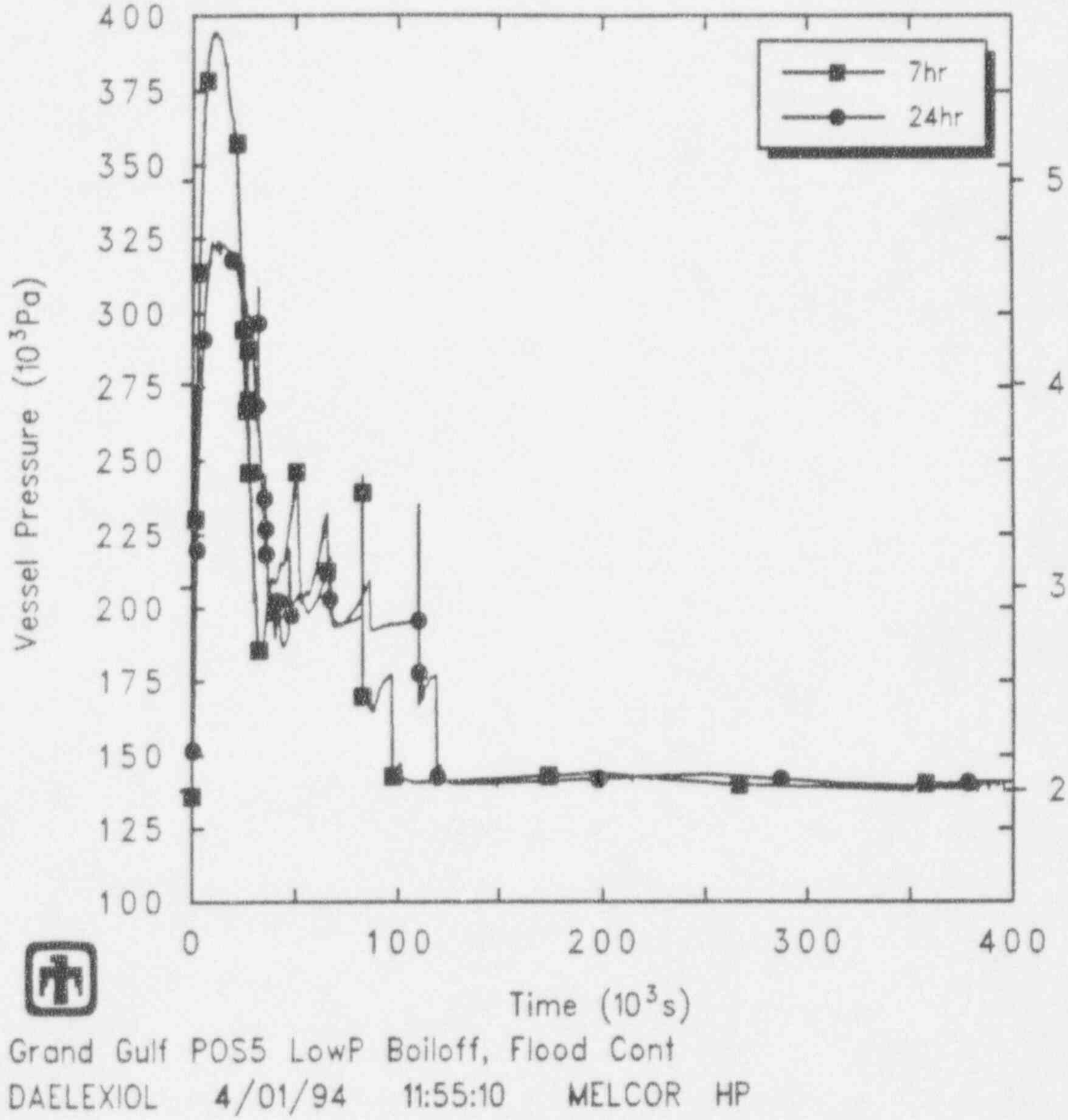


Figure 4.3.4.1. Reactor Vessel Pressures for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated at Various Times After Shutdown

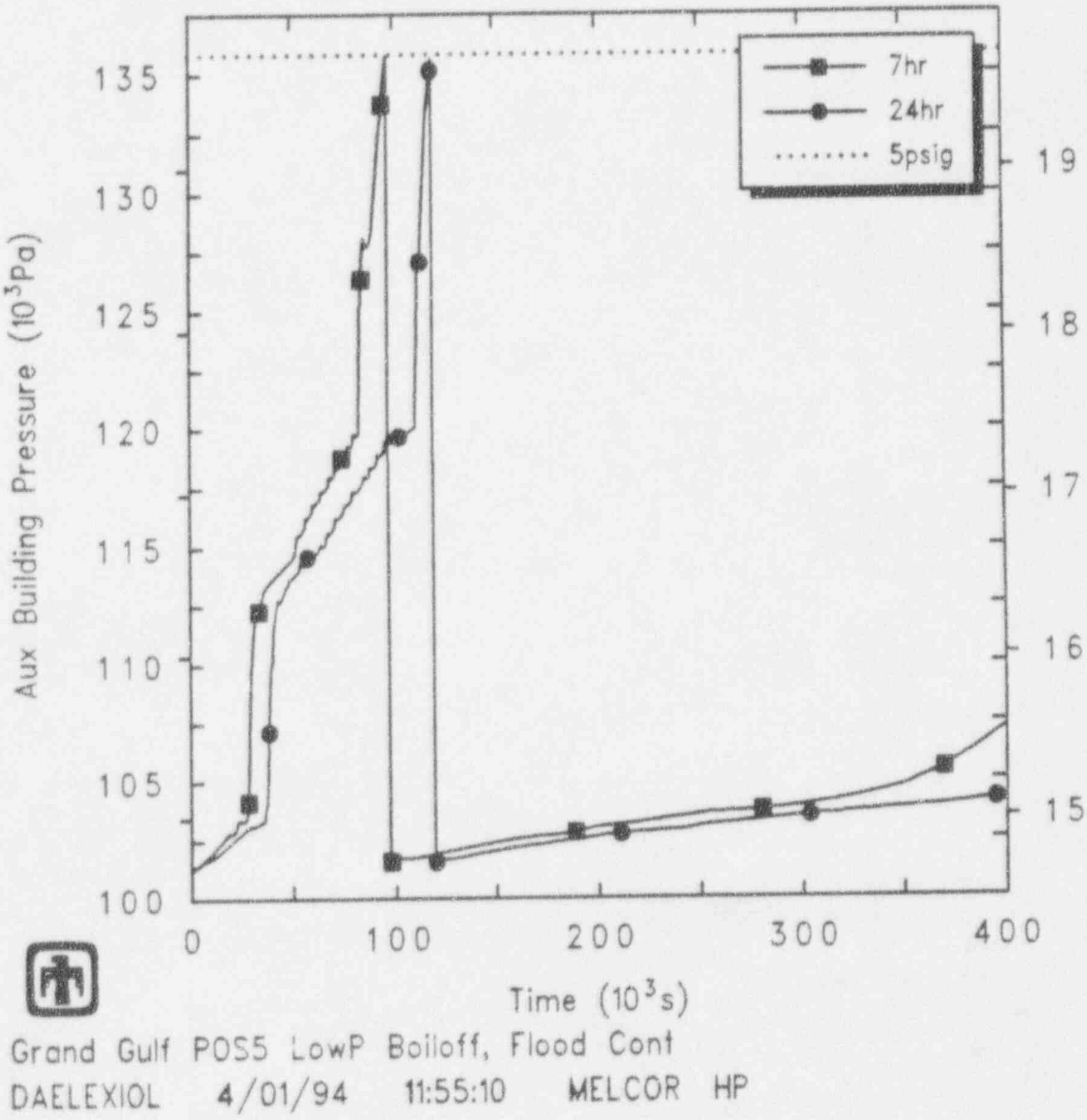


Figure 4.3.4.2. Auxiliary Building Pressures for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated at Various Times After Shutdown

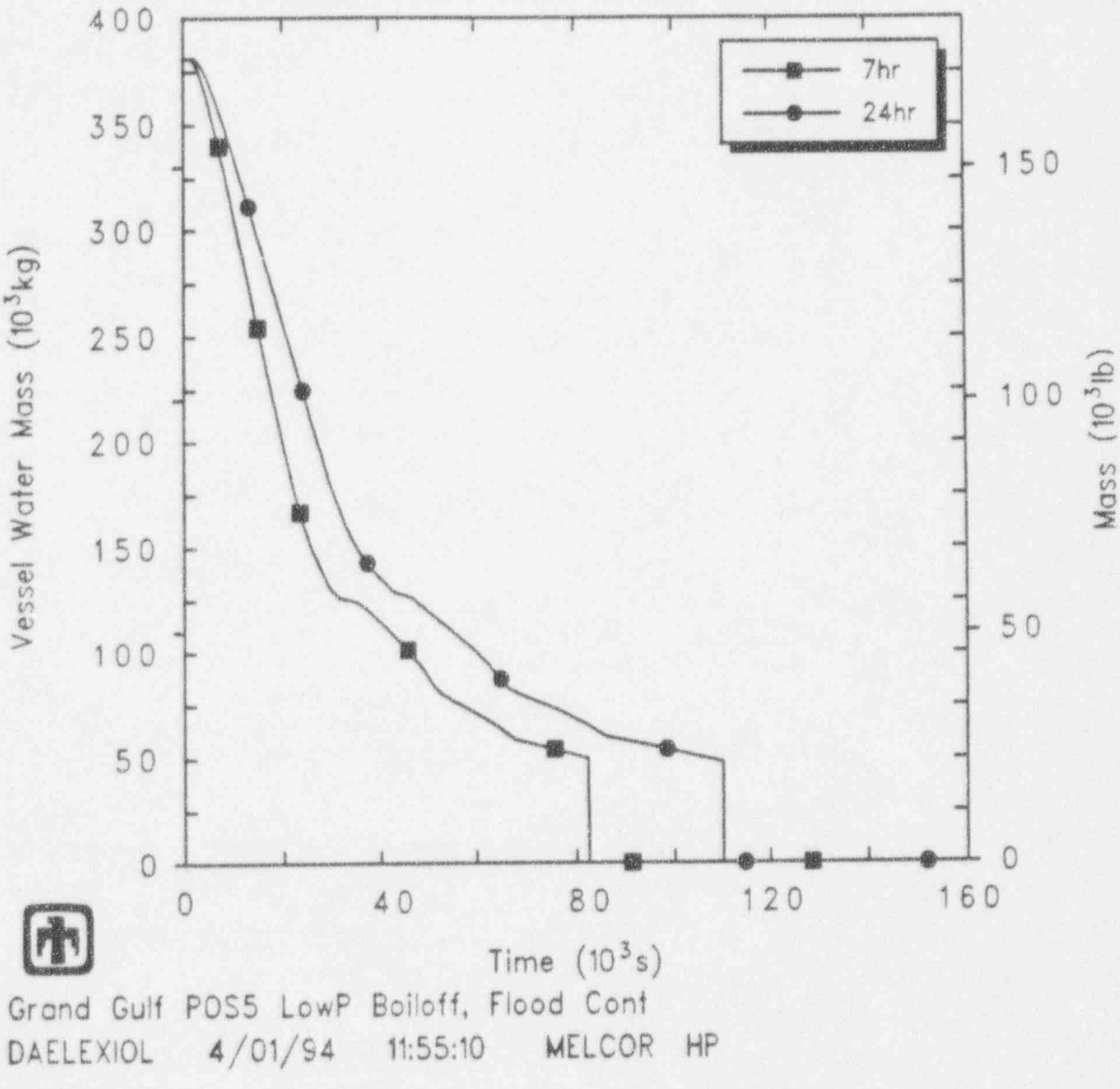


Figure 4.3.4.3. Reactor Vessel Water Masses for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated at Various Times After Shutdown

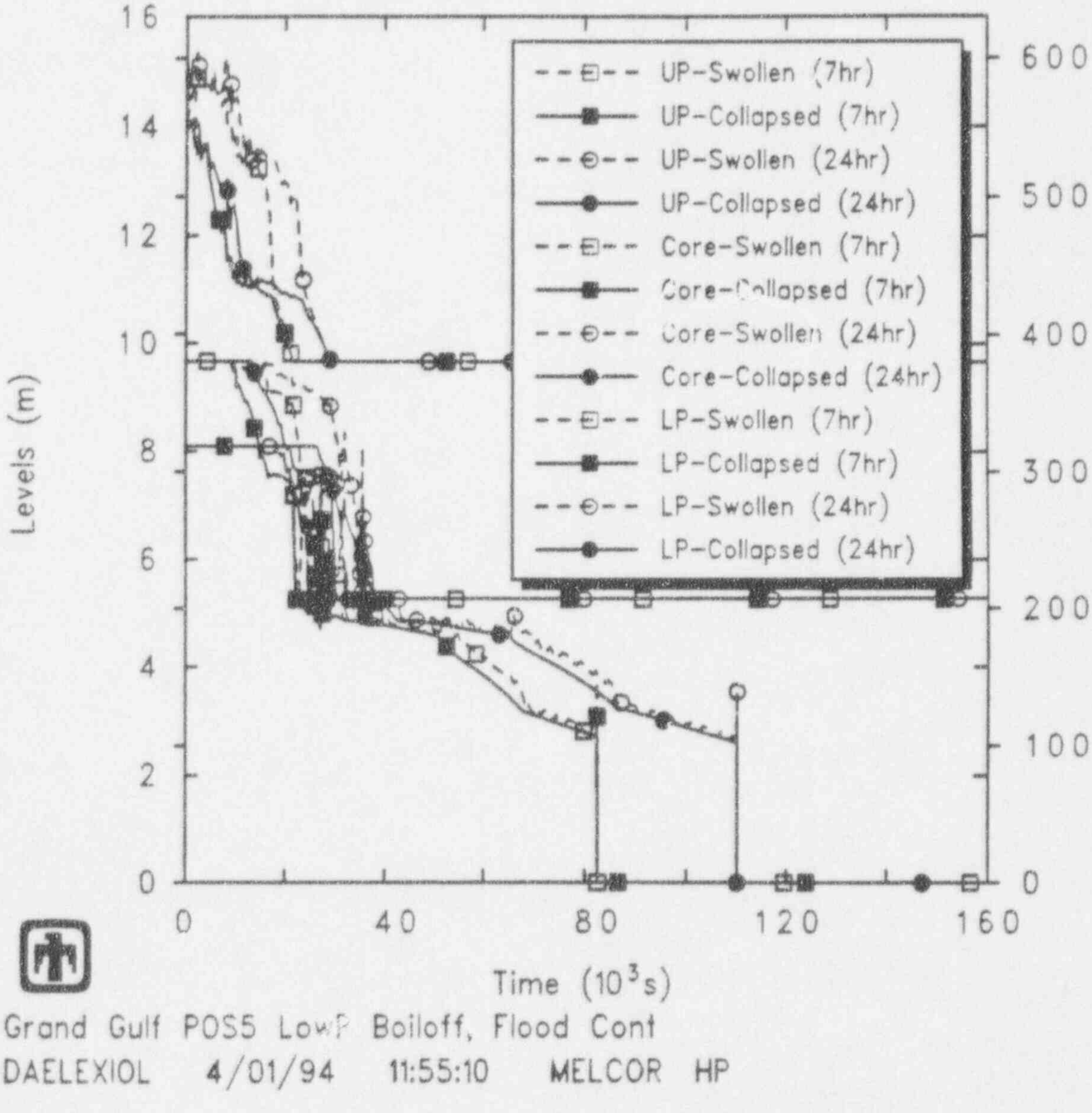


Figure 4.3.4.4. Upper Plenum, Core and Lower Plenum Liquid Levels for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated at Various Times After Shutdown

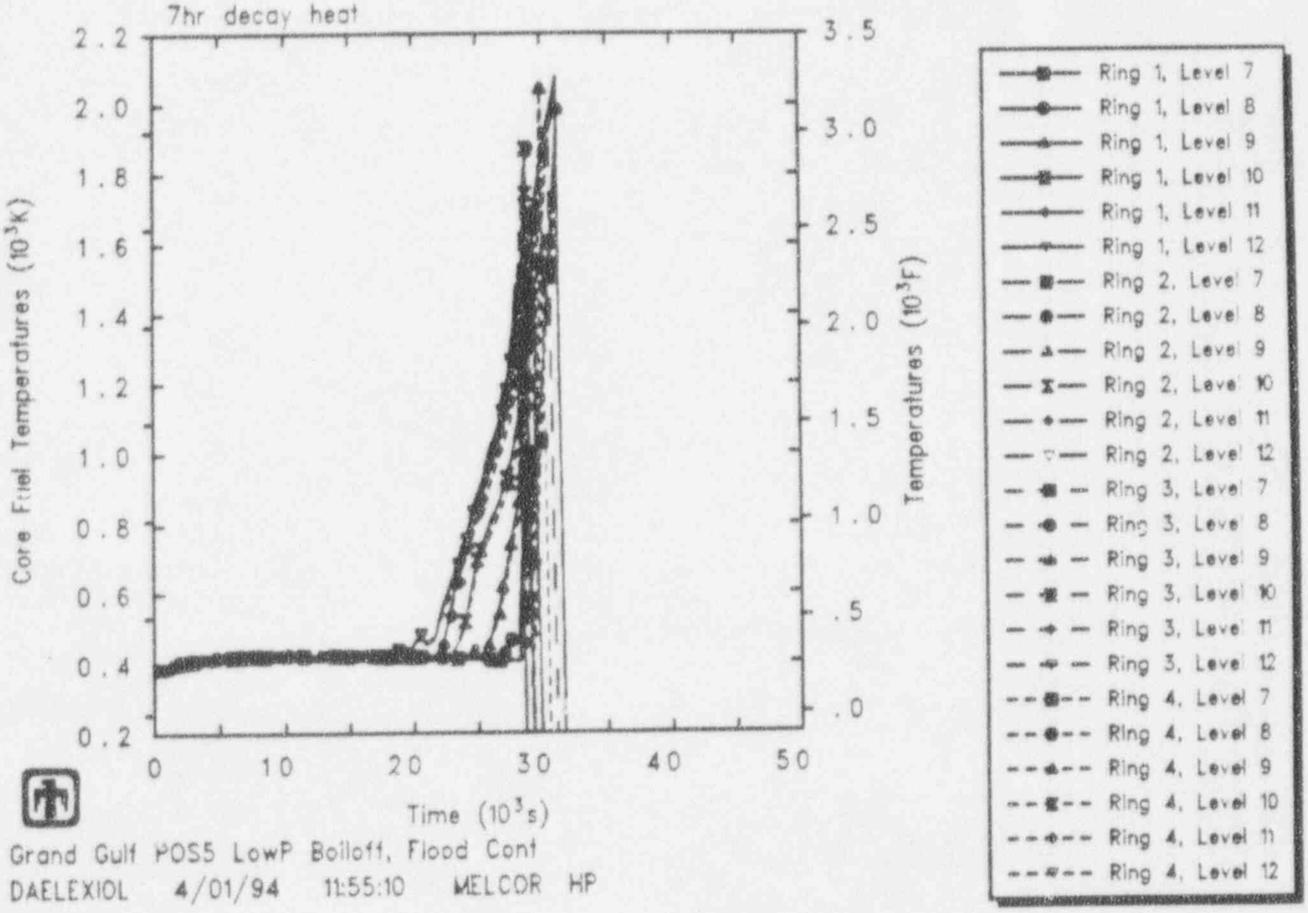


Figure 4.3.4.5. Core Intact Fuel/Clad Temperatures for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated 7 hr After Shutdown

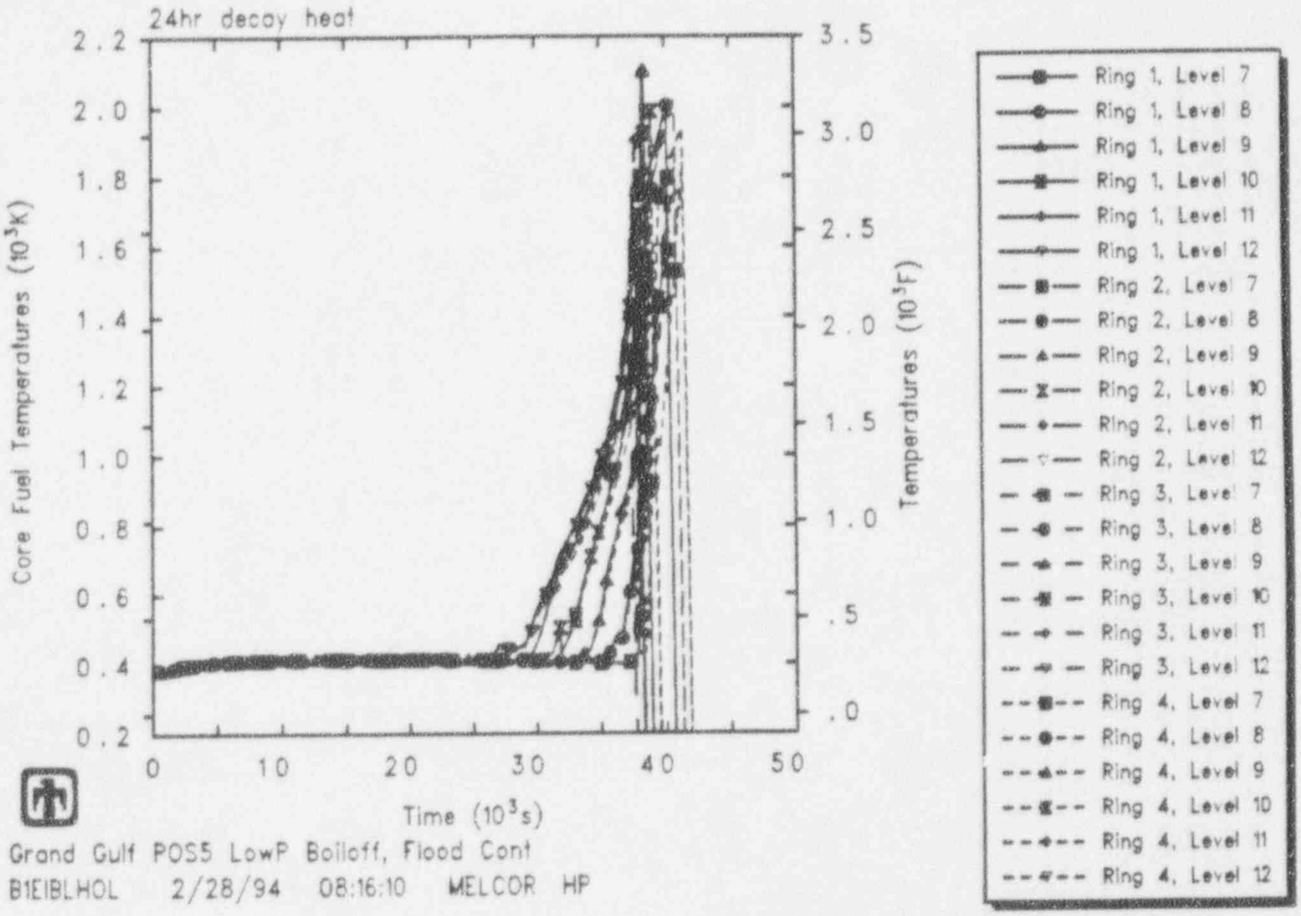


Figure 4.3.4.6. Core Intact Fuel/Clad Temperatures for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated 24 hr After Shutdown

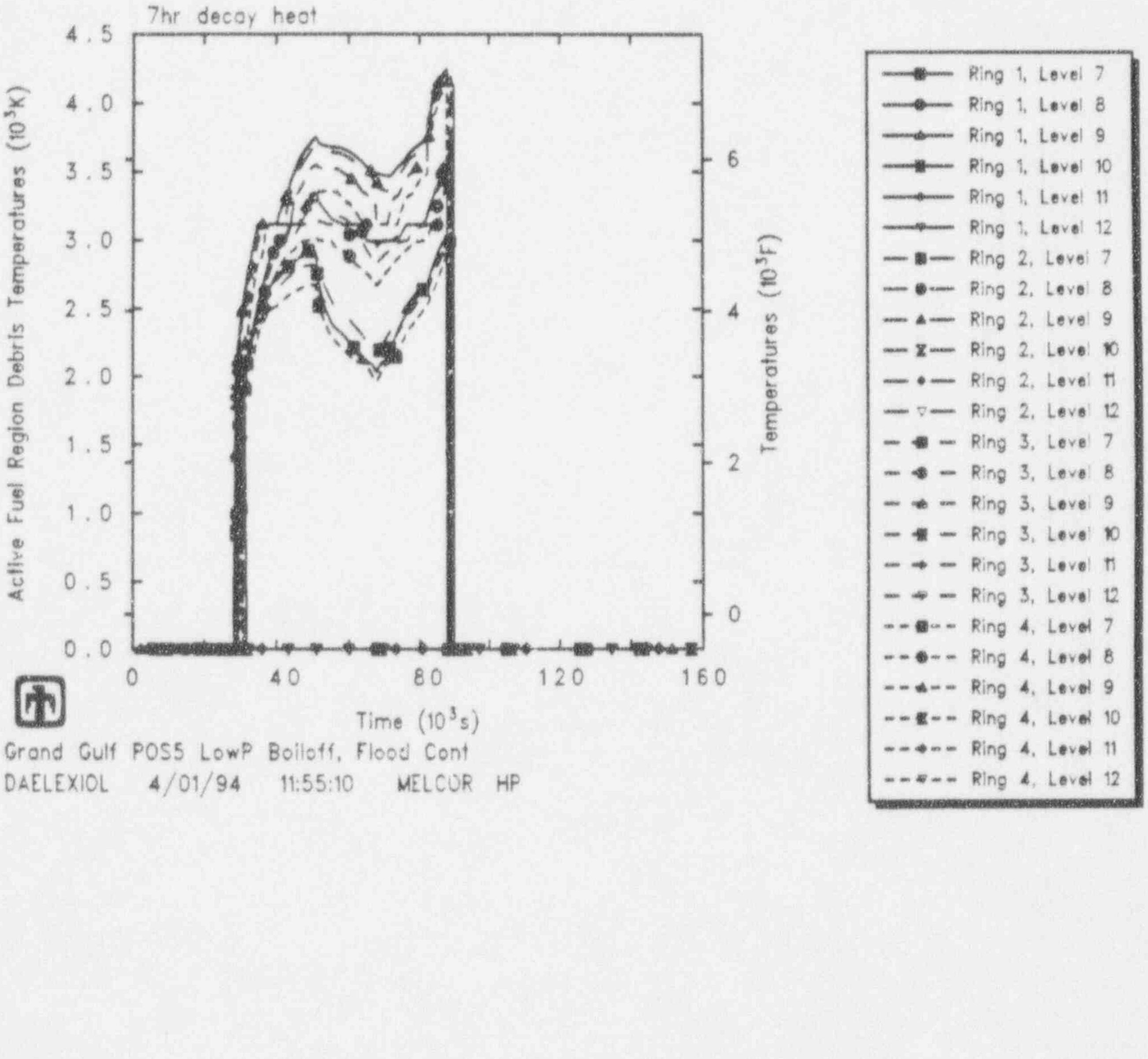


Figure 4.3.4.7. Core Active Fuel Region Debris Bed Temperatures for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated 7 hr After Shutdown

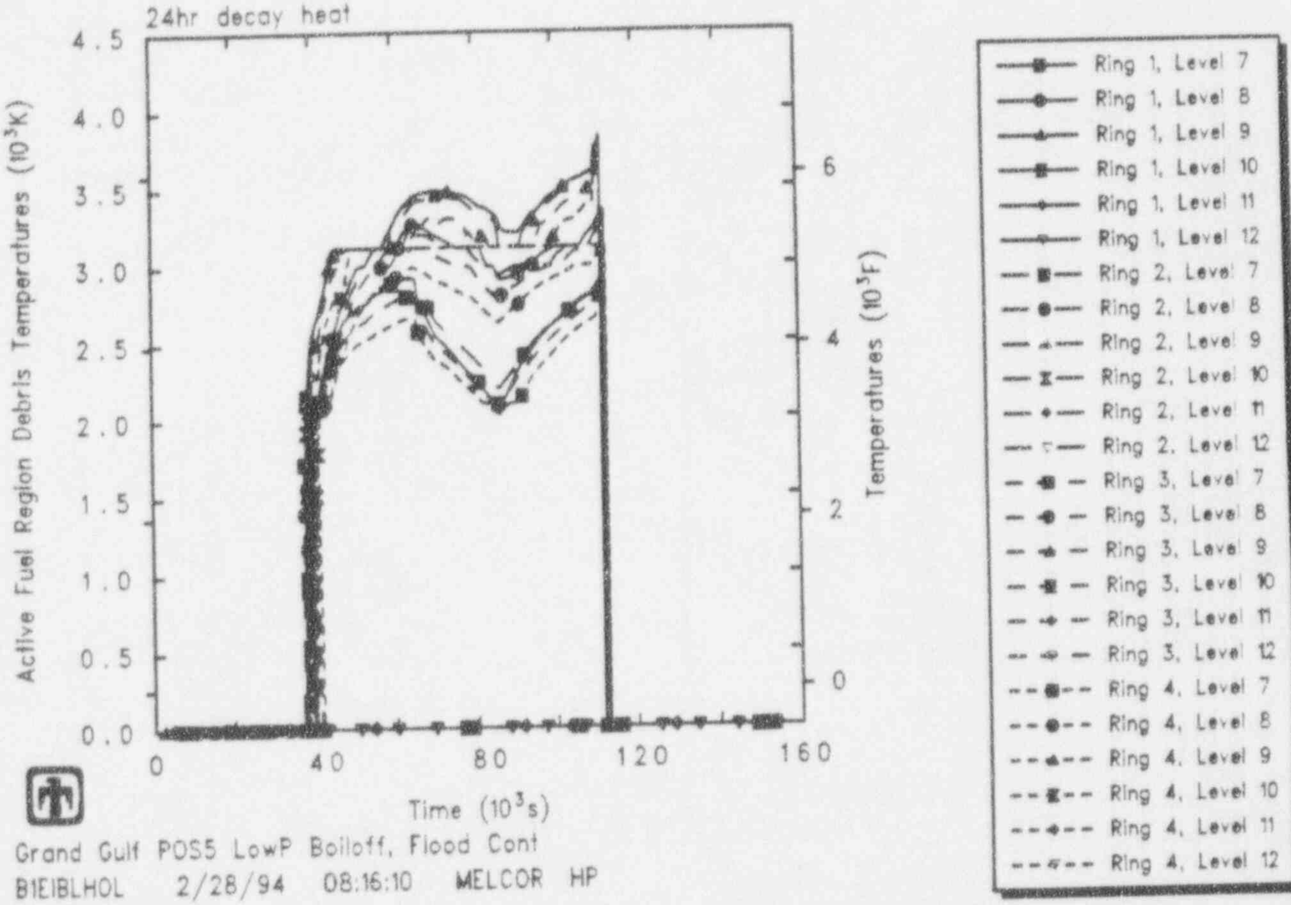


Figure 4.3.4.8. Core Active Fuel Region Debris Bed Temperatures for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated 24 hr After Shutdown

The temperatures of the active fuel region debris bed drop to zero when the core plate fails and the debris relocates to the lower plenum. The predicted temperatures in the debris bed in the lower plenum and core plate are given in Figures 4.3.4.9 and 4.3.4.10. for scenarios initiated at 7 hr and 24 hr after shutdown, respectively. In both cases, prior to core plate failure there is some cold, refrozen debris both on the core support plate and on the lower core structural material just above the core support plate; the cooling and refreezing of this debris is the cause of the continued gradual drop in lower plenum liquid level due to steaming seen in Figure 4.3.4.4. The debris temperature rises gradually to the core support plate failure temperature of 1273 K (1832°F). After core plate failure hot, high-temperature debris begins appearing the the lower plenum as debris falls from the active fuel region into the lower plenum. With the new debris radial relocation model added in MELCOR 1.8.2, the core plate needs to fail in only one ring before debris from cells in the active fuel region in all radial rings can flow sideways and down, fall through the failed plate, and then spread sideways into cells in the lower plenum in all radial rings. (Thus a lower head penetration can now fail in a ring before the core plate in that ring fails.) The lower head penetrations begin failing almost immediately, and the lower plenum debris temperatures begin dropping to zero as debris is ejected from the vessel to the cavity. Some cool, quenched debris can remain present in the lower plenum for a significant period of time, however, as indicated by the 1000-1250 K debris temperatures in the lowest level after vessel failure in the low pressure boiloff scenario initiated 24 hr after shutdown.

Figures 4.3.4.11 and 4.3.4.12 indicate what fraction of each material in the active fuel region has collapsed into a debris rubble bed held up by the core support plate, prior to core plate failure, debris relocation, lower head failure and debris ejection, for this low pressure boiloff with flooded containment initiated at 7 hr and 24 hr after shutdown, respectively. The fractions of each material and the overall fraction of total material in the active fuel region degraded into particulate debris and are similar in the two calculations. The majority of the debris bed is formed within about 1 hr, and the fractions of material collapsed from the intact geometry to a debris bed then remain very nearly constant for many hours, until vessel failure.

Figure 4.3.4.13 shows the total masses of core materials (UO_2, Zircaloy and ZrO_2, stainless steel and steel oxide, and control rod poison) remaining in the vessel. This includes both material in the active fuel region and in the lower plenum. Debris ejection began very soon after lower head failure. This figure illustrates that most of the core material was lost from the vessel to the cavity quickly, in step-like stages. In all cases, all of the UO_2 was transferred to the cavity within ~1 hr after vessel failure, as was the unoxidized zircaloy, the associated zirc oxide and the control rod poison. A small fraction (15%) of the structural steel in the lower plenum, and some associated steel oxide, was predicted to remain unmelted and in place in the low pressure boiloff scenario initiated 24 hr after shutdown.

Figure 4.3.4.14 presents the amounts of core debris, concrete ablated and the total debris mass (*i.e.*, core debris combined with concrete ablation products) in the cavity. There is a timing shift due to the slower core degradation and later vessel failure at the

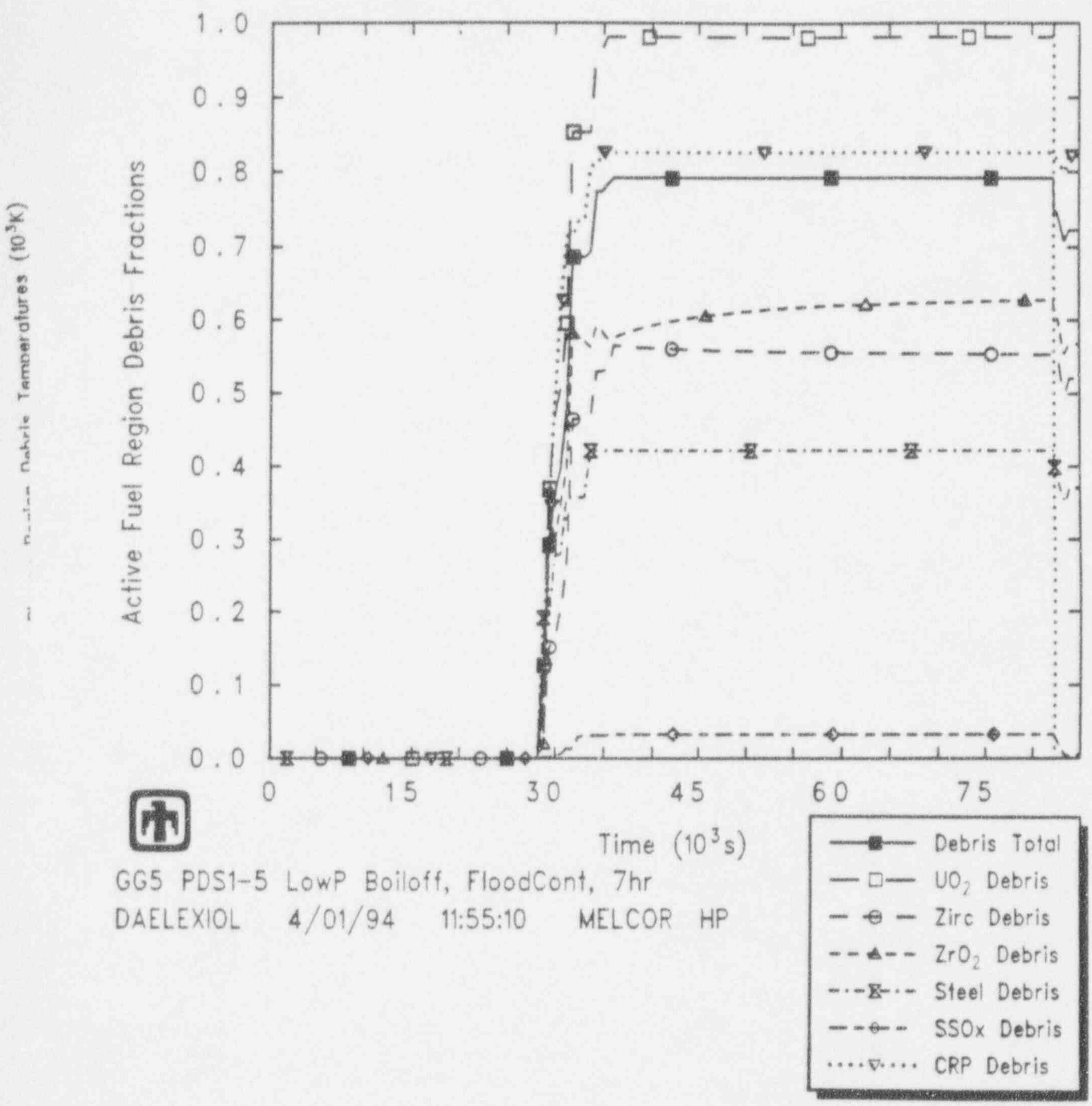


Figure 4.3.4.11. Core Active Fuel Region Degraded Material Fractions for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated 7 hr After Shutdown

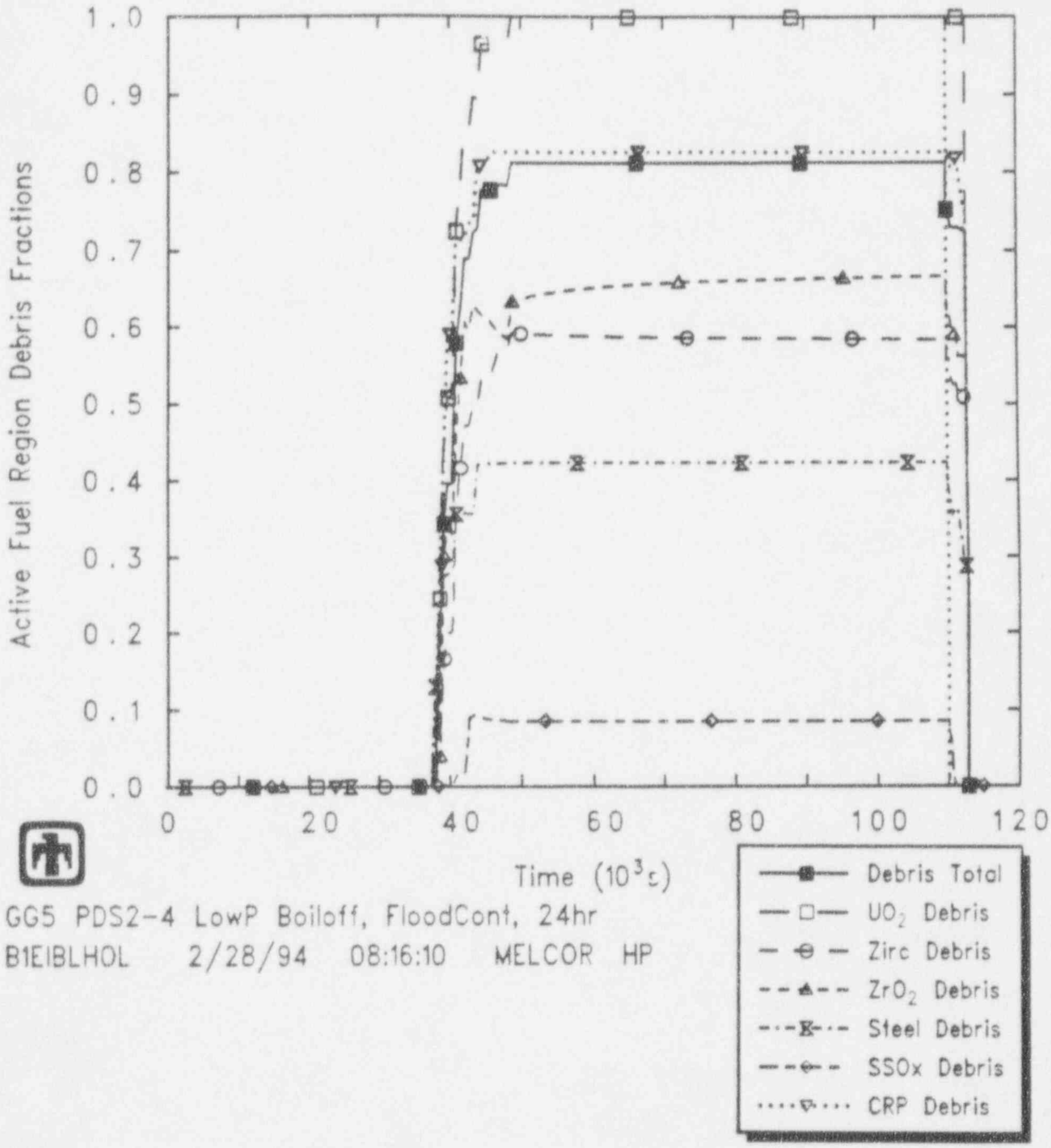


Figure 4.3.4.12. Core Active Fuel Region Degraded Material Fractions for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated 24 hr After Shutdown

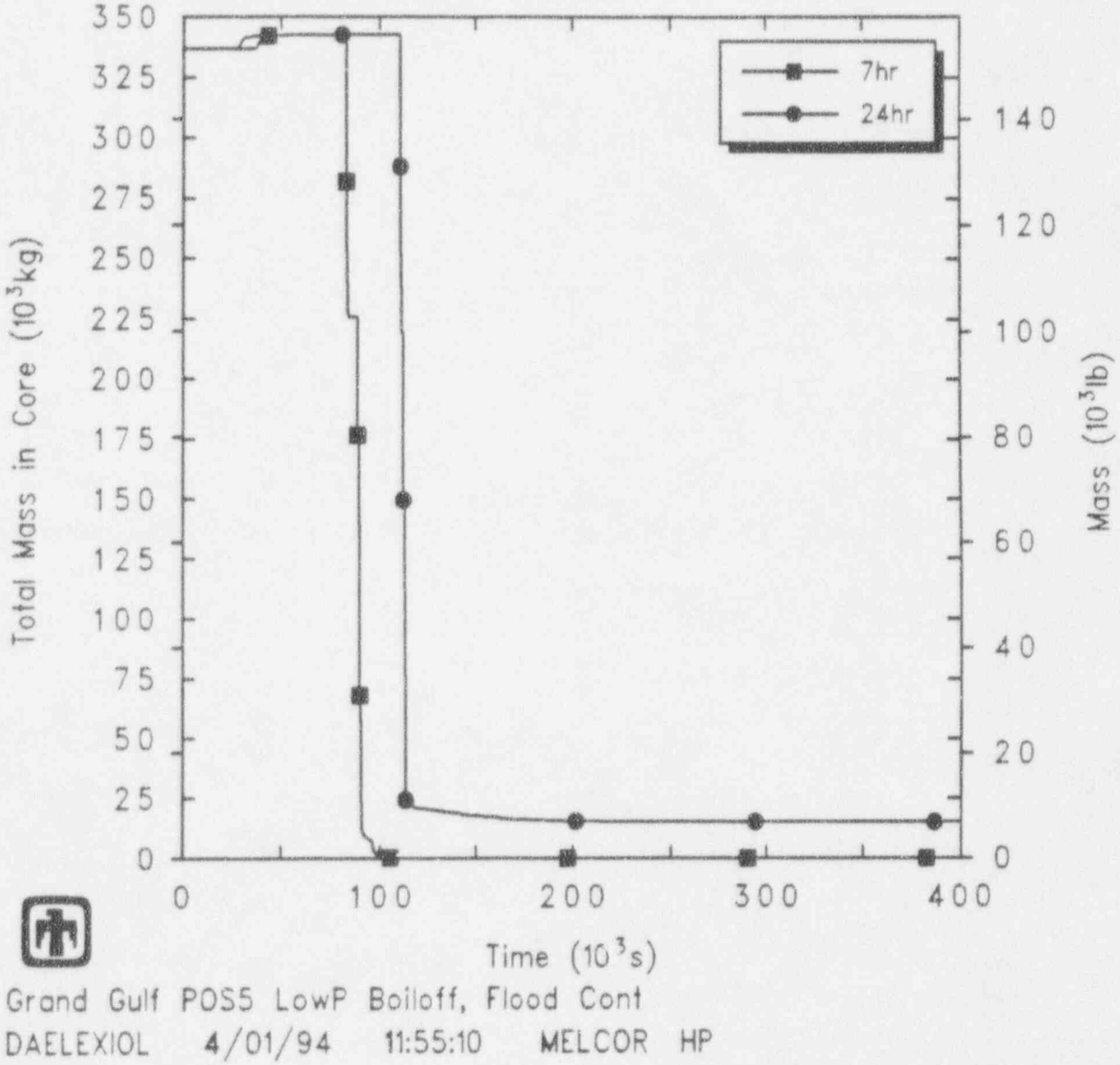


Figure 4.3.4.13. Total Core Material Masses for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated at Various Times After Shutdown

lower decay heat. Also, since almost all the material in the core active fuel region and lower plenum is ejected in this sequence initiated 7 hr after shutdown while some fraction of the lower plenum structural steel remains unmelted and in place in the same scenario initiated 24 hr after shutdown, the core debris mass in the cavity is slightly greater in the calculation initiated 7 hr after scram. However, the mass of concrete ablated and the total cavity debris mass are generally similar for this sequence initiated at two different times after scram. As in all our MELCOR analyses, concrete ablation is quite rapid soon after debris ejection (while the core debris is hot, >2000 K, and consists of a layer of metallic debris above a heavy oxide layer), and concrete ablation slows significantly after a short time (after enough concrete has been ablated for the debris bed configuration to invert to a light oxide layer above a layer of metallic debris, mixed to a lower average temperature of ~1500 K).

The calculated production of steam and noncondensable gases (H_2, CO, CO_2 and H_2O) is summarized in Figure 4.3.4.15. The hydrogen production shown includes both in-vessel production (the initial step increase) and ex-vessel production in the cavity (the later-time increase). The in-vessel hydrogen generation corresponds to the oxidation of about 10% of the zircaloy and about 1% of the steel in the core and lower plenum, prior to vessel failure and debris ejection. As soon as the core debris enters the cavity, core-concrete interaction begins, resulting in the production of carbon dioxide and hydrogen; reduction of these gases by the molten metal in the core debris also gives rise to carbon monoxide and hydrogen. The production rate of noncondensables from core-concrete interaction resembles the concrete ablation rate: quite rapid soon after debris ejection, later slowing after a CORCON "layer flip" has occurred. On a molar basis, much less CO_2 and steam are produced than H_2 and CO. More CO_2 and steam are calculated to be produced in this sequence initiated 24 hr after scram than initiated 7 hr after scram; this is a result of the reduced metal content in the core debris in the case initiated 24 hr after shutdown, due to the retention of some structural steel in the lower plenum.

The resulting mole fractions in the drywell, containment dome and auxiliary building (first and second floors) are presented in Figures 4.3.4.16 and 4.3.4.17 for this sequence initiated at two different times after shutdown, including vertical dotted lines at TAF uncovery and at vessel failure for reference. The mole fractions in the cavity resemble the behavior shown for the drywell; the mole fractions in the containment equipment hatch are very similar to those shown for the containment dome; and the mole fractions in the upper floors of the auxiliary building generally resemble the behavior shown for the second floor of the auxiliary building. The inner containment atmosphere consists mostly of steam, building up from accident initiation since the SRVs are locked open, decreasing somewhat after vessel failure and noncondensable gas generation due to core-concrete interaction, then increasing again throughout the remainder of the transient period simulated. The outer containment steam concentration remains generally low as steam condenses in the flooded containment until after vessel failure, when the core debris fallen into the cavity begins boiling the water flooding the containment in this scenario. The containment is open to the auxiliary building in the second and fourth floors. The atmosphere in the dead-end first floor of the auxiliary building remains near ambient with

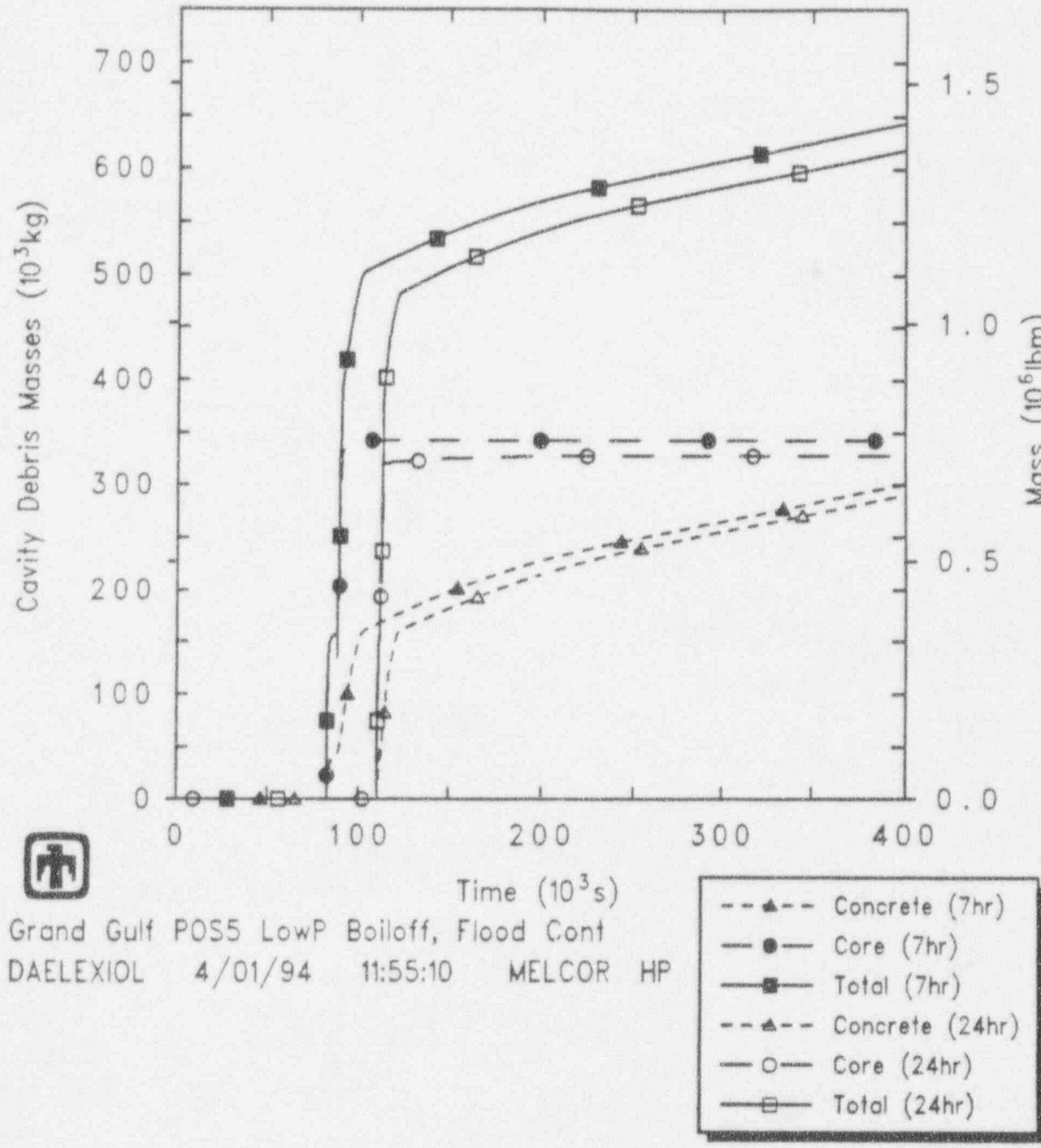


Figure 4.3.4.14. Cavity Total and Concrete Debris Masses for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated at Various Times After Shutdown

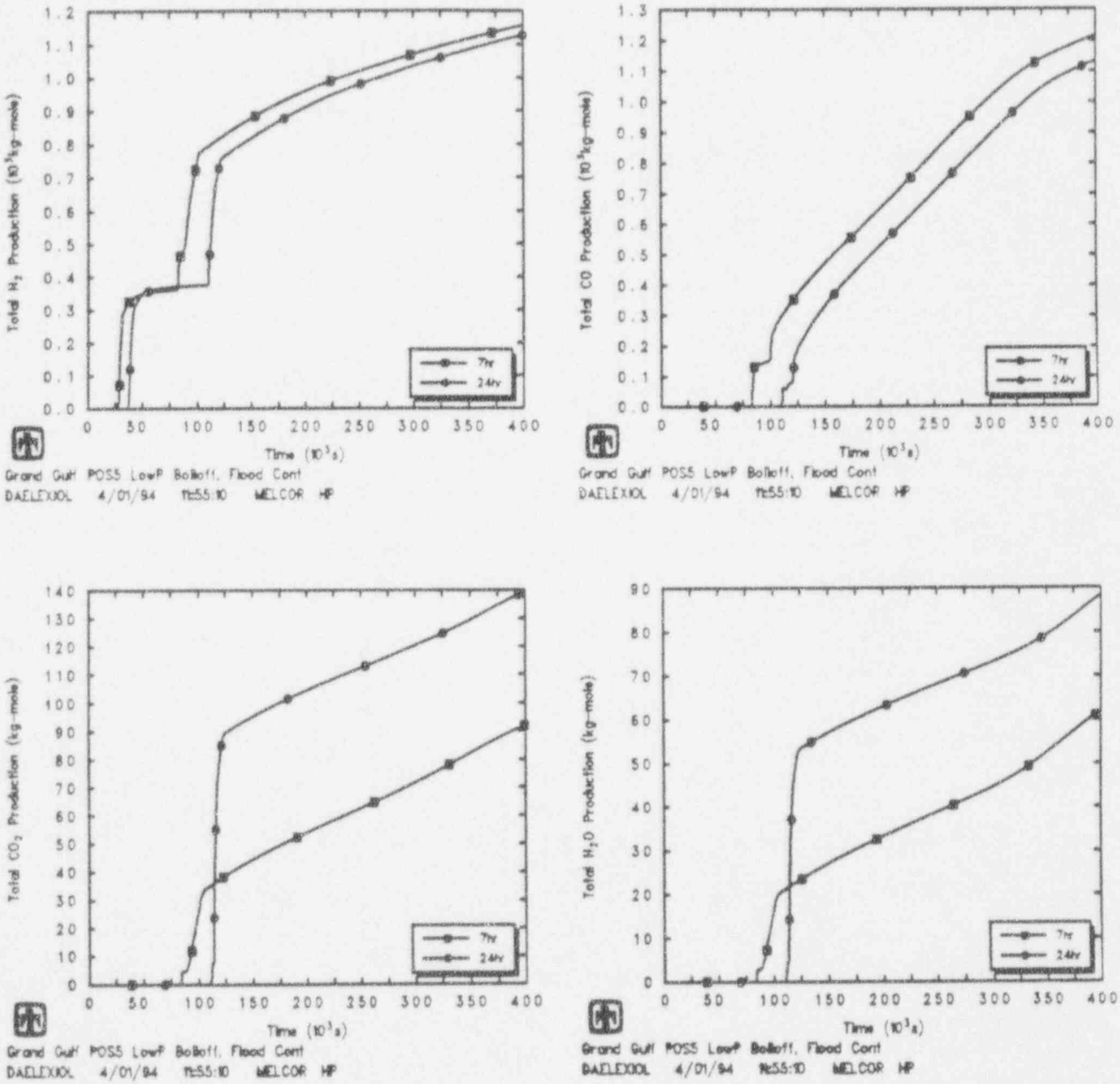


Figure 4.3.4.15. Hydrogen (upper left), Carbon Monoxide (upper right), Carbon Dioxide (lower left) and Steam (lower right) Generation for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated at Various Times After Shutdown

small fractions of steam and noncondensables from the upper floors; higher in the auxiliary building the atmosphere composition closely resembles that in the outer containment (because the containment equipment hatch and both of the containment personnel locks are open). The behavior is very similar in the calculations for this sequence initiated at two different times after shutdown, just shifted in time.

Figures 4.3.4.18 and 4.3.4.19 illustrate the time-dependent release of radionuclides from the fuel debris both within the vessel and in the cavity, for cases initiated 7 hr and 24 hr after scram, respectively. The vertical dotted lines within the plots mark the time of vessel failure, indicating that most of the in-vessel release occurs prior to vessel failure, from the hot debris bed in the active fuel region, while most of the ex-vessel release occurs within a short time period after vessel failure and debris ejection to the cavity, while the core debris is still hot, >2000 K, and consists of a layer of metallic debris above a heavy oxide layer, before enough concrete has been ablated for the debris bed configuration to invert to a light oxide layer above a layer of metallic debris, mixed to a lower average temperature of ~1500 K. Table 4.3.4.2 summarizes the in-vessel, ex-vessel and total amounts of each radionuclide class released, all normalized to the initial inventories of each class.

Note that these amounts generally consider only the release of radioactive forms of these classes, and not additional releases of nonradioactive aerosols from structural materials. Also note that the CORSOR-M fission product release model option used in these analyses has identically zero release in-vessel of Class 7 (Mo), Class 9 (La) and Class 11 (Cd). Finally, note that the MELCOR model for this low-pressure boiloff sequence included the formation of CsI from Cs and I_2 released from the fuel, and its subsequent transport, deposition and release. The initial radionuclide inventories are such that all the I_2 released reacts to form CsI while most of the Cs remains unreacted and forms CsOH, which is the default Cs form in MELCOR.

Figure 4.3.4.20 gives the total radioactive release to the environment in these two cases; the release fractions of individual classes to the environment are shown in Figures 4.3.4.21 and 4.3.4.22. The releases to the environment begin when the auxiliary building fails. The total releases and time history of the release for this accident initiated at two different decay heat levels are quite similar, except for a timing shift due to the slower core degradation and later vessel and auxiliary building failures at the lower decay heat. These environmental releases do not correspond to immediate release of all radionuclides released from the fuel; there is considerable retention of most radionuclide species within the containment and auxiliary building (as discussed below). Only the noble gases have substantial releases to the environment by the end of the transient periods simulated, because gaseous forms are not scrubbed, filtered, deposited or otherwise retained. There is a total of 484.63 kg of noble gases and halogens released from the fuel; the release to the environment is >90% of this by the end of these low-pressure boiloff simulations. The temperatures are low enough in these shutdown sequences with flooded containments that the other volatile species released from the fuel (*i.e.*, CsOH, CsI and Te) are found mostly in aerosol form and are retained in the primary system, containment and auxiliary building.

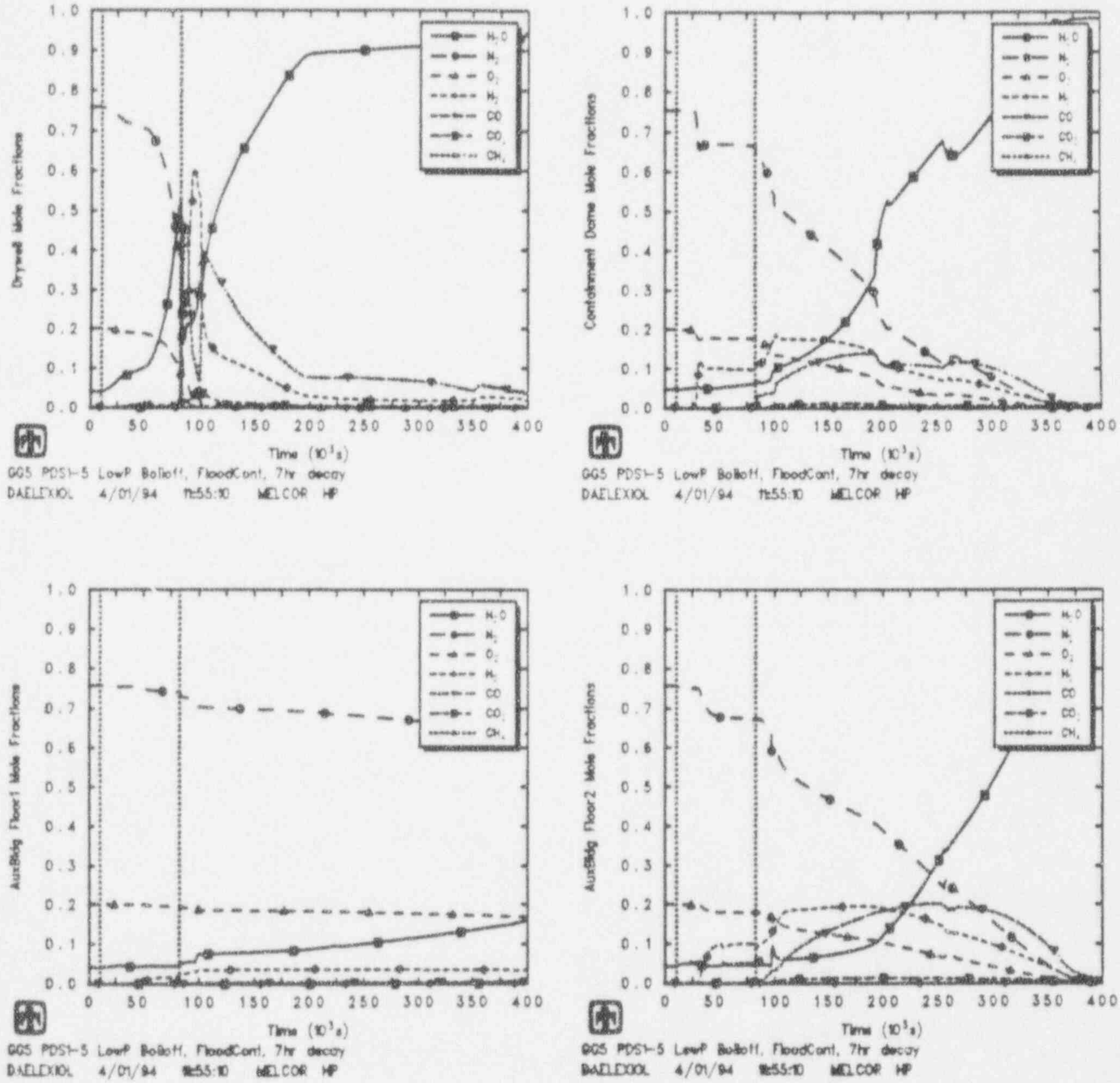


Figure 4.3.4.16. Mole Fractions in Drywell (upper left), Containment Dome (upper right), and Auxiliary Building First Floor (lower left) and Second Floor (lower right) for Grand Gulf POS 5 – Low Pressure Boiloff with Flooded Containment, Initiated 7 hr After Shutdown

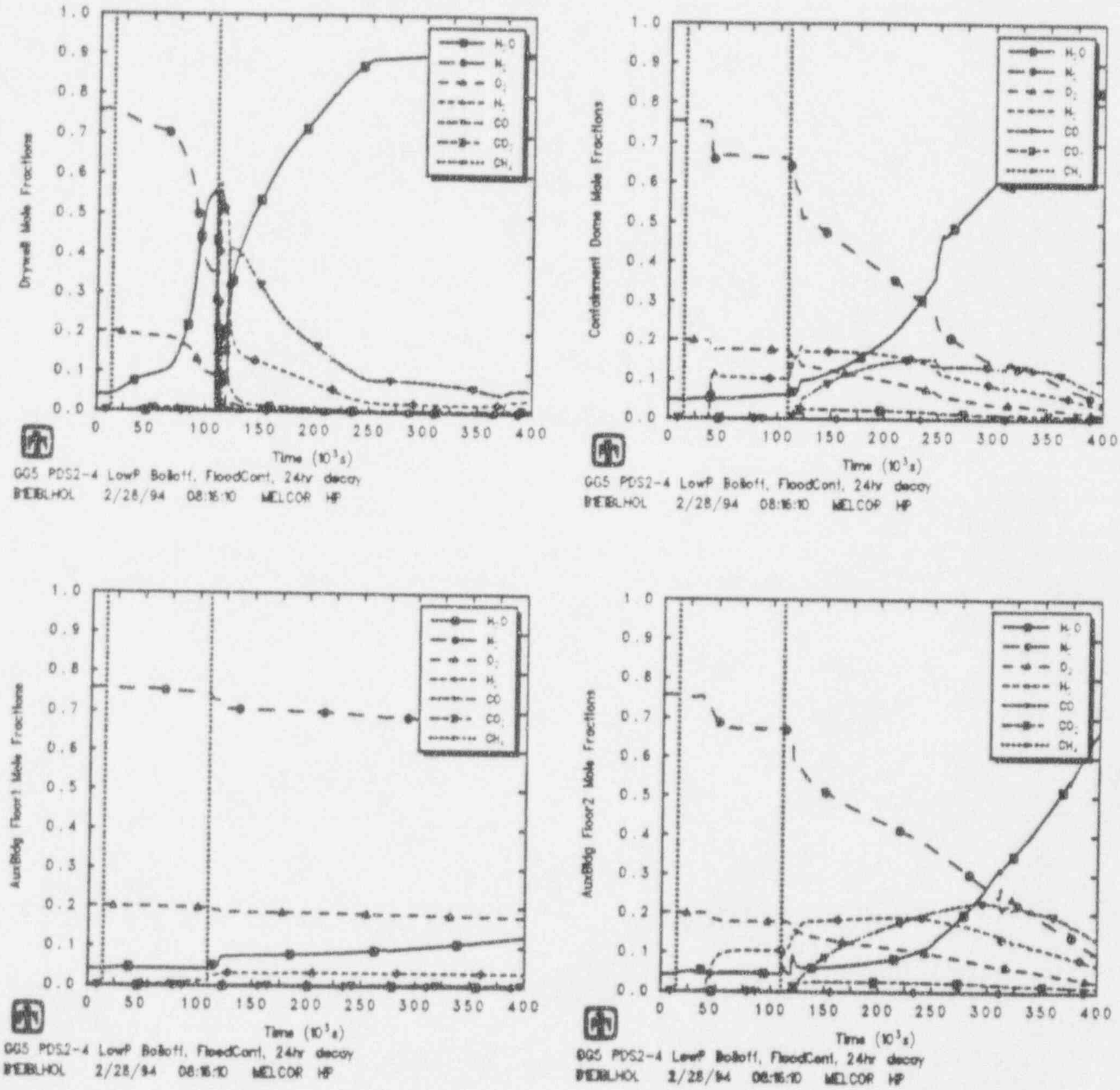


Figure 4.3.4.17. Mole Fractions in Drywell (upper left), Containment Dome (upper right), and Auxiliary Building First Floor (lower left) and Second Floor (lower right) for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated 24 hr After Shutdown

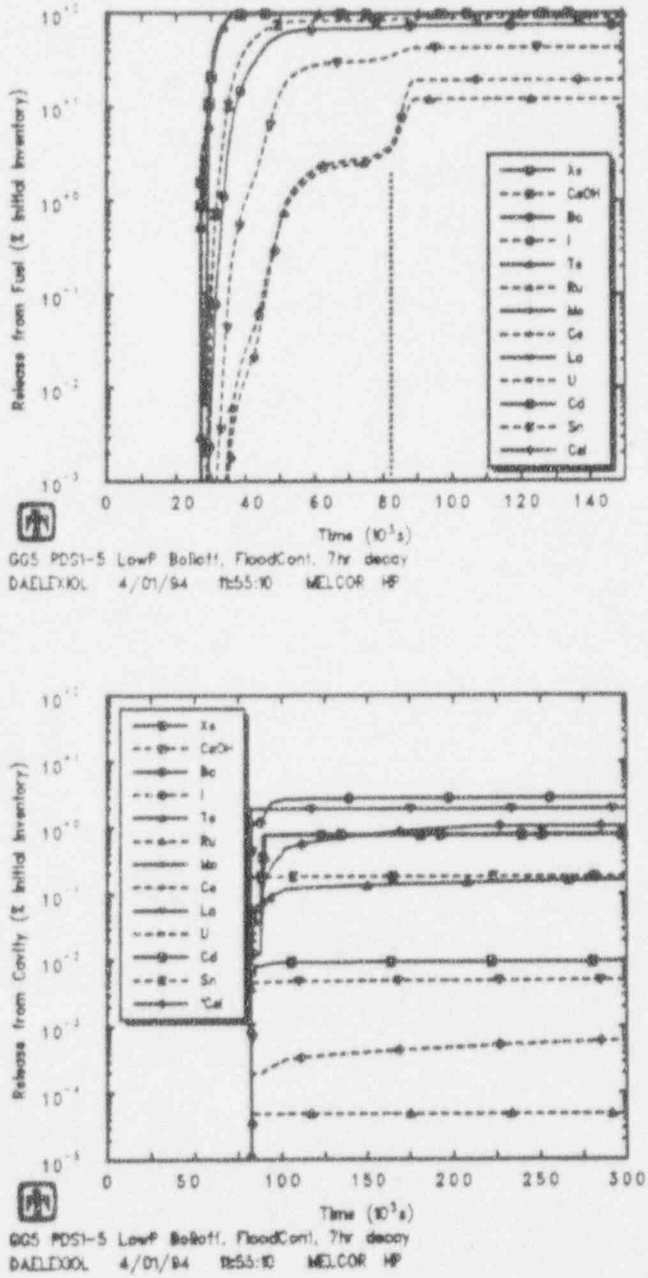
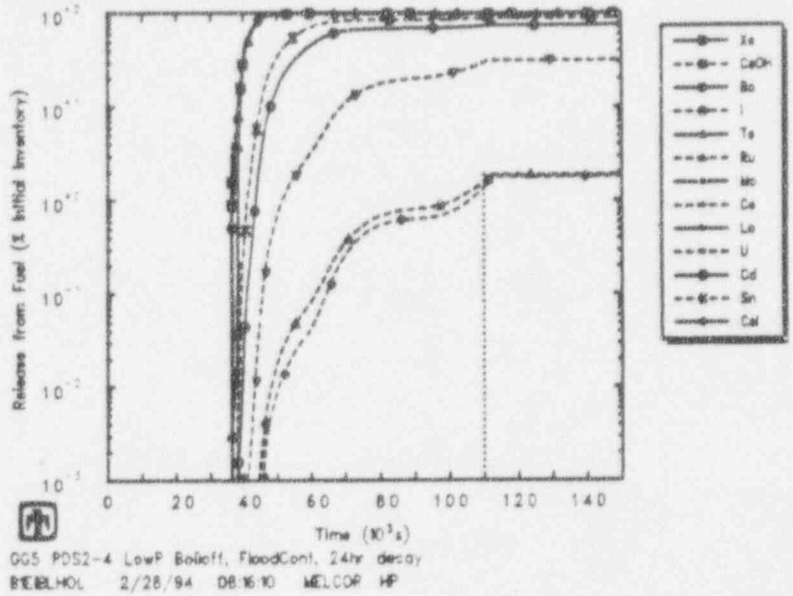


Figure 4.3.4.18. In-Vessel (top) and Ex-Vessel (bottom) Radionuclide Release Fractions for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated 7 hr After Shutdown



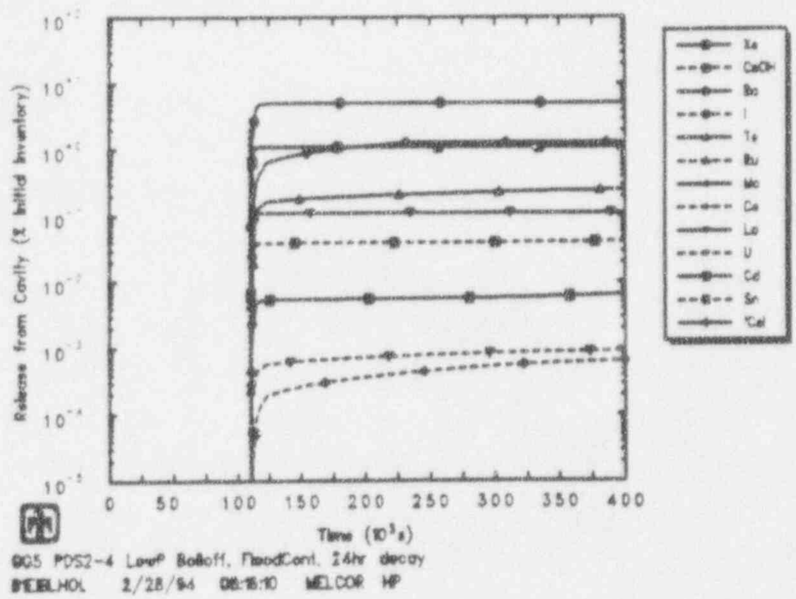


Figure 4.3.4.19. In-Vessel (top) and Ex-Vessel (bottom) Radionuclide Release Fractions for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated 24 hr After Shutdown

Table 4.3.4.2. Final Radionuclide Release Fractions for Grand Gulf POS 5 – Low Pressure Boiloff with Flooded Containment, Initiated at Various Times After Shutdown

Class	Fission Products Released from Fuel (% Initial Inventory)					
	7 hr			24 hr		
	In-Vessel	Ex-Vessel	Total	In-Vessel	Ex-Vessel	Total
Xe	99.18	0.79	99.97	98.86	1.11	99.97
CsOH	99.16	0.77	99.93	98.85	1.09	99.94
Ba	75.03	2.84	77.87	74.29	5.153	79.44
I	~0	~0	~0	~0	~0	~0
Te	99.09	0.175	99.27	98.77	0.248	99.02
Ru	12.09	0.00005	12.09	1.89	0.0000005	1.89
Mo	0	1.06	1.06	0	1.288	1.288
Ce	19.42	0.0007	19.42	1.79	0.0006	1.79
La	0	1.93	1.93	0	0.11	0.11
U	43.13	0.005	43.14	31.45	0.0009	31.45
Cd	0	0.010	0.010	0	0.0064	0.0064
Sn	88.86	0.185	89.05	89.68	0.04	89.72
CsI	99.18	0.80	99.98	98.84	1.12	99.96

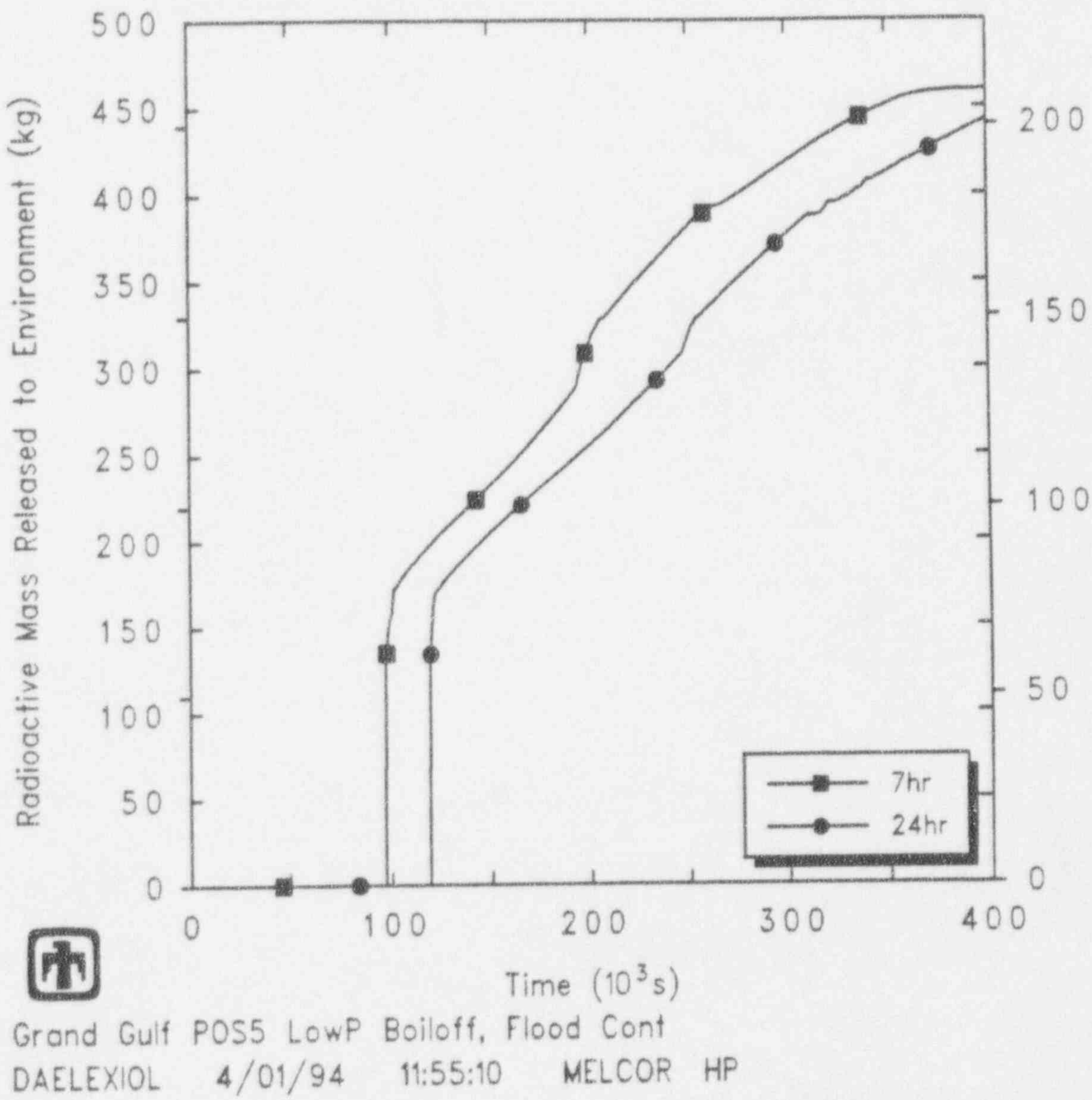


Figure 4.3.4.20. Total Environmental Radionuclide Releases for Grand Gulf POS 5 – Low Pressure Boiloff with Flooded Containment, Initiated at Various Times After Shutdown

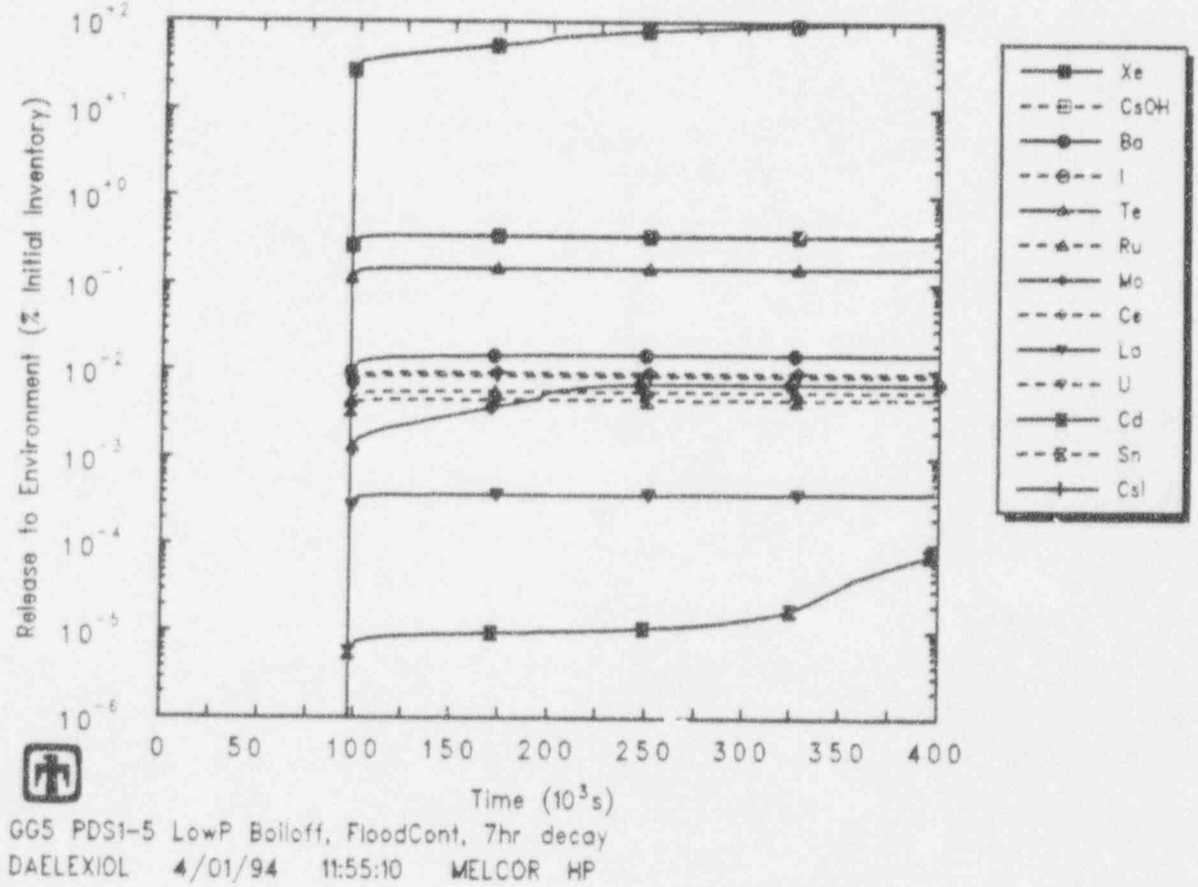


Figure 4.3.4.21. Environmental Radionuclide Release Fractions for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated 7 hr After Shutdown

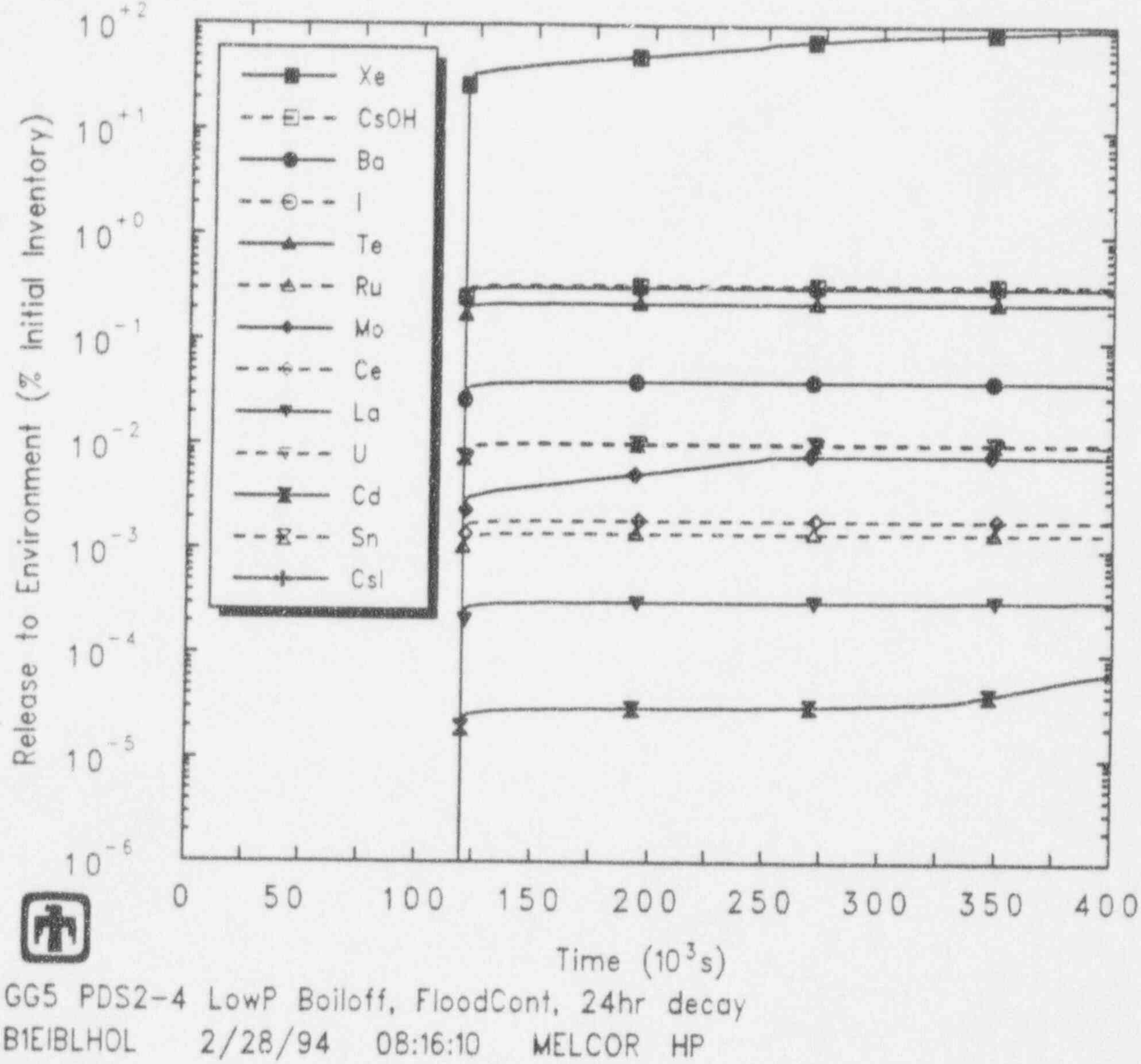


Figure 4.3.4.22. Environmental Radionuclide Release Fractions for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated 24 hr After Shutdown

Tables 4.3.4.3 and 4.3.4.4 summarize the distribution of the initial radionuclide inventory at the end of the two calculations initiated at different times after shutdown; they provide an overview of how much of the radionuclides remain bound up in fuel debris in either the core or the cavity, and of how much of the released radionuclides are retained in the primary system vs how much of the released radionuclides are released to, or released in, either the containment or the auxiliary building and the environment, all normalized to the initial inventories of each class. Table 4.3.4.5 presents a slightly different breakdown of the released radionuclide final distribution, giving the fractions of released inventory for each class in control volume atmospheres (including the environment), in pools, or deposited or settled onto heat structures at the end of the calculations. (As in Table 4.3.4.2, these amounts consider only the release of radioactive forms of these classes, and not additional releases of nonradioactive aerosols from structural materials.)

These tables show fission product distributions generally similar to those found for the large break LOCA sequence with flooded containment (discussed in Section ??) for the radionuclides with significant ($\geq$80% of initial inventory) release from fuel. In both accident scenarios, most of the noble gases released are in the environment, in the atmosphere. Most of the volatile species (CsOH, CsI and Te) releases occurred in-vessel in both scenarios, but most of those releases are retained in the containment, in water pools. The calculated releases of these volatiles to the environment are much lower for this low pressure boiloff sequence and for the large break LOCA scenario, both of which included flooded containments, than for the other accidents simulated. The two classes of radionuclides forming aerosols which had substantial releases (Ba and Sn, also occurring mostly in-vessel) were predicted to have about half those releases retained in the vessel, primarily deposited on structures, in both accident scenarios; the other half of those aerosol releases are retained in the containment, mostly in water pools and a small fraction deposited on structure surfaces.

4.3.5 High Pressure Boiloff with Open RPV Head Vent and Closed Containment, Initiated 24 hr After Shutdown

At the initiation of the accident, the reactor vessel is depressurized, the coolant is at the normal level and the SRV's are closed. Following the initiating event, all core cooling and makeup is lost and cannot be recovered. The operator fails to open the SRV's and steam the core at low pressure, *i.e.*, the SRV's remain closed during the accident and only open to relieve pressure at the safety setpoint. The vessel water inventory is at 366.5 K (200°F), which corresponds to the maximum temperature allowed by the Grand Gulf technical specifications for operation in POS 5. The reactor pressure vessel head vent is open at the beginning of the transient. The suppression pool level is 3.86 m (12.67 ft) above the suppression pool floor. In this scenario the operators successfully close the containment equipment hatch and both personnel locks 5 hr after the initiating event; however, the drywell personnel lock is still open. Containment is assumed to fail at 489 kPa (71 psia), with a 0.0929 m^2 opening above the auxiliary building roof (*i.e.*, "closed containment").

Table 4.3.4.3. Final Radionuclide Distribution for Grand Gulf POS 5 - Station Blackout with Flooded Containment, Initiated at 7 hr After Shutdown

Class	Fission Product Distribution (% Initial Inventory)				
	Fuel Debris	Primary System	Containment	Auxiliary Building	Environment
Xe	~0	0.033	0.003	3.59	96.4
CsOH	~0	3.35	85.38	10.9	0.350
Ba	22.1	38.6	36.87	2.33	0.0154
Te	0.72	4.2	85.93	8.92	0.151
Ru	87.8	2.64	9.24	0.201	0.0057
Mo	98.9	0.002	0.99	0.063	0.0070
Ce	80.6	3.13	15.98	0.304	0.0095
La	98.1	0.005	1.88	0.037	0.0004
U	60.3	20.2	18.60	0.883	0.0081
Cd	~100	0.00002	0.01	0.0003	0.00008
Sn	11.0	43.5	42.39	3.06	0.0048
CsI	~0	3.36	85.37	10.95	0.349

Table 4.3.4.4. Final Radionuclide Distribution for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated at 24 hr After Shutdown

Class	Fission Product Distribution (% Initial Inventory)				
	Fuel Debris	Primary System	Containment	Auxiliary Building	Environment
Xe	0.0012	0.0214	1.476	6.24	92.3
CsOH	0.0012	3.02	85.90	10.71	0.338
Ba	20.5	41.4	35.39	2.60	0.002
Te	0.934	3.92	85.43	9.54	0.229
Ru	98.1	1.18	0.67	0.043	0.0014
Mo	98.7	0.003	1.44	0.059	0.0079
Ce	98.2	1.04	0.718	0.04	0.0019
La	99.8	0.00008	0.11	0.0025	0.0003
U	71.1	19.4	8.84	0.68	0.0094
Cd	99.9	0.00001	0.006	0.0003	0.00007
Sn	10.3	43.5	42.76	3.41	0.0106
CsI	~0	3.07	85.89	10.62	0.325

Table 4.3.4.5. Final Radionuclide State for Grand Gulf POS 5 - Low Pressure Boiloff with Flooded Containment, Initiated at Various Times After Shutdown

Class	Fission Products Released from Fuel (% Released Inventory)					
	7 hr			24 hr		
	Atmosphere	Pool	Deposited	Atmosphere	Pool	Deposited
Xe	~100	0	0	~100	0	0
CsOH	0.35	96.30	3.36	0.34	96.67	3.03
Ba	0.02	50.23	49.75	0.05	47.71	52.25
I	~100	0	0	~100	0	0
Te	0.15	95.60	4.25	0.23	95.79	3.97
Ru	0.06	77.38	22.57	0.075	37.34	62.58
Mo	0.66	95.85	3.47	0.62	96.23	3.14
Ce	0.05	83.02	16.95	0.11	41.35	58.54
La	0.02	98.73	1.23	0.29	98.99	0.71
U	0.02	48.69	51.29	0.033	32.63	67.32
Cd	1.12	97.46	1.43	1.79	95.81	2.07
Sn	0.006	51.03	48.97	0.012	51.46	48.53
CsI	0.35	96.26	3.39	0.33	96.56	3.08

This sequence is almost identical to the high pressure boiloff scenario with open RPV head vent discussed in Section 4.2.4, except that in those Level 1 analyses the containment was open while in these Level 2 analyses the containment was assumed to be closed after 5 hr.

The sequence of events predicted by MELCOR for this accident with different initiation times is given in Table 4.3.5.1.

The vessel pressure response is very similar to that presented in Figures 4.2.5.1 in Section 4.2.5 for this sequence initiated 24 hr after shutdown. The vessel begins pressurizing as all core cooling is lost and continues pressurizing until reaching the SRV setpoint. The SRVs then cycle around the valve setpoints, intermittently opening. However, the system does not remain at the SRV cycling setpoints until vessel failure, but instead remains at the SRV cycling setpoints for only a few valve cycles before dropping due to continual inventory loss out the open RPV vent line. The vessel inventory response is also almost identical to the results discussed for the corresponding Level 1 analysis presented in Section 4.2.5 (shown in Figure 4.2.4.3).

The steam flow out both the SRVs and the RPV vent initially pressurizes both the containment and the auxiliary building, as shown in Figure 4.3.5.1. Closing the containment at 5 hr isolates the auxiliary building before it reaches its 5 psig overpressure failure setpoint. The closed containment continues to pressurize due to steam flow out both the SRVs and the RPV vent and later from the failed vessel lower head penetrations. There is a pressure spike in the containment at the time of vessel failure caused by flashing of the remaining lower plenum water by falling core debris. That pressure spike almost reached the containment failure pressure of 489 kPa (71 psia) locally in the cavity but did not challenge the containment global integrity. After that stepped increase in containment pressure at vessel failure, the containment continued to pressurize due to the generation of noncondensable gases from core-concrete interaction, until the containment failure pressure is reached.

Figure 4.3.5.2 presents the upper plenum, core and lower plenum swollen and collapsed liquid levels for this accident sequence. The level initially rises as the vessel pressurizes and then drops as inventory continues to be lost out the RPV vent and the SRV. The vessel liquid level drops smoothly through the upper plenum into the core and continue dropping smoothly partway into the lower plenum, followed by a more gradual loss of the remaining inventory due to boiling and steam outflow. There is very little pool frothing or swelling in any of the vessel volumes in this sequence. The lower plenum liquid level drops quickly to zero when the vessel lower head penetrations fail and any remaining water is dropped into the cavity together with falling core debris.

The heatup of the intact fuel and clad is illustrated in Figure 4.3.5.3. Because the fuel/clad component temperatures in MELCOR are set to zero in a cell when that component fails, this figure shows both the overall heatup rate and the time that the intact fuel/clad component fails through melting of the clad at 2100 K (3320°F). Figure 4.3.5.4 presents corresponding core debris temperatures in the active fuel region; these are the temperatures of the debris bed formed by the failure of the intact fuel/clad component in

Table 4.3.5.1. Sequence of Events Predicted by MELCOR for High Pressure Boiloff with Open RPV Head Vent and Closed Containment, Initiated 24 hr After Shutdown

Event	Time After Shutdown 24 hr
Accident initiation	0.0
Containment closed	18,000 s (5 hr)
Core uncovery (TAF) begins	42,875 s (11.91 hr)
Core heatup begins	43,500 s (12.08 hr)
Clad failure/Gap release	
(Ring 1)	49,945 s (13.87 hr)
(Ring 2)	49,857 s (13.85 hr)
(Ring 3)	49,931 s (13.87 hr)
(Ring 4)	50,362 s (13.99 hr)
(Ring 5)	51,959 s (14.16 hr)
(Ring 6)	70,680 s (19.63 hr)
Core plate failed	
(Ring 1)	73,667 s (20.46 hr)
(Ring 2)	73,628 s (20.45 hr)
(Ring 3)	76,631 s (21.29 hr)
(Ring 4)	78,988 s (21.94 hr)
(Ring 5)	80,344 s (22.32 hr)
(Ring 6)	85,320 s (23.70 hr)
Vessel LH penetration failed	
(Ring 1)	73,712 s (20.48 hr)
(Ring 2)	73,712 s (20.48 hr)
(Ring 3)	73,712 s (20.48 hr)
(Ring 4)	73,714 s (20.48 hr)
(Ring 5)	73,718 s (20.48 [illegible])
(Ring 6)	73,720 s (20.48 [illegible])
Commence debris ejection	73,712 s (20.48 hr)
Containment failed	308,264 s (85.63 hr)
Cavity rupture	343,883 s (95.52 hr)
End of calculation	343,883 s (95.52 hr)

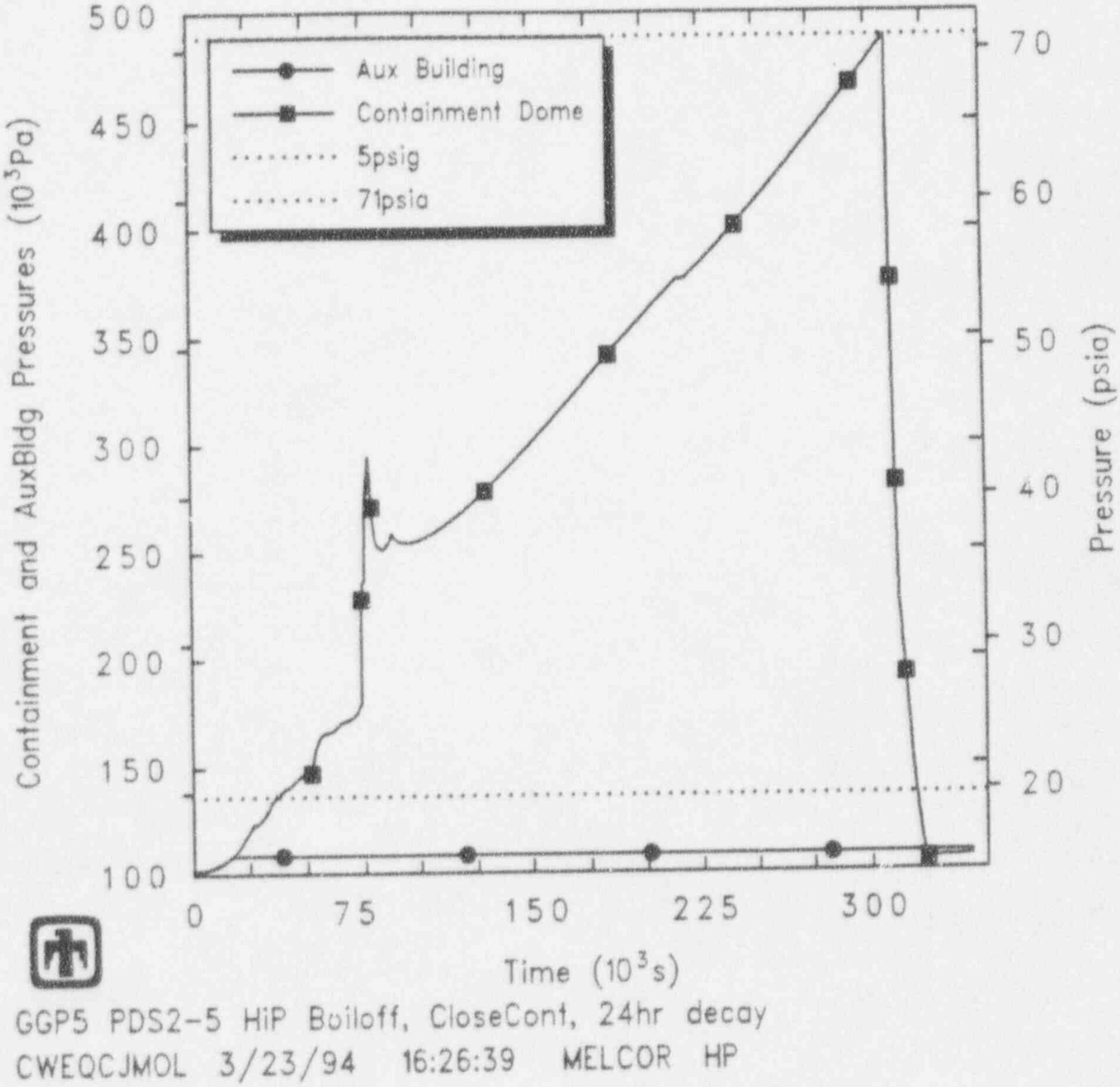


Figure 4.3.5.1. Containment and Auxiliary Building Pressures for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Vent and Closed Containment. Initiated 24 hr After Shutdown

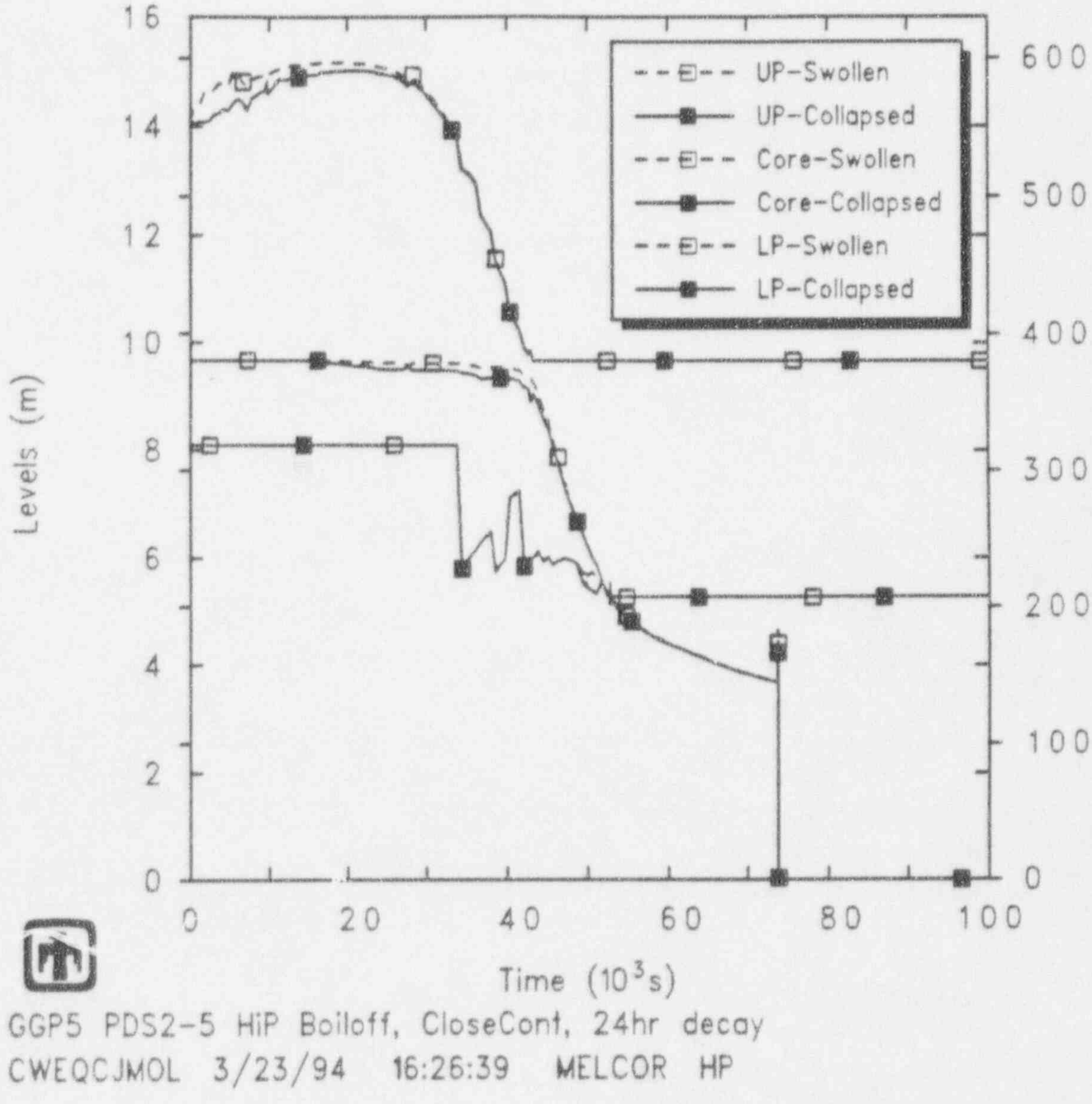


Figure 4.3.5.2. Upper Plenum, Core and Lower Plenum Liquid Levels for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Head Vent and Closed Containment, Initiated 24 hr After Shutdown

MELCOR in a core cell, whose (intact) temperatures were given in Figure 4.3.5.3. The debris bed in the active fuel region reaches peak temperatures about equal to the UO_2 melt temperature of 3113 K (5144°F).

The temperatures of the active fuel region debris bed drop to zero when the core plate fails and the debris relocates to the lower plenum. The predicted temperatures in the debris bed in the lower plenum and core plate are given in Figure 4.3.5.5. Prior to core plate failure there is some cold, refrozen debris both on the core support plate and on the lower core structural material just above the core support plate; the cooling and refreezing of this debris is the cause of the continued gradual drop in lower plenum liquid level due to steaming seen in Figure 4.3.5.2. The debris temperature rises gradually to the core support plate failure temperature of 1273 K (1832°F). After core plate failure hot, high-temperature debris begins appearing the the lower plenum as debris falls from the active fuel region into the lower plenum. With the new debris radial relocation model added in MELCOR 1.8.2, the core plate needs to fail in only one ring before debris from cells in the active fuel region in all radial rings can potentially flow sideways and down, fall through the failed plate, and then spread sideways into cells in the lower plenum in all radial rings. The lower head penetrations begin failing almost immediately, and the lower plenum debris temperatures begin dropping to zero as debris is ejected from the vessel to the cavity. Some cool, quenched debris remains present in the lower plenum for a significant period of time, however, as indicated by the 1000-1250 K debris temperatures in the lowest level after vessel failure.

Figure 4.3.5.6 illustrates what fraction of each material in the active fuel region has collapsed into a debris rubble bed held up by the core support plate, prior to core plate failure, debris relocation, lower head failure and debris ejection, for this high pressure boiloff scenario.

Figure 4.3.5.7 shows both the total and the individual masses of core materials (UO_2, Zircaloy and ZrO_2, stainless steel and steel oxide, and control rod poison) remaining in the vessel. This includes both material in the active fuel region and in the lower plenum. Debris ejection began very soon after lower head failure. This figure illustrates that most of the core material was lost from the vessel to the cavity quickly, in step-like stages. In all cases, all of the UO_2 was transferred to the cavity within a short time after the initial vessel lower head penetration failure, as was the unoxidized zircaloy, the associated zirc oxide and the control rod poison. A substantial fraction (75%) of the structural steel in the lower plenum, and some associated steel oxide, was predicted to remain unmelted and in place, more than in any of the other scenarios analyzed with MELCOR.

The debris material lost from the vessel is ejected to the drywell pedestal cavity. Figure 4.3.5.8 presents the amounts of ejected core debris, concrete ablated and the total cavity debris mass (*i.e.*, core debris combined with concrete ablation products). As in the other sequences analyzed, concrete ablation is quite rapid soon after debris ejection while the core debris is hot (>2000 K) and consists of a layer of metallic debris above a heavy oxide layer, and then slows noticably after enough concrete has been ablated for the debris bed configuration to invert to a light oxide layer above a layer of metallic debris, mixed to a lower average temperature of ~1500 K.

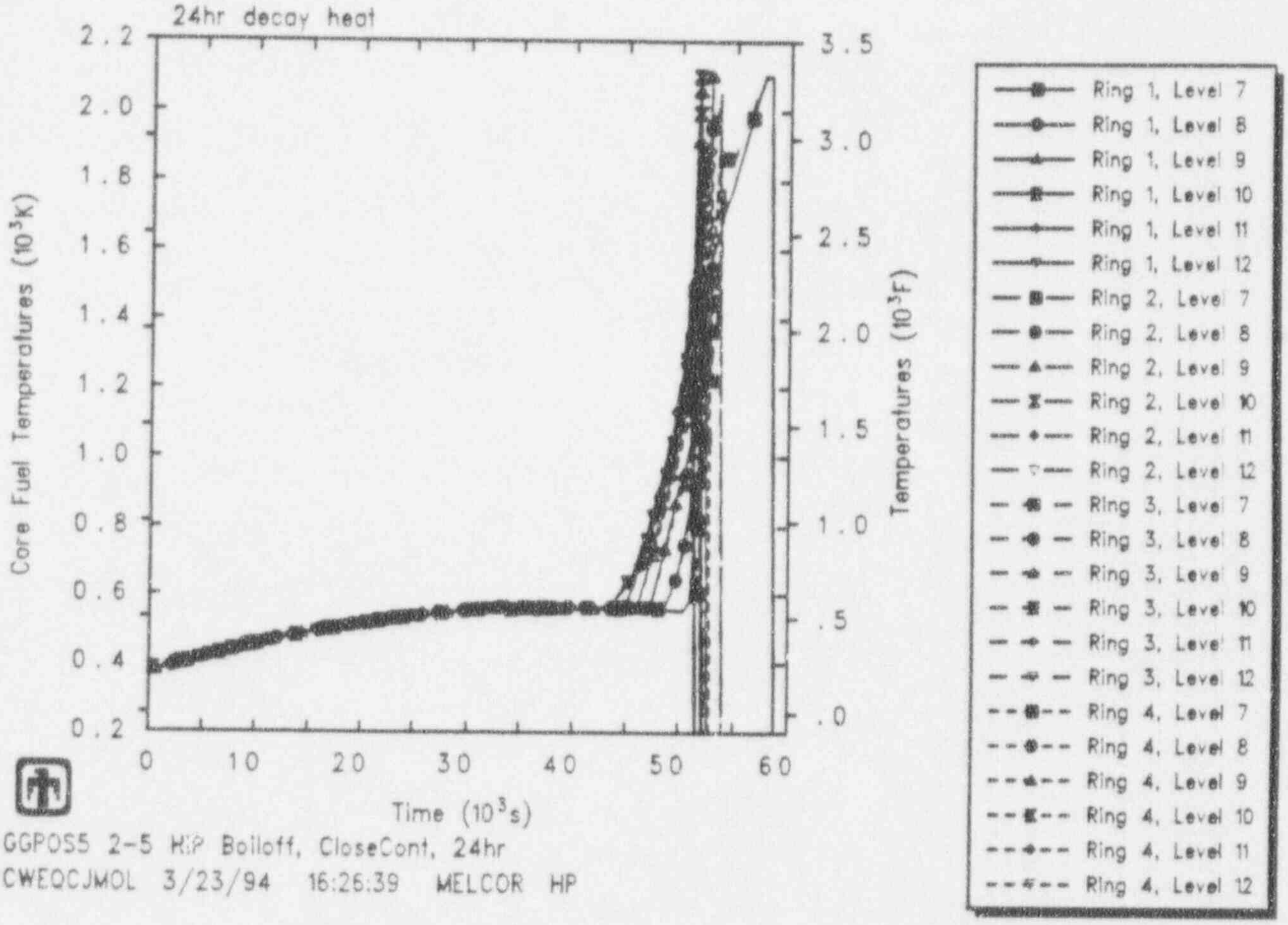


Figure 4.3.5.3. Core Intact Fuel/Clad Temperatures for Grand Gulf POS 5 – High Pressure Boiloff with Open RPV Head Vent and Closed Containment, Initiated 24 hr After Shutdown

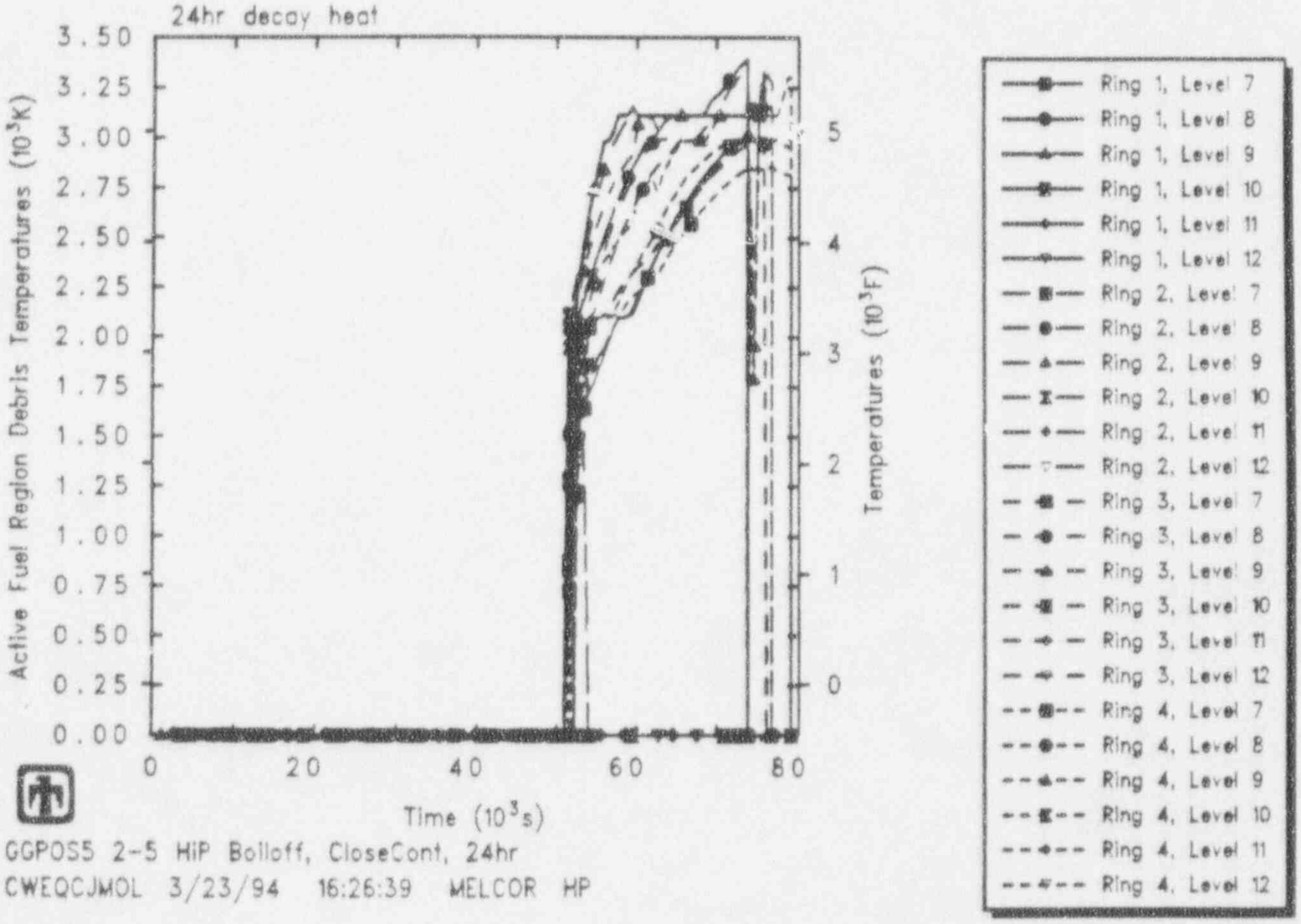


Figure 4.3.5.4. Core Active Fuel Region Debris Bed Temperatures for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Head Vent and Closed Containment, Initiated 24 hr After Shutdown

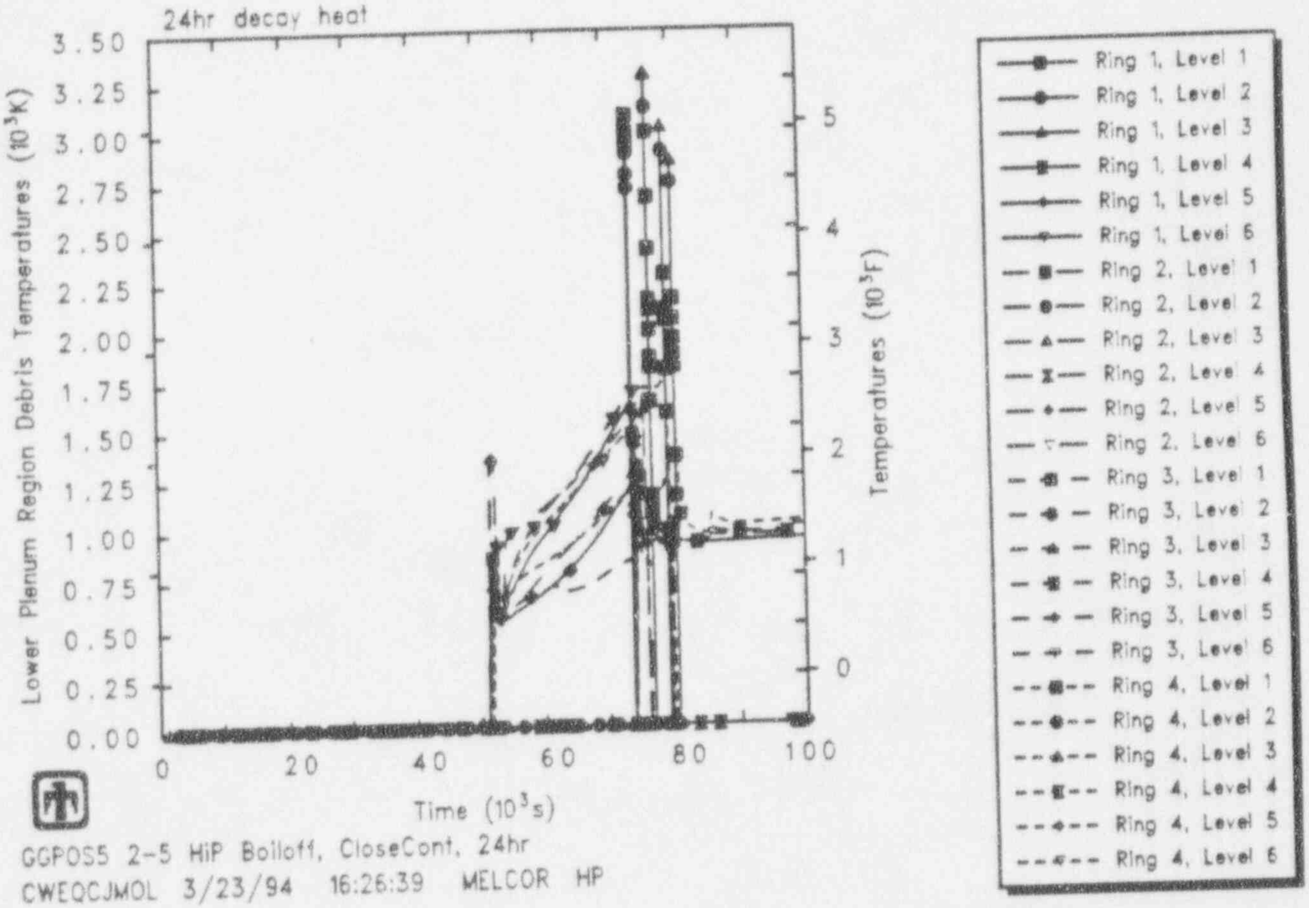


Figure 4.3.5.5. Core Lower Plenum and Core Support Plate Debris Bed Temperatures for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Head Vent And Closed Containment, Initiated 24 hr After Shutdown

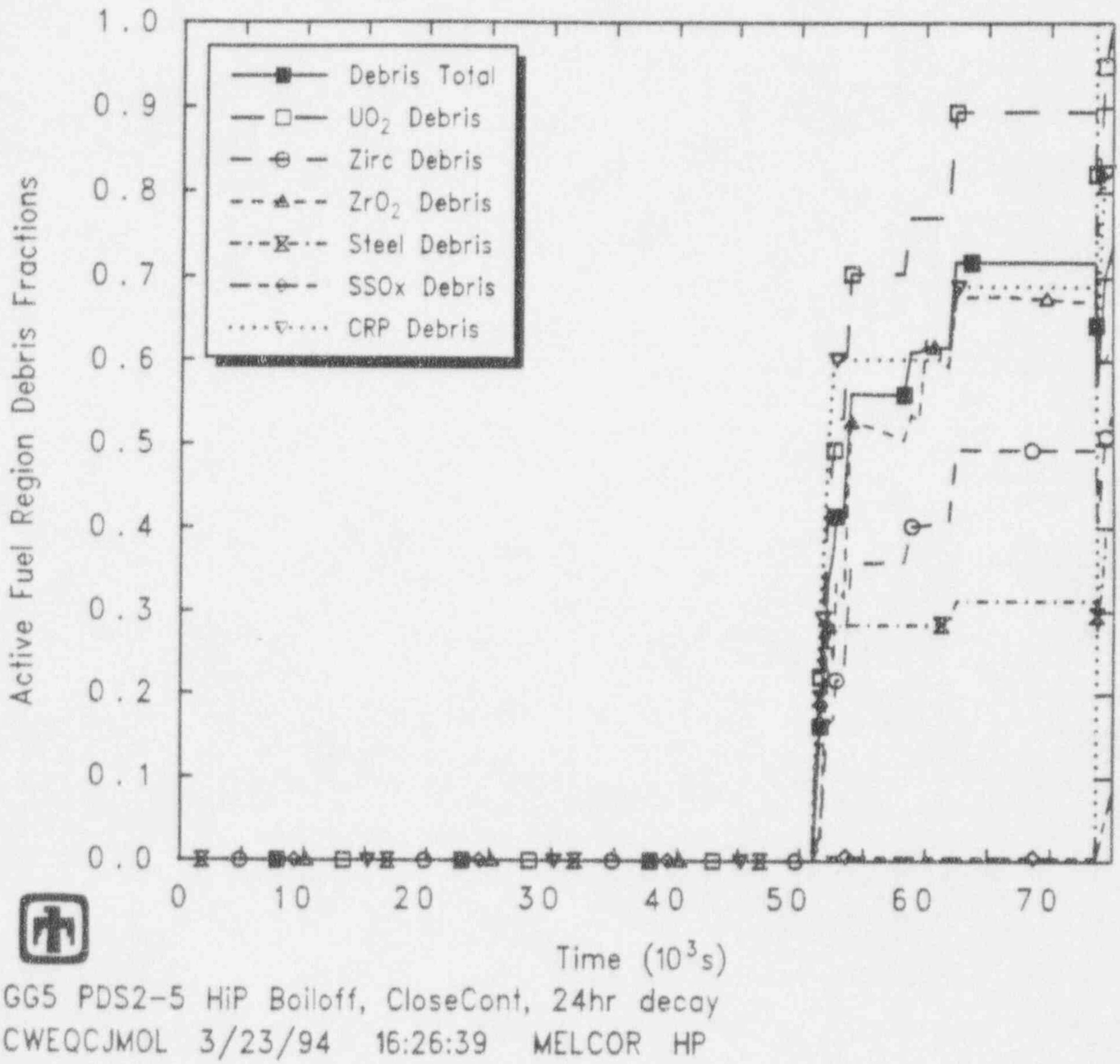


Figure 4.3.5.6. Core Active Fuel Region Degraded Material Fractions for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Head Vent and Closed Containment, Initiated 24 hr After Shutdown

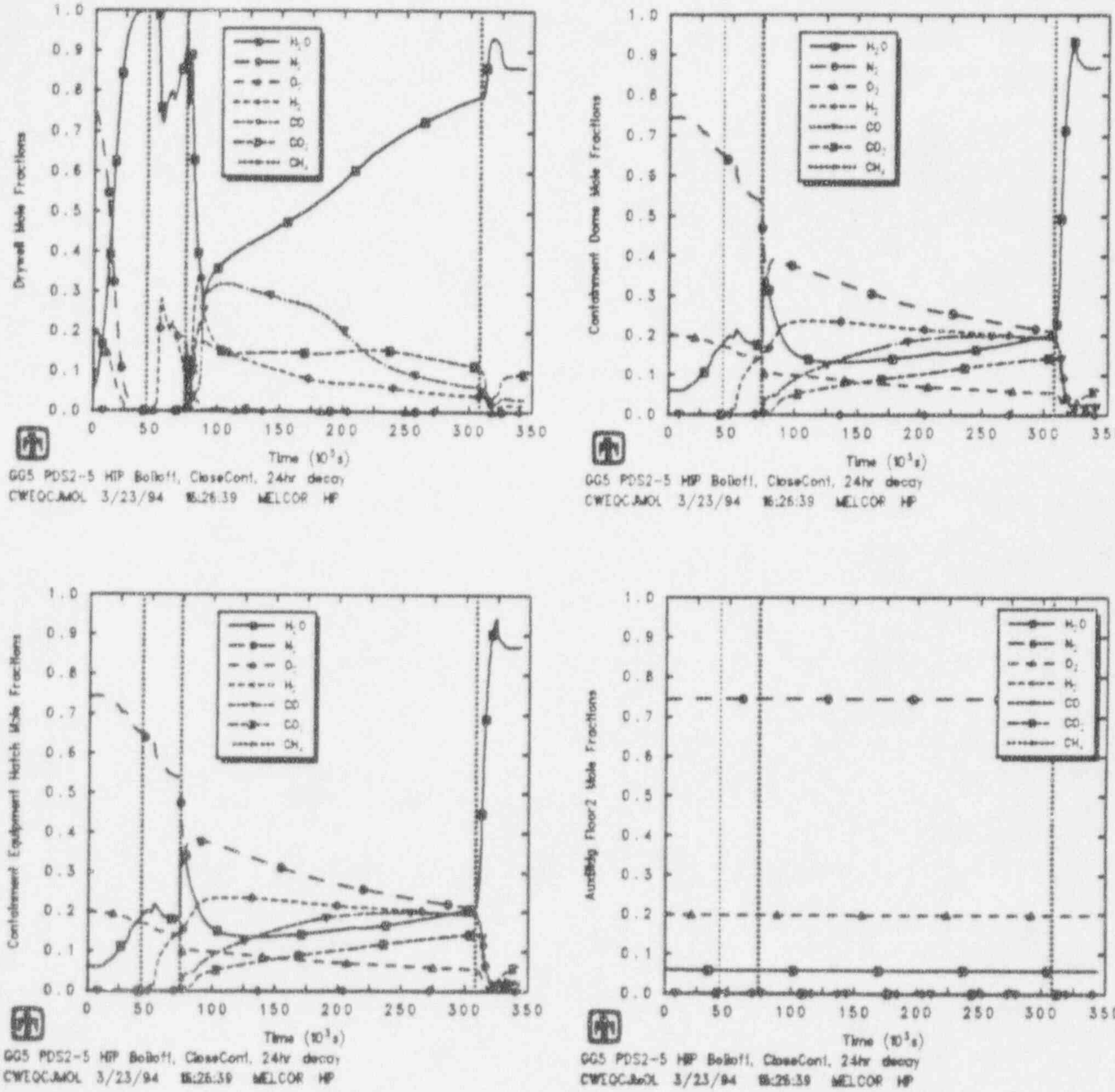


Figure 4.3.5.10. Mole Fractions in Drywell (upper left), Containment Dome (upper right), Containment Equipment Hatch (lower left) and Auxiliary Building (lower right) for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Head Vent and Closed Containment, Initiated 24 hr After Shutdown

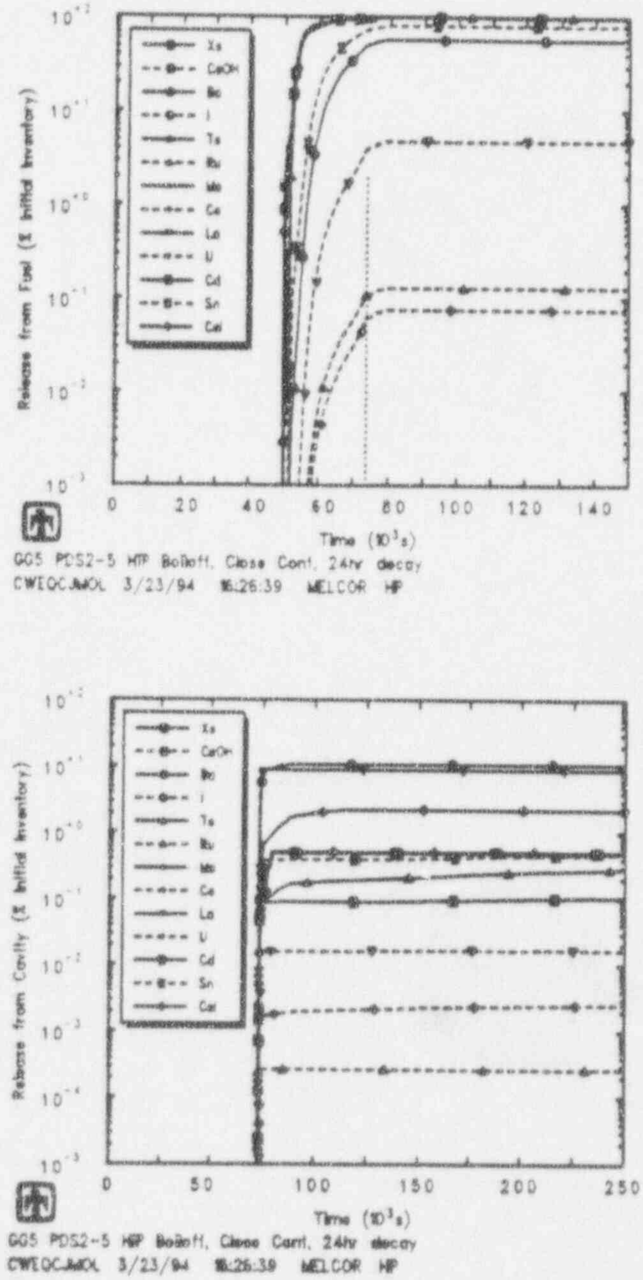


Figure 4.3.5.11. In-Vessel (top) and Ex-Vessel (bottom) Radionuclide Release Fractions for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Head Vent and Closed Containment, Initiated 24 hr After Shutdown

Table 4.3.5.2. Final Radionuclide Release Fractions for Grand Gulf POS 5 – High Pressure Boiloff with Open RPV Head Vent and Closed Containment. Initiated 24 hr After Shutdown

Class	Fission Products Released from Fuel (% Initial Inventory)		
	In-Vessel	Ex-Vessel	Total
Xe	99.46	0.50	99.96
CsOH	99.48	0.49	99.97
Ba	57.14	10.73	67.87
I	~0	~0	~0
Te	99.44	0.44	99.88
Ru	0.129	0.00026	0.129
Mo	0	2.19	2.19
Ce	0.076	0.0027	0.079
La	0	8.76	8.76
U	4.72	0.017	4.74
Cd	0	0.341	0.341
Sn	80.49	1.56	82.05
CsI	99.47	0.50	99.97

Figure 4.3.5.12 gives the total radioactive release to the environment, while the release fractions of individual classes to the environment are shown in Figure 4.3.5.13. The release to the environment does not begin at vessel failure in this sequence, but only after containment failure. These environmental releases do not correspond to immediate release of all radionuclides released from the fuel; there is considerable retention of most radionuclide species within the containment (but not within the isolated auxiliary building). Almost all the noble gases (~100%) are released to the environment soon after containment fails; in addition, there is some release to the environment of the other volatile species (*i.e.*, CsOH, CsI and Te) also.

Table 4.3.5.3 summarizes the distribution of the initial radionuclide inventory at the end of the calculation, and provides an overview of how much of the radionuclides remain bound up in fuel debris in either the core or the cavity, and of how much of the released radionuclides are retained in the primary system *vs* how much of the released radionuclides are released to, or released in, either the containment or the auxiliary building and the environment, all normalized to the initial inventories of each class. Table 4.3.5.4 presents a different breakdown of the released radionuclide final distribution, giving the fractions of released inventory for each class in control volume atmospheres (including the environment), in pools, or deposited or settled onto heat structures at the end of the calculations. (As in Table 4.3.5.2, these amounts consider only the release of radioactive forms of these classes, and not additional releases of nonradioactive aerosols from structural materials.)

These tables show fission product distributions somewhat similar to those found for the large break LOCA sequences (discussed in Section ??) for the radionuclides with significant (≥80% of initial inventory) release from fuel. In all the accident scenarios simulated, most of the noble gases released are in the environment, in the atmosphere. Most of the volatile species (CsOH, CsI and Te) releases occurred in-vessel in both the large break LOCA and in this high pressure boiloff, with most of those releases retained in the containment. More of the volatiles are released to the environment in this high pressure boiloff with closed containment than in the large break LOCA or station blackout scenarios. This is the only accident sequence analyzed with the calculated environmental release fraction increasing with the volatility (*i.e.*, CsI being the most volatile has the highest environmental release fraction, while CsOH being the least volatile has the lowest environmental release fraction), probably due to the fact that most of the releases to the environment occur with the containment at relatively high pressure compared to ambient. The two classes of radionuclides forming aerosols which had substantial releases (Ba and Sn, also occurring mostly in-vessel) were predicted to have about half those releases retained in the vessel and primarily deposited on structures in both accident scenarios, and the other half retained in the containment mostly in water pools but some deposited on structure surfaces.

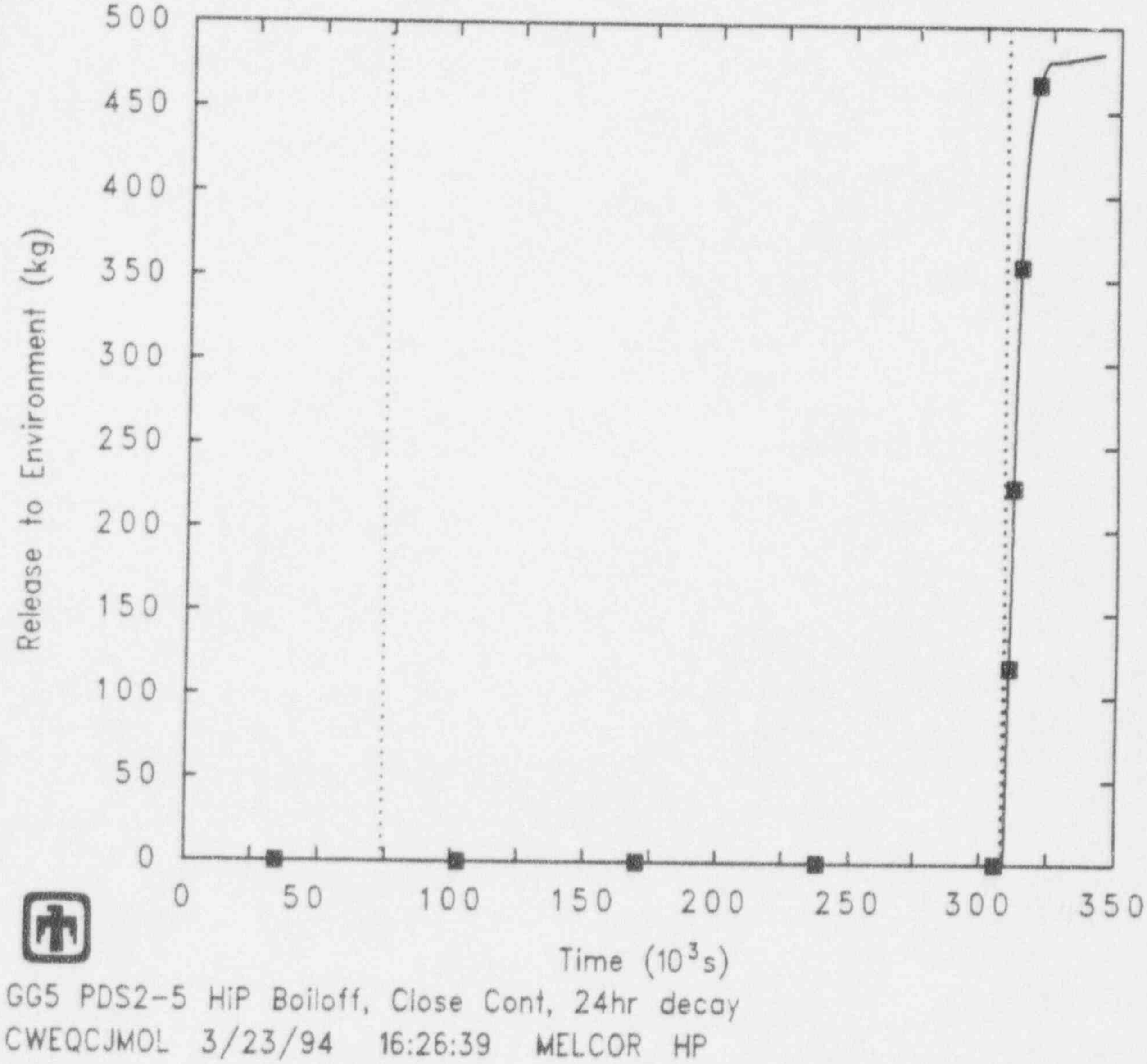


Figure 4.3.5.12. Total Environmental Radionuclide Releases for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Head Vent and Closed Containment, Initiated 24 hr After Shutdown

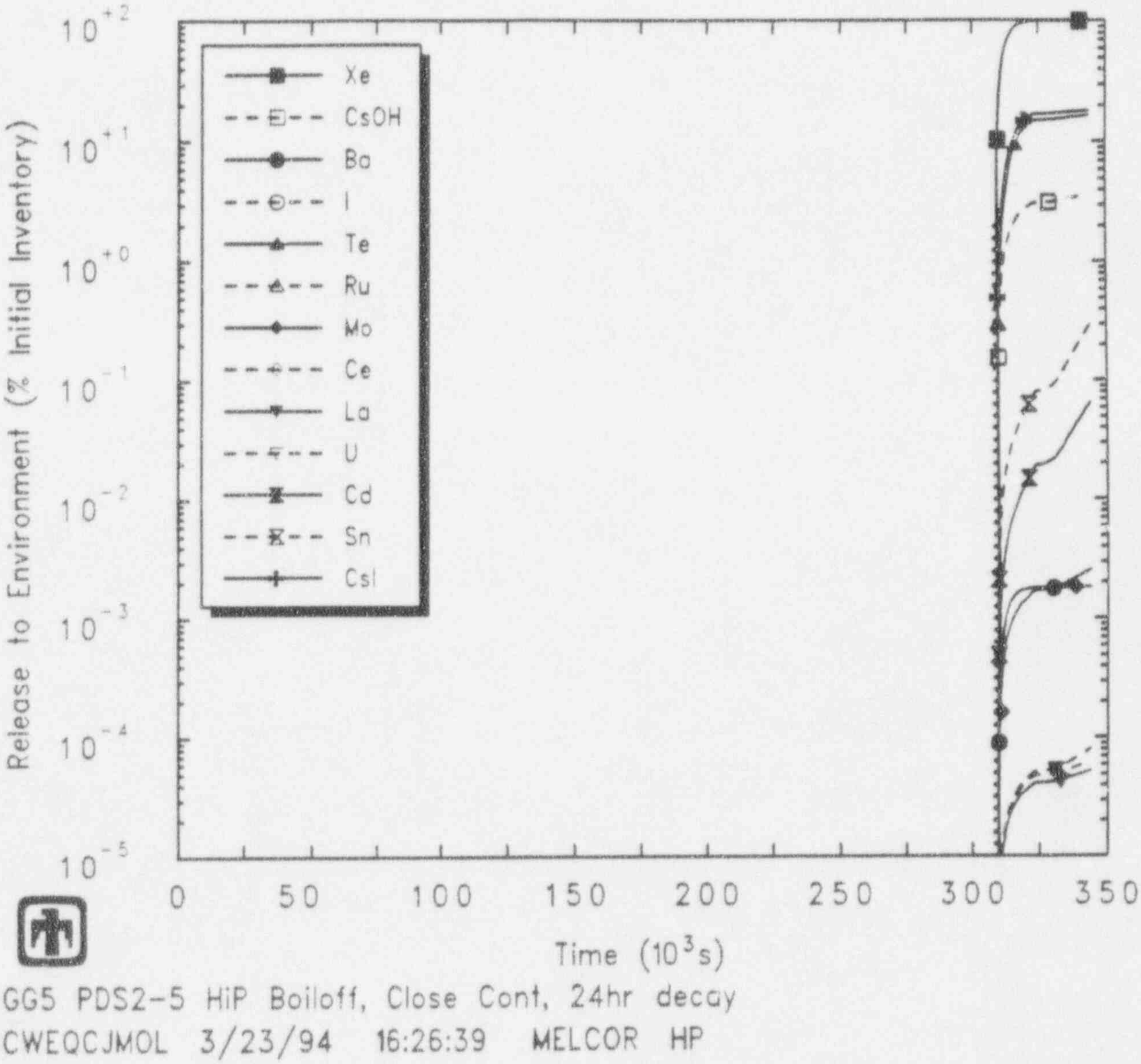


Figure 4.3.5.13. Environmental Radionuclide Release Fractions for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Head Vent and Closed Containment, Initiated at 24 hr After Shutdown

Table 4.3.5.3. Final Radionuclide Distribution for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Head Vent and Closed Containment. Initiated 24 hr After Shutdown

Class	Fission Product Distribution (% Initial Inventory)				
	Fuel Debris	Primary System	Containment	Auxiliary Building	Environment
Xe	0.04	0.028	0.431	0	99.5
CsOH	0.04	48.8	47.54	0	3.60
Ba	32.14	38.8	34.07	0	0.003
Te	0.12	11.3	64.65	0	16.5
Ru	99.8	0.069	0.060	0	0.0000007
Mo	97.8	0.217	1.98	0	0.0018
Ce	99.9	0.04	0.038	0	0.00006
La	91.2	0.955	7.80	0	0.00005
U	95.6	2.35	2.02	0	0.00007
Cd	99.6	0.011	0.266	0	0.0643
Sn	17.9	48.4	33.32	0	0.291
CsI	~0	3.49	78.52	0	18.0

Table 4.3.5.4. Final Radionuclide State for Grand Gulf POS 5 - High Pressure Boiloff with Open RPV Head Vent and Closed Containment, Initiated 24 hr After Shutdown

Class	Fission Products Released from Fuel (% Released Inventory)		
	Atmosphere	Pool	Deposited
Xe	~100	0	0
CsOH	4.69	36.9	58.4
Ba	0.006	22.0	78.0
I	~100	0	0
Te	24.40	36.6	39.0
Ru	~0	18.5	81.5
Mo	0.086	36.9	63.0
Ce	0.11	19.6	80.3
La	0.001	55.7	44.3
U	0.003	18.4	81.6
Cd	49.5	22.4	28.0
Sn	0.95	16.2	82.8
CsI	22.8	51.0	26.2

4.3.6 Open MSIVs with Closed Containment, Initiated 24 hr After Shutdown

The accident is initiated 24 hr after shutdown. The MSIVs are open; the reactor head vent is closed. The water level in the vessel is at the steam lines, and the water in the vessel is at 366.5 K (200°F), which corresponds to the maximum temperature allowed by the Grand Gulf technical specifications for operation in POS 5. The suppression pool level is at the ECCS suction strainers, 3.05 m (10 ft) from the suppression pool floor. The containment is at 305.4 K (90°F) and the suppression pool is at 308.2 K (95°F). Following the initiating event, the operators close the containment 5 hr after the initiating event, but the drywell personnel lock remains open. Injection is not restored to the core during the accident.

This sequence is virtually identical to the open-MSIV scenario discussed in Section 4.2.1; in those Level 1 analyses the containment was open while in these Level 2 analyses the containment was assumed to be closed after 5 hr but, because of the open MSIV line providing a path to the auxiliary building, that difference in scenario is not significant.

The sequence of events predicted by MELCOR for this accident with different initiation times is given in Table 4.3.6.1.

Figure 4.3.6.1 gives the vessel, containment and auxiliary building pressures predicted by MELCOR. The pressure responses for the vessel and for the auxiliary building are very similar to that presented in Figures 4.2.1.1 and 4.2.1.2 in Section 4.2.1 for this sequence initiated 24 hr after shutdown. The system begins pressurizing as all core cooling is lost but only pressurizes to ~160kPa before the steam flow out the open MSIVs is sufficient to remove all the decay heat. The steam flow out the MSIVs in turn pressurizes the auxiliary building and, through the open equipment hatch and personnel locks, pressurizes the containment. The auxiliary building fails on a 0.345 kPa (5 psig) overpressure. The closing of the containment at 5 hr allows a pressure differential of ~2 psig to build up between the reactor pressure vessel and the containment, but the open MSIV line keeps the vessel equilibrated and venting to the auxiliary building, which fails soon after 5 hr when the containment is closed.

The vessel inventory response is also almost identical to the results discussed for the corresponding Level 1 analysis presented in Section 4.2.1 (shown in Figure 4.2.1.3). Figure 4.3.6.2 presents the upper plenum, core and lower plenum swollen and collapsed liquid levels for this accident sequence. The level drops as inventory continues to be lost out the open MSIV line. The vessel liquid level drops through the upper plenum into the core and continue dropping smoothly partway into the lower plenum, followed by a more gradual loss of the remaining inventory due to boiling and steam outflow. There is substantial pool frothing and swelling in both the upper plenum and upper core regions during this boiloff. vessel volumes in this sequence. The lower plenum liquid level drops quickly to zero when the vessel lower head penetrations fail and any remaining water is dropped into the cavity together with falling core debris.

Table 4.3.6.1. Sequence of Events Predicted by MELCOR for Open MSIVs with Closed Containment, Initiated 24 hr After Shutdown

Event	Time After Shutdown 24 hr
Accident initiation	0.0
Core uncovery (TAF) begins	15,714 s (2.72 hr)
Containment closed	18,000 s (5 hr)
Auxiliary building failed	20,000 s (5.56 hr)
Core heatup begins	30,000 s (8.33 hr)
Clad failure/Gap release	
(Ring 1)	35,373 s (17.53 hr)
(Ring 2)	35,290 s (17.51 hr)
(Ring 3)	35,377 s (17.52 hr)
(Ring 4)	35,838 s (17.62 hr)
(Ring 5)	37,452 s (18.02 hr)
(Ring 6)	41,997 s (22.00 hr)
Core plate failed	
(Ring 1)	118,554 s (25.14 hr)
(Ring 2)	113,565 s (26.43 hr)
(Ring 3)	118,141 s (26.26 hr)
(Ring 4)	116,470 s (26.25 hr)
(Ring 5)	118,063 s (28.50 hr)
(Ring 6)	122,243 s (31.21 hr)
Vessel LH penetration failed	
(Ring 1)	113,652 s (25.16 hr)
(Ring 2)	113,666 s (25.124 hr)
(Ring 3)	113,565 s (25.124 hr)
(Ring 4)	113,642 s (25.18 hr)
(Ring 5)	113,647 s (28.54 hr)
(Ring 6)	113,653 s (31.36 hr)
Commence debris ejection	113,565 s (25.16 hr)
Cavity rupture	
End of calculation	250,000 s (55.32 hr)

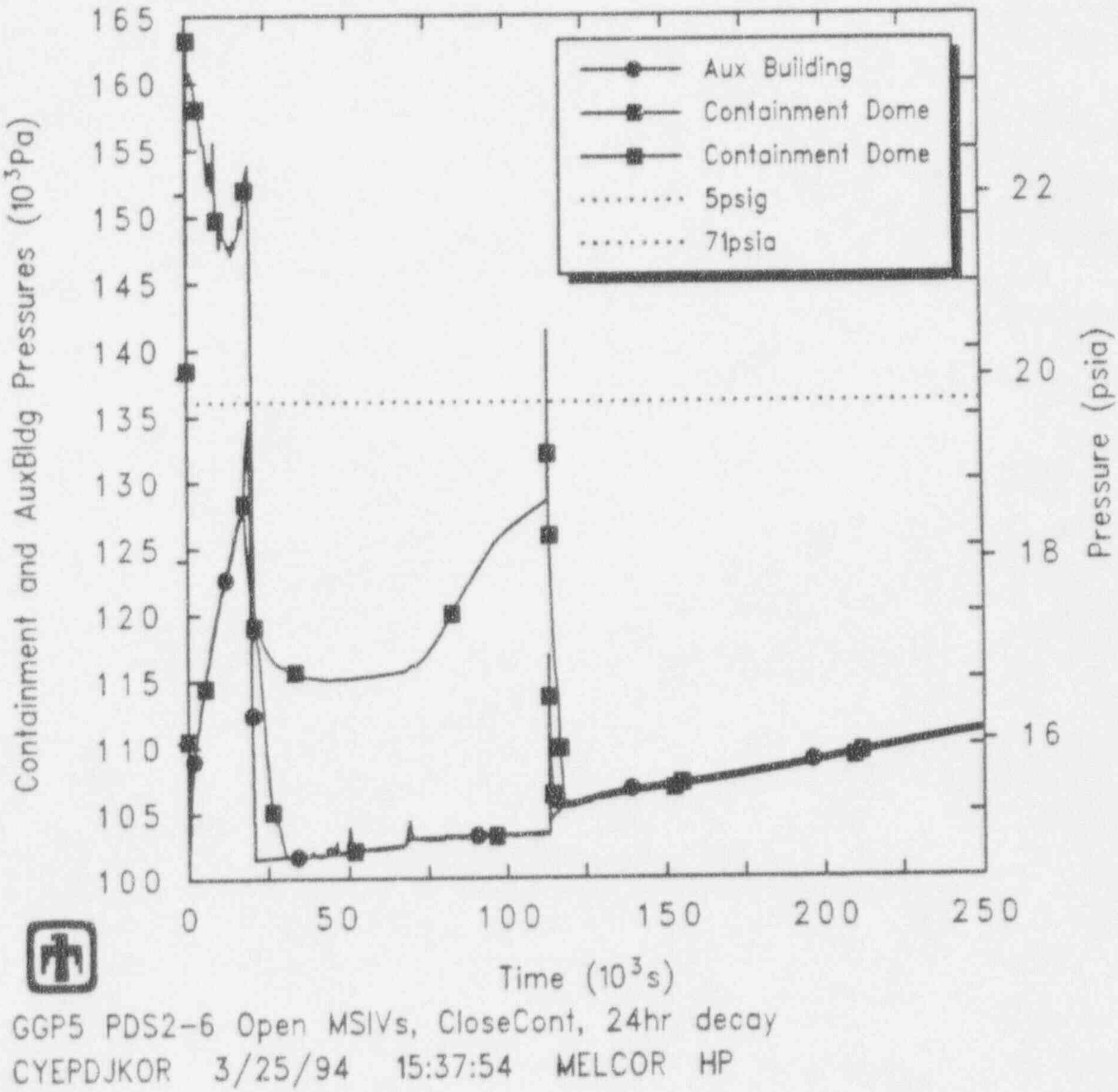


Figure 4.3.6.1. Vessel, Containment and Auxiliary Building Pressures for Grand Gulf POS 5 – Open MSIV's with Closed Containment, Initiated 24 hr After Shutdown

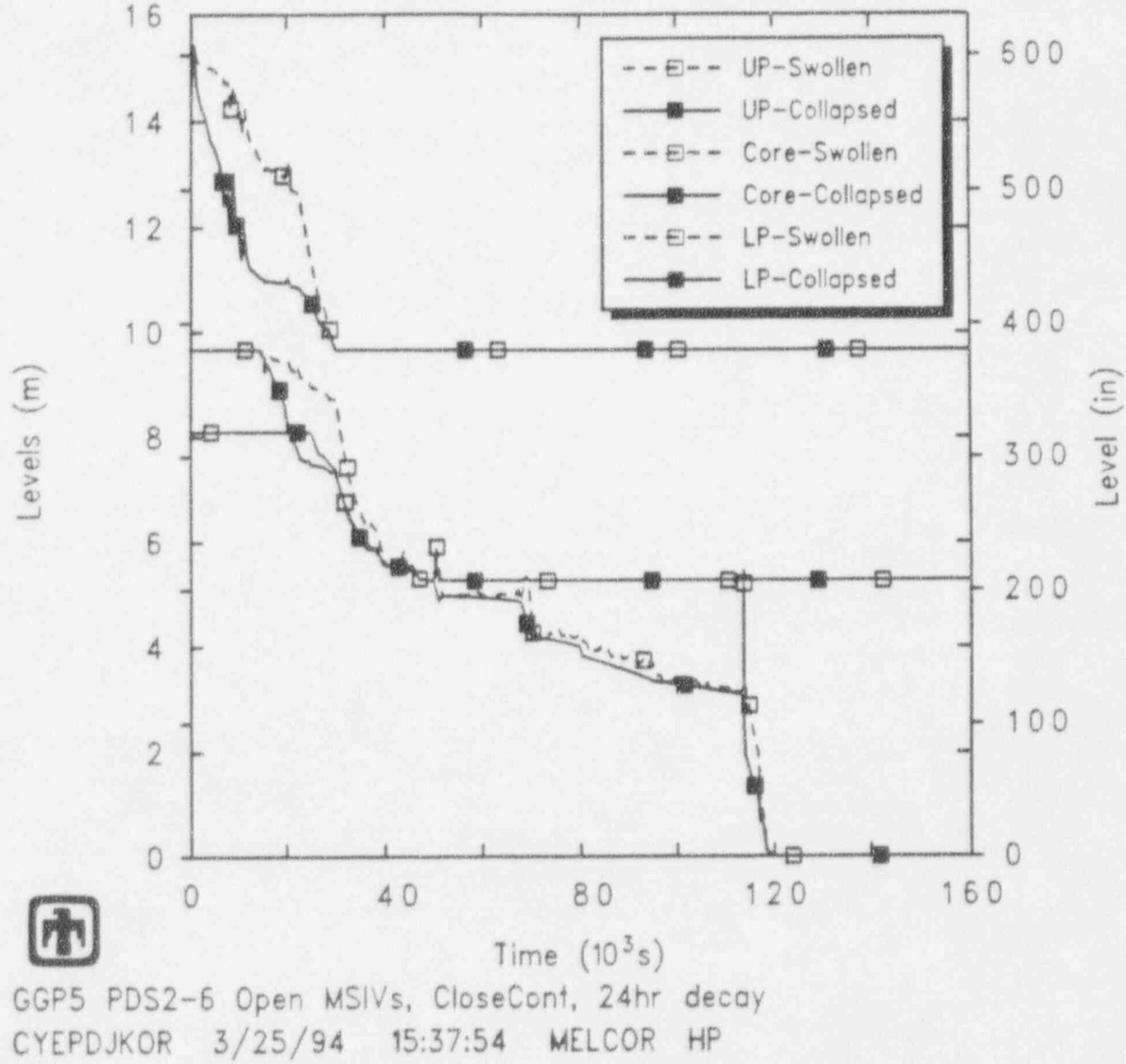


Figure 4.3.6.2. Upper Plenum, Core and Lower Plenum Liquid Levels for Grand Gulf POS 5 - Open MSIVs with Closed Containment, Initiated 24 hr After Shutdown

The heatup of the intact fuel and clad is illustrated in Figure 4.3.6.3. Because the fuel/clad component temperatures in MELCOR are set to zero in a cell when that component fails, this figure shows both the overall heatup rate and the time that the intact fuel/clad component fails through melting of the clad at 2100 K (3320°F). Figure 4.3.6.4 presents corresponding core debris temperatures in the active fuel region; these are the temperatures of the debris bed formed by the failure of the intact fuel/clad component in MELCOR in a core cell, whose (intact) temperatures were given in Figure 4.3.6.3. The debris bed in the active fuel region reaches peak temperatures $\geq$3500 K (5840°F), significantly above the UO_2 melt temperature of 3113 K (5144°F), except in the lowest active fuel level where the temperature never reaches the UO_2 melt temperature. The temperatures of the active fuel region debris bed drop to zero when the core plate fails and the debris relocates to the lower plenum.

The predicted temperatures in the debris bed in the lower plenum and core plate are given in Figure 4.3.6.5. Prior to core plate failure there is some cold, refrozen debris both on the core support plate and on the lower core structural material just above the core support plate; the cooling and refreezing of this debris is the cause of the continued gradual drop in lower plenum liquid level due to steaming seen in Figure 4.3.6.2. The lower core debris bed temperatures during this time period are substantially lower than predicted in the other transients analyzed, due to enhanced steam flow and cooling in the core region, and it takes a relatively long time for the debris temperature to rise to the core support plate failure temperature of 1273 K (1832°F). After core plate failure hot, high-temperature debris begins appearing the the lower plenum as debris falls from the active fuel region into the lower plenum. The lower head penetrations begin failing almost immediately, and the lower plenum debris temperatures begin dropping to zero as debris is ejected from the vessel to the cavity.

Figure 4.3.6.6 illustrates what fraction of each material in the active fuel region has collapsed into a debris rubble bed held up by the core support plate, prior to core plate failure, debris relocation, lower head failure and debris ejection, for this high pressure boiloff scenario. The debris bed forms as material (in particular, the zircaloy clad and the UO_2 fuel) reaches melting. The debris bed forms relatively slowly in this scenario, taking 10,000-20,000 s to reach its final configuration. The fraction of material in the debris bed later remains nearly constant as the debris material continues to heat up.

Figure 4.3.6.7 shows both the total and the individual masses of core materials (UO_2, Zircaloy and ZrO_2, stainless steel and steel oxide, and control rod poison) remaining in the vessel. This includes both material in the active fuel region and in the lower plenum. Debris ejection began very soon after lower head failure. This figure illustrates that most of the core material was lost from the vessel to the cavity quickly, in step-like stages. In all cases, all of the UO_2 was transferred to the cavity within a short time after the initial vessel lower head penetration failure, as was the unoxidized zircaloy, the associated zirc oxide and the control rod poison. A substantial fraction (45-50%) of the structural steel in the lower plenum, and some associated steel oxide, was predicted to remain unmelted and in place, more than in any of the other scenarios analyzed with MELCOR except the high pressure boiloff discussed in the previous section.

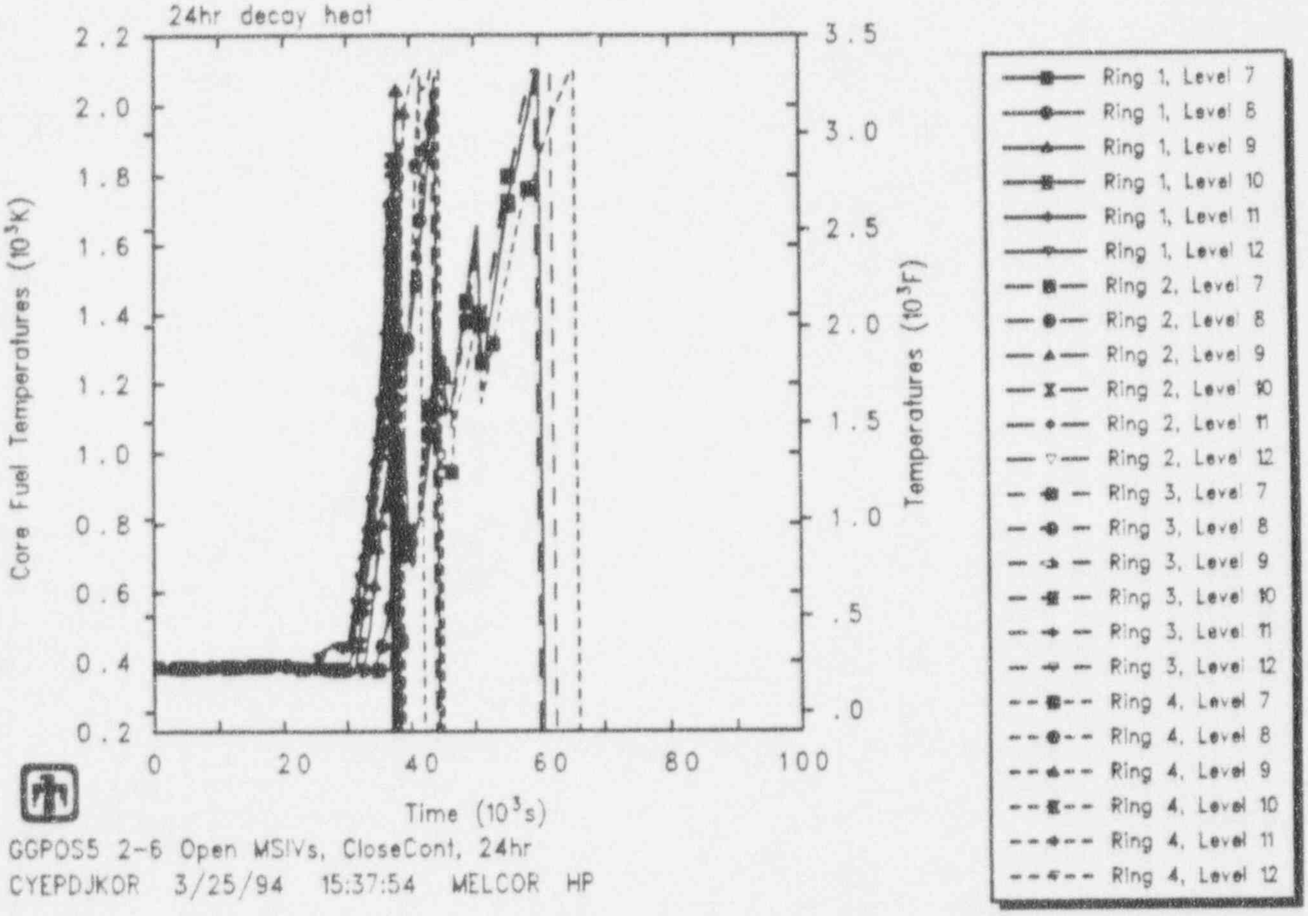


Figure 4.3.6.3. Core Intact Fuel/Clad Temperatures for Grand Gulf POS 5 - Pressure Open MSIVs with Closed Containment, Initiated 24 hr After Shutdown

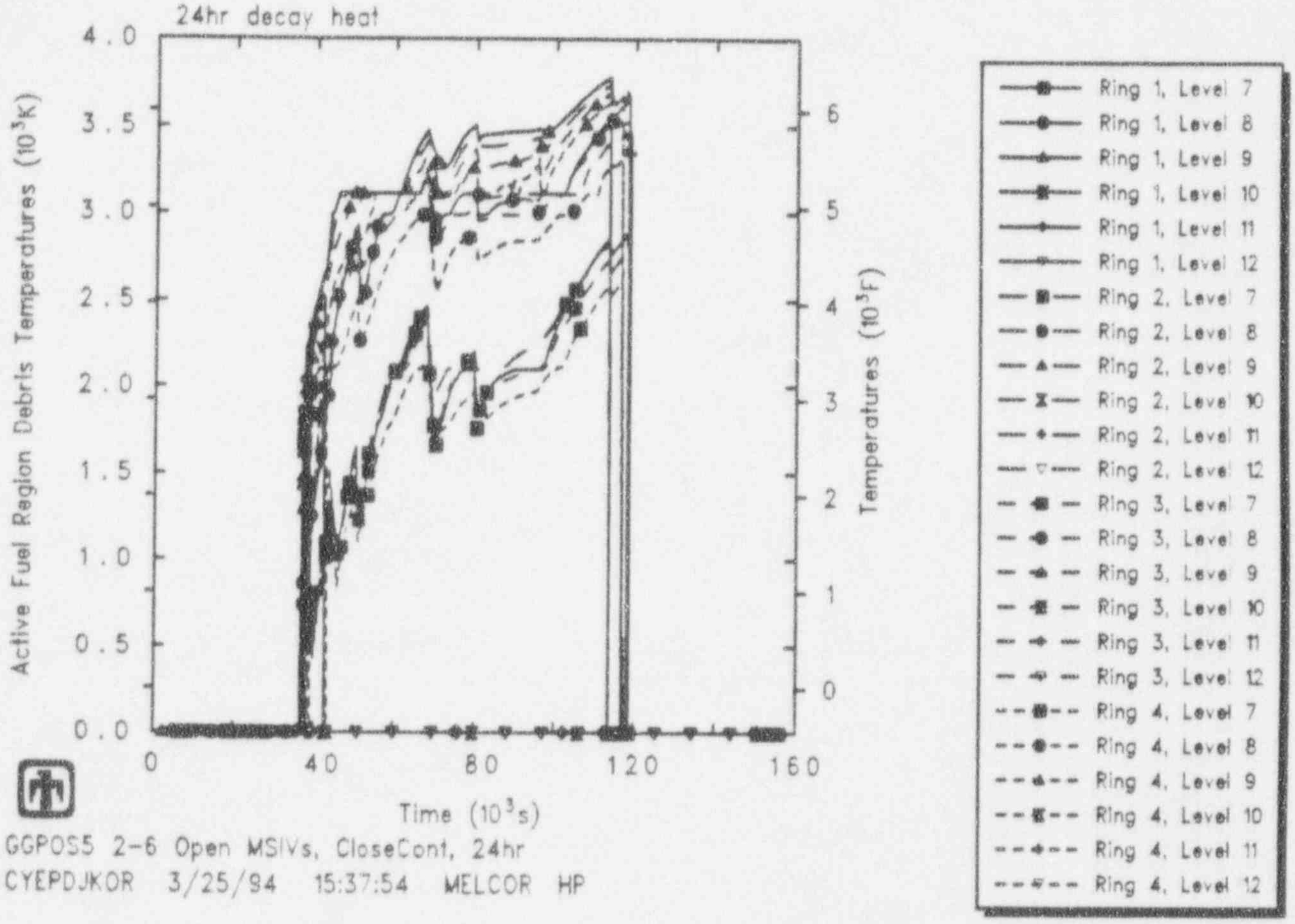


Figure 4.3.6.4. Core Active Fuel Region Debris Bed Temperatures for Grand Gulf POS 5 - Open MSIV's with Closed Containment, Initiated 24 hr After Shutdown

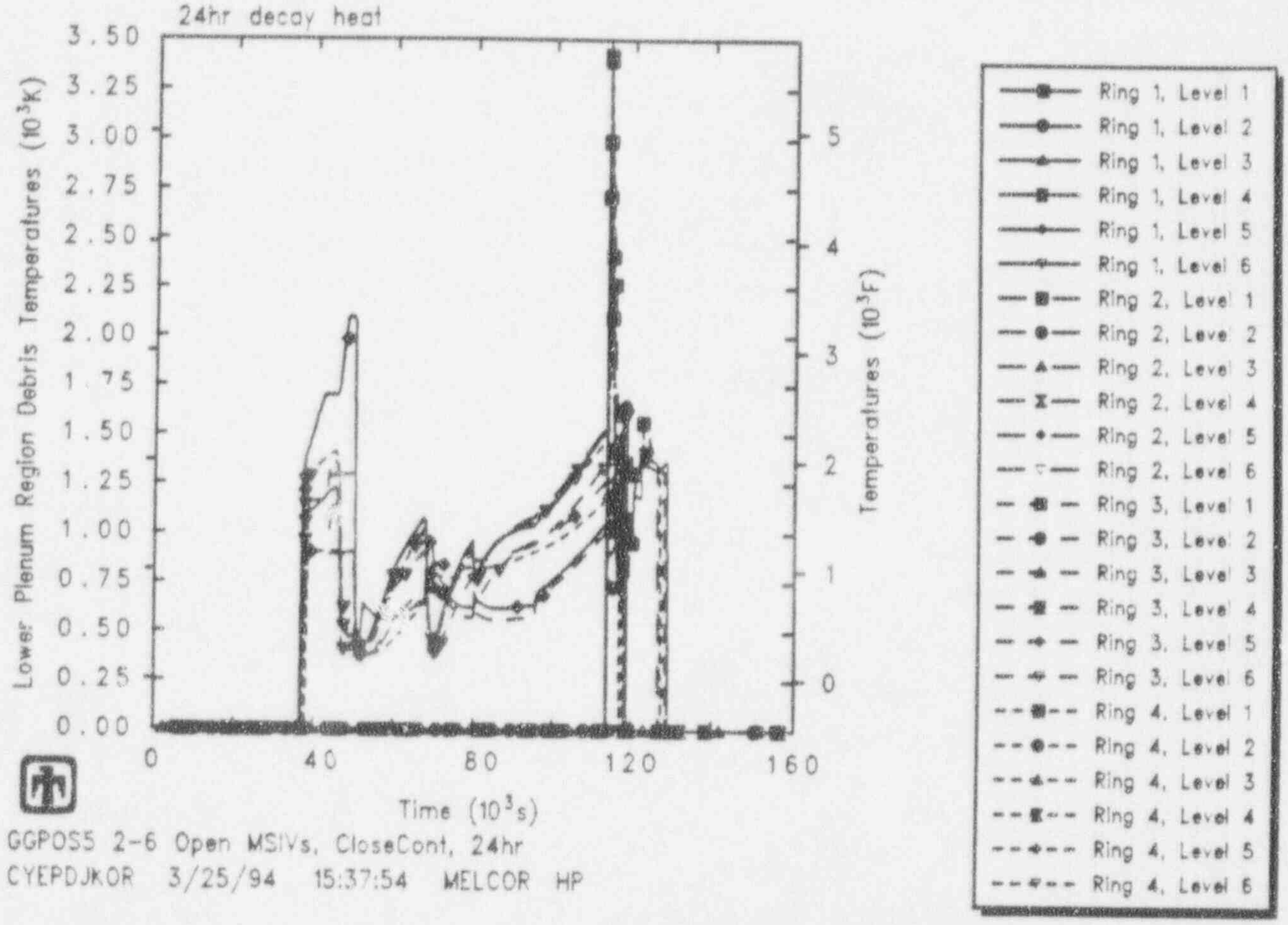


Figure 4.3.3.5. Core Lower Plenum and Core Support Plate Debris Bed Temperatures for Grand Gulf POS 5 – Open MSIVs And Closed Containment, Initiated 24 hr After Shutdown

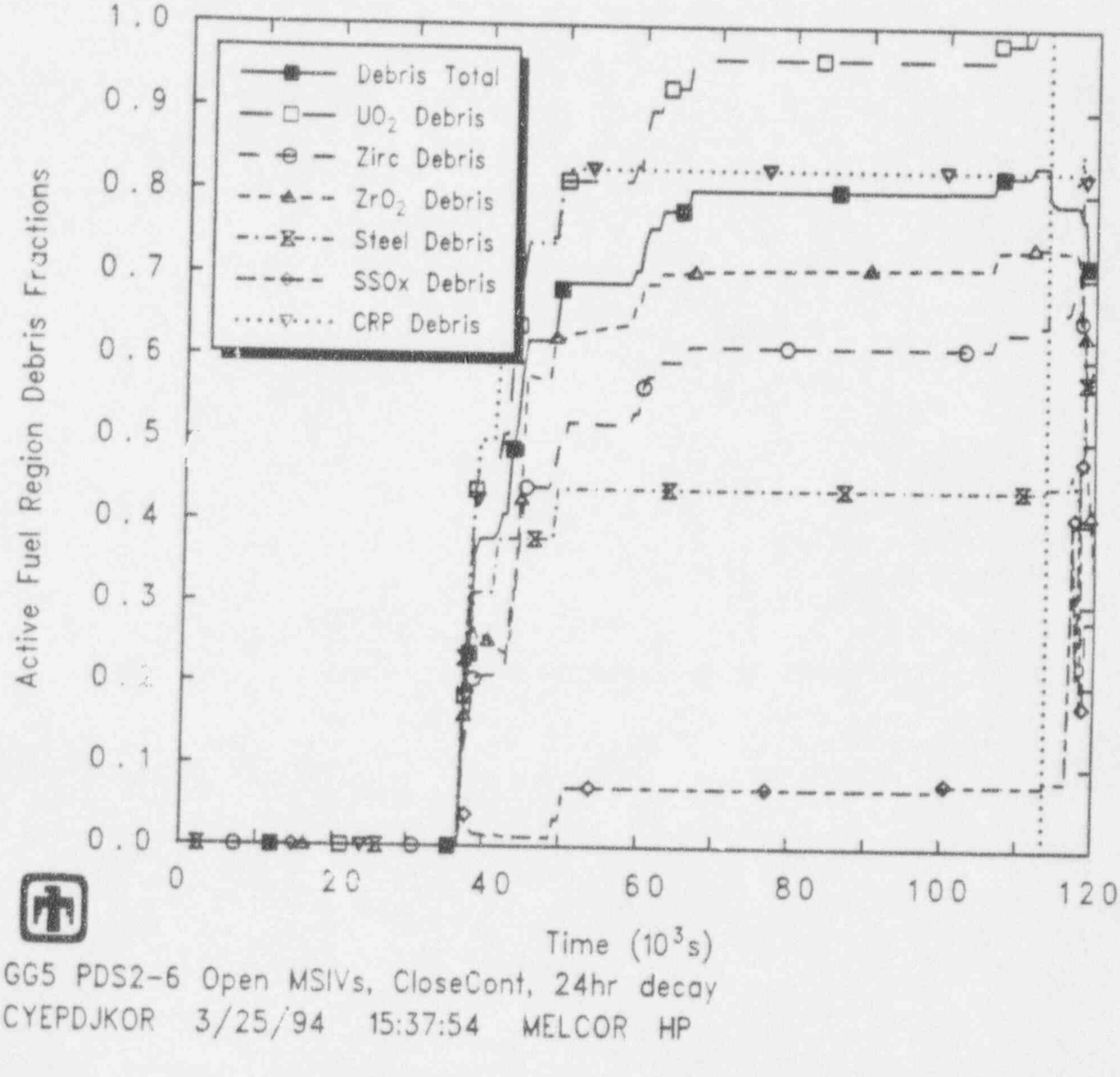


Figure 4.3.6.6. Core Active Fuel Region Degraded Material Fractions for Grand Gulf POS 5 - Open MSIV's with Closed Containment, Initiated 24 hr After Shutdown

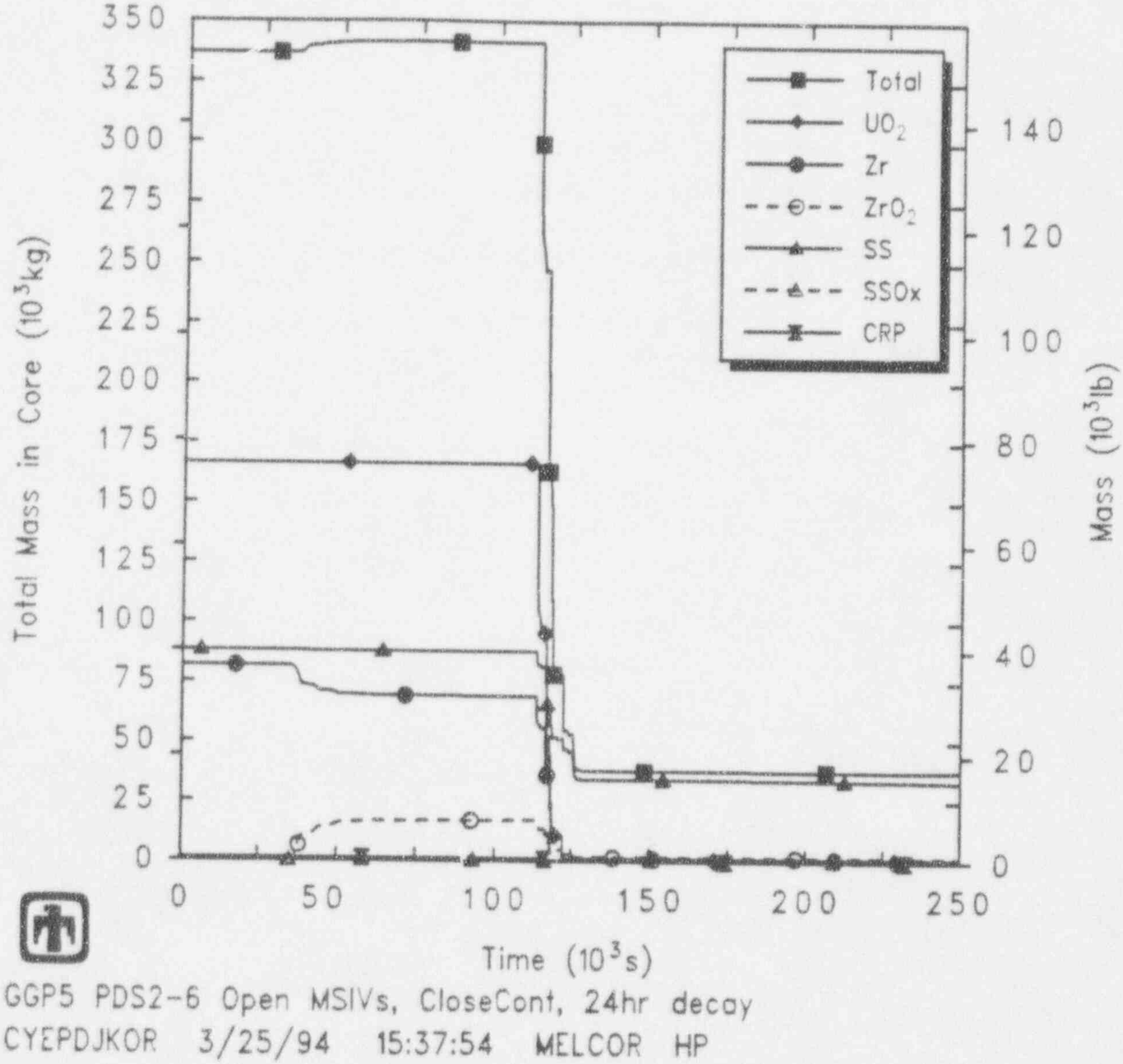


Figure 4.3.6.7. Total and Individual Core Material Masses for Grand Gulf POS 5 - Open MSIVs with Closed Containment, Initiated 24 hr After Shutdown

The debris material lost from the vessel is ejected to the drywell pedestal cavity. Figure 4.3.6.8 presents the amounts of ejected core debris, concrete ablated and the total cavity debris mass (*i.e.*, core debris combined with concrete ablation products). As in the other sequences analyzed, concrete ablation is quite rapid soon after debris ejection while the core debris is hot (>2000 K) and consists of a layer of metallic debris above a heavy oxide layer, and then slows noticably after enough concrete has been ablated for the debris bed configuration to invert to a light oxide layer above a layer of metallic debris, mixed to a lower average temperature of ~1500 K.

The calculated production of steam and noncondensable gases (H_2, CO, CO_2 and H_2O) is depicted in Figure 4.3.6.9. The hydrogen production shown includes both in-vessel production (the initial step increase) and ex-vessel production in the cavity (the later-time increase). The in-vessel hydrogen generation corresponds to the oxidation of about 10-20% of the zircaloy and about 1% of the steel in the core and lower plenum, prior to vessel failure and debris ejection. As soon as the core debris enters the cavity, core-concrete interaction begins, resulting in the production of carbon dioxide and hydrogen; reduction of these gases by the molten metal in the core debris also gives rise to carbon monoxide and hydrogen.

The mole fractions in the containment dome and in the auxiliary building (first, second and fourth floors) are shown in Figure 4.3.6.10, including vertical dotted lines at TAF uncovery and at vessel failure for reference. The mole fractions in the drywell, cavity and containment equipment hatch resemble the behavior shown for the containment dome, while the behavior in the third floor of the auxiliary building resembles the results shown for the second floor. Before vessel failure, the containment atmosphere consists of air with some steam vented out the open MSIV line to the auxiliary building and back into the containment; after vessel failure and debris ejection, the containment atmosphere consists of nearly early parts of steam, air and the noncondensable gases generated by core-concrete interaction. The open MSIV line vents to the third floor of the auxiliary building, causing a high concentration of steam to build up on the second and third floors after the containment is closed and before the vessel fails; after vessel failure, noncondensable gases generated by core-concrete interaction are added to the atmosphere, transported from the cavity into the vessel through the failed lower head, up through the vessel and out the open MSIV line to the auxiliary building. With the containment equipment hatch and upper personnel lock closed, the fourth floor is a dead-end volume resembling the first floor.

Figure 4.3.6.11 illustrate the time-dependent release of radionuclides from the fuel debris both within the vessel and in the cavity. The vertical dotted lines within the plots mark the time of vessel failure, indicating that most of the in-vessel release occurs prior to vessel failure, from the hot debris bed in the active fuel region, while most of the ex-vessel release occurs within a short time period after vessel failure and debris ejection to the cavity, while the core debris is still hot, before enough concrete has been ablated for the debris bed configuration to cool and invert; this behavior is seen in most of our MELCOR analyses. Table 4.3.6.2 summarizes the in-vessel, ex-vessel and total amounts of each radionuclide class released, all normalized to the initial inventories of each class.

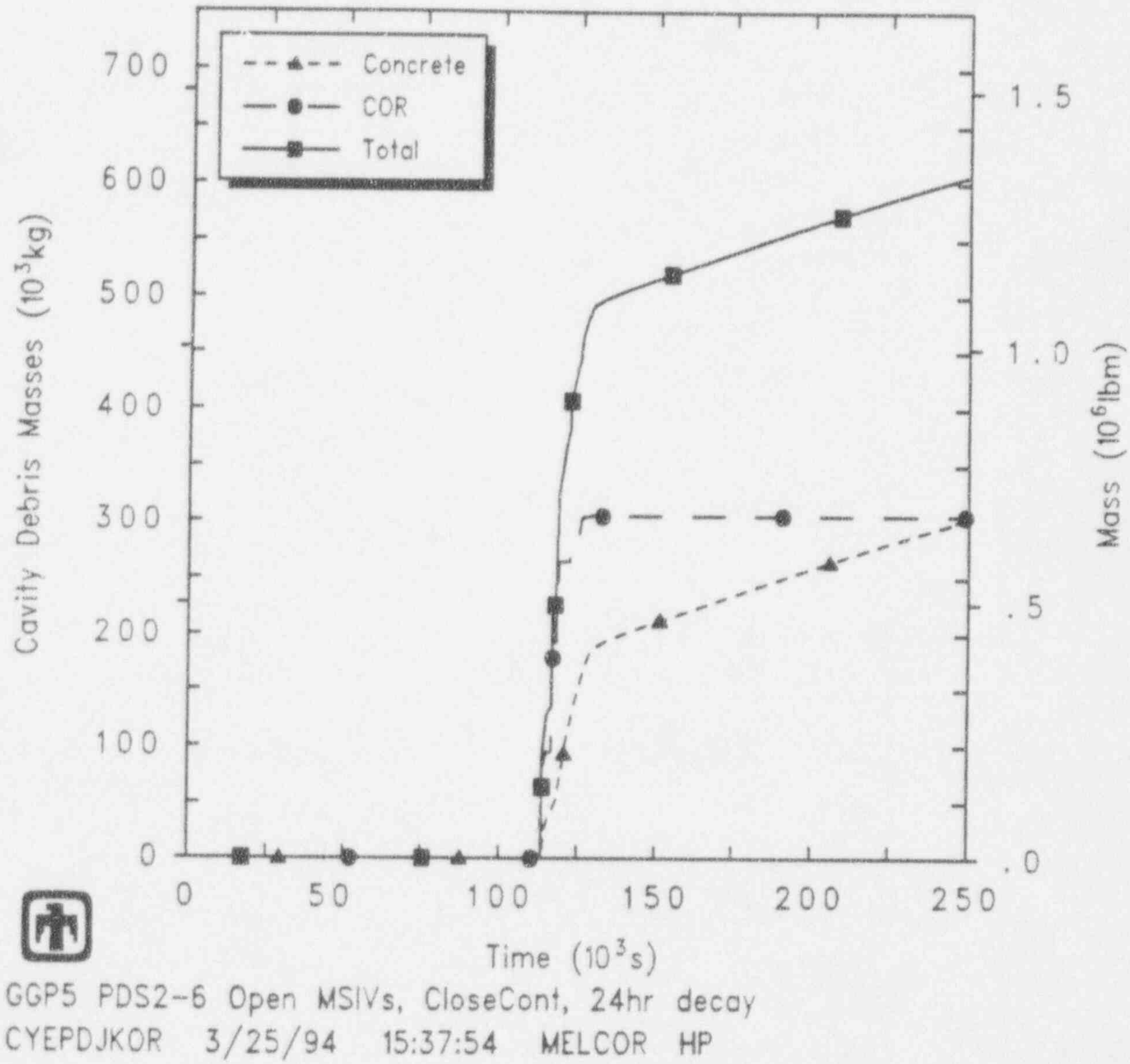


Figure 4.3.6.8. Cavity Total and Core and Concrete Debris Masses for Grand Gulf POS 5 - Open MSIV's with Closed Containment, Initiated 24 hr After Shutdown

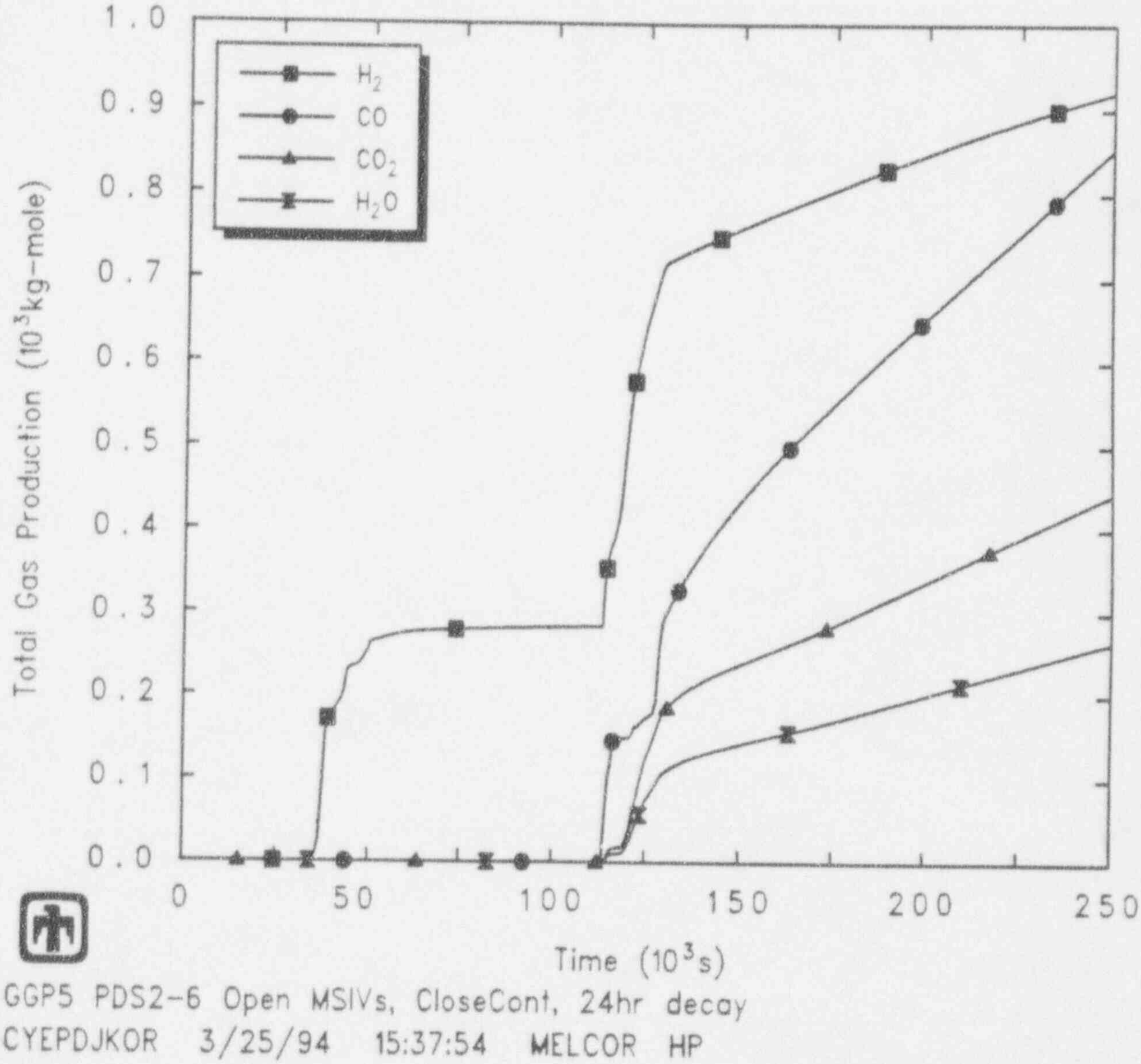


Figure 4.3.6.9. Hydrogen, Carbon Monoxide, Carbon Dioxide and Steam Generation for Grand Gulf POS 5 - Open MSIV's with Closed Containment, Initiated 24 hr After Shutdown

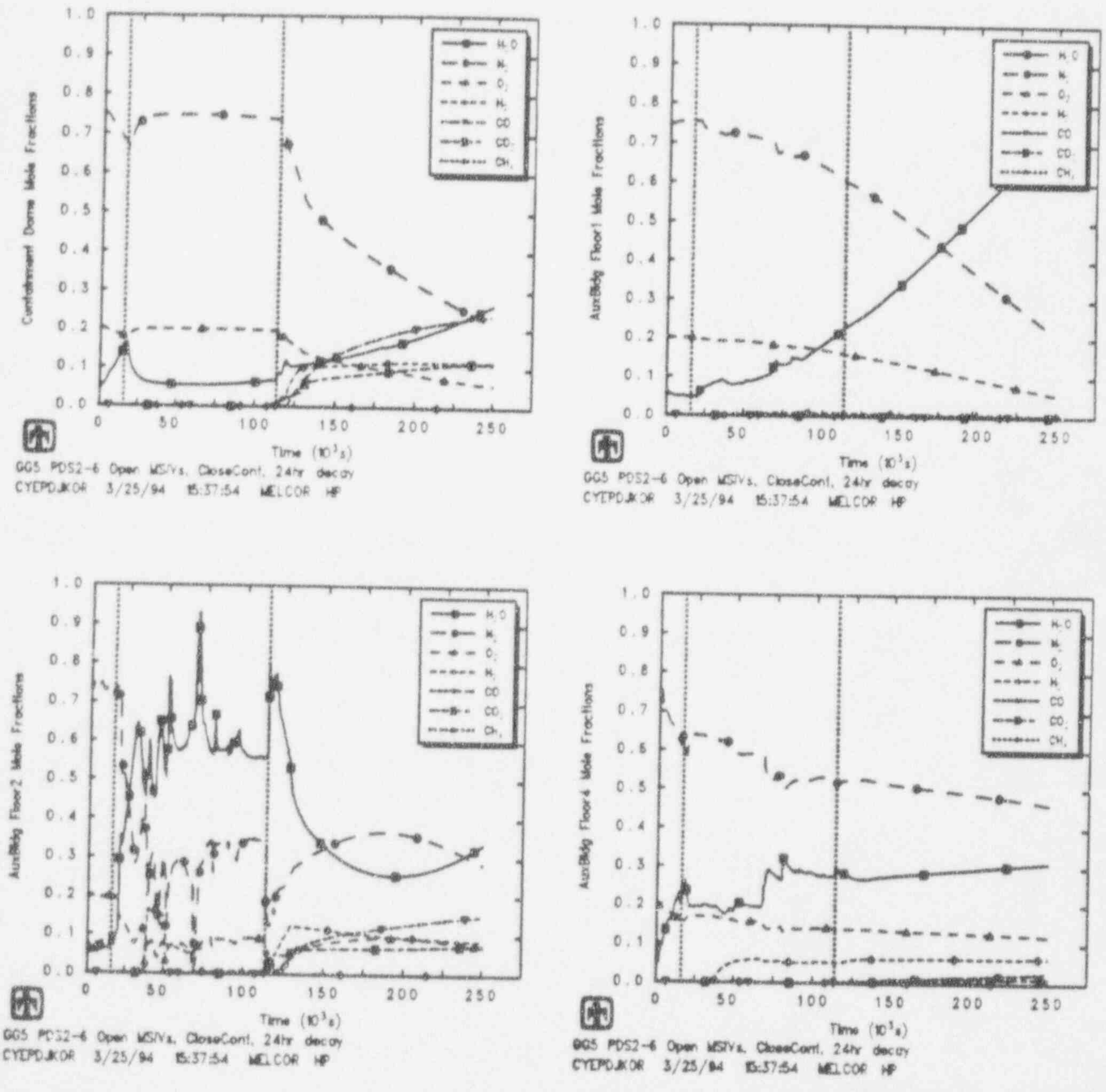


Figure 4.3.6.10. Mole Fractions in Containment Dome (upper left) and Auxiliary Building First (upper right), Second (lower left) and Fourth Floors (lower right) for Grand Gulf POS 5 - Open MSIVs with Closed Containment, Initiated 24 hr After Shutdown

(Note that th amounts generally consider only the release of radioactive forms of these classes, and not additional releases of nonradioactive aerosols from structural materials.)

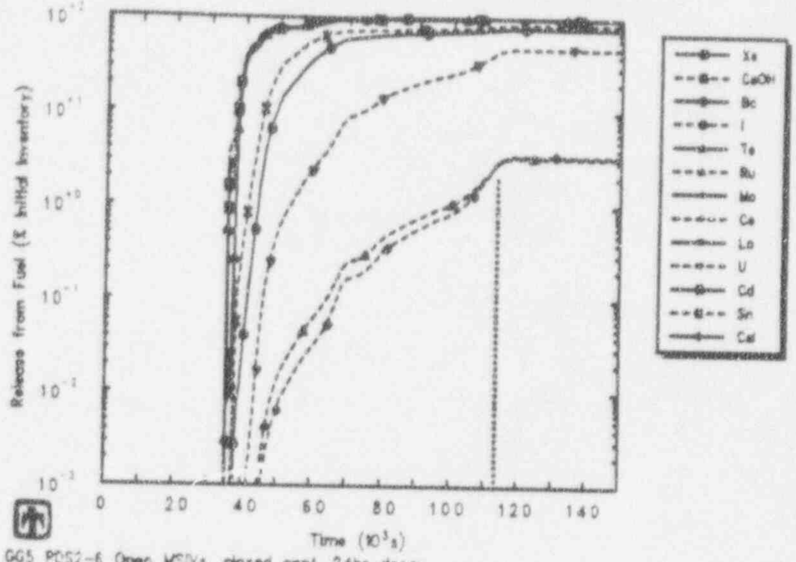
The release behavior predicted by MELCOR is somewhat different for this scenario that for the others analyzed. In all cases, almost all (~100%) of the volatile Class 1 (noble gases), Class 2 (CsOH), Class 5 (Te) and Class 16 (CsI) radionuclide species are released, primarily in-vessel, as are most (80-90%) of the Class 3 (Ba) and Class 12 (Sn) inventories. The next major release fraction is for uranium. Around 2-5% of the total inventories of Ru and Mo, Ce and La, are released. Finally, a total ≤0.05% of the initial inventory of Class 11 (Cd) is predicted to be released. Note that the CORSOR-M fission product release model option used in these analyses has identically zero release in-vessel of Class 7 (Mo), Class 9 (La) and Class 11 (Cd).

Figure 4.3.6.12 gives the total radioactive release to the environment, while the release fractions of individual classes to the environment are shown in Figure 4.3.6.13. The release to the environment does not begin at vessel failure in this sequence, but earlier after auxiliary building failure. Closing containment is an ineffective measure in this scenario unless the MSIVs are also closed.

...These environmental releases do not correspond to immediate release of all radionuclides released from the fuel; there is considerable retention of most radionuclide species within the containment (but not within the isolated auxiliary building). Almost all the noble gases (~100%) are released to the environment soon after containment fails; in addition, there is some release to the environment of the other volatile species (*i.e.*, CsOH, CsI and Te) also.

Table 4.3.6.3 summarizes the distribution of the initial radionuclide inventory at the end of the calculation, and provides an overview of how much of the radionuclides remain bound up in fuel debris in either the core or the cavity, and of how much of the released radionuclides are retained in the primary system *vs* how much of the released radionuclides are released to, or released in, either the containment or the auxiliary building and the environment, all normalized to the initial inventories of each class. Table 4.3.6.4 presents a different breakdown of the released radionuclide final distribution, giving the fractions of released inventory for each class in control volume atmospheres (including the environment), in pools, or deposited or settled onto heat structures at the end of the calculations. (As in Table 4.3.6.2, these amounts consider only the release of radioactive forms of these classes, and not additional releases of nonradioactive aerosols from structural materials.)

These tables show fission product distributions generally similar to those found for the station blackout sequence with failure to isolate SDC (discussed in Section 4.3.2) for the radionuclides with significant (≥80% of initial inventory) release from fuel. Most of the fission product release occurs in-vessel prior to vessel failure in all cases, and both these sequences vent from the vessel directly to the auxiliary building, either through the SDC break or through the open MSIV line, before vessel failure. In all the accident scenarios analyzed, most of the noble gases released are in the environment, in the atmosphere. In both scenarios venting directly to the auxiliary building most of the volatile species



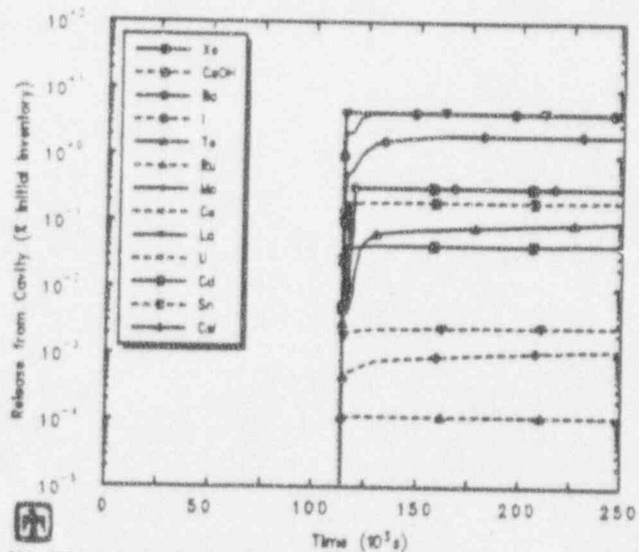


Figure 4.3.6.11. In-Vessel (top) and Ex-Vessel (bottom) Radionuclide Release Fractions for Grand Gulf POS 5 - Open MSIVs with Closed Containment, Initiated 24 hr After Shutdown

Table 4.3.6.2. Final Radionuclide Release Fractions for Grand Gulf POS 5 - Open MSIV's with Closed Containment. Initiated 24 hr After Shutdown

Class	Fission Products Released from Fuel (% Initial Inventory)		
	In-Vessel	Ex-Vessel	Total
Xe	99.64	0.32	99.96
CsOH	99.65	0.31	99.96
Ba	78.23	4.33	82.56
I	~0	~0	~0
Te	99.60	0.098	99.61
Ru	3.299	0.00011	3.30
Mo	0	1.98	1.98
Ce	3.444	0.0011	3.45
La	0	4.425	4.43
U	47.60	0.0026	47.60
Cd	0	0.045	0.045
Sn	85.99	0.204	86.20
CsI	99.67	0.32	99.99

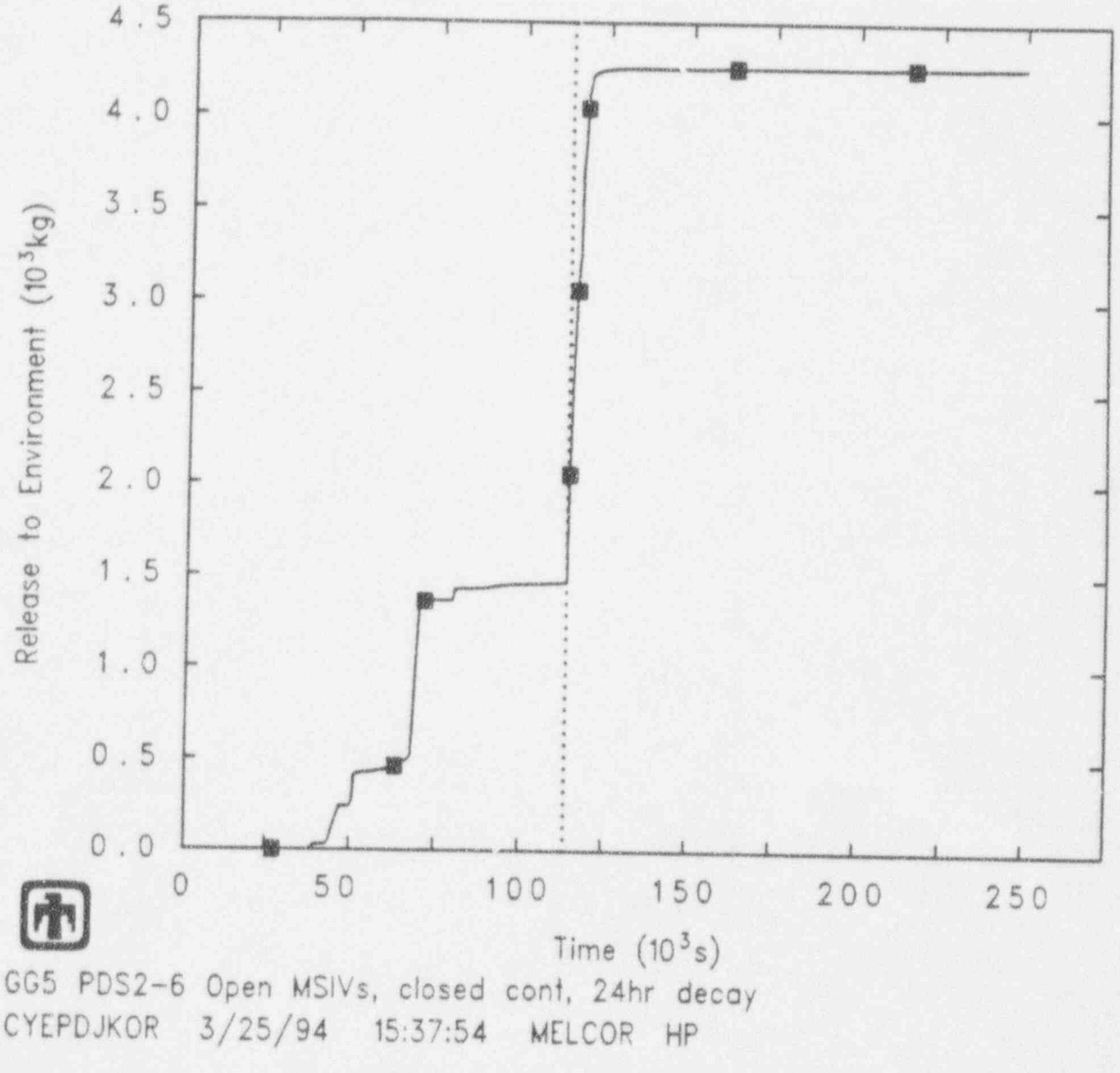


Figure 4.3.6.12. Total Environmental Radionuclide Releases for Grand Gulf POS 5 – Open MSIVs with Closed Containment, Initiated 24 hr After Shutdown

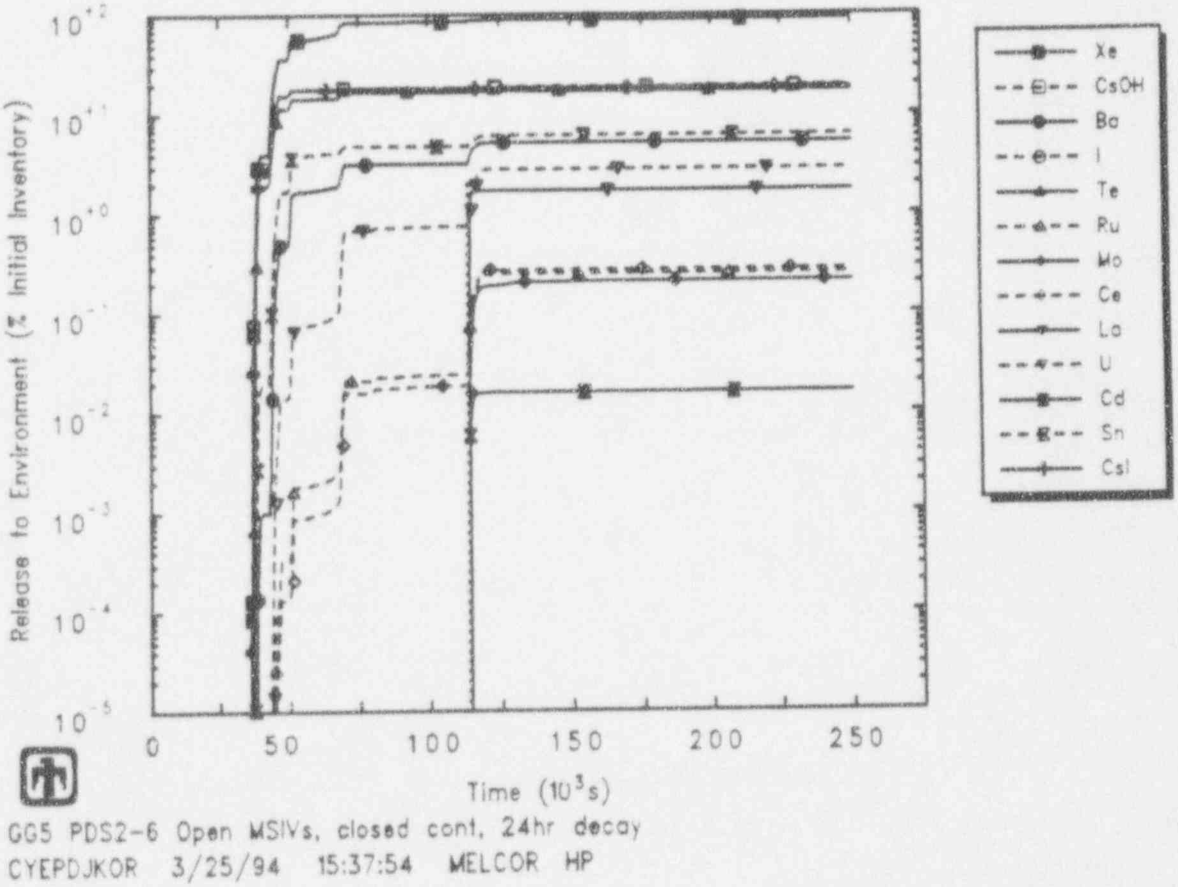


Figure 4.3.6.13. Environmental Radionuclide Release Fractions for Grand Gulf POS 5 - Open MSIVs with Closed Containment, Initiated 24 hr After Shutdown

Table 4.3.6.3. Final Radionuclide Distribution for Grand Gulf POS 5 – Open MSIVs with Closed Containment, Initiated 24 hr After Shutdown

Class	Fission Product Distribution (% Initial Inventory)				
	Fuel Debris	Primary System	Containment	Auxiliary Building	Environment
Xe	~0	0.0001	0.03	9.74	90.23
CsOH	~0	2.41	0.54	78.17	18.88
Ba	17.46	34.4	8.12	34.75	5.28
Te	0.237	2.73	0.36	79.21	17.43
Ru	96.7	1.82	0.453	0.775	0.252
Mo	98.0	0.112	1.14	0.519	0.212
Ce	96.6	1.89	0.532	0.753	0.275
La	95.6	0.292	1.21	1.18	1.74
U	56.1	24.66	4.62	11.96	2.62
Cd	99.9	0.0002	0.036	0.013	0.0016
Sn	13.8	32.8	7.67	40.36	6.19
CsI	~0	2.534	0.43	78.51	18.53

Table 4.3.6.4. Final Radionuclide State for Grand Gulf POS 5 - Open MSIVs and Closed Containment, Initiated 24 hr After Shutdown

Class	Fission Products Released from Fuel (% Released Inventory)		
	Atmosphere	Pool	Deposited
Xe	~100	0	0
CsOH	19.00	44.34	36.53
Ba	6.39	21.83	71.63
I	~100	0	0
Te	17.45	43.08	39.34
Ru	7.64	9.63	82.67
Mo	10.78	38.15	51.05
Ce	7.99	9.16	82.80
La	39.20	12.9	47.88
U	5.97	11.56	82.40
Cd	37.87	14.47	47.63
Sn	7.18	24.87	67.82
CsI	18.44	44.36	36.80

(CsOH, CsI and Te) released in-vessel are retained in the auxiliary building: the two classes of radionuclides forming aerosols which had substantial releases (Ba and Sn) were predicted to have about half those releases retained in the vessel, primarily deposited on structures, and the other half of the releases retained in both the containment and in the auxiliary building (with a slightly higher percentage retained in the auxiliary building compared to the containment).

4.3.7 Large Break LOCA with Flooded Containment and with Hydrogen Igniters, Initiated 7 hr After Shutdown

The analysis of the large break LOCA scenario with flooded containment initiated 7 hr after shutdown described in Section ?? was repeated with the hydrogen ignition system assumed functional. Igniters were modelled in every control volume in both the inner and outer containments.

The amounts of hydrogen and carbon monoxide burned in each control volume in the containment are shown in Figures 4.3.7.1 and 4.3.7.2. While combustion occurs throughout the containment, most of the hydrogen and carbon monoxide combustion occurs in the containment dome. In the first portion of the transient only hydrogen is produced, through metal-water reaction in the vessel, so only hydrogen is burned; later, after vessel failure at 92,500 s (25.7 hr), carbon monoxide generated by core-concrete interaction is burned also. Comparison to the total hydrogen and carbon monoxide production given in Figure 4.3.1.20 shows just over 50% of the hydrogen produced and just over 25% of the carbon monoxide generated is burned.

The combustion can be seen to occur in stepped stages. Each such set of burns generates large pressure and temperature excursions in the containment and, through the open equipment hatch, personnel locks and the recirculation pipe break generates large pressure and temperature excursions in the auxiliary building and vessel also. Figure 4.3.7.3 illustrates one impact of hydrogen ignition: the auxiliary building fails much earlier than in the same sequence with no hydrogen combustion, at 38,334 s (10.65 hr) on a sharp pressure spike due to combustion in the containment instead of about 7 hr after vessel failure, at 117,500 s (32.6 hr), due to pressurization by noncondensable gases generated during core concrete interaction in the cavity.

Figures 4.3.7.4 and 4.3.7.5 depict the magnitude of the temperature excursions generated by combustion in the containment dome and auxiliary building, respectively.

The combustion has the general effect of reducing the mole fractions of hydrogen and carbon monoxide present. Figures 4.3.7.6 and 4.3.7.7 present the mole fractions in the containment dome and in the second floor of the auxiliary building, respectively, comparing the results obtained with working igniters with the results assuming no hydrogen combustion. The behavior predicted in the containment equipment hatch is almost identical to that shown for the containment dome. The behavior predicted in the top two floors of the auxiliary building is very similar to that shown for the second floor: the first floor is a dead-end volume and remains more near ambient. Figure 4.3.7.6 shows

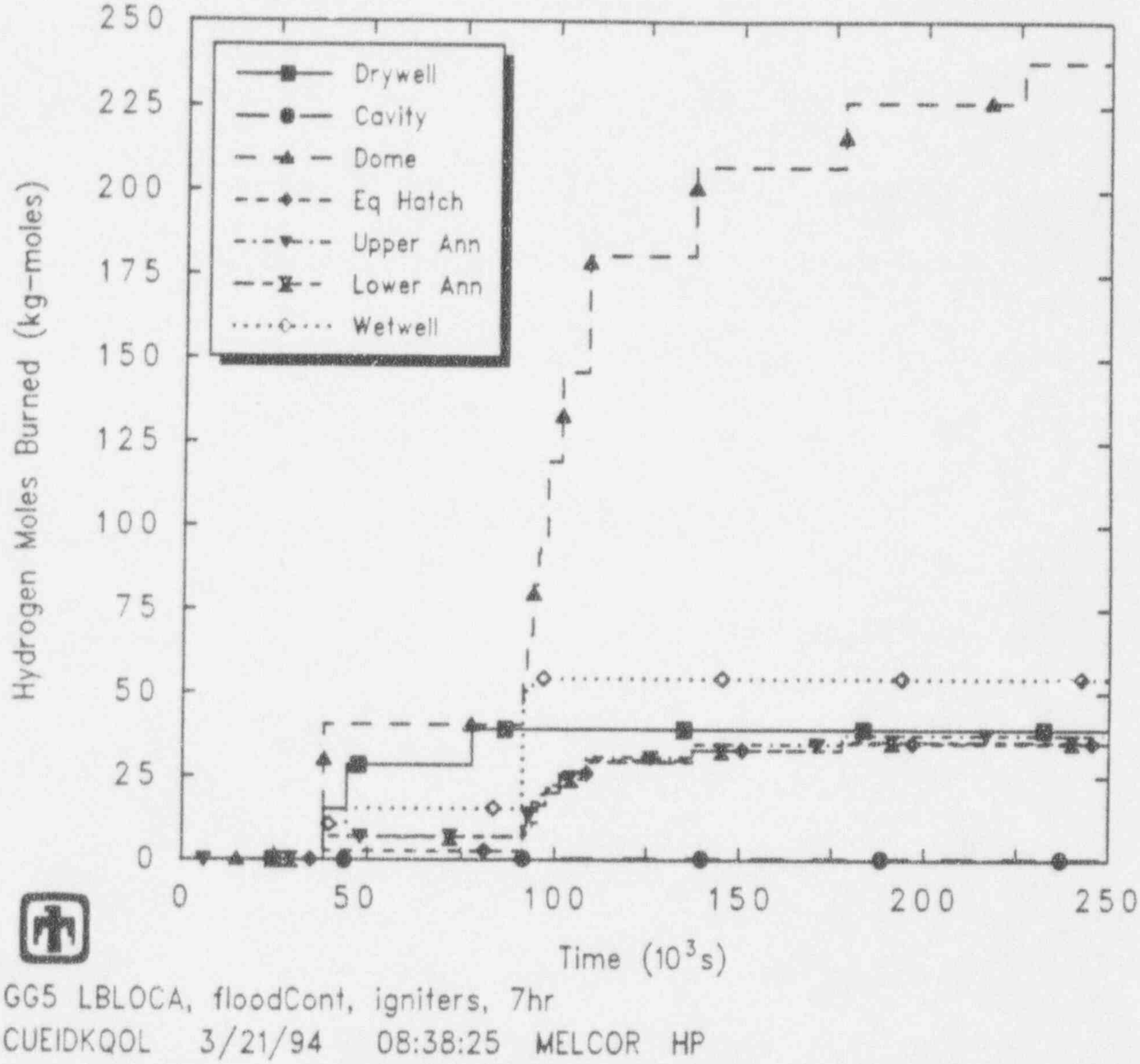


Figure 4.3.7.1. Hydrogen Combustion Generation for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, with Hydrogen Ignition, Initiated 7 hr After Shutdown

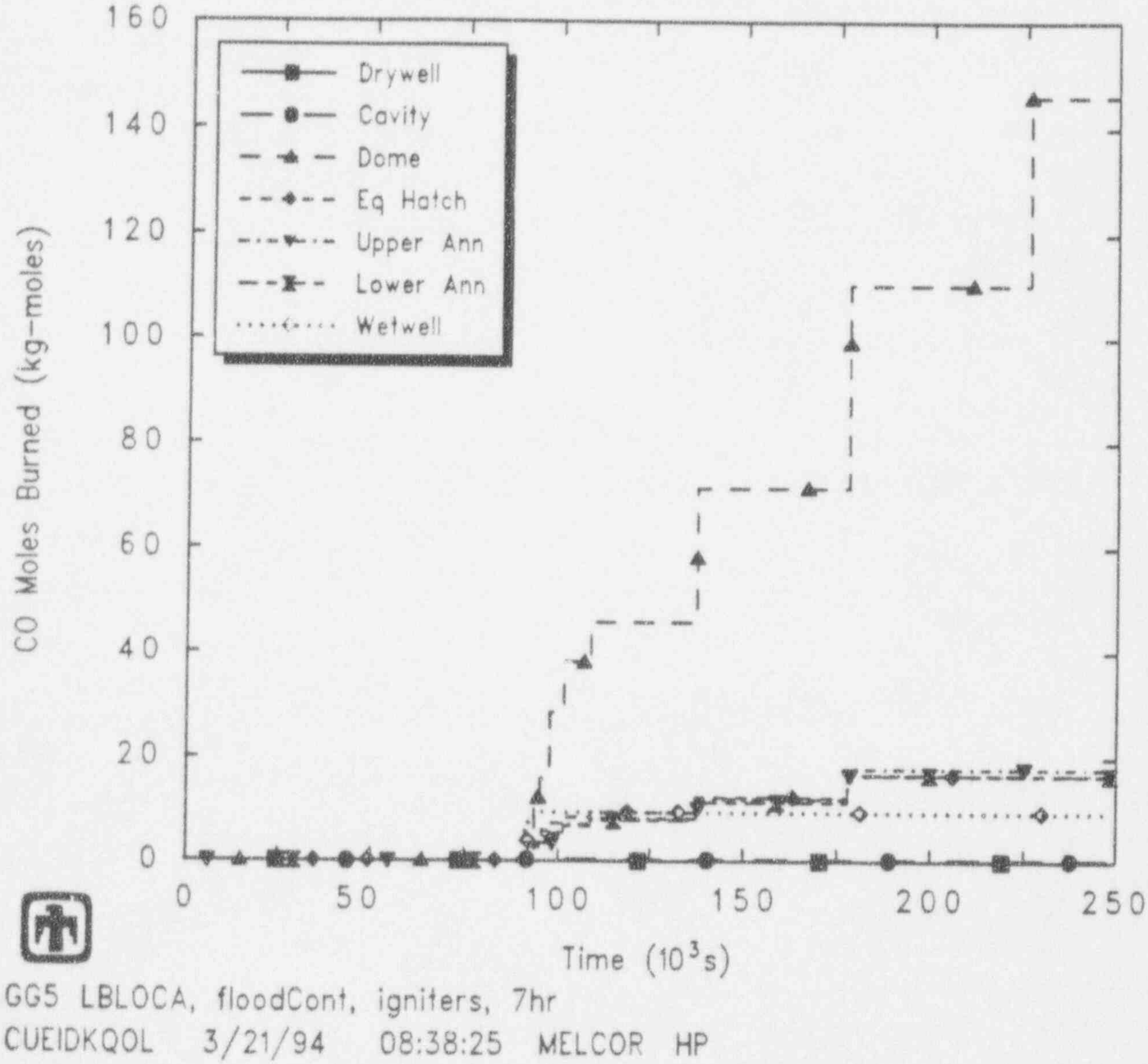


Figure 4.3.7.2. Carbon Monoxide Combustion Generation for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, with Hydrogen Ignition, Initiated 7 hr After Shutdown

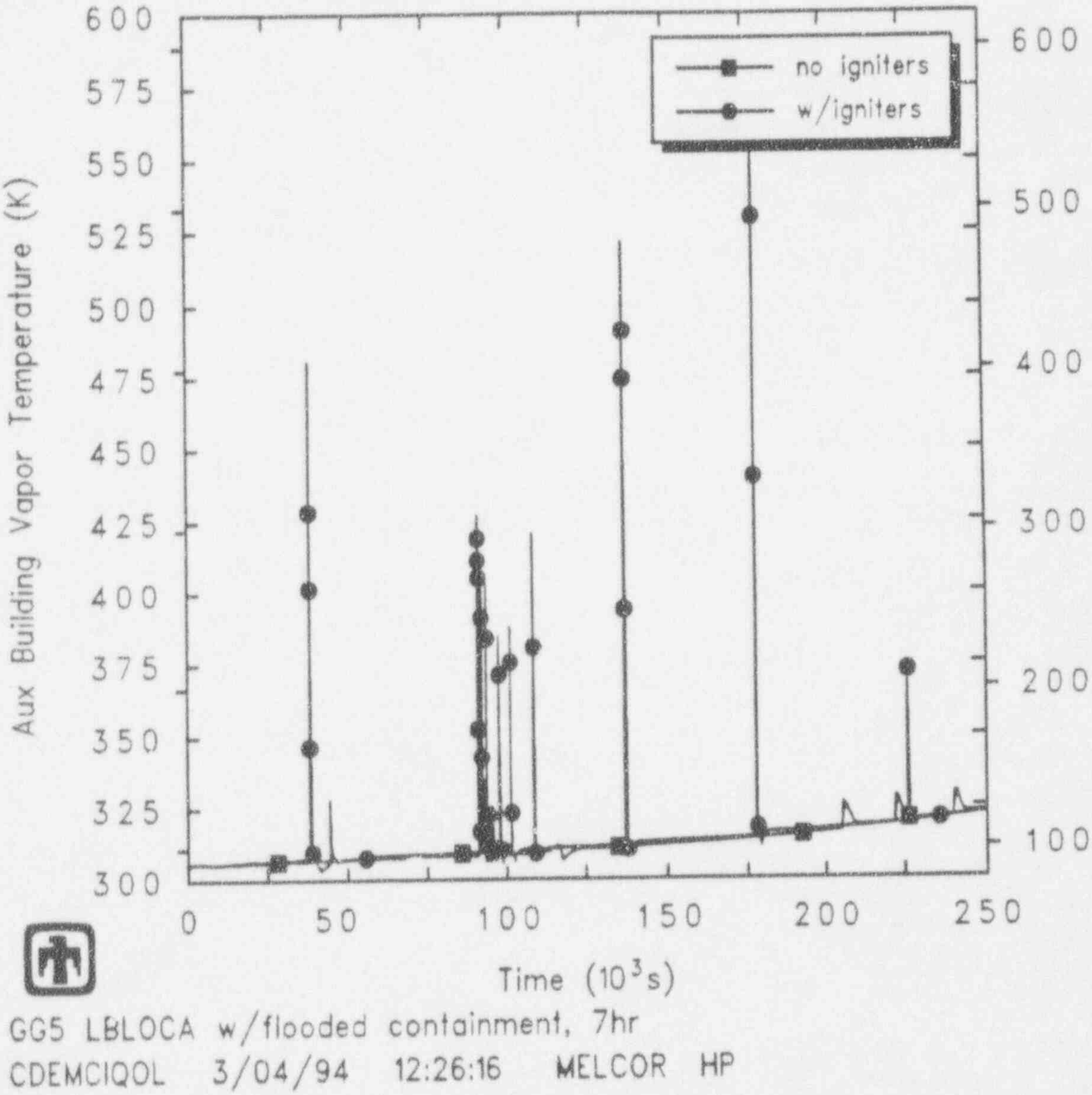


Figure 4.3.7.5. Auxiliary Building Temperatures for Grand Gulf POS 5 - Large Break LOCA with Flooded Containment, with Hydrogen Ignition and without Hydrogen Combustion, Initiated 7 hr After Shutdown

the combined steam and carbon dioxide mole fraction instead of just the steam mole fraction, because the sum determines whether the volume is inert, and also includes the mole fraction ignition limits. (In the MELCOR calculation, burn occurs in volumes with igniters if $x_{H_2} \geq 0.07$, $x_{CO} \geq 0.129$, $x_{O_2} \geq 0.05$, and $x_{H_2O} + x_{CO_2} \leq 0.55$.)

personnel locks and the containment equipment hatch are closed). This calculation did not include the auxiliary building model, but would vent directly to the environment after containment failure would occur.

3. The accident is initiated 4 days after reactor scram. The hydrogen ignition system is unavailable during the accident. Both the containment personnel locks are open; the containment equipment hatch is also open. The open auxiliary building model (which assumes some of the interior doors are open) is included in this calculation, with failure on a 5psi overpressure.

4. The accident is initiated 4 days after reactor scram. The hydrogen ignition system is unavailable during the accident. Both the containment personnel locks are open; the containment equipment hatch is also open. The closed auxiliary building model (which assumes some of the interior doors are closed) is included in this calculation, with failure on a 5psi overpressure.

5. The accident is initiated 4 days after reactor scram. The hydrogen ignition system is unavailable during the accident. The open auxiliary building model is included in this calculation, with failure on a 5psi overpressure. The containment equipment hatch is open; however, both of the containment personnel locks are closed.

6. The accident is initiated 4 days after reactor scram. The hydrogen ignition system is unavailable during the accident. The closed auxiliary building model is included in this calculation, with failure on a 5psi overpressure. The containment equipment hatch is open; however, both of the containment personnel locks are closed.

7. The accident is initiated 15 days after reactor scram. The hydrogen ignition system is unavailable during the accident. Both of the containment personnel locks are open; the containment equipment hatch is also open. This calculation did not include the auxiliary building model, but vented directly to the environment.

8. The accident is initiated 4 days after reactor scram. The hydrogen ignition system is operational during the accident. The containment is isolated (*i.e.*, the containment personnel locks and the containment equipment hatch are closed). This calculation did not include the auxiliary building model, but was assumed to vent directly to the environment after containment failure would occur.

In addition, a few sensitivity studies were done on various code options and/or parameters. In one calculation, the CORSOR fission product release model was used instead of the (MELCOR default) CORSOR-M fission product release model. Because it was sometimes necessary to back up and reduce the user-specified maximum time step in order to avoid a code abort and complete the analysis, a calculation was done in which that was the only change made, to determine how big an effect reducing the time step would have on the results.

Two calculations were done to address concerns [58, 59] raised about the lack of air oxidation modelling in MELCOR, and the associated lack of extensive release of ruthenium demonstrated to occur when irradiated reactor fuel is heated in air.

Table 5.2.1. Key Event Times for Grand Gulf POS 6 - Reference Calculation

Event	Time
Level below TAF	13.04 hr
Clad failure/Gap release	
(Ring 1)	18.80 hr
(Ring 2)	18.76 hr
(Ring 3)	18.81 hr
(Ring 4)	19.04 hr
(Ring 5)	19.93 hr
(Ring 6)	23.22 hr
Auxiliary building failure	20 hr
Vessel LH penetration failure	
(Ring 1)	24.52 hr
(Ring 2)	24.74 hr
(Ring 3)	25.49 hr
(Ring 4)	26.50 hr
(Ring 5)	27.87 hr
(Ring 6)	30.22 hr
Cavity rupture	91.41 hr

5.2 Reference Analysis

The calculation selected as the POS 6 reference, base case, analysis has the accident initiated 4 days after reactor scram. The hydrogen ignition system is unavailable during the accident. Both the containment personnel locks are open; the containment equipment hatch is also open. The closed auxiliary building model (which assumes some of the interior doors are closed) is included in this calculation, with failure on a 5 psi overpressure. The timing of key events as predicted in this reference analysis is presented in Table 5.2.1.

At the start of the accident, the primary system (*i.e.*, reactor vessel), containment and auxiliary building are all assumed to be at atmospheric pressure. The vessel is filled with water at 333 K (140 °F) to an elevation of 16.13 m, corresponding to the bottom of the main steam lines. The only assumption in the accident is no intervention, either manual or automatic.

Figure 5.2.1 presents the pressures calculated in various regions of the reactor vessel. A pressure gradient develops immediately, representing simply the head of the liquid water; thus, the lower plenum exhibits the highest pressure, the core and bypass the next highest, and the downcomer and upper plenum pressures nearest atmospheric. The vessel water mass predicted to remain at any given time is given in Figure 5.2.2. As the water inventory is steamed away by the core decay heat, the pressure gradient in the vessel diminishes due to the decreasing pressure head.

The vessel pressure does not drop to atmospheric as the liquid water inventory decreases but instead equilibrates to the containment pressure, shown in Figure 5.2.3. The containment (and the auxiliary building, whose pressure is virtually identical to the containment pressure) pressurizes rapidly as steam generated in the core rises in the vessel and flows out into containment through the removed upper head region. Figure 5.2.4 depicts that steam flow from the vessel out to containment through the removed upper head opening (as well as the breach flow when the vessel first fails at about 25 hr, when Figure 5.2.2 indicates most of the remaining vessel liquid inventory is lost very quickly).

The containment and auxiliary building are kept in pressure equilibrium by three large, open flow paths – the containment equipment hatch and the upper and lower personnel locks. The flows through these paths are illustrated in Figure 5.2.5. Throughout most of the transient, there is a substantial outflow from containment into the auxiliary building through the equipment hatch and a corresponding inflow into containment from the auxiliary building through the lower personnel lock; the flow through the upper personnel lock is more erratic, switching between periods of inflow and outflow. The auxiliary building reaches its specified 5 psi overpressure failure criterion at just over 20 hr, when the stairwell door to the environment is assumed blown open. After that, the primary system, containment and auxiliary building all remain at essentially atmospheric pressure, equilibrated with the environment. There are no substantive differences in the containment equipment hatch and the upper and lower personnel lock flows after the auxiliary building fails.

The temperatures calculated in the various reactor vessel control volume atmospheres are shown in Figure 5.2.6. The temperature remains low, at saturation, until after the top of the active fuel (TAF) is uncovered at about 13 hr; soon afterward, the temperatures rise rapidly as the core degrades. The temperature oscillations die down after vessel breach at just before 25 hr, but remain elevated throughout the transient. The temperatures calculated in the various containment control volume atmospheres are shown in Figure 5.2.7. The temperatures remain low until after more than 35 hr, when the cavity temperature rapidly rises to $\sim$1500 K, and the drywell and weirwall temperatures also rise. Figure 5.2.8 presents the atmosphere temperatures in the auxiliary building. The elevated temperatures in the drywell/cavity do not propagate through the outer containment and into the auxiliary building; the auxiliary building temperature rise remains limited on all floors.

The reactor vessel water inventory is steamed away by the core decay heat, as indicated by the vessel water mass remaining at any time given in Figure 5.2.2, and also by the primary system control volume liquid levels given in Figure 5.2.9. The top of the active fuel is uncovered at about 13 hr and the core is essentially dry at 20 hr. Most of the lower plenum inventory is lost at vessel breach at 25 hr, after which the last of the water, that trapped in the downcomer below the jet pump inlet, slowly boils away.

As the reactor vessel water is boiled away, the clad and fuel uncovered begin heating up. The clad temperature histories in the core level just below the active fuel midplane

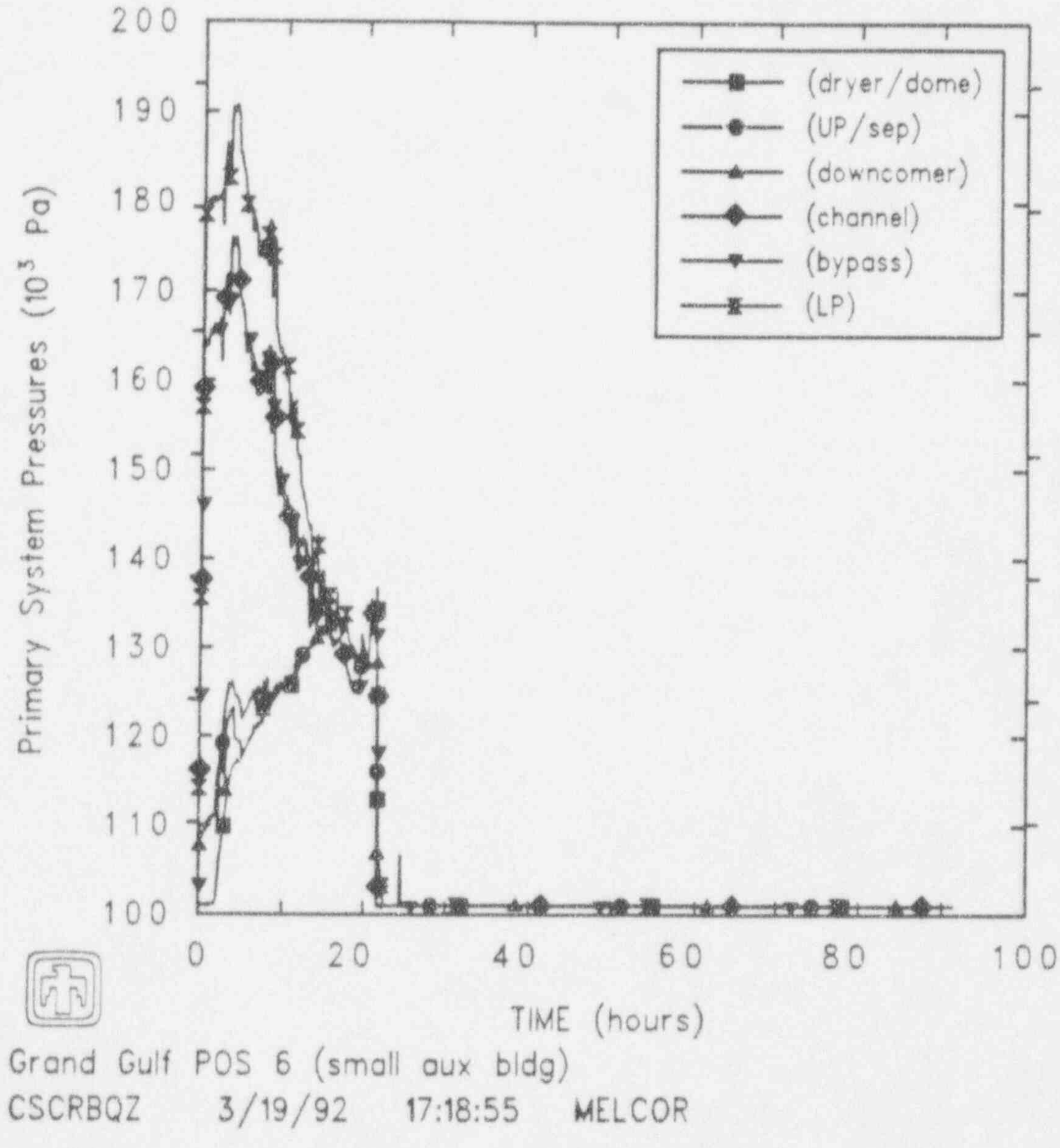


Figure 5.2.1. Reactor Vessel Pressures for Grand Gulf POS 6 - Reference Calculation

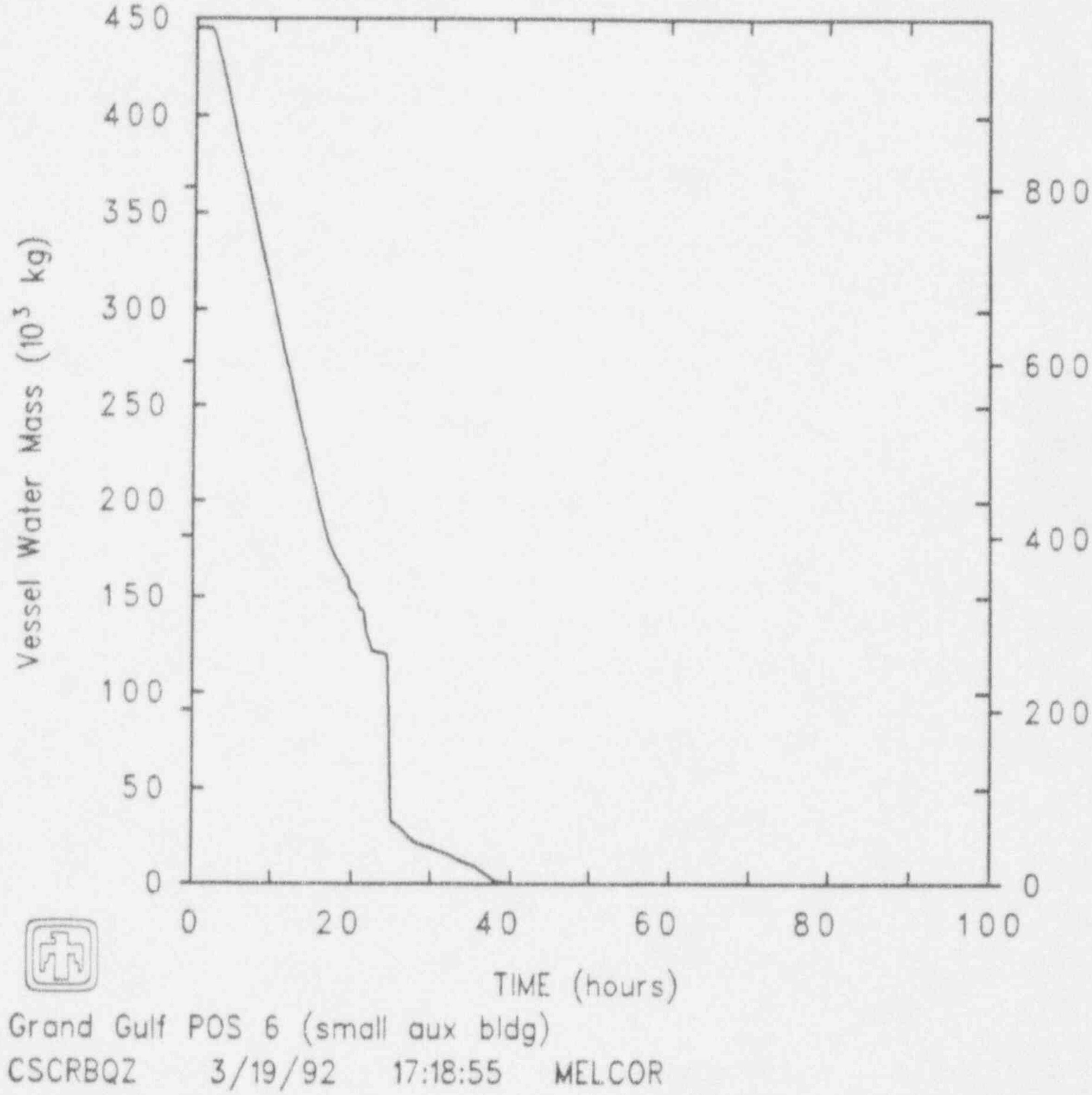


Figure 5.2.2. Reactor Vessel Water Mass for Grand Gulf POS 6 – Reference Calculation

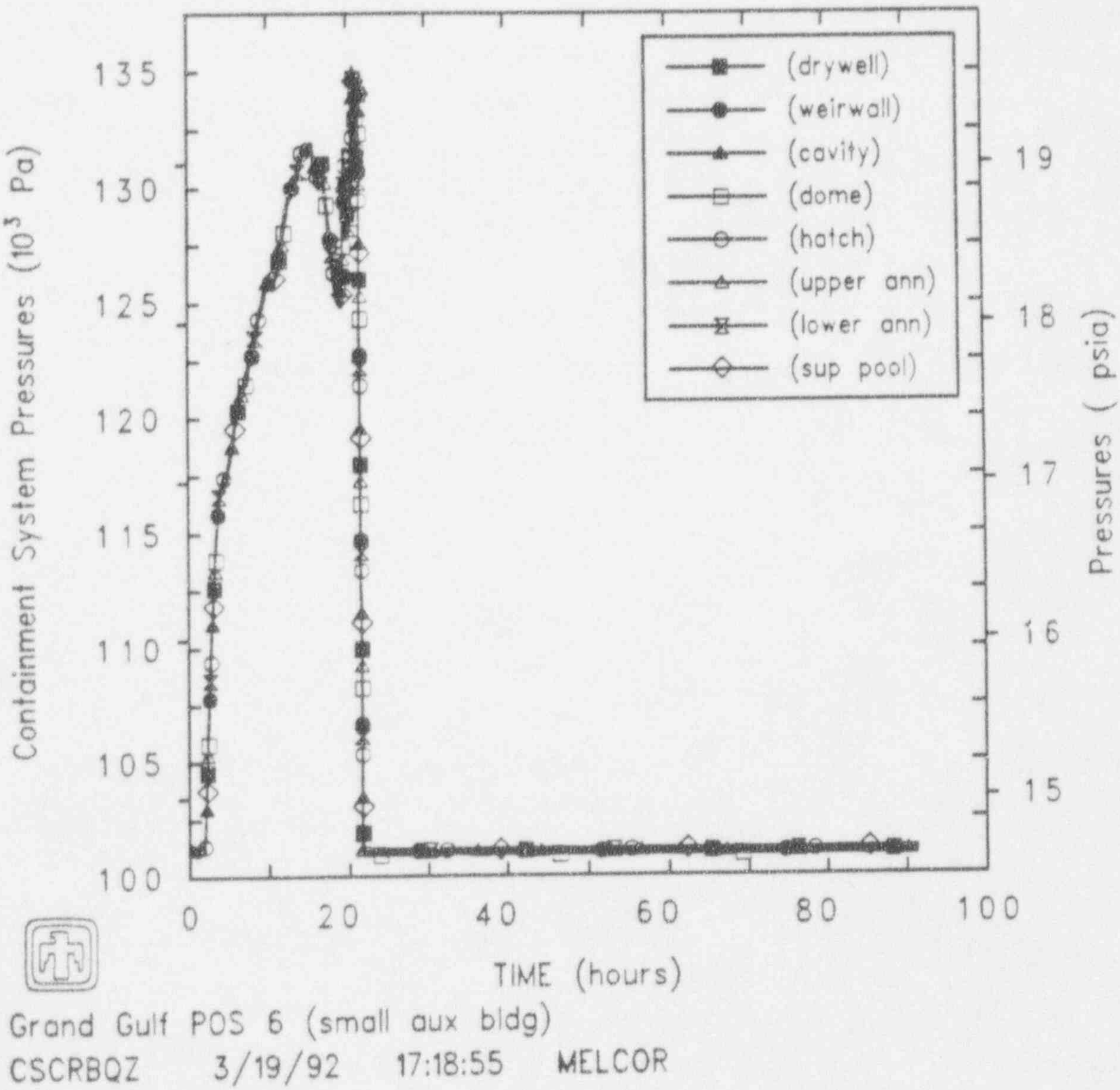


Figure 5.2.3. Containment Pressures for Grand Gulf POS 6 - Reference Calculation

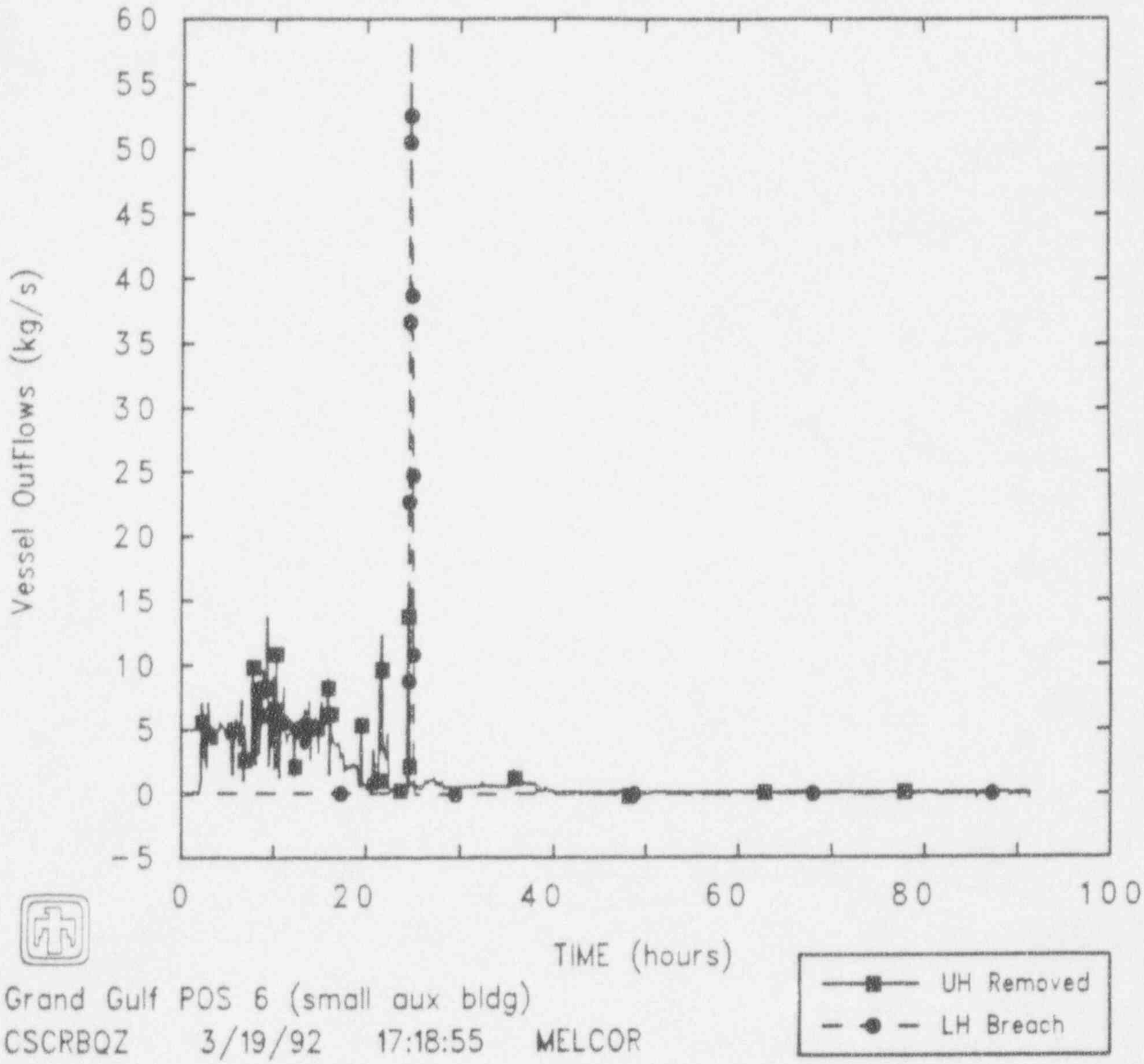


Figure 5.2.4. Reactor Vessel Outflows for Grand Gulf POS 6 - Reference Calculation

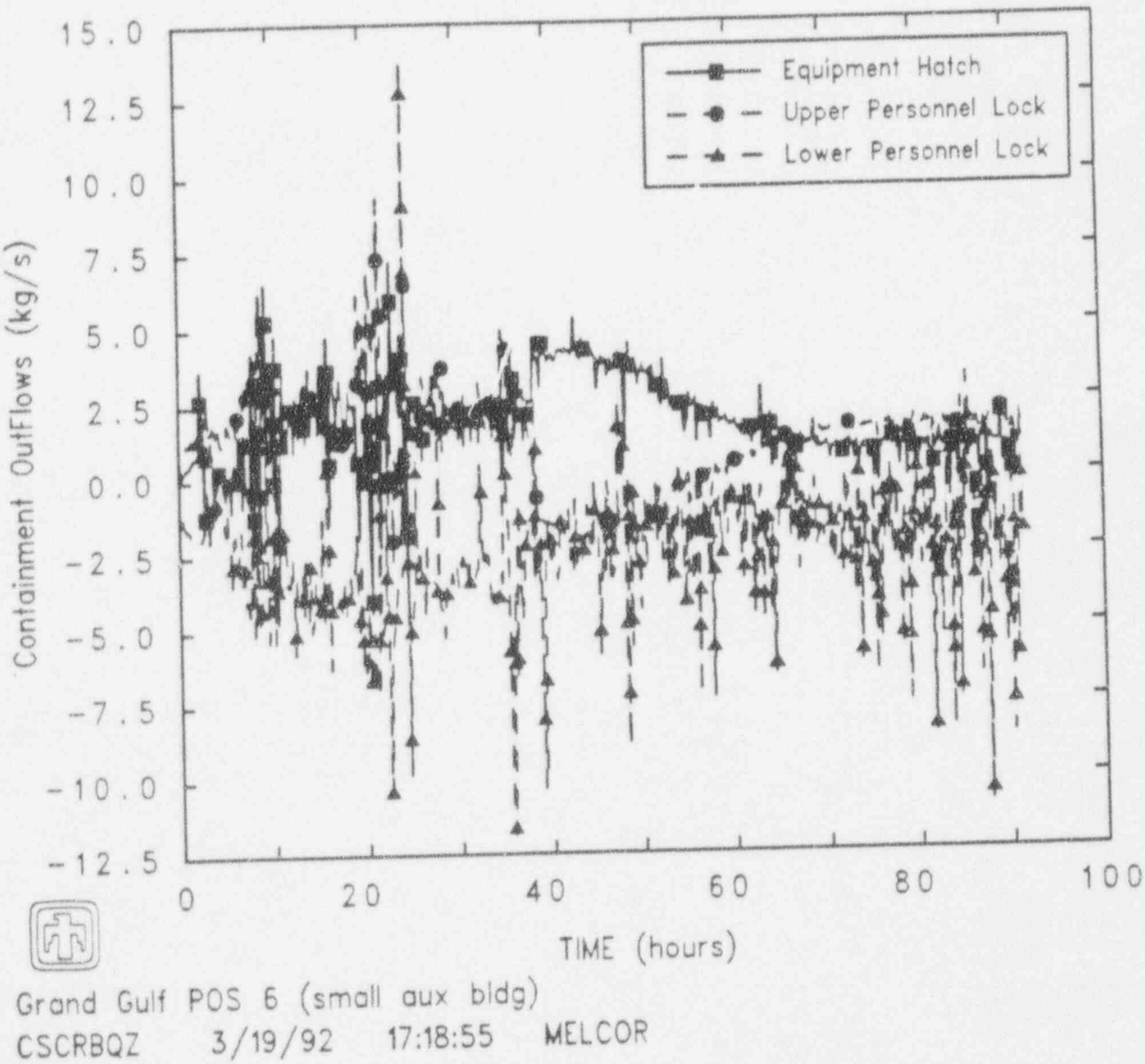


Figure 5.2.5. Containment Outflows for Grand Gulf POS 6 – Reference Calculation

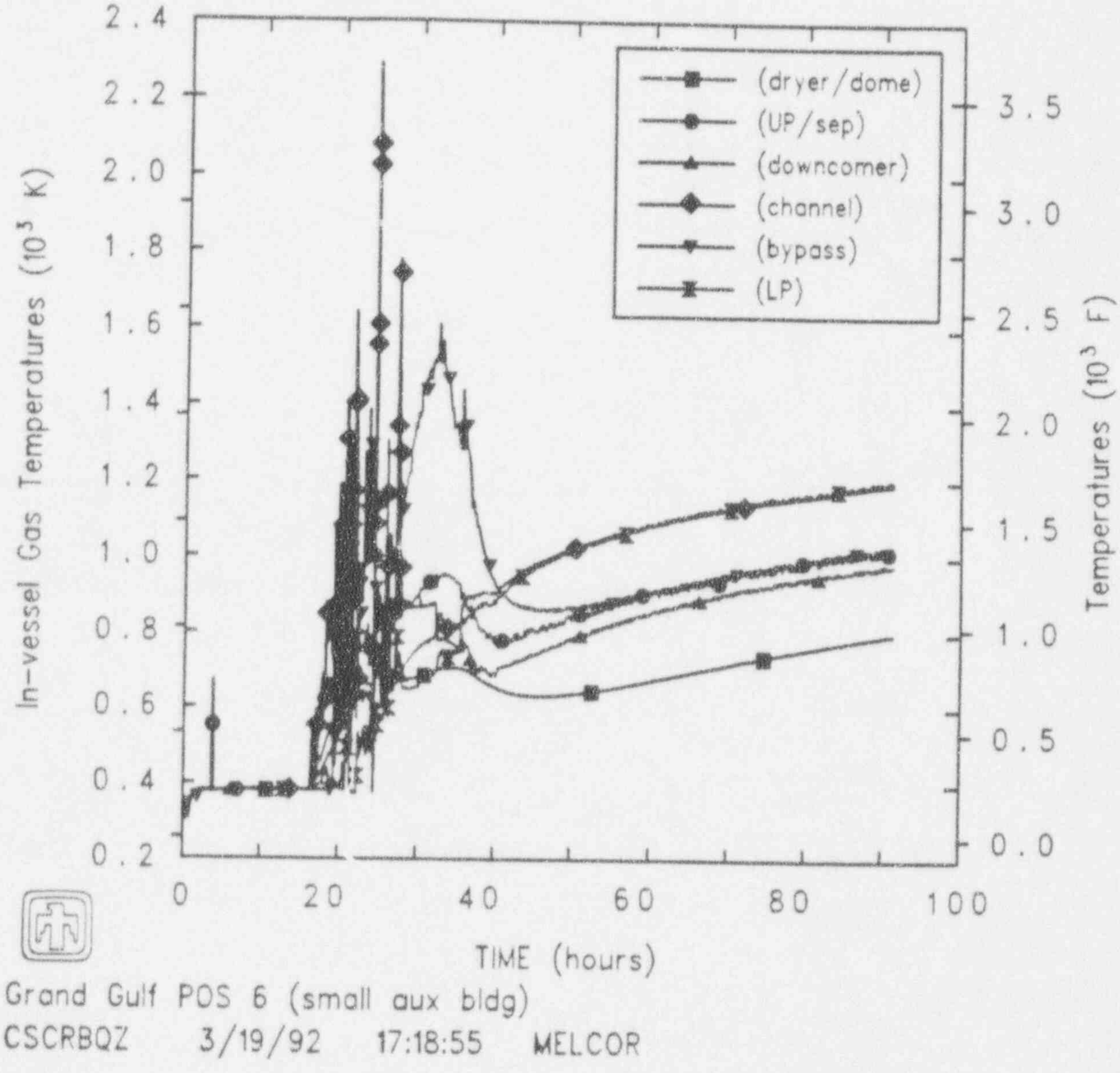


Figure 5.2.6. Reactor Vessel Atmosphere Temperatures for Grand Gulf POS 6 - Reference Calculation

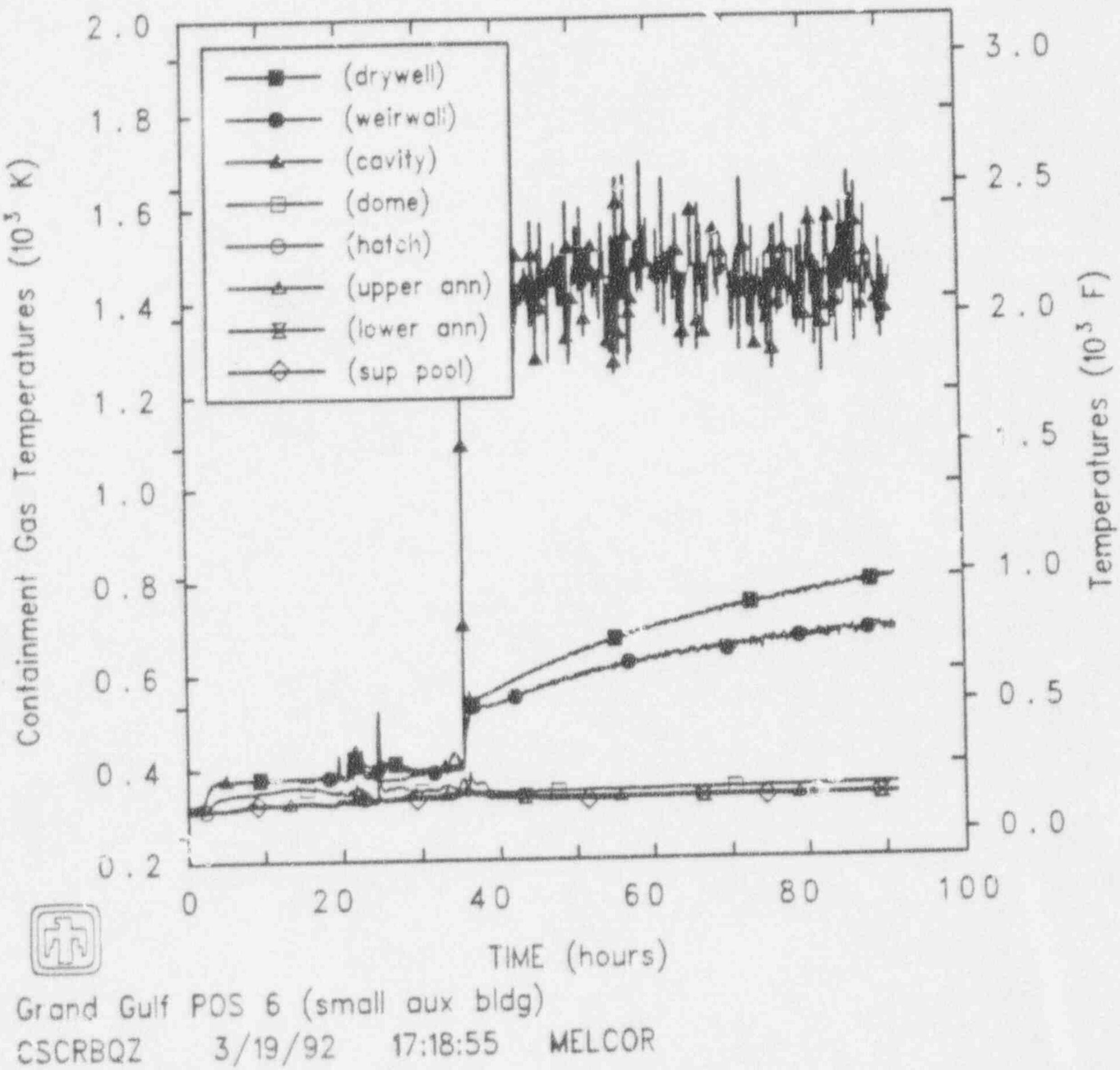


Figure 5.2.7. Containment Atmosphere Temperatures for Grand Gulf POS 6 - Reference Calculation

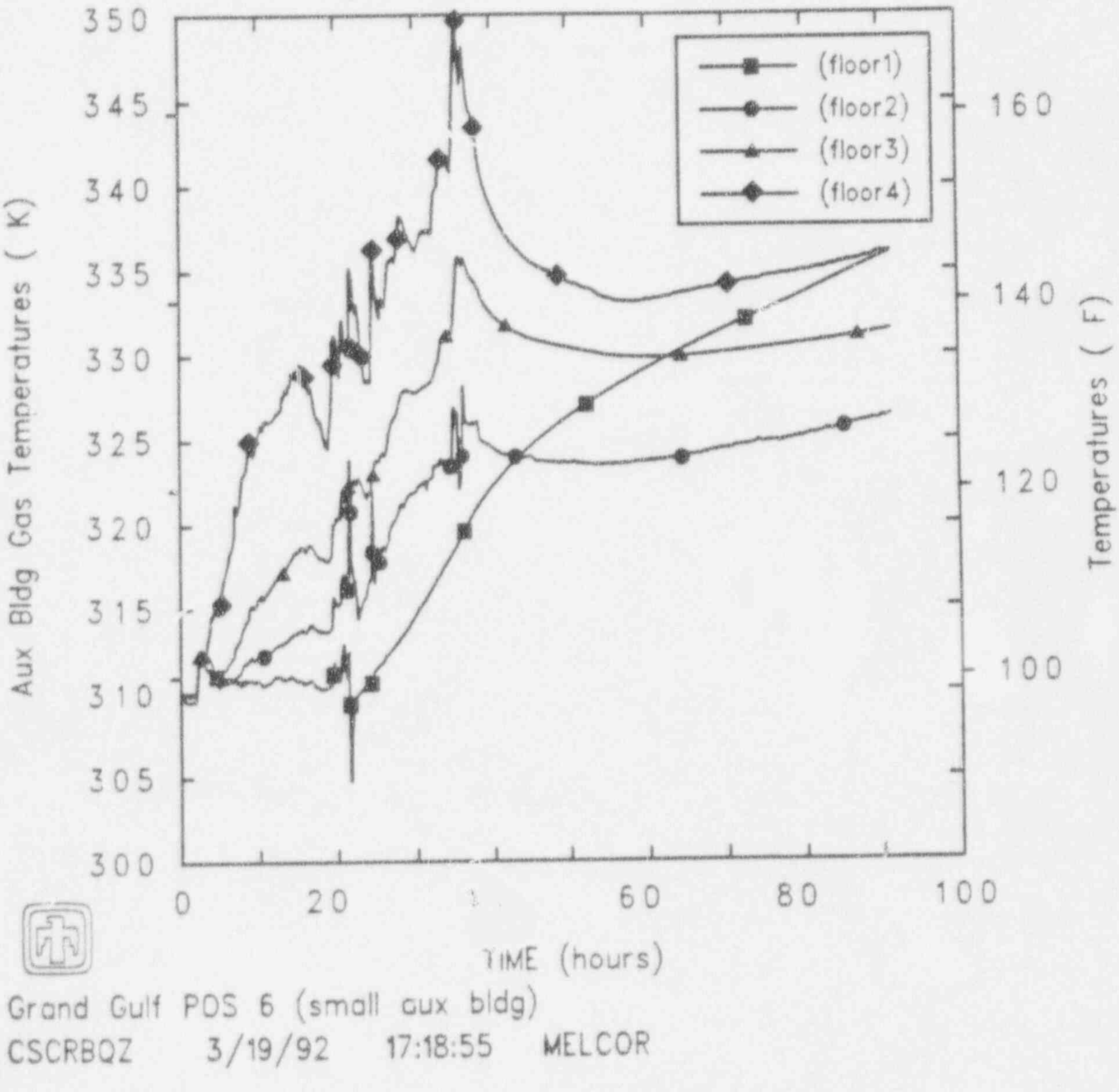


Figure 5.2.8. Auxiliary Building Atmosphere Temperatures for Grand Gulf POS 6 - Reference Calculation

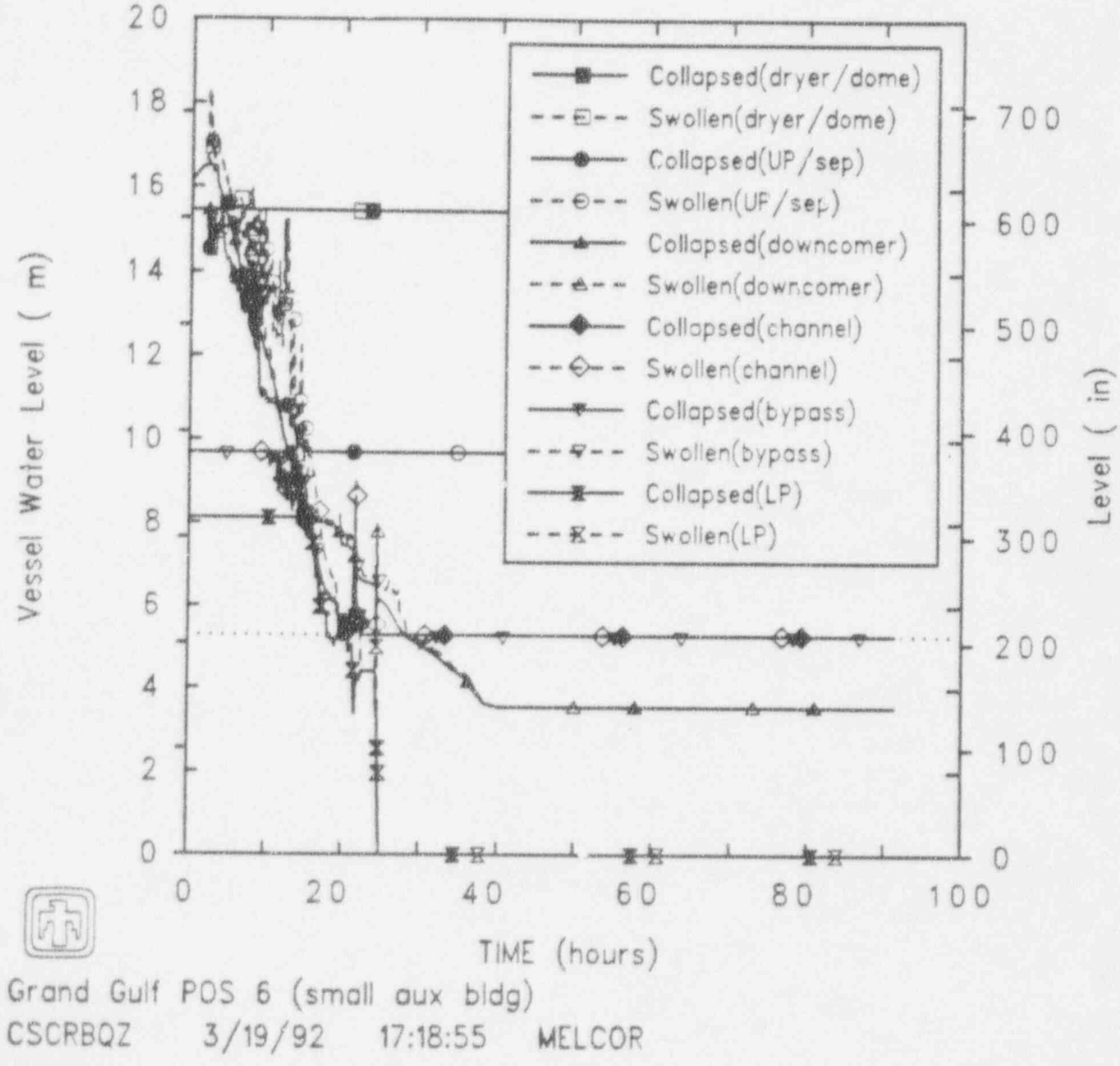


Figure 5.2.9. Reactor Vessel Liquid Levels for Grand Gulf POS 6 - Reference Calculation

in the six core rings are depicted in Figure 5.2.10, as representative of the overall core response. The clad is assumed to rupture at 1173 K, at times ranging from 19 hr to 23 hr in the various core rings, with consequent release of the gap radionuclides and beginning release of radionuclides from the fuel. Substantial clad oxidation occurs, generating hydrogen. The clad melts and relocates at about 2100 K (the zircaloy melt temperature), as does the still-solid fuel, forming debris continually moving downward. (The drop of clad temperatures to zero, as seen in Figure 5.2.10, indicates the disappearance of intact clad from the location being plotted.) The debris can be supported for a short time on the lower core support plate, but the core support plate also fails eventually, and drops the debris into the lower plenum where it attacks and eventually melts through the lower head. The entire process takes just under 12 hr from core uncovery to lower head failure, and about 6 hr from start of clad heatup and oxidation to lower head failure.

The hydrogen generated in the reactor vessel through oxidation of the zircaloy clad and canister, and steel other structure, is shown in Figure 5.2.11; the 1144 kg of hydrogen produced in the vessel corresponds to oxidation of about 20% of the zircaloy and around 5% of the steel. Hydrogen production stops in the vessel as core debris is ejected from the lower plenum in the reactor vessel to the cavity in the inner containment; the core debris in the cavity then continues to generate hydrogen through continued oxidation of zirconium and steel, and through corium-concrete interaction. The amount of hydrogen generated in the cavity by the end of the transient (in this case, by the time the cavity ruptures) is about equal to the amount of hydrogen generated in the vessel earlier in the accident. The hydrogen generated in the cavity in the latter stages of the transient is quite small when compared to the generation rates of other gases, such as CO, CO_2 and water, as illustrated in Figure 5.2.12.

The cavity layer masses and temperatures are given in Figure 5.2.13. Mass first appears in the cavity when the vessel first fails, through a lower head penetration melting, just before 29 hr. Initially, the material consists of mostly heavy oxides and some metals, but changes to a layer of light oxides on top of metals at about 38 hr. The melt temperature drops slightly after this layer inversion, corresponding to the time that elevated cavity atmosphere temperatures in near-equilibrium with the light oxide layer are first seen in Figure 5.2.7.

The MELCOR calculation was stopped when the cavity was ruptured, *i.e.*, when the concrete side and/or bottom walls were completely ablated at at least one point. Figure 5.2.14 shows that, in this calculation, it was the bottom, initially 2 m thick, that was broached first (although when the cavity is predicted to rupture in this calculation there is a minimum side wall thickness of only 0.2 m out of an initial side wall thickness of 1.752 m).

The total radionuclide releases predicted by MELCOR (given in terms of fraction of initial inventory) are presented in Table 5.2.2 at two specific times considered of interest: when a lower head penetration first fails (at about 25 hr in this analysis) and at the end of the calculation (*i.e.*, at 91.4 hr when the cavity is predicted to rupture). At the first time, radionuclides have been released within the reactor vessel as the core degrades; at the latter time, most of the additional release has come from core debris in the cavity

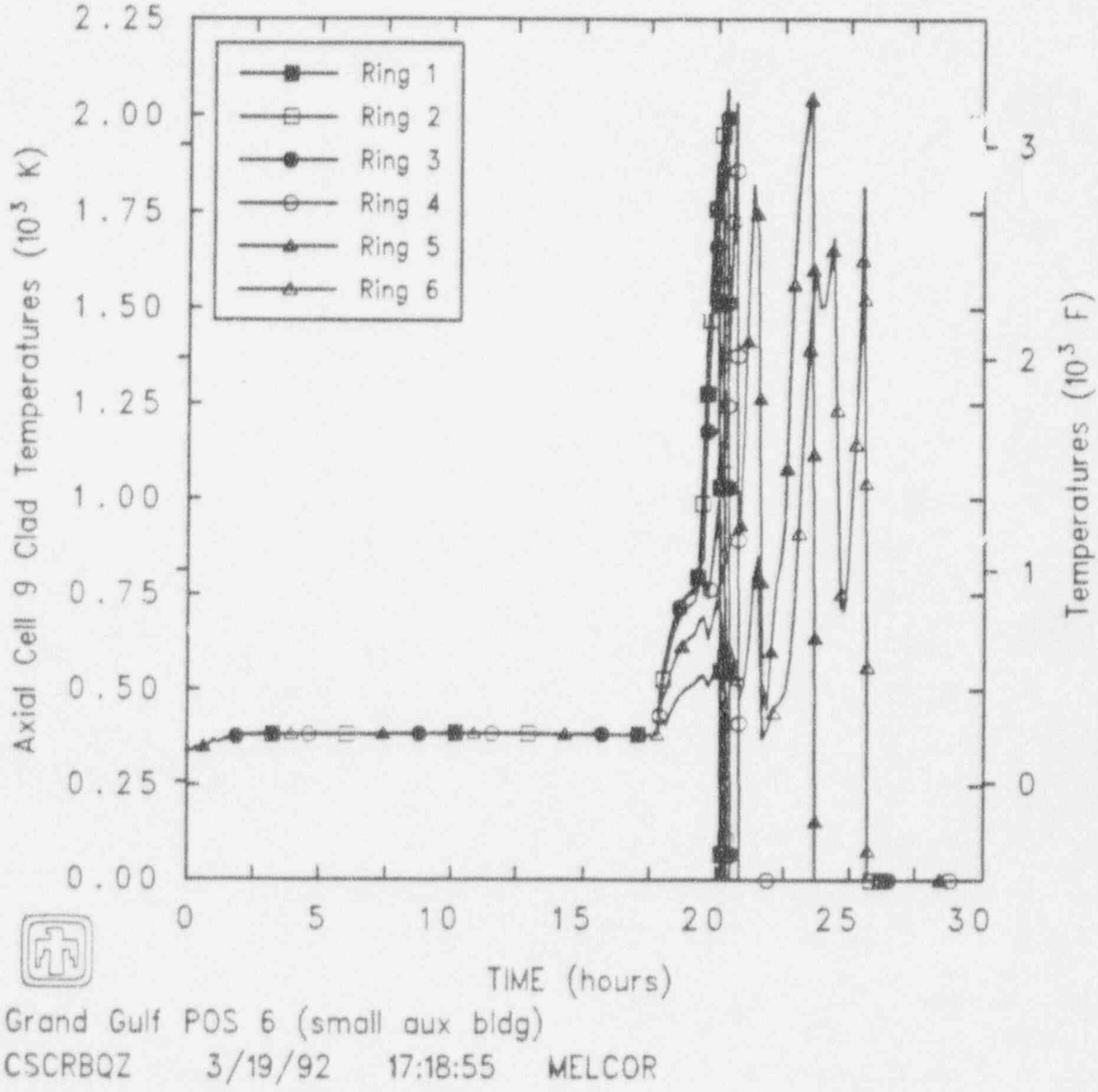


Figure 5.2.10. Level 9 Clad Temperatures for Grand Gulf POS 6 - Reference Calculation

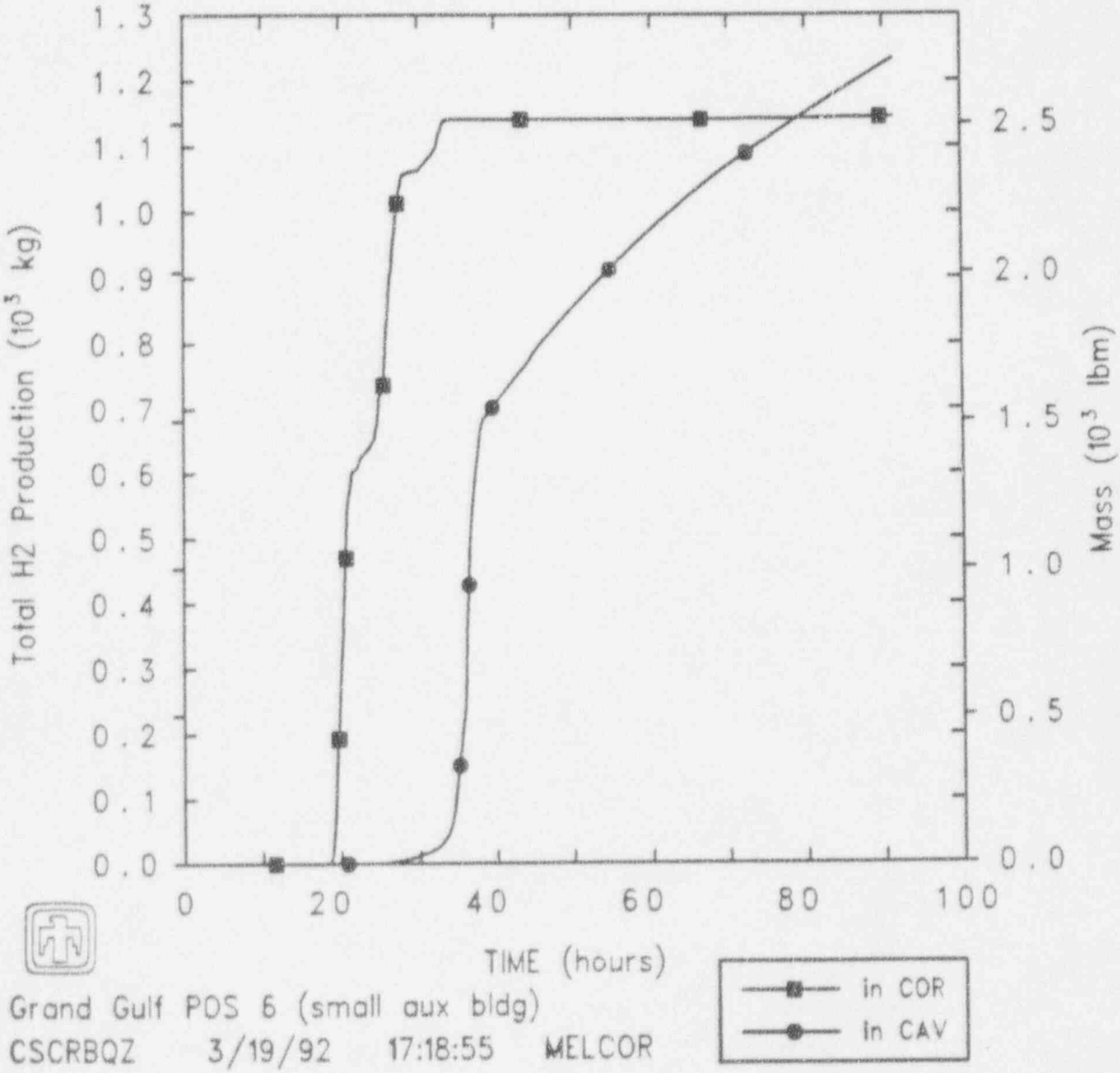


Figure 5.2.11. Total Hydrogen Generation for Grand Gulf POS 6 - Reference Calculation

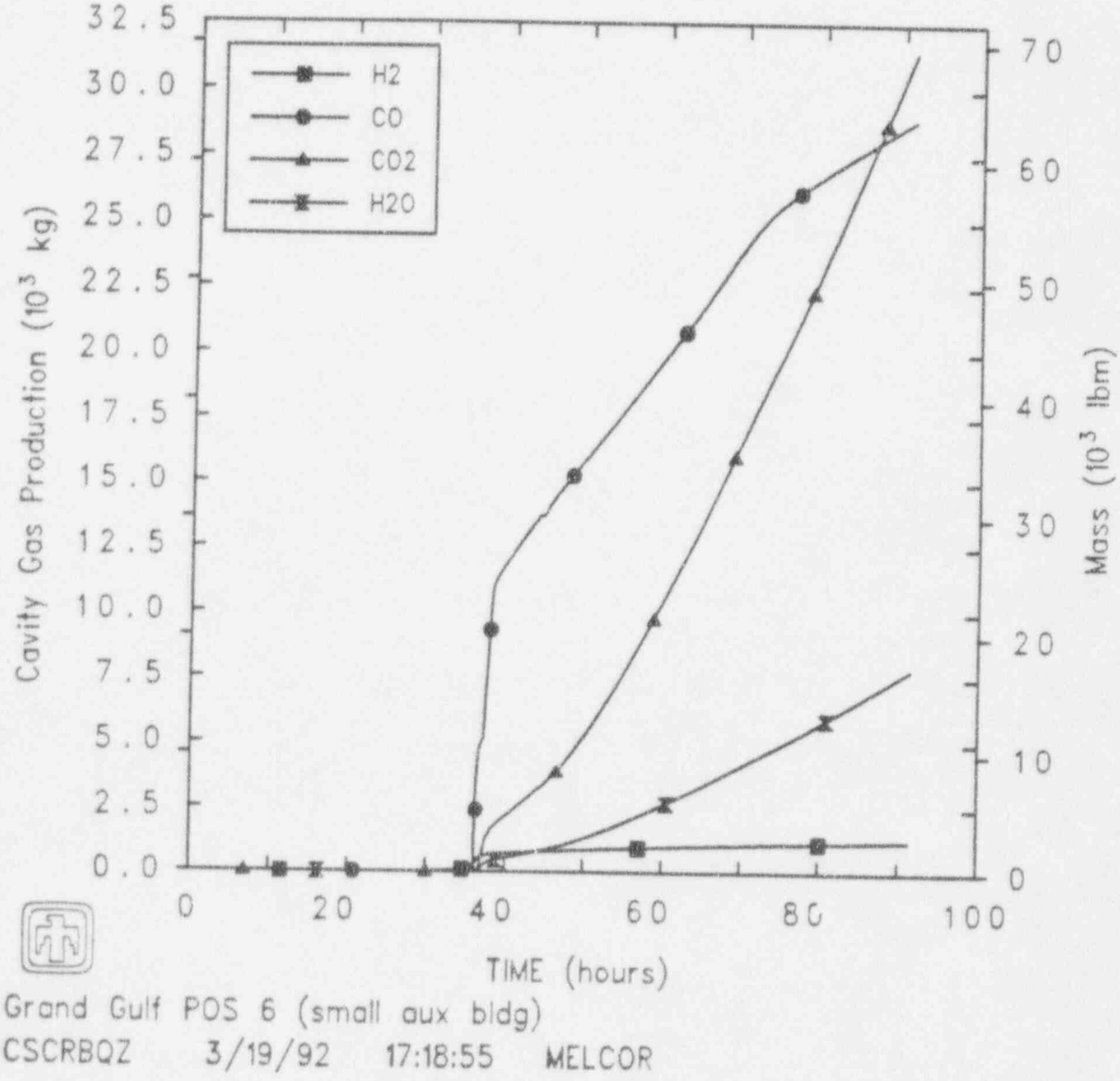
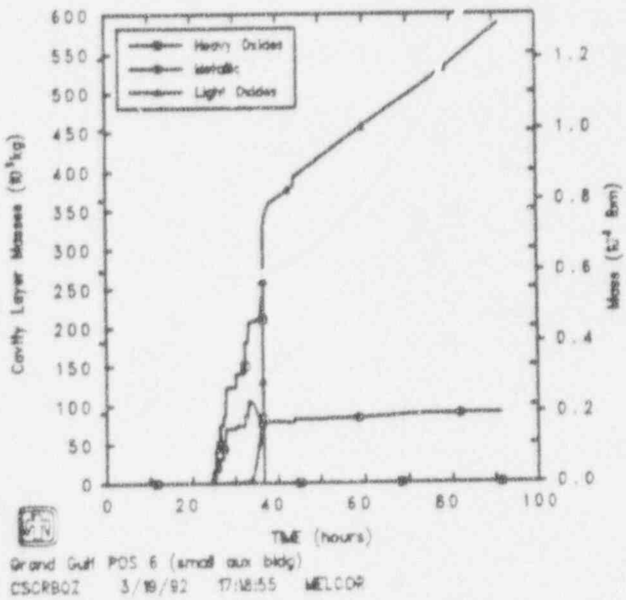


Figure 5.2.12. Cavity Gas Generation for Grand Gulf POS 6 - Reference Calculation



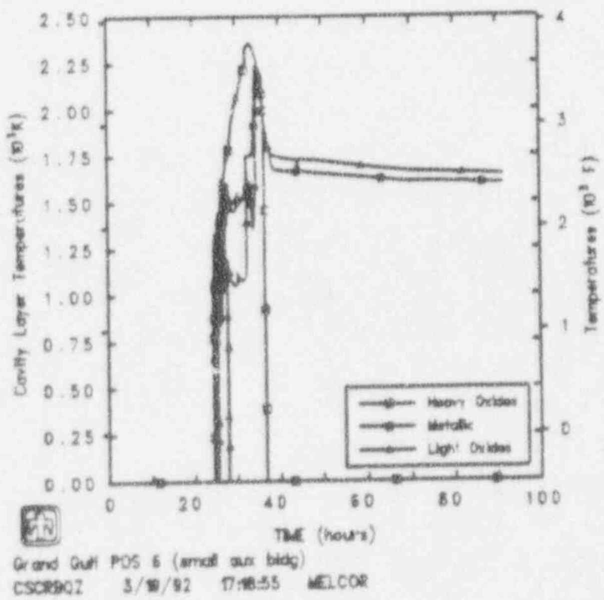


Figure 5.2.13. Cavity Layer Masses (top) and Temperatures (bottom) for Grand Gulf POS 6 – Reference Calculation

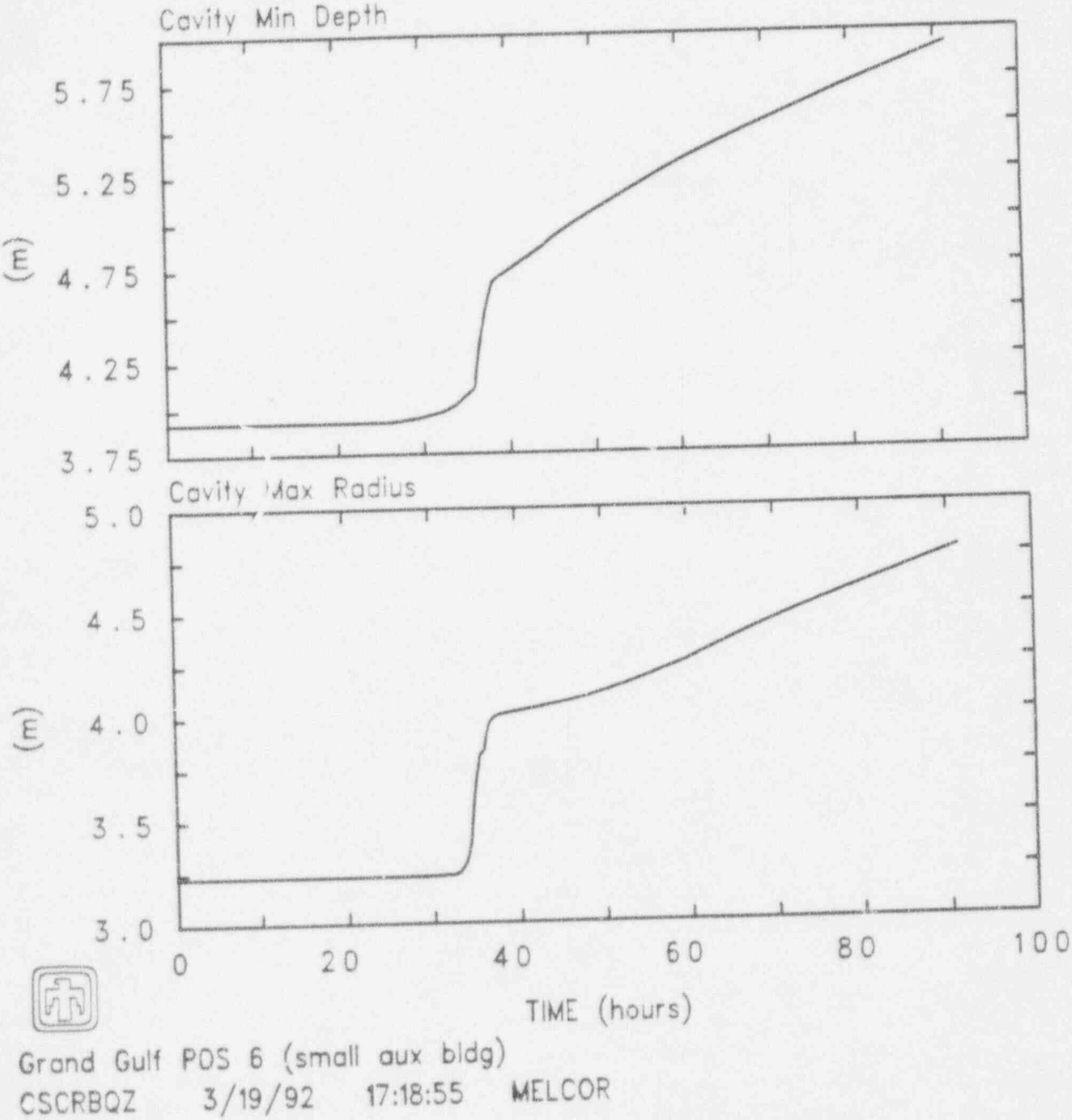


Figure 5.2.14. Cavity Maximum Radius and Minimum Depth for Grand Gulf POS 6 – Reference Calculation

Table 5.2.2. Total Fission Product Radioactive Masses Released from Fuel for Grand Gulf POS 6 – Reference Calculation

	% of Initial Inventory Released		
Class	Before Vessel Failure	Before Cavity Rupture	to Environment
1 (Xe)	76.4	100.0	93.0
2 (Cs)	76.9	100.0	7.35
3 (Ba)	4.22	47.4	0.0615
4 (I)	76.1	93.2	86.8
5 (Te)	62.8	93.0	5.84
6 (Ru)	0.0023	0.0508	0.00056
7 (Mo)	0	0.0128	0.0102
8 (Ce)	0.00076	0.0264	0.00034
9 (La)	0	0.2809	0.00689
10 (U)	0.1145	2.16	0.0233
11 (Cd)	0	0.0396	0.0018
12 (Sn)	13.53	37.7	0.6603

(although some release continues in the vessel until all the core material is ejected to the cavity). Table 5.2.2 also gives the amounts released to the environment by the end of the calculation.

A large percentage of volatile materials (the noble gases, cesium, iodine and tellurium) are released early and in-vessel, and all or almost all of the initial inventories of these classes are released by the end of the transient considered. Class 3 (the alkaline earths, such as Ba or Sr) and Class 12 (the less volatile main group elements like Sn) show significant releases, with almost half the initial inventories released by the end of the transient. The more refractory trivalents (La) and uranium show about a percent release by the time of cavity rupture, while the most refractory classes (Ru, Mo, Ce and Cd) release only 0.01-0.05% of their initial inventories by the end of the calculation.

Most of the release to the environment is in the form of the noble gases and iodine. This is expected because the volatiles (the noble gases, Cs and I) show the most release from fuel and debris, and most of that released inventory is released to the environment for those classes of volatiles which are assumed to be in the form of fission product vapors (the noble gases and I). This result could change if MELCOR considered iodine chemistry in detail.

The only other significant releases to the environment are for Cs and Te, in percentage terms, and for U, in absolute mass terms. Most of the classes either exhibit little release from fuel and/or debris, or substantial retention in the reactor vessel, containment and auxiliary building.

The releases in Table 5.2.2 gives a view at two distinct, different times in the transient. Additional information can be obtained by considering the time-dependent releases, in both the vessel and in the cavity, and also by considering the distribution of the radionuclides released.

Figure 5.2.15 presents release and distribution histories for Class 1 (Xe), with both the amounts released and the amounts in any given location at a particular time normalized by the initial mass of the class. (The results for Class 4, iodine and the other halogens, are virtually identical.) As was evident from the values given in Table 5.2.2, most of the noble gas inventory is released early in the in-vessel phase (>90%) with the remainder all released with in the cavity. Because it is in the form of a fission product vapor, it is quickly transported through the primary system and containment, to the auxiliary building and out to the environment. By the end of the transient considered, over 90% of the initial inventory of noble gases and 85% of the initial inventory of halogens have been released to the environment.

The release and subsequent distribution histories of the alkali metals (Class 2, characterized by Cs) and the chalcogens (Class 5, represented by Te) are similar to each other, with the results for cesium given in Figure 5.2.16. As with the noble gases and iodine, most of the initial inventories (>90% for Cs and >85% for Te) are released while still in the primary system, and almost all the remaining inventory is released in containment. Very little of the released inventory (less than 10%) finds its way to the environment, and these materials appear to settle into a stable distribution pattern with little transport after about 40 hr; the abrupt shift from the drywell to the sumps at around 35 hr appears to be due to the abrupt rise in drywell temperature (Figure 5.2.7). There is no one predominant location for these classes, with about 30-35% retained in the auxiliary building, and less than 20% each in the primary system, drywell, outer containment and sump pools.

Another set of similar release and distribution behavior is found in the platinoids (Class 6) and the tetravalents (Class 8): the results for both these classes closely resemble the behavior predicted for uranium (Class 10), shown in Figure 5.2.17, even though their release fractions are much lower. Most of the release occurs in-vessel, at the high temperatures characteristics of the degraded core, with little or no release predicted at the slightly lower temperatures predicted in the debris bed in the cavity. Of the material released, about 35% remains in the reactor vessel, with another 20-25% in the sump pools and 15-20% found in the auxiliary building. As with the Cs and Te classes, very little of the released inventory (around 1%) finds its way to the environment (or to the drywell or outer containment, either), and these materials also appear to settle into a stable distribution pattern with little transport after about 40 hr; the abrupt shift from the drywell and outer containment to the sumps at about 35 hr appears to be due to the abrupt rise in drywell temperature (Figure 5.2.7).

The remaining classes (such as Ba and Sn) do not appear to fall into such convenient groupings in terms of their release/distribution behavior. The results for the alkaline earths class (Class 3, characterized by barium) are illustrated in Figure 5.2.18. This is the only class showing about equal amounts released in the vessel and in the cavity.

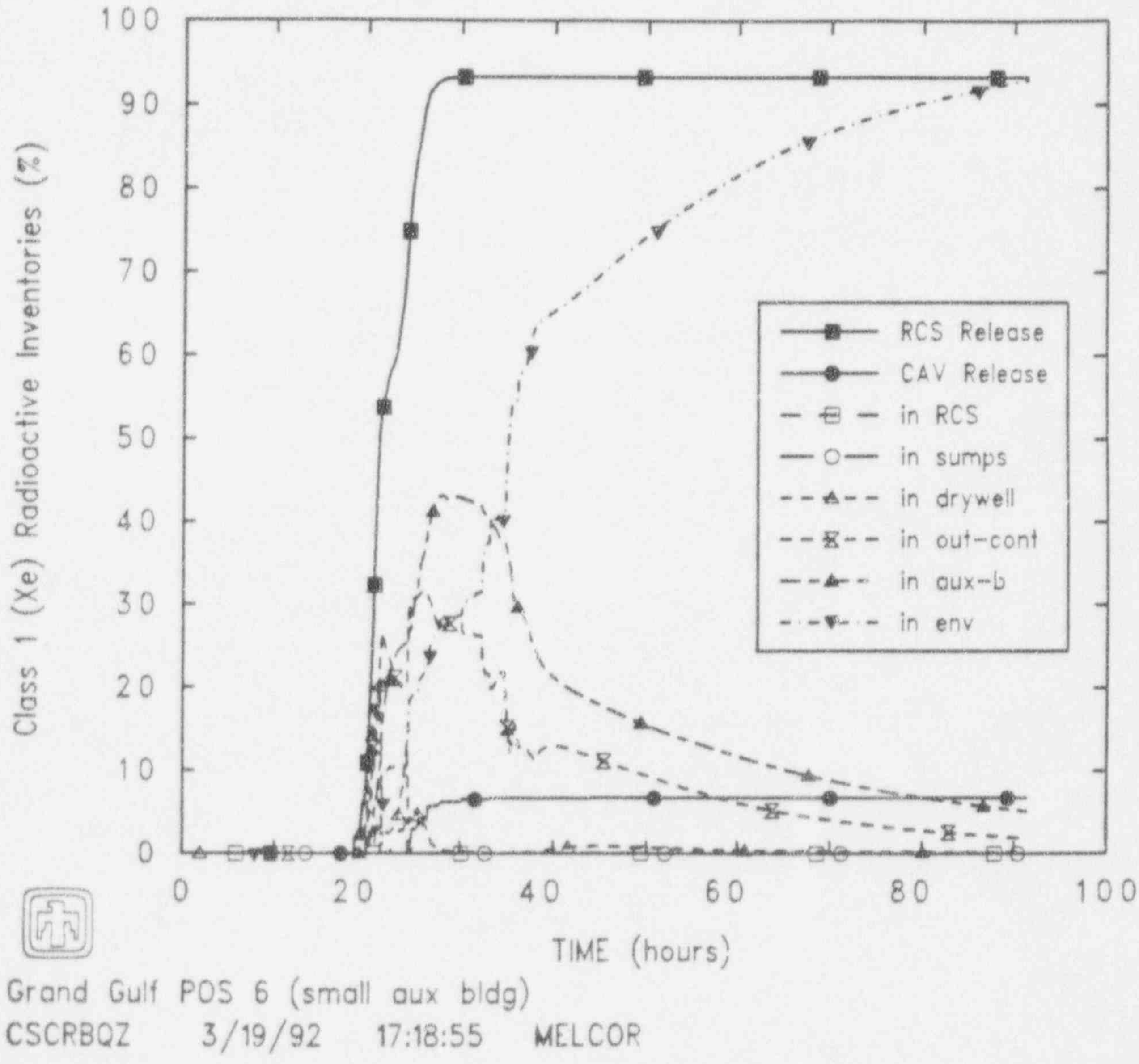


Figure 5.2.15. Class 1 Radioactive Mass Radionuclide Distribution for Grand Gulf POS 6 - Reference Calculation

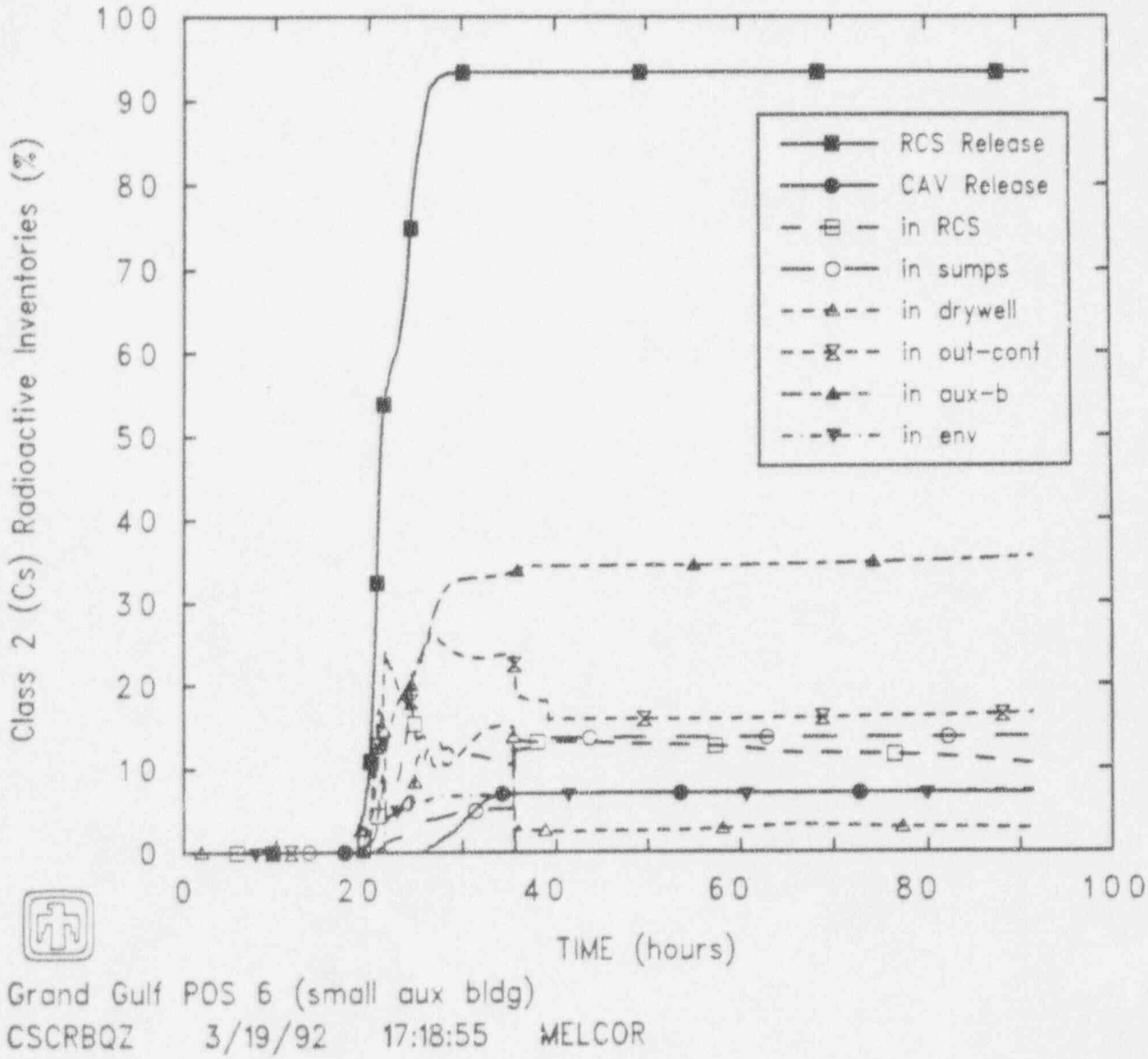


Figure 5.2.16. Class 2 Radioactive Mass Radionuclide Distribution for Grand Gulf POS 6 - Reference Calculation

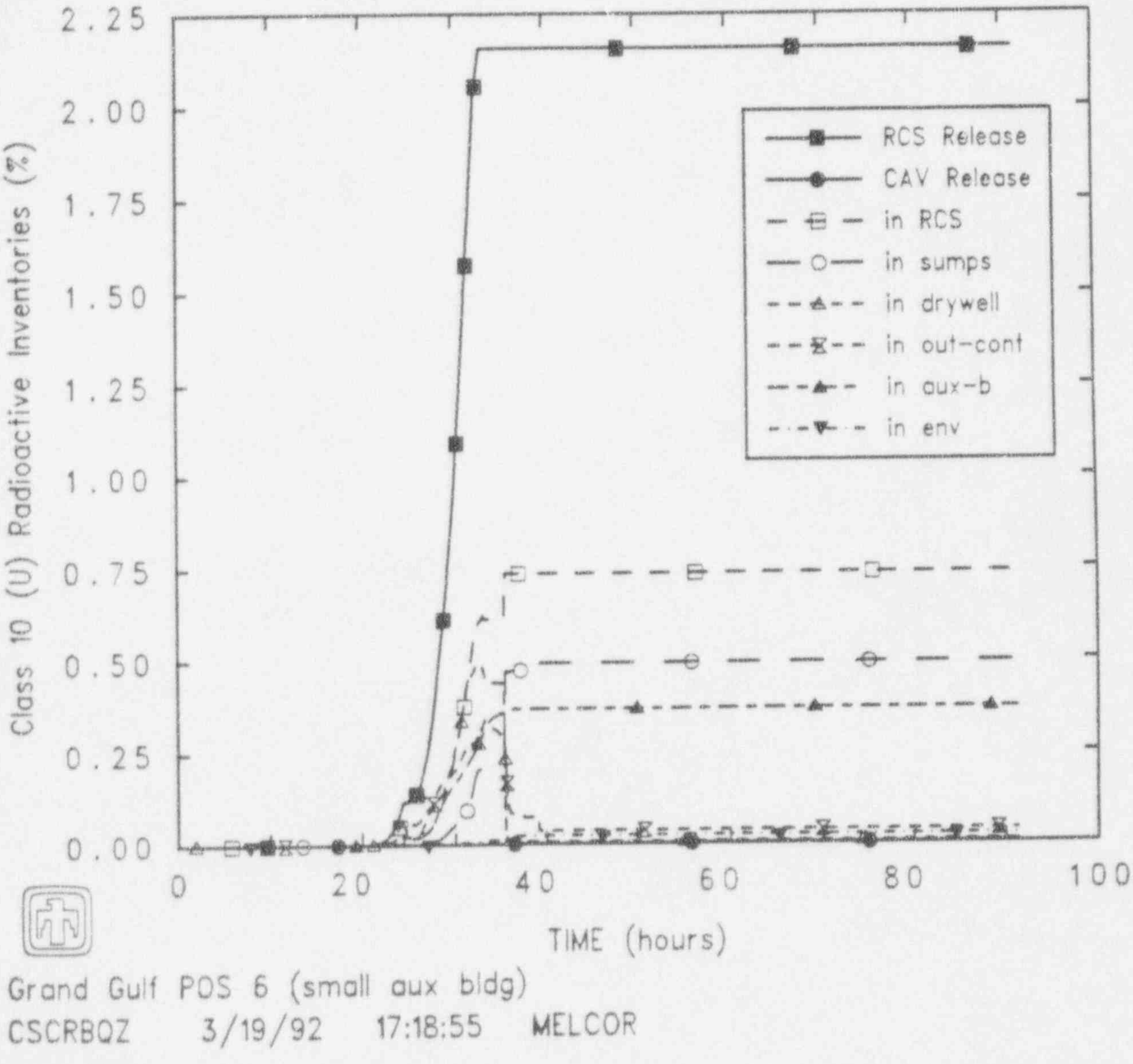


Figure 5.2.17. Class 10 Radioactive Mass Radionuclide Distribution for Grand Gulf POS 6 - Reference Calculation

About 25% of the mass released accumulates in the sump pools, where it is relatively immobilized; a second and third quarter of the mass released either remains in the reactor vessel or settles in the auxiliary building. Of the final 25%, most is in the drywell and outer containment, and little (around 2%) is released to the environment.

Figure 5.2.19 gives the corresponding results for Class 12 (Sn). Most of the release occurs at the higher in-vessel temperatures, with very little release at the lower, cavity-debris temperatures. Of the material released, the distribution somewhat resembles that just described for the barium class. The largest fraction (25-30%) is retained in the auxiliary building, with another 20-25% each in the vessel and sump pools. Of the remaining mass released, most is in the outer containment, and little is found in the drywell or released to the environment.

The behavior calculated for the early transition elements such as Mo (Class 7, in Figure 5.2.20), the trivalents represented by La (Class 9, in Figure 5.2.21), and the more volatile main group elements such as Cd (Class 11, in Figure 5.2.22), all share the common trait that the release occurs in the cavity after debris ejection; no release is seen in the primary system. As with all the classes discussed so far, the distribution of the trivalents does not change much after about 40 hr, *i.e.*, after the abrupt rise in drywell temperature (Figure 5.2.7). The distribution of the Class 7 radionuclides also stops changing but later in time, after about 50 hr, because there is still some release of this class occurring between 40 and 50 hr. In contrast, Class 11 shows continued release at a nontrivial, nearly linear rate for the remainder of the transient after the initial step release at ~35 hr. All three of these class have the largest fractions of their released inventories in the sump pool, auxiliary building, drywell and primary system, with little appearing in the environment (although the amount of Cd in the environment is still increasing at the end of the transient).

5.3 Plant Configuration Studies

The calculations done for POS 6 included variations on the plant configuration, as summarized in Section 5.1. The results of these sensitivity studies are described in this section. These analyses evaluated the effect of including the auxiliary building in the calculations, with various free volumes and deposition surface areas assumed to represent doors being open or closed. The effects of the containment personnel locks being open or closed were investigated also, as was the impact of the drywell head being open or closed. The influence of the time between scram and accident initiation was considered, as well as the effect of hydrogen igniters being active.

5.3.1 Auxiliary Building

As discussed in Section 3, a model for the auxiliary building (shown in Figure 3.4), was developed specifically for these analyses, primarily from the limited information in the FSAR [57]. Because of the uncertainties in the descriptions of the auxiliary building

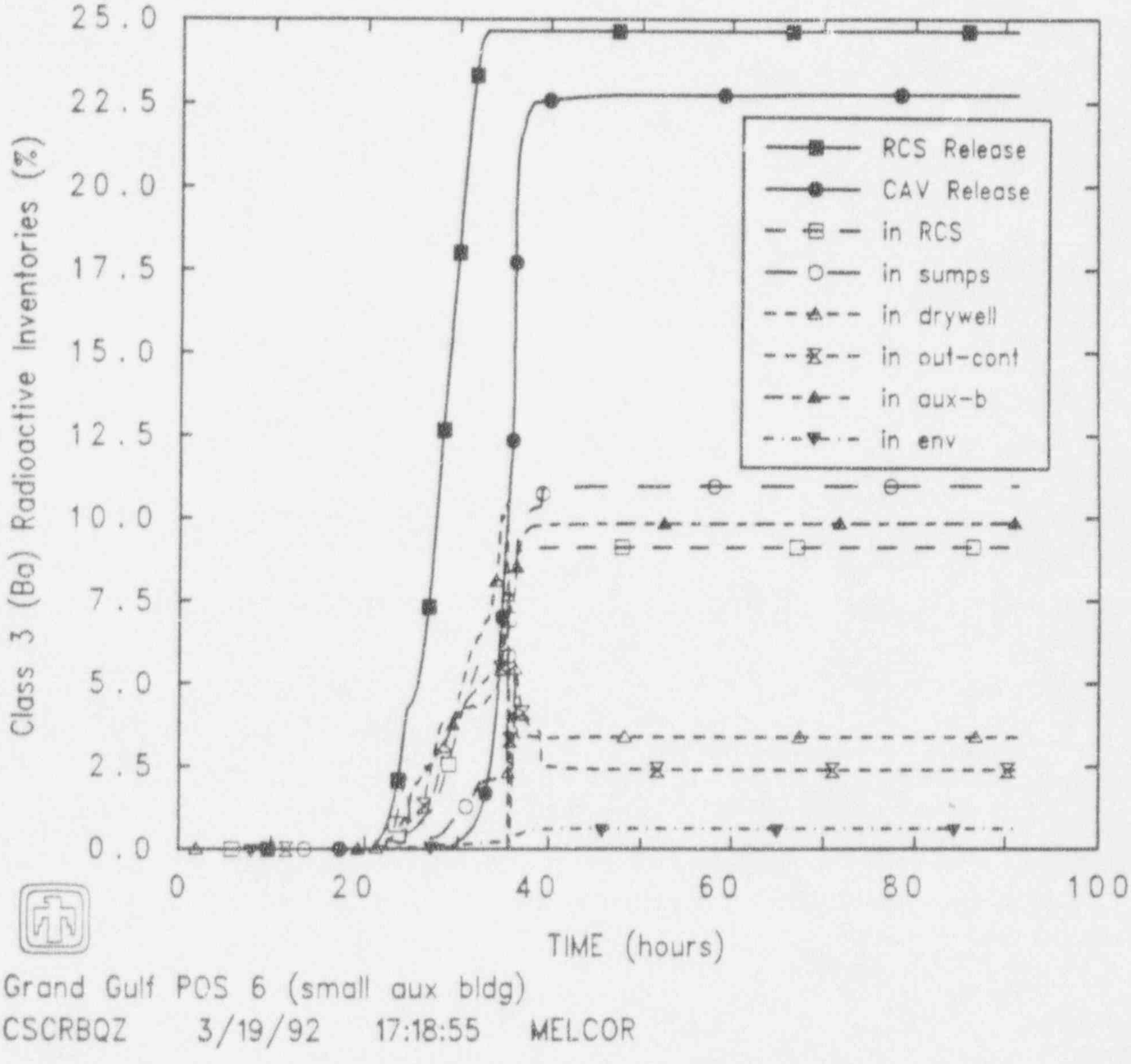


Figure 5.2.18. Class 3 Radioactive Mass Radionuclide Distribution for Grand Gulf POS 6 - Reference Calculation

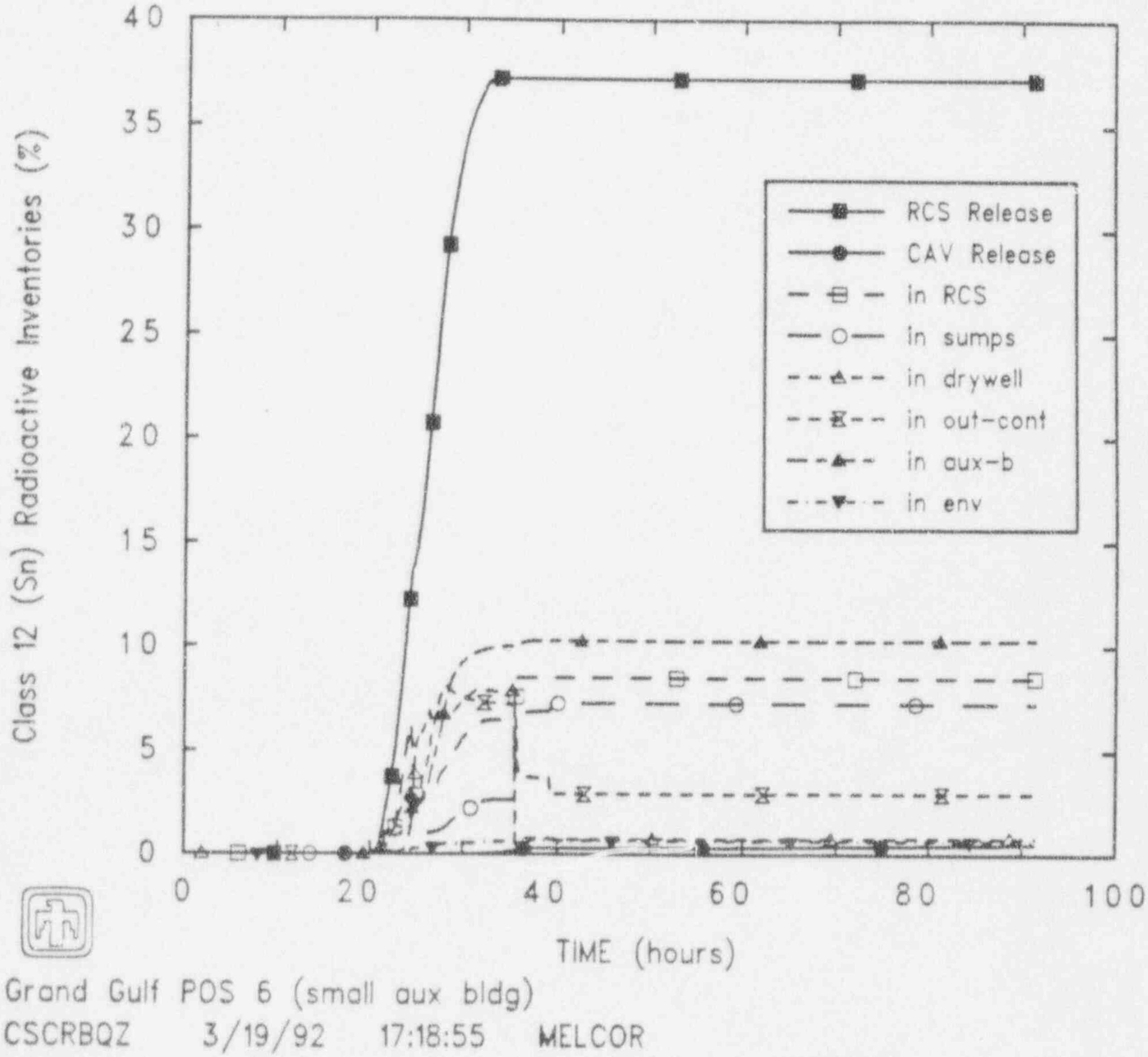


Figure 5.2.19. Class 12 Radioactive Mass Radionuclide Distribution for Grand Gulf POS 6 - Reference Calculation

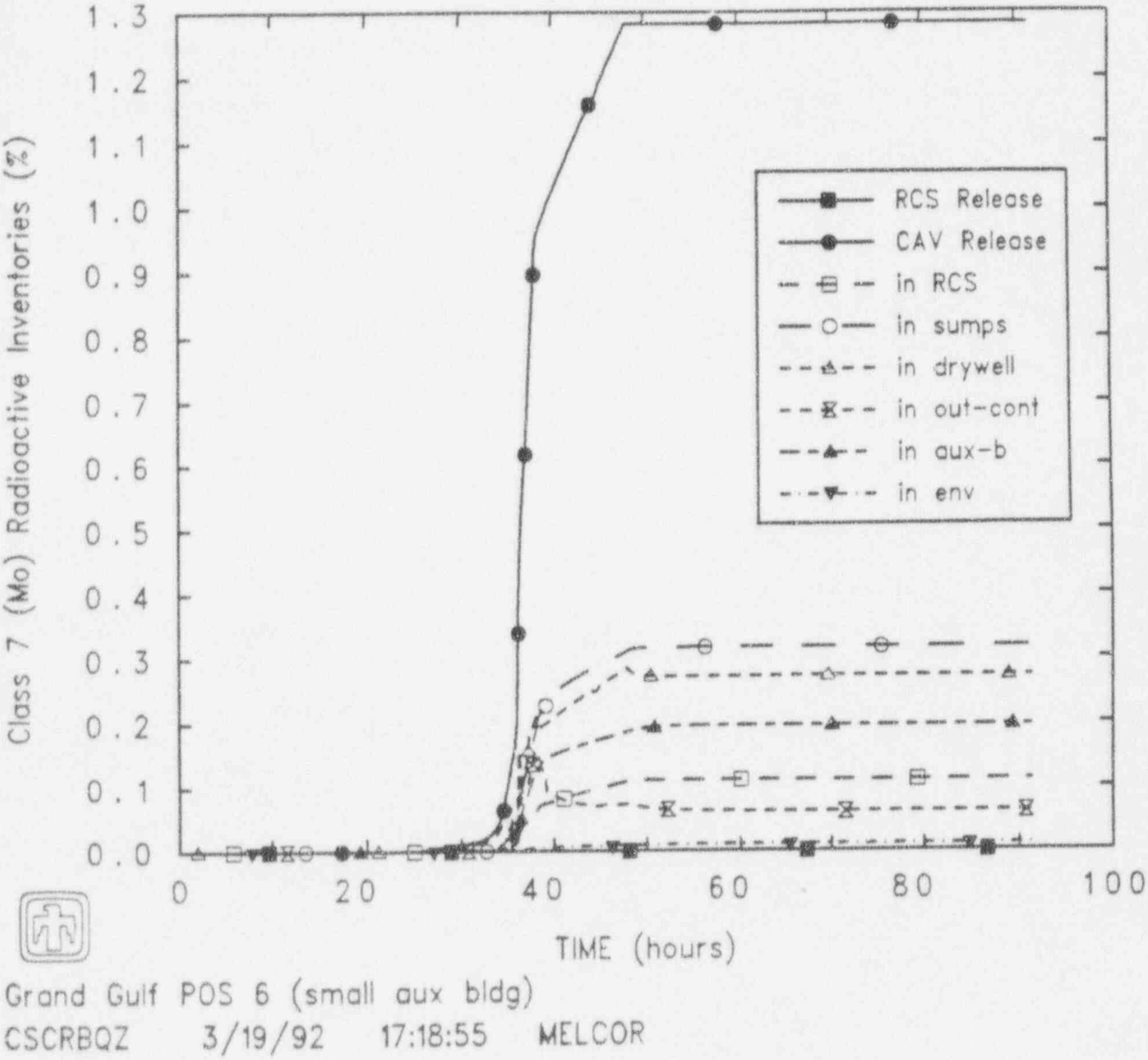


Figure 5.2.20. Class 7 Radioactive Mass Radionuclide Distribution for Grand Gulf POS 6 - Reference Calculation

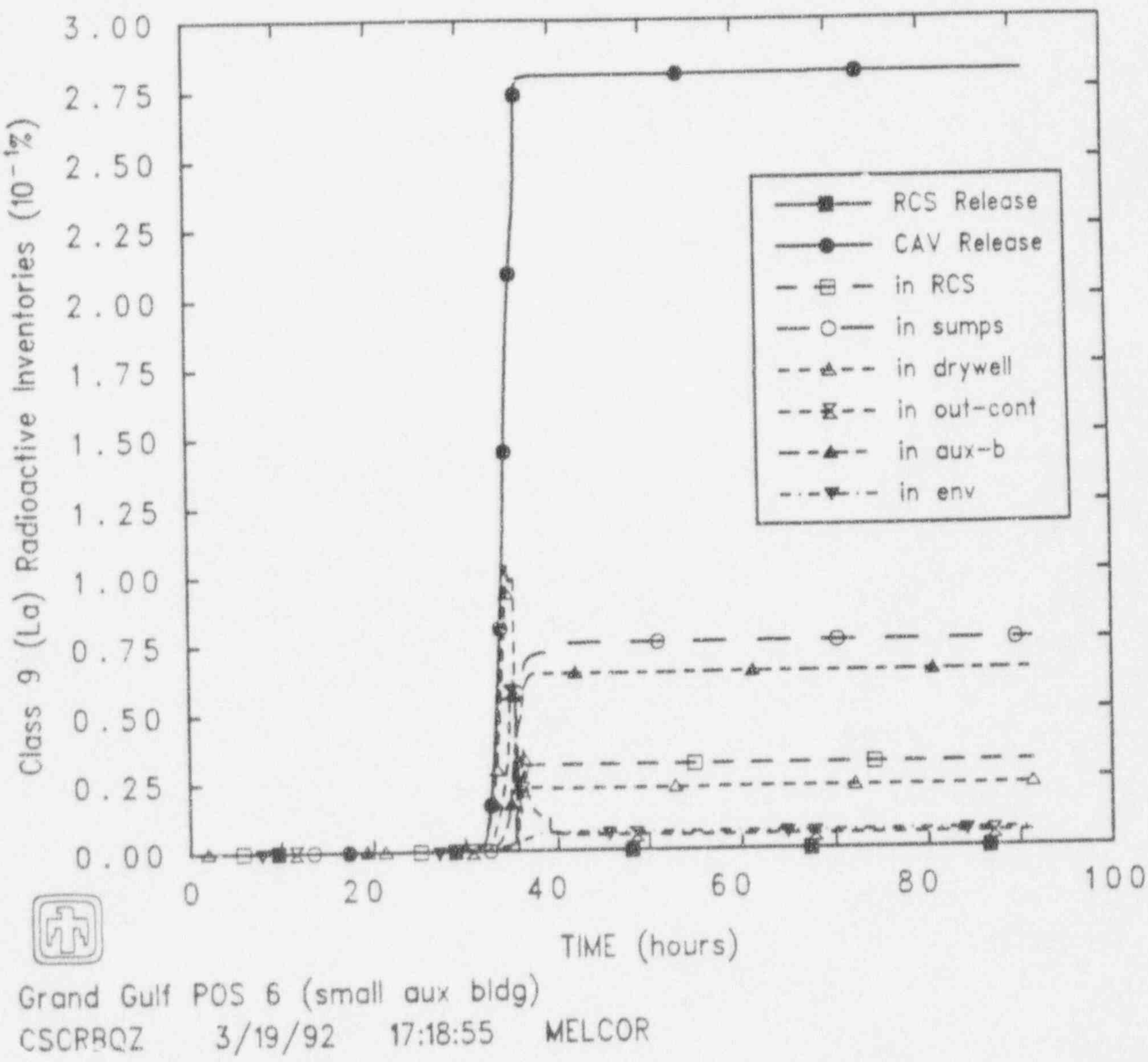


Figure 5.2.21. Class 9 Radioactive Mass Radionuclide Distribution for Grand Gulf POS 6 - Reference Calculation

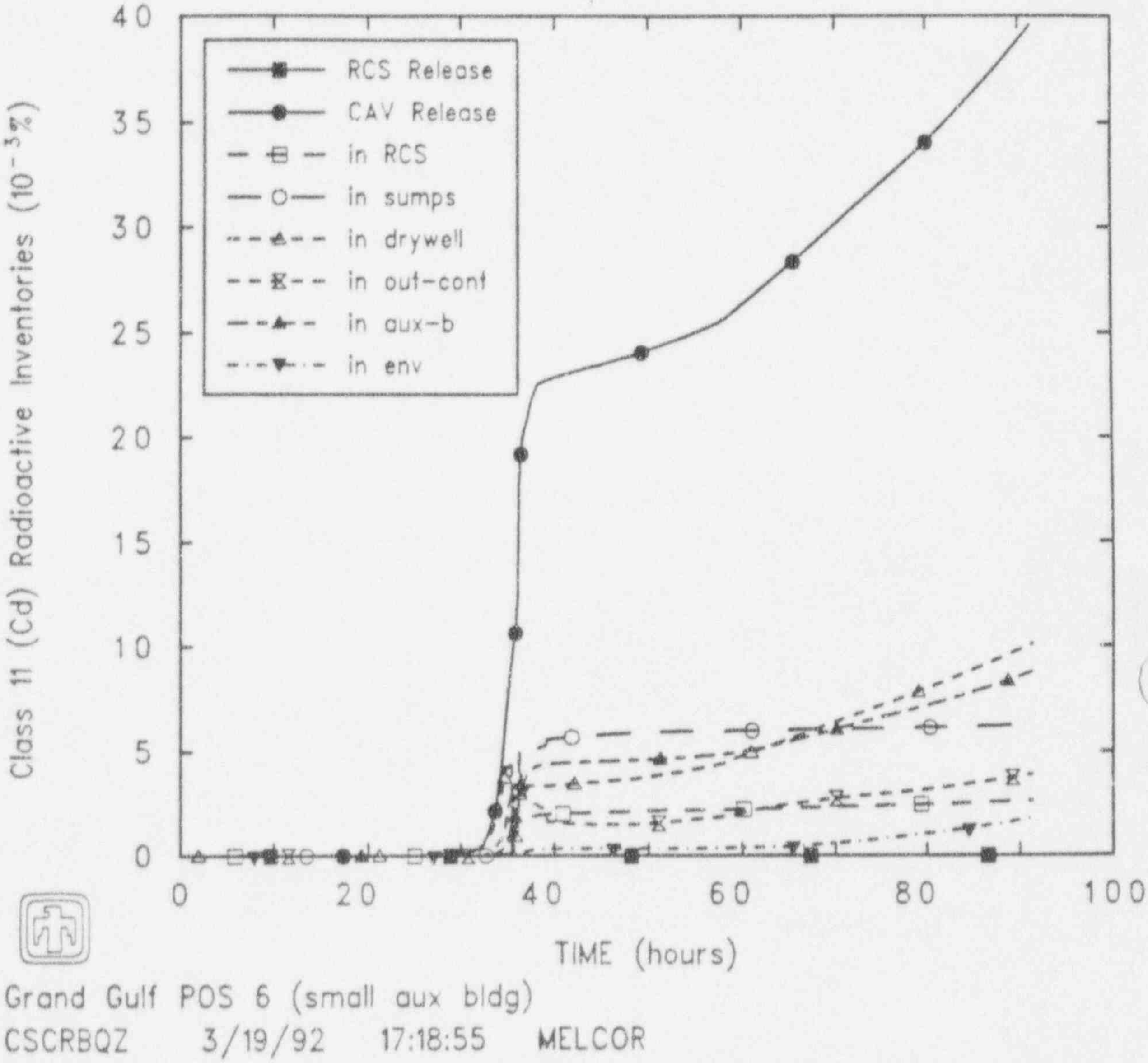


Figure 5.2.22. Class 11 Radioactive Mass Radionuclide Distribution for Grand Gulf POS 6 - Reference Calculation

geometry, especially the flow paths, two variations of the auxiliary building model were developed.

In both, the auxiliary building model consisted of the same number of control volumes, flow paths and heat structures, but the volumes and surface areas were changed: the opened (or "big") auxiliary building model represented open interior doors, resulting in larger open volumes and heat structure surface areas for flow-through and potential retention and/or deposition of aerosols before the stairwell door to the environment is blown open; the closed (or "small") auxiliary building model represented closed interior doors while the stairwell door to the environment is blown open. Both auxiliary building models assumed failure on a 5 psi overpressure.

The reference calculation with results described in detail in Section 5.2 used the closed auxiliary building model. To evaluate the impact of the uncertainties in the description of the auxiliary building geometry, calculations were done with the open auxiliary building model as well as with no auxiliary building model (*i.e.*, the containment open directly to the environment).

Table 5.3.1.1 compares the timings of various key events predicted in the calculations with no, opened and closed auxiliary buildings modelled. The start of core uncovery varies by at most about 22 min, while the first gap release varies by at most about 25 min, in the calculations with either auxiliary building *vs* no auxiliary building; the timing difference for these early events is much smaller (1-5 min) for the calculations with the two different auxiliary building models.

The timing differences shown in Table 5.3.1.1 grow larger at later times, with the first lower head penetration failure occurring 4 hr later in the open auxiliary building analysis (compared to less than 1 hr difference in lower head failure time in the other two calculations); however, this 4 hr difference is to some extent a numerical effect. It was necessary in this particular calculation (as in a few others) to back up and reduce the user-specified maximum time step in order to continue through and past numerical difficulties in modelling the core degradation process in order to be able to complete the analysis. That time-step reduction affected the results calculated to some degree, in addition to any effects of the different auxiliary building model (as shown in Section 5.4.2, presenting the results of a calculation in which a time-step cut during the core degradation process was the only change made, to determine how big an effect reducing the time step would have on the results).

The early-time differences found in timing of core uncovery and gap release are due to differences in the pressure response of the primary system and containment in the calculations using different auxiliary building models. Figure 5.3.1.1 presents the lower plenum pressures from these three calculations, as representative of the primary system response. With an auxiliary building modelled, the primary pressures slowly equilibrate to the (rising) containment pressure as water inventory is boiled away, until the auxiliary building fails, after which time the pressures drop rapidly to atmospheric; with no auxiliary building and the containment open directly to the environment, the primary pressures equilibrate directly to atmospheric pressure instead of to rising containment pressures, resulting in lower reactor vessel pressures during the first 20 to 30 hr.

Table 5.3.1.1. Key Event Times for Grand Gulf POS 6 - Auxiliary Building Model Sensitivity Study

Event	No Aux Bldg	Open Aux Bldg	Closed Aux Bldg
Level below TAF	12.68 hr	12.95 hr	13.04 hr
Clad failure/Gap release			
(Ring 1)	18.37 hr	18.79 hr	18.80 hr
(Ring 2)	18.34 hr	18.75 hr	18.76 hr
(Ring 3)	18.38 hr	18.80 hr	18.81 hr
(Ring 4)	18.61 hr	19.04 hr	19.04 hr
(Ring 5)	20.46 hr	20.93 hr	19.93 hr
(Ring 6)	21.80 hr	21.99 hr	23.22 hr
Auxiliary building failure	-	28.55 hr	21.50 hr
Vessel LH penetration failure			
(Ring 1)	25.45 hr	28.54 hr	24.52 hr
(Ring 2)	25.39 hr	28.83 hr	24.74 hr
(Ring 3)	26.10 hr	29.20 hr	25.49 hr
(Ring 4)	34.49 hr	29.26 hr	26.50 hr
(Ring 5)	30.24 hr	29.71 hr	27.87 hr
(Ring 6)	32.63 hr	32.52 hr	30.22 hr
Cavity Rupture	85.61 hr	73.32 hr	91.41 hr

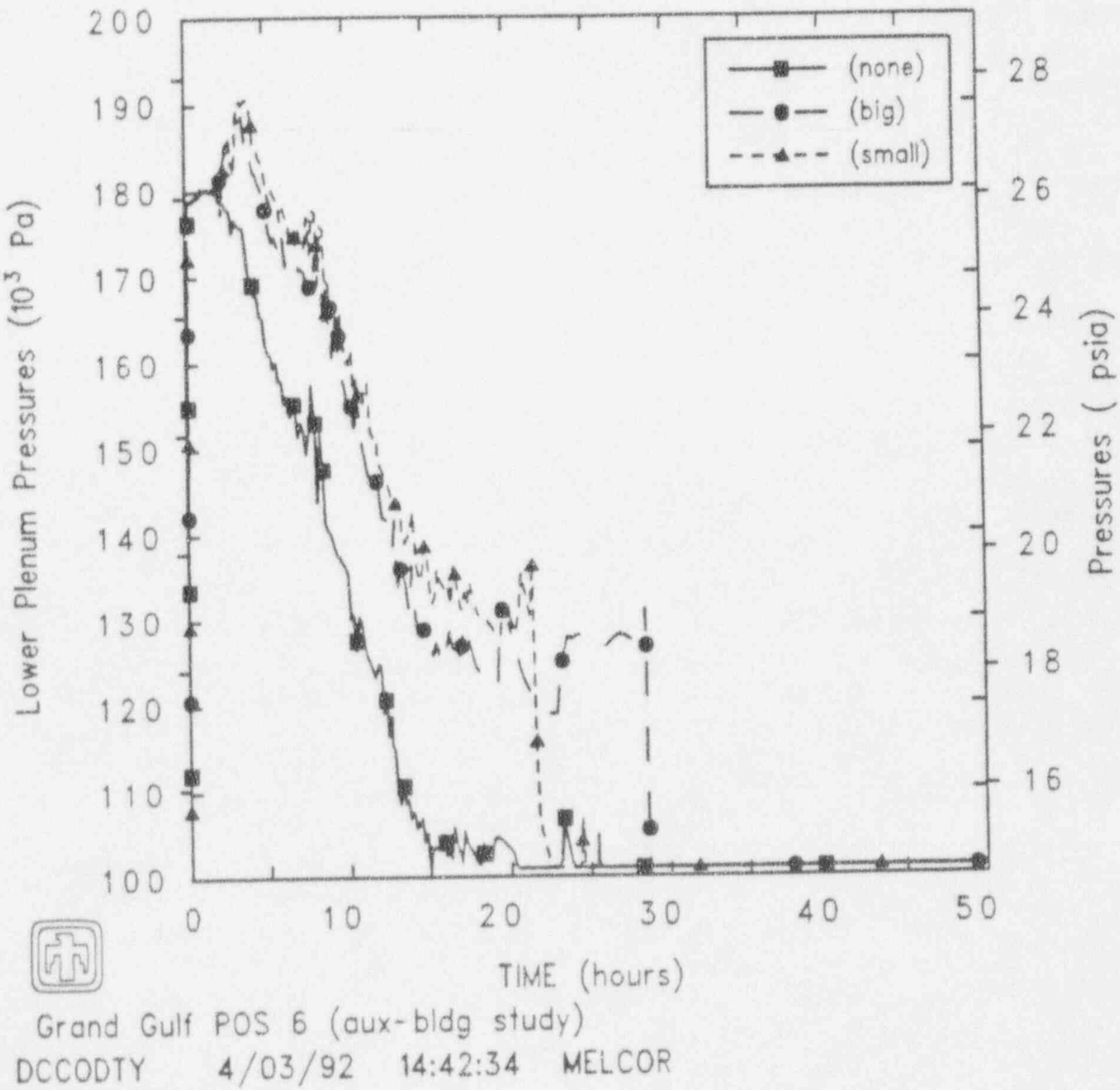


Figure 5.3.1.1. Lower Plenum Pressures for Grand Gulf POS 6 - Auxiliary Building Model Sensitivity Study

The pressures in the outer containment dome for these three cases are given in Figure 5.3.1.2; the pressures in other containment volumes are virtually identical in each calculation. With no auxiliary building and the containment open directly to the environment, the containment pressure remains constant at atmospheric pressure. With an auxiliary building in the model and assuming a 5 psi overpressure failure criterion, the containment (and auxiliary building) pressures rise as steam is generated in the vessel core as water inventory is boiled away, until the auxiliary building fails (at 21.5 hr with the smaller volume assumed and at 28.5 hr with the larger volume) after which time all the pressures drop rapidly to atmospheric.

The presence or absence of the auxiliary building in the MELCOR model affects the circulation flow found in the reference calculation. A substantial outflow from containment into the auxiliary building develops through the equipment hatch and a corresponding inflow into containment from the auxiliary building goes through the lower personnel lock, as illustrated in Figure 5.3.1.3. With no auxiliary building modelled, the flows go directly between the containment and the environment, primarily out through the upper personnel lock and back through the lower personnel lock, and are significantly greater in magnitude. Using either model of the auxiliary building model does not significantly affect this circulation flow.

Clad temperature histories in several core cells, in various axial levels and radial rings, are given in Figure 5.3.1.4, as representative of the overall core response in these three calculations. There is no dominant effect of these three different auxiliary building models on the core temperature response - some cells experience similar heatup and clad/fuel failure behavior, other cells experience faster heatup and earlier clad/fuel failure and yet others later failure.

Table 5.3.1.2 summarizes the radioactive masses released from the fuel and debris for each class, together with the amount released to the environment by the time of cavity rupture, normalized by the initial inventory of each class given in Table 3.1. The varying amounts released with no or different auxiliary building models primarily reflect the differences in core temperature histories and lower head failure times (*e.g.*, the later vessel failure time in the open auxiliary building analysis is a major factor in the higher fission product release fractions in the vessel prior to breach), and to a lesser degree differences in the cavity response. By the end of the transient, all or most of the volatiles (the noble gases, cesium, iodine and tellurium) are released by the end of the transient considered, in all three calculations. Class 3 (the alkaline earths, such as Ba or Sr) and Class 12 (the less volatile main group elements like Sn) show similar and significant releases, with almost half the initial inventories released by the end of the transient. The more refractory trivalents (La), the transition elements (Mo) and uranium show about a percent release by the time of cavity rupture, while the most refractory classes (Ru, Ce and Cd) release only 0.004-0.05% of their initial inventories by the time of cavity rupture. With an auxiliary building modelled, most of the release to the environment is in the form of the noble gases and iodine; with no auxiliary building modelled, a large fraction of the initial inventories of Ba, Te and Sn are also released to the environment.

Figure 5.3.1.5 presents release and distribution histories for Class 4 (I), with both the

1

IMAGE EVALUATION TEST TARGET (MT-3)

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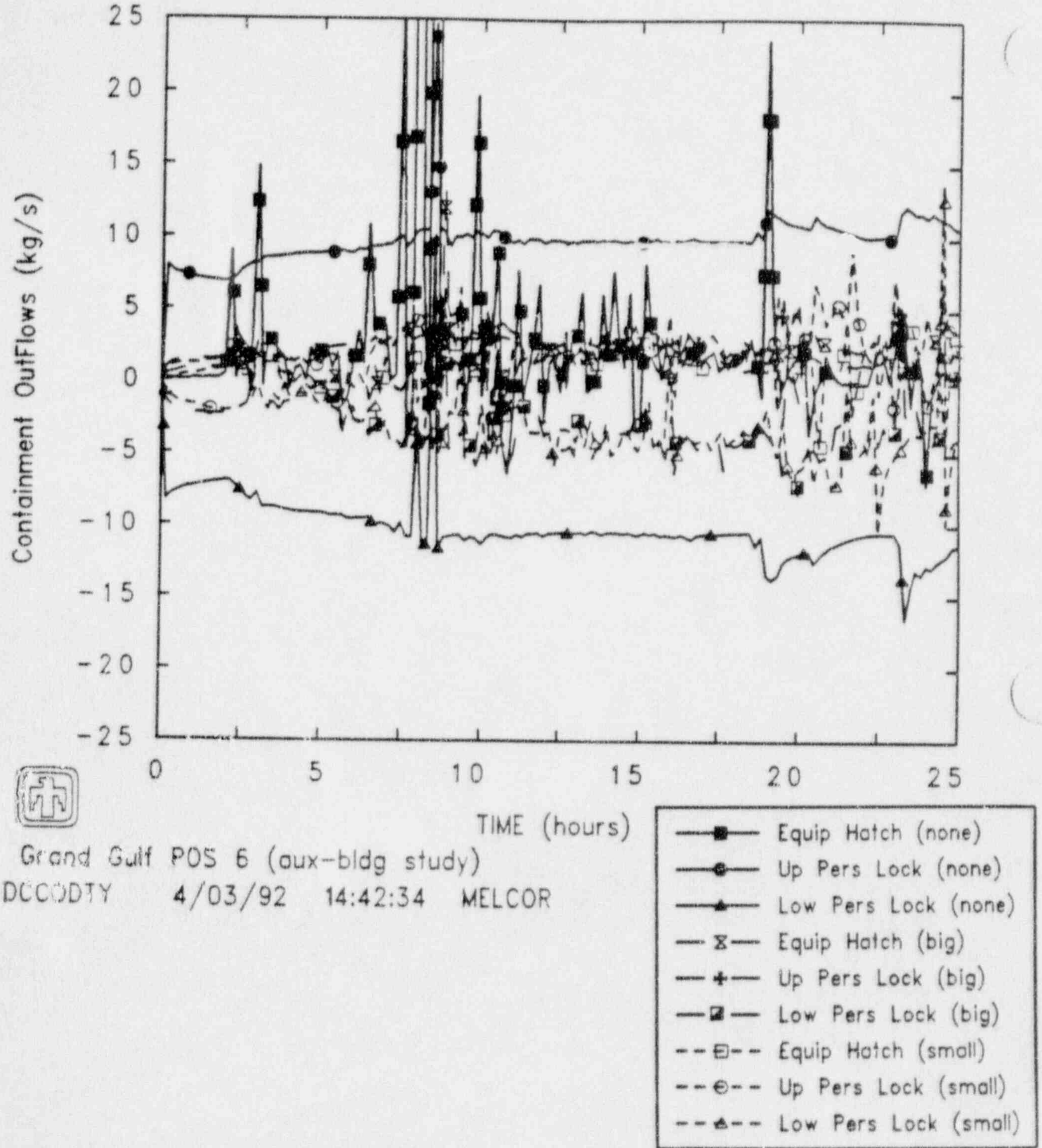


Figure 5.3.1.3. Containment Outflows for Grand Gulf POS 6 - Auxiliary Building Model Sensitivity Study

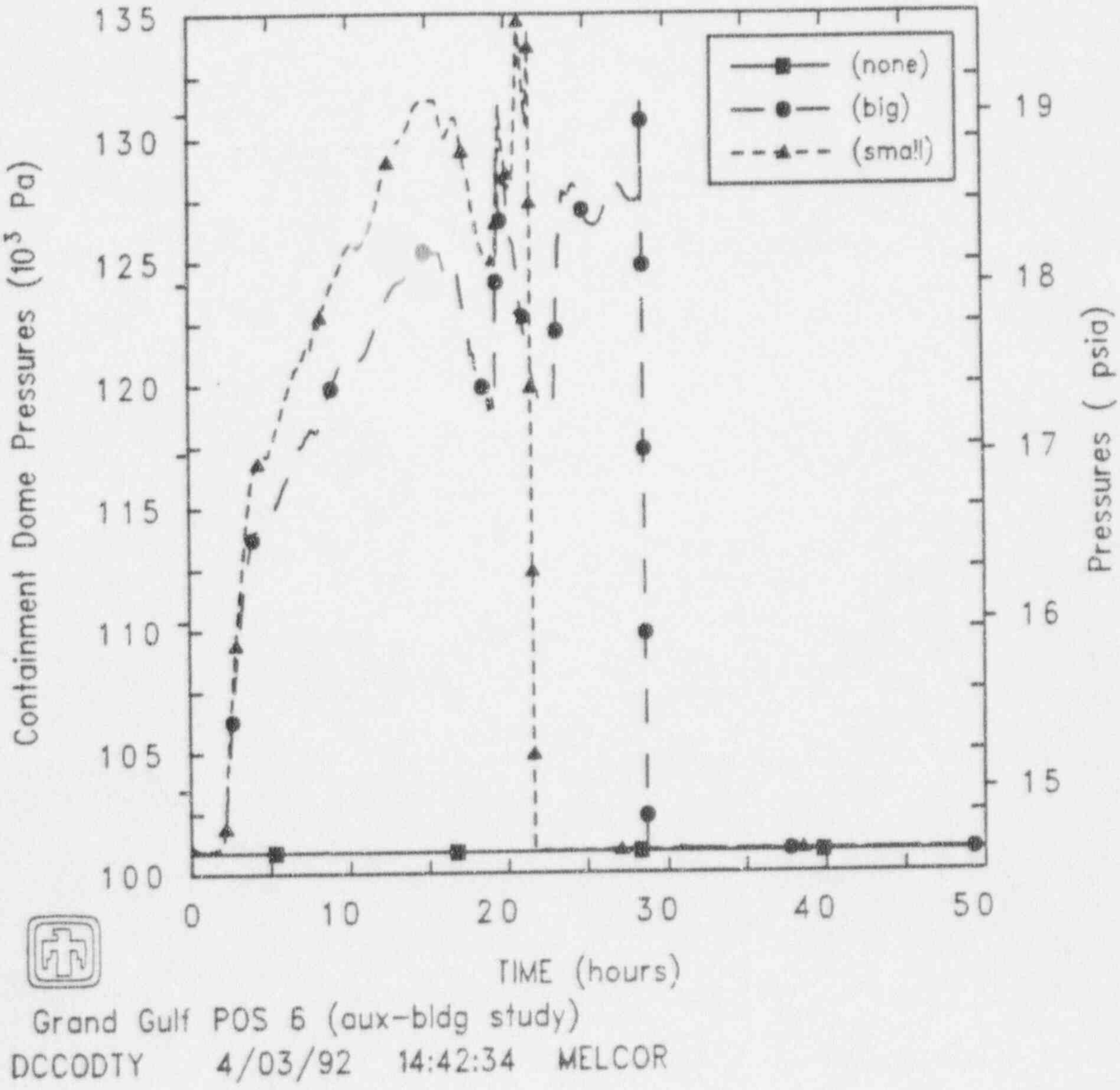


Figure 5.3.1.2. Containment Dome Pressures for Grand Gulf POS 6 - Auxiliary Building Model Sensitivity Study

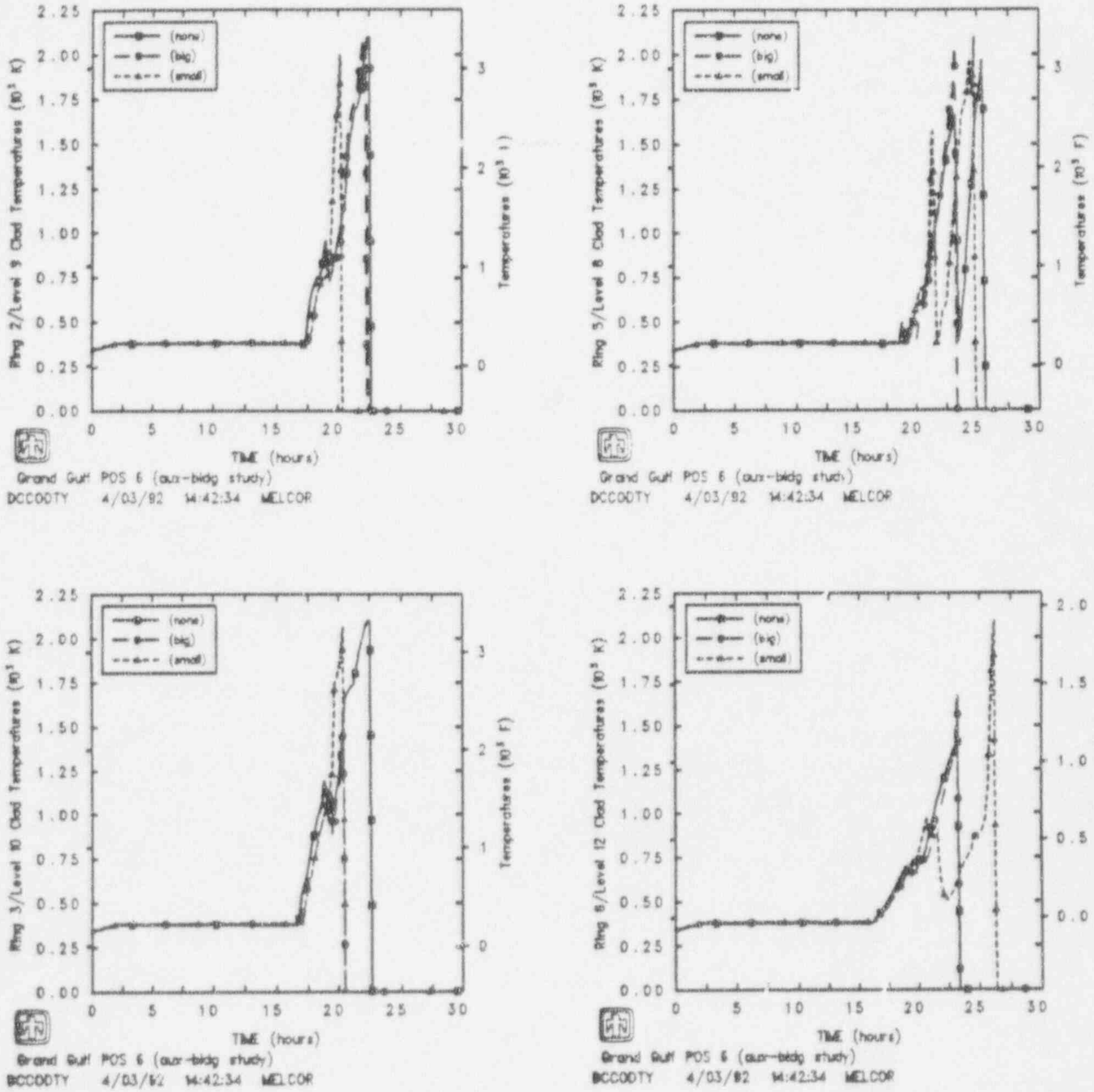


Figure 5.3.1.4. Clad Temperatures for Grand Gulf POS 6 - Auxiliary Building Model Sensitivity Study

Table 5.3.1.2. Total Fission Product Radioactive Mass Releases for Grand Gulf POS 6 - Auxiliary Building Model Sensitivity Study

Class	% of Initial Inventory Released from Fuel			to Environment		
	None	Open	Closed	None	Open	Closed
1 (Xe)	100.0	100.0	100.0	100.0	84.3	93.0
2 (Cs)	100.0	100.0	100.0	67.6	2.54	7.35
3 (Ba)	42.0	44.8	47.4	21.3	1.011	0.0615
4 (I)	95.5	97.3	93.2	95.5	82.5	86.8
5 (Te)	95.2	97.1	93.0	62.3	2.77	5.84
6 (Ru)	0.0070	0.0456	0.050	0.0045	0.00103	0.00056
7 (Mo)	1.61	1.57	0.012	0.642	0.0240	0.0102
8 (Ce)	0.0037	0.0237	0.026	0.00215	0.00056	0.00034
9 (La)	0.2170	2.94	0.280	0.1024	0.1562	0.00689
10 (U)	1.62	1.99	2.16	0.212	0.0447	0.0233
11 (Cd)	0.0385	0.0555	0.039	0.0189	0.0023	0.0018
12 (Sn)	22.5	55.6	37.7	15.5	1.163	0.6603

amounts released and the amounts in the environment at a particular time normalized by the initial mass of the class. (The results for Class 1, the noble gases, are very similar.) By the end of the transient considered, 100% of the initial inventory of noble gases and over 90% of the initial inventory of halogens have been released to the environment, with or without an auxiliary building modelled; the effect of the auxiliary building is seen primarily as a timing delay and a slower rate of release to the environment.

The release and subsequent distribution histories of the alkali metals (Class 2, Cs) and the chalcogens (Class 5) are similar, with the results for tellurium given in Figure 5.3.1.6. Interestingly, although only about 30-35% of these class masses released are retained in the auxiliary building (if modelled), the release to the environment increases much more (to over 60%) if the auxiliary building is neglected. The behavior predicted for the alkaline earth... (Class 3) and the less volatile main group elements (Class 12, Sn) also shows the release to the environment increasing much more (from about 1% to less than 20%) if no auxiliary building is modelled.

The changes in release and distribution of the other classes present much less coherent a pattern for these three different calculations. To a large degree this is because the amounts released are very low, and the behavior extremely sensitive to minor changes in temperature histories and flow patterns predicted. It is not clear whether the differences observed are significant, since for these other classes the releases to environment by the time of cavity rupture are under 1% in all three analyses.

These three calculations all ended on cavity rupture, at various times. The 6 hr difference between the no and closed auxiliary building models probably represents a more reasonable timing difference than the 18 hr difference between the opened and closed model calculations, because of the probable long-term impact of the perturbing time-step effects. Figure 5.3.1.7 shows that these three calculations all predict that the cavity concrete will first be ablated in depth, with various minimum side wall thicknesses of concrete remaining.

5.3.2 Personnel Locks

POS 6 begins when the vessel head is detached and ends when the upper reactor cavity has been filled with water. The steam dryers are removed, vessel water level is lowered to the bottom of the steam lines and the steam lines are plugged, water level is raised and the steam separators are removed, and vessel water level is raised to flood the upper reactor cavity. Prior to this mode of operation, the containment equipment hatch and personnel locks have been opened, the drywell head has been removed and the drywell equipment hatch and personnel locks have been opened.

A circulation flow was found in the reference calculation (Figure 5.2.5), consisting of a substantial outflow from containment into the auxiliary building through the equipment hatch and a corresponding inflow into containment from the auxiliary building through the lower personnel lock. Some of that strong recirculation flow may be physical, while some may be only numerical; the fraction of each contributing is hard to judge. To

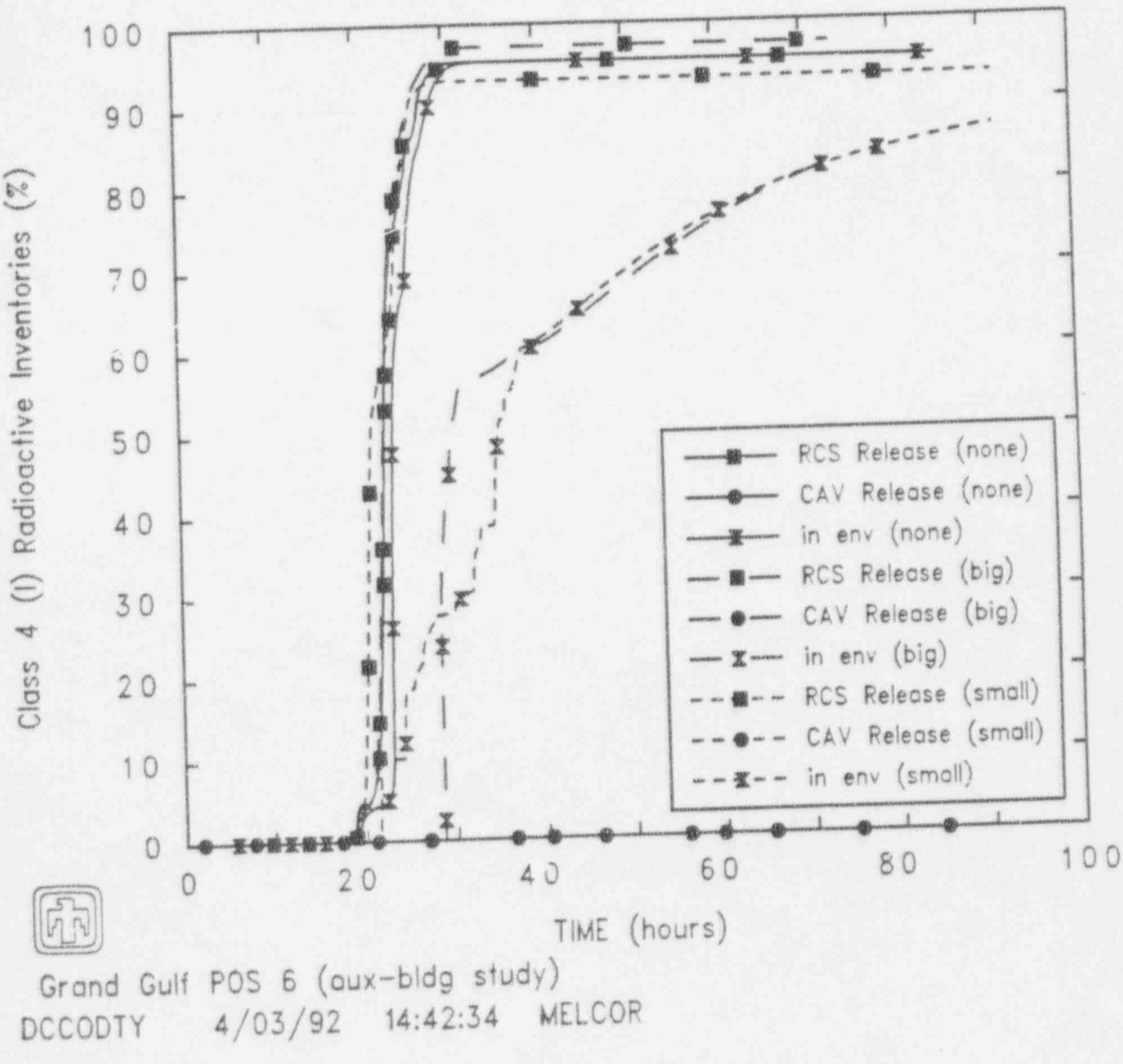


Figure 5.3.1.5. Class 4 Radioactive Mass Radionuclide Distribution for Grand Gulf POS 6 - Auxiliary Building Model Sensitivity Study

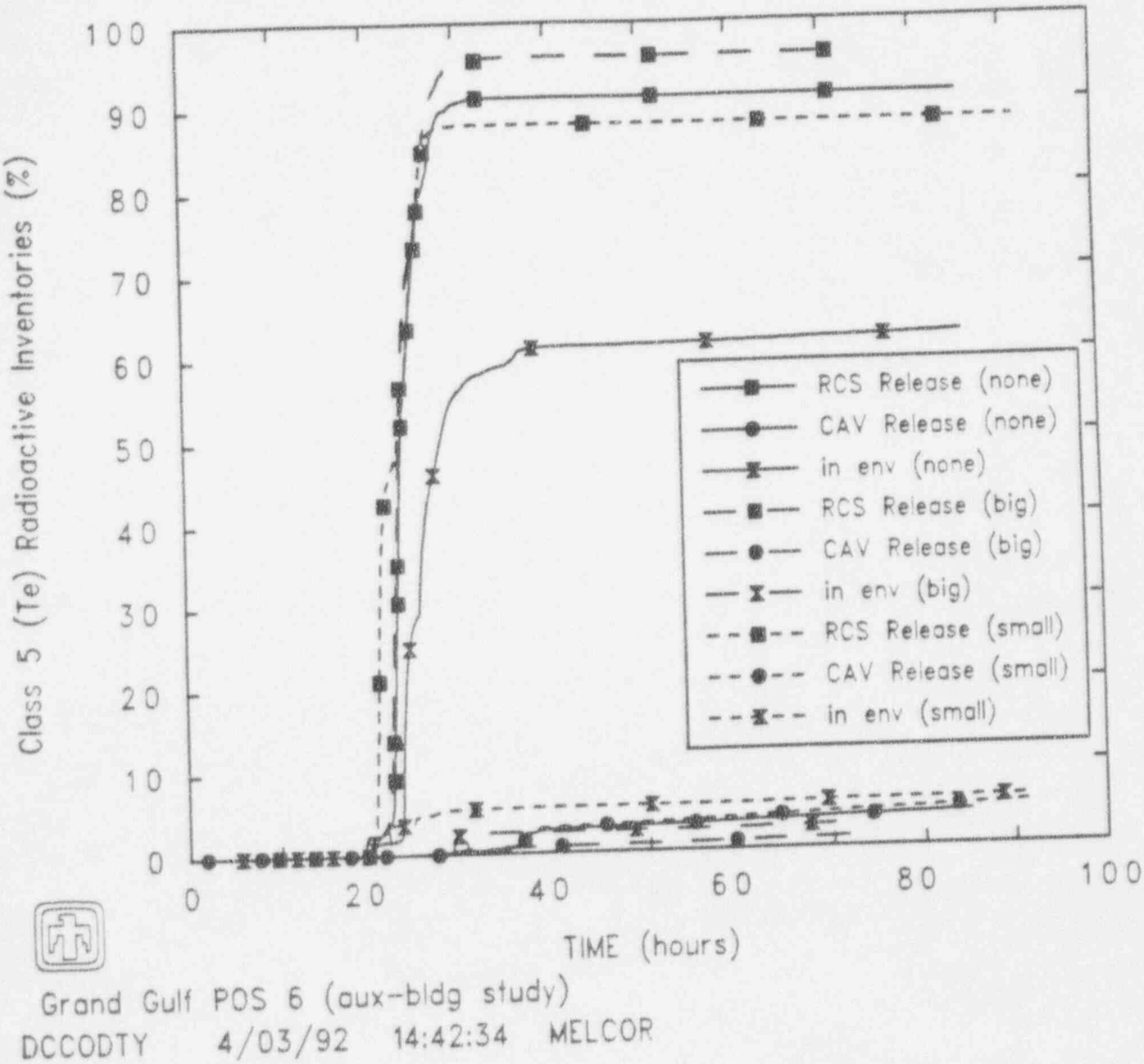


Figure 5.3.1.6. Class 5 Radioactive Mass Radionuclide Distribution for Grand Gulf POS 6 - Auxiliary Building Model Sensitivity Study

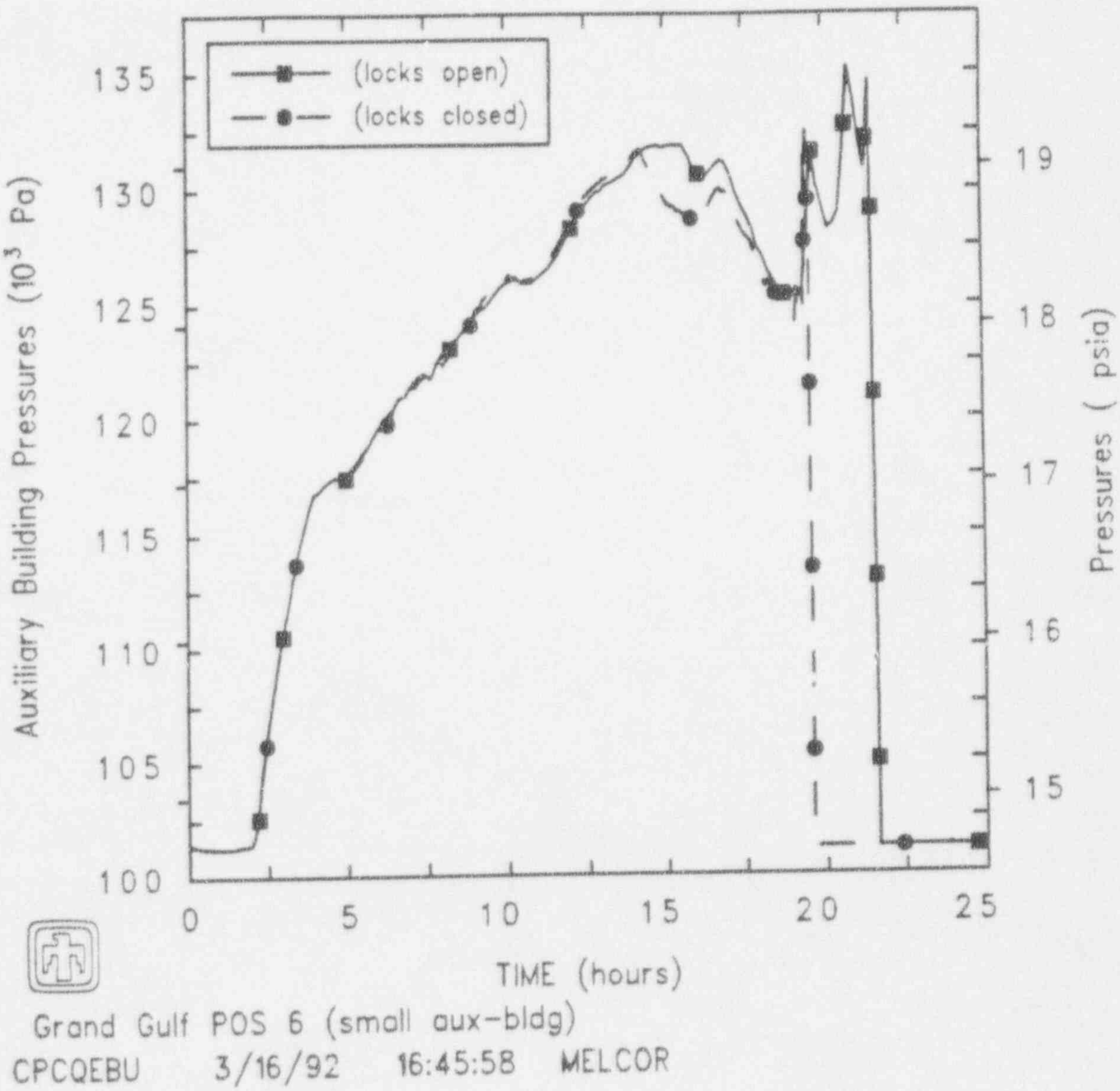


Figure 5.3.2.1. Auxiliary Building Pressures for Grand Gulf POS 6 - Personnel Locks Sensitivity Study

Table 5.3.3.1. Key Event Times for Grand Gulf POS 6 – Containment Isolation Sensitivity Study

Event	Containment Open	Containment Isolated
Level below TAF	12.68 hr	13.57 hr
Clad failure/Gap release		
(Ring 1)	18.37 hr	19.40 hr
(Ring 2)	18.34 hr	19.37 hr
(Ring 3)	18.38 hr	19.42 hr
(Ring 4)	18.61 hr	19.68 hr
(Ring 5)	20.46 hr	20.45 hr
(Ring 6)	21.80 hr	21.36 hr
Vessel LH penetration failure		
(Ring 1)	25.45 hr	28.95 hr
(Ring 2)	25.39 hr	28.96 hr
(Ring 3)	26.10 hr	28.56 hr
(Ring 4)	34.49 hr	28.88 hr
(Ring 5)	30.24 hr	29.75 hr
(Ring 6)	32.63 hr	62.22 hr
Cavity Rupture	85.61 hr	78.83 hr

5.3.3 Closed Containment

The sensitivity study just discussed studied the effects of open *vs* closed containment personnel locks, with the containment equipment hatch open in both cases. Another calculation was done in which the containment equipment hatch was assumed closed, in addition to closed personnel locks, so that the containment remains isolated until the assumed 71 psi containment failure pressure is reached. (This calculation was done with no auxiliary building model.)

The timings of various key events predicted assuming either an open or an isolated containment are presented in Table 5.3.3.1. With the containment isolated, most events take place progressively later, with the exception of cavity rupture terminating the analysis earlier.

Some primary system component and the outer containment dome pressures calculated assuming either an open or an isolated containment are given in Figures 5.3.3.1 and 5.3.3.2, respectively. With the containment open, the containment pressure remains atmospheric and the primary pressure quickly equilibrates to atmospheric as the vessel water inventory is boiled away by the core decay heat. With an isolated containment, the steam generated in the core pressurizes the containment, with the primary system and containment equilibrating at about 175 kPa when the vessel water has fully uncovered the core; afterwards, the reactor vessel, drywell and outer containment pressures are virtually identical. There is a wide pressure spike beginning when the vessel first fails and rising rapidly until all the condensate water drained into the cavity has been evaporated by the hot debris falling from the vessel; the pressure then drops as most of the steam condenses onto walls and pool surfaces, followed by a gradual pressurization later in the transient as the hot cavity atmosphere diffuses through and heats the rest of the containment. The containment failure pressure has not been reached by the time cavity rupture is predicted to occur.

The total amount of hydrogen produced by the time the cavity is breached is quite similar regardless of whether the containment is open or isolated, as demonstrated in Figure 5.3.3.3. With the containment open, 1001 kg of hydrogen is calculated to be produced in the vessel before the core debris falls into the cavity and 1280 kg of hydrogen is generated in the cavity before the cavity is ruptured, for a total of 2281 kg; in the sensitivity study analysis with the containment isolated, more hydrogen is produced through oxidation in the vessel before all the core debris falls into the cavity (1207 kg) but less hydrogen (1098 kg) is generated attacking concrete in the cavity by the time the cavity is ruptured, for a total of 2305 kg, or a 1% difference.

Figure 5.3.3.4 illustrates the clad temperature histories in a core level below the top of the active fuel region in the six core rings, predicted assuming either an open or a closed containment. The heatup rate appears slightly slower in the isolated-containment case than in the open-containment analysis, probably due to the higher system pressures calculated in the closed-containment scenario and resulting in the later lower head penetration failure times.

Table 5.3.3.2 compares the total radioactive masses of radionuclides released in this pair of MELCOR calculations, by the time when a lower head penetration first fails and at the end of the calculation (*i.e.*, when the cavity is predicted to rupture), normalized to the initial masses of each class. The later vessel breach time calculated with the containment isolated (28.5 *vs* 25.4 hr) results in significantly higher release fractions of all of the radionuclide classes (with nonzero releases) by the time of vessel breach. The most volatile classes (Xe, Cs, I and Te) all yield almost 100% release by the end of the transient in both analyses. For all of the less volatile classes, a larger fraction of the initial inventories is released by the time of cavity rupture. For several of the more refractory elements (*e.g.*, Ru and Ce) the amounts released by the end of the transient differ simply by the different amounts released prior to vessel breach; for the others (those with less than 1% release), the increase is not as great because the response is more nonlinear. Of the species with no in-vessel release, a difference is seen only in the trivalents (La), but not for the early transition elements such as Mo and the more volatile main group elements such as Cd. (Because the calculation with the containment isolated did not reach the containment failure pressure prior to transient termination on cavity rupture, there is no release to the environment in this sensitivity study.)

Both calculations ended on cavity rupture, at slightly different times, ~6-7 hr different. Figure 5.3.3.5 shows that both calculations predict that the cavity concrete will first

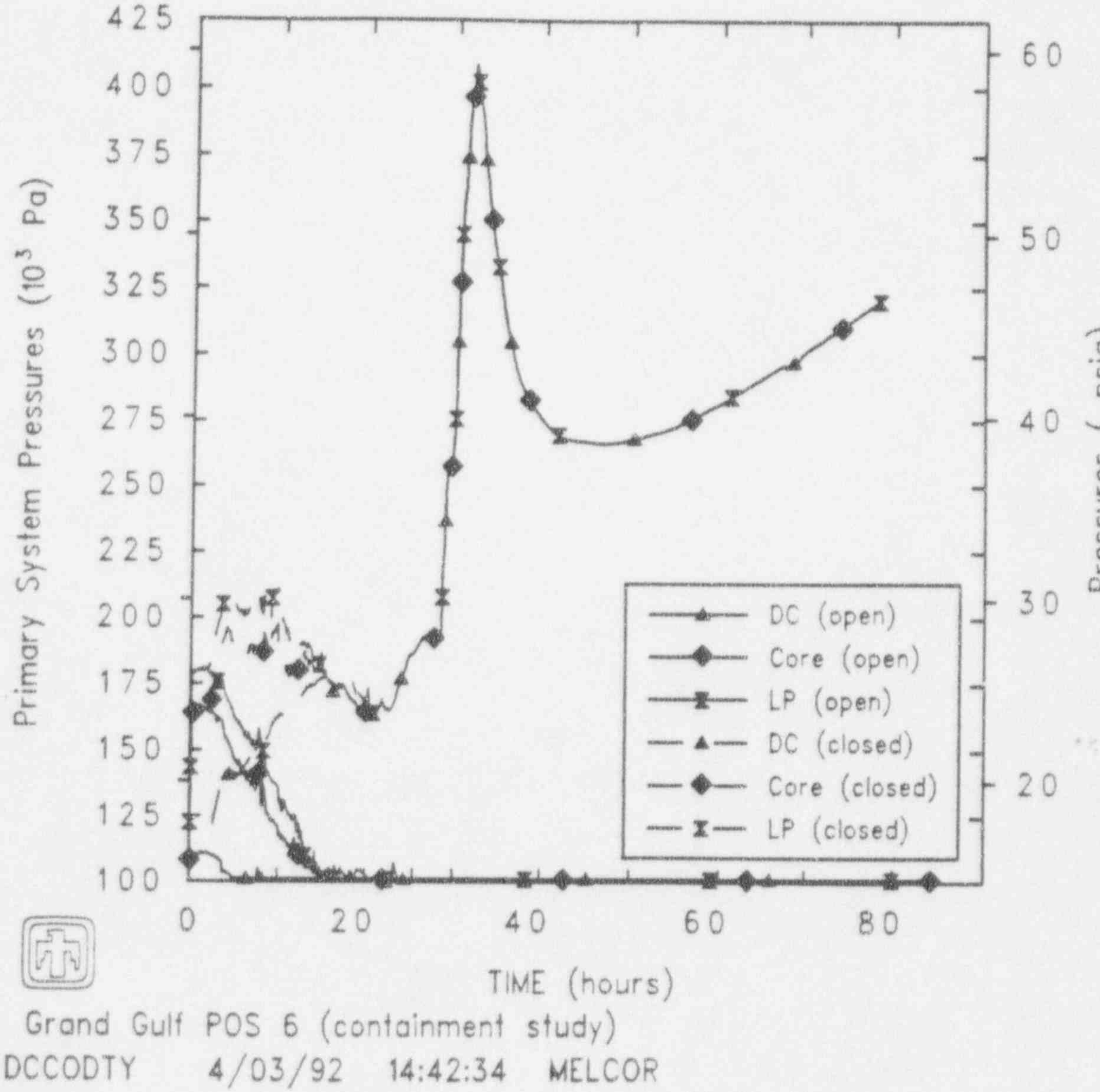


Figure 5.3.3.1. Primary System Pressures for Grand Gulf POS 6 - Containment Isolation Sensitivity Study

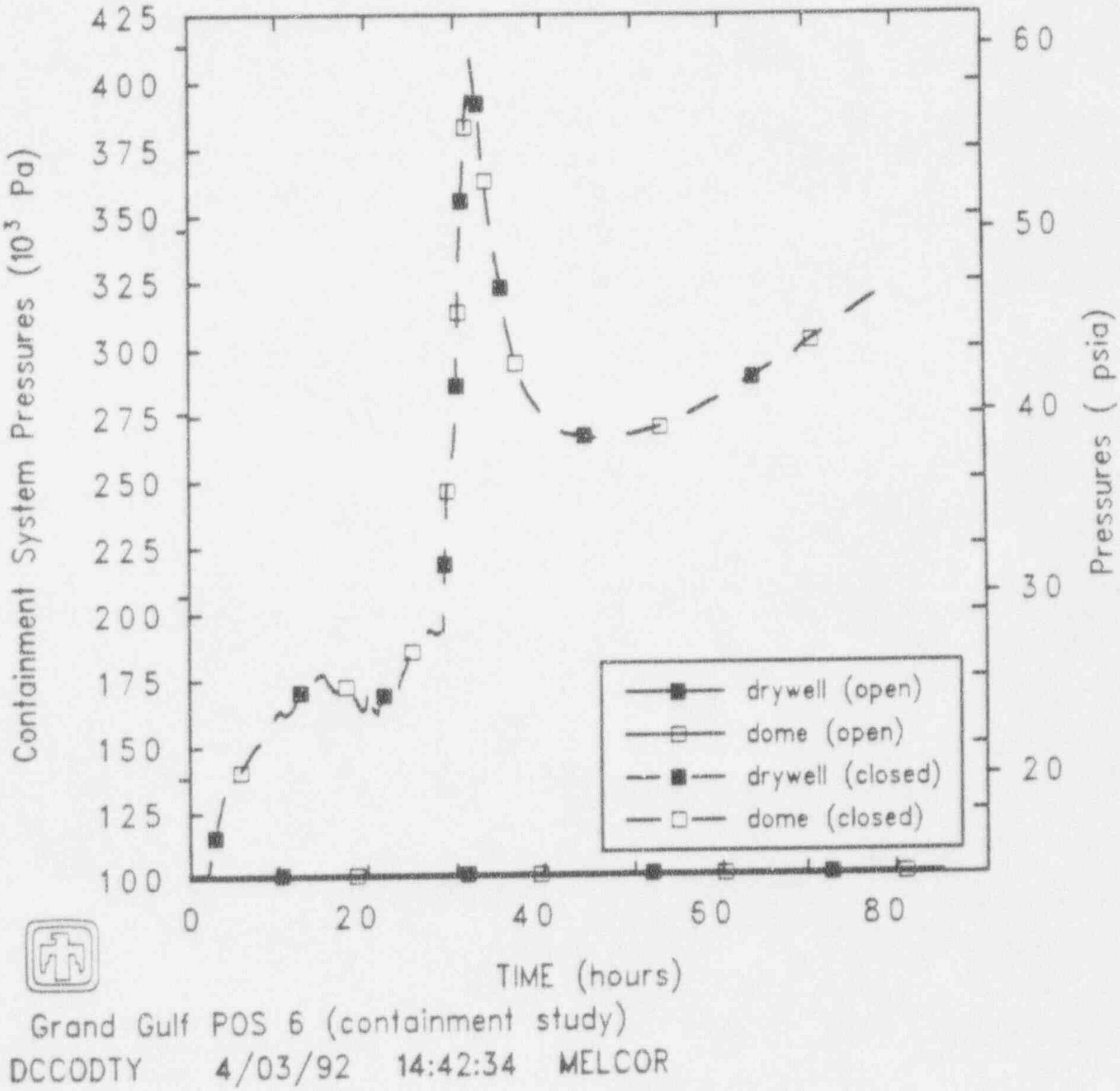


Figure.5.3.3.2. Containment Dome Pressures for Grand Gulf POS 6 - Containment Isolation Sensitivity Study

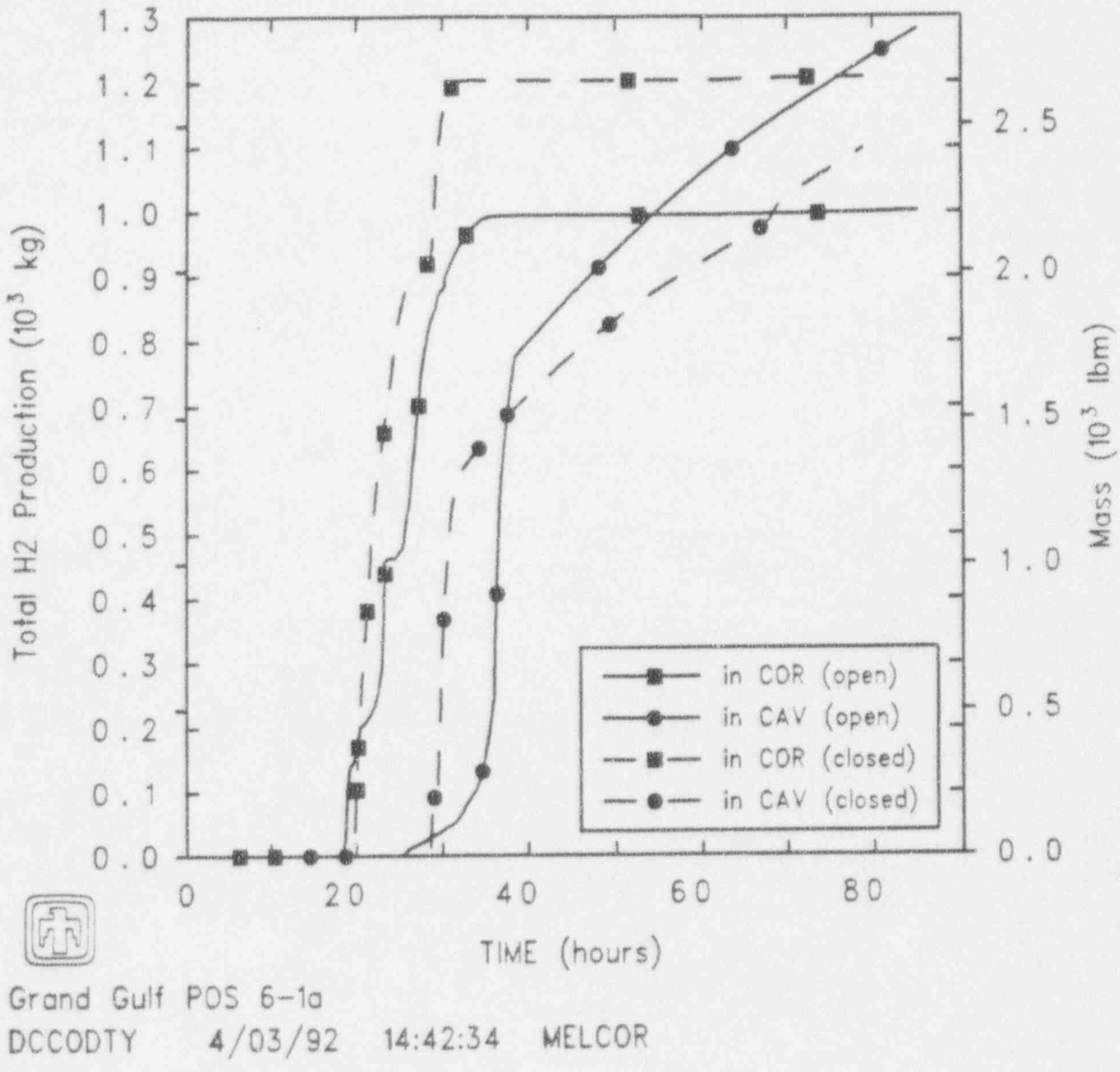


Figure 5.3.3.3. Hydrogen Generation for Grand Gulf POS 6 - Containment Isolation Sensitivity Study

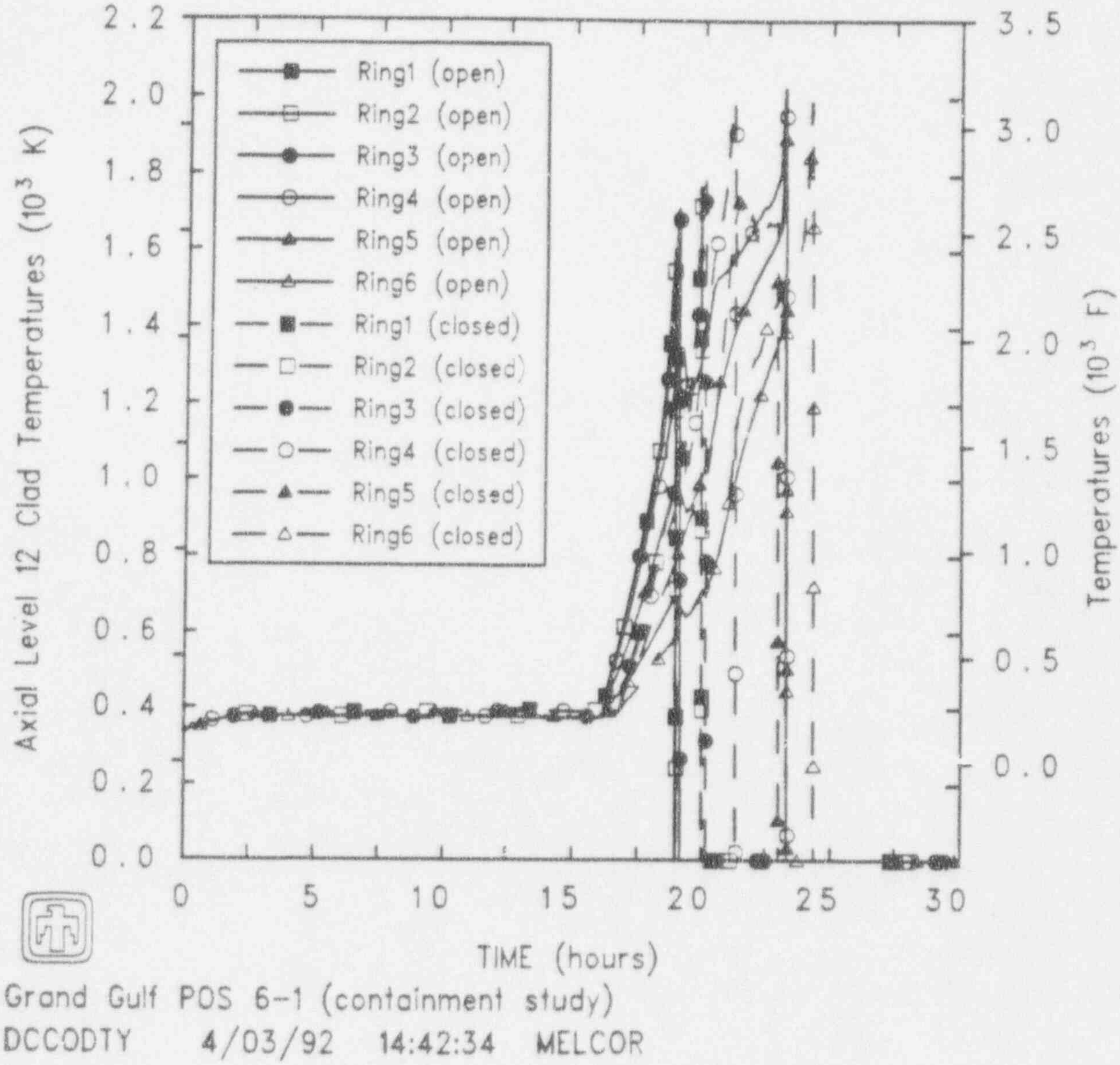


Figure 5.3.3.4. Level 12 Clad Temperatures for Grand Gulf POS 6 - Containment Isolation Sensitivity Study

Table 5.3.3.2. Total Fission Product Radioactive Masses Released from Fuel for Grand Gulf POS 6 - Containment Isolation Sensitivity Study

Class	% of Initial Inventory Released			
	Before Vessel Breach		Before Cavity Rupture	
	Open	Isolated	Open	Isolated
1 (Xe)	81.3	95.6	100.0	100.0
2 (Cs)	81.7	95.7	100.0	100.0
3 (Ba)	2.38	42.6	42.0	59.5
4 (I)	81.0	95.5	95.5	97.1
5 (Te)	72.5	93.5	95.2	97.0
6 (Ru)	0.00002	0.0757	0.0070	0.0847
7 (Mo)	0.0	0.0	1.61	1.61
8 (Ce)	0.000003	0.0376	0.0037	0.0436
9 (La)	0.0	0.0	0.2170	1.201
10 (U)	0.00156	3.26	1.62	3.62
11 (Cd)	0.0	0.0	0.0385	0.0361
12 (Sn)	2.866	61.9	22.5	64.7

be ablated in depth.

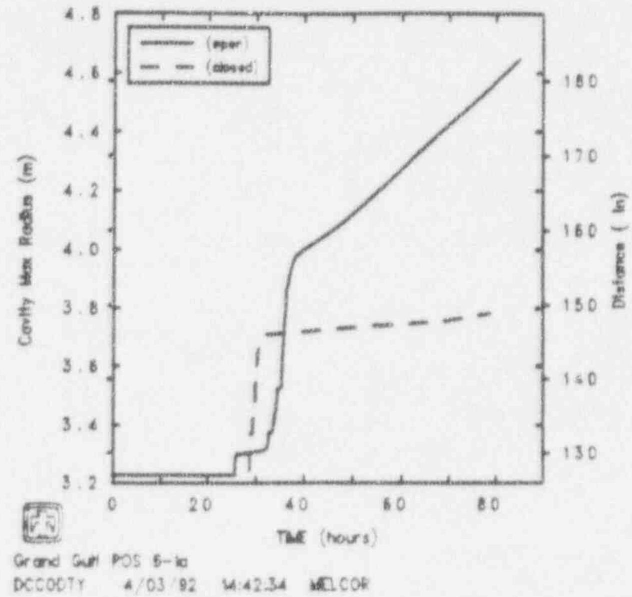
5.3.4 Initiation Time

Timing information for the initiation of the accident in POS 6 is based on Grand Gulf refueling outage (RFO) data. Based on this data, the fastest the plant will enter POS 6 from full power is approximately four days after shutdown and the longest the plant has been in POS 6 (in the going-down phase) is approximately 12 days (*i.e.*, 16 days from shutdown). In the Level 1 analysis the time window from the initiating event to core damage was based on the decay heat at four days; this assumption is carried through the Level 2/3 analyses. Our MELCOR analyses were therefore initiated at 4 days after shutdown, with the exception of a single sensitivity study which assumed the accident sequence to begin 15 days after shutdown. This initiation-time sensitivity study was run with the containment personnel locks and equipment hatch open and venting directly to the environment (*i.e.*, with no auxiliary building modelled).

The decay heat assuming the accident to begin 15 days after scram is about 70% of the decay heat level driving an accident beginning 4 days after scram, as shown in Figure 5.3.4.1. The main effect of the later accident initiation assumed is to delay the timing of all events, as illustrated by comparing the decay power in the primary system (also in Figure 5.3.4.1). The delay in timing is also clearly seen in the vessel water masses in Figure 5.3.4.2, and in the clad temperature histories just below the active fuel midplane presented in Figure 5.3.4.3, and is quantified by comparing the timings of various key events as done in Table 5.3.4.1.

Figure 5.3.4.4 shows that the total amount of hydrogen produced by the time the cavity is breached is quite similar regardless of whether the accident was initiated 4 or 15 days after scram. In the calculation begun 4 days after scram, 1001 kg of hydrogen is produced in the vessel before the core debris falls into the cavity and 1280 kg of hydrogen is generated in the cavity before the cavity is ruptured, for a total of 2281 kg; in the sensitivity study analysis initiated 15 days after scram, more hydrogen is produced through oxidation in the vessel before the core debris falls into the cavity (1318 kg) but less hydrogen (1035 kg) is generated attacking concrete in the cavity before the cavity is ruptured, for a total of 2353 kg, or a 3% difference.

Table 5.3.4.2 compares the total radioactive masses of radionuclides released in this pair of MELCOR calculations, by the time when a lower head penetration first fails and at the end of the calculation (*i.e.*, when the cavity is predicted to rupture), normalized to the initial masses of each class. Before vessel breach, the longer time period that core temperatures are elevated for an accident started 15 days after scram cause significantly higher releases; at the time of cavity rupture, the final releases to environment are some lower and some higher for accidents started 4 days *vs* 15 days after scram, but are generally similar for these two accident scenario calculations.



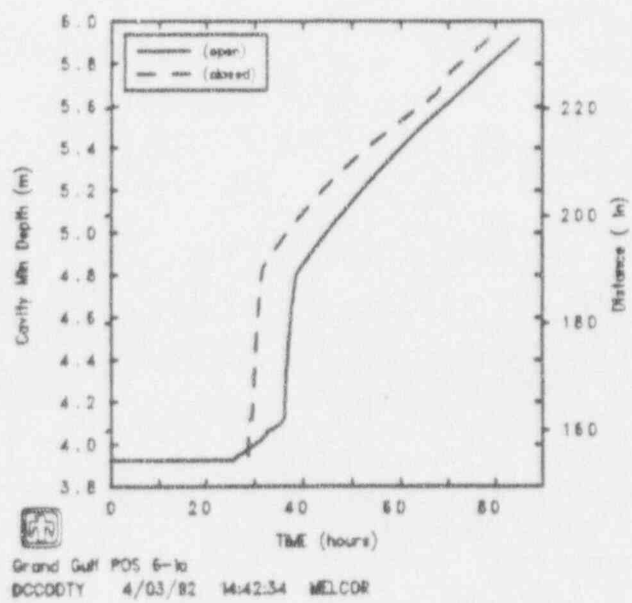


Figure 5.3.3.5. Cavity Maximum Radius and Minimum Depth for Grand Gulf POS 6 - Containment Isolation Sensitivity Study

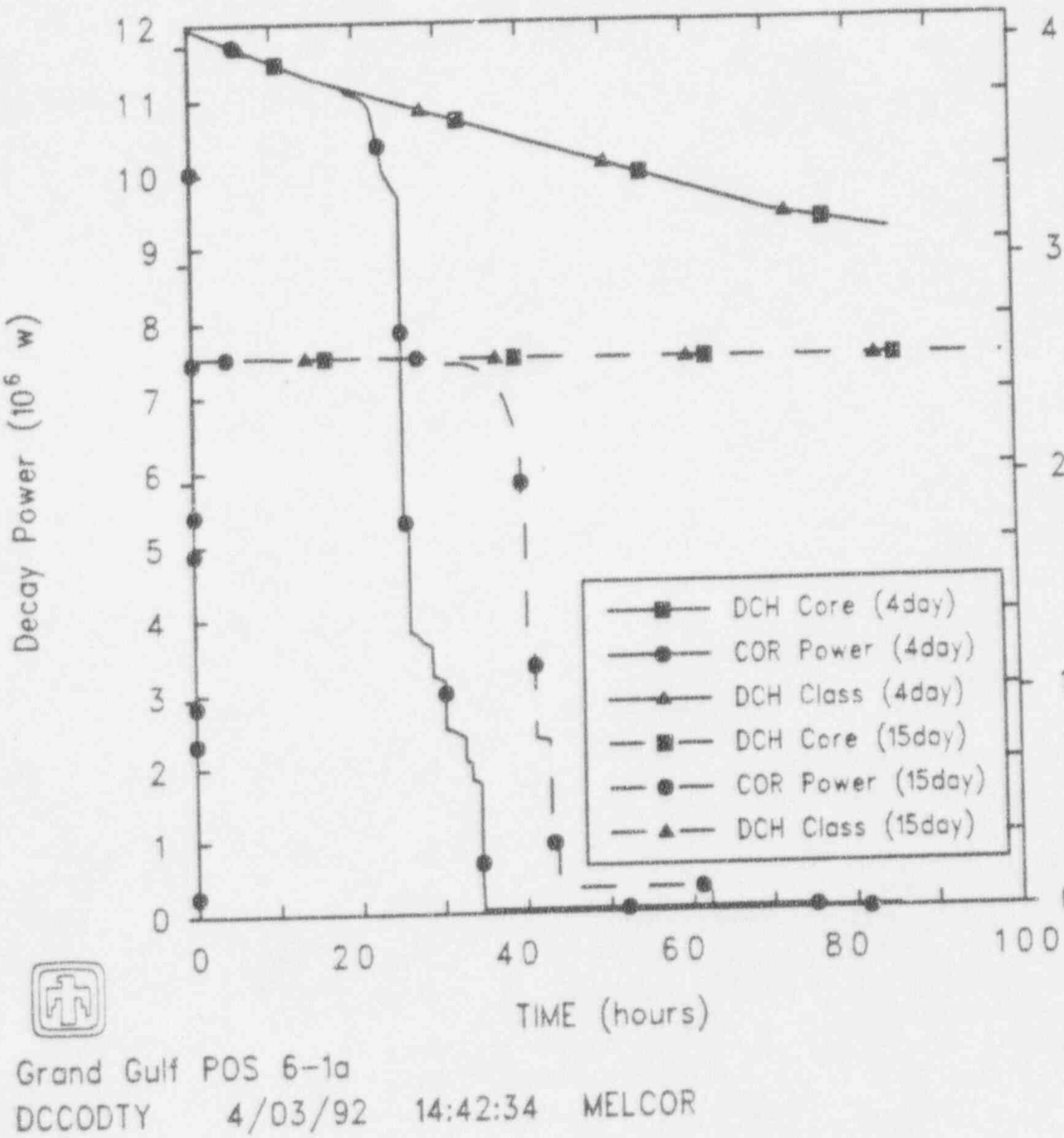


Figure 5.3.4.1. Decay Heat (Total and In-Vessel) for Grand Gulf POS 6 - Accident Initiation Time Sensitivity Study

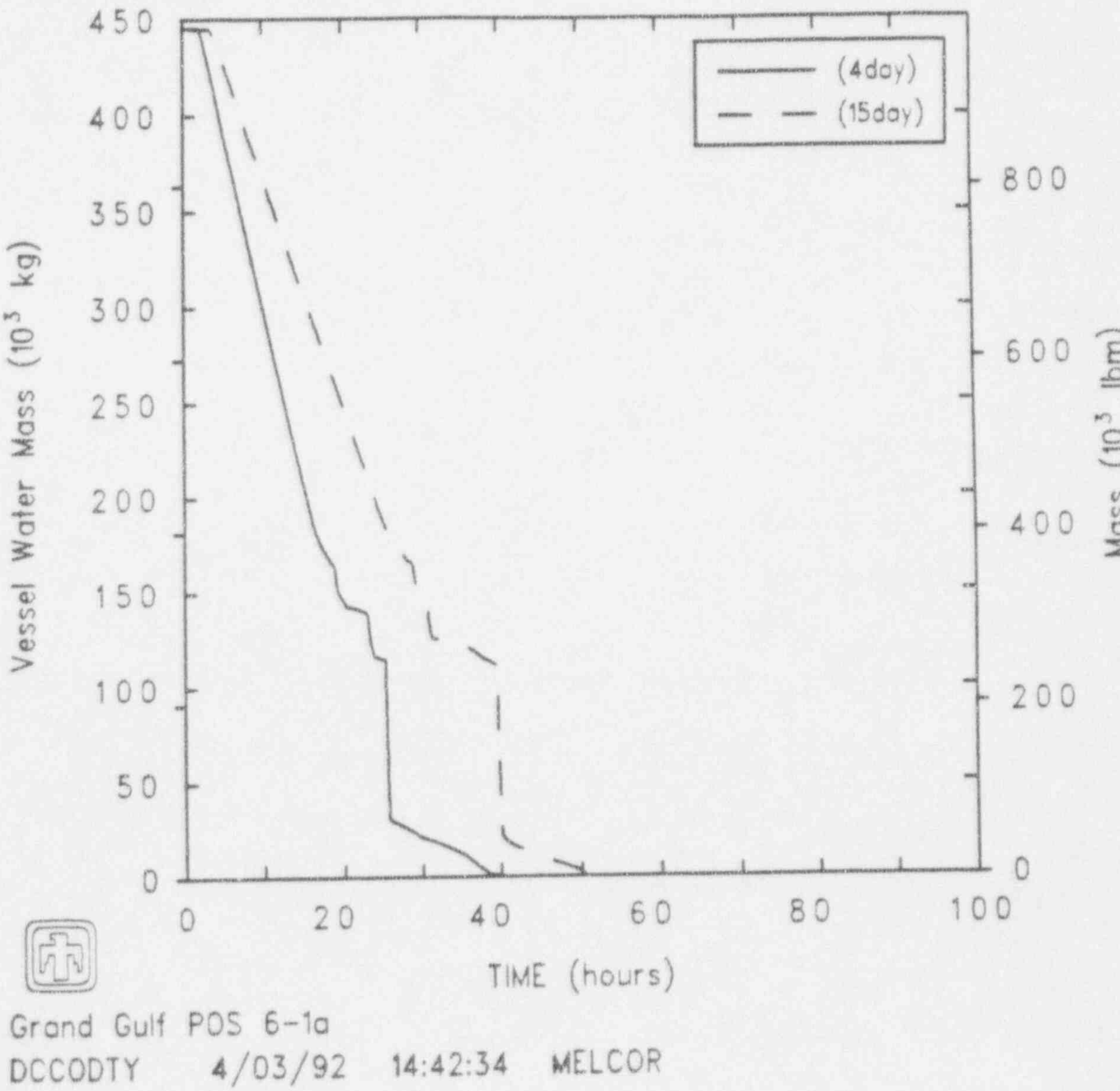


Figure 5.3.4.2. Vessel Water Masses for Grand Gulf POS 6 - Accident Initiation Time Sensitivity Study

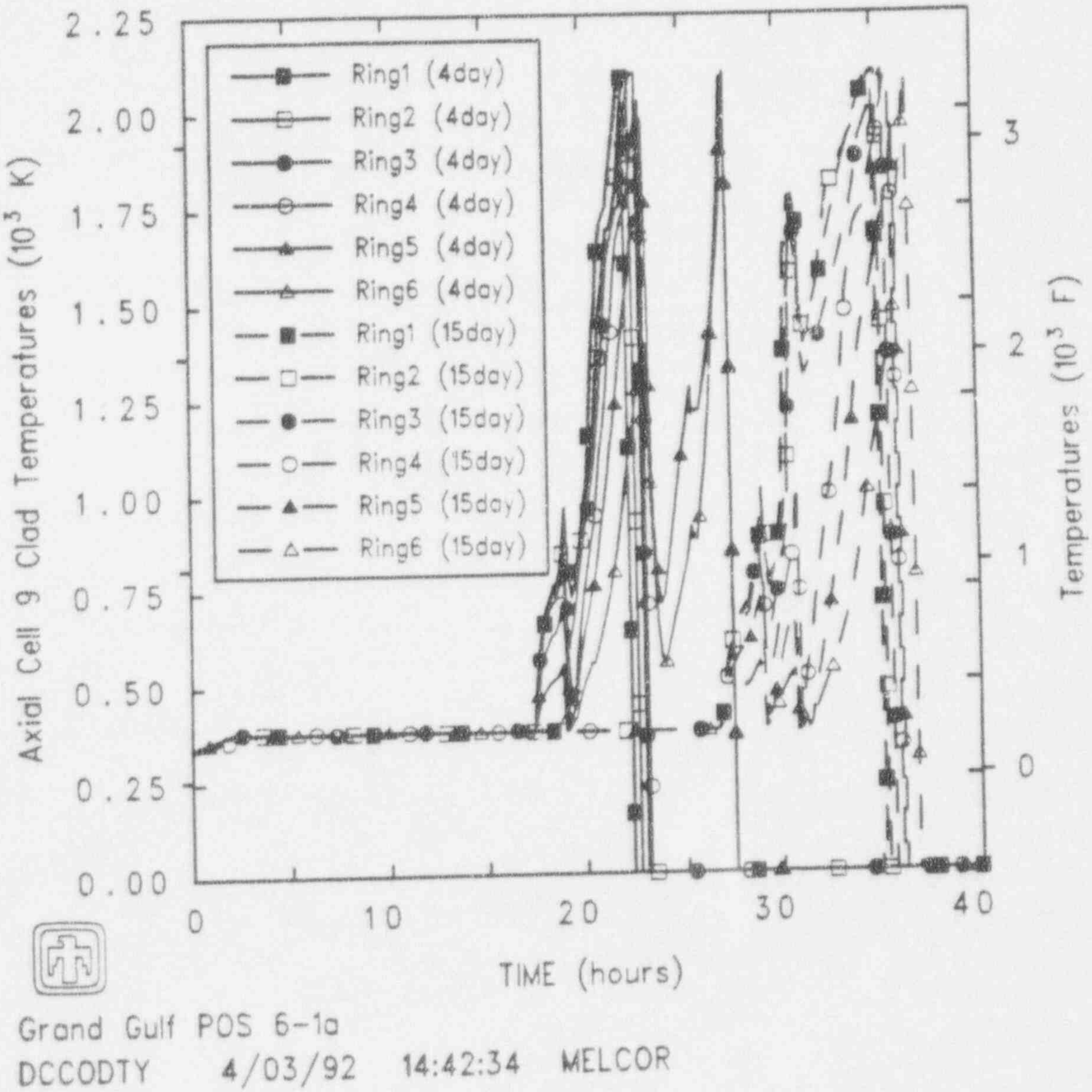


Figure 5.3.4.3. Level 9 Clad Temperatures for Grand Gulf POS 6 – Accident Initiation Time Sensitivity Study

Table 5.3.4.1. Key Event Times for Grand Gulf POS 6 – Accident Initiation Time Sensitivity Study

Event	Time after scram 4 days	Time after scram 15 days
Level below TAF	12.68 hr	19.71 hr
Clad failure/Gap release		
(Ring 1)	18.37 hr	28.35 hr
(Ring 2)	18.34 hr	28.31 hr
(Ring 3)	18.38 hr	28.37 hr
(Ring 4)	18.61 hr	28.62 hr
(Ring 5)	20.46 hr	30.28 hr
(Ring 6)	21.80 hr	34.57 hr
Vessel LH penetration failure		
(Ring 1)	25.45 hr	39.79 hr
(Ring 2)	25.39 hr	40.29 hr
(Ring 3)	26.10 hr	41.22 hr
(Ring 4)	34.49 hr	42.95 hr
(Ring 5)	30.24 hr	43.74 hr
(Ring 6)	32.63 hr	45.68 hr
Cavity rupture	85.61 hr	98.65 hr

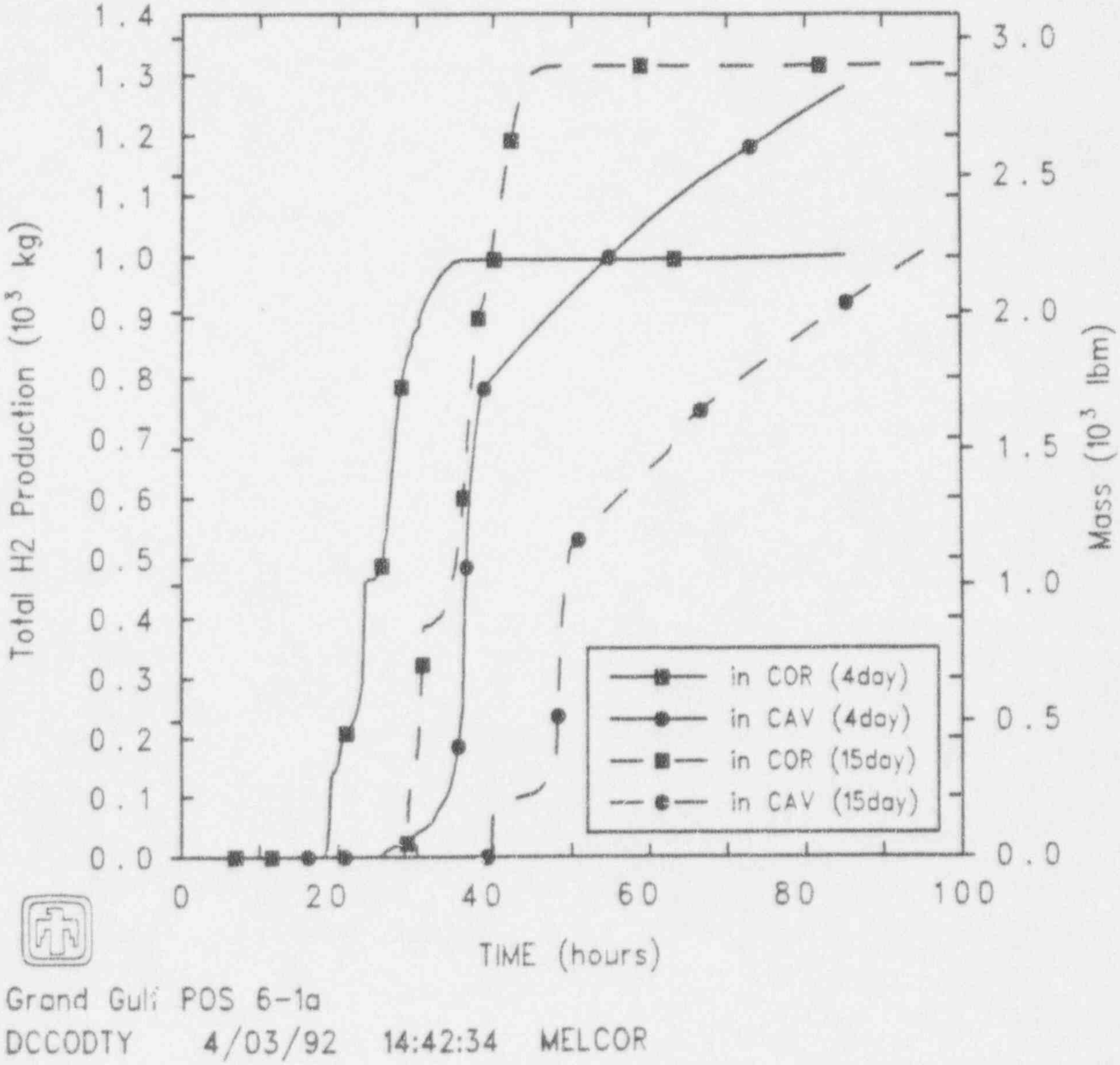


Figure 5.3.4.4. Hydrogen Generation for Grand Gulf POS 6 - Accident Initiation Time Sensitivity Study

Table 5.3.4.2. Total Fission Product Radioactive Masses Released from Fuel for Grand Gulf POS 6 - Accident Initiation Time Sensitivity Study

Class	% of Initial Inventory Released			
	Before Vessel Breach		Before Cavity Rupture	
	4 days	15 days	4 days	15 days
1 (Xe)	76.9	93.8	100.0	100.0
2 (Cs)	76.9	93.9	100.0	100.0
3 (Ba)	4.22	21.6	47.4	41.6
4 (I)	76.1	93.7	93.2	95.2
5 (Te)	62.8	91.6	93.0	95.9
6 (Ru)	0.0023	0.0326	0.0508	0.0403
7 (Mo)	0.0	0.0	0.0128	1.50
8 (Ce)	0.00076	0.0154	0.0264	0.0198
9 (La)	0.0	0.0	0.2809	0.7518
10 (U)	0.1145	1.43	2.16	1.77
11 (Cd)	0.0	0.0	0.0396	0.0334
12 (Sn)	13.53	36.15	37.7	44.4

5.3.5 Igniters

In most of our POS 6 calculations, the hydrogen igniters were assumed to be inactive. A calculation was done, assuming an isolated containment (and no auxiliary building model), in which the igniters were used. The isolated-containment case was chosen to evaluate the effect of the igniters on the calculated pressure rise.

Figure 5.3.5.1 compares the pressures in the outer containment dome, with and without the igniters active. Instead of a large and broad pressure peak around the time of vessel breach, a series of sharp pressure spikes indicating hydrogen burns are calculated prior to vessel breach; these hydrogen burns change the steam/noncondensable mixture sufficiently to ... The magnitude of the burn-generated pressure spikes is not much less than the peak pressure predicted in the absence of igniters. After vessel breach there is no indication of hydrogen burns even with active igniters (because the containment is then steam-inert), and the pressure rises more rapidly, nearing the containment failure pressure at the end of the transient.

The gas temperatures in the containment are presented in Figure 5.3.5.2, in the drywell, cavity and in the outer containment dome, for the calculations with and without active igniters. Temperature spikes indicating hydrogen burns are seen during the 20 to 30 hr period with active igniters. After vessel breach, the calculation with no igniters has a very hot cavity, a cold outer containment dome and an intermediate temperature in the drywell, for a very pronounced temperature gradient; the calculation with hydrogen burns earlier shows very little temperature gradient among the containment control volumes, with all temperatures remaining relatively low.

Table 5.3.5.1 compares the timings of various key events predicted in these closed-containment calculations with and without active igniters. Before the first hydrogen burn (just before 19 hr), the timing of events is identical. Afterwards, there are a few minor differences in first gap release in a few of the rings, and the failure of the lower head penetrations in most of the rings varies by only about 30 min. And, even with the large differences in later-time containment pressure and temperature histories, the end times for these two calculations, when the cavity is predicted to rupture, differ only by about 1 hr.

Table 5.3.5.2 compares the total radioactive masses of radionuclides released in this pair of MELCOR calculations, by the time when a lower head penetration first fails and at the end of the calculation (*i.e.*, when the cavity is predicted to rupture), normalized to the initial masses of each class. In this study, despite the later vessel breach time calculated with active igniters (29 *vs* 28.5 hr), slightly lower release fractions of all of the radionuclide classes (with nonzero releases) are predicted by the time of vessel breach. The most volatile classes (Xe, Cs, I and Te) all have almost 100% release by the end of the transient in both analyses. For all of the less volatile classes, except Mo, a smaller fraction of the initial inventories also is released by the time of cavity rupture in the calculation with igniters active, as much as 50% less than in the no-igniter analysis. (Because these calculations did not reach the containment failure pressure prior to transient termination on cavity rupture, there is no release to the environment in this sensitivity study.)

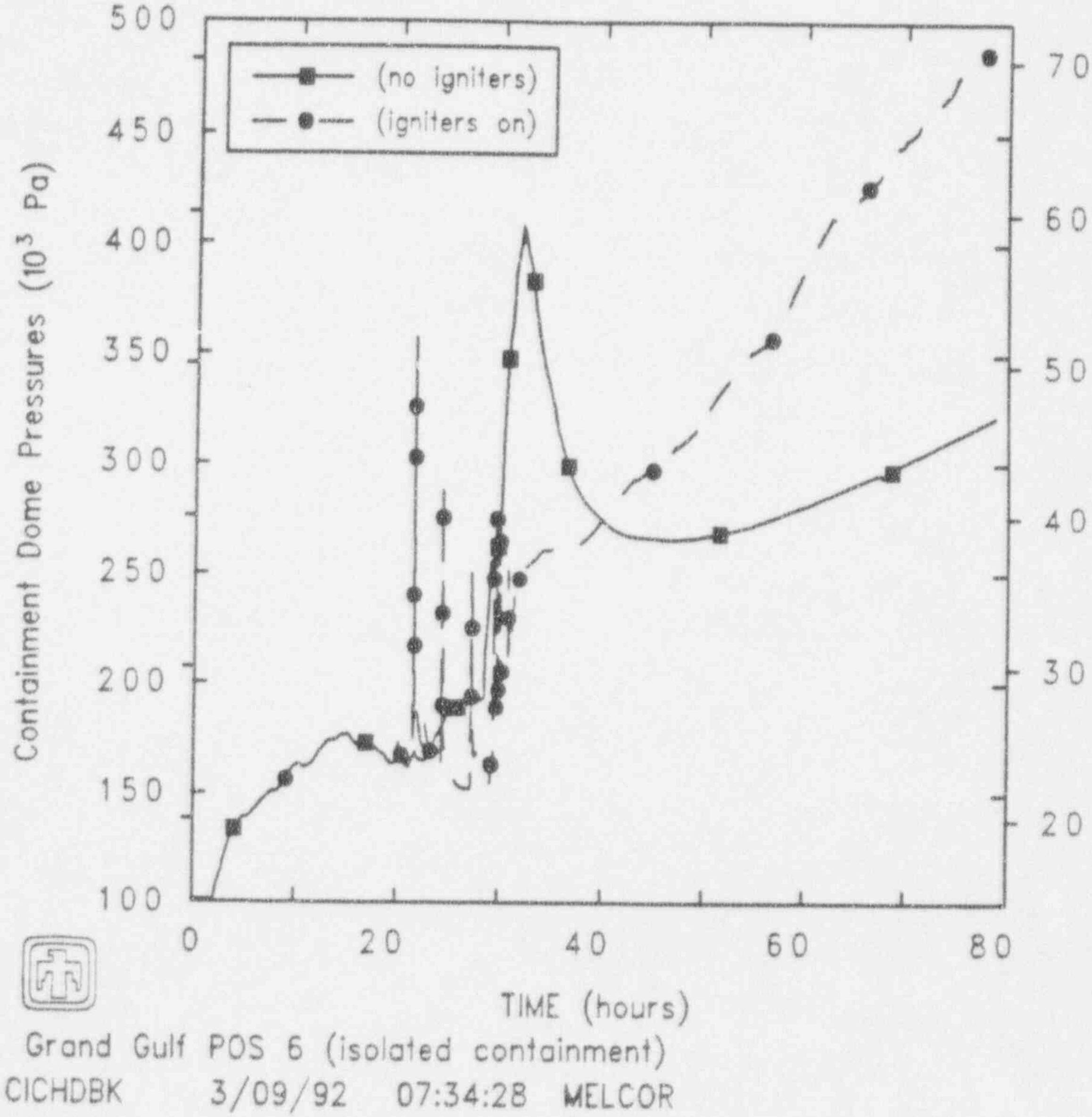


Figure 5.3.5.1. Containment Dome Pressures for Grand Gulf POS 6 - Igniter Sensitivity Study

Table 5.3.5.2. Total Fission Product Radioactive Masses Released from Fuel for Grand Gulf POS 6 - Igniter Sensitivity Study

Class	% of Initial Inventory Released			
	Before Vessel Breach		Before Cavity Rupture	
	Inactive	Active	Inactive	Active
1 (Xe)	95.6	94.3	100.0	99.5
2 (Cs)	95.7	94.4	100.0	99.8
3 (Ba)	42.6	37.5	59.5	58.2
4 (I)	95.5	94.2	97.1	95.6
5 (Te)	93.5	92.0	97.0	96.5
6 (Ru)	0.0757	0.0597	0.0847	0.0667
7 (Mo)	0.0	0.0	1.61	2.11
8 (Ce)	0.0376	0.0295	0.0436	0.0340
9 (La)	0.0	0.0	1.20	0.95
10 (U)	3.26	2.58	3.62	2.87
11 (Cd)	0.0	0.0	0.0361	0.0624
12 (Sn)	61.9	59.0	64.7	61.8

5.4 Code Option Studies

In addition to the plant-configuration sensitivity studies discussed in the previous section, a few sensitivity studies were done on various code options and/or parameters. In one calculation, the CORSOR fission product release model was used instead of the (MELCOR default) CORSOR-M fission product release model. Because it was sometimes necessary to back up and reduce the user-specified maximum time step in order to to complete the analysis, a calculation was done in which that was the only change made, to determine how big an effect reducing the time step would have on the results. Two calculations were done to address concerns [58, 59] raised about the lack of air oxidation modelling in MELCOR, and the associated lack of the extensive release of ruthenium demonstrated to occur when irradiated reactor fuel is heated in air.

5.4.1 Source Term

The POS 6 analysis has been run with a different release model option enabled in MELCOR, as a sensitivity study on fission product source term. The options available include the CORSOR and CORSOR-M models. (The new CORSOR-Booth model was not available in the code version used for these POS 6 analyses.) This source-term sensitivity study was run with the containment personnel locks and equipment hatch open and venting directly to the environment (*i.e.*, with no auxiliary building modelled).

The CORSOR model is a simple correlational relationship based on data from early experiments [63]. Release of volatiles is assumed to be limited by diffusion, and all volatiles share the same release parameters, obtained by averaging experimental results; release of nonvolatiles is assumed to be limited by vaporization, and vapor pressures are scaled for consistency with experimental observations. The fractional release coefficients in CORSOR are simple exponentials, with constants selected for each species in specific temperature ranges based upon fitting experimental data. The fractional release coefficients used in CORSOR-M (the MELCOR default) utilize an Arrhenius-type equation with constants representing empirical fits to experimental data.

Table 5.4.1.1 compares the radioactive masses of radionuclides calculated to be released using the CORSOR and CORSOR-M model options, when a lower head penetration first fails and at the end of the calculation (*i.e.*, when the cavity is predicted to rupture), and the amounts that have been released to the environment, all normalized to the initial masses of each class (given in Table 3.1).

In both calculations, most of the noble gases (Xe), alkali metals (Cs) and halogens (I) have been released by the time of first lower head penetration failure, and most or all of these three classes have been released by the end of the transient, with half or more released to the environment. The CORSOR correlations predict more release of the alkaline earths (Ba), the platinoids (Ru), the tetravalents (Ce) and the less volatile main group elements (Sn); the CORSOR correlations also predict non-zero releases of the early transition elements (Mo), the trivalents (La) and the more volatile main group elements (Cd) prior to vessel breach. The CORSOR-M relations give a higher release for

Table 5.4.1.1. Total Fission Product Radioactive Masses Released from Fuel for Grand Gulf POS 6 – CORSOR Option Sensitivity Study

Class	% of Initial Inventory Released: Before Vessel Breach		Before Cavity Rupture		to Environment	
	CORSOR	CORSOR-M	CORSOR	CORSOR-M	CORSOR	CORSOR-M
1 (Xe)	76.9	81.3	100.0	100.0	100.0	100.0
2 (Cs)	76.9	81.7	100.0	100.0	52.6	67.6
3 (Ba)	17.6	2.38	52.7	42.0	22.0	21.3
4 (I)	76.5	81.0	91.2	95.5	91.2	95.5
5 (Te)	17.1	72.5	60.1	95.2	29.3	62.3
6 (Ru)	0.743	0.00002	1.366	0.0070	0.574	0.0045
7 (Mo)	10.03	0.0	16.7	1.61	6.70	0.642
8 (Ce)	0.0168	0.000003	0.0305	0.0037	0.0128	0.00215
9 (La)	0.077	0.0	0.6844	0.2170	0.325	0.00056
10 (U)	0.077	0.00156	0.176	1.62	0.078	0.1562
11 (Cd)	38.2	0.0	57.0	0.0385	22.6	0.0023
12 (Sn)	38.2	2.866	57.7	22.5	23.0	15.5

the chalcogens (Te), as well as for the volatiles (*i.e.*, the noble gases, alkali metals and halogens). The total releases up to the time of cavity rupture (the end of the calculations) and the releases to the environment follow the qualitative trends seen comparing the in-vessel releases prior to lower head breach. (These trends are the same as seen in several recent MELCOR assessment calculations [64, 17].) However, the releases to the environment calculated for the two release options are not simply equal fractions of the amounts released from the fuel and debris; the fission product transport is apparently dependent to some extent on the amounts and relative amounts of the fission products present.

5.4.2 Time Step

Several of the grand Gulf POS 6 MELCOR calculations aborted with various error messages at assorted during the core degradation process. In all cases, it was possible to back up, reduce the user-specified maximum time step to below that used by the code just prior to developing problems, and complete the analysis. There has been a lot of discussion in the past few years [65] on numeric effects seen in various MELCOR calculations, producing either differences in results for the same input on different machines or differences in results when the time step used is varied. To determine how big an effect reducing the time step would have on the results, a calculation was done in which that was the only change made.

In most of our calculations, the maximum allowed time step was set through user input to be 99 s, so that the code used its internal logic to select a time step. In this sensitivity study, the time step was reduced to 0.5 s from 70,000 s (19.444 hr) to 100,000 s (27.778 hr).

The change in time step affects some of the event timings, as illustrated in Table 5.4.2.1. There is, of course, no difference in the timing of events before the time step reduction. The changes in timing of key events after the time step reduction are generally small.

There are no major differences observable in primary and containment systems pressure histories, or core inventory boiloff. Figure 5.4.2.1 compares clad temperature histories in a core level above the active fuel midplane in the six core rings as representative of the overall core response. Small offsets are visible in the temperatures predicted with the reduced time step, resulting in the slightly later lower head penetration failure times.

The total amount of hydrogen produced by the time the cavity is breached is greater in the calculation with the temporarily-reduced time step (2450 kg, ~7.4% high compared to 2281 kg). Figure 5.4.2.2 indicates that the major difference is in significantly more hydrogen generated during in-vessel core degradation (1299 kg *vs* 1001 kg of hydrogen produced in the vessel in the basecase); less is generated later in the cavity (~1151 kg compared to 1280 kg in the basecase) before the cavity is ruptured.

Table 5.4.2.2 compares the total radioactive masses of radionuclides released in this pair of MELCOR calculations, by the time when a lower head penetration first fails and

Table 5.4.2.1. Key Event Times for Grand Gulf POS 6 - Time Step Sensitivity Study

Event	Base Δt	Reduced Δt
Level below TAF	12.68 hr	12.68 hr
Clad failure/Gap release		
(Ring 1)	18.37 hr	18.37 hr
(Ring 2)	18.34 hr	18.34 hr
(Ring 3)	18.38 hr	18.38 hr
(Ring 4)	18.61 hr	18.57 hr
(Ring 5)	20.46 hr	20.85 hr
(Ring 6)	21.80 hr	21.69 hr
Vessel LH penetration failure		
(Ring 1)	25.45 hr	26.00 hr
(Ring 2)	25.39 hr	26.01 hr
(Ring 3)	26.10 hr	26.15 hr
(Ring 4)	34.49 hr	31.05 hr
(Ring 5)	30.24 hr	33.13 hr
(Ring 6)	32.63 hr	33.69 hr
Cavity rupture	85.61 hr	82.22 hr

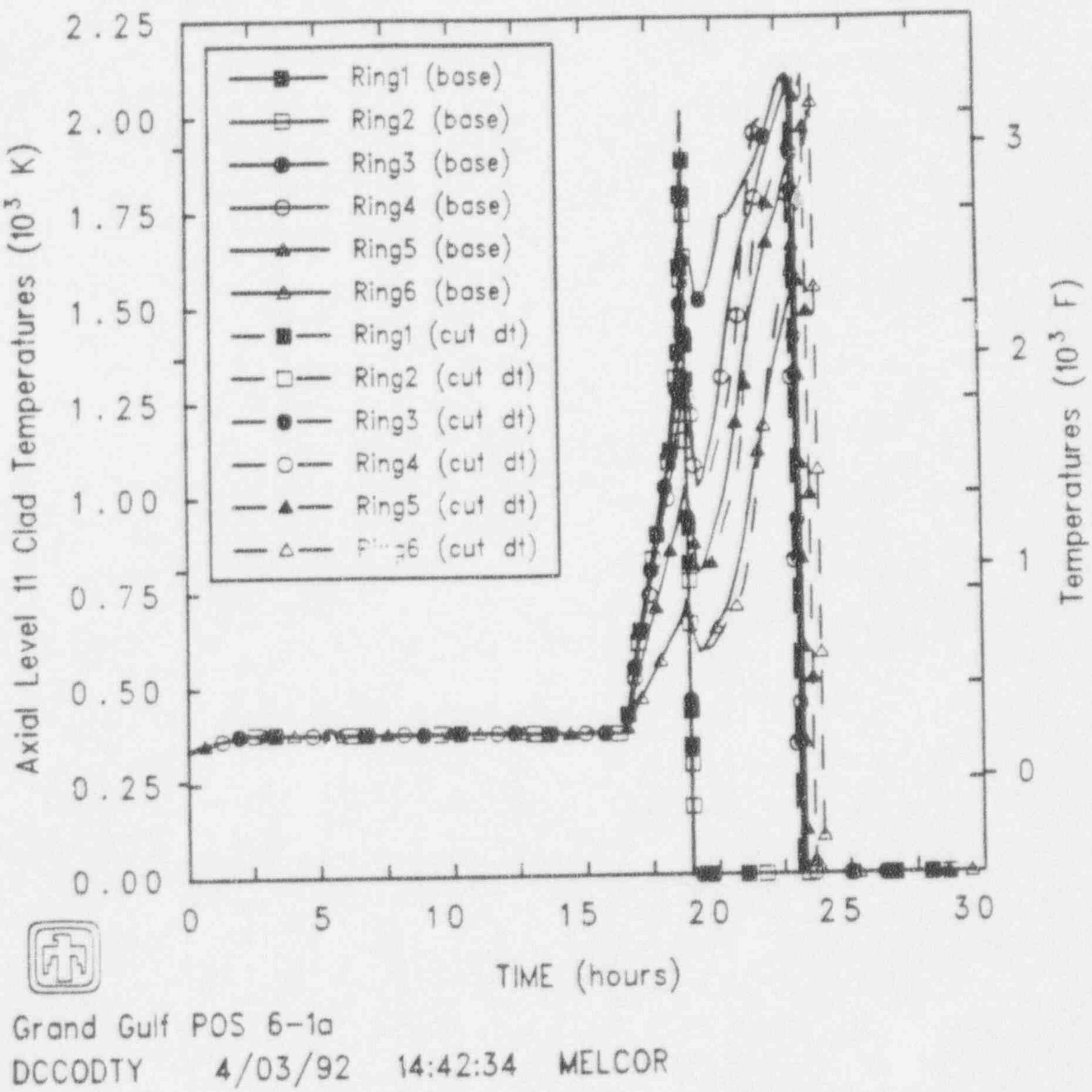


Figure 5.4.2.1. Level 10 Clad Temperatures for Grand Gulf POS 6 - Time Step Sensitivity Study

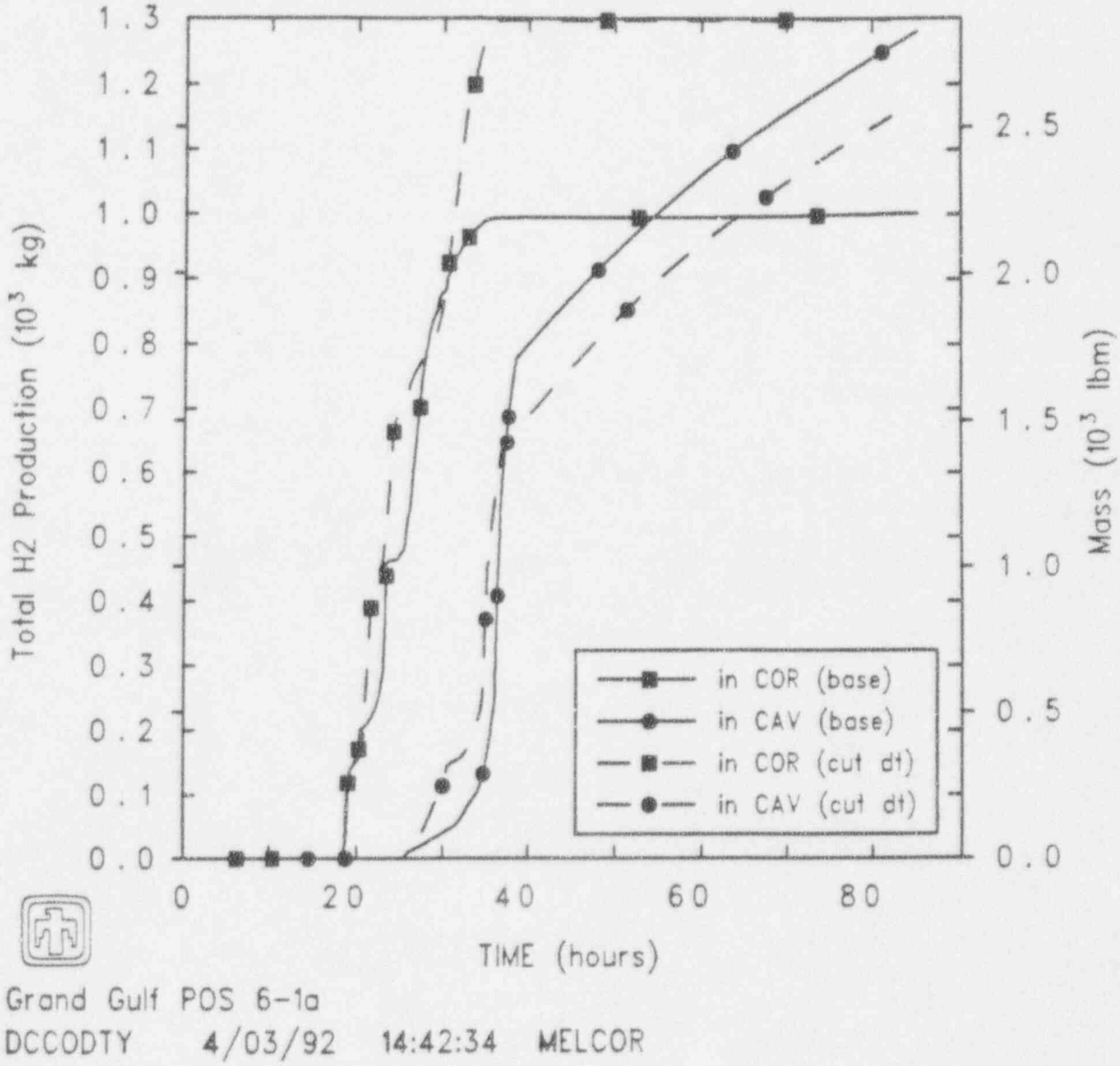


Figure 5.4.2.2. Hydrogen Generation for Grand Gulf POS 6 - Time Step Sensitivity Study

Table 5.4.2.2. Total Fission Product Radioactive Masses Released from Fuel for Grand Gulf POS 6 - Time Step Sensitivity Study

% of Initial Inventory Released

Class	Before Vessel Breach		Before Cavity Rupture		to Environment	
	Base Δt	Cut Δt	Base Δt	Cut Δt	Base Δt	Cut Δt
1 (Xe)	81.3	92.8	100.0	100.0	100.0	100.0
2 (Cs)	81.7	92.9	100.0	100.0	67.6	52.0
3 (Ba)	2.38	6.85	42.0	41.9	21.3	20.3
4 (I)	81.0	92.7	95.5	95.7	95.5	95.7
5 (Te)	72.5	80.2	95.2	95.3	62.3	58.2
6 (Ru)	0.00002	0.0048	0.0070	0.0068	0.0045	0.0030
7 (Mo)	0.0	0.0	1.61	1.76	0.642	0.739
8 (Ce)	0.000003	0.0019	0.0037	0.0035	0.00215	0.00156
9 (La)	0.0	0.0	0.2170	0.4974	0.00056	0.260
10 (U)	0.00156	0.226	1.62	0.332	0.156	0.151
11 (Cd)	0.0	0.0	0.0385	0.0757	0.0023	0.039
12 (Sn)	2.866	20.8	22.5	34.8	15.5	17.4

at the end of the calculation (*i.e.*, when the cavity is predicted to rupture), together with the release to the environment by the end of the transient, normalized to the initial masses of each class. The increased in-vessel hydrogen generation in the calculation with the time-step reduction is associated with increased release of all radionuclide classes prior to vessel breach. (The same trend, increased release fractions with reductions in time step, were found in MELCOR assessment analyses of the ACRR ST-1/ST-2 source term experiments [64].) These increased releases early in the transient do not significantly change the total amounts of most classes released by the end of the calculations, but larger amounts of the trivalents (La) and both the more and less volatile main group elements (Cd and Sn) are released by the time of cavity rupture; however, less uranium is released by the time of cavity rupture, even though more uranium is released in-vessel during the (relatively brief) reduced-dt period. The amounts released to the environment also vary somewhat for most of the classes, but not proportionally to the differences in either early-time or end-time releases. These variations are not very significant because the differences in amounts released to the environment are smallest for those classes with the greatest release from the fuel; the differences increase as the fractional amounts released to the environment decrease and only the release to the environment of the trivalents (La) and the more volatile main group elements (Cd) differ by more than an order of magnitude.

The calculation with the time-step reduction at the time of vessel breach predicts cavity rupture about 3 hr earlier; the comparison of cavity maximum radii and minimum altitudes in Figure 5.4.2.3 demonstrates that the axial ablation is very similar in both cases, but that there is much less radial ablation during the time the time step is cut, resulting in a constant offset in maximum radii throughout the remainder of the transient, even after the time step is increased back to its original value.

5.4.3 Air Oxidation

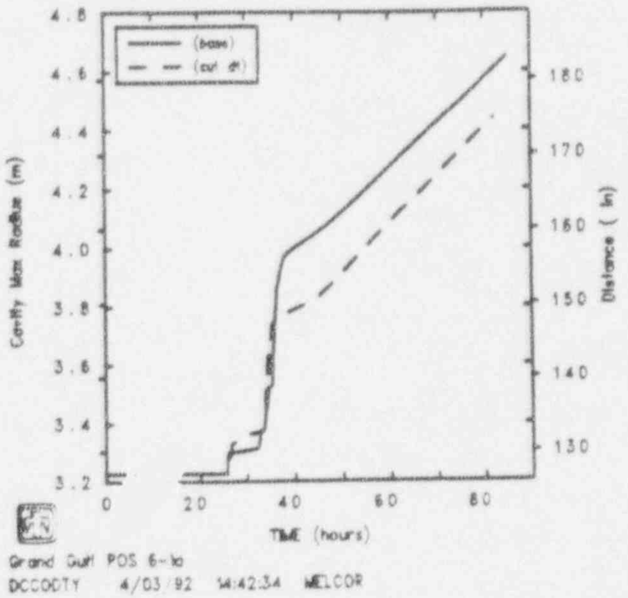
Two calculations were done to address concerns [58, 59] raised about the lack of air oxidation modelling in MELCOR 1.8.1, and the associated lack of extensive release of ruthenium demonstrated to occur when irradiated reactor fuel is heated in air. In both, the effect of oxidation with free oxygen in addition to the oxygen in steam was included in the code; in one calculation a constant release rate coefficient was used for Class 6 (Ru), while the other used a variable coefficient dependent on the partial pressure of oxygen in the core. These air-oxidation sensitivity studies were run with the containment personnel locks and equipment hatch open and venting directly to the environment (*i.e.*, with no auxiliary building modelled).

The POS 6 calculations done all indicate that the lack of an air-oxidation model in MELCOR, and the associated lack of extensive release of ruthenium, is not an issue because no oxygen is predicted to be drawn into the core until late in the transient, after the core material has fallen into the cavity; this is visible in both the oxygen mole fractions in the core and the oxygen mass flow rates in the core inlet and outlet junctions, shown for the reference calculation in Figures 5.4.3.1 and 5.4.3.2, respectively.

To investigate the impact of air oxidation and enhanced ruthenium release, we had to artificially introduce air directly into the core control volume. A total of 28,608 kg of O_2 (the amount that would be required to oxidize the clad in the core), at a uniform rate starting when the core liquid level drops below the top of the active fuel until a lower head penetration first fails (*i.e.*, from 13.04 hr to 18.76 hr). The free oxygen sourced into the core control volume during the core heatup period in these sensitivity study analyses is visible in both the oxygen mole fractions in the core and the oxygen mass flow rates in the core inlet and outlet junctions, in Figures 5.4.3.3 and 5.4.3.4, respectively.

Table 5.4.3.1 compares the timings of various key events predicted in the two air-oxidation calculations with a corresponding basecase analysis. There is no difference in timing on any events before the extra oxygen is first sourced in. The gap release and the failure of the lower head penetrations in the various rings are predicted to occur somewhat earlier, because of the slightly accelerated core heatup due to more clad oxidation.

There are no major differences observable in primary and containment systems pressure histories, or core inventory boiloff. Clad temperature histories in the core level just below the active fuel midplane in one of the six core rings are presented in Figure 5.4.3.5, as representative of the overall core response. The two air-oxidation sensitivity study calculations both show more rapid clad heatup due to the increased degree of (exothermic) clad oxidation, resulting in earlier melt, relocation and lower head failure.



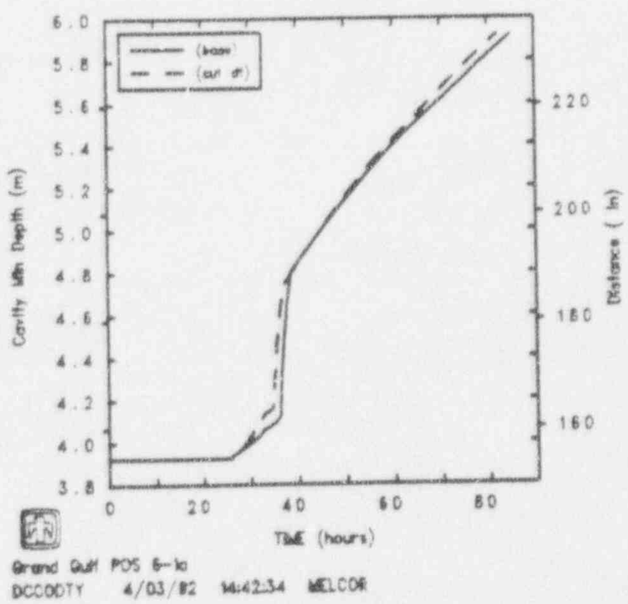


Figure 5.4.2.3. Cavity Maximum Radius and Minimum Depth for Grand Gulf POS 6 - Time Step Sensitivity Study

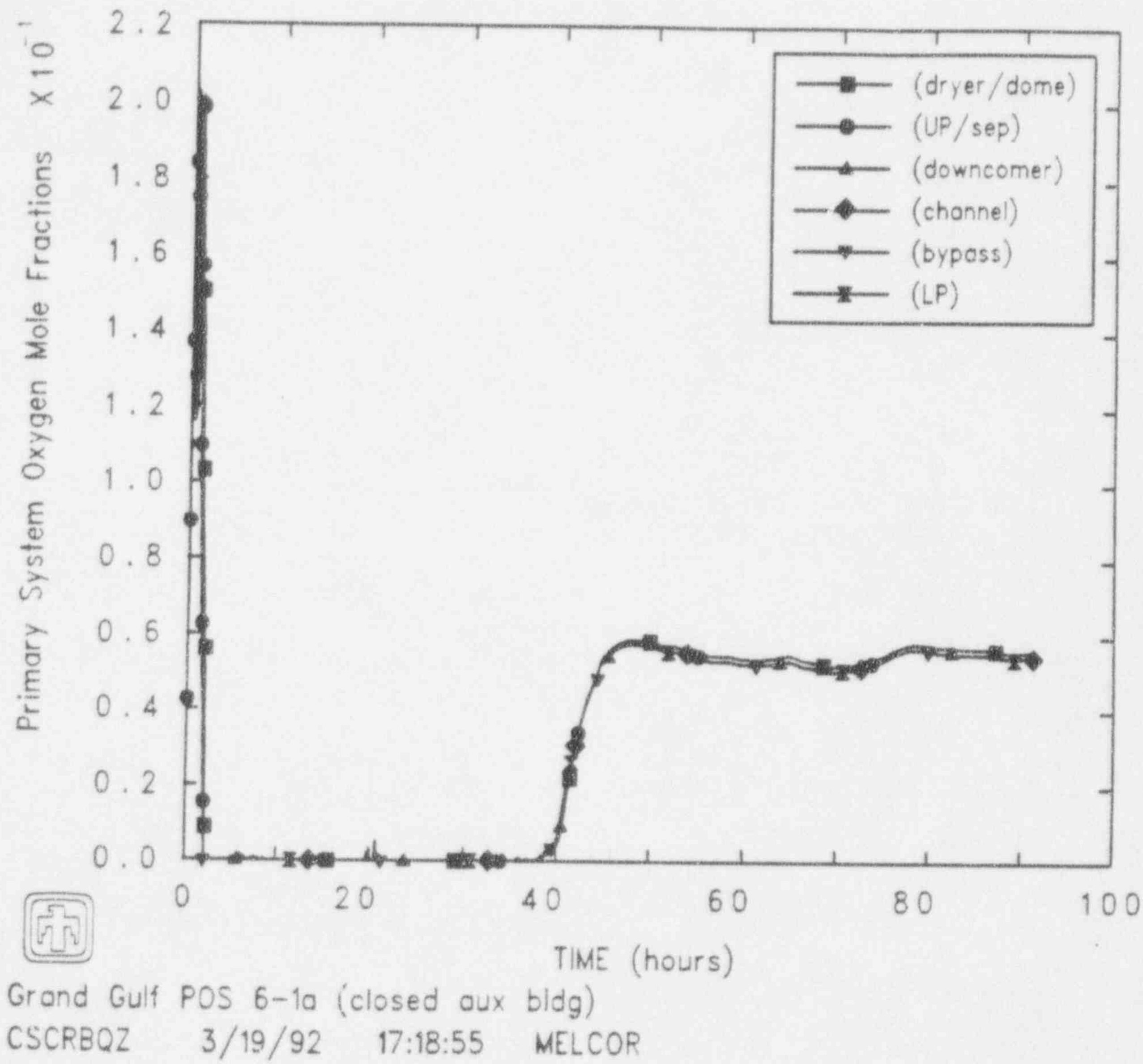


Figure 5.4.3.1. Primary Oxygen Mole Fractions for Grand Gulf POS 6 - Reference Calculation

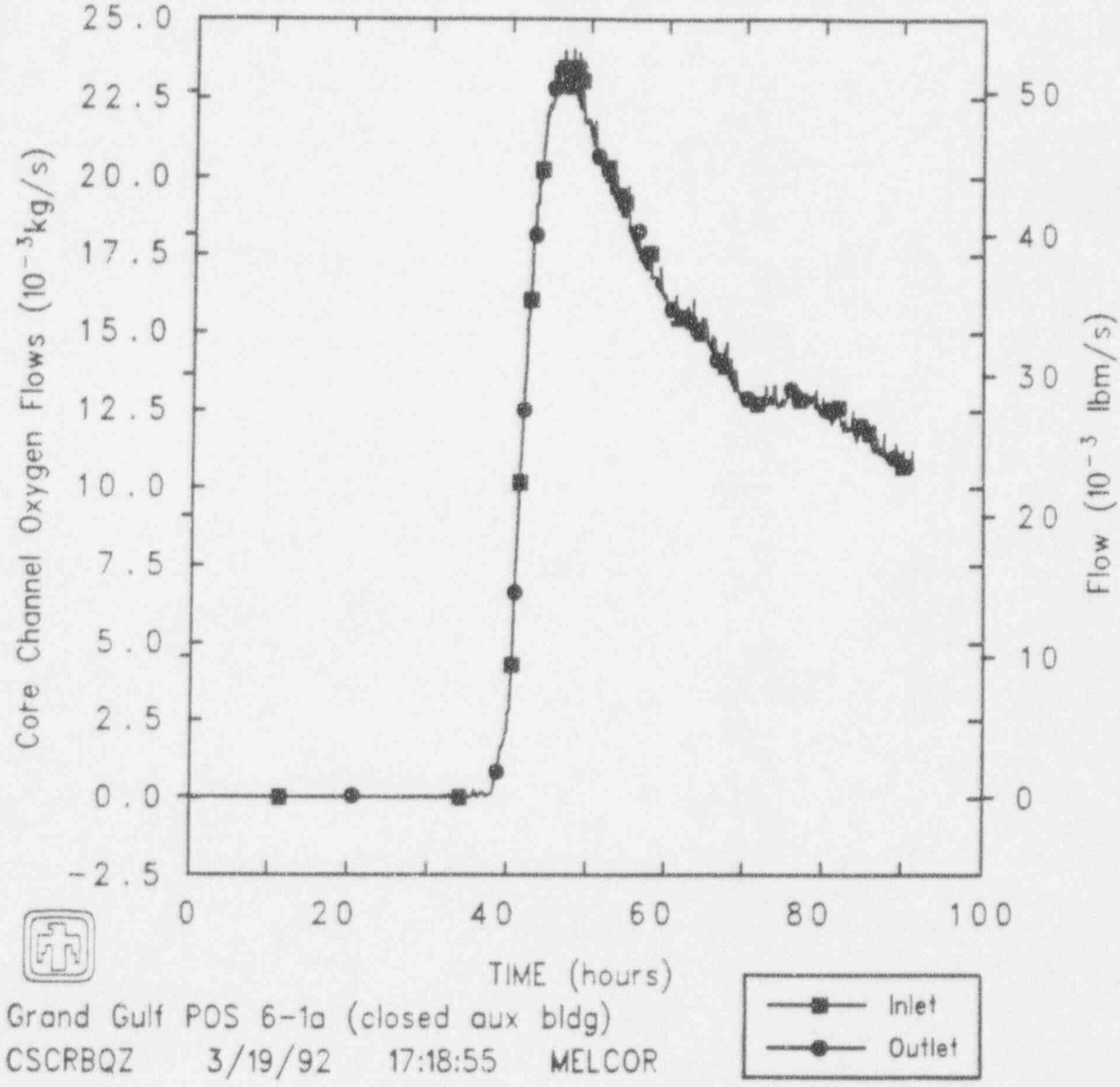


Figure 5.4.3.2. Core Oxygen Inlet and Outlet Mass Flows for Grand Gulf POS 6 - Reference Calculation

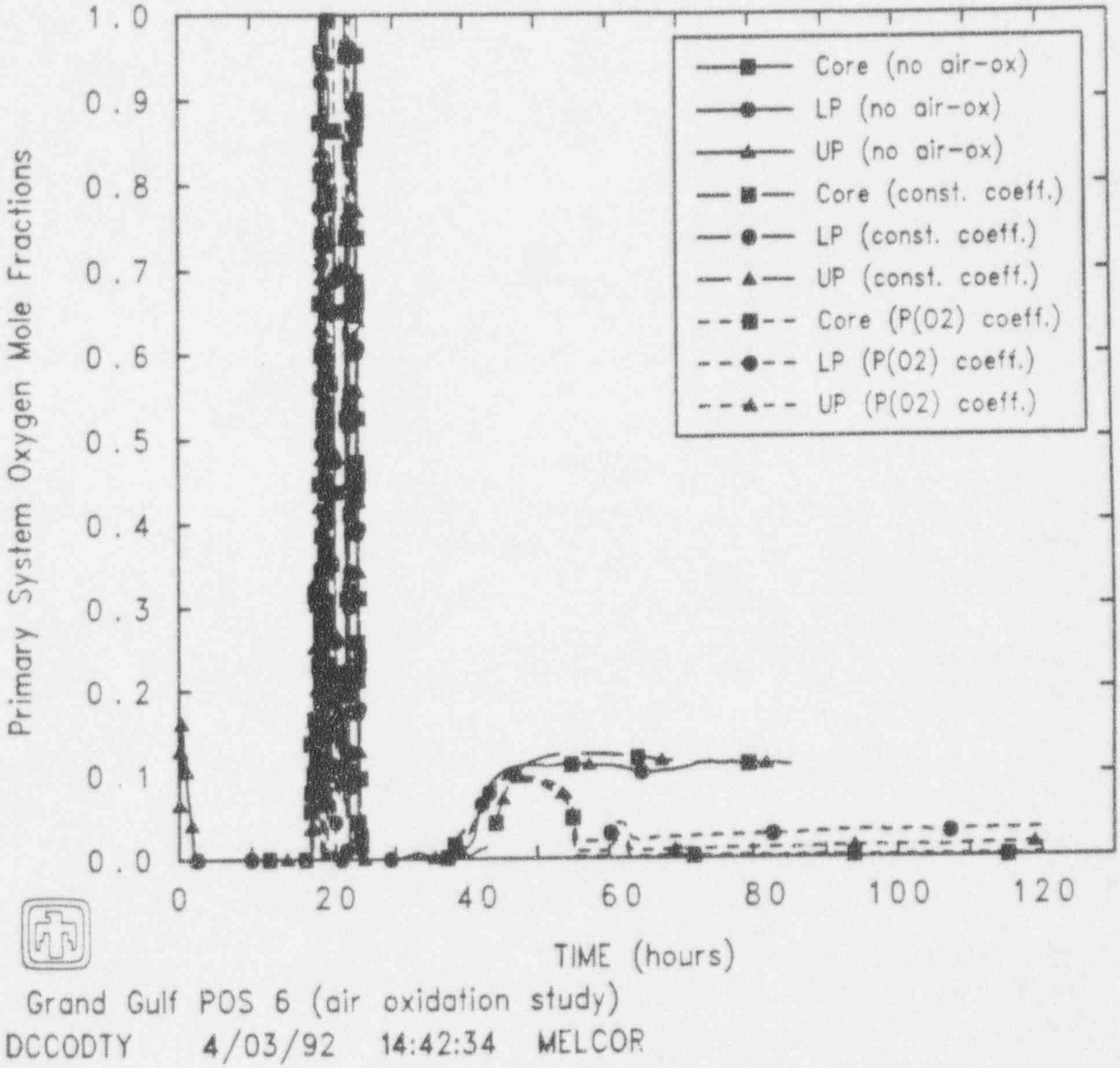


Figure 5.4.3.3. Primary Oxygen Mole Fractions for Grand Gulf POS 6 - Air Oxidation Sensitivity Study

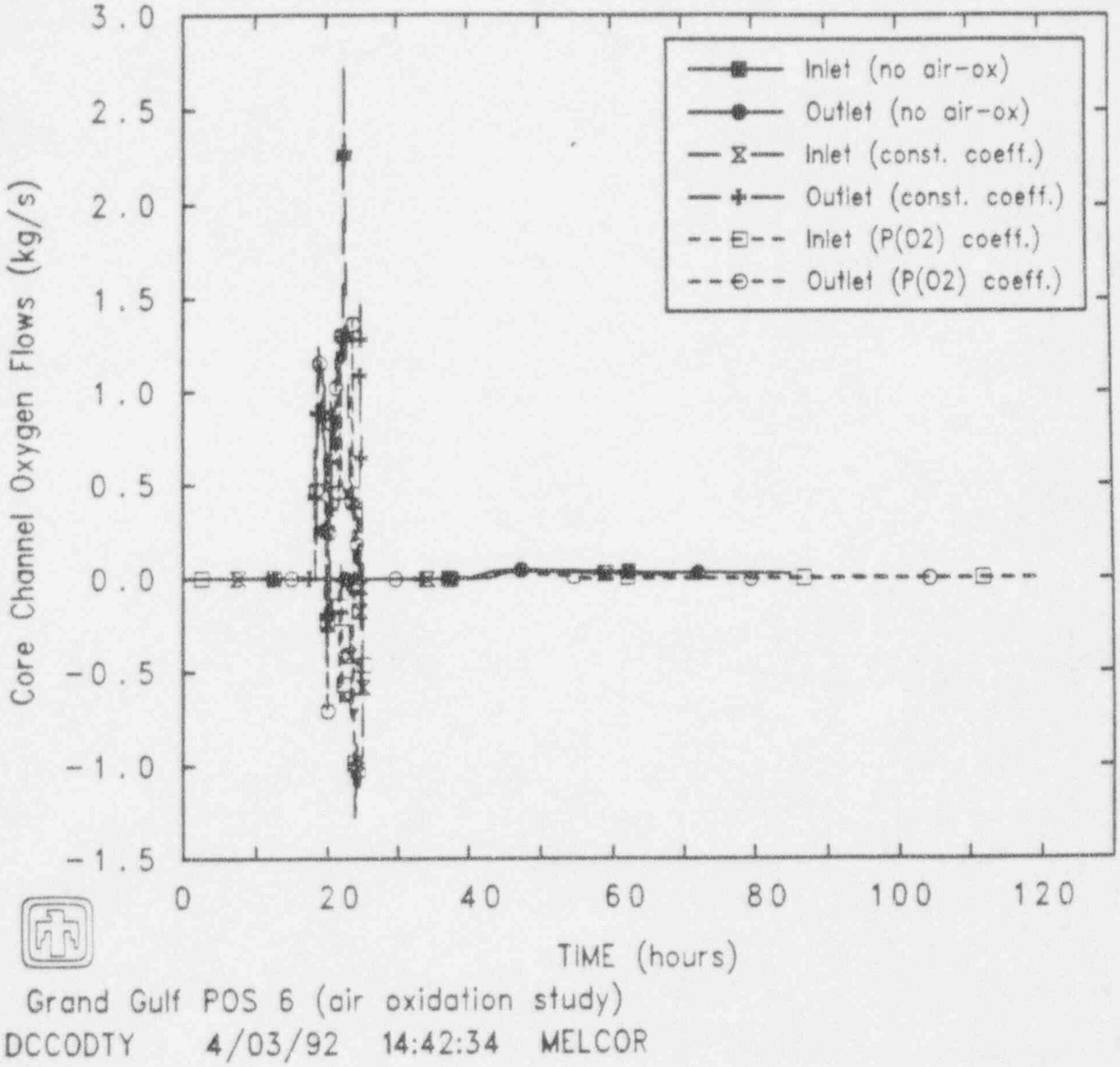


Figure 5.4.3.4. Core Oxygen Inlet and Outlet Mass Flows for Grand Gulf POS 6 – Air Oxidation Sensitivity Study

Table 5.4.3.1. Key Event Times for Grand Gulf POS 6 – Air Oxidation Sensitivity Study

Event	No Air-Ox	Constant Coeff.	$P(O_2)$ Coeff.
Level below TAF	12.68 hr	12.68 hr	12.68 hr
Clad failure/Gap release			
(Ring 1)	18.37 hr	18.34 hr	18.34 hr
(Ring 2)	18.34 hr	18.34 hr	18.34 hr
(Ring 3)	18.38 hr	18.34 hr	18.34 hr
(Ring 4)	18.61 hr	19.28 hr	19.24 hr
(Ring 5)	20.46 hr	20.04 hr	19.96 hr
(Ring 6)	21.80 hr	20.51 hr	20.02 hr
Vessel LH penetration failure			
(Ring 1)	25.45 hr	24.81 hr	25.78 hr
(Ring 2)	25.39 hr	24.81 hr	26.00 hr
(Ring 3)	26.10 hr	25.15 hr	24.61 hr
(Ring 4)	34.49 hr	24.83 hr	23.63 hr
(Ring 5)	30.24 hr	24.85 hr	24.21 hr
(Ring 6)	32.63 hr	–	54.88 hr
Cavity rupture	85.61 hr	68.79 hr	120.44 hr

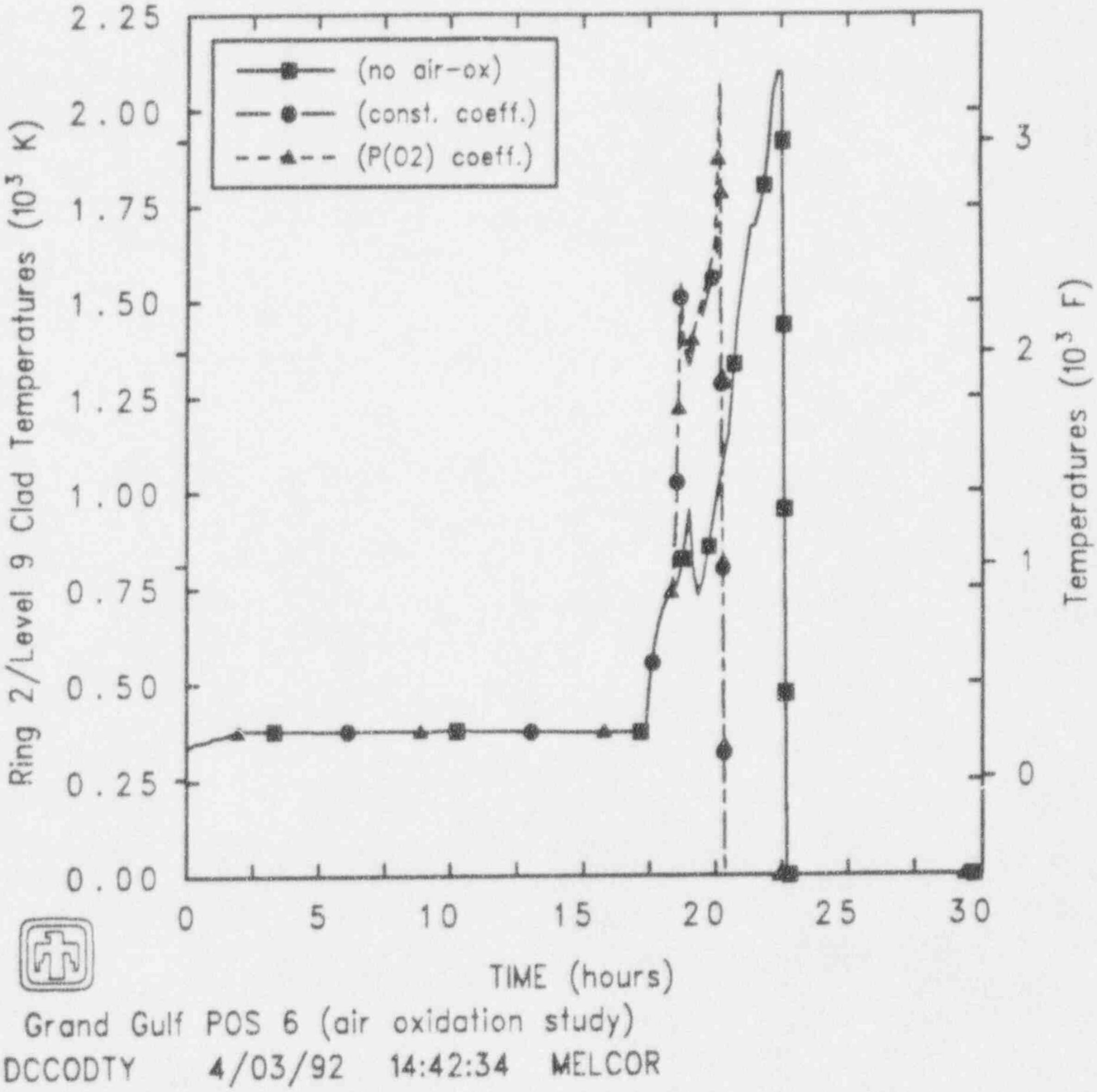


Figure 5.4.3.5. Level 9 Clad Temperatures for Grand Gulf POS 6 - Air Oxidation Sensitivity Study

Table 5.4.3.2. Oxidation Masses for Grand Gulf POS 6 - Air Oxidation Sensitivity Study

Material	Total Masses at End of Transient (kg)		
	No Air-Ox	Constant Coeff.	$P(O_2)$ Coeff.
In COR Package			
Zircaloy	12356	24551	6848
Zirc Oxide	7211	7890	8784
Stainless Steel	35299	35875	33650
Steel Oxide	1809	1688	3658
Steam Consumed	8750	3078	5738
Oxygen Consumed	–	4862	6291
In CAV Package			
Metal Layer	83959	7746	87965
(Light) Oxide Layer	591150	413250	618710
Hydrogen			
Produced in Vessel	1001	344	642
Produced in Cavity	1280	1019	1159
Total Produced	2281	1363	1801

The masses of zircaloy and zirc oxide, stainless steel and steel oxide, steam and oxygen consumed and hydrogen generated by the end of these transient calculations are presented for these air-oxidation sensitivity studies in Table 5.4.3.2. With the free oxygen source, 10-20% more zircaloy and 100% more steel is oxidized in-vessel. Because 30-60% less steam is consumed, 30-60% less hydrogen in generated in-vessel; with 10-20% less hydrogen generated in the cavity, the total amount of hydrogen generated is 20-40% less in the two air oxidation sensitivity studies. (Most of the oxygen sourced into the core control volume therefore escapes out through the upper head and vessel breach, to the containment and then the environment, without being consumed in oxidation processes.)

Figure 5.4.3.6 shows the hydrogen generation rates, both in-vessel and in the cavity. The lower amounts of hydrogen produced in the air oxidation sensitivity studies are seen to be primarily a result of sharp differences during the time period the free oxygen is being added, not gradual divergences throughout the remainder of the transient.

Table 5.4.3.3 compares the radioactive masses of radionuclides released in this set of MELCOR calculations, when a lower head penetration first fails and at the end of

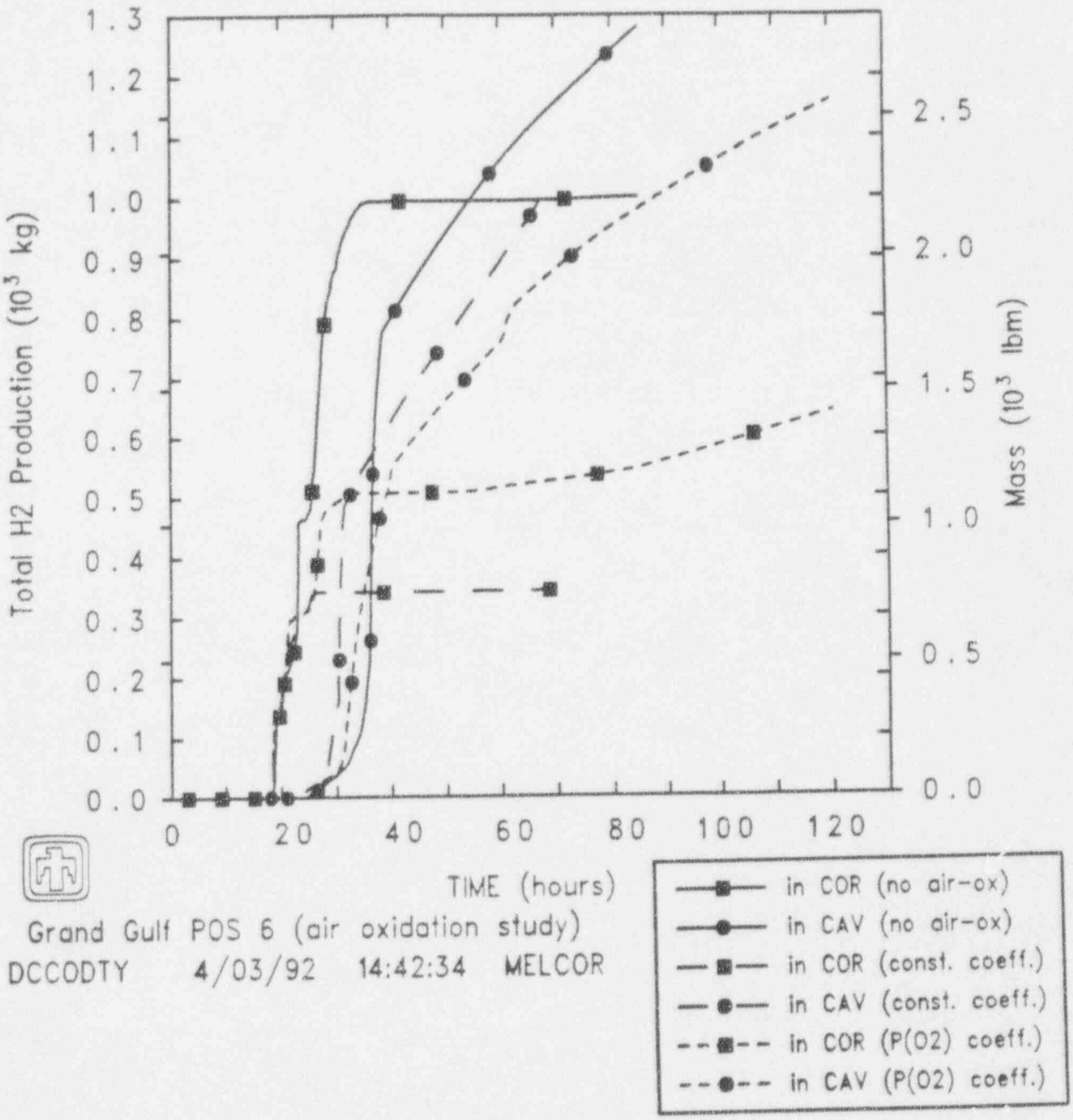


Figure 5.4.3.6. Hydrogen Generation for Grand Gulf POS 6 - Air Oxidation Sensitivity Study

Table 5.4.3.3. Fission Product Radioactive Masses for Grand Gulf POS 6 - Air Oxidation Sensitivity Study

Class	% of Initial Inventory Released					
	Before Vessel Breach			Before Cavity Rupture		
	No Air-Ox	Constant Coeff.	$P(O_2)$ Coeff.	No Air-Ox	Constant Coeff.	$P(O_2)$ Coef
1 (Xe)	81.3	93.2	79.5	100.0	97.7	100.0
2 (Cs)	81.7	93.3	79.9	100.0	97.9	100.0
3 (Ba)	2.38	8.65	22.1	42.0	42.7	47.1
4 (I)	81.0	93.1	79.2	95.5	93.4	89.0
5 (Te)	72.5	92.6	76.3	95.2	95.8	92.8
6 (Ru)	0.00002	99.9	100.0	0.0070	100.0	100.0
7 (Mo)	0.0	0.0	0.0	1.61	3.23	1.405
8 (Ce)	0.000003	0.0074	0.1186	0.0037	0.0082	0.1276
9 (La)	0.0	0.0	0.0	0.2170	0.666	0.3588
10 (U)	0.00156	0.520	4.59	1.62	0.522	5.10
11 (Cd)	0.0	0.0	0.0	0.0385	0.0808	0.0763
12 (Sn)	2.866	19.3	28.0	22.5	20.7	34.8

the calculation (*i.e.*, when the cavity is predicted to rupture), normalized to the initial masses of each class (given in Table 3.1). The primary difference is the (as expected) ~100% release of ruthenium in-vessel in the two air-oxidation sensitivity study analyses, both using a constant release rate coefficient and using a variable coefficient dependent on the partial pressure of oxygen in the core. But there are other differences. More of the more refractory classes (Ba, Ce, U and Sn) are released prior to vessel breach in the two air-oxidation sensitivity study calculations; unexpectedly, while more of the more volatile classes (Xe, Cs, I and Te) are released using a constant Ru release rate coefficient, slightly less are released using a variable Ru release coefficient dependent on the partial pressure of oxygen in the core than predicted with no air oxidation at all.

The comparison of released by the time of cavity rupture is more confused. The three classes with identically-zero in-vessel releases all show the greatest release fraction for the air-oxidation sensitivity study using a constant release coefficient for Class 6; the other more refractory classes (Ba, Ce, U and Sn) show higher release in the calculation with a variable Ru release coefficient dependent on the partial pressure of oxygen; the volatiles (Xe, Cs, I and Te) all show 90-100% releases with no clear pattern of variation.

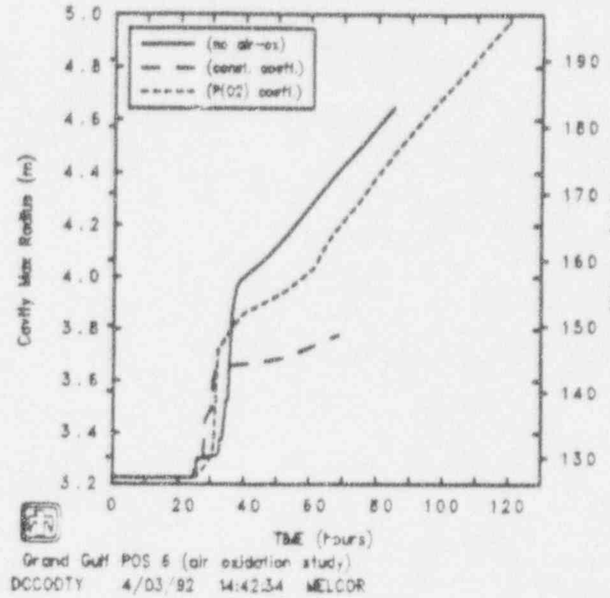
The total radioactive masses released from the fuel and debris for each class, and the amount released to the environment by the time of cavity rupture (given in terms of the initial inventory) are summarized in Table 5.4.3.4. Almost all of the ruthenium is

Table 5.4.3.4. Total Fission Product Radioactive Mass Released from Fuel for Grand Gulf POS 6 - Air Oxidation Sensitivity Study

	% of Initial Inventory					
Class	Released Before Cav-Rupture			Released to Environment		
	No Air-Ox	Constant Coeff.	P(O$_2$) Coeff.	No Air-Ox	Constant Coeff.	P(O$_2$) Coe
1 (Xe)	100.0	97.7	100.0	100.0	97.5	100.0
2 (Cs)	100.0	97.9	100.0	67.6	62.6	60.7
3 (Ba)	42.0	42.7	47.1	21.3	21.8	16.9
4 (I)	95.5	93.4	89.0	95.5	93.4	88.9
5 (Te)	95.2	95.8	92.8	62.3	60.6	47.0
6 (Ru)	0.0070	100.0	100.0	0.0045	62.2	54.6
7 (Mo)	1.61	3.23	1.405	0.642	1.26	0.487
8 (Ce)	0.0037	0.0082	0.1276	0.00215	0.00467	0.0259
9 (La)	0.2170	0.666	0.3588	0.1024	0.365	0.168
10 (U)	1.62	0.522	5.10	0.212	0.305	1.20
11 (Cd)	0.0385	0.0808	0.0763	0.0189	0.0388	0.040
12 (Sn)	22.5	20.7	34.8	15.5	12.5	14.3

released from the fuel in these two air-oxidation sensitivity study analyses, and over half of that is released to the environment (in the absence of any additional retention in the auxiliary building, not included in these calculations).

Both air-oxidation calculations ended on cavity rupture, at very different times. Figure 5.4.3.7 shows that the calculations with no air-oxidation and with air oxidation and a constant Ru release coefficient predict that the cavity concrete will first be ablated in depth, with the calculation using a variable coefficient dependent on the partial pressure of oxygen predicts that the cavity concrete will first be ablated radially, but with less than 3.5 cm depth remaining axially at that time.



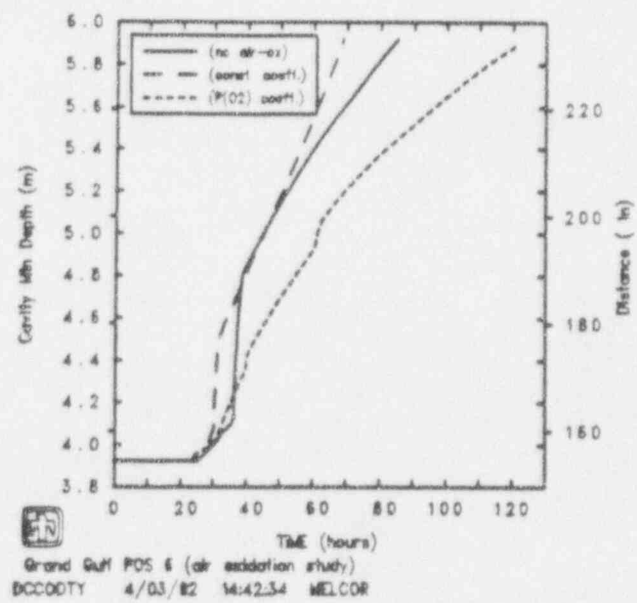


Figure 5.4.3.7. Cavity Maximum Radius and Minimum Depth for Grand Gulf POS 6 - Air Oxidation Sensitivity Study

The 2/3 of the core under water was cooled by heat transfer to the pool within the fuel assemblies. This pool remained subcooled due to heat transfer through the channel boxes to the bypass pool. Thus, the only steam generated within the fuel assemblies was from pool surface evaporation. The core water temperatures are shown in Figure 8 along with the atmospheric saturation temperature and the LPCI injection temperature. The channel pools remained subcooled by about 17, 16, 17, 21, 33, and 48 K for core rings 1 through 6, respectively. The injected water was heated by about 5.5 K before flowing out of the vessel.

A code error affecting the pool temperatures as seen in Figure 8 became apparent at the onset of hydrogen generation at about 9000 seconds. The cooling of these pools at this time was unrealistic and the cause of the problem is unknown at this time. However, since the objective of the calculation was to determine whether or not fuel damage could occur and this error cooled the convective fluid and fuel damage was predicted anyway, this problem should not affect any of the study conclusions.

The core heating is shown in the next five figures. Figure 9 shows the cladding temperatures of ring 2 which had the highest power density and therefore the highest temperatures of all the rings. The cladding temperature for the cells at axial level 20 for each ring are shown in Figure 10. The cell component temperatures for cells 220, 218, and 214 are shown in Figures 11, 12, and 13, respectively.

The upper most core cells which did not have any fuel and therefore did not have any decay power heated only by convection heat transfer from the rising hot gases within the channels. The localized channel fluid temperatures (DTDZ model) closely followed the cladding temperatures as the gases rose within the core. The cladding temperature of cell 222 reached as high as 496 K.

The highest cladding temperature in the calculation was for cell 220 near the top of the core. Its final temperature was 1217 K. The cladding of this cell was predicted to reach 1173 K and fail at 9570 seconds (2.66 hours) which would have released the first cladding gap fission products at this time. There was sharp increase in this temperature at about 9000 seconds due to the energy released from cladding oxidation.

Cell 219, 220, and 221 in ring 2 continue to increase throughout the calculation and would continue to increase further, perhaps melting, if the calculation was continued. These cells were above the core water levels in both the core channels and the bypass. The component temperatures for cell 220 in Figure 11 show that all components heat together with even the control rod approaching structural failure (at roughly 1273 K).

Cells 215 through 218 were uncovered inside the fuel assemblies but the outside of the canisters were cooled by the cold bypass water. The component temperatures for cell 218 in Figure 12 illustrate the heat transfer associated with these cells. The exposed fuel rods temperatures peaked at about 722 and 717 K for the fuel and cladding. At these code [6]. The first core cell cladding to reach this temperature was cell 211 (upper second ring) at 18.9 hours. Once oxidation began, the heating and damage to the core fuel progressed rapidly.

In conclusion, the more complex MELCOR calculation verified the hand calculation results for core uncovery and in addition provided an estimate of when initial fuel damage occurred. While the collapsed water reached the top of the active fuel at about 13.1 hours, the fuel did not begin to heat until about 16.5 hours with the onset of oxidation occurring at 18.8 hours.

Recirculation Pipe LOCA with One LPCI Pump

A low decay power shutdown LOCA was run involving the double-ended rupture of a pump suction pipe in a recirculation loop with ECCS provided by only one LPCI pump. The LPCI pumps water from the containment suppression pool into the core bypass region. The broken recirculation pipe allowed all the reactor vessel water above the jet pump throat to drain from the vessel leaving the upper 1/3 core exposed, without significant cooling, and subject to damage.

This calculation was run with 6 control volumes representing the core channels, i.e., one for each core ring. The upper head was removed, the recirculation loop flow paths simulating a LOCA were active, and the six ring fine node core model was used. The calculation was initialized at 4 days after the reactor was tripped with the water level initialized at the normal water level (569.7 inch), the vessel water temperatures at 333.15 K (140 F), and the LPCI water temperature at 305.37 K (90 F). The LPCI flow rate to the core bypass was 7620 gpm.

The reactor vessel water quickly (less than 3 minutes) drained from the vessel until the downcomer level dropped below the jet pump throats. The water levels which are shown in Figure 7, then remained relatively stable for the remainder of the calculation. The average channel and the bypass levels remained about .1 m and .7 m, respectively, above the top of the jet pump throats. The core channel water levels did vary slightly from channel to channel due to their variation in water density but the maximum difference was not more than a few centimeters. After the initial transient was complete, none of the LPCI water over flowed the core from the bypass into the core channel. After the initial phase of the transient, the downcomer water level remained below the jet pumps and so had no effect on the core water levels.

Basically, the LPCI water entered the core bypass, flowed downward to the lower plenum, upwards through the jet pump diffusers into the downcomer and then out of the vessel. Only a small amount of water entered the core channels to replace water lost to steaming within the channels. The steaming rates were quite small (less than .04 kg/sec for the total core) and were due to pool surface evaporation which was enhanced by radiative heat transfer from the exposed core.

subcooled pool temperatures, the vessel water levels, and the second ring cladding temperatures during the fuel heating.

The first portion of the calculation involved heating the subcooled pools until boiling occurred. Natural circulation, with the heated core water rising and the colder downcomer water falling, tended to equilibrate the water temperatures. The first boiling occurred at 2.0 hours in the dome volume due to its lower pressure and saturation temperature. Before boiling occurred, 340 kg of water were evaporated from the pool surface. This initial boiling was at a relatively slow rate until the upper plenum volume saturated at 2.1 hours and then the boiling rate increased to the rate sustained throughout most of the boiloff.

The time for boiling to occur calculated with the level I analysis [5] was 1.8 hours. There were two significant differences between the two calculations. First, the level I calculation used a decay power that was about 22% greater than that used in the MELCOR calculation (the two decay power correlations came from different sources). Second, the level I calculation assumed that the initial water level was at the flange, whereas, in the MELCOR calculation, it was set to just below the steam lines. Thus, the level I calculation was initialized with about 13% more water than was done in the MELCOR calculation. When 1.8 hours is multiplied by 1.22 and divided by 1.13, the result is 1.94 hours which is in excellent agreement with the MELCOR result.

The first voiding within the core occurred at 10.5 hours. The initial voiding was small and unstable as steam was formed and then replaced by water from above. The collapsed water level, as measured in the downcomer volume, reached the top of the core and the top of the active fuel at 12.6 and 13.1 hours, respectively.

The level I calculation predicted that the time to boil the water to the top of the fuel was 13.8 hours. The major difference between the level I hand boiloff calculation and the MELCOR calculation was that the level I calculation boiled away 26% more water to reach the top of active fuel than did MELCOR. This was primarily due to the water level being initialized at the flange.

Convective cooling of the core continued after exposure. The downcomer water level reached the jet pump throats at 16 hours, after which the core water levels dropped faster because the water flow from the downcomer through the jet pumps ceased. A water pool continued to exist in the upper plenum until 16.1 hours, held in place by steam flows exiting the core.

The first fuel heating began in cell 212 (top fuel in second ring) at about 16.5 hours as the convective cooling decreased. The cladding oxidation began at about 18.8 hours as indicated when 0.0001 kg of hydrogen had been produced. About 2 minutes later, the hydrogen production had reached 1 kg. Cladding was modeled to fail and release radioactive fission products from the fuel when it reached 1173 K. The cladding failure criteria of 1173 K (900 C) was adapted from the CORSOR nodalization. The channel inlet included the lower tie plate (which has an inlet orifice) and half of the seven grid spacers. The channel exits included the upper tie plate and the other half of the spacers. The channel loss coefficients are dominated by the lower tie plate orifice. The resulting coefficients are listed in Table 7 and despite their uncertainty, they should be adequate for these calculations.

Flow paths were included to simulate the recirculation pump suction lines, and the recirculation inlet nozzles during a double-ended break LOCA in a recirculation loop. The suction line flow path modeled two 24 inch OD lines of 10 m length which were always fully open. The inlet nozzles modeled twelve 10 inch OD lines and the header and pumps and were initially open but closed when the water level dropped below the nozzle entrance. A flow path with an area equal to reactor vessel cross sectional area was included to simulate the vessel with its upper head removed.

<u>Emergency Core Cooling Systems</u> During the low pressure LOCA calculations, ECCS was supplied to the core bypass control volume simulating LPCI. The FSAR (Table 5.4-2d) gives the flow rate per pump at 7620 gpm. These calculations involved either 1 or 2 pumps and the 2 pump flow was just double the 1 pump flow. The temperature of the injected water was 90 F which was the estimated suppression pool temperature. ECCS was not applicable to the boiloff calculation.

<u>Vessel Heat Structures</u> The reactor vessel heat structures in the LaSalle model were inserted unchanged into the Grand Gulf model. These heat structures were relatively unimportant to the objective of determining whether or not fuel damage would occur during these low decay power calculations.

<u>Containment</u> The containment was not modeled for these calculations. A large control volume was included to provide a dump for steam and water flows leaving the reactor vessel and to maintain a constant system pressure initialized at one atmosphere.

CALCULATION DESCRIPTIONS AND RESULTS

Boiloff Calculation

A simple boiloff calculation was run for Grand Gulf. This particular calculation used just one volume to model the core channels and the six ring course axial node core model. The upper head was removed, all piping remained intact, but all sources of cooling water to the core failed. The calculation was initialized at 4 days after the reactor was tripped with the water level just below the steam lines at an elevation of 635 inches and the vessel water temperatures were all initialized at 333.15 K (140 F). The initial water mass was 444,910 kg.

The boil-off results are illustrated by Table 8 which lists the timing of events during the calculation and in Figures 4 through 6 showing the resulted in the draining of the downcomer so that a loop model was not needed.

Vessel Flow Paths The primary system was modeled with eight internal flow paths and three external flow paths. The entrance and exit elevations, the forward and reverse loss coefficients, and the flow area for each path are listed in Table 5. The internal flow paths include the core channel and bypass inlets and exits, steam separators and separator returns, dryer drains, and the jet pump diffusers. The external flow paths include a path to simulate the vessel with its upper head removed, the recirculation pump suction lines, and the recirculation inlet nozzles.

Again the LaSalle input was used as a framework for the Grand Gulf models. The flow paths which are critical to these low power calculations were the jet pump diffusers and the core plate. Significant errors in the input of the other flow paths should not significantly impact the results of these calculations, therefore the LaSalle input was adopted for these paths.

Input data was developed specifically for the Grand Gulf jet pumps. The jet pump diffusers consist of three sections; the throat section, the diffuser section, and the extension section. Diameters and elevations were obtained or estimated from the FSAR and drawings. The jet pump flow path data which are listed in Table 6 were developed for the throat flow area. Most of this data are more than adequate for these calculations. The one parameter which has an uncertainty potentially important to the conclusions from these calculations is the throat exit loss coefficient judged at 1. This coefficient dominates the total reverse loss coefficient for the water flow through the jet pumps during the LOCA calculations which was in the reverse direction. This uncertainty is discussed further in the uncertainty section.

The most sensitivity parameter for the LOCA calculations was the reverse core plate loss coefficient for water flow from the core bypass to the lower plenum. This determined the water head in the bypass and whether or not the water overflowed the bypass into the fuel assemblies. Due to the lack of applicable Grand Gulf data, the core loss coefficients developed for LaSalle were used. The uncertainty of this parameter on the final conclusions was investigated and is discussed in the uncertainty section.

The LaSalle core loss coefficients were developed from the RETRAN input and since this input was developed by engineers with access to GE proprietary information, the LaSalle coefficients were the best available for these calculations. The RETRAN coefficients used in developing the LaSalle coefficients included the coefficients for forward and reverse flow through the fuel assemblies. The bypass loss coefficients were then calculated to establish the ratio of channel to bypass flow at 10 [4] for steady state operation (reverse flow assumed the same ratio as the forward flow). The MELCOR core flows were all based on the channel or bypass flow areas in an unrestricted portion of the core. The RETRAN coefficients were then modified by the ratio of squared areas (MELCOR/RETRAN) to get coefficiunts applicable to MELCOR and since the RETRAN nodalization was much more detailed, coefficients were summed for the more course MELCOR distribution outside of the fueled region (i.e., the handles, the lower tie plate, the fuel support pieces, control rod velocity limiters, etc.) were estimated from the available data and schematic drawings.

Other Input Other core model input were computed in a similar manner as were the masses. These include the component surface areas, the flow areas, cross sectional areas, and the equivalent diameters. Inputs for the vessel lower head and penetrations still reflect the LaSalle data. However, because the Grand Gulf calculations do not include core meltdown, the results will not be particularly sensitive to this input and, therefore, it is felt that the LaSalle numbers are adequate.

Hydrodynamic Models

Vessel Control Volumes The reactor vessel control volumes were adapted from the LaSalle input model with modifications. The available Grand Gulf data included a few basic dimensions and volumes from the FSAR and plant drawings. FSAR Figure 5.1-2 lists six primary system volumes which total to 21745 ft^3. The LaSalle model, which was developed from RETRAN input, totals 21444 ft^3 (with the volume of the steam lines deleted). These two totals differ by 1.4% and it is likely that the primary system designs are very similar. There are however differences when comparing the LaSalle model and the Grand Gulf data. The Grand Gulf core shroud has a larger diameter than LaSalle to accommodate the larger core. Grand Gulf has 24 jet pumps compared to 20 for LaSalle and have different jet pump designs.

Since the LaSalle input was derived from a more detailed RETRAN input model, the LaSalle model is a good framework for developing an adequate Grand Gulf primary system model for these low power calculations. The LaSalle input was adapted and modified to include the larger Grand Gulf core shroud and the Grand Gulf jet pump designs. A few other numbers such as elevations were changed to coincide with data from the plant drawings.

The volumes and elevations for the Grand Gulf models are listed in Table 4. The basic model consists of 6 volumes with a total volume of 22182 ft^3. This volume includes the steam line volume of 1454 ft^3 and excludes the recirculation loop volume of 1020 ft^3 and agrees closely with the Grand Gulf FSAR data. However, the MELCOR volume nodalization is not the same as the FSAR nodalization. The core volume nodalization (channels and bypass) go from the core plate to the top of the fuel assembly canisters. The volume within the jet pumps is contained in the lower plenum control volume.

A more sophisticated core volume model which has 6 control volumes representing the fuel assembly channels and 1 volume for the bypass region was developed for the LOCA calculations. The core channels were subdivided according to the core model ring volume fractions.

The recirculation loop piping was not modeled for these calculations. In the boiloff calculation, it was assumed that circulation within the recirculation piping would not significantly effect the boiloff results. In the LOCA calculations, the recirculation loop double-ended rupture

The core decay power distribution was developed from FSAR EOC data. The radial power factors are listed in FSAR Figure 4.3-21 for each fuel assembly. These power factors were used to determine the power factors for the six ring core model illustrated in Figure 1. Since the power distribution dips at the core center, the inner portion of the core was subdivided to focus on the region with the highest power density (second ring). It is important to remember that some fuel assemblies have higher power factors than their associated ring averages (the highest is 1.232). The number of assemblies in each ring, the volume fractions, the outer radii, the power fractions, and power factors are listed in Table 1.

The axial power factor distribution shown in FSAR Figure 4.3-22 was used to develop the axial power factors for MELCOR. The resulting axial power factors are listed in Tables 2a and 2b for a course and fine axial nodalization and these power factors were adjusted to include the non-fueled portions of the core. For the course axial nodalization, the entire active fuel region of the core was subdivided into 6 cells of equal height (25 inches) but in the fine nodalization, the upper half of the active fuel region was further subdivided into cells with a height of 6.25 inches. The core water level for the LOCA accidents involving a recirculation line break remained above the jet pump throats which is about 2/3 the way up the active fuel.

The core nodalization is shown in Figures 2 and 3 for the course and fine nodalization schemes, respectively. In the cell numbering system, each cell has a 3 digit identifying number. The first digit is the ring number, with the rings numbered from the core center outward, and the second and third digits indicate the axial level, beginning with level 1 at the bottom of the lower plenum. Level 5 represents the core plate, and levels 13 and 22 represent the top cells for the course and fine nodalization schemes, respectively. For example, cell 212 (course nodalization) is top cell containing fuel located in the second ring. The top cells did not contain fuel.

Component Masses The 800 assembly Grand Gulf core contains a total of 179,760 lbm of Zr. There is 98.7 lbm in each assembly canister and 126 lbm in the fuel rods. In addition, the FSAR lists the total fuel mass as 458 lbm/assembly for a total UO_2 mass of 366,400 lbm. The total fuel assembly and control masses are given as 699 and 218 lbm, respectively. There are 193 control rods in the core.

The Grand Gulf fuel rods appear to be identical to the LaSalle rods and both have an 8 by 8 matrix. Grand Gulf, however, has a thicker canister than LaSalle, in addition to 36 more fuel assemblies and 8 more control rods than LaSalle.

The fuel assembly and control rod masses are listed in Table 3. LaSalle data was used for the top guide, core plate, fuel supports, control rod tubes and housings masses. These masses were subdivided into radial and axial cells corresponding to the cells for the power distribution. The subdivided masses are reasonably accurate for the active fueled core region and the correct total masses were maintained. The mass

MELCOR has a structured, modular architecture that accesses only those modules called 'packages' required for a particular calculation and that facilitates the incorporation of additional or alternative phenomenological models. MELCOR has an input preprocessor called MELGEN which generates the initial restart and a plot processor called MELPLT. Separate input is required for each; MELGEN, MELCOR, and MELPLT.

Five major MELCOR packages were employed to model the thermal/hydraulic behavior for these calculations. The Control Volume Hydrodynamics Package (CVH) models the behavior of water and non-condensible gases in a control volume. The Flow Path Package (FP) models the movement of water and non-condensible gases between the control volumes. The Control Volume Thermodynamics package (CVT) handles thermodynamic calculations for the control volumes and together with the CVH and FP packages advance the thermal/hydraulic state in the control volumes from one time level to the next. The Heat Structures package (HS) calculates one-dimensional heat conduction within an intact solid structure and energy transfer across its boundary surfaces into control volumes. The core package (COR) treats the processes associated with chemical and mechanical degradation of the core and associated structures brought about as the core heats and degrades.

MELCOR MODEL DESCRIPTION

The following describes the MELCOR model development for the Grand Gulf low power/shutdown study. Previous Grand Gulf calculations [2] used a modified LaSalle core and reactor cooling system. These models, particularly the core model, have been improved to ensure that these calculations represent Grand Gulf. These models still contain LaSalle specific data [3] but the parameters of importance have been converted to or verified as Grand Gulf data to the extent possible given the limited available plant data. For instance, the core model has the proper fuel assembly and control rod masses, the primary system volumes are in reasonable agreement with the volumes stated in the FSAR [4] but certain flow loss coefficients which were critical to determining whether or not the bypass water overflowed the top of the core into the fuel assemblies were not known specifically for Grand Gulf.

Core Input Model

Core input was developed specifically for the Grand Gulf 800 assembly core as previous Grand Gulf calculations used a modified version of the LaSalle input.

Decay Power The time dependent decay power is calculated using the normalized time dependent power distribution developed for the LaSalle plant (this is the same power curve used in the previous Grand Gulf calculations). The operating power level was 3833 Mw when the reactor was tripped and these low decay power calculations begin 4 days after the reactor was tripped. The initial power level at 4 days is .309% of operating power (11.86 Mw).

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date May 17, 1991

to T. D. Brown, Org.6413

Clint Shaffer

from C. J. Shaffer, Org. 6418

subject Grand Gulf Low Power/Shutdown MELCOR Calculations

INTRODUCTION

Severe accident calculations with MELCOR were run to support the Grand Gulf Low Power/Shutdown PRA. The Grand Gulf Nuclear Station located in southwestern Mississippi is a BWR-6 boiling water reactor with an 800 fuel assembly core contained inside a Mark III containment. The calculations all assume that the reactor vessel upper head was removed when the accidents were initiated four days after the reactor tripped (PRA plant state 6).

Three calculations were done. First, a low decay power boiloff without any ECCS and all piping intact, then two LOCA accidents with a recirculation loop double-ended pipe rupture. The first LOCA calculation assumed only one Low Pressure Coolant Injection (LPCI) pump was operated and the other LOCA calculation assumed two pumps were available. The LPCI pumped water from the containment suppression pool into the core bypass region. The broken recirculation pipe allowed all the reactor vessel water above the jet pump throats to drain from the vessel which left the reactor core about 2/3 covered with water and allowed the upper 1/3 of the core to heat and possibly become damaged. The LPCI may or may not over fill the core bypass allowing water to flow into the core channels.

BRIEF MELCOR DESCRIPTION

MELCOR [1] is a fully integrated, relatively fast-running code that was developed at SNL to model the progression of severe accidents in light water reactor nuclear power plants. Characteristics of severe accident progression that can be treated with MELCOR include the thermal-hydraulic response in the reactor coolant system, reactor cavity, and containment; core heatup and degradation; core-concrete attack; combustible gas generation, transport, and combustion; plant-structure thermal response; radionuclide release and transport; and the impact of engineered safety features on thermal-hydraulic and radionuclide behavior. MELCOR has been designed to facilitate sensitivity and uncertainty analyses through the use of sensitivity coefficients. Many parameters in the correlations are coded as sensitivity coefficients changeable through user input.

A Additional Level 1 Supporting Calculations

Copy of Memo to T. D. Brown, SNL,
from C. J. Shaffer, SEA*

*Science and Engineering Associates, Inc.

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to the vessel during the accident, and suppression pool makeup is not dumped into the suppression pool. The MELCOR POS 6 calculations done included a number of variations on the exact plant configuration assumed. In addition, a few sensitivity studies were done on various code options and/or parameters.

Table 6.1. MELCOR Level 2 Support Calculations - Sequences and Relative Contribution of Plant Damage States to Core Damage Frequency

Plant Damage State	Time After Shutdown	Fraction Contributed	Sequence Description
PDS 3-1	40 day	0.338	LBLOCA with flooded containment
PDS 2-2	24 hr	0.242	SBO w/o firewater, break in SDC
PDS 2-1	24 hr	0.170	LBLOCA with flooded containment
PDS 2-4	24 hr	0.104	Low-P Boiloff with flooded containment
PDS 1-3	7 hr	0.032	SBO w/10 hr-firewater, High-P Boiloff
PDS 1-1	7 hr	0.019	LBLOCA with flooded containment
PDS 1-2	7 hr	0.015	SBO w/o firewater, break in SDC
PDS 1-5	7 hr	0.008	Low-P Boiloff with flooded containment
PDS 2-5	24 hr	0.007	High-P Boiloff with closed containment
PDS 2-6	24 hr	0.006	Open MSIV's with closed containment
PDS 2-3	24 hr	0.054	Same as PDS 2-2, but with potential to recover AC power
PDS 1-4	7 hr	0.005	Same as PDS 1-2, but with potential to recover AC power

7. station blackout with firewater addition.

8. station blackout with 10 hr firewater addition followed by high pressure boiloff. and

9. station blackout with 10 hr firewater addition followed by failure to isolate SDC.

In all these Level 1 cases, the drywell personnel lock is open; the containment equipment hatch and both of the containment personnel locks are open.

Calculations were performed for several different times from shutdown for each of these accident scenarios: 7 hr, 24 hr, 59 hr, 12 days, and 40 days. The first two times correspond to the times used to determine the decay heats for the first and second time windows; the third time corresponds to the midpoint of the second time window; the last time corresponds to the time corresponding to the decay heat level in the third time window. Because the primary interest was in time to core damage, these Level 1 support calculations were run until any of the following: vessel failure, code abort or 24 hr of transient. If any sequence produced no significant core damage within 24 hr for a given decay heat level, no further calculations were done with longer shutdown time s (*i.e.*, lower decay heat levels).

Based partly on the results of the MELCOR calculations done in support of the POS 5 Level 1 analysis, a number of accident sequences were eliminated from consideration as not resulting in core damage within the first 24 hr from the start of the accident. The remaining sequences, those leading to core damage within 1 day and with a frequency greater than the Level 1 truncation frequency, were grouped into plant damage states or PDSs. The plant damage states are ranked by their relative contribution to core damage frequency as:

Complete MELCOR accident analyses have been done for these sequences in support of the Level 2 PRA, with results described in detail. (The last two sequences in the table are identical to other sequences in the table with regard to MELCOR calculations, but with different recovery assumptions in the Level 2 PRA.)

An abridged risk analysis was performed on the early portion of the refueling mode of operation. In the Level 1 coarse screening analysis this mode of operation is referred to as plant operating state 6 (POS 6). During a refueling outage, the plant will enter POS 6 prior to loading fresh fuel (*i.e.*, going down) and then following fuel transfer on the way back up to power conditions (*i.e.*, going up). In this POS 6 study, only the going-down phase is analyzed. POS 6 begins when the vessel head is detached and ends when the upper reactor cavity has been filled with water. Prior to this mode of operation, the containment equipment hatch and personnel locks have been opened, the drywell head has been removed and the drywell equipment hatch and personnel locks have been opened. Thus the suppression pool is effectively bypassed both from the vessel and from the drywell (*i.e.*, steam lines are plugged and the drywell is open).

All the MELCOR POS 6 calculations were done assuming that, at the start of the accident, shutdown cooling, suppression pool cooling and containment sprays are all unavailable and remain unavailable during the accident; coolant injection is not provided is used, what the capabilities and features of MELCOR are, and how the code has been used by others in the past. Brief descriptions of the Grand Gulf plant and its configuration during LP&S operation, and of the MELCOR input model developed for the Grand Gulf plant in its LP&S configuration are given. The results of MELCOR analyses of various accident sequences for the POS 5 plant configuration are presented, for accidents initiated at several different times after scram and shutdown, including shortened thermal/hydraulic and core damage calculations done in support of the Level 1 analysis and full plant analyses, including containment response and source terms, supporting the Level 2 analysis. MELCOR calculations of various accident scenarios for POS 6 also are given; these include a reference calculation and sensitivity studies on both plant configuration assumed and on code input options used. brief summary of this work.

MELCOR [6] is a fully integrated, relatively fast-running, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants, being developed at Sandia National Laboratories for the NRC and the U. S. Department of Energy (USDOE). An entire spectrum of severe accident phenomena is modelled in MELCOR in a unified framework for both boiling water reactors and pressurized water reactors. Characteristics of severe accident progression that can be treated with MELCOR include the thermal/hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heatup, degradation and relocation; fission product release and transport; hydrogen production, transport and combustion; core-concrete attack; heat structure response; and the impact of engineered safety features on thermal/hydraulic and radionuclide behavior. The MELCOR computer code has been developed to the point that it is now being successfully applied in both experiment analyses, intended for code validation, and in plant analyses, in support of PRAs and accident management studies.

A series of MELCOR calculations were done to support the quantification of the Level 1 PRA models for POS 5. POS 5 is rigorously defined as: "Cold Shutdown (Operating Condition 4) and Refueling (Operating Condition 5) only to the point where the vessel head is off." For these calculations, the parameters of interest include the times to reach various pressure and/or level setpoints, the time to top-of-active-fuel (TAF) uncovery, the times to core heatup and clad failure and the time to vessel failure. Several general scenarios when the plant is in POS 5 have been considered:

1. open MSIVs,
2. low pressure boiloff,
3. high pressure boiloff with closed RPV head vent,
4. high pressure boiloff with open RPV head vent,
5. large break LOCA,
6. station blackout with failure to isolate SDC.

6 Summary

The safety of commercial nuclear plants during full power operation has been previously assessed in many probabilistic safety assessment studies. Recent events at several nuclear power generating stations, recent safety studies, and operational experience, however, have all highlighted the need to assess the safety of plants during low power and shutdown modes of operation. In contrast to full power operation, there is very little information on the safety of plants during low power and shutdown modes of operation. In the past, the assumption has been that power operation is the risk dominant mode of operation because the decay energy is greatest at the time of shutdown and then decays as a function of time. Thus, the rationale was that during shutdown modes of operation the decay heat would be sufficiently low that there would be plenty of time to respond to any abnormal event that may threaten the core cooling function. Furthermore, given the unlikely event that a release did occur, radioactive decay would lessen the radiological potential of the release. This argument's Achilles' heel is that the technical specifications allow for more equipment to be inoperable in off power conditions. Thus, while there may be more time to respond to an accident during shutdown, many of the systems that are relied on to mitigate an accident during power operation may not be available during shutdown.

To gain a better understanding of the risk significance of low power and shutdown modes of operation, the Office of Nuclear Regulatory Research at the NRC established programs to investigate the likelihood and severity of postulated accidents that could occur during low power and shutdown (LP&S) modes of operation at commercial nuclear power plants. To investigate the likelihood of severe core damage accidents during off power conditions, probabilistic risk assessments (PRAs) were performed for two nuclear plants: Unit 1 of the Grand Gulf Nuclear Station which is a BWR-6 Mark III boiling water reactor (BWR) and Unit 1 of the Surry Power Station which is three loop, subatmospheric, pressurized water reactor (PWR). These studies consist of the following five analysis components: accident frequency analysis, accident progression analysis, analysis of the release and transport of radioactive material (*i.e.*, source term analysis), consequence analysis, and a risk integration analysis. A principle product of such a Level 3 PRA is an expression for risk.

The analysis of the BWR was conducted at Sandia National Laboratories while the analysis of the PWR was performed at Brookhaven National Laboratory. This multi-volume report presents and discusses the results of the BWR analysis. Volumes 2-5 present the accident frequency analysis (*i.e.*, Level 1). Volume 6 presents the Level 2/3 analysis performed under FIN L1679. Part 1 of Volume 6 presents the accident progression, radionuclide release and transport, consequence and risk analyses. The subject of this part, *i.e.*, Part 2 of Volume 6, presents the deterministic code calculations, performed with the MELCOR code [6], that were used to support the development and quantification of the PRA models.

In this report, the background for the work documented in this report is first summarized, including how deterministic codes are used in PRAs, why the MELCOR code temperatures, the heat transfer from the fuel rods to the colder canister became sufficient to prevent the rods from heating further. The cooling of these rods after the onset of oxidation was associated with increased convection heat transfer coefficients due to the addition of hydrogen to the control volumes. This increased the heat transfer rates to the colder canisters. The canister and control rods for cell 218 were cooled by the bypass water pool.

Cell 214 was partially covered by water within the fuel assemblies. Cells 206 through 214 were all kept cooled. The component temperatures for cell 214 in Figure 13 show that the fuel rods peaked at about 362 K and the canisters at about 334 K. The heat generated within the fuel rods was transferred to the water within the fuel assemblies and then conducted through the canisters to the colder bypass water. The heat transfer through the canisters was sufficient to keep the water within the assemblies subcooled. These cells cooled after the onset of cladding oxidation because the pool temperatures were decreasing unrealistically due to the unknown code error discussed above.

Cladding oxidation is illustrated by the production of hydrogen as shown in Figure 14. The oxidation began at about 9010 seconds (2.50 hours). The convection heat transfer coefficient for a section of the core shroud is shown in Figure 15 along with bypass hydrogen mole fraction.

Recirculation Pipe LOCA with Two LPCI Pumps

A low decay power shutdown LOCA was run involving the double-ended rupture of a pump suction pipe in a recirculation loop with ECCS provided by two LPCI pumps. The LPCI pumps water from the containment suppression pool into the core bypass region. The broken recirculation pipe allowed all the reactor vessel water above the jet pump throat to drain from the vessel leaving the upper 1/3 core exposed, without significant cooling, and subject to damage. The initialization of this calculation was identical to the one pump calculation except that the LPCI flow rate to the core bypass was 15240 gpm.

The water levels for the two pump calculation are shown in Figure 16. The bypass volume completely filled and over flowed into core channel with the average channel level remaining about .24 m above the top of the jet pump throats. After the initial transient was complete, about 24% of the LPCI water over flowed the core from the bypass into the core channel. The downcomer water level remained below the jet pumps and so had no effect on the core water levels. The water that over flowed the core into the channels flowed downwards through the fuel assemblies and into the lower plenum. Core channel evaporation was very minor.

The 2/3 of the core under water was cooled by heat transfer to the pool within the fuel assemblies. This pool remained subcooled due to the bypass over flow and to heat transfer through the channel boxes to the bypass pool. The channel pools remained subcooled by about 57, 57, 57, 58, 61, and 64 K for core rings 1 through 6, respectively. The injected water was heated by only 1.0 K within the bypass and by about 3.0 K before flowing out of the vessel.

The core heating is shown in the next four figures. Figure 17 shows the cladding temperatures of ring 2 which had the highest power density and therefore the highest temperatures of all the rings. The cladding temperature for the cells at axial level 17 for each ring are shown in Figure 18. The cell component temperatures for cells 217, and 214 are shown in Figures 19, and 20, respectively.

The highest temperatures in this calculation were 691 and 685 K for the fuel and cladding of cell 217 as shown in Figure 19. Since the bypass over flowed the core in this calculation, all of the fuel assembly canisters were cooled which limited the fuel rod heating even for the uncovered cells.

Cell 214 was covered by water within the fuel assemblies. Cells 206 through 215 were all kept cooled. The component temperatures for cell 214 in Figure 20 show that the fuel rods peaked at about 319 K and the canisters at about 311 K. These cells were cooled by both the water over flowing the top of the core and by conduction through the canisters to the colder bypass water.

The temperatures of this calculation were over predicted because MELCOR lacks the fuel rod film model needed for calculating the heat transfer to the water running down the fuel rods from the bypass core over flow. Therefore, the actual fuel rod cooling would have been much greater than calculated. However, the temperatures predicted in this calculation did not even approach either the cladding failure temperature or the temperature needed to initiate oxidation. No hydrogen was produced and fuel damage was not predicted.

UNCERTAINTY DISCUSSION

Flow Loss Coefficients

The reverse bypass inlet and the reverse jet pump loss coefficients were major uncertainties in determining whether or not the core bypass water level over flowed the top of the core into the fuel assemblies. Core damage will generally be prevented if water over flows the top of the core into the fuel assemblies. The coefficients used in these calculations represent a reasonably good estimate considering the lack of data required to compute accurate numbers but uncertainty still exists. Therefore, a parameter study was performed to determine the sensitivity of the bypass water level to these coefficients.

The bypass water level is shown in Figure 21 as a function of the reverse bypass inlet coefficients at three different jet pump coefficients and at LPCI flow rates corresponding to 1 and 2 operating pumps. The bypass inlet coefficient ranges from zero to a number sufficient to cause the bypass to over flow the core. The jet pump coefficient values are 0, 1, and 5 for the unknown throat exit number plus 0.0531 calculated for flow into and through the diffuser. Also marked in the figure are levels for the jet pump throat exit, the top of the active fuel, and the top of the fuel assembly canisters. The bypass water level for the base case coefficients used in these calculations was .7 m above the jet pumps.

The bypass inlet coefficient would need to be more than a factor of two higher to force the flow from just one pump over the top of the core and it would have to be reduced by 40% or more to prevent the flow from two pumps from going over the top. The FSAR shows nine different paths for water to flow from the lower plenum to the core bypass. If one attempts to estimate the flow area of the lower tie plate holes and leakage between parts and treat this area as an orifice in a channel with the bypass flow area, a range of coefficients can be calculated which includes the base case coefficient. While it is not possible to prove with the limited available Grand Gulf data, it unlikely that one operating LPCI pump will prevent fuel damage and it is very likely that two pumps will.

Fuel Bundle Center Peaking

MELCOR calculates cell average temperatures which is appropriate to calculating the heating of a fully uncovered core. But for conditions encountered during these calculations where the dominate heat transfer was to a cooled canister, center bundle temperature peaking is a concern.

When a core cell was uncovered within the fuel assembly but the bypass was water filled, the heat transfer from the fuel rods to the canister has a radiative heat transfer component. The center bundle fuel rods are shielded from the canister by the outer rods and their temperatures would be higher than the outer rods. The input fuel-to-canister radiative exchange factor could also be considered an uncertainty factor. The question is how much higher is the peak temperatures than the bundle average temperature.

When a core cell was completely covered by water both inside and outside the canister, the fuel rods were cooled by a subcooled pool. Water near the inner rods would have been hotter than the pool average and possibly boiling could have occurred locally where it was not predicted by the volume average temperature. This would have enhanced the convective cooling of the upper exposed fuel.

Decay Power

The normalized time dependent decay power distribution used in these calculations was developed for the LaSalle plant and is another uncertainty in the results. Using a higher powered decay heat curve or initiating the calculation earlier would increase the predicted temperatures.

Some fuel assemblies had higher power factors than their associated ring averages (the highest was 1.232). Therefore, some fuel assemblies will heat to higher temperatures than predicted in these calculations.

Further, certain fuel rods within a particular fuel assembly have higher power densities than the assembly averages.

Nodalization

Higher temperatures may have been predicted with finer nodalization. For instance, finer control volume nodalization within the exposed core would have created cells with higher power densities resulting in higher temperature predictions. Finer control volume nodalization within the subcooled pool would have predicted portions of the pool with less subcooling and localized boiling would then have been more probable.

CONCLUSIONS

The more complex MELCOR boiloff calculation verified the results of the level I hand calculation for the time of core uncovery and in addition provided an estimate of the onset of fuel damage. While the collapsed water level reached the top of the active fuel at about 13.1 hours, the fuel did not begin to heat until about 16.5 hours with the onset of oxidation at 18.8 hours.

The MELCOR low decay power LOCA (recirculation loop pipe break) calculations predicted severe fuel damage with cladding oxidation beginning at about 2.5 hours if only one LPCI pump operated and no fuel damage if two pumps operated. Although uncertainties exist in these calculations, core damage will generally be prevented if water over flows the top of the core into the fuel assemblies. The loss coefficient sensitivity study generally showed that it is unlikely that one operating LPCI pump will over flow the top of the core but that it is likely that two pumps will.

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Table 1: Six Ring Core Model Data

Ring Number	Number of Assemblies	Outer Radius (ft)	Volume Fraction	Power Fraction	Power Factor
1	112	3.	.140	.1608	1.149
2	204	5.	.255	.2996	1.175
3	132	6.	.165	.1908	1.156
4	168	7.	.210	.2194	1.045
5	100	7.5	.125	.0923	.738
6	84	8.	.105	.0371	.353

Table 2a: Course Axial Power Distribution Model

Cell Number	Cell Height (m)	Volume Fraction	Power Fraction	Power Factor
13	.3591	.08170	0.	0.
12	.6350	.14446	.1069	.7400
11	.6350	.14446	.1648	1.1409
10	.6350	.14446	.1828	1.2655
9	.6350	.14446	.1936	1.3403
8	.6350	.14446	.2000	1.3846
7	.6350	.14446	.1519	1.0516
6	.2268	.05160	0.	0.

Table 2b: Fine Axial Power Distribution Model

Cell Number	Cell Height (m)	Volume Fraction	Power Fraction	Power Factor
22	.3591	.08169	0.	0.
21	.15875	.03611	.0152	.4209
20	.15875	.03611	.0244	.6757
19	.15875	.03611	.0316	.8751
18	.15875	.03611	.0357	.9886
17	.15875	.03611	.0385	1.0662
16	.15875	.03611	.0401	1.1105
15	.15875	.03611	.0425	1.1770
14	.15875	.03611	.0437	1.2102
13	.15875	.03611	.0445	1.2323
12	.15875	.03611	.0453	1.2545
11	.15875	.03611	.0461	1.2767
10	.15875	.03611	.0469	1.2988
9	.6350	.14446	.1936	1.3402
8	.6350	.14446	.2000	1.3845
7	.6350	.14446	.1519	1.0515
6	.2268	.05161	0.	0.

Table 3: Fuel Assembly and Control Rod Musses

Material	Fuel Assembly Each	Fuel Assembly Total	Control Rod Each	Control Rod Total
UO_2	458.0	366400	0	0
Zr	224.7	179760	0	0
Steel	16.3	13040	203.7	39314
B_4C	0	0	14.3	2760
Total	699.0	559200	218.0	42074

Table 4: Reactor Vessel Control Volumes

Volume Description	Volume (ft^3)	Volume (m^3)	Elevation Lower (m)	Elevation Upper (m)
Lower Plenum	3814.6	108.03	0.	8.0936
Downcomer	6935.6	196.42	3.5462	15.4304
Core - Channels	1304.7	36.95	5.2672	9.6630
Core - Bypass	1086.8	30.78	5.2672	9.6630
Upper Plenum & Separators	2280.4	64.58	9.6630	15.4304
Dryers & Steam Dome	6759.9	191.44	15.4304	22.2493
Total	22182.0	628.20		

Table 5: Reactor Vessel Flow Paths

Description	Elevations From (m)	Elevations To (m)	Loss Coefficients Forward	Loss Coefficients Reverse	Flow Area (m^2)
Core Channel Inlet	5.267	5.267	21.81	29.64	7.861
Core Bypass Inlet	5.267	5.267	1338.	1637.	5.528
Core Channel Outlet	9.663	9.663	9.13	9.37	7.861
Core Bypass Outlet	9.663	9.663	446.	546.	5.528
Separators	15.43	15.43	9.1	2.8	3.318
Dome to Downcomer	15.43	15.43	1.	1.	13.9
Separator Drains	13.1	13.1	3.	3.	3.2
Jet Pump Diffusers	8.064	3.459	.178	1.0531	.4981[a]
Upper Head	19.4	19.4	1.	1.	31.9
Recir Pump Suction	4.377	-5.7	2.	2.	.4576[b]
Recir Inlet Nozzle	8.750	-5.7	350.	350.	.4995[c]

a - throat area
b - two 24 inch nominal O.D. pipes
c - twelve 10 inch nominal O.D. pipes

Table 6: Jet Pump Flow Data

Parameter	Unit	Throat Segment	Diffuser Segment	Extension Segment	Totals
Individual Area [a]	m^2	.0208	.0564[b]	.110	
Total Area	m^2	.498	1.35	2.63	
Lower Elevation[c]	m	6.18	4.23	3.49	
Length	m	1.86	1.95	.740	4.54
Hydraulic Diameter	m	.163	.268	.373	
Surface Roughness	m	7.6E-7	7.6E-7	7.6E-7	
Loss Coefficients[d]					
Forward		.05[e]	.092[f]	.036[g]	.178
Reverse		.018[h]	.035[i]	1.[j]	1.053

a - 24 Individual Pumps
b - based on average diameter
c - top throat elevation estimated at 8.03 m
d - Ref. Crane and adjusted to throat area (i.e., d_1^4/d_2^4)
e - rounded protruding entrance
f - expansion (Crane, page A-26, formula 3)
g - pipe exit to reservoir (nominal 1.)
h - right angle entrance flush with wall (nominal .5)
i - contraction (Crane, page A-26, formula 1)
j - exit into empty reservoir with an obstruction (judgement)

Table 7: Core Loss Coefficients

	Forward	Reverse
Channel Inlet	21.8	29.6
Channel Outlet	9.1	9.4
Bypass Inlet	1340	1640
Bypass Outlet	450	550

Table 8: Boil-Off Calculation Event Times

		Event Times	
Event	Seconds	Minutes	Hours
On-Set of Boiling	7050	119	2.0
Rapid Boiling	7510	125	2.1
Core Cavitation	37920	632	10.5
Core Uncovery (Collapsed)	45392	757	12.6
TAF Uncovery (Collapsed)	47014	784	13.1
Jet Pump Throat Uncovered	57600	960	16.0
Upper Plenum Water Exhausted	57800	963	16.1
On-Set of Fuel Heating	59410	990	16.5
On-Set of Oxidation	67730	1129	18.8
1 kg of Hydrogen	67872	1131	18.9
First Fission Product Release	68034	1134	18.9

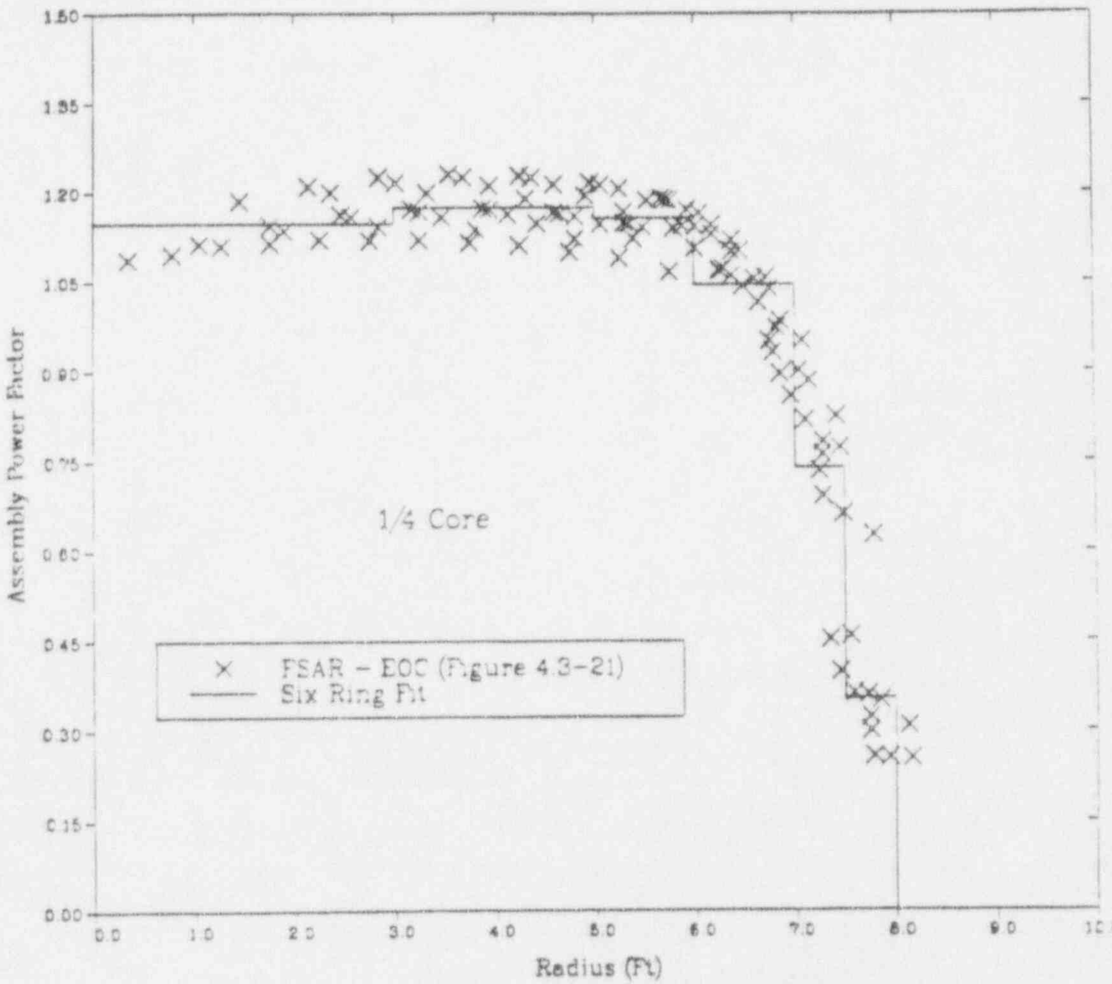

Figure 1: Grand Gulf Radial Power Factors

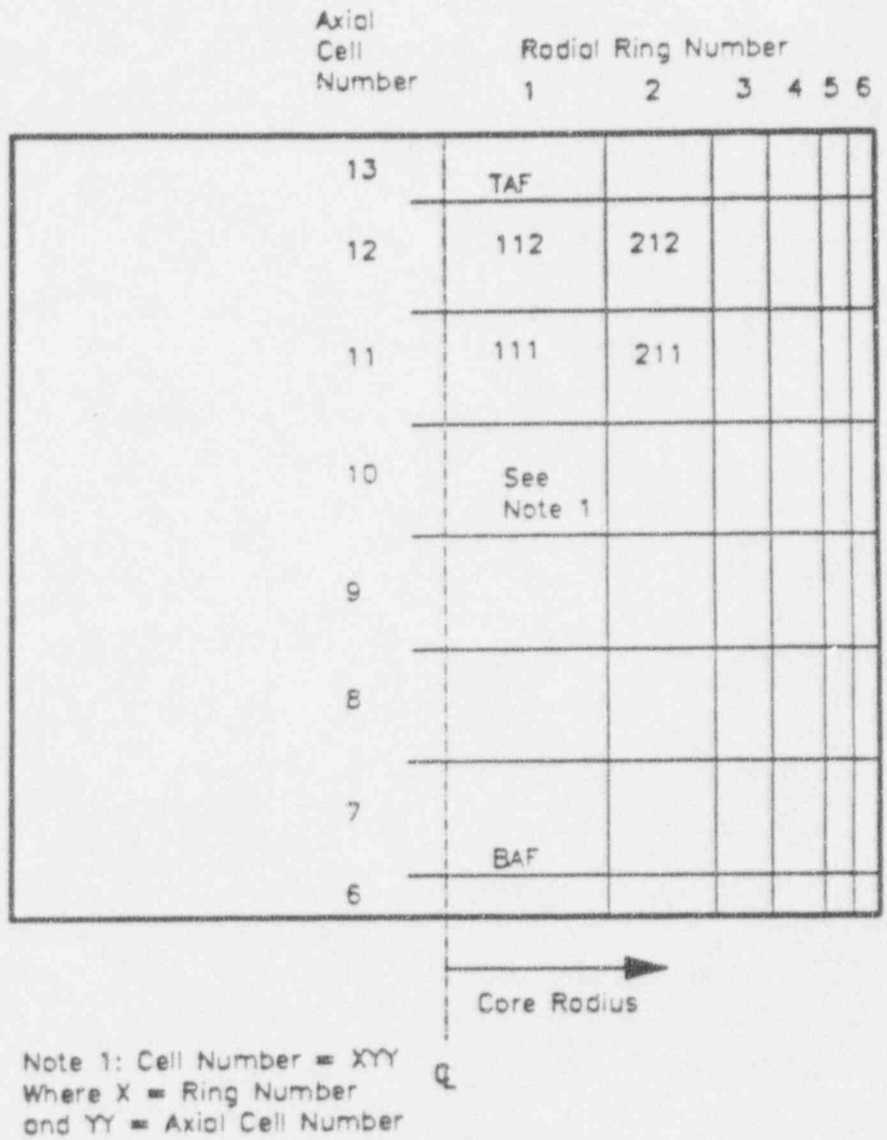

Figure 2

Six Ring Course Node Core Model

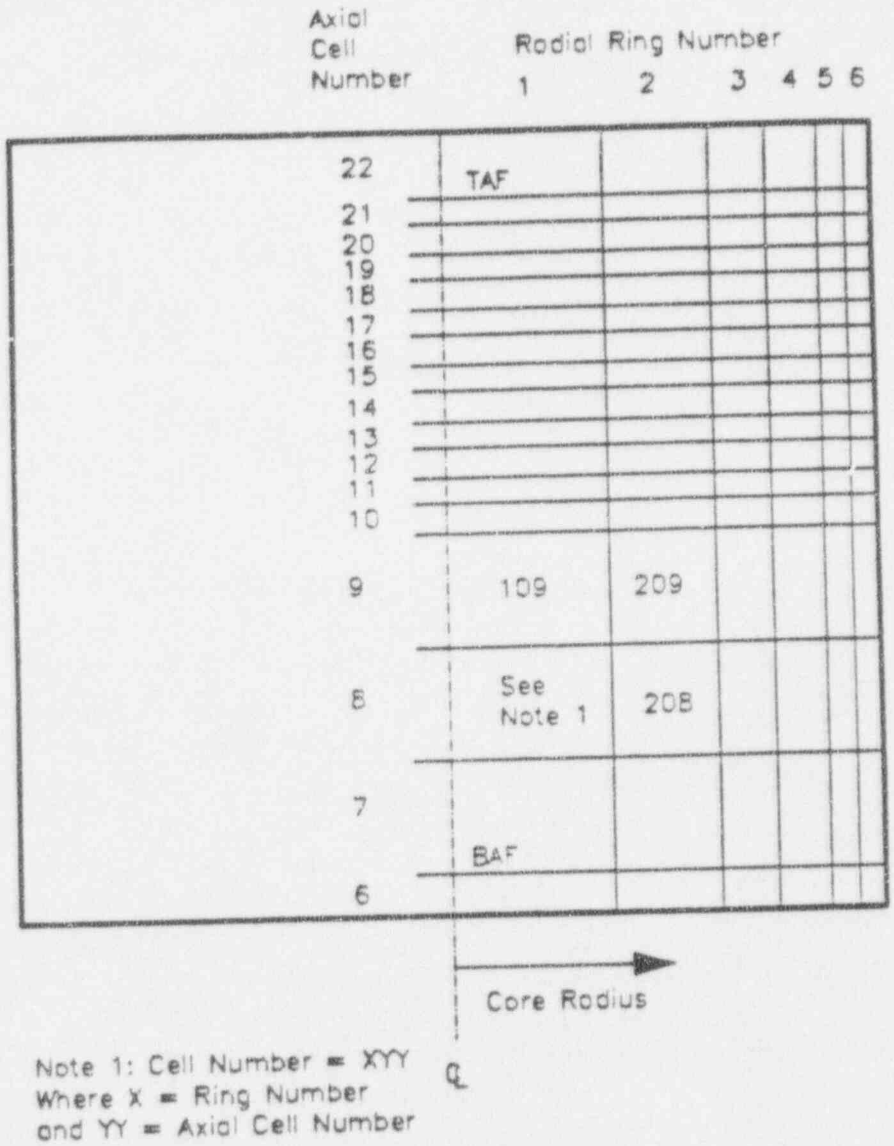

Figure 3

Six Ring Fine Node Core Model

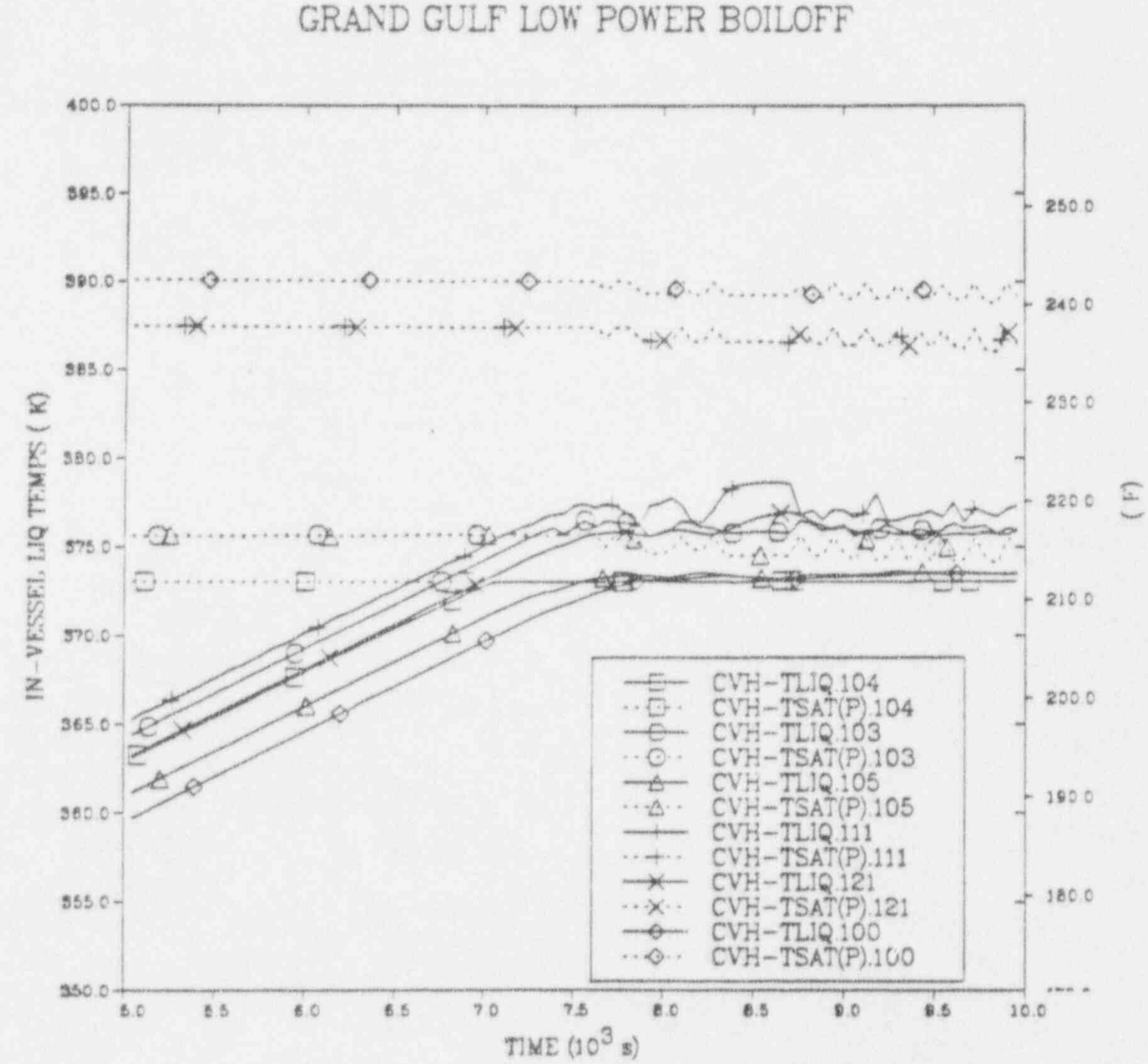

Figure 4: Pool and Saturation Temperatures

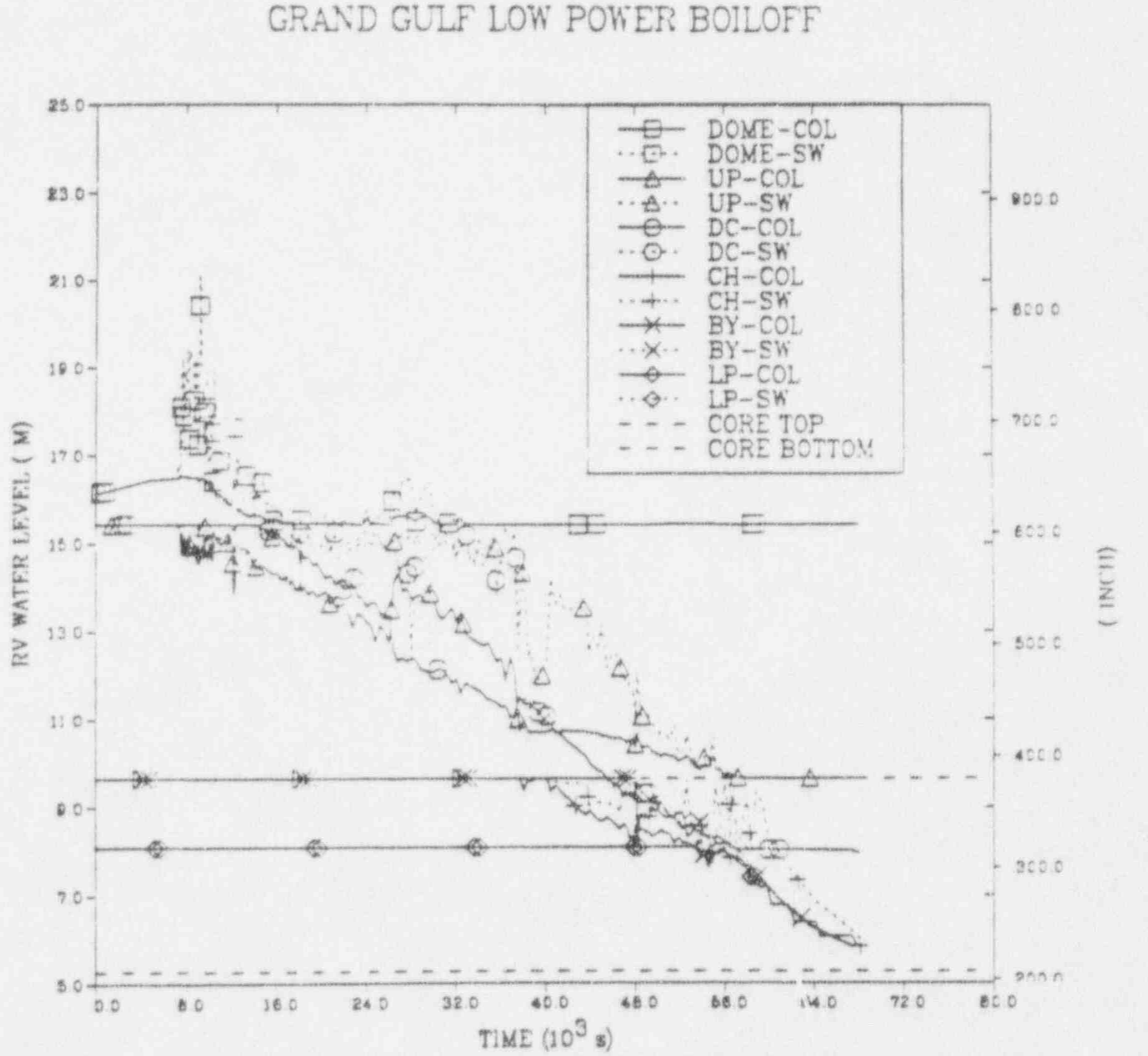

Figure 5: Collapsed and Swollen Water Levels

GRAND GULF LOW POWER BOILOFF

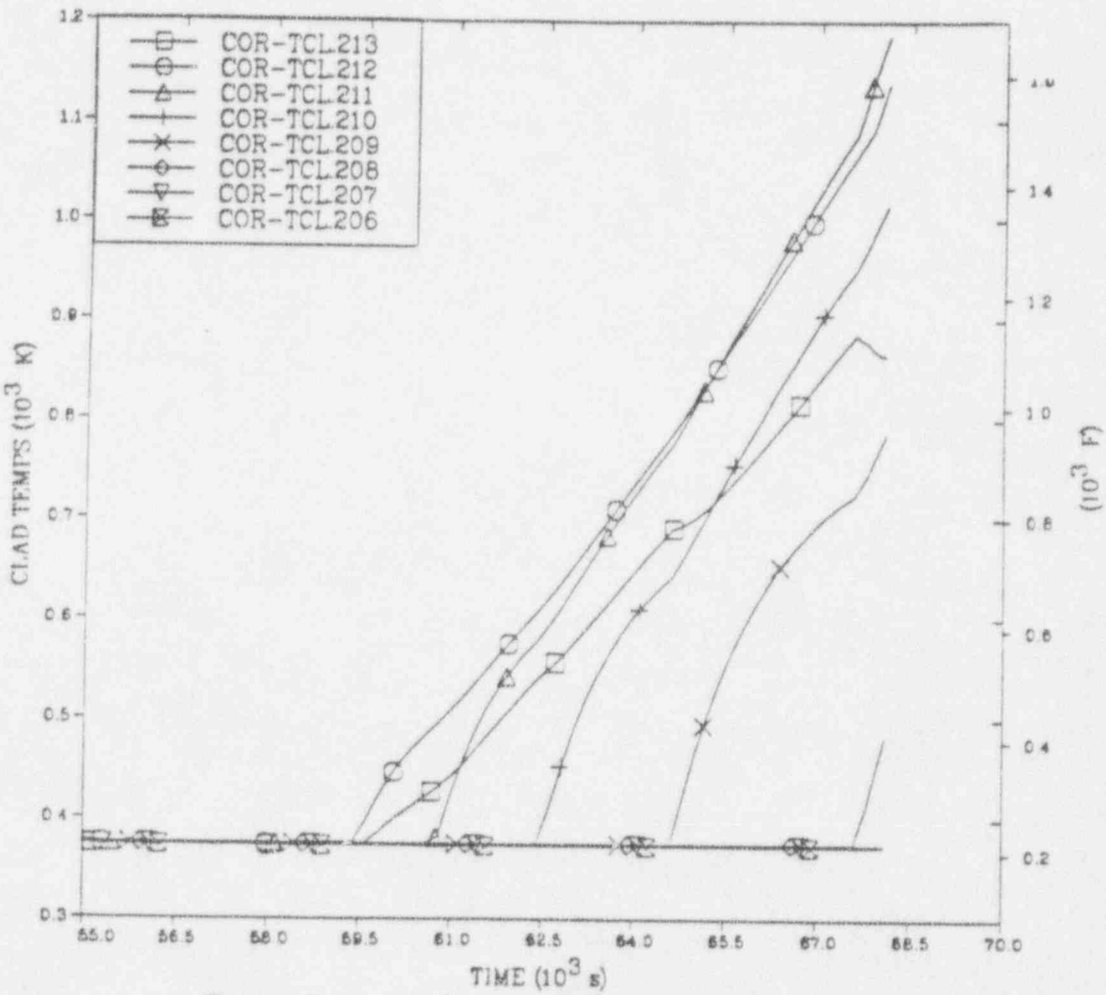

Figure 6: Second Ring Cladding Temperatures

GRAND GULF LOW POWER LOCA ACCIDENT

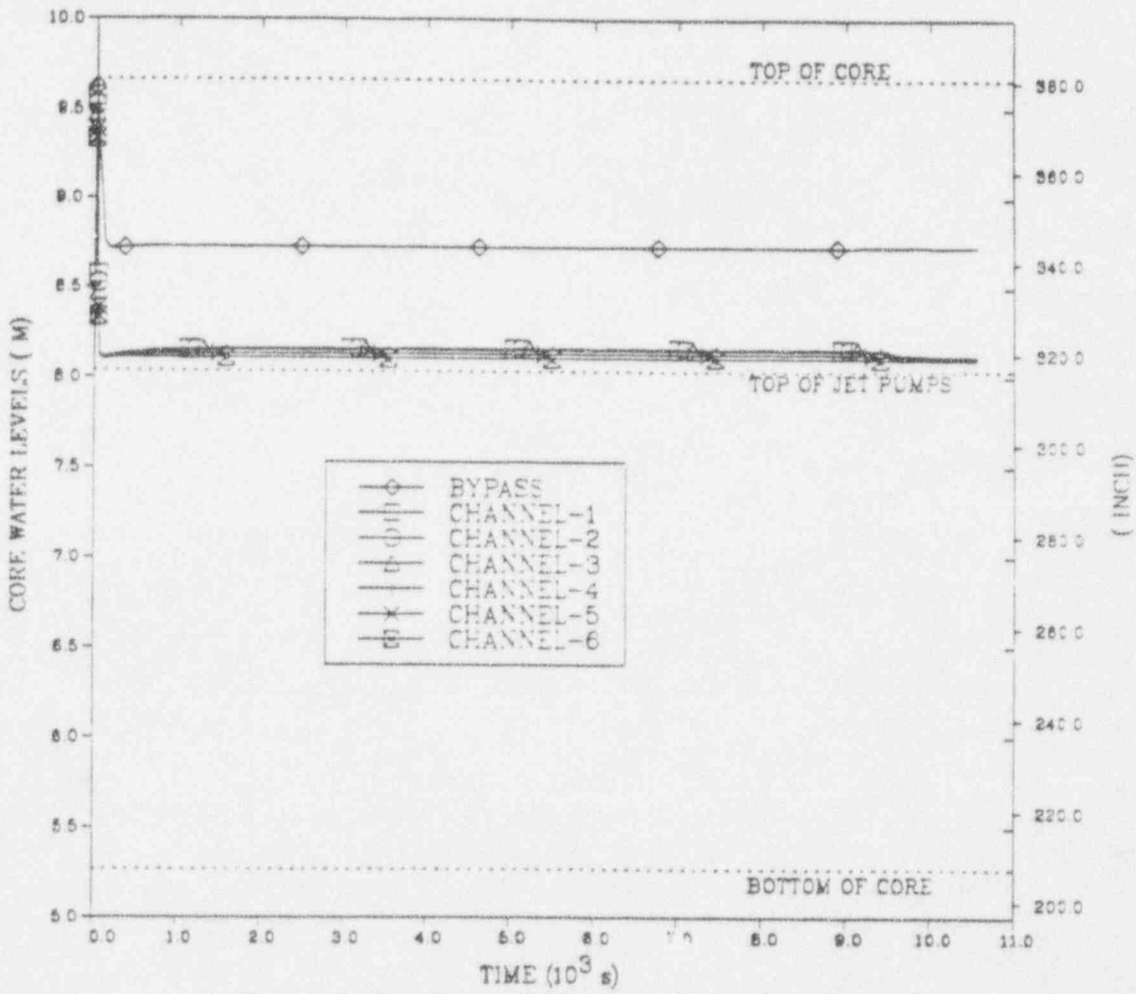

Figure 7: One Pump LOCA Core Water Levels

GRAND GULF LOW POWER LOCA ACCIDENT

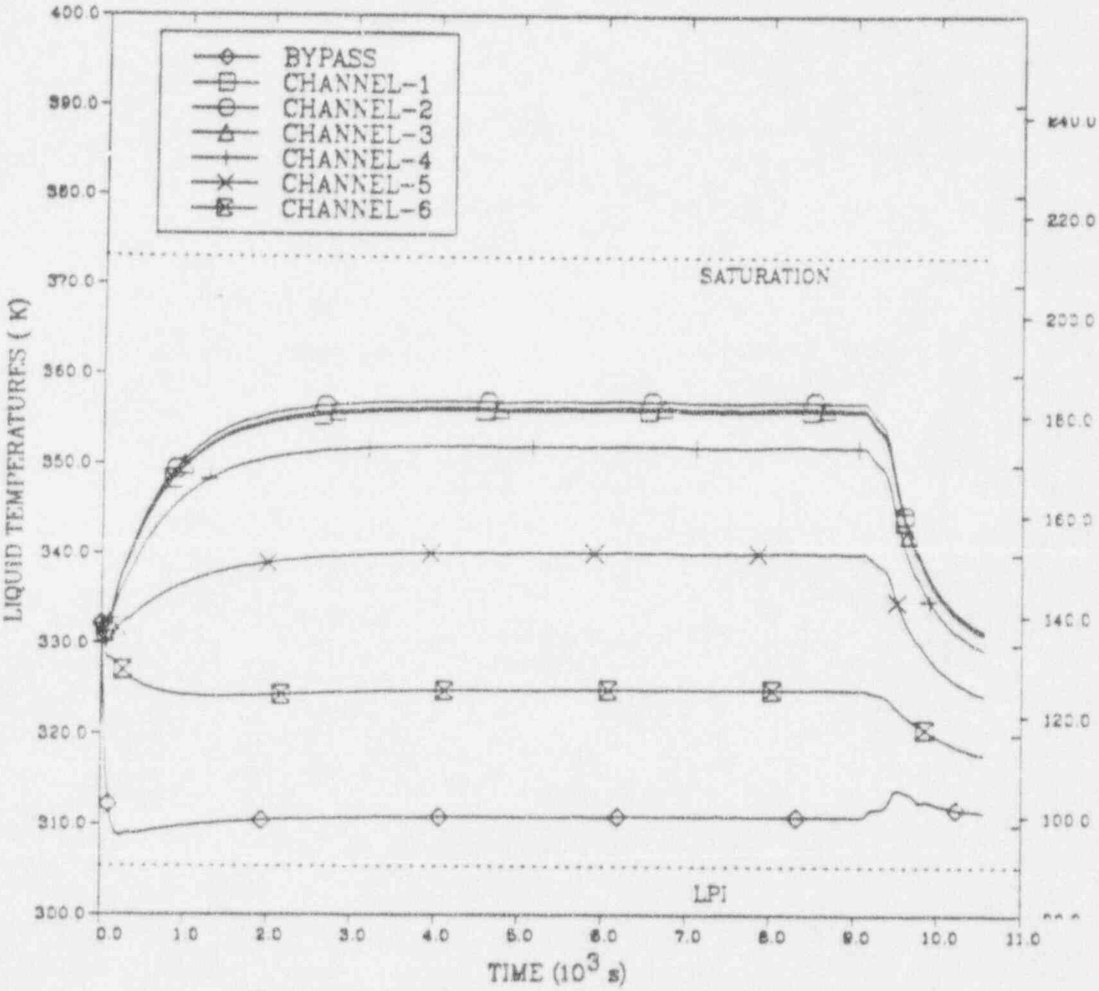

Figure 8. One Pump LOCA Water Temperatures

GRAND GULF LOW POWER LOCA ACCIDENT

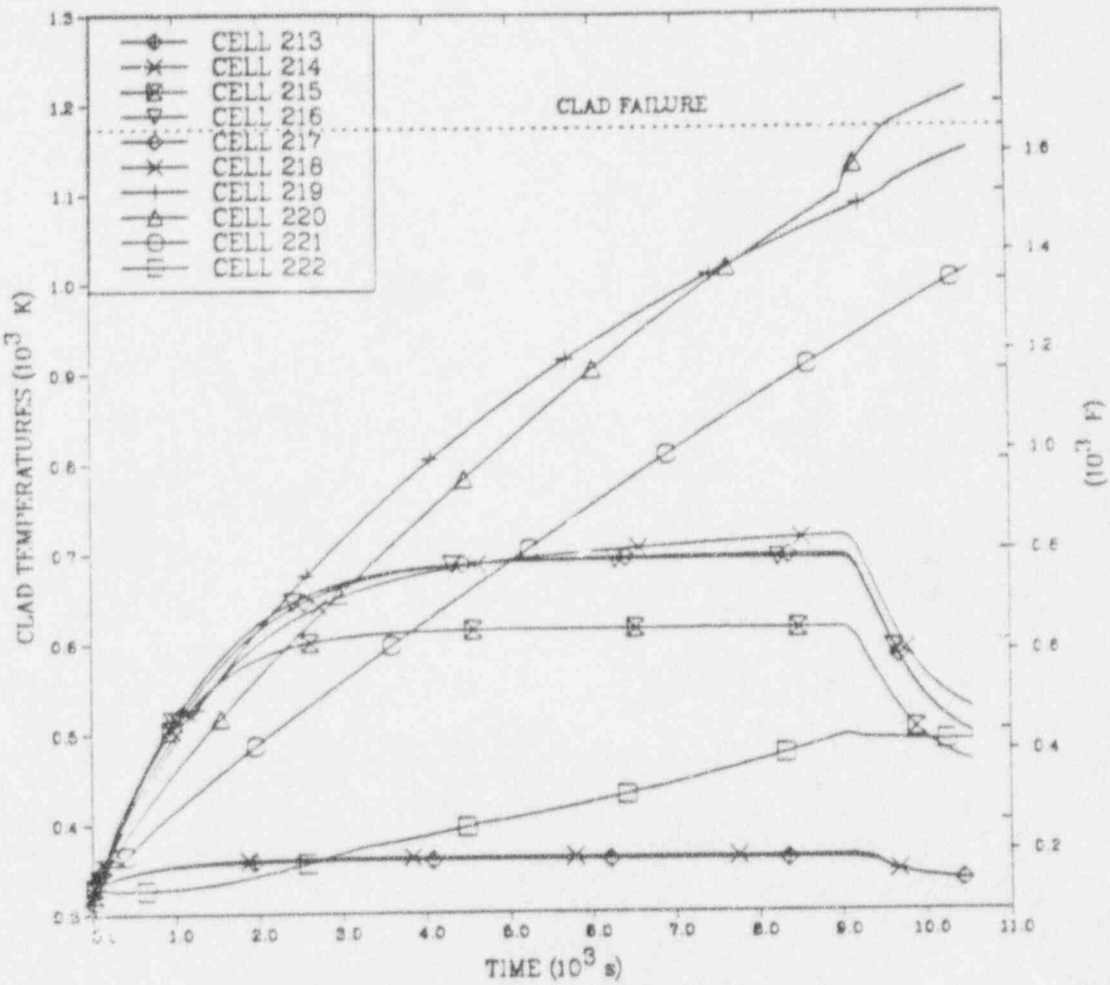

Figure 9: Second Ring Clad Temperatures

GRAND GULF LOW POWER LOCA ACCIDENT

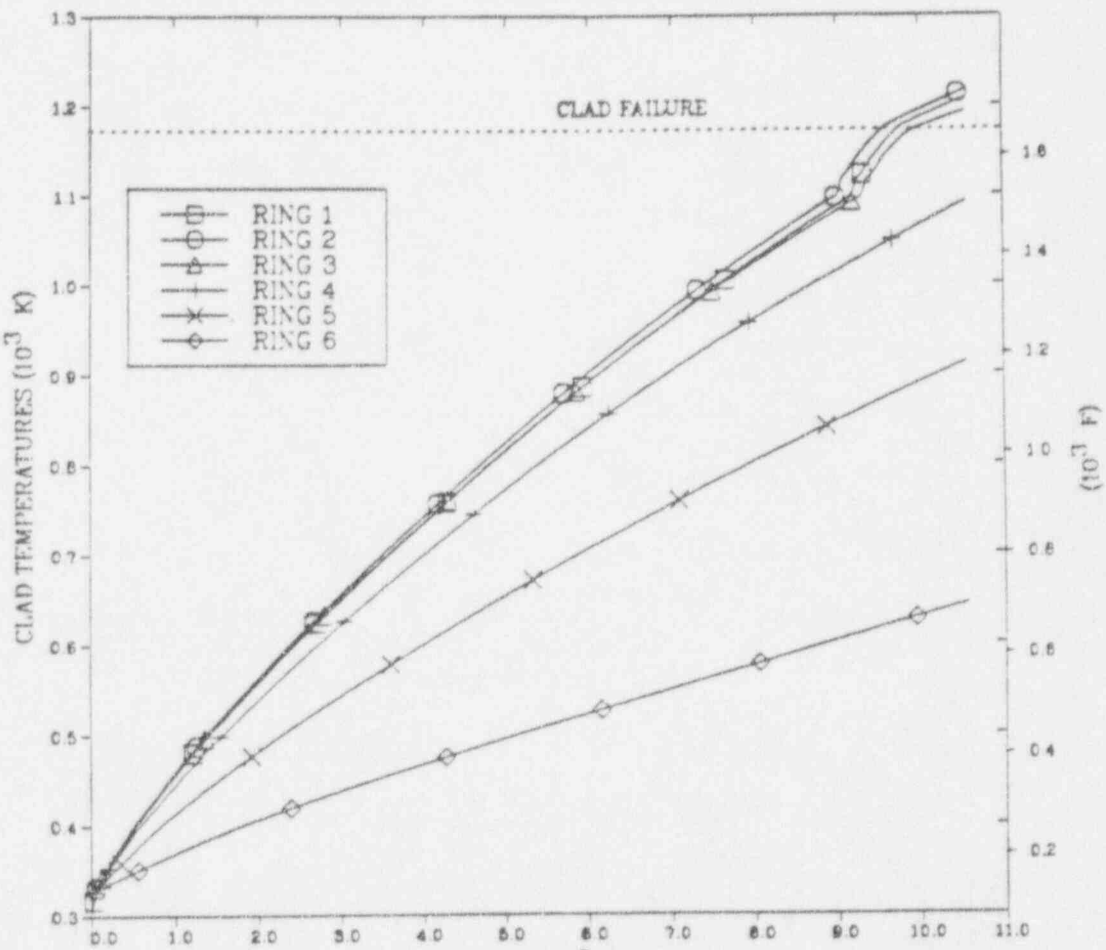

Figure 10: Axial Level 20 Clad Temperatures

GRAND GULF LOW POWER LOCA ACCIDENT

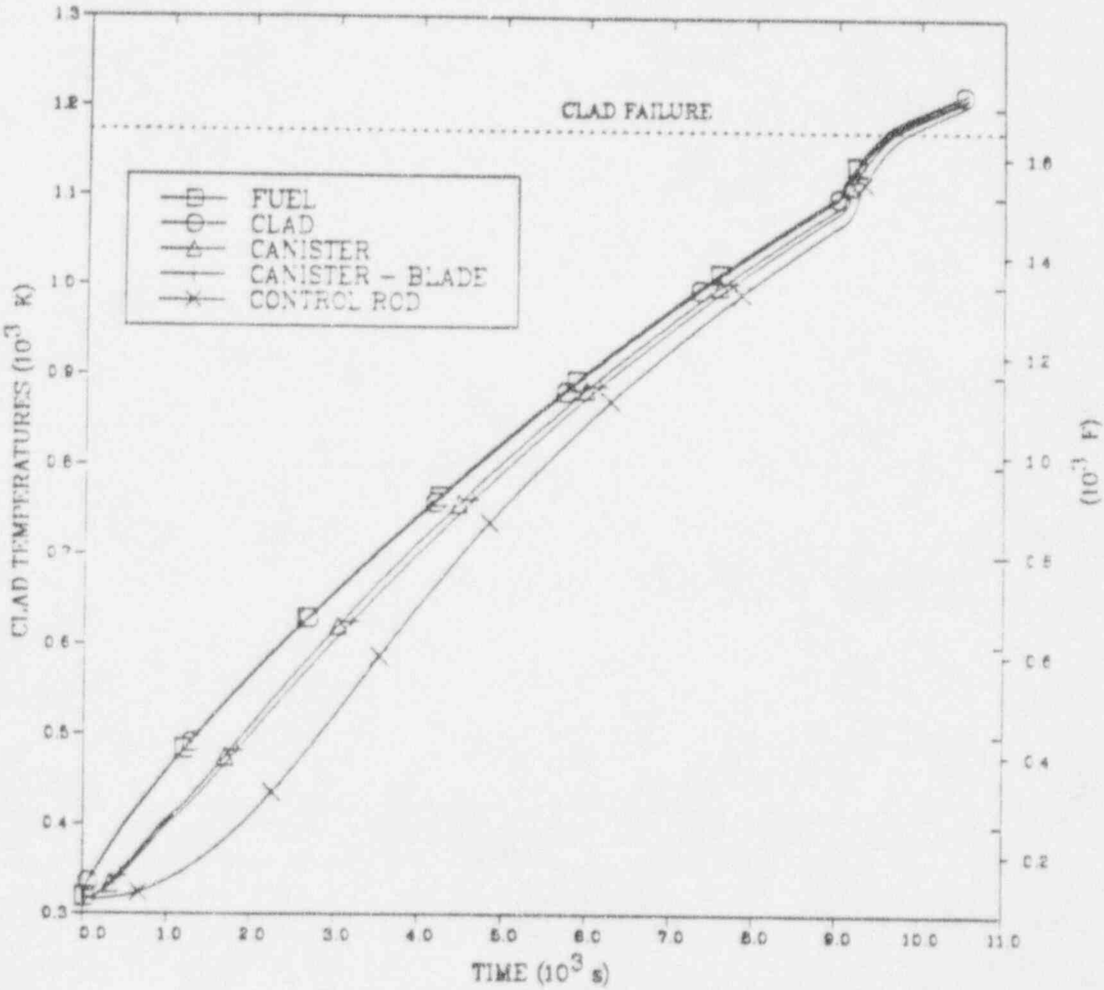

Figure 11: Cell 220 Temperatures

GRAND GULF LOW POWER LOCA ACCIDENT

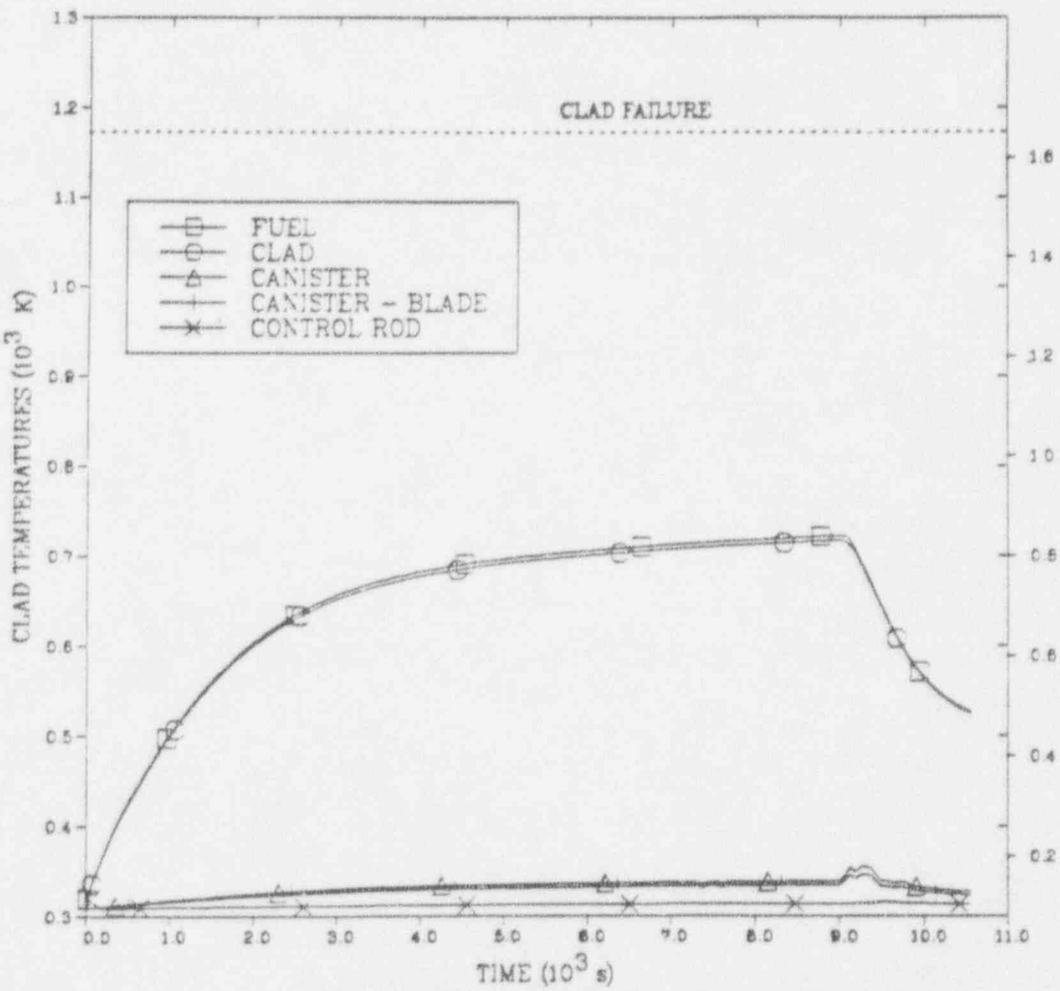

Figure 12: Cell 218 Temperatures

GRAND GULF LOW POWER LOCA ACCIDENT

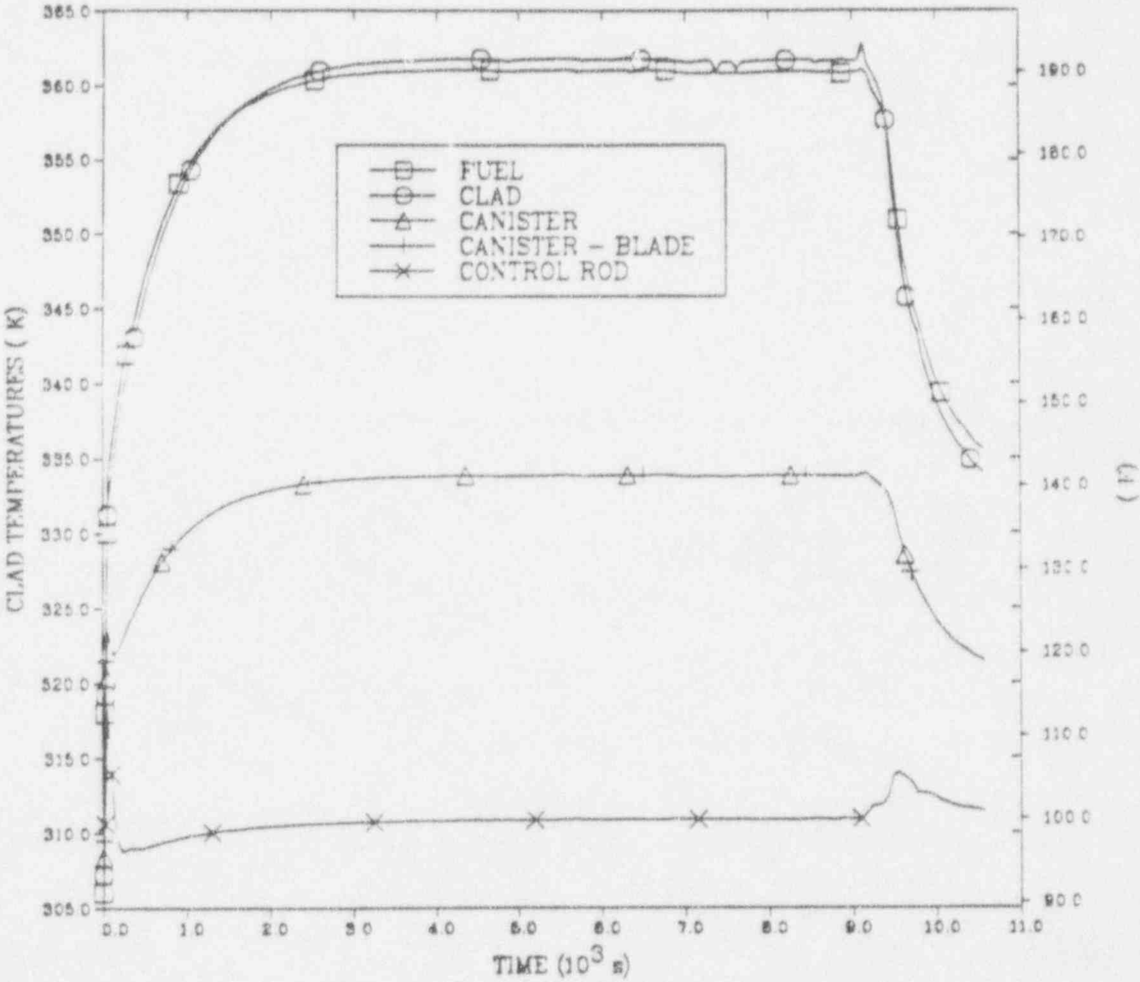

Figure 13. Cell 214 Temperatures

GRAND GULF LOW POWER LOCA ACCIDENT

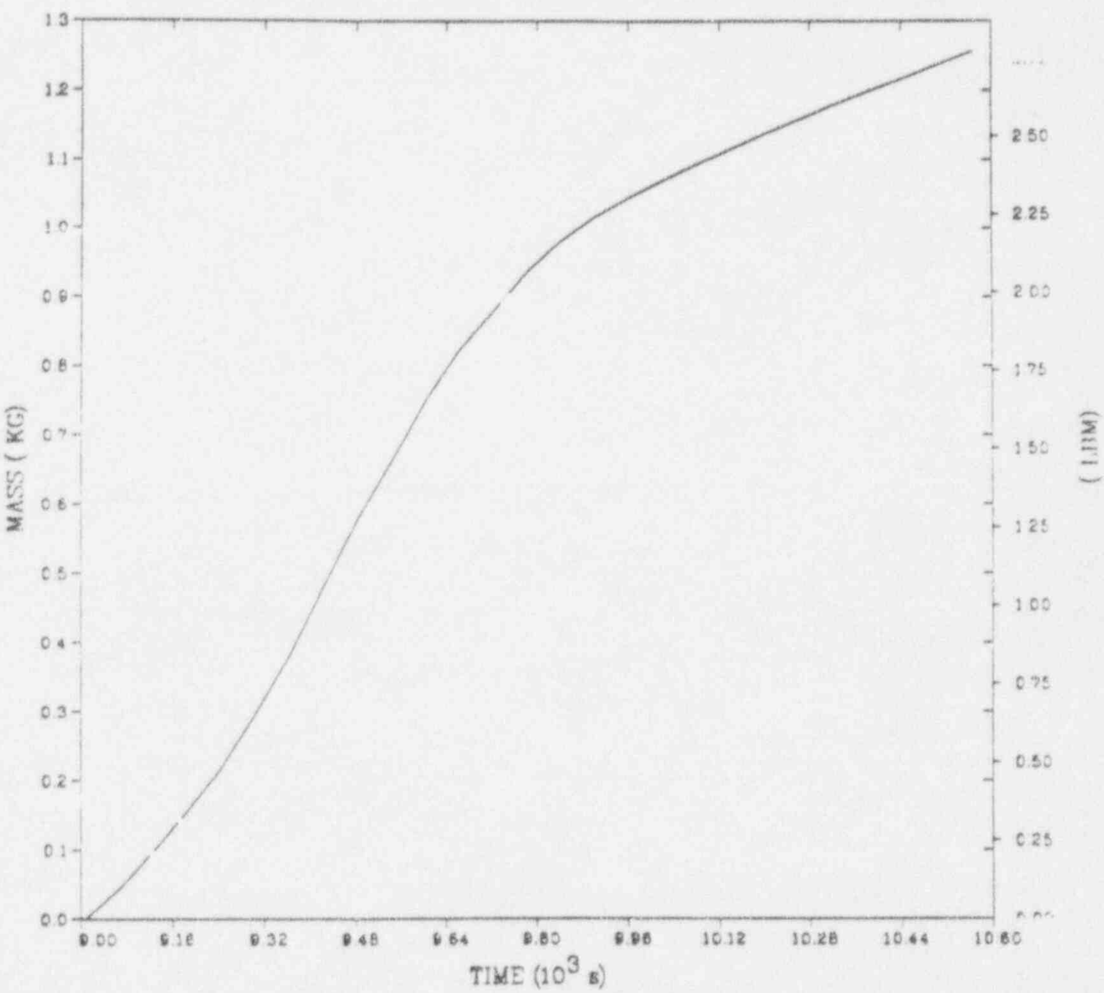

Figure 14: Hydrogen Produced

GRAND GULF LOW POWER LOCA ACCIDENT

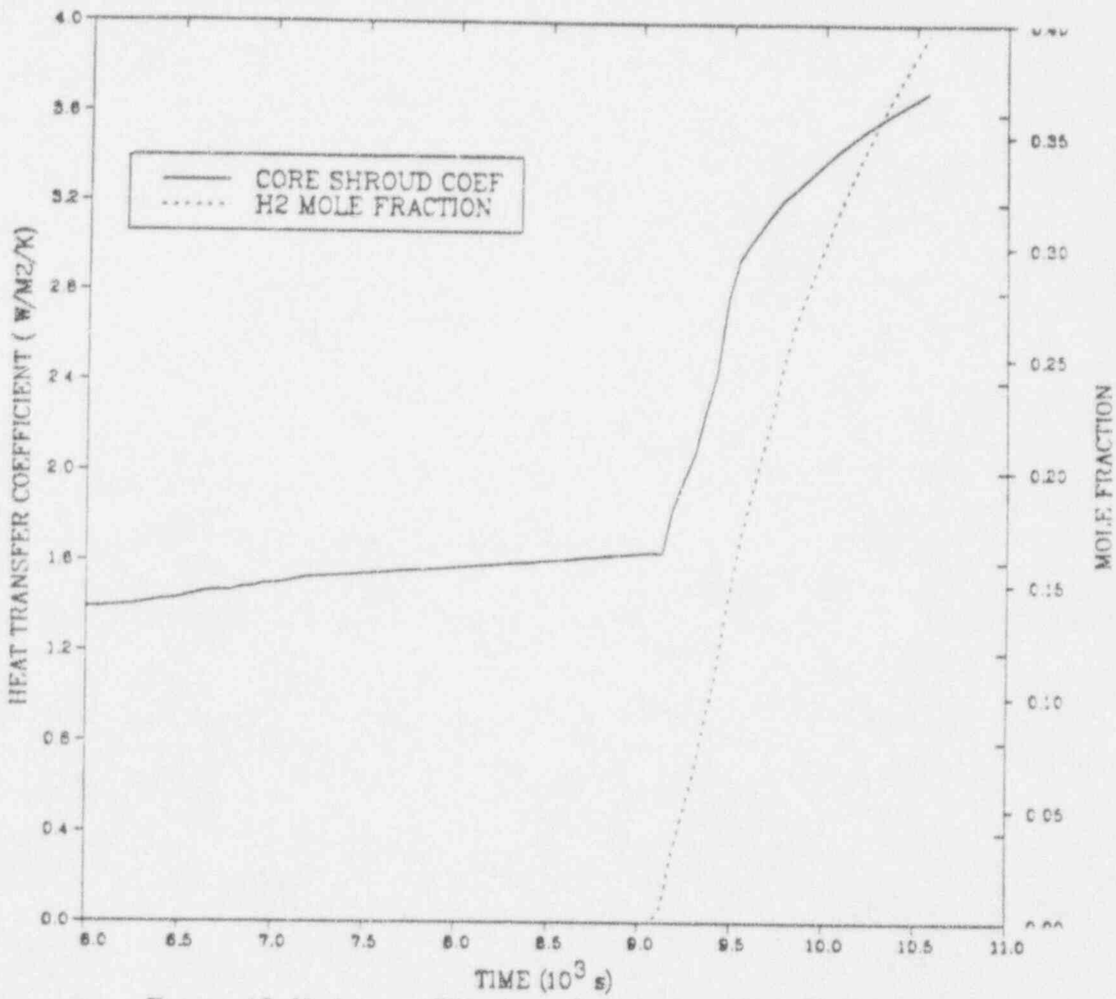

Figure 15: Hydrogen Effect on Convection Heat Transfer

GRAND GULF LOW POWER LOCA ACCIDENT

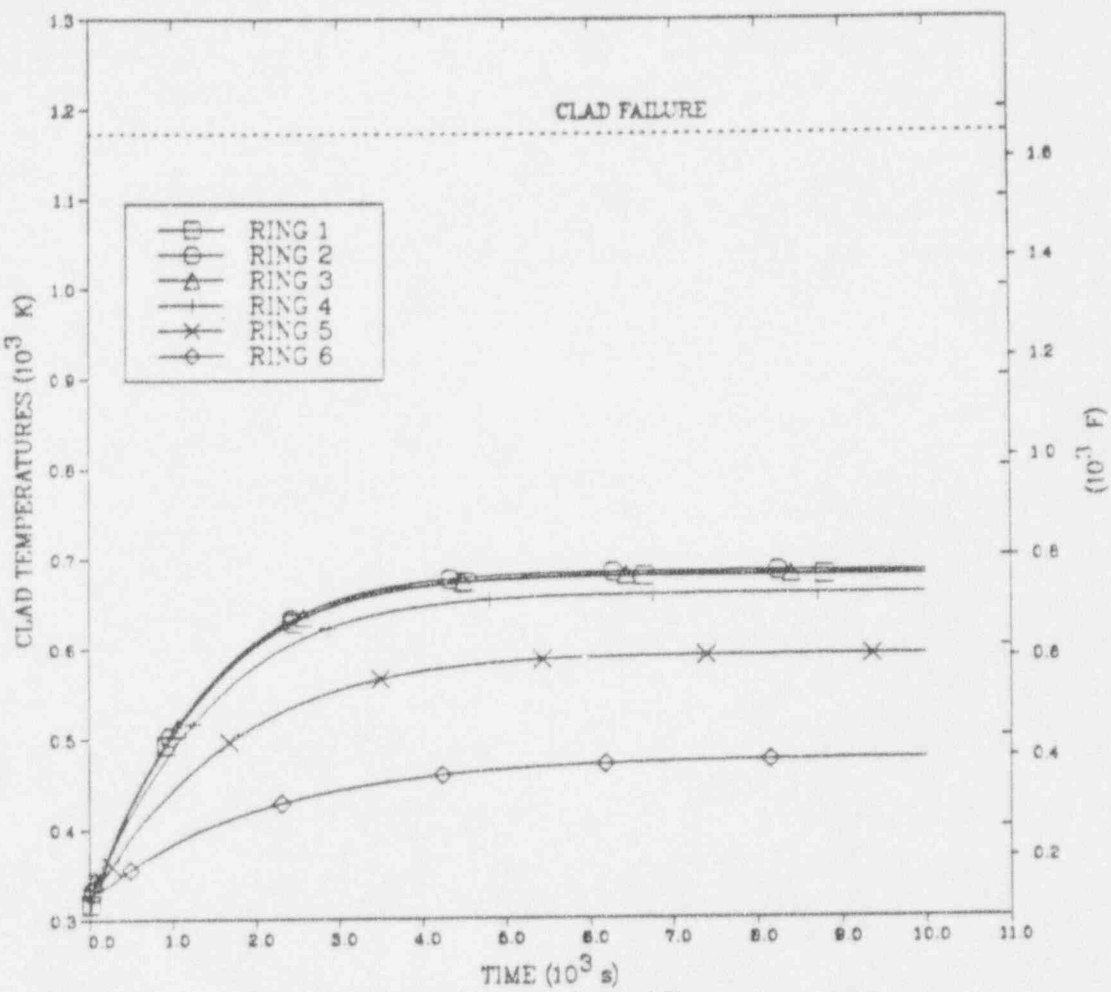

Figure 18: Axial Level 17 Clad Temperatures

GRAND GULF LOW POWER LOCA ACCIDENT

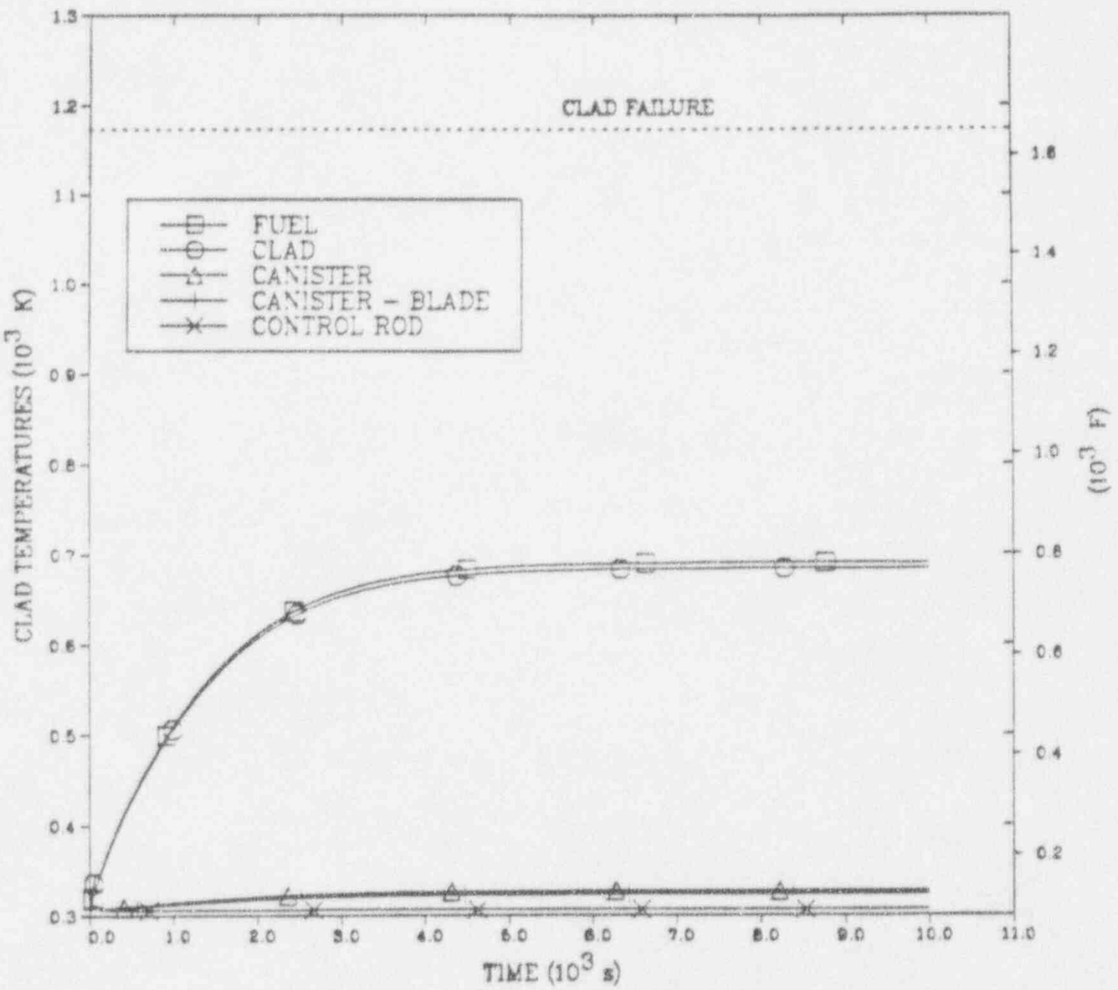

Figure 19: Cell 217 Temperatures

NUREG/CR-6144
BNL-NUREG-52399
Vol. 6, Part 1

EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY UNIT-1

Evaluation of Severe Accident Risks During Mid-loop Operations

Main Report

Draft Completed: June 1994

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NOTICE

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ABSTRACT

Traditionally, probabilistic risk assessments (PRA) of severe accidents in nuclear power plants have considered initiating events potentially occurring only during full power operation. Some previous screening analysis that were performed for other modes of operation suggested that risks during those modes were small relative to full power operation. However, more recent studies and operational experience have implied that accidents during low power and shutdown could be significant contributors to risk.

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects being performed by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL). Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied.

The objectives of the program are to assess the risks of severe accidents initiated during plant operational states other than full power operation and to compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150. The scope of the program includes that of a level-3 PRA.

A phased approach was used in the level-1 program. In phase 1 which was completed in Fall 1991, a coarse screening analysis including internal fire and flood was performed for all plant operational states (POSs). The objective of the phase 1 study was to identify potential vulnerable plant configurations, to characterize (on a high, medium, or low basis) the potential core damage accident scenarios, and to provide a foundation for a detailed phase 2 analysis.

In phase 2, mid-loop operation was selected as the plant configuration to be analyzed based on the results of the phase 1 study. The objective of the phase 2 study is to perform a detailed analysis of the potential accident scenarios that may occur during mid-loop operation, and compare the results with those of NUREG-1150. Volume 1 summarizes the results of the study. The scope of the level-1 study includes plant damage state analyses, and uncertainty analysis. The internal event analysis is documented in Volume 2. The internal fire and internal flood analysis are documented in Volumes 3 and 4 respectively. A separate study on seismic analysis, documented in Volume 5, was performed for the NRC by Future Resources Associated, Inc.

A phased approach was also used in the level 2/3 program however both phases addressed the risk from only mid-operation. The first phase of the level 2/3 PRA was initiated in late 1991 and consisted of an Abridged Risk Study. This study was completed in May 1992 and was focused on accident progression and consequences, conditional on core damage. Phase 2 is a more detailed study in which an integrated evaluation of risk during mid-loop operation was performed. The results of the phase 2 level 2/3 study are the subject of this volume of NUREG/CR-6144, Volume 6.

The off-site risk estimates for latent health effects of accidents during mid-loop operation were similar to the risk estimates for full power operation. The early health consequences are much lower than the full power results primarily due to the long time after scram when the accidents occur in mid-loop operation (i.e., because of the natural decay of the short-lived isotopes of iodine and tellurium, which are primarily associated with early health effects). The uncertainties in risk for accidents during mid-loop operating are largely due to uncertainties associated with isolating the containment and achieving a pressure retaining capability.

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FOREWORD

(NUREG/CR-6143 and 6144)
Low Power and Shutdown Probabilistic Risk Assessment Program

Traditionally, probabilistic risk assessments (PRA) of severe accidents in nuclear power plants have considered initiating events potentially occurring only during full power operation. Some previous screening analysis that were performed for other modes of operation suggested that risks during those modes were small relative to full power operation. However, more recent studies and operational experience have implied that accidents during low power and shutdown could be significant contributors to risk.

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects performed by Brookhaven National Laboratory(BNL) and Sandia National Laboratories(SNL), with the seismic analysis performed by Future Resources Associates. Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied.

The objectives of the program are to assess the risks of severe accidents due to internal events, internal fires, internal floods, and seismic events initiated during plant operational states other than full power operation and to compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150. The scope of the program includes that of a level-3 PRA.

The results of the program are documented in two reports, NUREG/CR-6143 and 6144. The reports are organized as follows:

For Grand Gulf:

NUREG/CR-6143 - Evaluation of Potential Severe Accidents during Low Power and Shutdown Operations at Grand Gulf, Unit 1

- Volume 1: Summary of Results
- Volume 2: Analysis of Core Damage Frequency from Internal Events for Operational State 5 During a Refueling Outage
  - Part 1: Main Report
    - Part 1A: Sections 1 - 9
    - Part 1B: Section 10
    - Part 1C: Sections 11 - 14
  - Part 2: Internal Events Appendices A to H
  - Part 3: Internal Events Appendices I and J
  - Part 4: Internal Events Appendices K to M
- Volume 3: Analysis of Core Damage Frequency from Internal Fire Events for Plant Operational State 5 During a Refueling Outage
- Volume 4: Analysis of Core Damage Frequency from Internal Flooding Events for Plant Operational State 5 During a Refueling Outage
- Volume 5: Analysis of Core Damage Frequency from Seismic Events for Plant Operational State 5 During a Refueling Outage

Volume 6: Evaluation of Severe Accident Risks for Plant Operational State 5 During a Refueling Outage
Part 1: Main Report
Part 2: Supporting MELCOR Calculations

For Surry:

NUREG/CR-6144- Evaluation of Potential Severe Accidents during Low Power and Shutdown Operations at Surry Unit-1

Volume 1: Summary of Results

Volume 2: Analysis of Core Damage Frequency from Internal Events during Mid-loop Operations
Part 1: Main Report
Part 1A: Chapters 1 - 6
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Volume 3: Analysis of Core Damage Frequency from Internal Fires during Mid-loop Operations
Part 1: Main Report
Part 2: Appendices

Volume 4: Analysis of Core Damage Frequency from Internal Floods during Mid-loop Operations

Volume 5: Analysis of Core Damage Frequency from Seismic Events during Mid-loop Operations

Volume 6: Evaluation of Severe Accident Risks during Mid-loop Operations
Part 1: Main Report
Part 2: Appendices

EXECUTIVE SUMMARY

S.1 Background

A systematic and integrated evaluation of risk has been performed for mid-loop operation at the Surry Unit 1 plant. Surry is a pressurized water reactor (PWR) with a subatmospheric containment building. The study was performed by Brookhaven National Laboratory (BNL) for the Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES). A sister study of the Grand Gulf nuclear power plant, a boiling water reactor (BWR), is being performed by Sandia National Laboratories.

Probabilistic Risk Assessments (PRAs) for low power and shutdown operations were initiated in support of the NRC's response to the Chernobyl accident, which was an accident initiated at low power conditions, the Diablo Canyon event of April 10, 1987 which led to the issuance of a Generic Letter 88-171 and later modified by the staff's follow-up actions to the incident at the Vogtle plant on March 20, 1990. An analysis of the core damage frequency (Level 1 PRA) for low power shutdown operation at Surry was initiated in late 1990 and carried out in two phases. Phase 1 undertook a coarse qualitative screening analysis of the accident sequences leading to core damage for all plant operational states during low power and shutdown, while in Phase 2 a detailed quantitative analysis of the core damage frequency was performed for mid-loop operation only.

The accident progression and consequence assessment (Level 2 and 3 PRA) was initiated in late 1991 and was also carried out in two phases. An Abridged Risk study was performed from January to May 1992. It was focused on accident progression and consequences, conditional on core damage. Phase 2 is a more detailed study in which the accident frequency analysis was combined with the accident progression and consequence analysis to calculate risk. This Phase 2 study is the subject of this volume of the report. The analysis of core damage frequency for accidents initiated by internal events, internal fire, internal flood, and seismic events are reported in separate volumes.

The objective of the Phase 2 study is to develop methods to compute the risk of the Surry plant during mid-loop operation and to perform the study. The approach used in the risk assessment was to utilize to the extent possible the component analyses developed as part of the NUREG-1150 program. The assessment also identified those factors that have the most impact on the risk estimates and highlights unique features of the risk analysis performed. The results of the study were also compared against the risk of full power operation as evaluated in the NUREG-1150 study of Surry and the NRC safety goals.

Mid-loop operation occurs when the reactor coolant system (RCS) level is lowered to the mid-plane of the hot leg. This allows the steam generators to be drained so that they can be tested. Mid-loop operation can occur during different types of outage and when the plant is in several different operational states. At Surry, mid-loop operation has occurred in four types of outages: refueling, drained maintenance, non-drained maintenance with the use of the residual heat removal (RHR) system, and non-drained maintenance without the use of the RHR system. Each outage type is characterized by several operating states with each state representing a unique set of operating conditions (temperature, pressure, configuration). Three mid-loop operating states were identified from Surry outage records, two during refueling outages and one in drained maintenance. Each of these operating states is characterized by different decay heat levels and plant configurations, such as number of RCS loops that are isolated and whether the safety/relief valves on the pressurizer have been pulled for maintenance.

The scope of this study was to perform an integrated risk analysis for mid-loop operation during three plant operating states. Risk estimates were made for accidents initiated by internal events due to equipment failure and human error. Risk estimates were however not made for accidents initiated by internal fires, internal floods or seismic events. In addition, as this study is limited to accidents at mid-loop operation it is not a complete risk estimate for accidents that could occur during low-power and shutdown. In fact, mid-loop operation was selected for more detailed study because in the Phase 1 study this was found to be one of the more vulnerable plant configurations. The current risk estimates for mid-loop operation are therefore likely to be higher than for other plant configurations during low-power and shutdown. Another related point deals with the impact of this study on plant operations during mid-loop conditions at Surry. The study has identified potential vulnerabilities over the last few years and the plant staff have responded (if they found that a response was warranted) by making improvements. While these responses are encouraging and lead to improved plant safety it has meant that we have been trying to analyze a moving target. In order to complete the study we therefore had to use procedures and other plant information available as of April 1993.

S.2 Method

The approach used in the risk assessment was to utilize to the extent possible the component analyses that were developed for the full power study. However, due to the long time periods over which an accident can occur and due to differences in plant configuration during mid-loop operation, the interface between the core damage frequency analysis and the accident progression analysis was sufficiently different that additional factors had to be incorporated in combining the accident sequences into appropriate plant damage states for entry into the accident progression event tree (APET). It was also possible to simplify (i.e., reduce the number of top events) the full power APET for use in the mid-loop study. In addition to reducing the size of the tree it was necessary to introduce a number of new top events related to containment isolation in order to appropriately describe the accident progression and plant configuration during mid-loop operation.

The source term model used in the full power study was considered suitable for use in the mid-loop study with only minor modifications. This suitability was based on comparisons with point calculations from a deterministic code, MELCOR, and the views expressed by an expert review panel drawn from staff at Sandia and Brookhaven National Laboratories. However the partitioning method used in the full power study to combine the source terms into a smaller number of representative source terms for input to the consequence model had to be modified. This was necessary in order to account for changing radionuclide inventories for the various accidents because they can occur during a long time period after shutdown. The latest version of the MACCS code was used to evaluate the offsite consequence measures. In addition, simple, scoping estimates of onsite doses in the open area of the plant adjacent to the containment (so-called parking lot dose) were also made. The method used to integrate risk was the same as that used in the full power study.

S.3 Results

Figure S.1 presents four statistical measures of the distributions of the major contributors (plant damage states) to the core damage frequency for accidents during mid-loop operation obtained from this study. Similar statistical measures for full power operation obtained from the NUREG-1150 study of Surry are also included in the figure. Figures S.2 and S.3 presented similar information for early cancer fatality risk and the population dose. Population dose is included in the figures rather than latent cancer fatalities to facilitate comparison with the NUREG-1150 results. The mid-loop study used the latest version of the MACCS code, which incorporates the BEIR V update to the latent cancer versus dose relationship, whereas NUREG-1150 used an older version of MACCS. The latest BEIR V update gives approximately a factor of three higher latent cancers for the same value of population dose.

From an inspection of Figure S.1 it is apparent that the mean core damage frequency of accidents initiated by internal events during mid-loop operation was calculated to be about an order of magnitude lower than the mean frequency of accidents during full power operation. In addition the mean and median frequencies of the two distributions were within a factor of approximately two which indicates that the means were not strongly influenced by the tails of the distribution. However the tails of the distributions do overlap and therefore for some cases the mid-loop core damage frequency could be higher than the full power frequency.

Figure S.2 indicates that the mean risk of offsite early health effects is over two orders of magnitude lower for accidents during mid-loop operation than for full power. This is due to the natural decay of the radionuclide inventory (because the accidents occur a long time after shutdown) particularly the short-lived isotopes of iodine and tellurium, which are primarily associated with early health effects. The distributions obtained for long-term health effects (measured by population dose in Figure S.3) for mid-loop and full power operation appear to be very similar. The reason why the population dose distributions are similar but the core damage frequency distributions are an order of magnitude lower for mid-loop operation is explored in the following paragraphs.

Accident sequences in which the operators did not correctly diagnose the situation or take proper actions (plant damage state 2 in Figure S.1) were the largest contributor to the total core damage frequency distribution for mid-loop operation. Accident sequences that lead to station blackout during mid-loop operation (plant damages states 1 and 4 in Figure S.1) contribute about 10 percent to the mean CDF. Other accidents (plant damage state 3 in Figure S.1) were identified that resulted in loss of core cooling after depletion of the refueling water storage tank and failure of recirculation. The leading cause of recirculation failure was found to be plugging of the suction from the sump. These accidents contribute about 20 percent to the mean core damage frequency.

From an inspection of Figure S.3 it is clear that plant damage state 2 is almost equivalent to the total risk distribution for the population dose. The distributions for PDS3 and PDS4 are almost entirely below the distribution for PDS2. The distributions for PDS3 and PDS4 consist of very low consequence estimates and do not impact the total risk distribution. This is because it was determined in the accident progression analysis for PDS2 that if operator error due to failure to diagnose the accident led to core damage then it was unlikely that the operators would have taken measures to isolate containment. The probability of the containment being open therefore, was very high for accident sequences in plant damage state 2. The probability of the containment not being isolated was found to be lower for the other plant damage states and thus their relative contribution to the offsite health effects was smaller. For example, while plant damage state 4 (recirculation failure due mainly to sump plugging) contributed almost 20% to the mean core damage frequency its contribution to the mean population dose was much smaller. This is because due to the recognition of the problem by the operators and the long times involved, the operators were assumed to have a high probability of being able to isolate the containment and the probability of a large source term from this type of accident was calculated to be small.

In summary, accident sequences involving human error were the largest contributors to the core damage frequency during mid-loop operation and even larger contributors to the offsite risk estimates because it was determined that during these sequences the operators would be unlikely to achieve containment isolation. Therefore, during mid-loop operation the probability of loss of containment integrity conditional on core damage was assessed to be high.

In comparison, in the full power study accident sequences that lead to station blackout were the largest contributors to core damage frequency but not to the offsite risk estimates. This is because containment performance at Surry was found to be very good for this class of accidents even if the molten core penetrates the lower head of the reactor vessel. Therefore accidents with lower frequencies but higher source terms which bypassed the containment, such as interfacing system loss of coolant accidents (ISLOCAs) and steam generator tube ruptures (SGTRs) were found to be the largest contributors to mean risk estimates in the full power study. Thus the loss of containment integrity conditional on core damage was determined to be small for severe accidents at full power.

Finally, the scoping estimates of onsite doses indicate that the parking lot dose rates for accidents involving unisolated containment were high. This would limit the ability to take corrective actions, which cannot be performed from the control room, for this class of accidents.

S.4 Conclusions

The main finding of the study is that during mid-loop operation the risk of consequence measures related to long-term health effects, latent cancer fatalities and population dose, are high, comparable to those at full power, despite the much lower level of the decay heat and the radionuclide inventory. The reason for this is that containment is likely to be unisolated for a significant fraction of the accidents initiated during mid-loop operation so the releases to the environment are potentially large and the radionuclide species which mostly contribute to long-term health effects (such as cesium) have long half-lives. Accident sequences involving failure to correctly diagnose the situation or take proper actions are the largest contributors to the integrated risk. Another finding of the study is that the risk of early fatalities is low despite the unisolated containment due to the decay of the short-lived radionuclide species such as iodine and tellurium which contribute to early fatality risk. The integrated risk estimates have a range of uncertainty extending over approximately two orders of magnitude from the 5th to the 95th percentile of the distribution.

Containment Status

The major factor driving the risk is the status of containment during mid-loop operation. It was determined that there is a high probability that the containment is either unisolated or that it would not have full pressure retaining capability during mid-loop operation. This is particularly the case if the operators fail to diagnose the accident as it was judged unlikely that they would take action to isolate containment or could succeed in doing so within the available time frame. This factor played a significant role in influencing the risk estimates of mid-loop operation. During the course of the study, Surry plant personnel made available new procedures for containment closure during mid-loop operation. However, it was difficult to assess the adequacy of these procedures in ensuring the pressure retaining capability of the containment within the time frame encompassed by this study. This feature contributed significantly to the uncertainty in containment status and the estimate of risk.

Availability of Containment Sprays

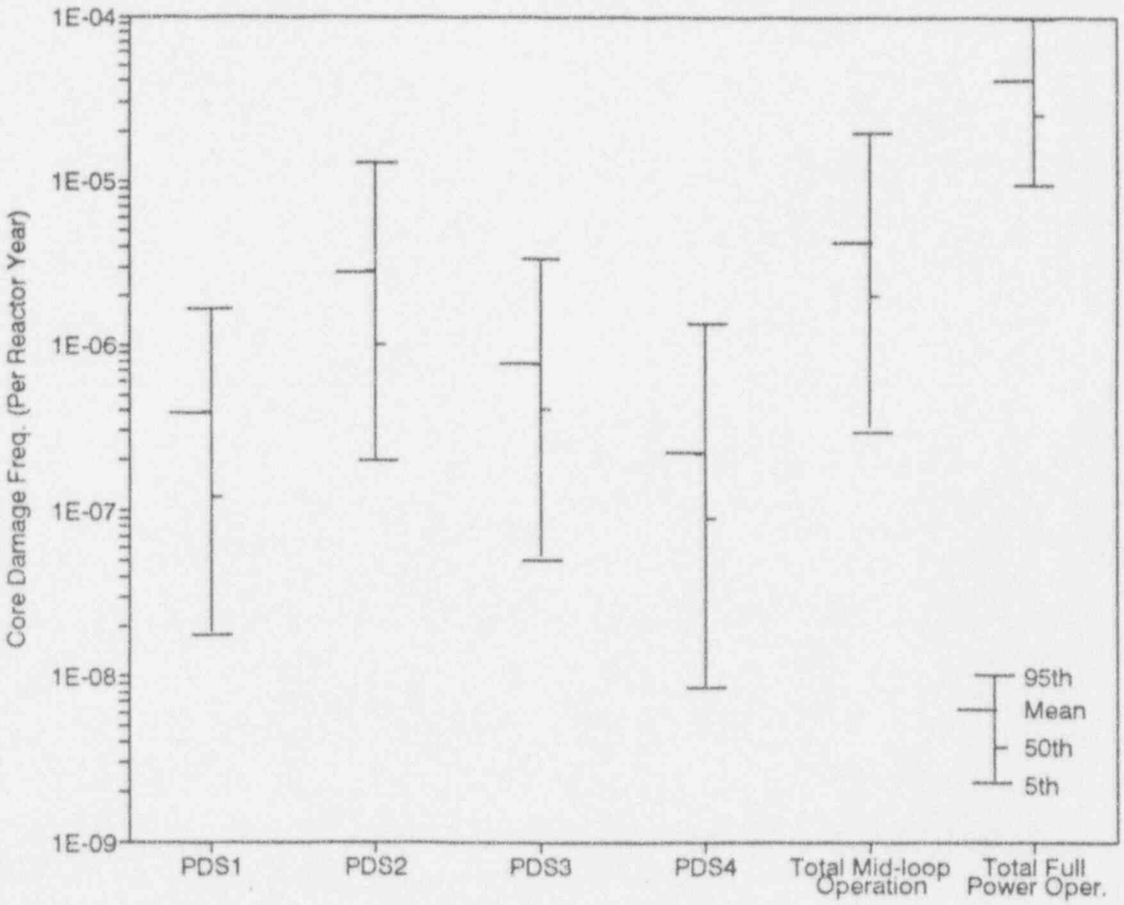
There is no requirement at Surry for the containment sprays to be available during shutdown. Plant records show that the spray systems could be inoperable because of maintenance. Spray availability was modeled as an uncertainty parameter in the integrated risk analysis. Since the sprays perform an important safety function in mitigating the effects of releases, spray unavailability contributed both to the risk and its uncertainty.

Possibility of Core Damage Arrest

The inclusion of the possibility of arresting the core degradation process before vessel failure is an important feature of this analysis as it was for the full power study. Termination of the accident in-vessel can significantly reduce some of the fission product releases and thus the risk. The potential for core recovery depends on the nature of the accident progression and is different for the various PDS Groups. Overall, the conditional probability of core damage arrest ranged from 0.23 (5th percentile) to 0.44 (95th percentile) with a mean of 0.35.

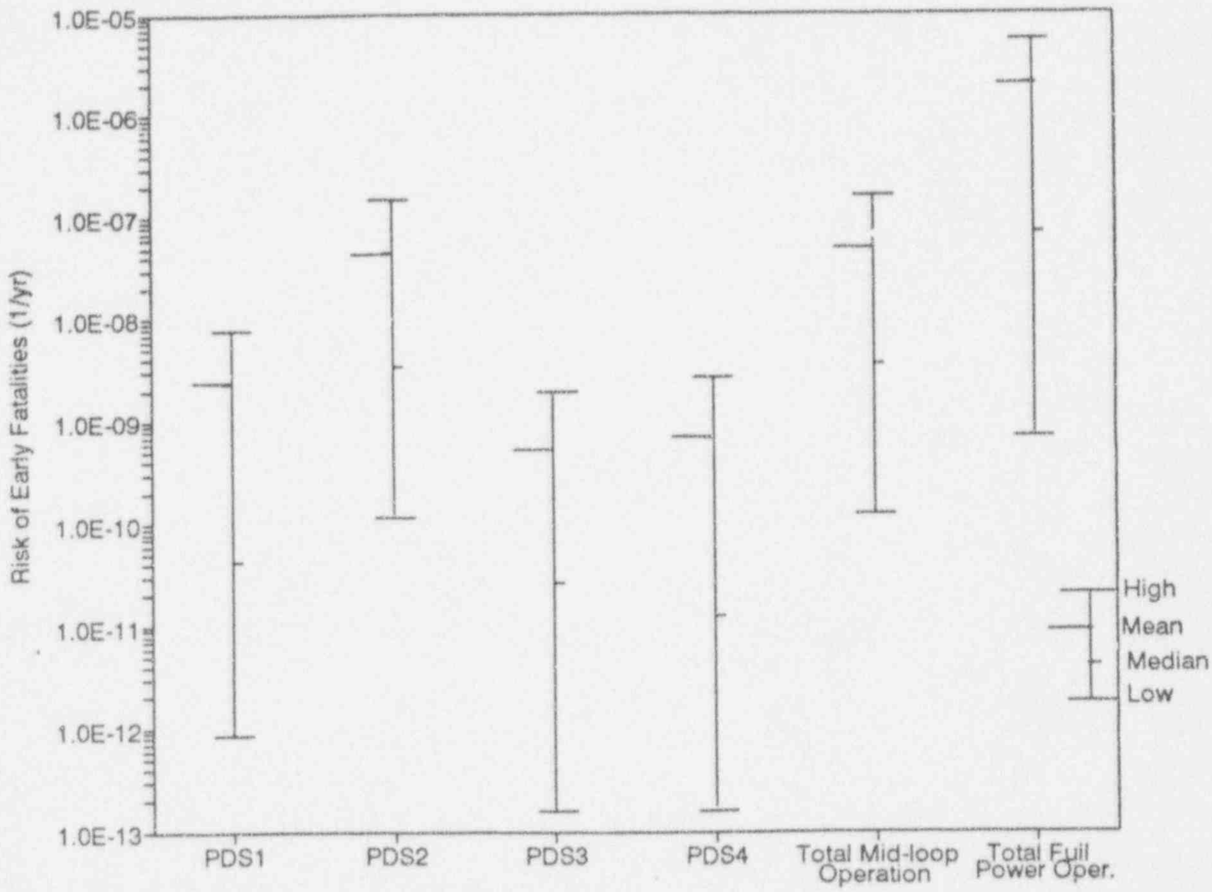
Comparison with Full Power Study

The mean core damage frequency for accidents during mid-loop operation is about an order of magnitude lower than the mean frequency of accidents caused by internal events at full power. However, the risk distributions obtained for comparable long term health consequences (measured by the population dose at 50 miles) are very similar in the two studies. What this finding implies is that the lower decay heat and lower radionuclide inventory of the mid-loop operating state, compared with full power, is offset by the likelihood of containment being unisolated. Finally, the mean risk of early health effects is over two orders of magnitude lower for accidents during mid-loop operation than for accidents during full power operation. This is due to the natural decay of those radionuclide species which have the greatest impact on early fatality risk because accidents during mid-loop operation occur a long time after shutdown.



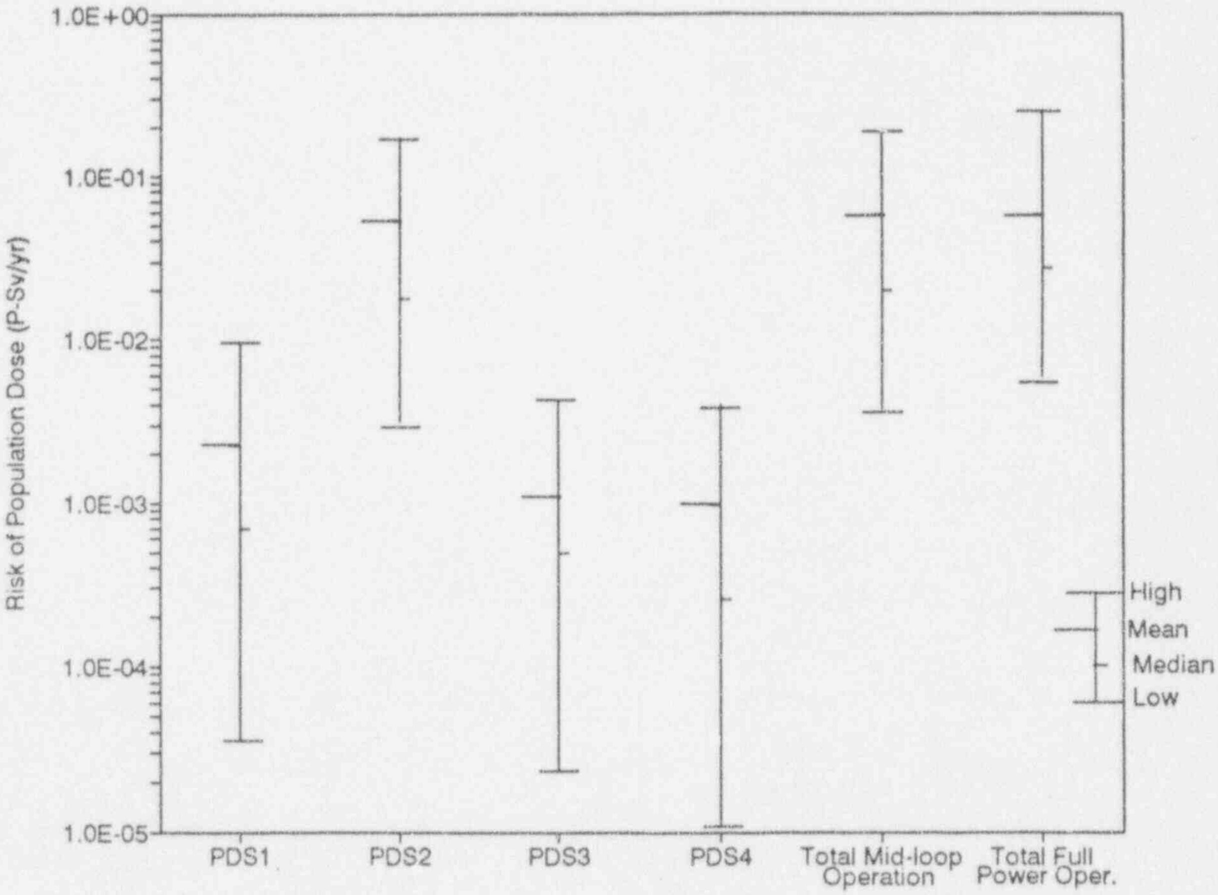
1. Station Blackout (SBO)
2. Human Error
3. Recirculation Failure
4. Loss of 4 kV bus (similar to SBO)

Figure S.1. Distribution for Core Damage Frequency for the Plant Damage States



1. **Station Blackout (SBO)**
2. **Human Error**
3. **Recirculation Failure**
4. **Loss of 4 kV bus (similar to SBO)**

Figure S.2. Distribution of Risk of Early Fatality for the Plant Damage States



1. Station Blackout (SBO)
2. Human Error
3. Recirculation Failure
4. Loss of 4 kV bus (similar to SBO)

Figure S.3. Distribution for Risk of Population Dose to 50 Miles for the Plant Damage States

1 INTRODUCTION

1.1 Background

This report presents the results of a Level 2 (accident progression) and Level 3 (consequence) analysis of the Surry nuclear power plant for possible accidents initiated while the plant is in mid-loop operation. The analysis was performed by Brookhaven National Laboratory (BNL) for the Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research (RES). This Level 2/3 analysis was combined with an analysis[1] of the core damage frequency (Level 1) for accidents initiated by internal events to produce an integrated risk assessment. A sister study of the Grand Gulf nuclear power plant, a boiling water reactor (BWR), is being performed by Sandia National Laboratories.

Probabilistic Risk Assessments (PRAs) for low power and shutdown operations were initiated in support of the NRC's response to the Chernobyl accident, which was an accident initiated at low power conditions, and later modified by the staff's follow-up actions to the incident at the Vogtle plant on March 20, 1990. The Level 1 PRA of Surry during low power and shutdown operation was initiated in late 1990 and carried out in two phases. Phase 1 undertook a coarse qualitative screening analysis of the accident sequences leading to core damage for all plant operational states during low power and shutdown, while in Phase 2 a detailed quantitative analysis of the core damage frequency was performed for mid-loop operation only. The Level 2 and 3 PRA was initiated in late 1991 and has also been carried out in two phases. In Phase 1 an Abridged Risk study was performed from January to May 1992. It was focused on accident progression and consequences, conditional on core damage. A summary of the Abridged study is contained in Chapter 2 of this report. Phase 2 is a more detailed study in which the accident frequency analysis was combined with the accident progression, source term and consequence analysis to calculate risk. This Phase 2 study is the subject of this volume of the report. The analysis of core damage frequency for accidents initiated by internal events, internal fires, internal floods, and seismic events are reported in separate volumes.

1.2 Study Objectives

The objective of this study is to develop methods to compute the risk of the Surry plant during mid-loop operation and to perform the study. The approach used in the risk assessment was to utilize to the extent possible the component analyses developed as part of the NUREG-1150 program.[2] The assessment also identified those factors that have the most impact on the risk estimates and highlights unique features of the risk analysis performed. Finally the results of the study were compared against the risk of full power operation as evaluated in the NUREG-1150 study of Surry and the NRC safety goals.

1.3 Scope of Study and Major Assumptions

The analysis reported in this volume is an integrated risk analysis of mid-loop operation at the Surry Unit 1 power plant, when the reactor coolant system level is lowered to the mid-plane of the hot leg. Mid-loop operation can occur in several plant operational states (POSs) of different outage types. At Surry, four types of outage: refueling, drained maintenance, non-drained maintenance with the use of the residual heat removal (RHR) system, and non-drained maintenance without the use of the RHR system, were defined in the Level 1 analysis. Each outage type is characterized by several POSs with each POS representing a unique set of operating conditions (temperature, pressure, configuration). Three mid-loop POSs were identified from Surry outage records in the Level 1 analysis, POS 6 and POS 10 in refueling outage, and POS 10 in drained maintenance. Each of these POSs is characterized by different decay heat levels and plant configurations, such as number of reactor coolant system (RCS) loops that are isolated and whether the safety/relief valves on the pressurizer have been pulled for maintenance.

The scope of the study is limited to an analysis of accidents initiated by internal events due to equipment failure and human error during mid-loop operation. In addition the study reflects procedures and other plant information available prior to April 1993. Risk estimates were not made for accidents initiated by internal fires, internal floods, and seismic events. As this study is limited to accidents during mid-loop operation it is not a complete risk estimate for accidents that could occur during low power and shut down operation. Mid-loop operation was selected for detailed study because the screening analysis carried out in Phase 1 indicated that this was one of the more vulnerable plant configurations during low power and shutdown. Therefore, the risk estimates for mid-loop operation are likely to be higher than for other plant configurations at low power and shutdown.

The major assumptions of the analysis are as follows:

(1) Discretization of time windows and decay heat levels: Decay heat level is a key parameter in the accident analysis due to the long time interval, depending on the POS and outage type, over which an accident can potentially occur during mid-loop operation. Four time windows with corresponding decay heat levels were constructed in the Level 1 analysis and it was assumed that the decay heat level (which varies continuously) of each time window can be adequately represented by its value at the mid-point of the time window.

(2) Containment Status: Several assumptions had to be made on the status of the containment during mid-loop operation. These assumptions are documented in more detail in Chapter 4 and relate to the pressure capability of the containment. The pressure capability ranges from no pressure retaining capability (leakage at inception of release) to full design capability (as at full power operation). The ability of containment to retain fission products released from damaged fuel is the dominant factor affecting risk.

(3) Source Term: It was assumed that the source term code, SURSOR[3], which was developed for the full power study would adequately apply to low power and shutdown conditions as well. This assumption was checked through spot comparisons with calculations based on the mechanistic code MELCOR[4] and by a review performed by a Source Term Advisory Group comprised of BNL and SNL staff.

(4) Accident Progression: Assumptions were made in various parts of the accident progression event tree on branch point probabilities, split fractions, etc. These are documented in Chapter 6.

(5) Consequences: The consequence calculation assumed the same emergency response for the offsite population in the low population zone surrounding Surry and the same long-term protective actions as the NUREG-1150 study.

(6) Onsite Doses: The scoping calculation of onsite doses assumed that the releases were directly from the containment to the environment through the equipment hatch and not through the personnel hatch so no in-building doses were calculated.

(7) Human Reliability: Several assumptions have been made regarding human errors, including failure to diagnose or failure to take action, which play a large role in both the Level 1 and Level 2 analyses. These assumptions are documented in Volume 2 report and in Chapters 5 and 6 of this volume.

1.4 Strengths and Limitations

The strengths of this study are:

(1) It is a systematic and integrated evaluation of risk during mid-loop operation at the Surry Unit 1 plant, including accident frequency, accident progression, source term and consequence analysis with a determination of uncertainty in each of the component analyses and in the final risk measures.

(2) The integrated analysis takes into account the long time after shutdown that the accidents can occur and the impact of the consequent decay in power level and radionuclide inventory on the risk. In particular, new latent cancer weights were derived for source term partitioning and used in the consequence calculation.

(3) The newest version of the MELCOR code, version 1.8.2, was used to calculate the timing of key events in the accident progression which were then used in the accident progression event tree.

(4) The accident progression event tree has sufficient detail to account for a significant portion of the likely paths of evolution of the accident.

(5) The study includes a scoping calculation of onsite dose rates at locations in the vicinity of the plant during the accident.

The study has the following main limitations:

(1) There was no formal expert elicitation process used, as in the NUREG-1150 study, to provide values and distributions for key variables in the accident progression. Assignments for these variables had to be made internally at BNL or derived from analogy with full power conditions. The selection of the key variables themselves was also made internally at BNL. Thus the uncertainty analysis is not as robust as it could have been with input from an expert panel.

(2) The scope of the study is limited to accidents initiated by internal events (due to equipment failure or human error) during mid-loop operation. Risk estimates were not made for accidents initiated by internal fires, internal floods and seismic events. The final risk numbers should therefore not be interpreted to reflect the risk of all plant operational states during low power and shutdown operation.

1.5 Organization of This Report

This report is published in six volumes as described briefly in the Foreword. The first volume of NUREG/CR-6144 provides a summary of the results of the full scope PRA (levels 1, 2, and 3) that has been performed for the Surry plant for severe accidents that might occur during mid-loop operation.

Volumes 2 through 6 present a detailed description of the results of the constituent analyses. Volume 2 describes the analysis of the core damage frequency (CDF) from internal events initiated during mid-loop operation. An analysis of the CDF from internal fire and internal floods is presented in Volumes 3 and 4, respectively. The CDF from seismic events is addressed in Volume 5.

This volume of NUREG/CR-6144, Volume 6, presents the risk results for accidents during mid-loop operation at Surry. Part 1 of this volume presents the analysis and the results in some detail; Part 2 consists of appendices which contain further detail. Following a summary and introduction, Chapter 2 of this volume presents the results of Phase 1 of this study which consisted of an abridged risk study.

The rest of the chapters describe the more detailed Phase 2 study. Chapter 3 briefly describes the methods used in the study. A description of the plant is given in Chapter 4. The interface between the level 1 and level 2 analyses is described in Chapter 5. Chapter 6 presents the results of the accident progression analysis. Chapter 7 presents the results of the source term analysis, and Chapter 8 gives the results of the consequence analysis. Chapter 9 summarizes the risk results, including the contributors to uncertainty. The results for mid-loop operation are compared with the full power results in Chapter 10. Remaining open issues are addressed in Chapter 11, and finally Chapter 12 presents the conclusions drawn from the study.

1.6 References

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2. U. S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, Vols. 1–3, December 1990 - January 1991.

3. Jow, H. N., W. B. Murfin and J. D. Johnson, "XSOR Codes User's Manual," NUREG/CR-5360, Sandia National Laboratories, December 1989.

4. Summers, R. M., et. al., "MELCOR 1.8.0: A Computer Code for Nuclear Reactor Severe Accident Source Term and Risk Assessment Analyses," NUREG/CR-5531, Sandia National Laboratories, SAND90-0364, January 1991.

2 SUMMARY OF ABRIDGED STUDY ON RISK DURING MID-LOOP OPERATION

2.1 Background and Objectives

The abridged risk study was conducted from January through April 1992. The objective of the analysis was to make a preliminary determination of risk of the accident progression and the consequences of accidents during mid-loop operation at the Surry plant, Unit 1. The study was designed to obtain results for regulatory decisions that were to be made in the early summer of 1992.

The abridged risk study was carried out to compute estimates of the conditional consequences (probability of the various events during the accident progressions multiplied by the consequences), given the occurrence of core damage. Traditional risk estimates, computed by multiplying the conditional consequences and the frequency of the sequences leading up to core damage, could not be made because the frequencies had yet to be determined in companion Level 1 and human reliability analysis (HRA) studies. Uncertainty was taken into account in a manner consistent with the detail of the abridged study.

2.2 Methodology

The methodology of the abridged study was an abbreviated version of the NUREG-1150[1] study. The calculations began with the assumption that core damage had occurred, making the consequences conditional. Given core damage, the possible accident progressions were delineated with a simple accident progression event tree (APET) limited to nine top event questions. The timing of key events in the accident progression was based on deterministic calculations with the MELCOR[2] code. The likelihood of the various accident progressions is reflected by branch point probabilities in the APET. In large-scale risk studies, such as NUREG-1150, the assignment of such probabilities is made by a formal expert elicitation process; in the abridged study, because of resource limitations, these assignments were made by the BNL staff. Thus, the probabilities are not as rigorous as they could be; this is one of several limitations of the study. This lack of rigor was partially offset by repeating the calculations with other reasonable input values; together, these repeated calculations constitute an uncertainty analysis.

Through the uncertainty analysis, distributions, instead of point values, were assigned to the branch points. The distributions are subjective, but account for many possible values of the branch points. Point values were selected from the distributions with a form of Monte Carlo sampling known as Latin Hypercube Sampling (LHS).[3] After making sets of inputs, each set is assigned to the branch points and multiplied through to the ends of the APET. The calculations were repeated using the sets of inputs to build a probability distribution at the end of each pathway.

Having delineated accident progressions with the APET, the source terms of the progressions were calculated with the parametric SURSOR[4] code developed for the NUREG-1150 program. SURSOR determines source terms from the characteristics of the pathways through the APET and other inputs. As in the APET calculations, distributions are assigned to the variables and sampled with LHS to form many sets of input values for repeated calculations. The result is a distribution of source terms for each accident progression pathway.

Two sets of consequence measures were determined; an onsite dose rate (within the site boundary and designated as a *parking lot* dose rate), and offsite consequences, including early fatalities, population dose, and latent cancers.

- The parking lot dose rate was computed using a recent model by Ramsdell[5] and a combination of the older Wilson[6] and Regulatory Guide 1.145 models.[7] (Dose rates inside the containment or the reactor building were not calculated because the releases were assumed to take place through the equipment hatch directly to the outside).

- Offsite consequences were computed using the MACCS code.[8] Uncertainty was not propagated through the consequences as it was through the APET and the source term calculations.

Conditional risk was computed for each accident progression pathway by multiplying the consequences by their associated probabilities determined from the APET. The products of the pathways were summed. This process was repeated for each Monte Carlo sample of the source terms. Then, high, medium, and low results were reported. In the NUREG-1150 study,[1] high, medium and low results are represented by the 95th percentile, median and 5th percentile values of the distribution of the results. However, in this study, the number of samples taken was not sufficient enough to define them statistically. Therefore, they were referred to as high, medium and low results. They would approach the statistical 95th, 50th or 5th percentiles, if sufficient numbers of samples were taken.

2.3 Accident Progression Analysis

The abridged analysis was based on a preliminary screening analysis of the systems reliability and characterization of the accident sequences leading to core damage for the internally initiated events at the Surry Unit 1 plant.[9] From this coarse screening analysis[9] mid-loop operation, which can occur during drained maintenance or refueling outages, was determined to be one of the most vulnerable plant conditions, mainly due to the reduced inventory in the RCS. The dominant causes of accidents during mid-loop operation are loss of residual heat removal (RHR) and loss of offsite power. Loss of RHR accident sequences occur largely due to operator errors, such as failure to diagnose an accident or failure to take proper action. Operating experience at nuclear power plants indicate a relatively high incidence of loss of RHR. For this category of accidents, the recovery probability is largely determined by the human reliability analysis (HRA). Since the HRA results have a large band of uncertainty, it also was included as an uncertainty parameter. For accidents initiated by a loss of offsite power, the probability of recovering offsite power during the accident progression determines the probability of recovering core cooling capability, and terminating the accident.

For the abridged analysis, it was assumed that all the reactor loops were isolated and the safety/relief valves were removed for maintenance, which provides a vent path from the RCS to the containment.

The time to enter mid-loop after shutdown and the duration of mid-loop operation vary widely; from one day to more than one month. These times were selected as an uncertainty parameter to be varied in the sampling process. Longer times after shutdown are characterized by a lower decay heat level which potentially increases the time available to take actions to recover core cooling capability before the core is uncovered, and also reduces the inventory of fission products available for release.

To determine the extent of detail needed for the APET, extensive use was made of the accident progression analysis for the Surry plant carried out for the NUREG-1150 program,[1] which was a PRA of the plant at full power. That study showed that the major cause of release at Surry was containment bypass, followed by basemat melt-through. The probability of early failure of the containment caused by various mechanisms such as a hydrogen or steam explosion and late failure resulting from gradual pressurization was either very small or negligible. Thus, once the containment boundary is closed, the containment retains the fission products most of the time (except by very late basemat melt-through). In other words, phenomena such as direct containment heating or steam explosions were not important contributors to the estimated probability of containment failure and the eventual release of fission products. For accidents during low power and shutdown operation where the decay heat is significantly less and the reactor pressure is generally low, there are no particular reasons to believe that the performance of a closed containment would be any worse than for accidents occurring at full power. Two important factors for determining the containment's response during an accident in mid-loop operation are the status of its integrity and the availability of containment sprays.

From several discussions with the Surry personnel, it was learned that while the containment is considered "closed" during mid-loop operation at Surry, closure does not ensure that the containment can contain the pressure which could be generated during a severe accident and prevent release of fission products.[10] This is due primarily to the presence of a temporary restraining plug, that has no overpressure capability, in place of the escape tunnel in the containment equipment hatch. Therefore, in the abridged study, no credit was given to the containment barrier; it was assumed that the fission products would leak to the environment once they were released to the containment. This aspect simplified the abridged APET. Because the integrity of the containment is already lost at accident initiation, many questions normally needed to assess the potential for containment failure are no longer relevant.

Sprays are important because they are the major containment cooling system during severe accidents, and can reduce the source terms by scrubbing. There is no requirement under the existing technical specifications to have any of the containment sprays available once the plant enters the RHR entry condition at Surry.[10] Consequently, all of the containment sprays could be out of service during mid-loop operation. Spray availability was used as one of the uncertainty parameters in this study.

Figure 2.1 shows the APET used in the abridged study. The first three questions refer to the status of containment. In the abridged study, the containment was assumed to be leaking from the start of the accident. Once the status of the containment is identified, the fourth question asked is the timing of core-cooling recovery, which determines the extent of core damage. Arrest of core degradation before failure of the vessel during a severe accident could significantly decrease the magnitude of release of fission products. Therefore, the timing of recovery of core-cooling capability was divided into five periods; Very early, Early, Intermediate, Late, and Never. The timing of *Very early* extends to the point where core cooling is recovered without any core damage. *Early* is recovery of cooling during the relatively short period after the cladding rupture of the fuel rods, but before significant core melting. *Intermediate* is the period in which the recovery of core cooling will arrest the progress of core melt without breaching the vessel. From consultation with the Source Term Advisory Group, this intermediate period was assumed to extend until 45% core melting occurred. If core cooling is recovered during the *Late* period (which, here, is defined to be more than 45% of the core melted), the vessel is assumed to be breached by the core debris. *Never* indicates no core cooling recovery.

The time of accident initiation varies widely. Therefore, the BNL staff determined the time of [illegible] initiation by sampling from the joint distributions of the time to enter the mid-loop operation and the duration of mid-loop operation for each observation. Data from the Surry plant, which were collected for the screening level 1 analysis,[9] were used to determine the distribution of the time of accident initiation; the MELCOR-calculated timing of the core-melt progression was adjusted by the decay heat to determine the time available for recovery of core cooling for the accident sequences whose times of accident initiation were different from those selected for the MELCOR calculations. The recovery probability was based on the HRA recovery curve for human error,[11] the offsite power recovery curve,[9] and hardware availability for each of the time periods (the latter from data used in the screening level 1 study).

The next three questions in the APET address spray availability and whether the cavity is dry or wet, which determines the extent of core-concrete interaction. Spray availability was included as an uncertainty parameter. Because the containment during mid-loop operation at Surry was assumed to have no pressure-holding capability, the branches related to *Closed* and *Open* containment were not developed further in this study. This APET was applied to each of the major cutsets leading to core damage sequences identified in the preliminary screening level 1 study in which core damage was defined to have occurred when the coolant level dropped to the top of active fuel.

The outcomes of the accident sequences in the APET were classified into eight bins (including a *No Release* bin) depending on the extent of core damage, vessel breach and spray availability (Fig. 2.1). In estimating the final risks conditional on core damage, only accident sequences which were actually predicted to result in core damage were included; accident sequences which were terminated in the *Very early* period were not included in the calculations of conditional risk. The conditional probability of arrest of core damage before vessel breach for the abridged analysis was estimated to vary from a high of about 0.75 to a low of about 0.4 with both the mean and median being approximately 0.55. The corresponding conditional probability at full power estimated in the NUREG-1150 study of Surry ranged from a high (95th percentile) of 0.7 to a low (5th percentile) of about 0.2 with a mean of 0.5 and a median of 0.45.

2.4 Source Term Analysis

The parametric code, SURSOR,[4] that was developed in NUREG-1150 for Surry, was used to define source terms in the abridged study. Two measures were taken to assure the adequacy of the source terms: The first involved comparing the calculations from MELCOR with the data used in and results from SURSOR. Second, a Source Term Advisory Group was established to provide guidance, and any additional information on modifying the SURSOR code for the present study.

Considering the differences between full power and shutdown operations, the Source Term Advisory Group identified two parameters in SURSOR as important and possibly different than the values used in NUREG-1150. The first parameter is the fraction of the fission products in the core that are released to the vessel before vessel breach. The second parameter is the fraction of the fission products released to the vessel that are subsequently released to the containment. The distributions of these two parameters as defined in NUREG-1150 were compared with MELCOR calculations to establish the values to be used.

SURSOR was used to predict the fission product release fractions for the five accident progression bins (APBs), APB-4 through APB-8, presented in Figure 2.1. Source terms for APB-1 through APB-3 were not considered because they were not expected to lead to any significant offsite consequences. (Due to early recovery, there is no core damage for APB-1 and only clad damage for APB-2 and APB-3). Two hundred sets (or observations) of release fractions were produced for each of the five bins to address source term uncertainty.

To limit the number of offsite consequence calculations, but still provide a range of uncertainty, 19 source term samples were randomly selected (from the 200 source term samples, using the LHS sampling method) for each of the five APBs for offsite consequence calculations.* When combined with the two time parameters, associated with the duration of mid-loop operation during the drained maintenance and the refueling outages respectively, this gave 38 source term samples for each APB.

In addition to release fractions, a complete description of a source term requires specification of the timing, energy, and height of the release. The timing of the release affects both the radioactive decay of the inventory and the warning time for offsite emergency response (e.g., evacuation). The release times and durations were obtained from MELCOR calculations. Since the release time is measured from accident initiation (which in the case of mid-loop operation may be many days after shutdown of the reactor) it is not meant for calculating radioactive decay after reactor shutdown, but can be used to determine the timing for emergency response. The warning time for offsite emergency response is the time at which notification is provided to the public to begin emergency response procedures as measured from the time of accident initiation. In the abridged study, the warning time was assumed to be 60 minutes after accident initiation.

An important parameter in the source term definition for accidents initiated during mid-loop operation, which is not considered in a full power analysis, is the time of accident initiation measured from reactor shutdown. This parameter determines the inventory available for release at accident initiation. The extended time period between accident initiation and reactor shutdown for an accident during mid-loop operation will result in significant radioactive decay, and consequently, a much reduced fission product inventory available for release. Because of its importance to consequences, it was treated as one of the uncertainty parameters in the abridged study. A randomly selected value of the time of accident initiation over the duration of mid-loop operation was assigned to each source term defined in this study.

2.5 Consequence Analysis

2.5.1 Onsite Consequences

The total onsite dose rate is a sum of the inhalation and cloud exposure dose rates based on the radionuclide concentration in the wake region of a building. A scoping value of onsite dose rate was estimated using the following wake centerline concentration models: Ramsdell,[5] Wilson,[6] and Reg. Guide 1.145.[7]

The scoping calculations were performed for three source terms referred to as high, medium, and low (Gap release). The Wilson/Reg. Guide 1.145 labelled box in Figure 2.2 is based on the Reg. Guide 1.145 prediction, limited from above by the values predicted by the Wilson model. The results in Figure 2.3 for the onsite dose rate (Rem/h) indicate a variation of about two orders of magnitude as a function of the source term. The

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* This is the minimum number of source term samples needed to provide a 5% to 95% range for the consequence measures. However, because of the low confidence level associated with such a small sample, they are simply referred to as the upper and lower limits of the calculations, with no percentiles associated with them.

onsite dose rates are high, and are likely to lead to early fatalities for exposed workers. In view of the relatively large number of onsite personnel during shutdown operations, these dose rates outside the containment suggest that a careful examination should be made of onsite evacuation schemes to limit the consequences in the event of an accident.

2.5.2 Offsite Consequences

MACCS[6] calculations of the offsite consequences were made for all the source terms generated by LHS sampling[3] of the SURSOR results. There were nineteen sample groups (one for Drained Maintenance and one for Refueling outages), each containing four distinct sets of release fractions for the nine radionuclide groups represented in the MACCS calculations. The time of release for each group was determined using the LHS technique. The radionuclide core inventories for Surry at various times after shutdown were taken from Reference 12. Then, the initial inventory for each source term was calculated using a logarithmic interpolation between the two closest data points.

The following additional assumptions were used:

- Release power: 1.0 MW (sensitivity calculations with 0.0 MW).
- Release elevation: 28' (8.54 m), the height of the equipment hatch above ground.

Figure 2.3 shows the results for the early and latent fatalities predicted by MACCS. APB-5 through APB-8 contain thirty eight data points each. The median value of early fatalities is shown only for APB-7; zero values were predicted for the remaining bins. The calculations predict the highest number of early fatalities and latent fatalities for APB-7. However, the number of early fatalities, as expected, is very small (a high of less than 1.0 and a mean and median of less than 0.1) even for the most severe accidents (APB-7) involving failure of the vessel's bottom head.

2.6 Conditional Probabilities of Consequences

Once the consequences are calculated for each of the release bins, conditional risks can be evaluated by combining the accident progression analysis, source term analysis, and consequences. If the core damage frequencies of the PDS had been available from the level 1 analysis, absolute integrated risks could have been calculated for this particular POS. However, since they were unavailable, the risks were calculated as conditional on core damage; i.e., the results presented are averaged over various accident progressions, given core damage.

Figure 2.5 shows ranges of four risk measures (conditional on core damage), the early fatalities, late cancer fatalities, the population dose at 50 miles, and the population dose at 1000 miles, calculated for the accidents initiated by internal events during mid-loop operation at Surry. The upper and lower bounds shown in the figures do not represent any particular statistical measures because the number of samples was too small to attach any statistical significance to them. However, if sufficient samples are used, these bounds are expected to asymptotically approach the 5th and 95th percentiles. For comparison, the figure show the same risk measures for full power operation at Surry from the NUREG-1150 study; to make this comparison, the integrated risk results of NUREG-1150 were converted in an approximate fashion to conditional probabilities of the various consequence measures; conditional on core damage and on containment failure. The comparison shows that the conditional probability of early fatalities during mid-loop operation is considerably less than the conditional probability of early fatalities at full power either given core damage or given containment failure. This result is expected since the fission products will have had a longer time to decay and the species which have the greatest influence on the early fatalities generally have short half lives.

Figure 2.5 also shows that the latent cancer fatalities and population doses are higher than those predicted for the full power accidents conditional on core damage. However, these long-term health effects are comparable for accidents conditional on containment failure because these risk measures are more affected by slow-decaying species and the longer decay time has less impact on these species. Therefore, the risks are similar once containment is failed. Since the containment is assumed to be essentially open during mid-loop operation in the abridged study, the offsite risk of latent health effects averaged over core damage sequences is higher for mid-loop operation than for full power operation.

These comparisons of the conditional probabilities of consequences for mid-loop and full power operation are conditional on the occurrence of core damage or containment failure, i.e., assuming the same frequency of core damage or the same probability of containment failure. However, the risk profile is determined by the product of these conditional risks with the frequencies of occurrence of the conditions giving rise to the risk. If the frequencies of core damage or containment failure accidents are significantly different during mid-loop operation from those at full power, the integrated risk profiles will be dominated by those accident frequencies.

2.7 References

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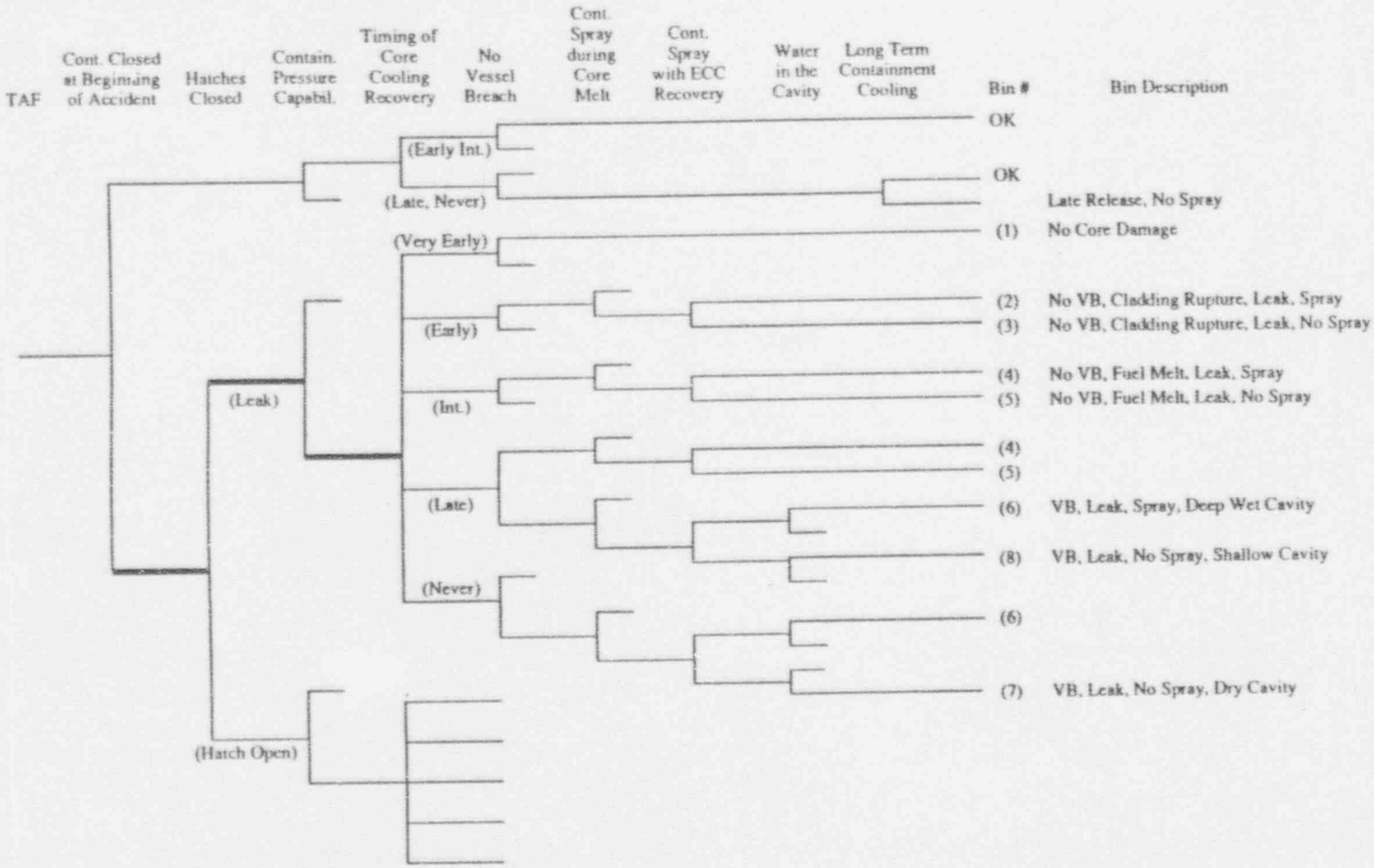


Figure 2.1 Accident Progression Event Tree for the Abridged Low Power/Shutdown Risk Analysis

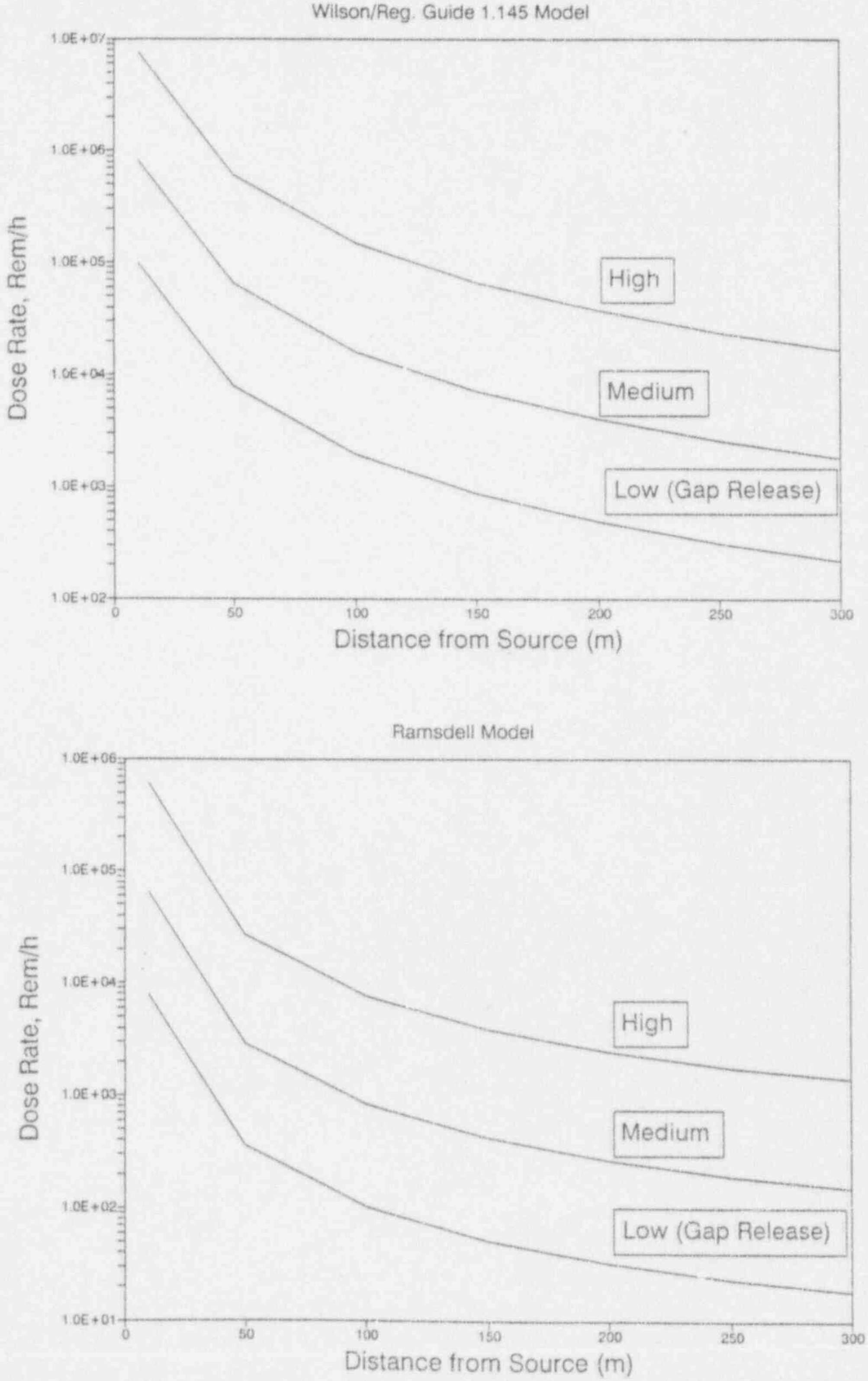
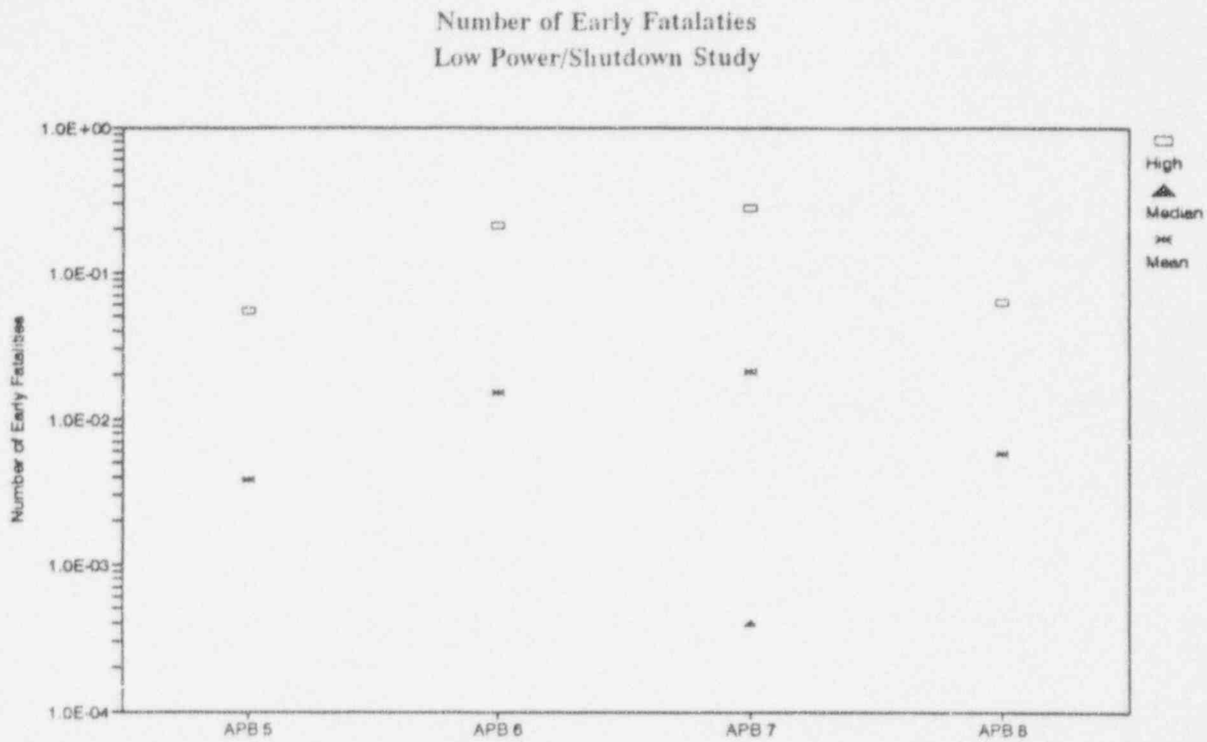


Figure 2.2 Onsite Parking-Lot Dose Rate



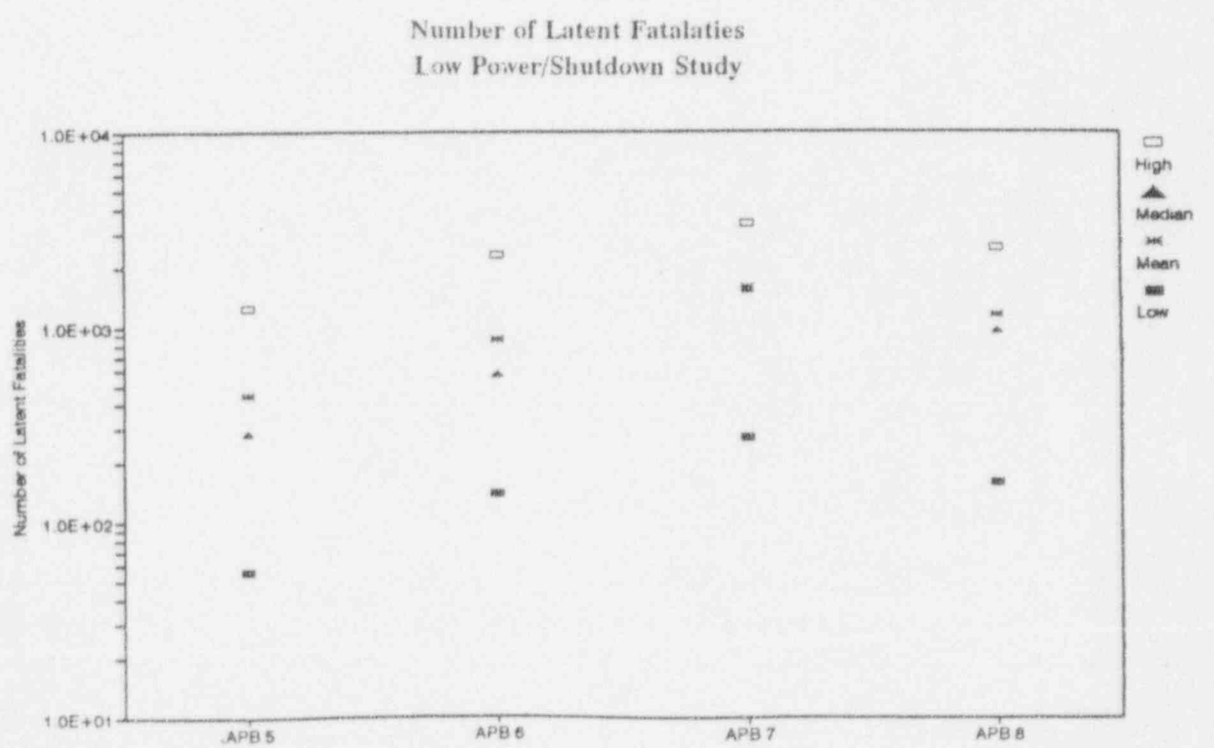
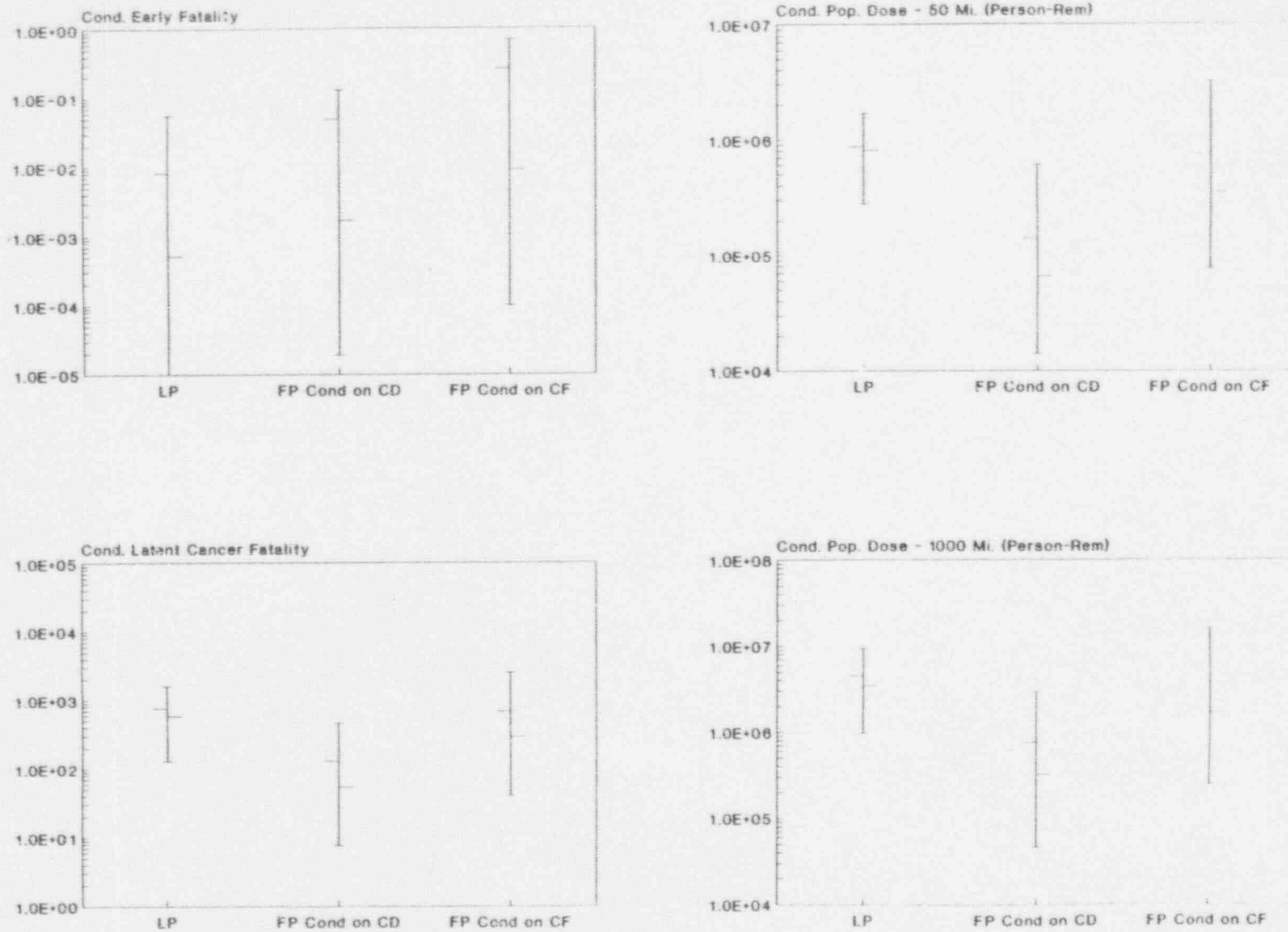


Figure 2.3 Offsite Consequences Calculated by MACCS



* FP results based on NUREG-1150.

Figure 2.4 Comparison of Conditional Probabilities of Consequences for Mid-loop and Full Power Operation (Conditional on Core Damage and Containment Failure)

3 METHODOLOGY

The approach used in the current risk assessment was to utilize to the extent possible the component analyses developed as part of the NUREG-1150 program.[1] The components of the analyses process used to compute risk are displayed schematically in Figure 3.1. This figure is very similar to a figure presented in reference 2 which describes the risk assessment performed for full power operation at Surry. Both approaches have the same four elements (accident frequency, accident progression, source term, and consequence analysis) but some of the components of the analysis had to be modified to reflect differences between the plant conditions at full power and during mid-loop operation.

3.1 Overview of NUREG-1150 Methodology

Figure 3.1 displays schematically the components of the analysis process which consists of four elements:

1) Systems analysis and models of plant response to various initiating events, quantification of accident sequences leading to core damage;

2) Analysis of the accident progression to determine various possible ways the accident could evolve given core damage;

3) Source term analysis, the releases to the environment for the various outcomes of the accident progression;

4) Consequence analysis, the health and economic impacts of each of the source terms.

Integrated risk is obtained by combining the frequency of core damage, the conditional probability of the release paths, and the value of the consequences of each source term conditional on the release into a single risk measure. By repeating the calculation several times with different input values (over specified ranges) of key parameters, a distribution of risk estimates is obtained from which the uncertainty in the risk can be determined.

3.1.1 Accident Frequency Analysis

The accident frequency analysis consists of the fault trees and event trees delineating the sequences leading to core damage. In four out of the five NUREG-1150 studies, the SETS code[3] was used to perform the initial accident frequency analysis. The ultimate outcome of the initial accident frequency analysis is the group of minimal cut sets leading to core damage. The minimal cutsets are then grouped into plant damage states (PDSs), based on similarity of plant conditions, to define the entry points for the subsequent accident progression analysis.

3.1.2 Accident Progression Analysis

In NUREG-1150, accident progression was analyzed using a single accident progression event tree (APET) developed for each plant which was evaluated with the EVNTRE code.[4] The specification of each PDS defines the entry conditions to the APET. The APET developed for Surry in NUREG-1150 had 71 event questions and many of the questions had several outcomes; there were thus far too many paths through the tree to allow consideration of each individual path in terms of the subsequent source term and consequence analysis. The outcomes of the paths were grouped into accident progression bins (APBs) which have similar characteristics and define the entry conditions for the source term analysis. Mechanistic and deterministic code calculations, experimental observations, and a formal expert elicitation process were employed in NUREG-1150 to determine values of key parameters and their ranges in quantifying the model of accident progression.

3.1.3 Source Term Analysis

For each accident progression bin (APB), the source term in NUREG-1150 was calculated by a parametric code, SURSOR.[5] This code is based on a mass-balance approach which considers the fractions of the radionuclide inventory released to the vessel, from the vessel to the containment, and from the containment to the environment. SURSOR integrates the results of detailed mechanistic codes such as MELCOR,[6] MAAP,[7] and the Source Term Code Package[8] as well as distributions provided by expert judgement into a fast-running code which can be executed repeatedly with different values of input parameters to provide distributions of source terms for each APB. The number of APBs is large enough so that evaluating the consequences of each source term in each bin (there are potentially tens of thousands of source terms) is not practical. In NUREG-1150, the source terms were classified into source term groups by the PARTITION program.[9] Partitioning is a procedure for grouping of the source terms based on the similarity of their consequences, that is, the early and late health effects arising from the magnitude and timing of the release of the radionuclide core inventory specified in each source term. Each source term is assigned an early fatality effect weight (which may be zero) and a chronic fatality effect weight and the source terms are divided into groups which have similar values of the weights. A further subdivision of the groups is made on the basis of the release timing relative to the warning and emergency evacuation times. In each source term group, an average (mean) source term is defined and then used for the detailed consequence calculation.

3.1.4 Consequence Analysis

The consequence measures, early fatalities, population dose (person-rem), and latent fatalities, are calculated for each source term group by the MACCS[10] code. The output of MACCS for each source term group is a distribution of the consequences, conditional on occurrence of the source term, which incorporates the uncertainty (variability) due to weather. However, in the NUREG-1150 process, the consequence analysis differs from the three earlier components, accident frequency/plant damage state analysis, accident progression, and source term analysis, in that uncertainties due to important variables and phenomena in the consequence analysis were not propagated through the integrated risk analysis via the Latin Hypercube sampling process[11] as they were for the other three constituent analyses.

3.1.5 Risk Integration

Integrated risk is obtained by combining the output from each of the component analyses as shown in Figure 3.1. The uncertainty and sensitivity analysis of the integrated risk is carried out by using a stratified form of Monte Carlo analysis called Latin Hypercube Sampling.[10] This approach is based on assigning distributions to important variables (through a formal expert elicitation process, for example), creating samples by randomly picking values from the distributions and propagating them through the integrated analysis. The result is a distribution of risk for each of the consequence measures.

3.2 Methodology of Current Study

The methodological approach adopted in the current study is mostly based on the NUREG-1150 approach, which is described in detail in references 1 and 2. The sections below therefore describe only those elements of the methodology which are different from the NUREG-1150 approach.

3.2.1 Accident Frequency Analysis

The Level 1 analysis, including fault trees, event trees, recovery actions, etc., of the significant accident sequences leading to core damage and their frequencies, was carried out by the IRRAS code.[12] This analysis is documented in Volume 2 of this report. A summary of the level 1 analysis results is presented in Chapter 5 of this volume. A newly added feature of IRRAS which became available recently was used to group the minimal cut sets into the plant damage states. Seven characteristics were used to construct the plant damage states.

The first characteristic identifies the time frame in which the accident occurs. A major difference between a PRA at full power and at mid-loop operation is the extended time period following shutdown during which an accident can occur. This time period allows for a significant decay of the power level, extends the time available for various phenomena and for recovery actions and leads to a lower value of the radionuclide inventory which can potentially be released. This feature of the shutdown PRA was modeled in the Level 1, accident frequency analysis through the construction of four "time windows" for various time periods following shutdown. Each time window has its own decay heat level and success criteria for accomplishing various recovery actions prior to core damage. The first PDS characteristic therefore identifies the time window in which the accident occurs.

The second characteristic provides the status of ac power. Of particular interest is whether or not ac power is available and if it is not available whether it can be recovered. Human error is an important contributor to the core damage accident frequency for mid-loop operation. The third characteristic therefore identifies if human error contributed to the accident and if it did what was the type of error. The status of the reactor coolant system can significantly impact accident progression and this is therefore addressed in the fourth PDS characteristic. For some plant damage states recovering coolant injection after the start of core damage can prevent further core damage and terminate the accident prior to the core melting through the reactor vessel. The fifth characteristic deals with the issue of restoring coolant injection. If core damage occurs and containment integrity is lost then operation of the containment sprays can reduce the airborne fission product aerosol concentration and reduce the amount of radionuclides released to the environment. The sixth characteristic gives the status of the spray system. Finally the status (injected or not injected) of the refueling water storage tank is given in the seventh characteristic. The construction of the plant damage states is described in more detail in Chapter 5.

3.2.2 Accident Progression Analysis

The accident progression event tree developed for mid-loop operation has been developed largely based on the APET developed for full power operation in the NUREG-1150 study. Some questions were removed and other questions pertinent to mid-loop operation (such as time windows and containment closure) were added. The APET for mid-loop operation consists of 40 questions compared to 71 questions in the full power APET. Due to resource limitations, a formal expert elicitation procedure could not be implemented to construct ranges of values and distributions for key variables. Assignments of these values and ranges were therefore made internally at BNL. The timing of key events in the accident progression is based on calculations carried out with the MELCOR code.[6] A very important issue which has a major impact on the result is the status of the containment at accident initiation. Assignments of the possible values of this status were made based on discussions and exchange of written communications with the Surry plant personnel. The APET developed for mid-loop operation is described in detail in Chapter 6. The APET was quantified using the EVNTRE[4] code as in the full power study. It was again necessary to combine the numerous outcomes of the APET into accident progression bins for input to the source term analysis. A similar approach to that used in the full power study was also used in the current analysis. The only additional information needed for the current study was the time window in which the accident occurred. Thus information identifying the time window was carried throughout all of the constituent analyses.

3.2.3 Source Term Analysis

The source term analysis used for mid-loop operation was similar to the approach used for full-power operation. The SURSOR[4] code was reviewed for its applicability to shutdown conditions by an expert group consisting of staff from BNL and SNL. The source term ranges in SURSOR were also compared against predictions with the MELCOR code. In general SURSOR was considered appropriate for use in the current study. The APBs were therefore processed through the SURSOR code in a similar manner to the full power study. The output from SURSOR is a larger number of source terms which need to be grouped into representative source terms. The process was done in the full power study using PARTITION.

Methodologically, the important difference between NUREG-1150 and the present study is a reworking of the partition approach to reflect the long time interval and consequent decay of the inventory in the current study. In effect, accident progression bins and source term groups are defined for each time window. The partitioning of the source terms and the assignment of health effect weights is carried out through a partition procedure designated PARTITION-LPS, which is described in Chapter 7.

3.2.4 Consequence Analysis

The consequence measures for the average source term in each source term group were calculated by the MACCS code.[10] The latest release of MACCS,[10] version 1.5.11.1, which incorporates the important BEIR V update to the latent cancer - dose relationship,[13] was used to compute consequences. In contrast, the NUREG-1150 study used an earlier version of MACCS, Version 1.5.11, to compute consequences. The more recent version of MACCS gives a higher (by approximately a factor of 3) number of latent cancers, than the earlier MACCS version for the same value of population dose.

3.2.5 Risk integration and Uncertainty

The integrated risk was obtained by combining the individual results of each of the constituent analyses as shown in Fig. 3.1. The approach was similar to the NUREG-1150 approach. Distributions were assigned to important variables (some distributions were identical to those used in NUREG-1150, others were developed specifically for mid-loop operation) and samples were then created by randomly picking values from the distributions using Latin Hypercube Sampling.[11] For each sample the values assigned to each variable were propagated through the integrated analysis to determine risk estimates for each consequence measure. By repeating the calculation for 100 samples (or observations) distributions of risk estimates were obtained from which the uncertainty in risk was determined.

3.3 Scoping Calculation of Onsite Doses

A scoping calculation of onsite doses, outside of the main risk calculation, was carried out at the request of NRC to gain some insight into plant conditions at the time of the release which could possibly have an effect on various recovery actions in different locations of the plant. This calculation is based on taking ranges of high and low source terms in various time windows and interfacing them with deterministically estimated weather conditions. Two different correlations, one of which includes recent work on building wake correlations, were used to compute the "parking lot" dose as described in Chapter 8.

3.4 References

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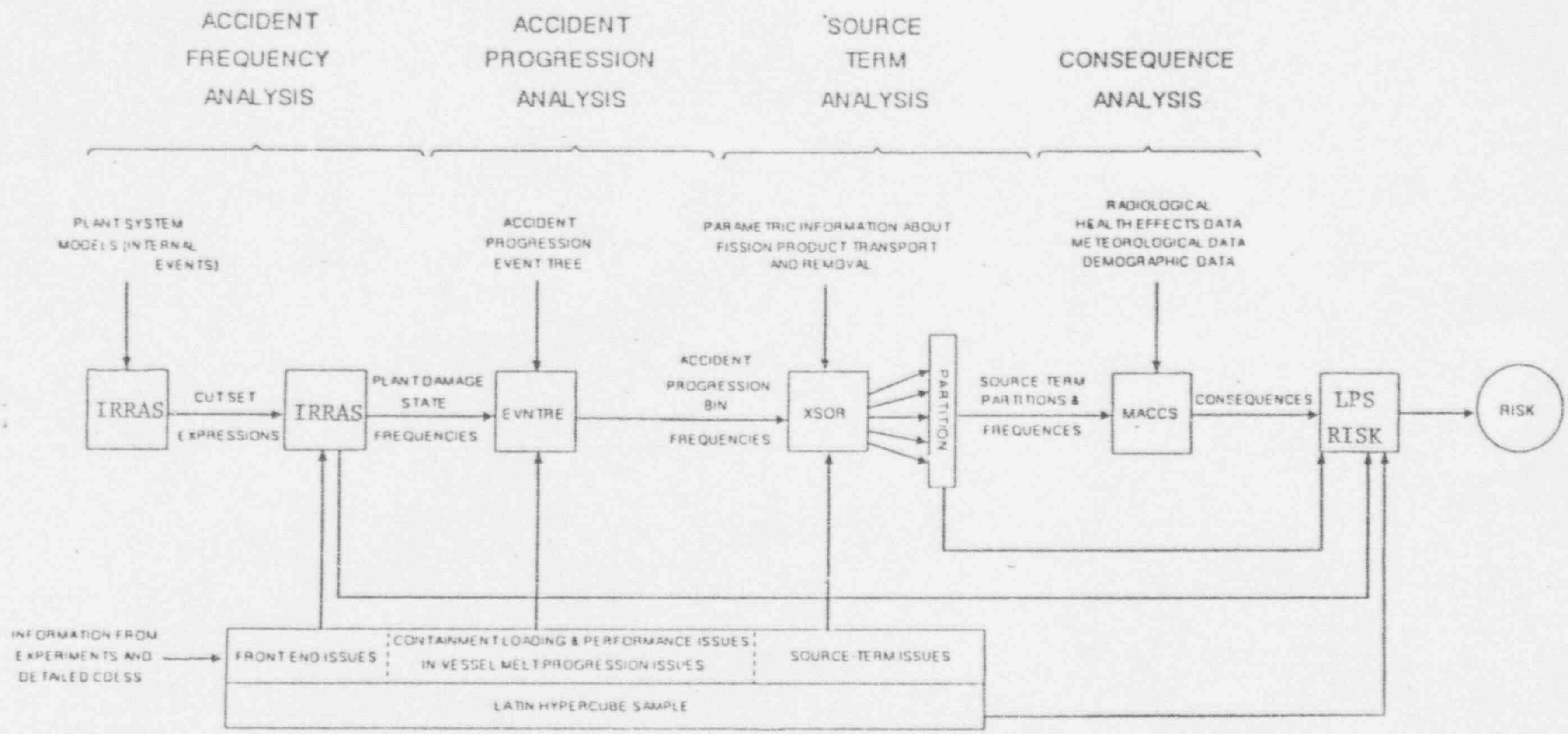

Figure 3.1 Overview of Integrated Plant Analysis Used for Mid-loop Operation

4 PLANT DESCRIPTION

4.1 General Description

Surry Unit 1 is a 2441 MWth pressurized water reactor (PWR) designed and constructed by Westinghouse. It is operated by the Virginia Electric Power Company. Surry is a three-loop plant; the reactor coolant system has three U-tube steam generators and three reactor coolant pumps. The containment and balance of plant were designed and constructed by Stone and Webster. Commercial operation of Unit 1 began in 1972.

Emergency ac power at the site is supplied by three diesel generators (DGs). One DG is aligned to Unit 1, the other to Unit 2, and the third DG functions as a swing diesel which can be aligned to either unit. Emergency dc power is supplied by separate battery banks at each unit. The DGs have their own separate set of batteries for starting power. The auxiliary feedwater (AFW) system has three trains. Two trains have electric pumps, the third train has a steam turbine driven pump. The condensate storage tank provides suction for the AFW system. The chemical volume and control system has three charging pumps which also serve as the high-pressure injection (HPI) pumps. There are two low pressure injection (LPI) pumps. Both the HPI and the LPI systems can function in the injection or recirculation mode. In the injection mode, they take suction from the refueling water storage tank (RWST) while in the recirculation mode they take suction from the sump. A more detailed description of the safety injection/recirculation systems is provided in sections 4.2.2.1 and 4.2.2.2 below. Surry also has three accumulators which provide a source of immediate, low-pressure, high flow injection.

Overpressure protection for the reactor coolant system is provided by three code safety/relief valves (SRV) and two power operated relief valves (PORV). Surry has an unique service water system which is supplied by gravity flow from an elevated canal. The canal is continuously supplied by river water from the Jamestown river through electric pumps. If ac power is lost, the service water canal will drain in about 30 minutes unless a large number of valves are closed manually.

The Surry containment is a reinforced concrete cylinder with a hemispherical dome. The free volume of the containment is 1.8 million cubic feet and the design pressure is 45 psig. A welded steel liner covers the inner surface of the containment and forms the pressure boundary. A section of the Surry containment is shown in Figure 4.1. Due to design conservatisms, realistic estimates of the loads needed to fail the containment are between two and three times the design pressure. The mean of the distribution for the failure pressure of the Surry containment provided by the expert panel in the NUREG-1150 study was 126 psig. During full-power operation, the Surry containment is maintained at a sub-atmospheric pressure of about 10 psia, i.e., about 5 psia below ambient atmospheric pressure. This feature and the Technical Specifications prevent plant operation much in excess of this pressure therefore the probability of pre-existing leaks and isolation failure during normal operation is extremely low.

Cooling of the containment is normally provided by fan coolers which are not safety grade and will be partially submerged if the sump is filled with water. Emergency cooling of the containment is provided by the containment spray systems (CSS). The CSS is described in more detail in section 4.2.4 below. Another feature of the Surry containment at a low elevation is that there is no connection between the sump and the reactor cavity. If a pipe break occurs, the water will flow to the sump. The cavity remains dry unless the containment sprays operate.

The general description given above indicates the main plant systems available during full power operation at Surry. However, during shutdown the plant is configured differently than during full power operation and some of the systems described above will not be available.

4.2 Plant and System Configuration During Mid-loop Operation

Three mid-loop operating states were identified and analyzed in the level 1 analysis (refer to volume 2 of this report); two mid-loop operating states during refueling outages (one early in the outage during cooldown using the residual heat removal (RHR) system and the other later after completion of refueling), and another mid-loop operating state during the cooldown period of a drained maintenance outage. A detailed analysis of plant systems, their response to various accident initiators and their status in accident sequences leading to core damage are contained in Volume 2 of this report.[1] In this volume, the focus is on those plant systems and features which are important to the progression of the accident and to the possible releases to the containment and the environment following core damage. Accident progression can be influenced by the status of the reactor coolant system, recovery of coolant injection systems, containment integrity, containment spray systems and cavity flooding. These systems and plant features during mid-loop operation are described in the following sections.

4.2.1 Status of the Reactor Coolant System

The reactor coolant system (RCS) is at low pressure during mid-loop operation as soon as the plant is placed in the RHR entry level condition. This implies that potential accidents during mid-loop operation will not involve any high pressure sequences such as those modeled in the full power PRA. Also during mid-loop operation the relief valves in the pressurizer are open connecting the pressurizer to the pressurizer relief tank which is vented to the process vent system. The vessel head vent is connected to the discharge side of the through piping that consists of a section of tygon tube which can withstand about 40 psia of pressure. Additionally, the safety valves could be removed for maintenance during mid-loop and a temporary partition placed on the opening. This creates the possibility of a direct vent path into containment for any released fission products in the event of any accident. These features of the RCS during mid-loop operation have been incorporated in the accident progression event tree described in Chapter 6.

4.2.2 Emergency Core Cooling Systems (ECCS)

The ECCS at Surry consists of the High Pressure Injection/Recirculation (HPI/HPR) system and the Low Pressure Injection/Recirculation (LPI/LPR) system. ECCS is important to the accident progression because for some plant damage states it could be restored after the start of core damage. If the ECCS is restored while the damaged core is still in the reactor vessel it may be possible to terminate the accident prior to vessel meltthrough. A relatively high probability of terminating the accidents invessel was estimated in the accident progression analysis for three out of the four plant damage states. If the core debris has melted through the vessel and is attacking the reactor cavity restoration of the ECCS will supply water to the cavity and flood the core debris. A flooded cavity could terminate the core-concrete interaction and considerably mitigate the associated source term. If core-concrete interactions continue, flooding of the cavity would lead to a scrubbing of the fission product release. The possibility of flooding the cavity is also incorporated in the accident progression analysis described in Chapter 6.

4.2.2.1 High Pressure Injection/Recirculation System

The Surry high pressure injection/recirculation (HPI/HPR) system consists of three centrifugal charging pumps and associated piping and valves. Following an accident, the charging pumps are used to provide primary coolant injection and recirculation as well as maintain flow to the RCP seals. The charging pumps are one of the three major components of the Safety Injection (SI) System. The other two components are accumulators and low pressure injection pumps. The primary purposes of the SI system are: (i) to inject borated water into the RCS to flood and cool the core following a LOCA; and (ii) to remove heat from the core for extended periods of time following a LOCA. The HPI system also functions to deliver boric acid to the RCS from the boric acid transfer system if emergency boration is required. The HPI/HPR system provides coolant makeup, early and late core heat removal or emergency boration for reactor shutdown.

The suction source of the charging pumps in the high pressure injection mode is the RWST. Before the contents of the RWST are exhausted, the Engineered Safety Features (ESF) system automatically initiates a recirculation mode transfer (RMT) signal. The operator can also terminate the injection mode and initiate the recirculation mode by manually repositioning the required valves. In the recirculation mode of operation, the HPR is used to provide core heat removal late in an accident sequence. The charging pumps draw suction from the discharge of the low pressure safety injection pumps in the low pressure recirculation (LPR) system.

To minimize the possibility of accidentally overpressurizing the RCS, technical specifications require that whenever the average temperature of the core is less than 350°F, the following charging pump conditions must be maintained: (i) a maximum of one charging pump operable; and (ii) two charging pumps must be demonstrated inoperable at least every 12 hours by verifying that their circuit breakers are racked out or that their control switches are in the pull-to-lock position. The surveillance requirements specify that a complete systems test be performed during refueling shutdown to demonstrate correct response to an activation of the safety injection signal. The HPI/HPR system configuration at shutdown, including mid-loop operation, is discussed at length in Volume 2 of this report.[1]

4.2.2.2 Low Pressure Injection/Recirculation System

The low pressure injection/recirculation (LPI/LPR) system provides emergency coolant injection and recirculation following a loss of coolant accident (LOCA) when the reactor coolant system (RCS) depressurizes below 180 psig. In the injection mode, the LPI system takes suction from the RWST. In the recirculation mode, the LPR system is aligned to take suction from the containment sump. During the recirculation mode following drainage of the RWST to a low-low level, the LPR discharge also provides the net positive suction head (NPSH) for the high pressure recirculation system.

The LPI/LPR system consists of two 100% capacity pump trains. During normal plant operations, the low pressure injection pumps are in standby, lined up to pump borated water from the RWST to the RCS cold legs. Each LPI pump has a capacity of 3250 gpm at a temperature of 300°F and a pressure of 300 psig with a design head of 225 ft. Actual pump capacity, however, depends on pump discharge pressure.

Technical specifications require that the pumps, piping, valves, etc. of the LPI/LPR system be operable at all times. A detailed description of the LPI/LPR system configuration in the shutdown POSs, including mid-loop operation, is contained in Volume 2 of this report.[1]

4.2.3 Containment Configuration During Mid-Loop Operation

At the inception of the Abridged Study, the status of containment isolation during mid-loop operation was raised with VEPCO staff during a visit to Surry. At that time it was determined that while containment was considered "closed" during mid-loop operation, what closure meant was that all penetrations were isolated from the outside, some with temporary barriers, so that there is no air/vapor exchange with the environment. However, "closure" in the above sense did not mean that the containment was capable of withholding the pressure that could be generated during the course of a severe accident and prevent the release of fission products to the environment. The operating procedure checklist 1-OP-1G (Surry Power Station Unit 1 Refueling Containment Integrity and RCS Mid-Loop Containment Closure Checklist), which was acquired from Surry staff on this visit, had as its objective the achievement of containment integrity in the above-mentioned sense of an air/vapor barrier not the design pressure capability. The containment closure procedure and closure time mentioned in the January 6, 1989 letter of VEPCO to NRC in their response to Generic Letter GL87-12 were also the procedures and times required to achieve containment isolation in the sense of no air/vapor exchange with the environment.

The difference in containment configuration between normal operation and POS 6, mid-loop operation, of the refueling outage appears to be that during the refueling outage there is a temporary plug in place where the emergency escape trunk is usually installed during normal operation. This is shown schematically in Figure 4.2. The temporary plug was estimated to have only a 3 psi overpressure capability. In normal operation, the escape trunk is installed with the O-rings in place in the interface with the equipment hatch to achieve the design pressure capability.

Based on these considerations, it was assumed in the Abridged Study (refer to Chapter 2 of this report) that for accidents initiated during mid-loop operation which progressed to core damage the containment would leak to the environment from the start of a release into containment. The leak was assumed to take place through the temporary plug in the equipment hatch.

Recognizing perhaps the potential problems regarding containment status during low power and shutdown operation, the Surry staff have developed additional procedures to address the concerns about the closure of the containment during POS 6 or shortly after the initiation of an accident during shutdown operation. According to the most recent Surry procedures, one of the minimum equipment requirements for forced feed and bleed (feed and spill) cooling is containment isolation (Surry Procedure 1-OSP-ZZ-004). The procedure for containment isolation is provided by Surry Operations Surveillance Procedure 1-OSP-CT-214, Containment (CTMT) Closure for Reduced or Potentially Reduced RCS Inventory Conditions. The procedure provides instructions for the preparation, implementation, and documentation of containment closure activities and indicates that it should be carried out before commencing to drain the RCS to a reduced or potentially reduced inventory condition with fuel in the reactor vessel. The procedure lists all the containment penetrations that need to be closed. A single barrier containment isolation is required. However, some penetrations that are required for normal operation of the plant may not be isolated during containment closure. The procedure states that valves associated with these penetrations that are in the open position for normal plant operation should be noted as such, but should not be considered as a CTMT closure discrepancy.

Discrepancies that would prevent the achievement of a containment pressure capability of 45 psig are treated as CTMT Closure Concerns. One action that may be taken to resolve a CTMT Closure Concern is to implement the use of an acceptable barrier to meet Pressurized CTMT Isolation: i.e., metal flanges are sealed with red rubber gaskets, bolting material, and torqued to specifications capable of retaining a pressure of 45 psig. There is no discussion in these procedures about what kind of leak tightness is required (or expected) or how it can be assured (e.g. through a leak test) if such a capability exists.

Because only a single barrier is specified, and there exists the possibility that temporary barriers are used for some penetrations, it is likely that the leak tightness and the pressure capability of the containment during POS 6 will not be as good as during normal power operation. On the other hand, if the above procedure succeeds in isolating the containment during POS 6, leaks from the containment atmosphere in accident conditions, and thus the fission product release, to the outside environment may be significantly reduced.

In addition to the questions about the leak tightness and the pressure capability of the containment, the probability of achieving containment isolation* within the time frame of interest in the presence of degraded containment conditions is another issue which is not addressed by the above procedure.

Since containment status during shutdown is, perhaps, the single most important feature of the plant which affects risk, additional questions on the procedure 1-OSP-CT-214 were addressed to the Surry staff to clarify its scope and intent. The first question was whether implementation of the "single barrier pressurized CTMT Isolation" procedure 1-OSP-CT-214 implies that "the containment is completely isolated during all phases of reduced inventory condition?" Surry staff were also asked a second question related to the pressure capability of the containment under the "single barrier containment isolation condition" specified in the procedure and asked if this capability, for both the design and the ultimate capability, was similar to that established for Surry by the NUREG-1150 study (during normal full power operation).

Surry's response to the first question was that the 1/2-OSP-CT-214 procedure "does not ensure that the containment is completely isolated during all phases of reduced inventory conditions." It was pointed out that, under the proposed procedure, the isolation barrier requiring the bulk of the closure time and the resources, the equipment hatch and the escape trunk, would be installed prior to entering reduced inventory and that the procedure "ensures that the majority of the remaining penetrations are closed or capable of being closed by a single isolation barrier (i.e. containment isolation valves) from the main control room". It was also mentioned that "penetrations, which are not or cannot be isolated from the main control room, are listed as discrepancies". However, the statement that the discrepancies would be isolated (by a containment closure team established prior to entering reduced inventory) "in case of an event" coupled with the inability to provide an estimate of the time required to resolve discrepancies and reestablish containment closure leads to some uncertainty about the level of assurance provided by the Surry response, despite the assertion that "any required reestablishment of containment closure should be performed well within the time to core uncovery recommended by Technical Report 865." Regarding the second question, Surry noted that "As an alternative to the escape hatch, a preliminary design for a new barrier is being considered" which would be capable of withstanding 45 psig. It is not clear why a new barrier design is needed (to address the closure/isolation concerns) if the existing barriers can isolate containment "well within" the time frame of relevance to the accident.

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* There will be open penetrations that need to be closed after accident initiation. These penetrations include those that are required for normal plant operation during containment closure.

Given this uncertainty, particularly the fact the new procedures may still be evolving, it seems prudent to model the probability of pre-existing leakage (as assumed in the Abridged Study) and the containment failure pressure as an uncertainty parameter and perform sensitivity analysis to evaluate the impact of different assumptions regarding containment status. The assumptions made in this regard are described in more detail below in Chapter 6 where the accident progression event tree is developed.

4.2.4 Containment Spray System

4.2.4.1 Containment Spray Injection System

Containment heat removal in an emergency at Surry is by means of the containment spray system (CSS) in the injection mode. The containment spray injection system provides initial cooling of the containment atmosphere following an accident which can pressurize the containment, for example a LOCA. The CSS pumps take suction from the refueling water storage tank (RWST), which is filled with chilled (45°F) borated water, and spray it inside the containment to condense the steam. Containment sprays also provide a mechanism for scrubbing fission products from the released inventory. The Technical Specifications require that the RWST contains between 387,100 and 398,000 gallons of 2300-2500 ppm borated water chilled to at least 45°F.

The containment spray system consists of two 100% capacity trains. Each train is connected to a separate spray ring, and there is an additional ring shared by both trains. The spray pattern from the rings covers about 73% of the containment atmosphere.

4.2.4.2 CSS Configuration During Mid-Loop Operation

The requirements on the availability of the CSS apply when the RCS temperature and pressure is in excess of 350°F and 450 psig, respectively. There are no requirements below these limits. When the reactor is operating at power, both CSS trains must be operable.

Considering the operating parameters of mid-loop operation, there are no Technical Specifications which require CSS to be available during this plant operational state. Discussions with Surry personnel indicated, informally, that the probability of at least one train of CSS being available was likely to be fairly high, on the order of 70 percent. Accordingly, spray availability was treated as an uncertainty parameter in the development of the Surry APET in Chapter 6. If CSS is available during POS 6, it would have to be manually actuated since automatic actuation is not available at RCS temperature below 350°F.

4.2.4.3 Recirculation Spray System

The Inside Spray Recirculation (ISR) and the Outside Spray Recirculation (OSR) systems provide the long term containment cooling and pressure reduction following an accident. At Surry, these systems also provide long term core cooling after the accident. Each of the recirculation spray systems, ISR and OSR, contains two independent pump trains. Each train takes suction from the containment sump and discharges through a separate heat exchanger. There are four heat exchangers, one for each train. The heat exchangers are cooled by service water. Each ISR or OSR train has a separate 180 degree spray ring so there are a total of four spray rings for the recirculation sprays. Two recirculation spray trains, in any combination, are sufficient to provide long term cooling following a loss of coolant accident (LOCA).

The ISR pumps are located inside the containment and are qualified for the harsh post-accident environment. They provide about 3500 gpm of flow. The OSR pumps are located outside the containment and also provide about 3500 gpm of flow each. The recirculation spray systems depend on either the CSS or the ECCS injection system to provide sufficient inventory of water in the sump for their operation and on the service water system for the ultimate heat sink. The requirements on the ISR and OSR systems are for RCS temperatures and pressures in excess of 350°F and 450 psig, respectively. Above these limits, all four trains must be available. There are no requirements below these limits.

4.2.4.4 Recirculation Spray System Configuration During Mid-Loop Operation

There are no Technical Specifications for ISR and OSR systems below the above limits of 350°F and 450 psig, which are the operational parameters in POSs 3-13. Thus it is possible that neither of the recirculation spray systems, ISR or OSR, would be available during mid-loop operation. In discussions with Surry plant personnel it was indicated, informally, that the likelihood of availability of at least one train of either ISR or OSR is high (about 70%) during shutdown.

4.2.5 Reactor Cavity

The reactor cavity at Surry is normally dry as all water in the containment drains to the sump and there is no connection between the sump and the cavity. This feature of the Surry cavity has important implications for the progression of severe accidents and the source terms where the vessel is breached and core-concrete interactions occur. The only way for the cavity to have water is either if the containment sprays operate or if core injection is recovered after vessel breach. This feature has been incorporated in the accident progression event tree described in Chapter 6.

4.3 References

1. Chu, T. L., et al., "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry Unit-1: Analysis of Core Damage Frequency from Internal Events During Mid-loop Operations," Brookhaven National Laboratory, NUREG/CR-6144, Volume 2, December 1993.

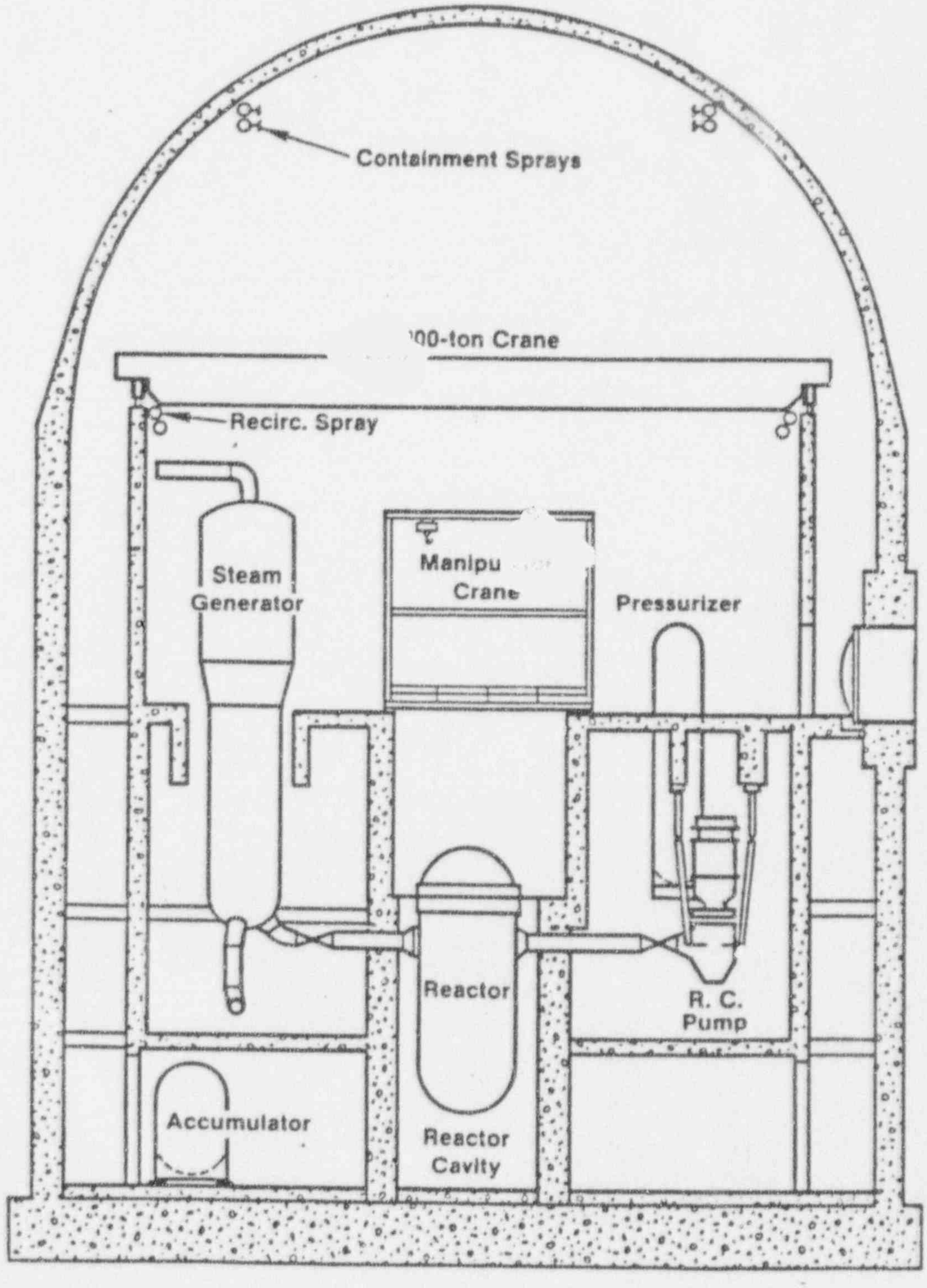
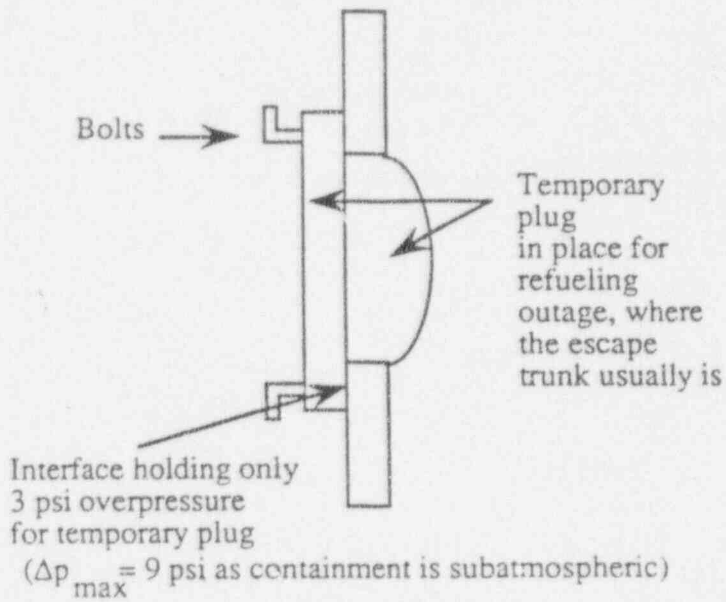


Figure 4.1 Section of Surry Containment

Figure 4.2 Equipment Hatch Configuration During Mid-loop and Full Power Operation



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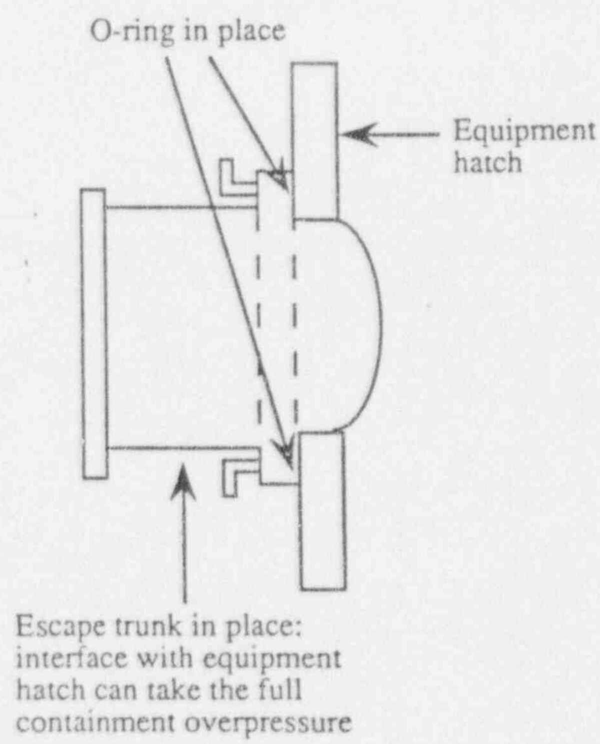

Full Power Operation

5 ACCIDENT FREQUENCY ANALYSIS INTERFACE

The analysis[1] of the core damage frequency (level 1 PRA) for accidents during mid-loop operation is presented in Volume 2 of this report. Volume 2 contains a detailed description of the characteristics of the operational states of the plant during mid-loop operation and describes the initiating events considered. An analysis of the plant's response to the initiating events (including human actions) and the calculation of the frequency of the sequences leading to core damage is also presented. A summary of the core damage frequency analysis is given in Section 5.1 below. The way in which the different accident sequences that can lead to core damage are interfaced with the accident progression analysis is discussed in the remainder of this Chapter. The accident sequences are binned into plant damage states which have the characteristic that all sequences in a given plant damage state behave in a similar manner during the subsequent accident progression. The starting point for the accident progression analysis is therefore defining the plant damage states and calculating their frequency. The characteristics of the plant damage states are described in section 5.2. The calculation of their frequency is presented in section 5.3.

5.1 Summary of Level 1 Core Damage Accidents

5.1.1 Definition of Plant Outage Types and Operational States

The initiating events and systems analysis performed for Surry Unit 1 in the Level 1 analysis included all operational modes at less than 15% power defined in the Updated Surry Final Safety Analysis Report. Seven operational modes are defined in the Surry Technical Specifications in terms of reactor criticality, reactor power, temperature, and pressure; five of these, refueling shutdown, cold shutdown condition, intermediate shutdown condition, hot shutdown condition, and refueling operation refer to low power and shutdown conditions.

Four different types of outage were considered in the Level 1 analysis: refueling, drained maintenance, non-drained maintenance with the use of the residual heat removal (RHR) system, and non-drained maintenance without the use of the RHR system. Each outage type is further subdivided into phases defined by the following parameters: Frequency, Plant Configuration, System Availability, Shutdown Activities, Time to Core Uncovery, Maintenance Unavailability, Reactor Coolant System (RCS) Integrity, Containment Integrity, Reactivity, Reactor Coolant Temperature, Reactor Coolant Pressure, Reactor Vessel Level, Time after Shutdown and Duration. The phases are described as plant operating states (POSs). Each outage type is characterized by a number of POSs, the time spent in each POS (a variable determined from plant operating records), and the activities carried out in that POS. The refueling outage, for example, is characterized by 15 POSs as follows:

POS 1: Initiation of low power operation (10-15% power level) proceeding to hot shutdown (Average Core Temperature = 547°F, RCS pressure = 2235 psig).

POS 2: Cooldown with Steam Generators (SGs) to 345°F.

POS 3: Cooldown with RHR system to 200°F.

POS 4: Cooldown to ambient temperatures using RHR system.

POS 5: Draining the RCS to Mid-Loop.

POS 6: Mid-Loop Operation.

POS 7: Fill for Refueling.

POS 8: Refueling.

POS 9: Draining RCS to Mid-Loop after Refueling.

POS 10: Mid-Loop Operation after Refueling.

POS 11: Refill RCS completely after Mid-Loop Operation.

POS 12: RCS Heatup Solid and draw Bubble.

POS 13: RCS Heatup to 350°F.

POS 14: Startup with SGs.

POS 15: Reactor startup and Low Power Operation.

For a drained maintenance outage, POSs 7 through 10 given above would not be applicable. To provide a perspective on the amount of time spent in each of the above POSs, Table 5.1 shows the average duration of each POS for the four outage types based on 1985-1989 plant data.

5.1.2 Mid-Loop Operation POSs

Mid-loop operation occurs when the reactor coolant level is drained to the mid-plane of the hot leg. Three mid-loop operation POSs were considered in the accident frequency analysis (refer to Volume 2 of this report). These are POS 6 and POS 10 of a refueling outage, and POS 6 of a drained maintenance outage. In POS 6 of a refueling outage, mid-loop operation is used to perform eddy current testing of the SG tubes, while in POS 6 of drained maintenance it is used to carry out needed maintenance.

In mid-loop the pressurizer power-operated relief valves (PORVs) are open connecting the pressurizer to the pressurizer relief tank (PRT) which is vented to the process vent system through a 3/4 inch line. The vessel head vent is connected to the discharge side of the PORVs through piping that consists of a section of tygon tube which can withstand a pressure of about 40 psia.

During mid-loop operation, based on past practice, there is a possibility that one or more of the reactor loops would be isolated. A review of past data showed that during refueling outages, the loops were isolated prior to reaching POS 6 in one case, and during POS 6 in the other cases. The loops remained isolated until POS 12 was entered, so that all three loops were isolated completely in POS 10. In drained maintenance, on the other hand, one loop is typically isolated. Loop isolation has a very important impact on the ability to use the steam generators to remove decay heat (reflux cooling) in the case of loss of the residual heat removal (RHR) system during mid-loop operation. If the loops are isolated, reflux cooling is not available as a means of controlling the accident.

One important difference between the various mid-loop POSs (in different outage types) is the time after shutdown that the POS occurs and the duration of the POS. These times determine the decay heat level which, in the event of an accident, strongly influences the time of the subsequent accident progression. The time after shutdown at which the accident occurs also strongly influences the fission product inventory potentially available for release if the fuel is damaged. These times, therefore, play a key role in all aspects of the integrated risk analysis. Past outage data at Surry were analyzed to determine distributions of the time to reach mid-loop and the duration of mid-loop operation. Details of the statistical analysis performed are given in Appendix D of Volume 2 of this report. Four statistical measures (i.e., mean, 5th, 50th, and 95th percentile) of these distributions are given in Tables 5.2 (time to reach mid-loop) and 5.3 (duration of mid-loop). These times vary over a wide range; the duration of POS 10, for example, ranges from less than 6 hours to over 2500 hours with a mean of 444 hours. This range greatly affects the evolution of the accident and the likelihood of recovery. The duration times given in Tables 5.1 and 5.3 are different because in Table 5.1 the times are simple averages of the data whereas the mean times in Table 5.3 were obtained by fitting distributions to the data.

A review of past data also showed that during mid-loop operation in a refueling outage the safety valves on the pressurizer were removed for maintenance (sometimes for extended periods). This removal opens a large vent in the RCS. In the event of an accident this vent would allow relief of system pressure and makeup from the refueling water storage tank (RWST). However, it would make reflux cooling impossible due to loss of inventory through the opening. In the accident progression following core damage, the removal of the safety valves also implies the RCS would be at low pressure during the various accident scenarios and that a direct path would exist from the vessel to the containment for the release of fission products. Safety valves are not removed during POS 6 in a drained maintenance outage.

The practice of using mid-loop operation during outages appears to be changing at Surry. In the most recent refueling outage Unit 1 avoided mid-loop operation. The core damage frequency analysis in Volume 2 of the report is based on past practice and does not reflect the most recent outage. It is not clear to what extent the plant can completely avoid going to mid-loop in future outages.

5.1.3 Description of Time Windows

The time to enter mid-loop and the average duration of mid-loop operation are important parameters, which have a large impact on the probability of recovering from the accident. The criteria used for success of the safety systems to prevent core damage differ depending on the decay heat level, which is a function of the time that the accident occurs after shutdown. These times also have a significant impact on the progression of the accident and on possible releases and the consequences. In order to incorporate these times formally into the analysis, a "time window" approach was developed. This approach is based on dividing the distribution of times shown in Tables 5.2 and 5.3 into sub-periods called "windows". A total of four time windows after shutdown were defined in the accident frequency analysis. Table 5.4 shows the definition of the time windows. The information in Table 5.4 was reproduced from a more detailed description of the time windows given in Volume 2 of this report. For POS 6, in both refueling and drained maintenance outages, all four windows were applicable; for POS 10 only windows 3 and 4 were applicable.

Each window is characterized by a time interval (measured from the time of reactor shutdown), and a representative level of decay heat, which corresponds to the mid-point of the time interval. The decay heat then determines subsequent parameters such as the time to boiling if the RHR system is lost, the time to reach various pressures which will challenge sub-systems such as the (temporary) tygon tubing, and the pressurizer relief tank (PRT), time to core uncovery and eventually core damage. These times are displayed for each time window in Table 5.4 The definition of time windows used in the accident progression analysis was also used in the definition of the plant damage states, and the accident progression event tree.

5.1.4 Summary of Initiating Events

Initiating events were identified in the Level 1 analysis by reviewing relevant studies and Surry operating procedures and searching licensee event reports. Initiating events that are specifically applicable to mid-loop operation include: loss of RHR, loss of offsite power, loss of 4 kV bus, loss of vital bus, loss of outside instrument air, loss of component cooling water, loss of emergency switchgear room cooling, inadvertent safety feature actuation, and boron dilution events. The most important initiator both in terms of frequency and its impact on the accident analysis is the loss of RHR, followed by loss of offsite power and loss of a 4 kV emergency bus.

5.1.5 Human Reliability Analysis

Human error events modeled in the Level 1 analysis included pre-accident errors and post-accident errors. The former were partly adopted from the full power study[2] performed for Surry, while the latter were specifically developed in the Level 1 analysis through a detailed definition of the event scenario, required actions and the factors which affect operator action. The success likelihood index methodology[3] (SLIM) was used to derive human error probabilities based on a qualitative evaluation of the actions and parameters affecting operator performance, the performance shaping factors (PSFs). Human errors modeled included failure to diagnose and failure to take action.

5.1.6 Accident Sequence Quantification and Results

The accident frequency analysis for mid-loop operation is presented in Volume 2 of this report. The results are summarized here to give a perspective on the subsequent accident progression analysis. A more detailed discussion including consideration of uncertainties is given in Volume 2. Accidents initiated in POS 6 in both drained maintenance and in refueling outages are the most important contributors to the mean core damage frequency. Accidents initiated in POS 10 were estimated to have an order of magnitude lower mean core damage frequency than accidents in POS 6. This result is expected because POS 6 occurs much earlier in an outage than POS 10 so that less time would be available for recovery actions and thus the core damage frequency should be higher in POS 6. The Level 1 analysis generated a total of 2186 core damage cutsets with frequencies greater than the truncation limit of 10^{-10} per year. All cutsets with frequencies above 10^{-10} per year were incorporated into the plant damage state analysis.

5.2 Plant Damage State (PDS) Characteristics

Information about the many different accidents that lead to core damage is passed from the core damage frequency analysis to the accident progression analysis by means of PDSs. Because the accident progression analysis is similar for many of the cutsets identified in the core damage frequency analysis the 2186 cutsets were grouped together into a smaller number of plant damage states. The prime consideration when assigning a core damage accident cutset to a PDS is similarity in the progression of the accident in the vessel and in containment, and in ways to terminate the accident. Therefore all of the sequences binned into a PDS should have similar behavior following the uncovering of the core.

The plant damage states were identified by a seven-letter indicator that defines seven characteristics that largely determine the progression of the accident. For each characteristic possible attributes are discussed below:

1. Time Window – The time window in which the accident occurs can be easily determined by the basic event names used. Attributes 1, 2, 3, or 4 are assigned depending on the time window in which the accidents occur.

2. AC Power – This question determines whether or not recovery of offsite power after core damage can prevent further degradation of the condition. If core damage is caused by loss of offsite power, then it may be possible to re-establish injection after offsite power is recovered.

   Y: If AC power is available in the cutset, then recovery of offsite power is not relevant.

   U: This attribute is used when the initiating event is a loss of emergency switchgear room cooling and cooling is not recovered. For such cutsets, the loss of power is not recoverable and vessel breach is unavoidable.

   B: If the cutset represents a station blackout, then recovery of offsite power should restore power to the equipment that can be used to prevent vessel breach. Recovery of offsite power is characterized by the recovery curve given in Volume 2 of this report.

   F: If the cutset involves a loss of the 4 kV Bus, then restoring power to the bus should restore power to the equipment needed to prevent vessel breach. The recovery of 4 kV bus is characterized by a different recovery curve than that of offsite power.

3. Human Error – If the core damage is the result of human error, then with more time available after core damage and additional alarms as a result of core damage, it is possible that the operators could recover from the error and initiate safety injection to prevent vessel breach. The type of human error can be easily identified by the names of the human error events used. If the error is diagnostic then the attribute is "D". If it is an action error is involved then the attribute is "A". If no human error or an unrecoverable human error is involved then the attribute is "N".

4. RCS Status at Onset of Core Damage – Based on the thermal hydraulic analysis described in Volume 2 of this report, the RCS pressure could reach 600 psi if core damage occurs in time window 1 with only 1 PORV open to relieve system pressure. This is a condition where the potential exists for direct containment heating (DCH) if the core debris melts through the reactor pressure vessel. For those window 1 cutsets in which the pressurizer safety valves are not removed, a letter "G" is assigned. For these cutsets, only two or less 2 PORVs are potentially available to relieve the pressure. For all other configurations the RCS is expected to be at low pressure, which is designated by letter "L".

5. ECCS Status – This question determines the cause of failure of the emergency core cooling system (ECCS), which in turn influences the possibility for restoring safety injection to prevent vessel breach.

   U: This identifies hardware failure as the reason why ECCS is not available. Under these circumstances it is not possible to establish safety injection.

R: If the cause of ECCS failure is due to either human error, loss of offsite power, or loss of the 4 kV bus, recovery from these events would allow coolant injection to prevent vessel breach.

C: This indicates that ECCS fails during recirculation. The main cause of recirculation failure is plugging of the sump suction. Recovery of ECCS was considered unlikely under these circumstances.

6. Recirculation Spray Status – The operation of the recirculation spray system can reduce the airborne fission product aerosol concentration in containment after core damage. The unavailability of the recirculation spray was determined by a set of basic events that were identified by reviewing the cutsets involving loss of recirculation spray. If the recirculation spray is recoverable then the attribute is "R" if the spray is unrecoverable then "U" is assigned.

7. RWST Status – The RWST inventory is needed if the ECCS is recovered after loss of power or human error. The RWST would not be available for those sequences in which failure occurs in recirculation, gravity feed is successful, or use of the Unit 2 charging pump is successful.

5.3 Plant Damage State (PDS) Results

In this section the allocation of the cutsets to the plant damage states is described. Additional details on how individual cutsets were assigned to particular PDSs is given in Appendix A. Initially an algorithm was developed for assigning the cutsets to the various plant damage states using the characteristics discussed in Section 5.2. Designators within the cutsets were identified and matched with the alphanumeric descriptors of the PDS. A total of 48 PDSs were obtained when the algorithm was applied to the 2186 core damage cutsets. The Level 1 analysis generated 10,000 LHS observations using the IRRAS code.[4] Each sample calculated a frequency for each of the 48 PDSs. The uncertainty ranges included in Appendix A for the 48 PDSs are therefore based on 10,000 LHS observations.

The 48 PDSs generated by the Level 1 analysis were regrouped into four PDS groups to be processed by the accident progression event tree analysis. The main consideration in regrouping was again the similarity of accident progression after core damage and for the convenience of the event tree logic. Appendix A indicates how the 48 PDSs were placed into four more general PDS groups.

It was noted above that the distributions for the 48 PDSs were obtained using 10,000 LHS observations. It would be impractical however to use 10,000 observations for processing through the accident progression analysis and it was therefore decided to use 100 LHS observations. It was therefore necessary to regenerate the uncertainty ranges based on 100 LHS observations for the four PDS groups for input to the accident progression analysis. The uncertainty ranges for the four PDS groups obtained from the smaller number of observations are given in Table 5.5. The uncertainty ranges for the various time windows within each PDS group are also given in Table 5.5.

The distributions obtained using 10,000 LHS observations differed slightly from the distributions based on only 100 LHS observations. Four statistical measures for the core damage frequency obtained from both sets of observations are given below:

	Core Damage Frequency Based on 10,000 LHS Observations	Core Damage Frequency Based on 100 LHS Observations
95th Percentile	1.5E-5	1.94E-5
Mean	4.9E-6	4.2E-6
50th Percentile	2.1E-6	2.0E-6
5th Percentile	4.8E-7	3.2E-7

One can see that the median values of the two distributions are close but that the mean values are further apart. This is to be expected because the means are more influenced by the tails of the distribution. However, the distributions are close and risk numbers obtained using the smaller number of observations should be valid.

From an inspection of the information in Table 5.5 it is apparent that accident sequences in which the operators did not correctly diagnose the situation or take proper actions (plant damage state 2) were the largest contributor (approximately two-thirds of the total) to the mean core damage frequency for mid-loop operation. Table 5.5 also indicates that most of the accidents are predicted to occur in the earlier time windows. This is expected because less time is available for recovery actions in the earlier time windows.

Each of the four plant damage state groups are briefly described below:

PDS Group 1 consists of five blackout PDSs. The PDSs in this group contribute approximately 10% to the mean core damage frequency. The accidents belonging to this group are initiated by a loss of offsite power. The diesel generators fail to respond, causing a station blackout condition. Attempts are unsuccessful in restoring power in time to provide cooling before core damage. In some of the accidents, gravity feed and bleed delays the onset of core damage until the water in the RWST runs out. The recirculation and containment systems are not available due to the loss of power. In this PDS group, the dominant factor in the accident progression is the recovery of offsite AC power.

PDS Group 2 consists of accidents attributable to human error. This PDS group contains 20 PDSs and is the largest contributor to the mean core damage frequency for accidents initiated by internal events during mid-loop operation at Surry. About two-thirds of all core damage accidents belong to this group. In this group operators either fail to diagnose the accidents or to take correct actions, following loss of core cooling due to some initiator. The progression of accidents is somewhat different depending on whether the human error is diagnostic or action. For example, if it is a diagnostic error, it is assumed that the same error results in failure to recognize the need for isolating the containment. However, if the error is in action, it was assumed that the containment would most likely be closed before core damage. In most cases, electric power and some core cooling systems are available. In this PDS group, the dominating factor in the accident progression is recovery from human errors.

PDS Group 3 consists of accident sequences where core cooling is lost during recirculation. This group contains 17 PDSs and accidents in this group contribute about 18% to the mean CDF for Surry. The accidents in this group occur only in Time Windows 1 and 2. In Windows 3 and 4 core cooling does not require recirculation within 24 hours which is the mission time for the Level 1 analysis. In this group, core cooling was successfully initiated and continued until the RWST was emptied; but the recirculation failed and the accident progressed to core damage. The leading cause of recirculation failure was found to be plugged suction from the sump.

PDS Group 4 consists of accidents where core cooling is lost because of loss of the 4 kV bus. This PDS group contains six PDSs and contributes about 5% to the mean CDF. There are no occurrences of this PDS in Windows 3 and 4. The accidents in this group are similar to those of PDS group 1 (SBOs) except that accidents are initiated by loss of the 4 kV bus. This group is separated from Group 1 since the recovery probabilities are different. The progression of accidents in this group are similar to those in Group 1.

5.3 References

1. Chu, T. L., et al., "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry Unit-1: Analysis of Core Damage Frequency from Internal Events During Mid-loop Operations," Brookhaven National Laboratory, NUREG/CR-6144, Volume 2, December 1993.

2. Breeding, R. J., et al., "Evaluation of Severe Accident Risks: Surry Unit 1," NUREG/CR-4551, Vol. 3, Rev. 1, Parts 1 and 2, Sandia National Laboratories, October 1990.

3. Embrey, D.E., et al., "SLIM-MAUD: An Approach to Assessing Human Error Probabilities Using Structured Expert Judgment," Brookhaven National Laboratory, prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-3518, Vols. 1-2, March 1984.

4.. Russell, K. D., et al., "Integrated Reliability and Risk Analysis System (IRRAS) Version NEW5," Developed by Idaho National Engineering Laboratory, 1993.

Table 5.1 Estimated Average Durations (Hours) of Plant Operational States (based on data from 1985-1989)

Plant Operational State	Refueling	Drained Maintenance	Non-Drained Maintenance (with RHRS)	Non-Drained Maintenance (w/o RHRS)
1	0.6	0.7	0.1	0.6
2	22.3	15.1	12.3	15.0
3	10.7	13.6	16.8	-
4	154.4	196.3	127.9	-
5	45.5	20.2	-	-
6	183.0	202.0	-	-
7	374.0	-	-	-
8	810.8	-	-	-
9	206.0	-	-	-
10	107.0	-	-	-
11	118.0	44.1	-	-
12	1840.0	175.0	-	-
13	34.4	10.3	-	-
14	69.0	40.4	21.0	-
15	56.1	12.7	18.6	9.9

Table 5.2 Time to Mid-loop (hours)

POS	Mean	5th Percentile	50th Percentile	95th Percentile
POS 6 Refueling	191	72	168	389
POS 10 Refueling	2619	833	968	4828
POS 6 Drained Maintenance	190	27	105	618

Table 5.3 Duration of Mid-loop (hours)

POS	Mean	5th Percentile	50th Percentile	95th Percentile
POS 6 Refueling	238	14	112	876
POS 10 Refueling	444	6	151	2586
POS 6 Drained Maintenance	255	12	109	958

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Table 5.4 Definition and Characterization of Time Windows*

	WINDOW 1	WINDOW 2	WINDOW 3	WINDOW 4
Definition	≤ 75 hours	> 75 hours and ≤ 240 hours	> 240 hours and ≤ 32 days	> 32 days
Representative Decay Heat	13 MW (2 days)	10 MW (5 days)	7 MW (12 days)	5 MW (32 days)
Time to Boiling	15 min.	20 min.	27 min.	37 min
Time to Tygon Tube Rupture (40 psia)	23 min.	31 min.	43 min.	59 min.
Time to PRT Rupture (100 psig)	51 min.	63 min.	78 min.	96 min.
Time to 165 psia	41 min. with 2 PORV 43 min. with 1 PORV	63 min. with 2 PORV 60 min. with 1 PORV	227 min. with 2 PORV 89 min. with 1 PORV	352 min. with 2 PORV 147 min. with 1 PORV
Time to 615 psig	145 min. with 1 PORV	—	—	—
Time to RWST Depletion	10 hrs	13.5 hrs	18.7 hrs	38.6 hrs
Time to AFW Initiation (with 25% SG inventory remaining)	743 min.	669 min.	925 min.	628 min.
Time to Core Uncovery	120 min.	157 min.	209 min.	273 min.
Time to Core Damage	219 min.	297 min.	411 min.	557 min.

*Reproduced from Table 5.4-2 in Volume 2 of this report.

Table 5.5 Distribution of Core Damage Frequencies for Each Time Window and PDS Group (per reactor year)

PDS Group	TIME WINDOW	5th Percentile	50th Percentile	95th Percentile	MEAN
1 (Station Blackout)	1	3.32E-9	4.67E-8	9.39E-7	1.98E-7
	2	3.11E-9	3.35E-8	7.81E-7	1.33E-7
	3	1.60E-9	1.36E-8	2.00E-7	4.05E-8
	4	8.56E-10	1.07E-8	8.87E-8	2.38E-8
	Total	1.87E-8	1.20E-7	1.65E-6	3.95E-7
2 (Human Error)	1	8.19E-9	1.27E-7	2.32E-6	4.90E-7
	2	6.07E-8	3.55E-7	6.07E-6	1.24E-6
	3	4.80E-8	2.83E-7	4.12E-6	9.14E-7
	4	7.49E-9	6.51E-8	7.43E-7	1.85E-7
	Total	2.07E-7	1.03E-6	1.28E-5	2.83E-6
3 (Recirculation Failure)	1	3.06E-8	3.34E-7	3.20E-6	6.69E-7
	2	7.37E-9	4.64E-8	3.94E-7	1.08E-7
	3	0	0	0	0
	4	0	0	0	0
	Total	5.34E-8	4.08E-7	3.26E-6	7.77E-7
4 (Loss of 4 kV Bus)	1	2.13E-9	3.86E-8	5.45E-7	1.33E-7
	2	2.22E-9	2.82E-8	4.21E-7	8.82E-8
	3	0	0	0	0
	4	0	0	0	0
	Total	8.87E-9	8.88E-8	1.32E-6	2.21E-7
Total		3.18E-7	1.99E-6	1.94E-5	4.22E-6

6 ANALYSIS OF THE ACCIDENT PROGRESSION

The accident progression analysis starts with information received from the accident frequency analysis: frequencies and definitions of the plant damage states (PDSs). The results of the accident progression analysis are passed to the source term analysis and the risk analysis. The methods used in this accident progression analysis are similar to those used in the NUREG-1150[1] study. Details of the methodology and the results of the PRA performed for accidents during full power operation at Surry are presented in NUREG/CR-4551, Vol. 3.[2]

The main tool for performing the accident progression analysis is an event tree. The Accident Progression Event Tree (APET) treats the progression of an accident from the onset of core damage to the release of fission products, if any, or a successful termination of the accident. The APET involves modelling of the physical processes occurring in the vessel and containment during the various accident sequences (such as hydrogen burning and direct containment heating), and the availability and status of various safety equipment which could be used to mitigate the severity of the accident (such as safety injection systems and containment sprays). The APET also evaluates the capability of the containment to retain the fission products under severe accident loads. A series of questions are asked which represent these events and phenomena. Each path through the APET defines a unique accident progression path that potentially could give rise to the release of fission products. The number of questions in a APET can vary, depending on the details desired, and the number of relevant, important phenomena to be modelled.

The APET is not meant to be a substitute for detailed, mechanistic codes, rather it forms a high-level model of the accident progression. The APET is an integrating framework for synthesizing the results of these codes together with expert judgment on the strengths and weaknesses of the codes. In this way, the full diversity of possible accident progressions can be considered and the uncertainty in the many phenomena involved can be included. The APET was evaluated by the computer code EVNTRE.[3]

The following section contains an overview of the APET used for this study. Details, including a complete listing of the APET and a discussion of the possible outcomes to each question, may be found in Appendix B of this report. Section 6.2 summarizes how the APET is quantified. It explains how the many numerical values for branching ratios and parameters were obtained. Section 6.2 also lists the variables that were sampled in the accident progression analysis for Surry shutdown study. A brief summary of supporting deterministic calculations is provided in section 6.3. Section 6.4 describes the binning process and the binning characteristics. The results of the accident progression analysis for each PDS and "Time Window" are presented in Section 6.5. Section 6.6 discusses the sensitivity of conditional containment failure probability due to the containment failure pressure.

6.1 Description of the Accident Progression Event Tree

In constructing the APET for Surry during mid-loop operation, extensive use was made of the results of the accident progression analysis for the Surry plant[2] carried out for the NUREG-1150 program, which was a PRA of the plant at full power. The NUREG-1150 study showed that the major cause of fission product release was from accidents in which the containment was bypassed, followed by basemat melt-through (BMT) by the molten core debris. The study indicated that phenomena such as direct containment heating (DCH) or steam explosions were not important contributors to the estimated probability of containment failure and the eventual release of fission products. Nor did hydrogen burning or gradual pressurization of the containment significantly contribute to the containment failure. Thus, an important finding of the full power study was that once the Surry containment boundary is closed, the containment retains the fission products most of the time (except for very late basemat melt-through) even when excessive core damage occurs. For accidents during low power and shutdown operation the decay heat is significantly less, the reactor pressure is generally low and the pressure generated in the containment is lower than for accidents occurring at full power. Therefore, early containment failure modes such as DCH and hydrogen burning could be excluded in the low power and shutdown risk study if the capability of the containment to hold pressure is the same as that of full power.

However, containment failure pressure was assumed to be 45 psig in this study while the results of full power risk analysis were based on a containment failure pressure of 126 psig as discussed in section 4.2.3. Therefore, containment failure caused by such phenomena as DCH and hydrogen burning could not be dismissed based on the full power analysis results and thus is included in this study. A sensitivity study with an assumed failure pressure of 126 psig was also performed as discussed in Section 6.6.

The APET for this study contains 40 questions while the full power study has 71 questions. Table 6.1 lists the 40 questions used in the Surry APET for accidents during mid-loop operation. The complete listing of the APET and a discussion of each question is found in Appendix B. The APET for mid-loop operation is largely based on the APET of the NUREG-1150 full power study. It was modified to reflect the conditions during mid-loop operation by removing some questions and adding several questions pertinent to the shutdown conditions.

Some of the modifications made to the NUREG-1150 APET to reflect the conditions during mid-loop operation are listed below:

1) Questions on fan coolers were removed. They were not relevant even in the full power study.

2) Questions on accumulators were removed; accumulators are blocked out during mid-loop operation.

3) Questions on Interfacing Systems LOCAs were removed; the level 1 analysis[4] does not have these sequences.

4) A question on scram was removed since the reactor has already been successfully shutdown.

5) Questions concerning steam generator tube ruptures and heat removal by the steam generators were removed. Level 1 results do not include steam generator tube rupture sequences. Failure of heat removal by steam generators is already considered in the level 1 analysis.

6) Several questions on temperature-induced reactor coolant pump seal failure and other breaks were removed. This phenomenon contributes to reducing the high pressure vessel failure probability. Since the majority of accidents initiated during the shutdown period are with the reactor vessel at low pressure (only a very small fraction of the accidents during Window 1 are at intermediate pressure) as discussed in the previous chapter (Section 5.2), these questions were considered not pertinent.

7) Containment failure by the "rocket mode" were removed; this failure mode requires a high vessel pressure (2500 psi) which is not possible for accidents during mid-loop operation.

8) Containment failure by BMT was not included in the tree because MELCOR[5] calculations showed that the rate of erosion of the basemat is very slow compared to the thickness of the basemat (refer to section 6.3).

9) Containment failure by gradual overpressurization was not considered to be feasible based on MELCOR calculations which indicated very slow pressurization rates (refer to section 6.3).

10) A question on the "Time Window" was added.

11) Questions on the status of containment closure and containment pressure capability were added.

12) Questions on human error and recovery from human error were added; human error is the largest cause of accidents during mid-loop operation in the level 1 analysis (Section 5.2).

13) Questions on recirculation failure were added; failure of recirculation due to sump plugging is a significant cause of accidents during mid-loop operation in the level 1 analysis (Section 5.2).

14) Questions on loss of a 4 kV bus and its recovery were added; loss of the 4 kV bus and the resultant loss of the injection pumps was found to be a significant cause of accidents during mid-loop operation in the level 1 analysis (Section 5.2).

15) Since the results of the NUREG-1150 study indicated that the pressure generated by hydrogen burning could be substantially above 45 psig, it was assumed that a large hydrogen burn would fail containment if it occurred (i.e., detail calculations on the magnitude of the pressure generated during a hydrogen burn were not performed specifically for this study). The APET still includes questions on whether conditions exist for hydrogen ignition in the containment.

The APET is broken into five time periods. The time periods are:

Initial: Questions 1 through 11 determine the conditions at the beginning of the accident.

Early: Questions 12 through 20 cover the in-vessel accident progression period up to the time of vessel breach (VB).

Intermediate: Questions 21 through 25 determine the progression of the accident at and immediately after vessel breach (VB), including the possibility of containment failure at VB.

Late: Questions 25 through 37 determine the progression of the accident during core-concrete interaction (CCI).

Very Late: Questions 38 through 40 determine the accident progression in the period following CCI, including the possibility of containment failure due to hydrogen combustion.

The clock time for each period varies depending on the time window when the accident is initiated. The table below shows these time periods for each time window based on the results of MELCOR calculations. Details of the MELCOR calculations are provided in Appendix F.

Accident Timing Used in the Surry APET for Each Time Window (Minutes Measured from the Start of the Accident)

	Window 1	Window 2	Window 3	Window 4	Remarks
End of Early Period	90	125	170	240	Core damage
Intermediate Period	216	300	364	530	Vessel breach
End of Late Period	400	480	550	710	Prompt CCI
Very Late	1440	1440	1440	1440	24 hrs

6.2 Overview of the APET Quantification

This section discusses the types of questions used in the APET and summarizes the quantification method.

In addition to the number and name of the question, Table 6.1 shows how the questions were evaluated or quantified. If the question is sampled, the table also indicates how it is sampled, i.e., if the distribution is based on the distribution of frequencies of core damage accidents and PDSs provided by the level 1 analysis, assigned internally by BNL staff, from the electric power recovery distribution,[2] from recovery distribution for human errors,[6] or from a distribution provided by one of the expert panels of the NUREG-1150 study. The item sampled may be either the branching ratio or the parameters defined at that question. For questions which are sampled and which were quantified internally, the entry ZO in the sampling column indicates that the question was sampled zero-one, and the entry SF means the questions was sampled with split fractions. If the sampling column is blank, the branching ratios for that question, and the parameter values defined in that question, if any, are fixed. The branching ratios of the PDS questions change to indicate which PDS is being considered. Some of the branching ratios depend on the relative frequency of the PDSs which make up the PDS group being considered. These branching ratios can change for every sample observation, and may change for some PDS groups but not for others. If the branching ratios change from observation to observation for any one of the four PDS groups, SF is placed in the sampling column for the PDS questions. The abbreviations in the quantification and sampling columns of Table 6.1 are explained at the end of the table.

Twenty-one variables, listed in Table 6.2, are sampled for the accident progression analysis. That is, every time the APET was evaluated by EVNTRE, the original values of these 21 variables were replaced with values selected for the particular observation under consideration. These values were either based on the distribution of accidents provided by the level 1 analysis or selected by the LHS program[7] from distributions that were defined before the APET was evaluated. As explained earlier, these distributions were determined internally or information provided by expert panels in the NUREG-1150 analysis. Some are branch fractions, others are parameter values used in the calculations such as pressure generated by DCH. Several variable listed are used to select the probability that off-site power will be recovered in a specified time interval given that it was not recovered in a previous time interval, each with different start and end times, for each time window. There are similar questions regarding recovery from loss of 4 kV bus or human errors. Additional information on the ranges of distribution of these variables can be found in Appendix B of this volume.

6.3 Supporting Deterministic Calculations

Several calculations were performed with the MELCOR code[5] to support the determination of the various time windows and associated success criteria and also to help quantification of the APET. Predictions of the MELCOR code were also used to compare against the source term distributions calculated by the SURSOR code (refer to Chapter 7). The MELCOR calculations are described in Appendix F, which also includes detailed results for several possible accident scenarios during mid-loop operation.

The major impact of the MELCOR calculations on the APET quantification relates to two potential containment failure mechanisms, namely basemat melt-through by the molten core debris and overpressurization of the containment by steam and noncondensible gases.

6.3.1 Basemat Melt-through

As noted above this failure mechanism was found to be a significant cause of fission product releases for accident during full power operation although the core debris was determined to penetrate the basemat very late in an accident sequence. However, the MELCOR calculations presented in Appendix F indicate much slower concrete erosion rates for accidents during mid-loop operation. This lower concrete erosion is caused by the relatively low decay heat for accidents during mid-loop operation. The erosion depth was calculated to be about 0.75 m (compared with a basemat thickness of 3 m) 30 hours after the start of an accident in time window 1 (recall that the representative decay heat is highest in this time window). Even in the full power analysis, it was calculated to take several days to breach the basemat. Since the probability of not recovering some safety injection system in this time period is extremely small, it was determined that basemat melt-through is not a credible failure mode for accidents during mid-loop operation.

6.3.2 Containment Overpressurization

Overpressurization of the containment by steam and noncondensible gases was found to be not a credible failure mode for accidents during mid-loop operation also based on MELCOR calculations. This is true even if the containment is assumed to leak at pressures above 45 psig. Again the low decay heat levels associated with accidents during mid-loop operation means that the driving force for containment pressurization is also low and the rate of pressurization is very slow. Detailed results are presented in Appendix F.

6.4 Description of the Accident Progression Bins

As each path through the APET is evaluated, the result of that evaluation is stored by assigning it to an Accident Progression Bin (APB). The accident progression bins are the means by which information is passed from the accident progression analysis to the source term analysis. A bin is defined by specifying the attribute or value for each of 12 characteristics or quantities which define a certain feature of the evaluation of the APET. The definition of APBs and the method of their assignment in this study is similar to those of the NUREG-1150 study except for one very important difference; an additional attribute was added to characterize the "Time Window". The "Time Window" information is passed to the consequence calculation to account for the different fission product decay times. Table 6.3 lists the 12 characteristics of the APBs: the detailed listing of the attributes of these characteristics may be found in Appendix B. The binner, which follows directly after the APET is the data file which forms the input to EVNTRE, is also listed in Appendix B. Some of the bin characteristics such as SGTR and RCS-hole size are not relevant or significant for the mid-loop risk analysis, but were still included to match the requirement of the SURSOR code.[8] The "Time Window" characteristic is not required by the SURSOR code but is passed to the consequence analysis.

Characteristic 1 primarily concerns the time of containment failure. There are seven attributes. Four of these attributes concern the time of containment failure, two concern Event V, and one is for no containment failure. Interfacing systems LOCAs (Event V) were not applicable to mid-loop operation and therefore attributes A and B were not used. BMT and eventual overpressure failure due to the inability to restore containment heat removal in the days following the accident were the failures that occurred in the Final period in the full power study. These failure modes were determined to be not credible for accidents during mid-loop operation and therefore attribute F was not used.

Characteristic 2 concerns the periods in which the sprays operate. The division into the nine attributes is a straightforward sorting out of the various combinations of time periods. The final time period is of little consequence for the fission product release, but it must be included because there are cases where the sprays operate only in this period and, for each characteristic, the binner must have a location in which to place every outcome.

Characteristic 3 concerns the CCIs. There are six possibilities which cover the meaningful combinations of prompt CCI, delayed CCI, and no CCI, with the amount of water in the cavity.

Characteristic 4 concerns the pressure in the reactor vessel before vessel breach; there are four levels. The pressures shown in parentheses are approximate pressures just before VB. The RCS pressures during most of the core degradation period for accidents in mid-loop were in the intermediate to low pressure range. Attributes A and B were therefore not used.

Characteristic 5 concerns the mode of vessel breach; there are six possibilities, including no VB.

Characteristic 6 concerns SGTR. Steam generator tube ruptures were not identified in the Level 1 analysis so that the attribute for this characteristic was always C.

Characteristic 7 concerns how much of the core not in HPME that is available to participate in the CCI. The fractions 0.30 and 0.70 divide the range into three portions. The fourth attribute is no CCI. As SURSOR subtracts out the fraction of the core involved in HPME, when HPME occurs the fraction of the core available for CCI is always set to Large.

Characteristic 8 concerns the amount of the core zirconium which is oxidized in-vessel before vessel breach. There are two possible values for this characteristic: low and high. The demarcation point between the two ranges is 40%.

Characteristic 9 concerns the amount of the core involved in HPME; there are four attributes. The possible range is divided into three portions by 20% and 40%. No HPME is the fourth attribute.

Characteristic 10 concerns the size of the hole that results from containment failure or the type of containment failure. There are four possible attributes.

Characteristic 11 concerns the number of large holes in the RCS after breach. The experts on the Source Term Expert Panel assembled for the NUREG-1150 study who provided distributions for revolatilization from the RCS surfaces after VB gave different distributions depending on whether an effective natural circulation flow would be set up within the vessel. A significant flow could be expected only if there were two large, effective holes in the RCS.

Characteristic 12 concerns the time at which the accident occurs after the reactor has been shutdown. Four time windows are possible as defined in the Level 1 analysis.

A set of "summary" bins was adopted for presentation purposes, as in NUREG-1150. Instead of the 12 characteristics and the hundreds of possible bins that describe the evaluation of the APET in detail, the summary bins place the outcomes of the evaluation of the APET into a few, very general number of groups. They are:

No VB, No CF
No VB, Open Containment
VB, No CF
VB, Open Containment
VB, CF (including steam explosions, DCH, & Hydrogen burn)

In some cases the DCH and hydrogen burn failure are reported separately and in other cases the results are presented without distinguishing between vessel breach and no vessel breach.

The "Open Containment" group includes leakage through the equipment hatch or other temporary barriers (which can occur even after "successful" isolation of containment) as well as failure to isolate containment before the onset of core damage.

6.5 Results of the Accident Progression Analysis

This section presents the results of the APET evaluation. As evaluating the APET produces a large number of accident progression bins (APBs), the discussion is primarily focused on events that result in core damage arrest before vessel breach, and loss of containment integrity. Therefore Tables 6.4 and 6.5 were included to provide information on the distributions of conditional probabilities for core damage arrest and loss of containment integrity. The term "loss of containment integrity" is used for this mid-loop study rather than "containment failure" because of the importance of failure to isolate containment, which is not really "failure" of containment. The term "containment failure" is reserved for energetic events (such as steam explosions, DCH and Hydrogen Combustion) that cause structural failure of the containment. Four statistical measures of the conditional probability distributions are included in Tables 6.4 and 6.5.

In order to assess the relative importance of the various accident progression bin groups the mean conditional probabilities are given for all four plant damage state groups in Tables 6.6 and 6.7. The APB groups in which the status of the reactor pressure vessel (no vessel breach or vessel breach) and the containment are identified in Table 6.6. The APB groups in which only the containment status is given are included in Table 6.7. Similar information is given for the various time windows in Table 6.8. Although information on the contribution of the mean of the various distributions is helpful the results should also be displayed with more information on the distribution. Therefore Table 6.9 and Figure 6.1 were included to provide additional statistical measures for the distributions of the frequencies for various accident progression bin groups.

Generally, the containment failure probability is dominated by the probability of whether the containment is successfully isolated prior to core damage. Containment failure due to energetic pressurization either because of DCH or hydrogen burning is relatively small as in the full power study even if the containment is assumed to leak at pressures above 45 psig. This is partly because the fraction of accidents with high or intermediate vessel pressure is very small, and partly because the fraction of accidents where the containment was not isolated is high. Very late containment failure due to basemat melt-through and gradual pressurization due to loss of containment cooling was assumed not to happen based on the results of MELCOR calculations as discussed in section 6.3.

6.5.1 Results for Each PDS Group

6.5.1.1 Results for PDS Group 1: Station Blackout (SBO)

This PDS group contains five PDSs as discussed in Section 5.2. The PDSs in this group contribute approximately 10% to the mean total core damage frequency. The accidents belonging to this group are initiated by a loss of off-site power and coupled with other failures result in a station blackout. The recirculation and containment systems are not available due to the loss of power. In this PDS, an important factor in the accident progression is the recovery of the off-site AC power.

The mean conditional probability of core damage arrest prior to vessel failure ranges from approximately 0.5 for Time Window 1 to 0.7 for Time Window 4 as shown in Table 6.4. The probability of arresting core damage generally increases with each Time Window (as expected) because more time is available in the later time windows to recover the power before vessel breach. The mean conditional probability averaged over all four time windows is about 0.55.

The mean conditional probability of loss of containment integrity for this PDS group averaged over all time windows is approximately 0.51 as shown in Table 6.5. This conditional probability slightly decreases as the time window increases (0.54 for window 1 vs. 0.45 for Window 4). This is also expected as loss of containment integrity is largely attributable to failure to isolate containment. Therefore as the time available to isolate containment increases the conditional probability of loss of containment integrity should decrease. Energetic containment failure is significant for this PDS group, with a conditional probability of about 0.15 (Table 6.7). This mostly comes from hydrogen burning later in the accident. This mode of failure is prominent in this PDS group, since hydrogen burning is more likely when the power is recovered after a substantial amount of hydrogen has accumulated in the containment. If the containment failure pressure is assumed to be 126 psig (as in full power), then this mode of failure becomes very unlikely as shown in the sensitivity calculation (refer to section 6.6), because the pressure generated by hydrogen burning is substantially below 126 psig.

6.5.1.2 Results for PDS Group 2: Human Errors (HX)

This PDS group contains a large number of PDSs and is the largest contributor to the internal event core damage frequency for mid-loop operation at Surry. About two thirds of all core damage accidents belong to this group. The accidents belonging to this group are attributable to human errors. Following loss of core cooling due to some initiators, operators either fail to diagnose the accidents or to take correct actions. The progression of accidents is somewhat different depending on whether the human error is in diagnosis or action. For example, if it is a diagnostic error, then it is assumed that the same error results in failure to recognize the need for containment isolation. If the error was a failure to take the correct action, it was more likely that the containment was closed before core damage. In most cases, the electric power and some core cooling system are available. In this PDS group, the dominating factor in the accident progression is the recovery from human errors.

The mean conditional probability of core damage arrest without vessel failure is about 0.42 averaged over all windows (Table 6.4). This probability is lower than that of PDS group 1 indicating that the recovery probability from human error is less likely than recovery of electric power once the accident progresses to core damage.

The mean conditional probability of loss of containment integrity for this PDS group is very high, about 0.9 (Table 6.5). This result reflects the assumption that the containment would most likely remain unisolated in this PDS group. Energetic containment failure is insignificant for this PDS group (Table 6.7). Since this PDS group is the largest contributor to the core damage frequency, it significantly contributes to the overall probability of loss of containment integrity.

6.5.1.3 Results for PDS Group 3: Recirculation Failure

The PDSs in this group contribute about 18% to the TCDF for Surry, although it contains a large number of PDSs. The accidents in this group occur only in Windows 1 and 2. In this group, core cooling was successfully initiated and was continued until the RWST is emptied; but the recirculation fails and the accident progresses to core damage. The conditional probability of core damage arrest before vessel failure in this PDS group is zero (Table 6.4) since it is assumed that core cooling is permanently lost once recirculation is lost.

The mean conditional probability of loss of containment integrity for this PDS group is relatively low, about 0.13 (Table 6.5). The probability of isolating the containment in this PDS group is considered to be high because core cooling is established and the reactor has been in a stable condition for a relatively long time. Energetic containment failure is unimportant for this PDS group, contributing only about 3% to containment failure (Table 6.7), mostly from DCH. Hydrogen burning is not likely to fail containment since power is available and hydrogen combustion was determined to occur at relatively low concentrations.

6.5.1.4 Results for PDS Group 4: Loss of 4 kV Bus

This PDS group contributes about 5% to the mean core damage frequency for accident in mid-loop. There are no occurrences of this PDS in Windows 3 and 4. The accidents in this group are similar to those of PDS group 1 (SBOs) except that accidents are initiated by loss of 4 kV bus. This group is separated from Group 1 since the recovery probabilities are different, however, the accident progression for this group is similar to that of Group 1.

The mean conditional probability of core damage arrest without vessel failure was determined to be about 0.6 (Table 6.4) which is slightly higher than that of Group 1. The mean conditional probability of loss of containment integrity for this PDS group is approximately 0.45 (Table 6.5), which is about same as Group 1. Hydrogen burning is a significant contributor to the conditional containment failure probability as in Group 1 (Table 6.7).

6.5.2 Core Damage Arrest and Avoidance of Vessel Breach

It is possible to arrest the core damage process and avoid vessel breach if injection is restored before the core degradation process has gone too far. Recovery of injection depends on the PDS groups. For Groups 1 (SBO) and 4 (Loss of 4 kV bus) the dominant factor is recovery of the off-site A/C power or the 4 kV bus. For Group 2 (Human Errors), recovery depends on the operators making correct diagnosis or taking proper actions. For Group 3 (Failure of Recirculation), it is assumed that no recovery action is possible once core damage occurs.

Table 6.4 shows four statistical measures for the distributions of the conditional probability of halting the degradation of the core before the lower head of the vessel fails, for each PDS and Time Window. Overall the mean conditional probability of core damage arrest without vessel failure is about 0.35 for all windows and PDS groups. The core damage arrest for each PDS is discussed above. For each window the average conditional probability of core damage arrest is roughly similar to the conditional probability of PDS Group 2. This result reflects the fact that PDS Group 2 is the largest contributor to the total core damage frequency (refer to Figure 5.1). The average conditional probability of core damage arrest for window 1 is lower than for the other windows. This is mainly because PDS Group 3 is significant contributor in this window (Table 5.8) and the core damage arrest probability for PDS Group 3 is zero.

6.5.3 Loss of Containment Integrity

There are four possibilities for loss of containment integrity:

1. Failure to isolate containment;
2. Containment leak due to failure of isolation barriers;
3. CF at VB due to the events at VB; and
4. CF due to hydrogen combustion before or after VB.

Very late containment failure due to gradual pressurization caused by the loss of containment cooling, or due to basemat melt-through was assessed to be very unlikely based on the results of MELCOR[5] calculations.

Table 6.5 shows the conditional probability of loss of containment integrity for each PDS and Time Window. The overall mean conditional probability of loss of containment integrity is about 0.67 for all windows and PDS groups. There are no apparent trends among the time windows for the overall probability, although some trends were observed in each PDS as discussed above. The trend in overall probability is obscured by the different composition of the PDS groups for each window. For example, in Window 1, the probability of loss of containment integrity is relatively low since the contribution of PDS group 3 to the window is high and this PDS group has a low probability of containment failure.

Table 6.7 shows the mean conditional probability of the APB groups in which only containment status is identified (isolation failure and early leak are combined into 'Open Containment'). The table shows that the conditional probability of loss of containment integrity is dominated by the probability of whether the containment is successfully isolated prior to core damage. Containment failure due to energetic pressurization either by DCH or hydrogen burning is relatively small as for accidents at full power even if the containment failure pressure (45 psig) is much less than that of full power (126 psig). This is partly because the fraction of accidents with high or intermediate vessel pressure is very small (i.e., minimizing the potential for DCH) and partly because the conditional probability of the containment not being isolated at the start of the accident was high.

The range of uncertainties associated with various containment failure frequency estimates are generally narrower than the full power results. This is because of domination by the isolation failure and a relatively small contribution by other early failure modes which usually have much wider range of uncertainties.

6.6 Sensitivity Analyses for Containment Failure Pressure

This section reports the results of a sensitivity analysis performed to determine the effect of higher containment failure pressure (i.e., a failure pressure of 126 psig which was the mean failure pressure used in the NUREG-1150 full power analysis) on the conditional probability of containment failure by hydrogen combustion.

Table 6.10 shows the mean probability of the APB groups (in which the vessel and containment status is identified) assuming a mean containment failure pressure of 126 psig. By comparing the results in Tables 6.6 and 6.10 the impact of the higher failure pressure becomes apparent. The conditional probabilities of containment failure caused by such events as DCH and hydrogen combustion become very small when the higher failure pressure is assumed (Table 6.10). This is particularly obvious for PDS groups 1 and 4 where failure by hydrogen combustion was a significant contributor to the mean conditional probability of containment failure when the lower failure pressure was assumed (Table 6.6). However, as PDS Group 2 is the largest contributor to the risk estimates (refer to Chapter 9) and because containment failure by DCH and hydrogen are such small contributors to this particular plant damage state assuming that the containment failure pressure is higher has a small impact on the overall mean conditional probability of containment failure as shown in Table 6.11. This would not have been the case if PDS groups 1 and 4 had been larger contributors to the risk estimates.

6.7 References

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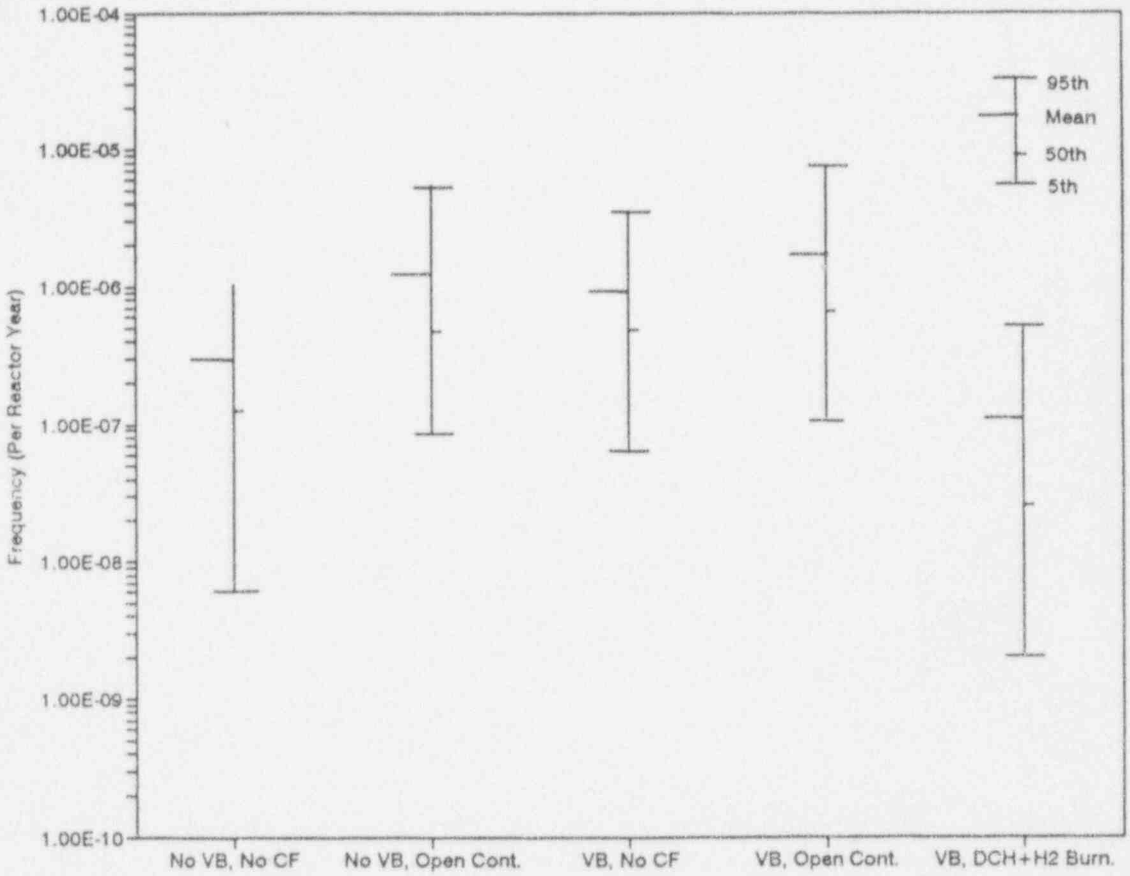

Figure 6.1 Distribution of Frequencies of APB Groups

Table 6.1 Questions in the APET

Question Number	Question	Quantification	Sampling
1	Which time window?	PDS	SF
2	Size of RCS break?	PDS	SF
3	RCS depressurization by the operators?		SF
4	Status of ac power?	PDS	ZO
5	Core damage accident due to human error?	PDS	ZO
6	Status of ECCS?	PDS	SF
7	Status of sprays?	PDS	SF
8	RWST injected into containment?	PDS	SF
9	Initial containment condition?	Internal	SF
10	Is containment isolated before core damage?	Internal	SF
11	Containment pressure capability	Internal	SF
12	Is ac power available early?	ROSP	DS
13	Recovered from human error before vessel breach?	HRAH	DS
14	Is core damage arrested before vessel breach?	Summary	
15	Vessel pressure just before vessel breach?	Internal	P
16	Does an alpha event fail both vessel and containment?	NUREG-1150	SF
17	Type of vessel breach?	NUREG-1150	ZO
18	Early sprays?	Summary	
19	Is the reactor cavity wet at vessel breach?	Summary	
20	Baseline containment pressure before VB?	Internal	P
21	Total pressure rise at vessel breach?	NUREG-1150	P
22	Does a significant ex-vessel steam explosion occur?	NUREG-1150	SF
23	Containment failure pressure?	NUREG-1150	DS
24	Containment failure at VB?	NUREG-1150	ZO
25	Containment status at VB?	Summary	
26	Is ac power available late?	ROSP	DS
27	Recovered from human error late?	HRAH	DS

Table 6.1 (continued)

Question Number	Question	Quantification	Sampling
28	Late sprays?	Summary	
29	Does late ignition occur?	NUREG-1150	P
30	Containment failure due to hydrogen burning?	Summary	
31	Status of containment?	Summary	
32	Is the debris bed in a coolable configuration?	NUREG-1150	SF
33	Does prompt CCI occur?	Summary	
34	Is ac power available very late?	ROSP	DS
35	Recovered from human error very late?	HRAH	DS
36	Very late sprays?	Summary	
37	Does delayed CCI occur?	Summary	
38	Does very late ignition occur?	NUREG-1150	P
39	Containment failure due to hydrogen burning?	Summary	
40	Final containment condition?	Summary	

Key to Abbreviations in Table 6.1

DS	The branch probabilities are taken from a distribution.
HRAH	Available from the Reliability Analysis Handbook (Ref. 6)
Internal	The quantification was performed at Brookhaven National Laboratory by the plant analyst with the assistance of other members of the laboratory staff.
NUREG-1150	Information from NUREG-1150 analysis
P	A parameter is determined from a distribution or deterministic calculations.
PDS	The quantification follows directly from the definition of the Plant Damage State.
ROSP	This question was quantified by sampling from a distribution derived from the off-site power recovery data for the plant.
SF	Split Fraction sampling—the branch probabilities are real numbers between zero and one.
Summary	The quantification for this question follows directly from the branches taken at preceding questions, or the values of parameters defined in preceding questions.
ZO	Zero-one Sampling: the branch probabilities are either 0.0 or 1.0.

Table 6.2 Variables Sampled in the Accident Progression Analysis

Question No.	Description
1	Frequencies of accident occurring at each time window for the given PDS group.
2	Pressure rise at VB when RCS pressure is at intermediate pressure and reality cavity is dry.
10	Probability that containment is successfully closed before core damage at each time window.
11	Probability that containment is isolated but does not have pressure retaining capability.
12	Probability that off-site power is recovered before vessel breach for each time window.
12	Probability that power is recovered to injection pumps from 4 kV loss for each time window.
13	Probability of recovery from human errors for each time window.
16	Probability that Alpha mode DF occurs, given that RCS is at low pressure
17	Probability that VB mode is pressure melt ejection (HPME) given that RCS is at intermediate pressure.
17	Probability that VB mode is bottom head failure
20	Baseline containment pressure just before VB.
21	Pressure rise at VB when VB mode is poor.
21	Pressure rise at VB when RCS pressure is at intermediate pressure and reactor cavity is wet.
23	Containment failure pressure when mean containment capability pressure is 162 psig.
23	Containment failure pressure when mean containment capability pressure is 45 psig.
26	Probability that off-site power is recovered late for each time window.
26	Probability that power is recovered late to injection pumps from loss of 4 kV bus for each time window.
27	Probability of late recovery from human errors for each time window.
34	Probability that off-site power is recovered very late for each time window.
34	Probability that power is recovered very late to injection pumps from loss of 4 kV bus for each time window.
35	Probability of very late recovery from human errors for each time window.

Table 6.3 Description of APB Characteristics

Characteristic	Abbreviation	Description
1	CF-Time	Time of containment failure
2	Sprays	Periods in which sprays operate
3	CCI	Occurrence of CCIs
4	RCS-Pres	RCS pressure before vessel breach
5	VB-Mode	Mode of VB
6	SGTR	Steam generator tube rupture
7	Amt-CCI	Amount of core available for CCI
8	Zr-Ox	Fraction of zirconium oxidized in-vessel
9	HPME	Fraction of the core in the high pressure melt injection (HPME)
10	CF-Size	Size or type of containment failure
11	RCS-Hole	Number of large holes in the RCS after VB 2.53
12	Window	Time window when core damage accident is initiated

Table 6.4 Conditional Probability of Core Damage Arrest

PDS Group / Time Window	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	All	Statistical Measures
Window 1	0.40	0.42	0.0	0.00	0.06	5th percentile
	0.49	0.45	0.0	0.64	0.20	Median
	0.49	0.44	0.0	0.58	0.22	Mean
	0.59	0.46	0.0	0.86	0.41	95th percentile
Window 2	0.48	0.38	0.0	0.05	0.28	5th percentile
	0.56	0.41	0.0	0.67	0.40	Median
	0.57	0.41	0.0	0.61	0.39	Mean
	0.67	0.43	0.0	0.87	0.46	95th percentile
Window 3	0.47	0.42	No Bins	No Bins	0.42	5th percentile
	0.55	0.45			0.45	Median
	0.55	0.44			0.45	Mean
	0.64	0.46			0.48	95th percentile
Window 4	0.59	0.38	No Bins	No Bins	0.40	5th percentile
	0.69	0.41			0.45	Median
	0.70	0.41			0.46	Mean
	0.83	0.43			0.56	95th percentile
All	0.45	0.39	0.0	0.03	0.23	5th percentile
	0.54	0.43	0.0	0.65	0.37	Median
	0.55	0.42	0.0	0.59	0.35	Mean
	0.71	0.45	0.0	0.86	0.44	95th percentile

Table 6.5 Probability of Loss of Containment Integrity* for Each PDS

PDS Group / Time Window	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	All	Statistical Measures
Window 1	0.18	1.00	0.02	0.12	0.18	5th percentile
	0.48	1.00	0.13	0.36	0.39	Median
	0.54	1.00	0.13	0.46	0.44	Mean
	0.97	1.00	0.23	0.97	0.81	95th percentile
Window 2	0.15	0.65	0.01	0.11	0.41	5th percentile
	0.42	0.91	0.10	0.33	0.76	Median
	0.50	0.87	0.11	0.44	0.73	Mean
	0.97	0.99	0.21	0.97	0.96	95th percentile
Window 3	0.15	0.69	No Bins	No Bins	0.63	5th percentile
	0.43	0.93			0.90	Median
	0.51	0.90			0.87	Mean
	0.97	0.998			0.997	95th percentile
Window 4	0.14	0.24	No Bins	No Bins	0.24	5th percentile
	0.33	0.71			0.65	Median
	0.45	0.66			0.63	Mean
	0.97	0.998			0.91	95th percentile
All	0.17	0.67	0.02	0.11	0.39	5th percentile
	0.44	0.93	0.12	0.35	0.70	Median
	0.51	0.89	0.13	0.45	0.67	Mean
	0.97	0.997	0.22	0.97	0.88	95th percentile

*Loss of containment integrity includes failure to isolate containment and containment failure caused by energetic events such as steam explosions, DCH and H_2 combustion

Table 6.6 Mean Conditional Probability of APB Groups for Each PDS (Including Vessel Status and Containment Status)

Accident Progression Bin Groups		PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	All
Vessel Status	Containment Status					
No VB	No CF	0.351	.4046	0.0	0.383	0.079
	CF, Open Containment	0.200	0.379	0.0	0.208	0.274
	Total	0.551	0.425	0.0	0.592	0.353
VB	No CF	0.138	0.066	0.870	0.164	0.248
	CF, Open Containment	0.159	0.509	0.103	0.150	0.372
	CF, DCH	0.004	0.0005	0.027	0.005	0.005
	CF, Hydrogen burning	0.148	0.0	0.0	0.089	0.021
	Total	0.449	0.575	1.00	0.409	0.646

Table 6.7 Mean Conditional Probability of APB Groups for Each PDS (Containment Status Only)

Accident Progression Bin Groups		PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	All
No CF		0.489	0.112	0.870	0.547	0.327
CF	Open Containment	0.359	0.888	0.103	0.358	0.646
	DCH	0.004	0.0005	0.027	0.005	0.005
	Hydrogen burning	0.148	0.0	0.00	0.089	0.021

Table 6.8 Mean Conditional Probability of APB Groups for Each Time Window (Including Vessel Status and Containment Status)

Accident Progression Bin Groups		Window 1	Window 2	Window 3	Window 4	All
Vessel Status	Containment Status					
VB	No CF	0.068	0.085	0.050	0.171	0.079
	CF, Open Containment	0.169	0.306	0.399	0.281	0.274
	Total	0.237	0.391	0.449	0.452	0.353
No VB	No CF	0.465	0.163	0.046	0.179	0.248
	CF, Open Containment	0.253	0.426	0.496	0.355	0.372
	CF, DCH	0.012	0.0003	0.000	0.000	0.005
	CF, Hydrogen burning	0.033	0.019	0.009	0.013	0.021
	Total	0.763	0.597	0.551	0.547	0.647

Table 6.9 Distribution of Frequencies of APB Groups

	5th Percentile	50th Percentile	Mean	95th Percentile
No VB, No CF	6.25E-9	1.25E-7	2.91E-7	1.05E-6
No VB, Open Containment	8.63E-8	4.66E-7	1.23E-6	5.48E-6
VB, No CF	6.58E-8	4.76E-7	8.90E-7	3.33E-6
VB, Open Containment	1.09E-7	6.49E-7	1.69E-6	7.48E-6
VB, DCH + H_2 Burning	2.06E-9	2.57E-8	1.08E-7	4.80E-7

Table 6.10 Mean Conditional Probability of APB Groups for Each PDS With a Mean Containment Failure Pressure of 126 psig (Including Vessel Status and Containment Status)

Accident Progression Bin Groups		PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	All
Vessel Status	Containment Status					
No VB	No CF	0.351	.4046	0.0	0.383	0.079
	CF, Open Containment	0.200	0.379	0.0	0.208	0.274
	Total	0.551	0.425	0.0	0.592	0.353
VB	No CF	0.286	0.066	0.870	0.253	0.269
	CF, Open Containment	0.159	0.509	0.103	0.150	0.372
	CF, DCH	0.004	0.0005	0.027	0.005	0.005
	CF, Hydrogen burning	0.0	0.0	0.0	0.0	0.0
	Total	0.449	0.575	1.00	0.409	0.646

Table 6.11 Comparison of Conditional Containment Failure Probability for Base Case and the Sensitivity Case

CF Status		Containment Failure Pressure	
		Base Case 45 psig	Sensitive Case 125 psig
No CF		0.327	0.348
CF	Open Containment	0.646	0.646
	DCH + H_2 Burning	0.026	0.005
	Total	0.673	0.651

7 SOURCE TERM ANALYSIS

The source term is the information required to calculate the offsite consequences for each group of accident progression bins (APBs). The source term for a given APB consists of the release fractions of the core inventory for nine radionuclide groups, and additional information about the timing of the releases, the energy associated with the releases, and height of the releases. The nine radionuclide groups defined for the source term analysis are: noble gases, iodine, cesium, tellurium, strontium, ruthenium, lanthanum, cerium, and barium. A source term is calculated for each APB for each observation in the APET analysis.

Because of the large number of source terms and the similarity of many of the source terms, it is not practical to perform consequence calculations for all of them. The source terms were therefore grouped through a partitioning process based on their potential health effects into a much smaller number (25 in the present analysis). Consequence calculations were performed only for these 25 source term groups.

The methods and computer codes used for the source term analysis of accidents during mid-loop operation are based on those developed in the NUREG-1150 program.[1] The applicability of these to mid-loop operation is discussed below. Section 7.1 describes the computer code used for the source term calculations. Section 7.2 discusses the quantification of the source terms. Section 7.3 describes the method used for source term partitioning and presents the results of the partitioning process.

7.1 Source Term Analysis Model

The source term analysis for Surry[2] in NUREG-1150 was performed by a parametric computer code: SURSOR.[3] The purpose of this code is not to calculate the behavior of the fission products from their chemical and physical properties and the flow and temperature conditions in the reactor and the containment. Instead, SURSOR provides a means of incorporating into the analysis the results of the more detailed codes that do consider these quantities. For example, SURSOR has a parameter that identifies the fraction of fission products in the core that are released to the vessel before vessel breach. Other parameters identify the fraction of fission products released to the containment and the environment. In all 12 parameters are used in SURSOR to define fission product behavior following a core damage accident. SURSOR also provides a framework for synthesizing the results of experiments and mechanistic codes as interpreted by experts in the field to develop uncertainty distributions of the release parameters. Volume 2, Part 4 of NUREG/CR-4551 provides a detailed description of how the various distributions were developed for the 12 parameters in SURSOR. The application of these distributions for accidents during full power operation at Surry is described in Reference 3.

A simple parametric approach is needed because the detailed codes require too many computer resources to be able to compute the source terms for the numerous APBs that resulted from the quantification of the APET. The use of SURSOR for source term estimation for accidents during mid-loop operation at Surry was first investigated in the abridged study (refer to Chapter 2). Two measures were taken in the abridged study to assure the adequacy of the source terms:

1) The first involved comparing the calculations from the MELCOR[4] code using initial and boundary conditions appropriate to mid-loop operation with the parameter data used in and the source term results obtained from SURSOR.

2) Second, the Source Term Advisory Group was established to provide guidance, and any additional information on modifying the SURSOR code for the study of mid-loop operation.

Considering the differences between full power and shutdown operations, the Source Term Advisory Group identified two parameters in SURSOR as important and possibly different than the values used in NUREG-1150. The first parameter is the fraction of the fission products in the core that are released to the vessel before vessel breach. The second parameter is the fraction of the fission products released to the vessel that are subsequently released to the containment. The distributions of these two parameters as given in NUREG-1150 were compared with MELCOR calculations to establish their values to be used in the study of mid-loop operation. In addition to the above comparison, the environmental releases of fission products obtained from SURSOR and MELCOR were compared. The comparisons show that generally, the MELCOR values fall within the ranges of SURSOR predictions. Although, for some radionuclide categories, the MELCOR calculated values are closer to the upper ranges of the SURSOR predictions, there are no apparent reasons to modify the SURSOR distributions. Consequently, the Source Term Advisory Group did not recommend any change to the SURSOR code for application to the study of mid-loop operation. Appendix C provides a more detailed discussion of the source term comparison.

The distributions for the parameters used in SURSOR have very large ranges. The 5th and 95th percentile values for some of distributions vary by several orders of magnitude. This signifies the uncertainty in source term estimation and reflects the large differences within the reactor safety community surrounding the source terms for any given accident sequence, even if the initial and boundary conditions are well characterized. Furthermore, the initial and boundary conditions are seldom well known, and this lack of knowledge adds additional uncertainty.

7.2 Quantification of Source Terms

Most of the parameters in SURSOR are determined by sampling from distributions of the parameters during Monte Carlo simulations. The distributions for the nine radionuclide groups are assumed to be correlated as they were in NUREG-1150. That is, a single LHS variable applies to each parameter in the release fraction equation, and it applies to the distributions for all nine radionuclide groups. For example, if the random number for the release fraction from the core is 0.8, the 80th percentile value is chosen from the iodine distribution, the cesium distribution, the tellurium distribution and the other six distributions. However, there are separate distributions for each fission product class.

Of the twelve parameters in SURSOR, the following ten parameters listed below were sampled for source term analysis:

Fraction of the radionuclide in the core released to the vessel before or at vessel breach (VB)

Fraction of the radionuclide released from the vessel to the containment before or at VB

Fraction of the radionuclide in the containment from the RCS release that is released from the containment in the absence of any mitigating effects

Fractional release of radionuclide from corium during core concrete interaction (CCI)

Containment transport fraction for ex-vessel release

Decontamination factor for containment sprays

Fraction of the iodine deposited in the containment which is revolatilized and released to the environment late in the accident

Fractional release of material deposited in the RCS due to revaporization

Fraction of core radionuclide released to the containment due to direct containment heating (DCH) at VB

Decontamination factor for a pool of water overlying the core debris during CCI

Source terms were computed for all the APBs for each of 100 observations. There are about 150 APBs in each observation. The total number of source terms obtained was 15,443. An approach used in the full power analysis was to summarize the source terms as complementary cumulative distribution functions for the release fractions of eight of the nine radionuclide groups. Four statistical measures of the distributions were used that give the frequencies at which the release fractions are exceeded. Similar curves were generated in this study; they are presented in Appendix C.

Besides the release fractions four other parameters are needed to specify the source term. These are: the height of the release, the energy of the release, the release timing and the release duration. Since the reason for unisolated containment during mid-loop operation was assumed to be the temporary plug in the escape tunnel of the equipment hatch all releases were assumed to take place through this opening. The height of the release was therefore the level of the equipment hatch, 28 ft (8.4 m) above ground. The release energy of a source term which is input to the consequence code is the average energy release rate over the duration of the release (joules/sec or watts). Energetic releases (containing a large amount of sensible heat) can result in a buoyant plume which can rise to heights much greater than the initial release height leading to greater dilution and smaller consequences near the point of release. The MACCS code models a criterion for plume buoyancy based on atmospheric conditions, windspeed and the sensible heat release rate. The energy during mid-loop operation is low, less than 1% of the energy at full power. Calculations were performed with a range of possible energy release rates during mid-loop operation and compared with the plume lift-off criterion; it was determined that there were no possible energy release rates which could result in a buoyant plume. Thus the release energy was set equal to zero for all releases during mid-loop operation. The timing of the release and the duration of the release were based on selected calculations performed by the MELCOR code. Details of these calculations are provided in Appendix F. Most of the releases occur in window 1 and window 2 with an unisolated containment. All releases were modeled as single-puff releases. Based on MELCOR results, a release timing of 1 hour and a release duration of 6 hours was assumed for these releases. Somewhat conservatively, these times were assumed for the other releases as well.

7.3 Partition of Source Terms

The accident progression and source term analyses resulted in a total of 15,443 source terms for internally initiated accidents during mid-loop operation. It is computationally impractical to carry out a consequence calculation for each source term to obtain the integrated risk for the selected consequence measures. To create an interface between the source term analysis and the consequence calculation, the total number of source terms are grouped into a much smaller number of source term groups. The groups are created such that the source terms within each group have similar properties with respect to consequences, i.e their potential for causing early fatalities and latent cancer fatalities is similar. A frequency weighted mean source term is determined for each group and the consequence calculation is performed for the mean source term.

The grouping of the source terms is designated as "partitioning". The process is described in detail in NUREG/CR-4551, Volume 1 and in Reference 5. It involves definition of an early health effect weight, EH, and a latent health effect weight, LH, for each source term and a grouping of source terms based on these weights. A further subdivision is made on the basis of the timing of the release relative to the time of the emergency evacuation. Then the frequency weighted mean source term is calculated for each group. In the present study, the timing of the release was conservatively estimated to be the same for all releases. Thus no grouping was necessary based on timing of the evacuation relative to the time of release. On the other hand, the effect of radioactive decay as a function of time window was included in the calculation of the early health effect weight. All source terms in Window 4, for example, were assigned an EH=0 based on the equivalent I-131 inventory of Window 4.

The early health effect weight was calculated by converting the radionuclide release associated with each source into an equivalent I-131 release. Surry site-specific calculations of the number of early fatalities as a function of equivalent I-131 release were performed in each time window. This estimated number of early fatalities is the EH for each source term. Details of the calculations including the relationship between the number of early fatalities and the equivalent I-131 release are provided in Appendix C.

The latent health effect, LH, was calculated by assuming a linear relationship between the number of latent cancer fatalities due to a particular radionuclide and the amount of release of that radionuclide. Surry site-specific consequence calculations were carried out for each time window assuming a fixed release of each of the 60 radionuclides in the nine radionuclide groups contained in MACCS. Based on these calculations and the linear relationship between latent cancer fatalities and the amount of radionuclides released, the amount of latent cancer fatalities for each source term was estimated (a window adjustment factor to adjust the radionuclide inventory in each time window was used to estimate the total release in curies associated with each source term depending on the time window to which it belonged). This estimated number of latent cancer fatalities is the latent health effect weight, LH, associated with each source term. Details of the calculation and results are provided in Appendix C.

Based on the estimates of EH and LH, the source terms were divided initially into three categories: EH>0 and LH>0, EH=0 and LH>0, and EH=0 and LH=0. The number of source terms and the percentage of total frequency associated with each of these categories is as follows:

Category	Number of Source Terms	% Total Frequency
EH>0, LH>0	213	2.8
EH=0, LH>0	15230	97.2
EH=0, LH=0	0	0

Each of the above categories was treated separately for partitioning.

For the EH>0, LH>0 category a grid was created by examining the ranges of the EH and LH values, placing the source terms within each cell on the grid and then pooling cells which either have a small frequency or a small number of source terms. Four source term groups were created for this category through this process which is described in more detail in Appendix C. For the EH=0, LH>0 category, the source terms were grouped along one dimension (the value of LH) by creating cells based on the range of values of LH. A total of 21 groups for this category were obtained through this process which is discussed in Appendix C.

A total of 25 source term groups was thus obtained after partitioning. A frequency weighted mean source term was then identified for each of the groups. The mean source terms were used for the consequence calculations. They are displayed in Appendix C.

7.4 References

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5. Iman, R. L., J. C. Helton, and J. D. Johnson, "A User's Guide for PARTITION: A Program for Defining the Source Term/Consequence Analysis Interfaces in the NUREG-1150 Probabilistic Risk Assessments," NUREG/CR-5253, SAND88-2940, Sandia National Laboratories, May 1990.

8 CONSEQUENCE ANALYSIS

Offsite consequences were calculated for the mean source term groups resulting from the partitioning process described in Chapter 7. The calculations were performed using the latest version (Version 1.5.11.1) of the MACCS code.[1] An approximate scoping calculation of onsite dose rates in the vicinity of the containment (so-called parking lot dose) was also carried out for three selected mean source terms and selected weather conditions based on dose models described in section 8.2 below.

8.1 Offsite Calculations

The offsite consequence calculations were performed by the latest version of the MACCS code,[1] Version 1.5.11.1, which incorporates the BEIR V recommended risk factors for the latent cancer - dose relationship. MACCS tracks the dispersion of the radioactive material in the atmosphere from the plant and computes its deposition on the ground. MACCS then calculates the effects of this radioactivity on the population and the environment. Doses and the ensuing health effects from 60 radionuclides are computed for the following pathways: immersion or cloudshine, inhalation from the plume, groundshine, deposition on the skin, inhalation of resuspended ground contamination, ingestion of contaminated water and ingestion of contaminated food.

MACCS treats atmospheric dispersion by the use of multiple, straight-line Gaussian plumes. Each plume can have a different direction, duration, and initial radionuclide concentration. Dry and wet deposition are treated as independent processes. The weather variability is treated by means of a stratified sampling process.

For early exposure, the following pathways are considered: immersion or cloudshine, inhalation from the plume, groundshine, deposition on the skin, and inhalation of resuspended ground contamination. For the long-term exposure, MACCS considers the following four pathways: groundshine, inhalation of resuspended ground contamination, ingestion of contaminated water and ingestion of contaminated food. The direct exposure pathways groundshine, and inhalation of resuspended ground contamination, produce doses in the population living in the area surrounding the plant. The indirect exposure pathways, ingestion of contaminated water and food, produce doses in those who ingest food or water emanating from the contaminated area around the accident site. The contamination of water bodies is estimated for the washoff of land-deposited material as well as direct deposition. The food pathway model includes direct deposition onto crops and uptake from the soil.

Both short-term emergency response actions and long-term mitigative measures are modeled in MACCS. Emergency response actions include evacuation, sheltering and emergency relocation out of the emergency planning zone (EPZ). Long-term actions include later relocation and restrictions on land use and crop disposition to reduce projected doses below a pre-determined level. Relocation and land decontamination, interdiction, and condemnation are based on projected long-term doses from groundshine and inhalation of resuspended radioactivity. The disposal of agricultural products is based on reducing the yearly doses induced by consumption of these products below a preset criterion based on the Protective Action Guides of the Food and Drug Administration. The removal of farmland from crop production is based on ground contamination criteria.

The health effects models link the dose received by an organ to predicted morbidity or mortality. MACCS calculates both early and latent (long-term) health effects. The model for latent cancers implemented in the latest version of MACCS is based on the recommendations of the BEIR V committee. Results for the following consequence measures calculated by MACCS are given in this report: early fatalities, population dose to 50 miles, and latent cancers to 50 miles.

Early fatalities are defined as those occurring within one year of the release. Population dose, expressed in effective dose equivalents for whole body exposure (person-Sv or person-rem), due to both early exposure and chronic exposure is calculated within a 50 mile radius of the plant. The latent cancers due to both early exposure and chronic exposure are calculated to 50 miles from the plant.

8.1.1 MACCS Input for Surry

The input parameters for the offsite consequence calculations were mainly based on Reference 2. Certain modifications, based on conditions specific to mid-loop operation and its accident characteristics, are noted below.

Site-specific data for Surry include: reactor power level, weather, population, exclusion zone distance, emergency response, shielding parameters, long-term mitigative actions, land and land use fractions, and economic parameters for calculating offsite costs. Apart from reactor power level, discussed below, the other site data were based on the values in Reference 2. For example, the emergency response assumes that 99.5% of the population in the EPZ (within a 10 mile radius of the plant) evacuates at a speed of 1.8 m/s (4 mph) after a 2 hour delay following the declaration of an emergency, i.e., a general warning by the local authorities. The remaining 0.5% of the population are assumed to follow normal activity. The long-term mitigative actions include relocation of people from contaminated land which could lead to a dose to an individual of 4 rem or more over 5 years (2 rem in the first year following the accident and 0.5 rem per year thereafter for 4 years). One year of hourly meteorological data at Surry and the site population distribution as in Reference 2 were used in the input.

The initial reactor power level (13.2 MW) used in the calculation was at the mid-point of Window 1 as defined in Table 5.5. The core inventory of 60 radionuclides at this power level for Surry was evaluated by interpolation from the calculations of the core inventory at Surry as a function of time after reactor shutdown reported in Reference 6. This inventory, displayed in Table 7.4 of Chapter 7 was used for calculating the consequences of all 25 source term groups resulting from the partitioning process. A window adjustment factor, WF_w, defined in section 7.3 of Chapter 7 was used in calculating the consequences for Windows 2, 3, and 4 using the predictions for Window 1.

8.1.2 Results of Offsite Consequence Calculations

The results for the offsite consequence measures given in this section are conditional on the occurrence of a release. The tables and figures in this section contain no information about the expected frequency of occurrence of these consequences. Information about the frequency of the consequences of different magnitudes is contained in the integrated risk results reported in Chapter 9.

The results of MACCS calculations for the twenty five mean source term groups are reported in two ways below. In the first way, a complementary cumulative distribution function (CCDF) is calculated for each consequence measure. Each CCDF is conditional on the occurrence of the source term and gives the probability of exceedance of individual consequence values due to the variability of the weather at the time of the release. The CCDFs are displayed in Figure 8.1 for early fatalities, Figure 8.2 for latent fatalities to 50 miles, and Figure 8.3 for population dose to 50 miles. Each of the above figures displays the CCDF for a particular consequence measure for all 25 mean source term groups. (The source term group number, ordered by increasing severity of consequences, increases from the plots located on the left to the plots located on the right in each of the above Figures 8.1, 8.2 and 8.3).

In the second way, by averaging the CCDFs over all weather bins, a single mean result is reported for each consequence measure. Given the 25 source term groups and the three consequence measures considered, this produces a 25 × 3 matrix of mean consequence measures, shown in Table 8.1.

Figure 8.1 shows that only four source term groups have an exceedance probability greater than 0.01 for causing one early fatality. Since there are only 17 CCDF curves shown in Figure 8.1, this means that 8 source term groups have zero early fatalities for all values of the exceedance probability, i.e there were now weather bins which gave rise to an early fatality for these 8 source term groups. For the case of latent cancer fatalities to 50 miles shown in Figure 8.2, most of the source term groups have an exceedance probability greater than 0.1 for causing 100 or more latent cancers.

The mean consequence values shown in Table 8.1 are a result of reducing one of the CCDFs for a particular source term group to a single number by averaging over the weather. 13 source term groups have a zero mean value of early fatalities and only four source term groups have a mean value of early fatalities which exceeds 0.1. In contrast, 16 source term groups have more than 100 mean latent cancer fatalities to 50 miles.

8.2 Onsite Consequences

The total onsite dose rate is modeled as a sum of the inhalation and cloud exposure dose rates in the wake region of a building due to the release of the radionuclide inventory following an accident. For the scoping calculation, an uniform release rate was assumed.

The dose rate is calculated as a sum of the cloud inhalation dose rate, D_i^{inh}, and the cloudshine dose rate, D_i^{cloud} (based on the 60 radionuclides in the MACCS dosimetry routine):

$$D = \sum_{i=1}^{i=60} \left[D_i^{inh} + D_i^{cloud} \right],$$

where,

$$D_i^{inh} = (DFI)_i \, \beta \, Q(t) \left\{ \frac{\chi}{Q} \right\}, \text{ rem/sec}$$

$$D_i^{cloud} = (DFC)_{\infty i} \, Q(t) \left\{ \frac{\chi}{Q} \right\}, \text{ rem/sec}$$

$(DFI)_i$ - inhalation dose-conversion factor, rem/Ci for the radionuclide i;

$(DFC)_{\infty i}$ - semi-infinite cloud dose-conversion factor for the radionuclide i $\frac{rem - m^3}{Ci - sec}$;

β - breathing rate, m^3/s. In these calculations, the breathing rate $\beta = 2.66 \times 10^{-4}$ m^3/s following the MACCS code default value;

r_i - fraction of nuclide's i inventory released over the release duration;

I_i - total inventory of nuclide i, Ci;

τ - release duration, sec.;

$Q(t)$ - release rate, assumed to be uniform over the release duration = $r_i I_i / \tau$, Ci/sec;

χ/Q - dilution factor calculated, as explained below, by three different models, sec/m^3.

The average concentration in the building wake was estimated using the following wake centerline concentration models: Ramsdell,[3] Wilson,[4] and Reg. Guide 1.145.[5] Brief descriptions of each model are given below.

8.2.1 Ramsdell Model

The Ramsdell model[3] calculates the concentration in the far-region of the wake by including the effects of the lateral and vertical diffusion due to background turbulence:

$$\chi/Q = \frac{1}{\pi U \left[\sigma_y^2 + (KA/a^2U^2)\,F(T_s)\right]^{1/2} \times \left[\sigma_z^2 + (KA/a^2U^2S^2)\,F(T_w)\right]^{1/2}}$$

where K is a characteristic dispersion factor for large structures typical of reactor buildings and recommended to have a value of 0.5 in Reference 3.

In the correlation above,

$$F(T) = 1 - [1 + x/(UT)]\,\exp[-x/UT],$$

where $T = T_s$ or T_w; $T_s = A^{1/2}/u^*$, sec;

$T_w = T_s$ for extremely unstable weather (Class A, Pasquill-Gifford), and $T_w = T_s/2.5$ for extremely stable conditions (class G);

$S = 1$ for extremely unstable weather (Class A, Pasquill-Gifford), and $S = 2.5$ for extremely stable conditions (class G);

$u^* = aU$, friction velocity, m/s; $a = 0.4/ln(z/z_0)$;
U is the average wind speed at $z = 10$ meters, m/s,
surface roughness length $z_0 = 0.1$ m; based on this, $a = 0.0869$,
A = building area, m^2;

σ_y and σ_z = diffusion coefficients due to the background turbulence.

x = downwind distance from the source.

8.2.2 Wilson Model

The Wilson model suggests a correlation for calculating the lower limit on the dilution in the wake (which corresponds to maximum concentration in the wake). This leads to the following expression for χ/Q:

$$\left(\frac{\chi}{Q}\right)_{max} = \frac{1}{0.11}\,\frac{1}{Ux^2}\,.$$

where U = average windspeed
x = downwind distance from the source.

As recommended in Ref. 3, a multiplier of 5.0 was used to calculate the ground level release (elevation lower than 0.2 H, where H is the height of the building).

8.2.3 NRC Regulatory Guide 1.145 Model

NRC Regulatory Guide 1.145 contains guidance on the calculation of χ/Q values for releases through vents that are effectively lower than two and one-half times the height of adjacent solid structures during neutral or stable weather stability conditions. The recommended correlations allow for horizontal plume meander when the windspeed at the 10-meter level is less than 6 m/s. Equation (2) of the Reg. 1.145 model was used for calculating χ/Q:

$$\chi/Q = \frac{1}{\bar{U}_{10}(3\pi\sigma_y\sigma_z)}$$

σ_y and σ_z are the standard dispersion parameters

$\bar{U}_{10}$ is the average windspeed at 10 m height

$\sigma_y = ax^b$ and $\sigma_z = cx^d$,

x is the distance from the source, m, and

the dispersion constants $a = 0.0722$, $b = 0.9031$, $c = 0.2$, and $d = 0.602$ for stable weather, Pasquill-Gifford Class F.

8.2.4 Calculation Assumptions

The scoping values of dose rates were calculated with the following set of input parameters corresponding to the Surry building and site.

- distance from source, $x = 10$ to 300 meters,
- building projected area, $A = 1500\ m^2$, and, finally,
- wind speed at 10 m elevation, $U = 1.2\ m/s$.

Wind speed was obtained by an arithmetic averaging of the wind speeds observed at the Surry site during the most stable weather conditions (Class F stability).

8.2.5 Results

The bounding calculations were performed for three source terms referred to as high, medium, and low (Gap release). The results are shown in Figure 8.4. The Wilson/Reg. Guide 1.145 result is based on the prediction of the Reg. Guide 1.145 correlation, limited from above by the values predicted by the Wilson model. The results in Figure 8.4 for the dose rate (Rem/h) indicate a variation of about two orders of magnitude as a function of the source term. The onsite dose rates are high, and are likely to lead to early fatalities or early injuries for exposed workers depending on the exposure period. In view of the relatively large number of onsite personnel during shutdown operations, these dose rates outside the containment suggest that a careful examination should be made of onsite evacuation schemes to limit the consequences.

8.3 References

1. Chanin, D., J. Rollstin, J. Foster, and L. Miller, "MACCS Version 1.5.11.1: A Maintenance Release of the Code," NUREG/CR-6059, October 1993.

2. Sprung, J. L., J. A. Rollstin, J. C. Helton, H-N Jow, "Evaluation of Severe Accident Risks: Surry Unit 1, Main Report, NUREG/CR-4551, October 1990.

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5. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.145, Rev. 1, November 1982.

6. Denning, R. S., R. Freeman-Kelly, P. Cybulskis, L. A. Curtis, "Source Term Calculations for Assessing Radiation Dose to Equipment," Battelle Columbus Laboratories, NUREG/CR-4949, 1989.

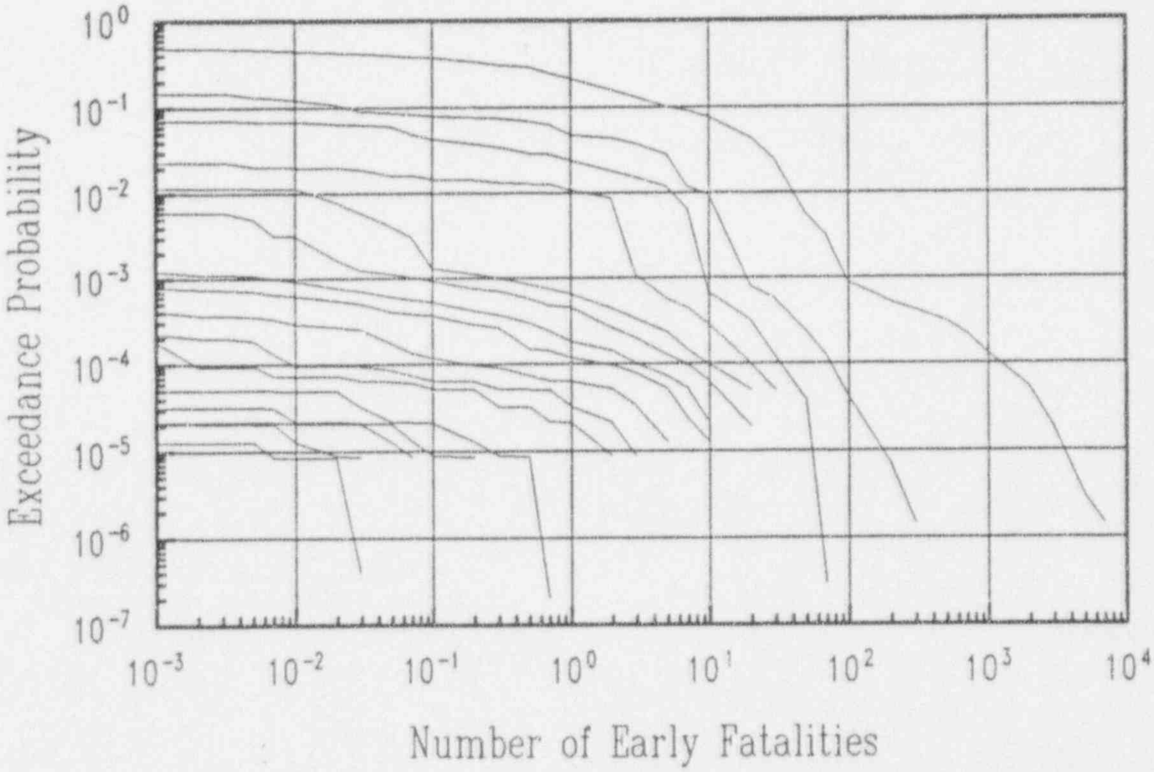


Figure 8.1 Surry, Mid-Loop Operation: CCDFs for Early Fatalities Conditional on Source Terms

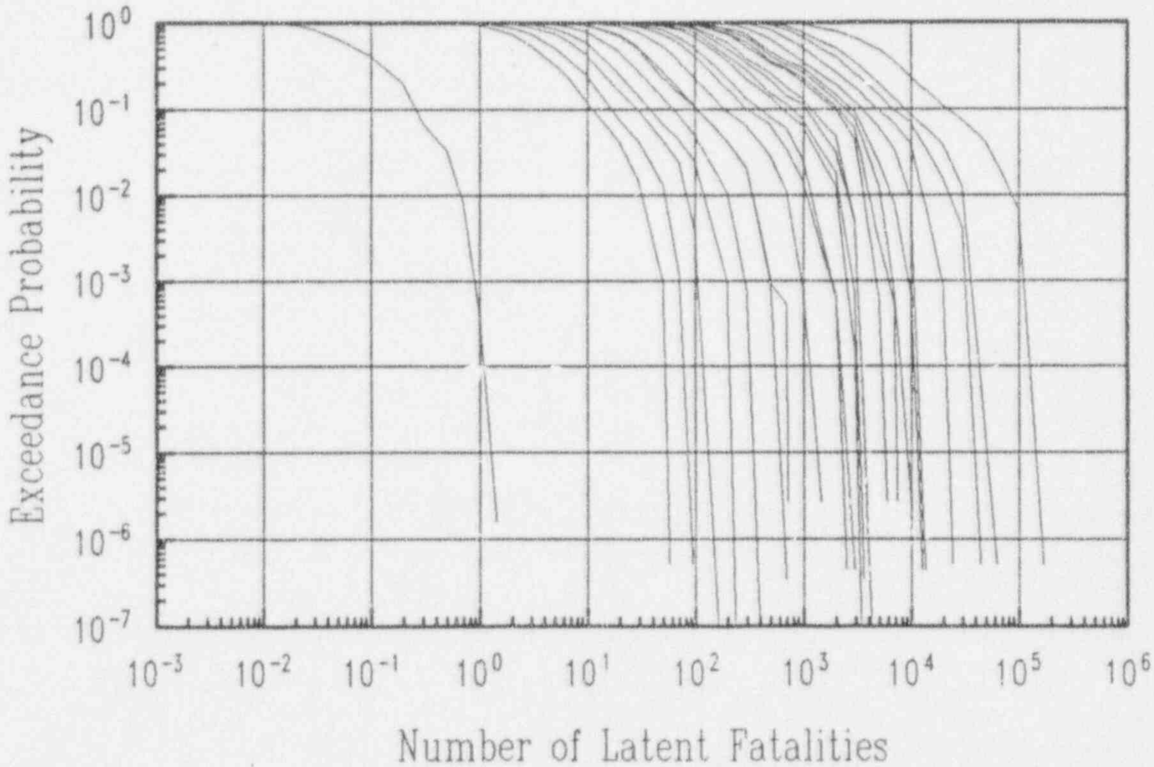


Figure 8.2 Surry, Mid-Loop Operation: CCDFs for Latent Fatalities to 50 Miles Conditional on Source Terms

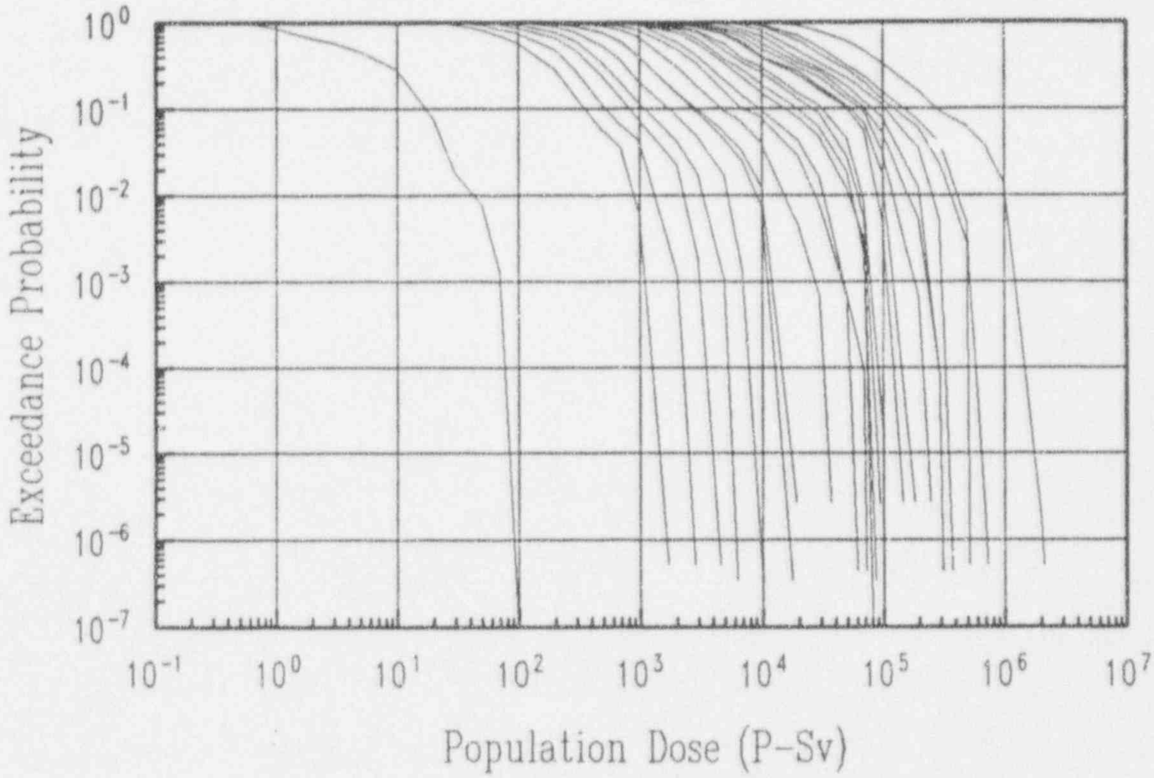


Figure 8.3 Surry, Mid-Loop Operation: CCDFs for Population Dose to 50 Miles Conditional on Source Terms

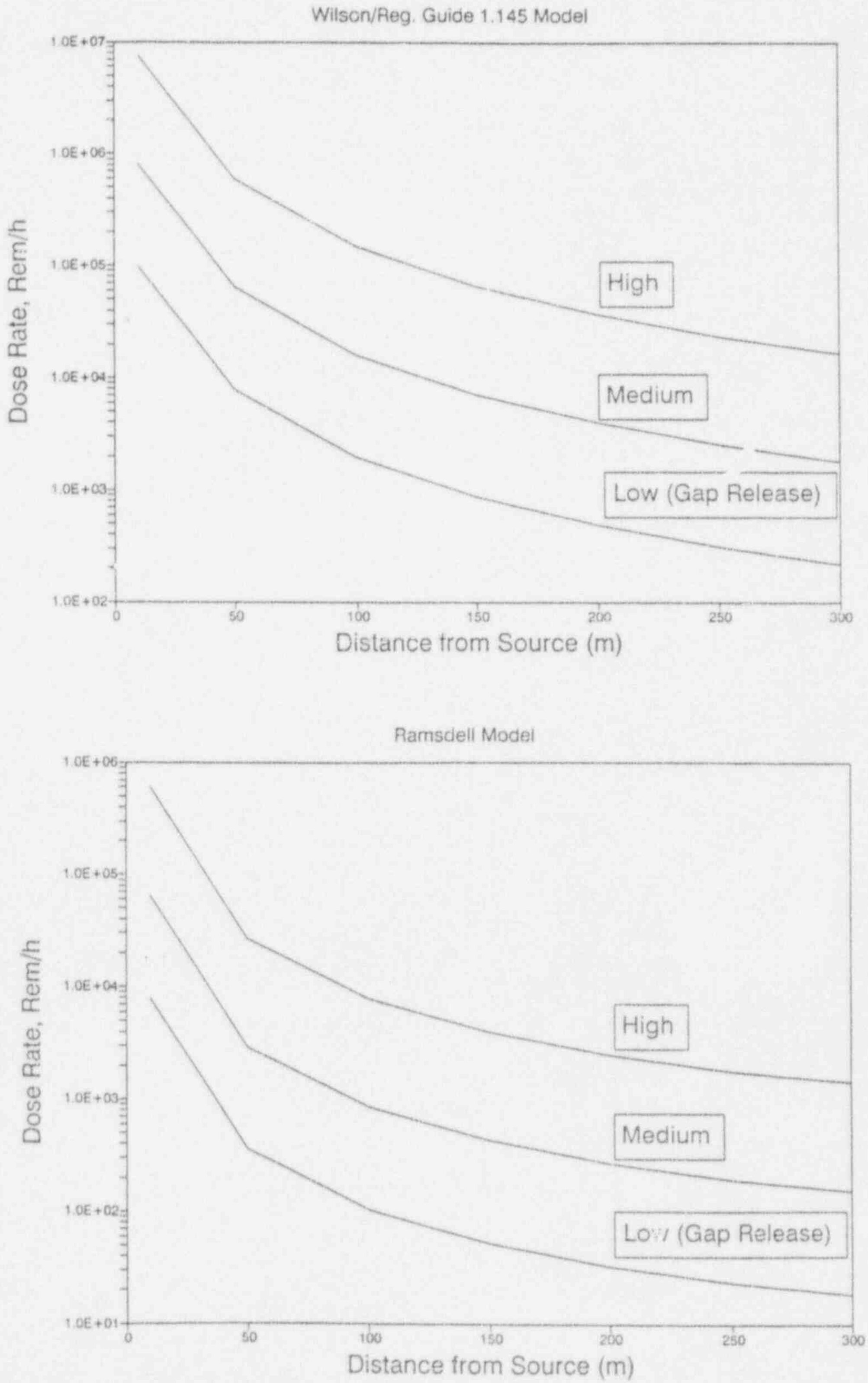


Figure 8.4 Onsite Parking Lot Dose Rate as a Function of Distance from the Containment for Three Source Terms

9 Risk Integration

This chapter provides the results of the integrated risk analysis of the Surry plant during mid-loop operation. Risk is determined by bringing together the results of the four constituent analyses; accident frequency (discussed in Volume 2 of this report and summarized in Chapter 5), accident progression (Chapter 6), source terms (Chapter 7) and consequence analyses (Chapter 8). The methods used to perform the risk integration have been broadly described in Chapter 3. Details of the calculations carried out for each of the risk results are briefly reviewed below. The results are presented in the form of CCDFs, distributions of risk, and fractional contributions to mean risk for the selected consequence measures, early fatalities, population dose to 50 miles and latent cancer fatalities to 50 miles.

9.1 Risk Results

9.1.1 Exceedance Frequencies for Risk

The integrated risk analysis was performed for a sample of 100 observations; each observation consists of a frequency for each of the 25 mean source term groups, calculated by summing the frequencies of the accident progression bins assigned to each source term group. For each consequence measure, these 100 observations were combined with the CCDFs for each of the mean source term groups (shown in Figures 8.1, 8.2, and 8.3 in Chapter 8) which contain the uncertainty due to weather variation. This calculational step produces 100 CCDFs for each consequence measure which display the relationship between the frequency of the consequence and the magnitude of the consequence. Four statistical measures were generated from these 100 CCDF curves by analyzing the curves in the vertical direction. For each value of a particular consequence, there are 100 values of the exceedance frequency (one for each observation). From these 100 values the mean, median, 95th percentile, and 5th percentile were calculated. This was done for each value of the consequence measure, to obtain the plots shown in Figures 9.1, 9.2 and 9.3. These figures show the relationship between the magnitude of the consequence and the frequency at which the consequence is exceeded, as well as the variation in that relationship.

Figure 9.1 shows that the risk of early fatalities during mid-loop operation is low. At the upper end (95th percentile) of the range, the risk of one early fatality is below 5 × 10^{-8} per year. Comparison of these curves with the corresponding risk at full power shows that the early fatality risk during mid-loop is between one and two orders of magnitude below the full power risk. This reduction is due mainly to the decay of the radionuclide inventory, especially those species such as iodine and tellurium which impact early fatalities. At high values of the early fatality consequence plots, the mean exceeds the 95th percentile. This indicates that the mean is being influenced by a few large values within the sample.

The curves for latent cancer fatalities in Figure 9.2 are relatively flat from 0.1 to about 100 fatalities. (A flat portion of an exceedance curve indicates a very low probability since probability is the first derivative of the exceedance curve). This means that latent cancer fatalities in this range are very unlikely. All accidents involving containment failure or unisolated containment are likely to lead to more than 100 latent cancer fatalities.

The 5th to the 95th percentiles indicates the uncertainty in the risk estimates due to uncertainty in the basic parameters in the three sampled constituent analyses (the accident frequency, accident progression, and source term analyses). For latent cancer fatalities and population dose to 50 miles (Figures 9.2 and 9.3), this uncertainty is approximately two orders of magnitude across most of the exceedance curve. Three parameters, in particular, contribute to this uncertainty; the uncertainty in the human error probabilities, the uncertainty in the status of containment, and the uncertainty in the availability of containment sprays. The uncertainties in the release fractions constituting the source terms also contribute to the overall uncertainty.

The variation along a curve in Figures 9.1, 9.2 and 9.3 is indicative of the variation in risk due to different types of accidents and due to different weather conditions at the time of the accident. Thus the individual curves can be viewed as representing stochastic variability (i.e., the effects of probabilistic events in which it is possible for the accident to develop in more than one way), and the variability between curves can be seen as representing the effects of imprecisely known parameters and processes that are mostly nonstochastic. Insights into the risk from different types of accidents, represented by plant damage states, is discussed below.

As the magnitude of the consequence measure increases, the mean curve typically approaches or exceeds the 95th percentile curve. This happens when the mean is dominated by a few large values within the sample because only few observations have nonzero exceedance frequencies at large values of the consequences.

9.1.2 Estimates of Total Risk and Mean Risk

Based on the CCDF of risk for each observation a single number may be generated for each consequence measure for each observation. This value, called total risk, is determined by summing the product of the frequencies and consequences for all the points that are used to construct the CCDF for each observation in the sample. The total risk estimate averages over the different weather states and includes contributions from all the different types of accidents that can occur. Since the complete analysis consisted of a sample of 100 observations, there are 100 values of total risk for each consequence measure. The distribution of total risk for the three consequence measures, early fatalities, latent cancer fatalities to 50 miles, and population dose to 50 miles, based on these 100 values is shown in Figure 9.4 The same four statistical measures utilized above, that is the median, mean, 5th, and 95th percentiles are shown in these plots.

The plots in Figure 9.4 show the distribution of the total risk for the three consequence measures. The distribution for early fatalities shows that the ratio of the 5th to the 95th percentile is approximately three orders of magnitude. For the cases of latent cancer fatalities and population dose to 50 miles, the distributions show that the ratio of the 5th to the 95th percentile is a little less than two orders of magnitude. Where the mean is close to the 95th percentile, as it is for the early fatality distribution, it may be inferred that a relatively small number of observations dominate the mean value. This is more likely to occur for the early fatality consequence measure than for the latent cancer fatality or population dose consequence measures due to the threshold effect for early fatalities. A minimum or threshold value of dose is required to induce an early fatality. This threshold value (150 rem for bone marrow, one of the organs in the early fatality model in the MACCS code) may be exceeded for a few weather bins which include washout of the release over a populated sector or a few source term bins where the timing of the release and the evacuation is such that the population evacuates under the plume.

These plots for the distribution of risk can be compared with the corresponding plots for the full power study to obtain insights into how accident progression during mid-loop operation differs from full power operation. This comparison is performed in Chapter 10.

9.1.3 Contributors to Mean Risk

To evaluate the contribution to risk from different types of accidents represented by the plant damage states, the fractional contribution to mean risk (FCMR) has been calculated as follows. If $FCMR_{jk}$ represents the fractional contribution to risk for consequence measure j from plant damage state k, then

$$FCMR_{jk} = \frac{\left\{ \sum_i rC_{ijk} \Big/ \sum_i \text{observations} \right\}}{\left\{ \sum_i rC_{ij} \Big/ \sum_i \text{observations} \right\}}$$

$$= \sum_i rC_{ijn} \Big/ \sum_i rC_{ij}$$

where,

rC_j = risk (consequences/year) for consequence measure j,

rC_{ij} = value of rC_j for observation i,

rC_{jk} = risk of consequence measure j due to PDS group k, and

rC_{ijk} = value of rC_{jk} for observation i.

The summation is over all observations (100 in this case).

Figure 9.5 shows the fractional contribution to mean risk of early fatality, latent cancer fatality and population dose to 50 miles from the four PDS groups: PDS 1 (station blackout), PDS 2 (human error), PDS 3 (recirculation failure and PDS 4 (loss of 4 kV bus). This figure shows that PDS 2 contributes over 90% of the mean risk for all three consequence measures. The reason for this is that for accidents where operator errors, such as failure to diagnose or take proper action, play a major role in determining the progression of the accident, it was judged unlikely that actions to isolate containment would be taken. As shown in Chapter 6, for PDS 2 the mean conditional probability of the containment being unisolated was estimated to be almost 0.95 for the accident sequences belonging to this plant damage state.

A similar analysis was performed for the fractional contribution to mean risk for the three consequence measures from each of the four time windows over which mid-loop operation extends. These results show which of the time periods over the duration of mid-loop operation have the most vulnerability from the standpoint of risk. The results are shown in Figure 9.6. The largest contribution, about 43% in the case of the mean risk of latent cancer fatalities and population dose and about 39% for the mean risk of early fatality, comes from Window 2. There are two reasons for the higher contribution of Window 2 to the mean risk. First, Window 2 has a higher contribution to the total core damage frequency compared with the other windows; second, PDS 2 has a large weight in Window 2 as shown in Table 5.8. Since the containment is largely unisolated in PDS 2, as discussed above, the combined effect of the contribution of Window 2 to the core damage frequency and the finding that a significant portion of this contribution arises from PDS 2 makes Window 2 the largest contributor to the mean risk. Windows 1 and 3 contribute roughly similar amounts to the mean risk, less than 30% for all three consequence measures, while the contribution of Window 4 is small, less than 10%. Window 1 contributes more than Window 3 to the risk of early fatality; this is mainly due to the reason that the inventory of radionuclides important to early fatalities is larger in Window 1 than in Window 3. However, the risk distributions for the three consequence measures are subject to a significant amount of uncertainty. Hence these estimates of the contribution of each time window to the mean risk should be considered in the presence of this uncertainty.

9.1.4 Distribution of Risk for Each PDS

Figures 9.7, 9.8 and 9.9 show the distribution of risk for each of the PDSs for the three consequence measures. The total risk is also shown in the figures for comparison. The distribution for PDS 2 is almost equivalent to the total risk distribution for all three consequence measures. As pointed out above, the mean conditional probability of the containment being unisolated is very high for accidents in PDS 2 and this contributes strongly to the total risk distribution. The distributions for PDS 3 and PDS 4 lie almost entirely below the distribution for PDS 2. Thus, even though the distributions for PDS 3 and PDS 4 have a wide range of uncertainty, they consist of very low estimates of consequences and do not impact the total risk distribution significantly. The upper end (95th percentile) of the distribution for PDS 1 lies near the median of the distribution for PDS 2 and the total distribution. A small fraction of the accidents, about 16%, in PDS 1 involve an initially isolated containment which is subsequently challenged by phenomena such as hydrogen combustion. These types of accidents contribute to the upper end of the risk range for PDS 1.

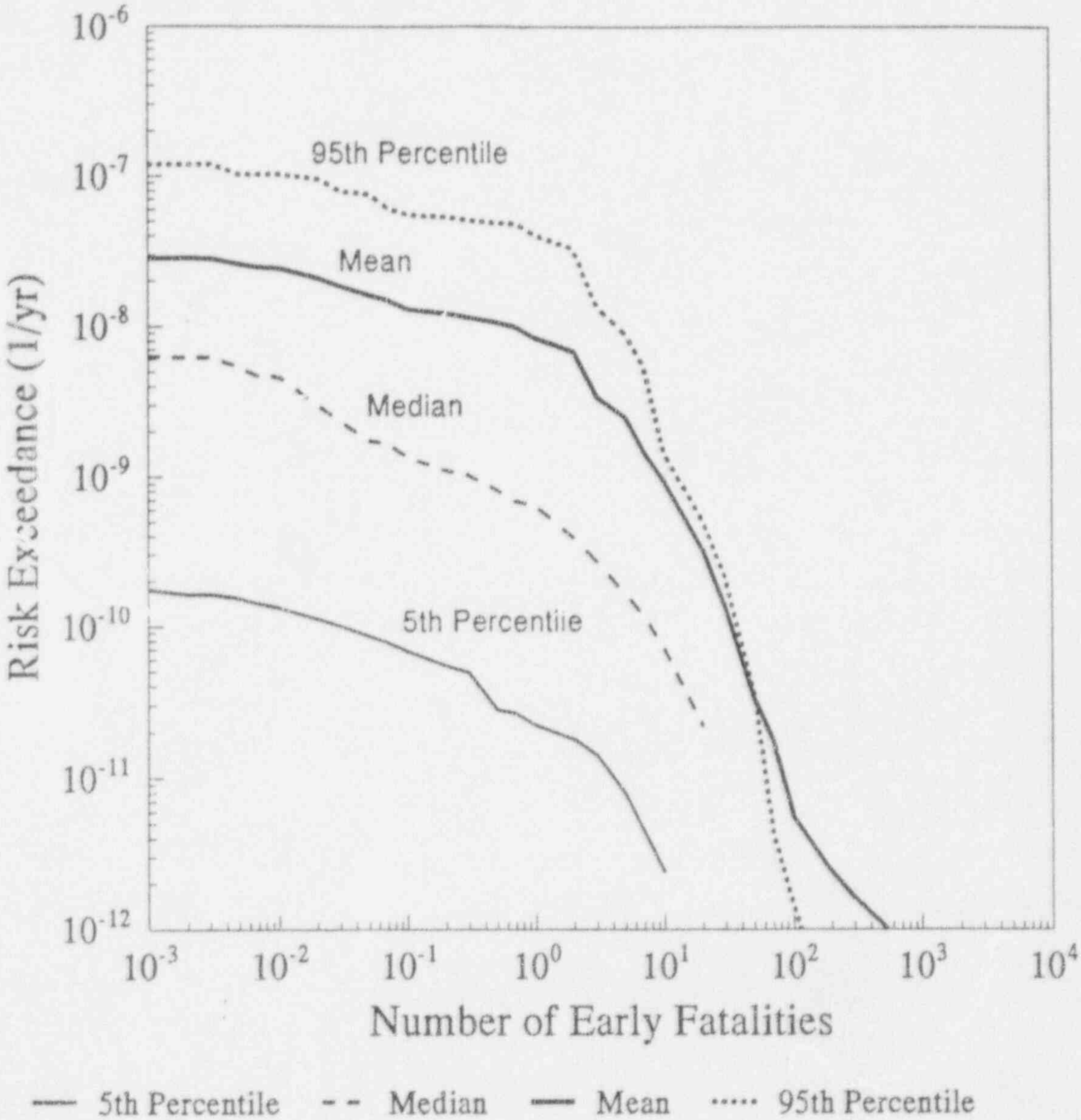


Figure 9.1 Exceedance Frequencies for Risk of Early Fatalities

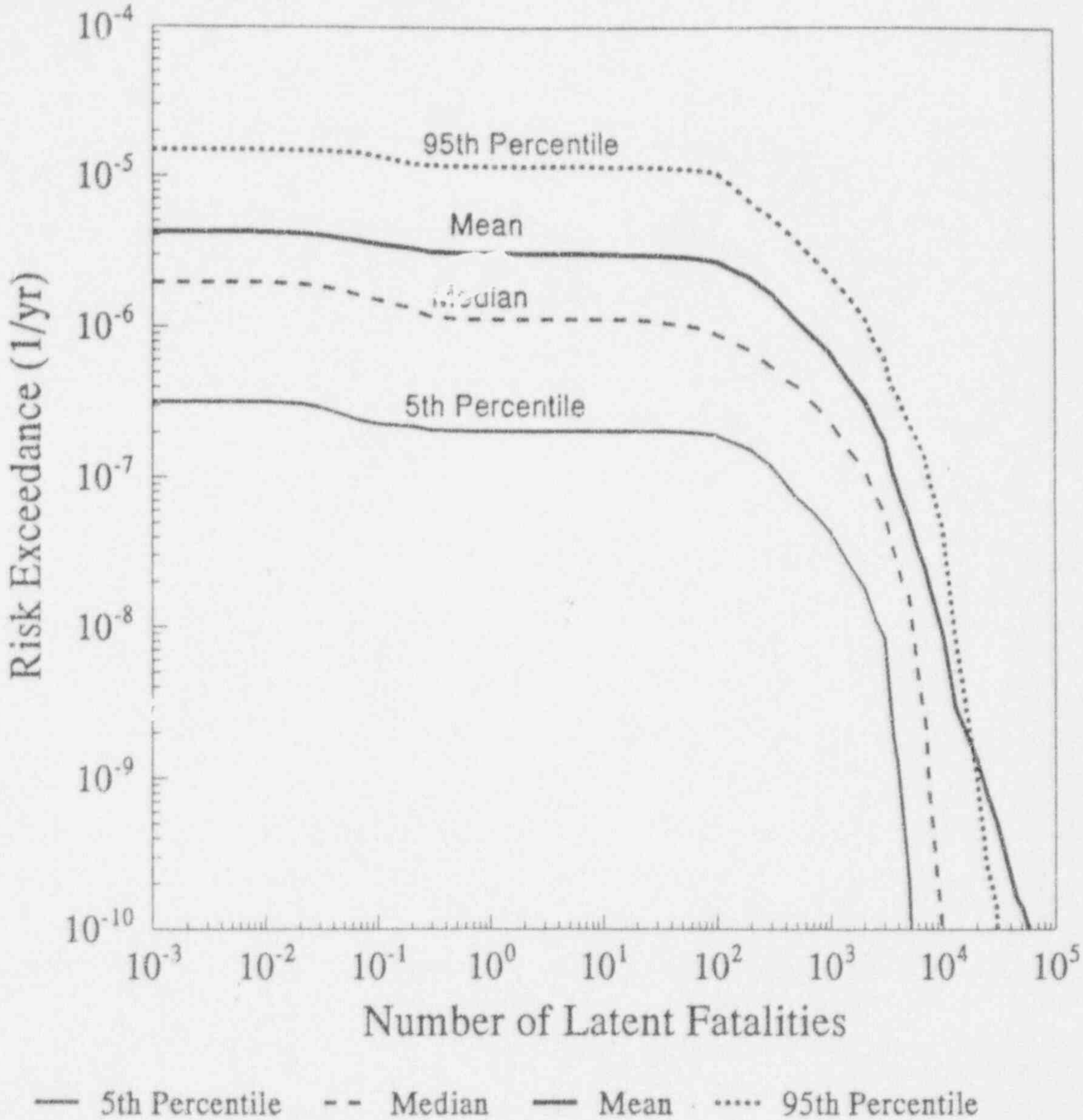


Figure 9.2 Exceedance Frequencies for Risk of Latent Cancer Fatalities to 50 Miles

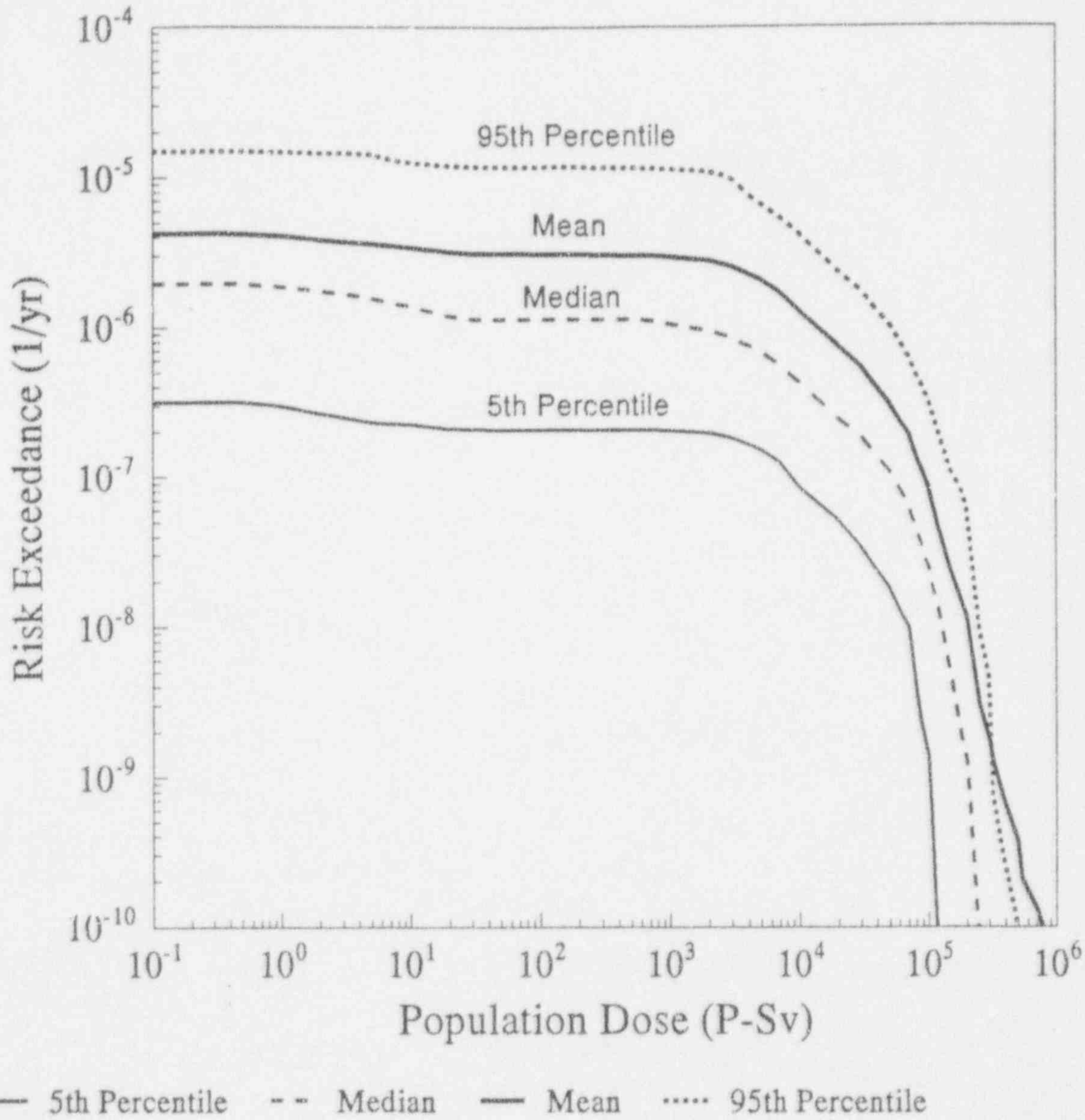


Figure 9.3 Exceedance Frequencies Risk of Population Dose to 50 Miles

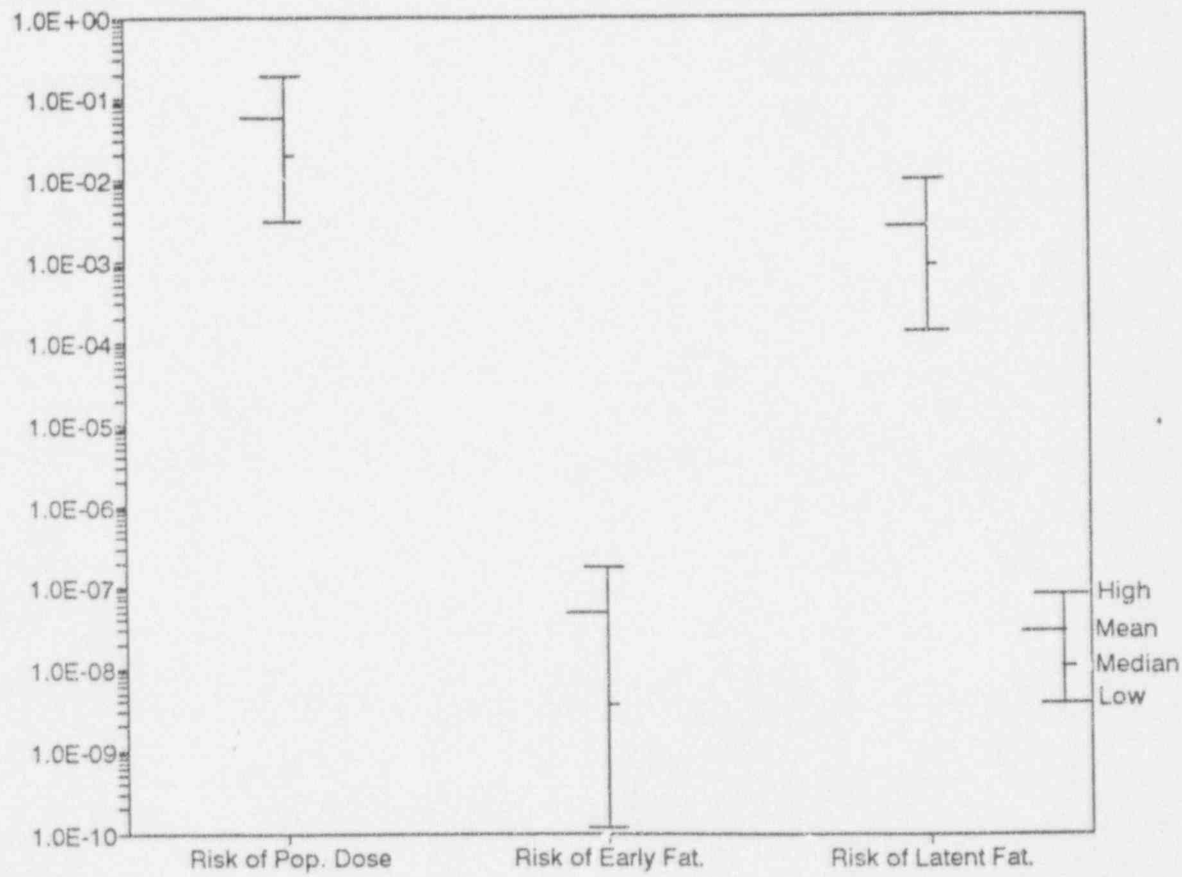


Figure 9.4 Surry Mid-Loop Operation: Distributions of Total Risk for the Three Consequence Measures

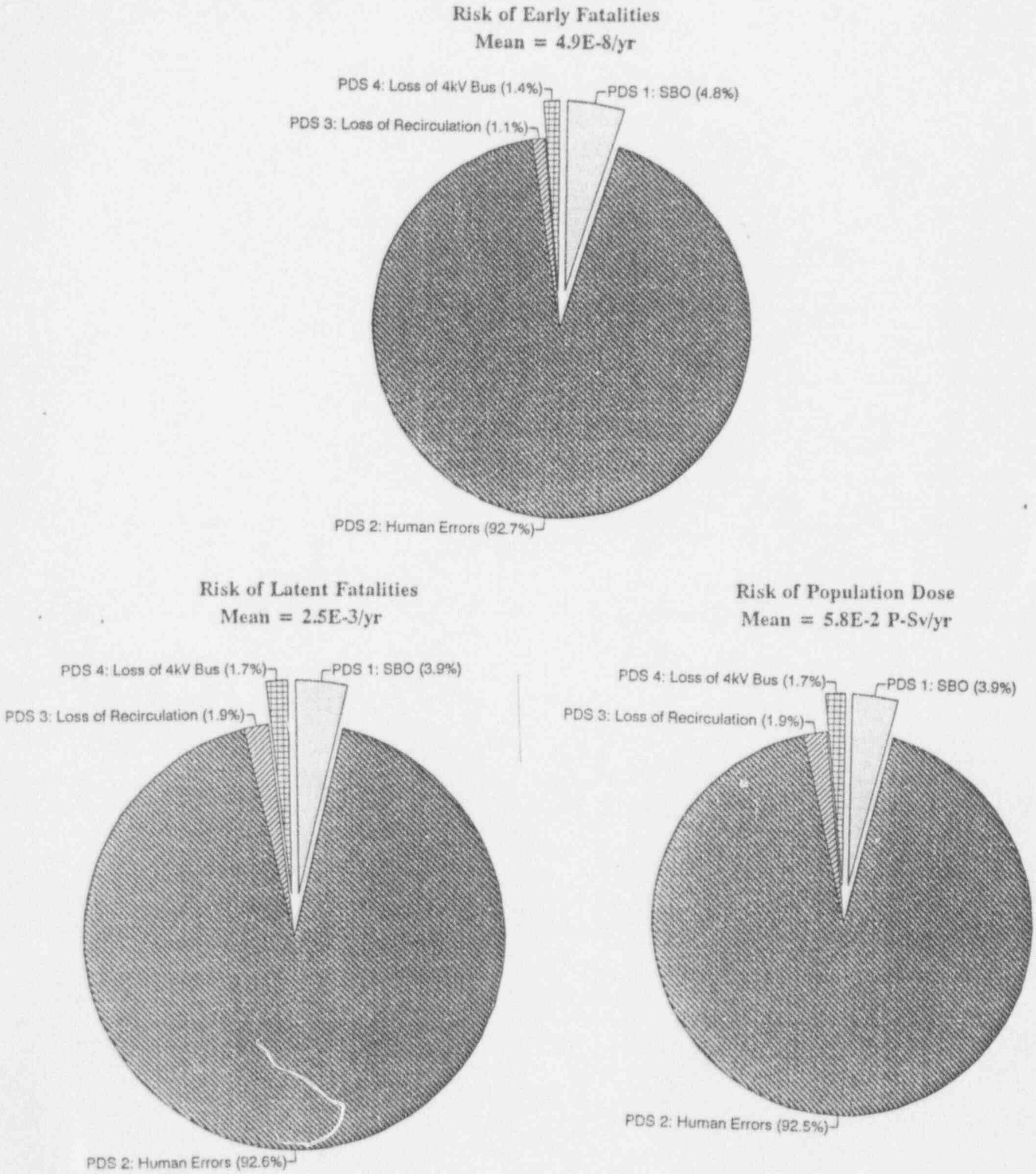


Figure 9.5 Contributions of Plant Damage States to Mean Risk of Consequences

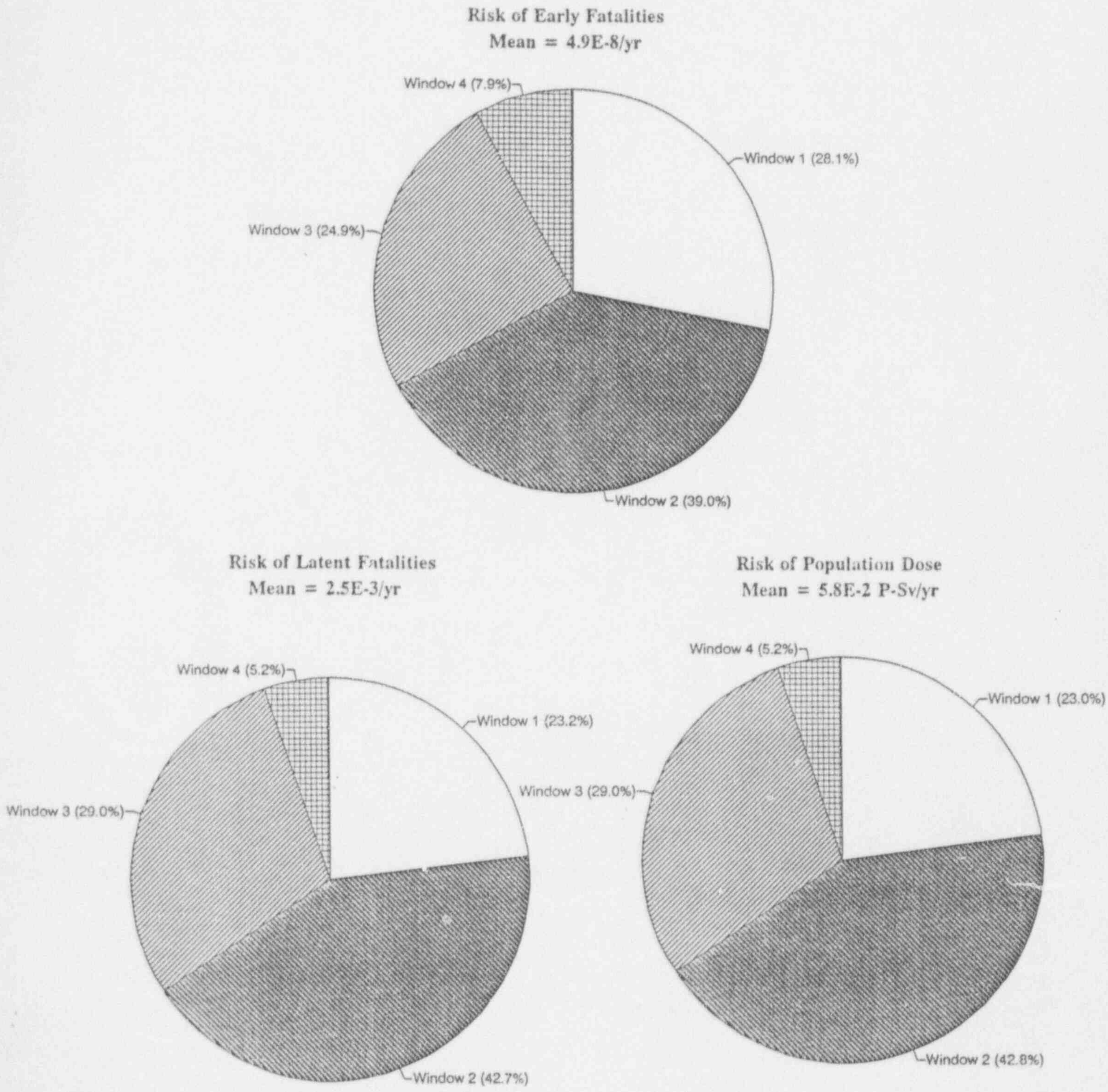


Figure 9.6 Contribution of Time Windows to Mean Risk of Consequences

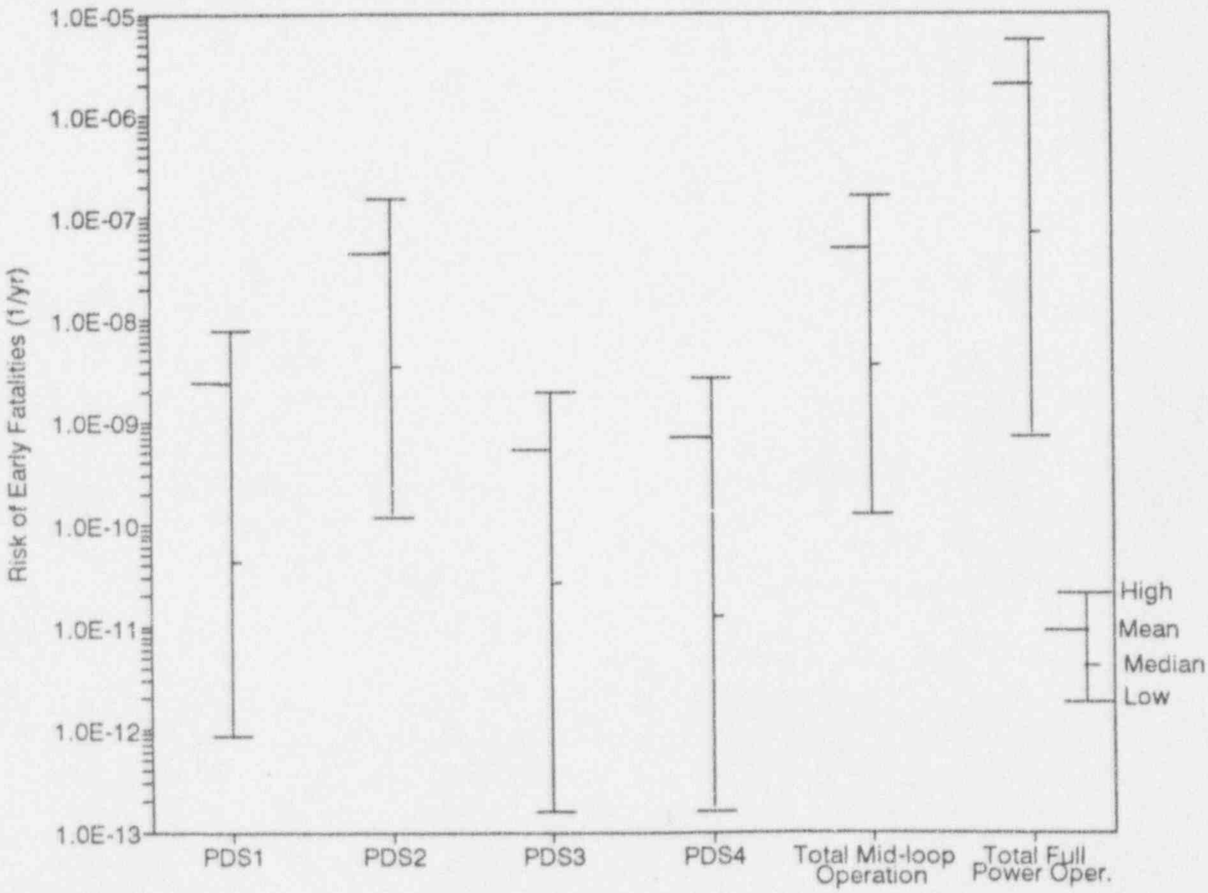


Figure 9.7 Distribution of Risk of Early Fatality for the Plant Damage States

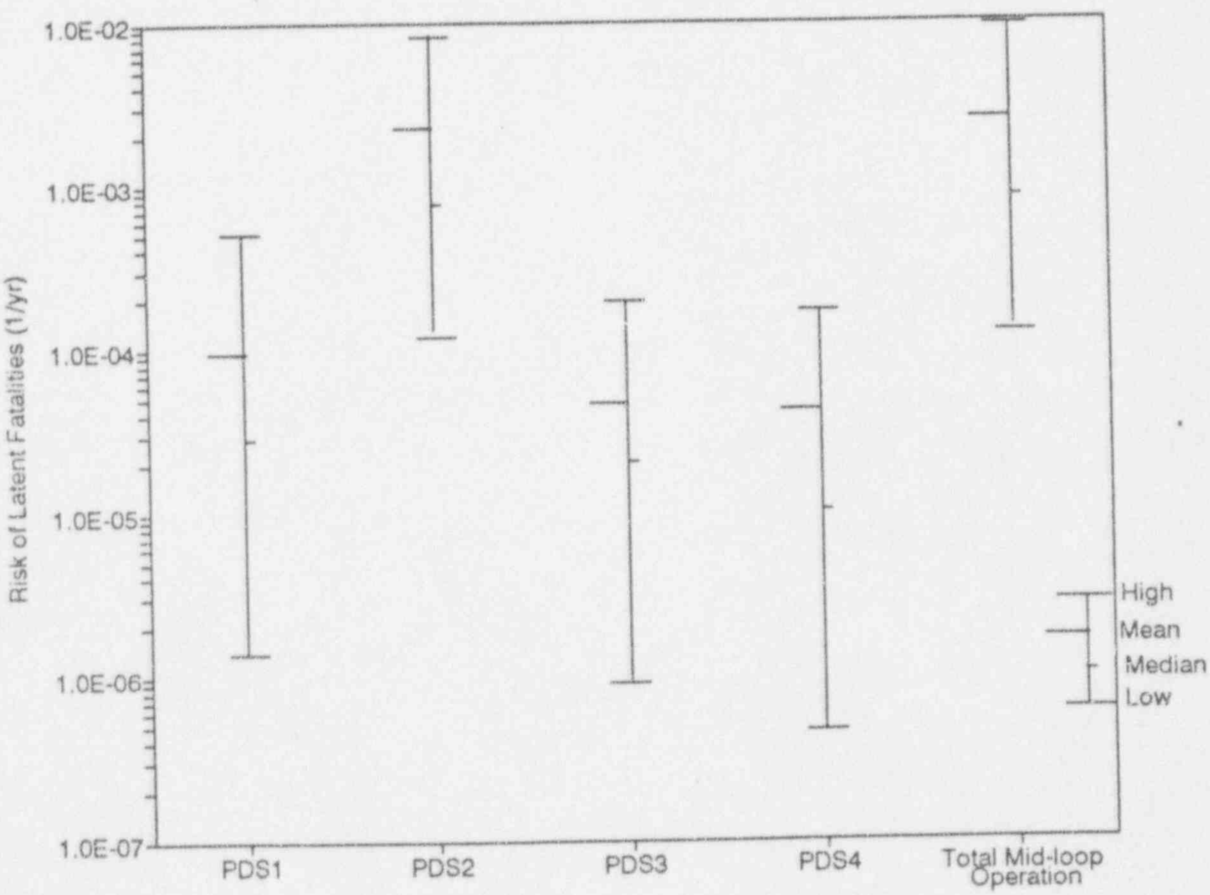


Figure 9.8 Distribution of Risk of Latent Cancer Fatalities to 50 Miles for the Plant Damage States

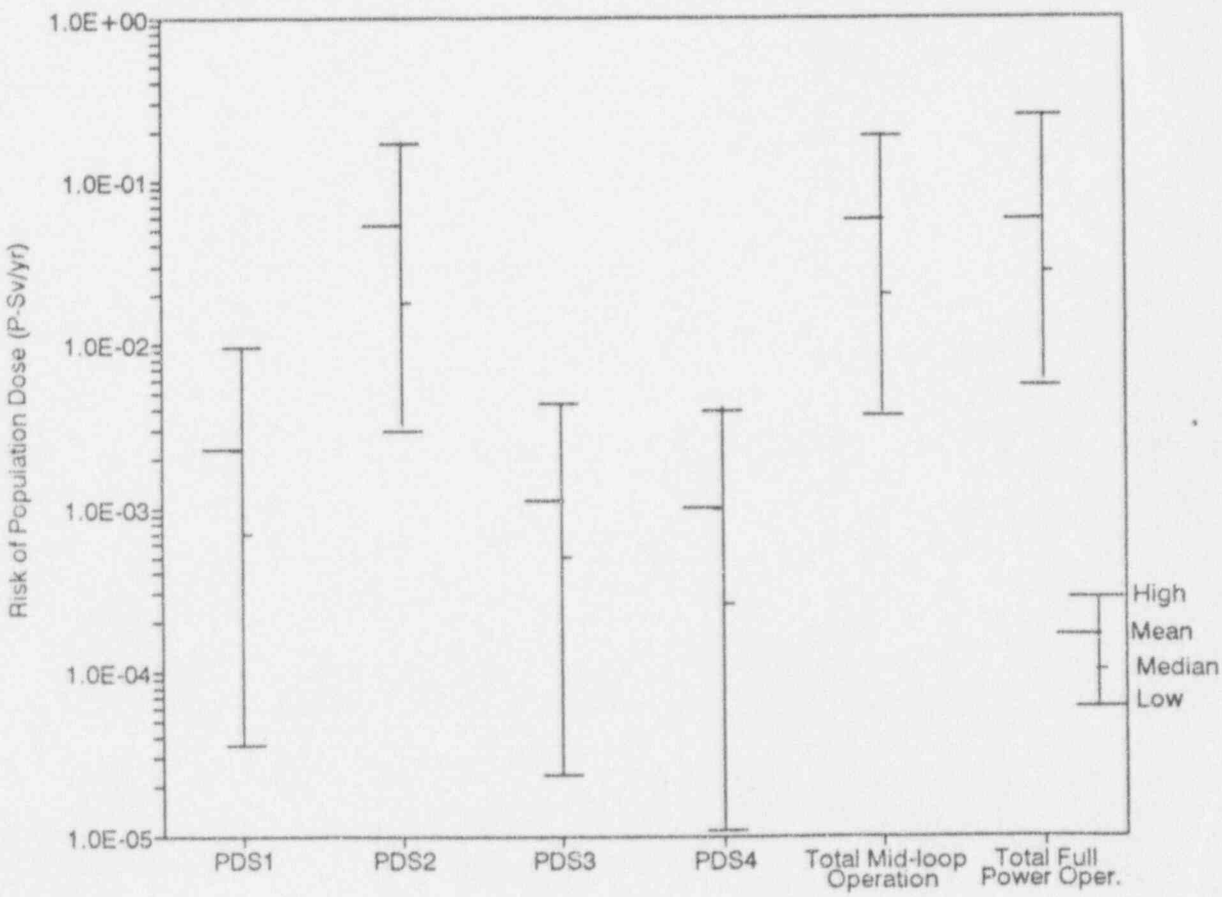


Figure 9.9 Distribution of Risk of Population Dose to 50 Miles for the Plant Damage States

10 COMPARISON TO FULL POWER RESULTS

This chapter compares the results of the integrated risk estimates for accidents during mid-loop operation with the risk estimates[1] for accidents occurring at power. The mid-loop risk estimates are for three mid-loop POSs, namely POS 6 and POS 10 of a refueling outage and POS 6 of a drained maintenance outage. The risk results therefore do not represent the risk from all low power and shutdown operations. The risk estimates (because they are on a yearly basis) also reflect the rather short time that the plant is at mid-loop.

10.1 Core Damage Frequency Analysis

The results of the core damage frequency analysis are discussed in detail in Volume 2 of this report, which also includes a comparison with the full power study. In order to appreciate the accident progression analysis and risk estimates a description of the core damage frequency results is included in this chapter. Four statistical measures of the core damage frequency distribution (CDF) for accidents during mid-loop operation are compared with similar measures for accidents during power operation below:

	Core Damage Frequency for Mid-Loop Operation (per reactor year)	Core Damage Frequency for Power Operation (per reactor year)
95th Percentile	1.9E-5	1.0E-4
Mean	4.2E-6	4.1E-5
50th Percentile	2.0E-6	2.5E-5
5th Percentile	3.2E-7	9.8E-6

The mean core damage frequency of accidents initiated by internal events during mid-loop operation is about an order of magnitude lower than the mean frequency of accidents during full power operation. In addition the mean and median frequencies of the two distributions were within a factor of approximately two which indicates that the means were not strongly influenced by the tails of the distribution. However, the tails of the distributions do overlap and therefore for some cases the mid-loop core damage frequency could be higher than the full power frequency.

The CDF analysis is coupled to the accident progression analysis through the plant damage states (PDS). Of particular interest is the characteristics of the PDS groups and their relative contribution to the core damage frequency estimates. The PDS characteristics are important because they strongly influence the subsequent accident progression. Table 10.1 displays the PDS contributors to the core damage frequency for both studies. Four statistical measures (namely the 5th percentile, median, mean and 9th percentile) on the distributions of the various PDS groups are given in Table 10.1.

Accident sequences in which the operators did not correctly diagnose the situation or take proper actions were the largest contributor (approximately two-thirds of the total) to the mean core damage frequency for mid-loop operation. Accident sequences that lead to station blackout during mid-loop operation (loss of the 4 kV Bus is similar to a station blackout) contribute about 10 percent to the mean CDF. Other accidents were identified that resulted in loss of core cooling after depletion of the refueling water storage tank and failure of recirculation. The leading cause of recirculation failure was found to be plugging of the suction from the sump. These accidents contribute about 20 percent to the mean core damage frequency.

Station Blackout accidents were the largest contributor (approximately two-thirds of the total) to the mean core damage frequency for accidents initiated by internal events during power operation. Other accidents initiated by loss-of-coolant accidents (LOCAs), transient events and anticipated transient without scram (ATWS) contributed about 25 percent to the mean CDF. Accidents that result in containment bypass (steam generator tube ruptures (SGTR) and interfacing systems LOCAs) contributed less than 10 percent to the mean CDF.

The plant damage states in Table 10.1 cannot be directly compared because the plant configuration during mid-loop operation is different than the configuration during full power operation. For example a Station Blackout during full power operation will have a different accident progression than a Station Blackout during mid-loop operation. An important difference is that the containment may not be isolated during mid-loop operation whereas the containment was found to be isolated for most of the accidents at full power. Differences in the status of containment integrity during mid-loop and full power operation have an important influence on the accident progression analysis and risk estimates. In the following sections differences in the plant configuration (and hence plant damage states) between mid-loop and low power are indicated.

10.2 Accident Progression Analysis

The plant damage states developed for the mid-loop and full power studies cannot be directly compared. An attempt was therefore made to summarize the results of the accident progression analyses performed for the two studies in such a way that differences in containment status could be ascertained for each of the plant damages. Table 10.2 summarizes the probability of accident progression bins (APB) conditional on the various PDS groups for full power operation (NUREG/CR-4551, Volume 3) and for mid-loop operation (Chapter 6 of this report). The table has been constructed in such a way that APBs have the same meaning in both studies. For example accidents that "bypass" containment were identified in the full-power study but not in the mid-loop study, whereas "containment not isolated" was an important APB for accidents during mid-loop operation but not for full power.

The most significant difference in the results given in Table 10.2 relates to the probability of the containment not being isolated. In the full power study the probability for the containment not being isolated was very small because during power operation the Surry containment is maintained at a subatmospheric pressure and therefore containment leakage would be detected. However, the probability of the containment not being isolated was determined to be high for most of the plant damage states during mid-loop operation. In fact, the plant damage state that is the largest contributor to the mean core damage frequency (PDS 2 - Human Error) has a very high conditional probability for the containment being open. This is because it was determined in the accident progression analysis that if operator error due to failure to diagnose the accident led to core damage then the operators probably would not have taken measures to isolate containment.

Another difference between the results in Table 10.2 relates to accidents that bypass the containment. In the full power study accidents that bypass the containment contribute less than 10 percent to the mean CDF but, because they are high consequence events, they are large contributors to the risk estimates (as indicated in

Section 10.5 below). However, in the mid-loop study accidents that bypass the containment (such as SGTRs or interfacing systems LOCAs) were not included because the configuration of the plant precludes such events.

The probability for early containment failure caused by such phenomena as hydrogen combustion, direct containment heating and steam explosions was found to be very low for all PDS groups in the full study. This is because the failure pressure of the containment was determined to be much higher than the design pressure and the loads predicted from the phenomena were generally lower than the failure pressures. The probability for early containment failure was also found to be small for accidents during mid-loop operation except for accidents involving station blackout and loss of the 4 kV bus. For these accident sequences the mean conditional probability for early containment failure was determined to be between 0.1 and 0.2. The cause of early containment failure was determined to be hydrogen combustion, which is a problem during mid-loop operation for two reasons. Firstly, if the operators are able to isolate containment during an accident in mid-loop operation there is a possibility that they may not be able to achieve full pressure retaining capability in the time available. The higher containment failure probability in the mid-loop study therefore reflects the lower pressure retaining capability of the containment relative to the capability expected during power operation. Secondly, for accidents involving station blackout it is unlikely that an ignition source would be available to ignite the hydrogen until power is recovered. This means that large quantities of unburned hydrogen could accumulate in containment. The higher early containment failure probability for station blackout accidents during mid-loop operation therefore also reflects the possibility that power will be restored after a large quantity of hydrogen has accumulated in containment.

The conditional probability of late containment failure, caused by the core debris penetrating the basemat or by overpressurizing the containment (due to the accumulation of steam and noncondensible gases) was determined to be between 0.01 and 0.1 for accidents during full power operation. Both of these failure mechanisms were eliminated for accidents during mid-loop operation based on deterministic calculations (described in Chapter 6). The calculations indicated that the decay heat levels for accidents during mid-loop operation were not sufficiently high to cause late containment failure by basemat penetration or containment overpressurization.

Finally, the mean conditional probability of the containment being intact (i.e., isolated, not bypassed, no excessive leakage, and no containment failure) was determined to be high (i.e., between 0.8 and 0.9) for all PDS groups in the full power except for the PDS group containing bypass accidents. As noted above, bypass accidents contribute less than 10 percent to the mean CDF in the full power study. The mean conditional probability of the containment being isolated varied over a wide range for accidents during mid-loop operation. The range varied from 0.05 (Human Error PDS) to about 0.9 for accidents involving loss of recirculation. However, as the Human Error PDS is the largest contributor to the mean CDF the probability of the containment being intact conditional on the mean CDF for all internal events during mid-loop operation was less than 0.3. This compares with a probability of the containment being intact conditional on the mean CDF for accidents during power operation of over 0.8. This difference in containment integrity during mid-loop and full power operation has an important influence on the risk estimates as indicated in Section 10.2.5.

10.3. Results of the Source Term Analysis

The source term model (SURSOR) used in the full power study was considered suitable for use in the mid loop study with only minor modifications. This suitability was based on comparisons with calculations from a deterministic code, MELCOR, and the views expressed by an expert review panel drawn from staff at Sandia and Brookhaven National Laboratories. Therefore, as the same source term model was used in the two studies the source terms are similar for similar accident progression bins. Although the source term calculations are similar for the two studies the risk estimates for mid-loop operation are influenced by the changing radionuclide inventories for the various accidents because they can occur a long time after shutdown. In order to account for the changing radionuclide inventory the partitioning method used in the full power study to combine the source terms into a smaller number of representative source term groups had to be modified for the mid-loop study. Therefore, a direct comparison of the source term groups determined for the two studies would be difficult because of the changing inventory associated with accidents during mid-loop operation.

10.4. Consequence Analysis

The approach used to calculate offsite consequences was similar in both studies. The major difference was that the latest version of the MACCS code was used to evaluate the offsite consequence measures in the mid-loop study. The latest version of MACCS incorporates the BEIR V update to the latent cancer versus dose relationship, whereas the full power study used in an earlier version of MACCS, which did not include the latest BEIR V update. The latest BEIR V update gives a factor of approximately three times higher latent cancers for the same value of population dose. Therefore population dose is used rather than latent cancer fatalities to facilitate comparison with the full power results in Section 10.2.5 below.

10.5 Integrated Risk Analysis

Figures 10.1, 10.2 and 10.3 present four statistical measures of the distributions of the major contributors (plant damage states) to three consequence measures for accidents during mid-loop operation obtained from this study. Similar statistical measures for full power operation obtained from the NUREG-1150 study of Surry are also included in the figure.

Figure 10.1 indicates that the mean risk of offsite early health effects is over two orders of magnitude lower for accidents during mid-loop operation than for full power. This is due to the natural decay of the radionuclide inventory (because the accidents occur a long time after shutdown) particularly the short-lived isotopes of iodine and tellurium, which are primarily associated with early health effects. The distributions obtained for long-term health effects (measured by population dose in Figure 10.3) for mid-loop and full power operation appear to be very similar. The reason why the population dose distributions are similar but the core damage frequency distributions are an order of magnitude lower for mid-loop operation is explored in the following paragraphs.

Accident sequences in which the operators did not correctly diagnose the situation or take proper actions (plant damage state 2 in Table 10.1) were the largest contributor to the total core damage frequency distribution for mid-loop operation. Accident sequences that lead to station blackout during mid-loop operation (plant damages states 1 and 4 in Table 10.1) contribute about 10 percent to the mean CDF. Other accidents (plant damage state 3 in Table 10.1) were identified that resulted in loss of core cooling after depletion of the refueling water storage tank and failure of recirculation. The leading cause of recirculation failure was found to be plugging of the suction from the sump. These accidents contribute about 20 percent to the mean core damage frequency.

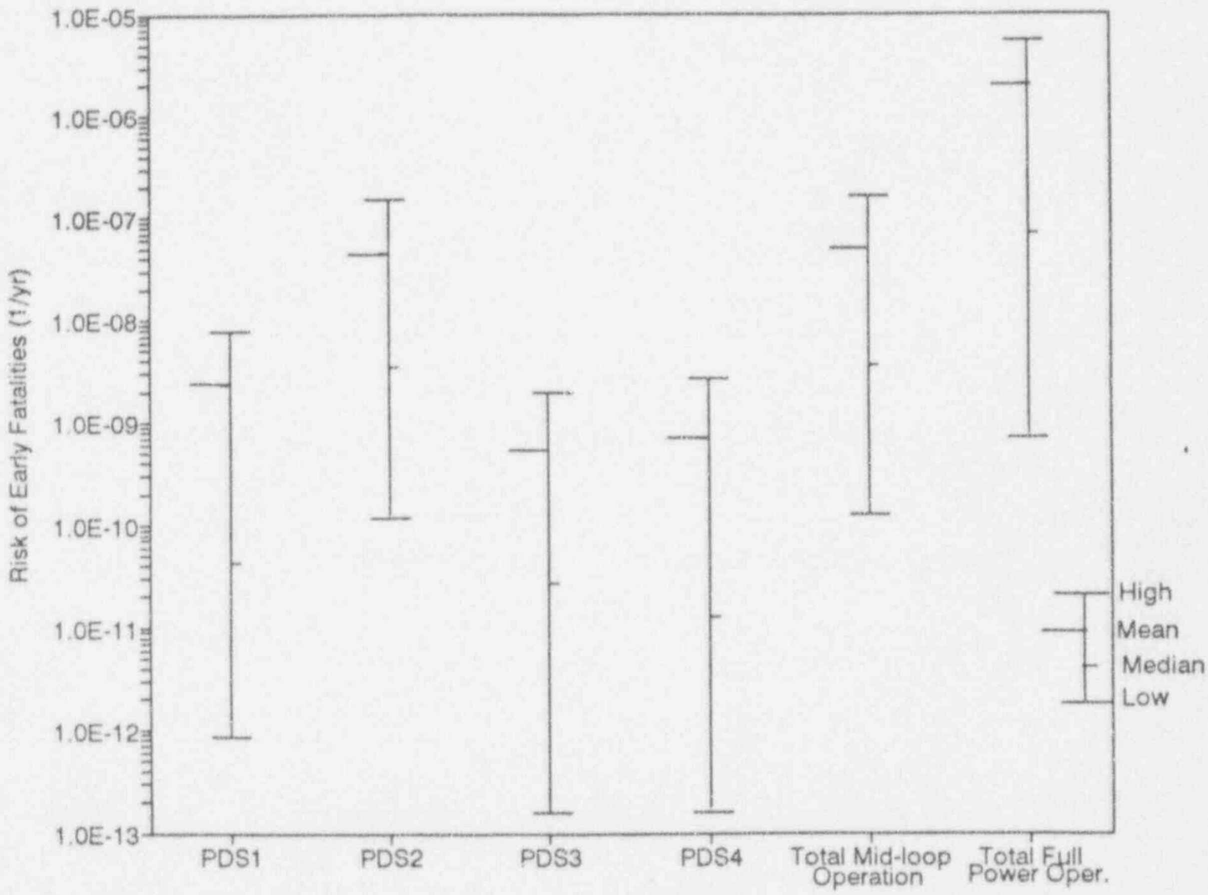
From an inspection of Figure 10.3 it is clear that plant damage state 2 is almost equivalent to the total risk distribution for the population dose. The distributions for PDS3 and PDS4 are almost entirely below the distribution for PDS2. The distributions for PDS3 and PDS4 consist of very low consequence estimates and do not impact the total risk distribution. This is because it was determined in the accident progression analysis for PDS2 that if operator error due to failure to diagnose the accident led to core damage then it was unlikely that the operators would have taken measures to isolate containment. The probability of the containment being open therefore, was very high for accident sequences in plant damage state 2. The probability of the containment not being isolated was found to be lower for the other plant damage states and thus their relative contribution to the offsite health effects was smaller. For example, while plant damage state 4 (recirculation failure due mainly to sump plugging) contributed almost 20% to the mean core damage frequency its contribution to the mean population dose was much smaller. This is because due to the recognition of the problem by the operators and the long times involved, the operators were assumed to have a high probability of being able to isolate the containment and the probability of a large source term from this type of accident was calculated to be small.

In summary, accident sequences involving human error were the largest contributors to the core damage frequency during mid-loop operation and even larger contributors to the offsite risk estimates because it was determined that during these sequences the operators would be unlikely to achieve containment isolation. Therefore, during mid-loop operation the probability of loss of containment integrity conditional on core damage was assessed to be high.

In comparison, in the full power study accident sequences that lead to station blackout were the largest contributors to core damage frequency but not to the offsite risk estimates. This is because containment performance at Surry was found to be very good for this class of accidents even if the molten core penetrates the lower head of the reactor vessel. Therefore accidents with lower frequencies but higher source terms which bypassed the containment, such as interfacing system loss of coolant accidents (ISLOCAs) and steam generator tube ruptures (SGTRs) were found to be the largest contributors to mean risk estimates in the full power study. Thus the loss of containment integrity conditional on core damage was determined to be small for severe accidents at full power.

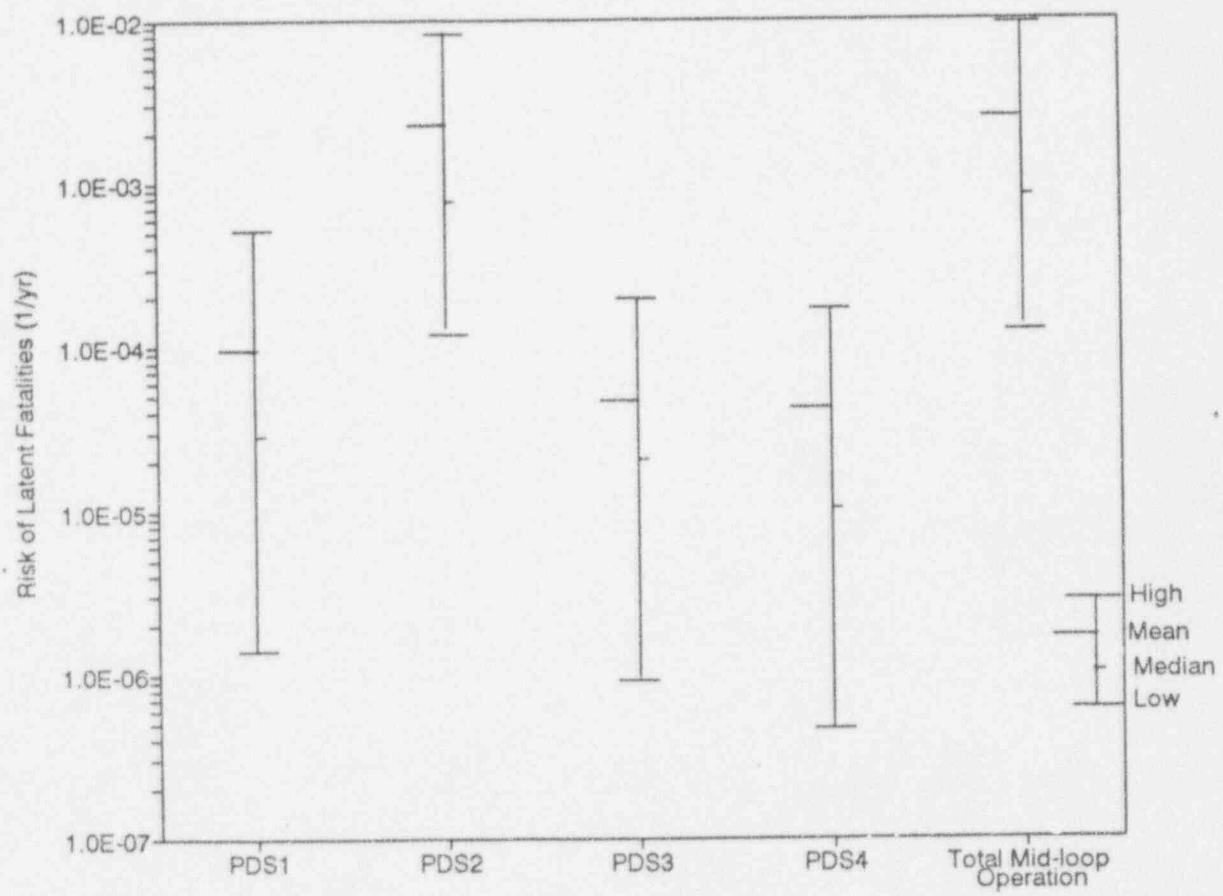
10.6 References

1. Breeding, R. J., et al., "Evaluation of Severe Accident Risks: Surry Unit 1," NUREG/CR-4551, Volume 3, Revision 1, Parts 1 and 2, 1990.



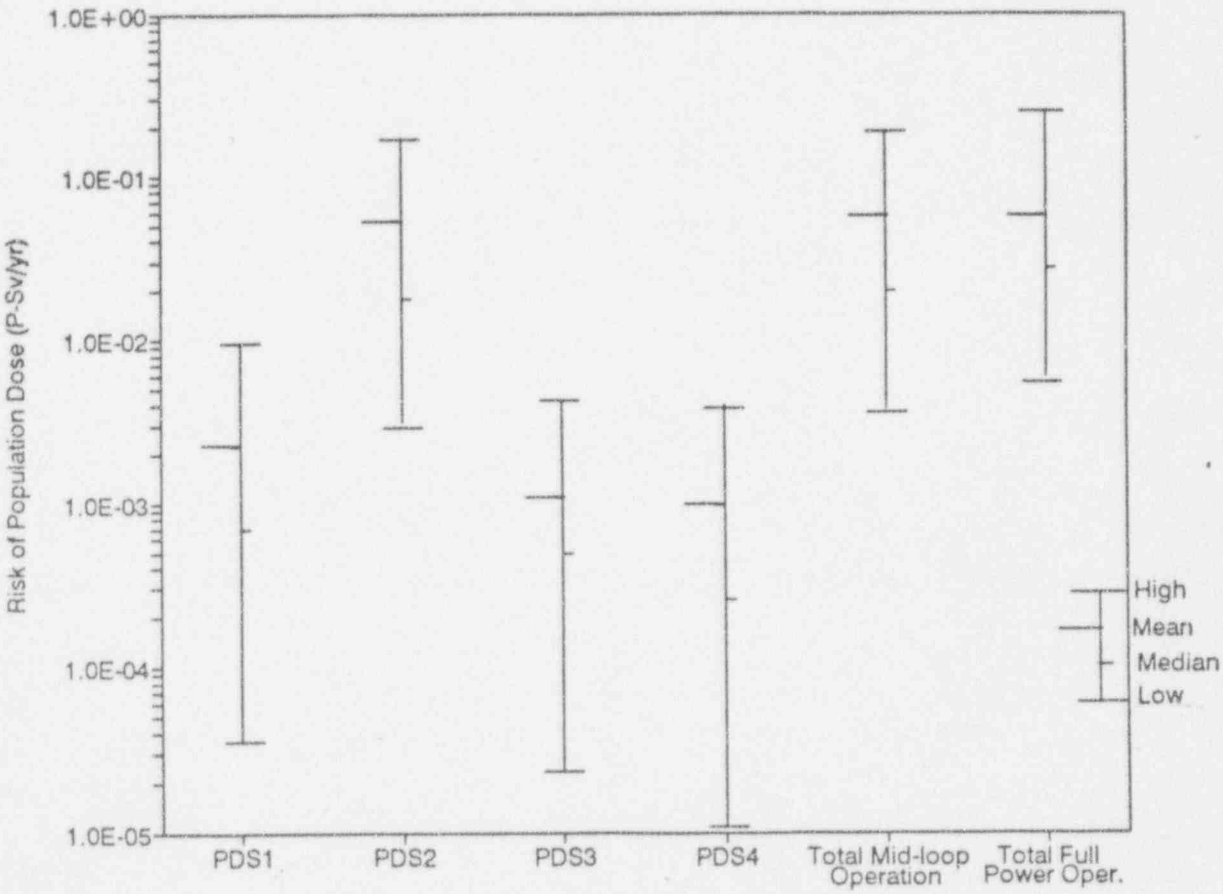
1. **Station Blackout (SBO)**
2. **Human Error**
3. **Recirculation Failure**
4. **Loss of 4 kV bus (similar to SBO)**

Figure 10.1. Distribution for Risk of Early Fatalities for the Plant Damage States



1. **Station Blackout (SBO)**
2. **Human Error**
3. **Recirculation Failure**
4. **Loss of 4 kV bus (similar to SBO)**

Figure 10.2. Distribution of Risk of Latent Fatalities for the Plant Damage States



1. **Station Blackout (SBO)**
2. **Human Error**
3. **Recirculation Failure**
4. **Loss of 4 kV bus (similar to SBO)**

Figure10.3. Distribution for Risk of Population Dose to 50 Miles for the Plant Damage States

Table 10.1 Comparison of the PDS Core Damage Frequencies (per reactor year) for Mid-Loop and Full-Power Operation (Internal Events Only)

Full-Power Operation				
PDS	**5th Percentile**	**50th Percentile**	**Mean**	**95th Percentile**
Station Blackout				
Short Term	1.2E-7	1.5E-6	5.4E-6	2.1E-5
Long Term	1.6E-6	1.1E-5	2.2E-5	6.4E-5
ATWS	2.9E-8	4.2E-7	1.4E-6	6.5E-6
LOCAs	1.2E-6	3.9E-6	6.1E-6	2.0E-5
Interfacing LOCA	3.6E-11	4.9E-8	1.6E-6	8.2E-6
SGTR	4.5E-7	1.4E-6	1.8E-6	4.7E-6
Total	9.8E-6	2.5E-5	4.1E-5	1.0E-4

Mid-Loop Operation				
PDS	**5th Percentile**	**50th Percentile**	**Mean**	**95th Percentile**
Station Blackout	1.9E-8	1.2E-7	4.0E-7	1.7E-6
Human Errors	2.1E-7	1.0E-6	2.8E-6	1.3E-5
Loss of Recirculation	5.3E-8	4.1E-7	7.8E-7	3.3E-6
Loss of 4 kV Bus	8.9E-9	8.9E-8	2.2E-7	1.3E-6
Total	3.2E-7	2.0E-6	4.2E-6	1.9E-5

Table 10.2 Comparison of the Mean Probabilities APBs for Conditional on PDS Groups for Mid-Loop and Full-Power Operation (Internal Events Only)

Accident Progression Bin Groups	Plant Damage State Groups										
	Full-Power Operation*						Mid-Loop Operation**				
	SBO (2.8E-5)	ATWS (1.4E-6)	Transients (1.8E-6)	LOCAs (6.1E-6)	Bypass (3.4E-6)	All (4.1E-5)	SBO (4E-7)	Human Error (2.8E-6)	Loss of Recirculation (7.8E-7)	Loss of 4 kV Bus (2.2E-7)	All (4.2E-6)
Early Containment Failure	.008	.003	.001	.006	-	.007	.17	-	.03	.11	.03
Late Containment Failure	.079	.046	.013	.055	-	.059	-	-	-	-	-
Containment Bypass	.003	.078	.007	-	1.0	.122	-	-	-	-	-
Containment Not Isolated	-	-	-	-	-	-	.28	.95	.10	.28	.69
No VB, No CF	.310	.528	.217	.586	-	.346	.37	.02	-	.39	.07
VB, No CF	.599	.350	.762	.352	-	.466	.18	.03	.87	.22	.21

* Reproduced from NUREG/CR-4551, Volume 3
** Reproduced from Table 6.7
VB Vessel Breach
CF Containment Failure

11 OPEN ISSUES

Several open issues were identified in the course of the study which potentially impact the risk of mid-loop operation and the uncertainty in the risk. A number of these issues relate to modeling of physical processes while others relate to lack of information. In some cases, if more information was made available then the uncertainty in the risk estimates could be reduced. In other cases significant additional analysis would be required to reduce uncertainty. The open issues have been grouped under four categories: (i) the status of procedures in place for dealing with accident conditions, (ii) the availability of systems for terminating the progress of an accident or mitigating its effects, (iii) environmental conditions in the plant which could impede recovery actions and (iv) recent changes in plant configuration during mid-loop operation.

11.1 Status of Procedures

An important issue surfaced by the study is the status of containment isolation during mid-loop operation and the adequacy of the procedures in place for achieving isolation if an accident occurs. This issue is discussed in more detail in Chapter 4. In the abridged study it was assumed that the containment could not be isolated in the time frame available before core damage and the start of the release of the core inventory. New procedures have been subsequently developed at Surry to address containment closure during mid-loop operation. However, questions still remain in the present study as to the adequacy of these procedures in ensuring the pressure retaining capability of the containment even if it is successfully isolated. This issue therefore remains an important contributor to the uncertainty associated with containment performance and determination of risk during mid-loop operation.

There are no procedures in place to ensure that the containment sump will be available as a source of water for recirculation cooling during an accident occurring in mid-loop operation. Plugging of the sump by temporarily stored materials required for performing plant maintenance during shutdown was found to be one of the contributors to core damage and risk due to failure of recirculation cooling.

11.2 Systems Unavailability

There is no requirement during mid-loop operation at Surry for the containment sprays to be available. Containment sprays are an important system during accident conditions for condensing steam and removing heat. Sprays are also potentially effective as a mitigation system for scrubbing fission products released as an aerosol and reducing the source term to the environment. Spray availability was therefore treated as an uncertainty parameter in the analysis; its potential availability during mid-loop operation was based on discussions with Surry plant personnel. However, if the sprays are available they would have to be manually actuated during mid-loop operation as automatic actuation is disabled at RCS temperature below 350°F.

One open issue relates to the effect of spray activation after core damage when a large amount of radioactive aerosols and gases could be present in the containment atmosphere. If the containment is unisolated water droplets from the sprays could displace the atmosphere inside containment and cause the aerosols and gases to be released through the opening in the containment boundary at a faster rate than if the sprays had not been activated. This effect could exacerbate the release to the environment; however, it was not modeled in the present study.

11.3 Impact of Environmental Conditions on Recovery Actions

The impact of environmental conditions in the plant after the start of bulk boiling on the potential for successfully performing recovery actions is another important issue. It may be difficult to carry out recovery actions, which cannot be carried out from the control room, after bulk boiling of the reactor coolant inventory begins. There are several actions during mid-loop operation that can only be performed by entering the containment, for example, restoring RHR and, for station blackout sequences, opening valves to feed the steam generator. The HRA considered the impact of environment as part of the quantification of recovery actions. At temperatures around 140–150°F, the air is too hot for normal pulmonary function and self-breathing respirators may be required for emergency personnel which would also significantly decrease the possibility of success of recovery actions. The uncertainty in the status of containment, referred to above, cuts across this issue. If the containment was isolated, it is unlikely that it would be re-opened to undertake a recovery action once it was recognized that core uncovery was imminent or had occurred as indicated by the radiation monitors. On the other hand, if the containment were unisolated or had no pressure holding capability, the high radiation levels in its immediate vicinity as shown by the onsite dose rates would also make recovery actions inside it unlikely. The impact of environment on the ability of operators to perform recovery actions remains an important issue which contributes to the overall uncertainty.

The impact of recovering cooling water early in the accident progression after core uncovery but before vessel breach is also an open issue. If the clad becomes embrittled on heat up it could fracture on quenching releasing the gap inventory. Water could then enter the ruptured fuel rods and leach out iodine (and other volatile fission products) from the fuel matrix. Depending on temperature and solubility limits, the iodine would be partitioned between the water in the vessel and the containment atmosphere. While this accident scenario is not likely to have any significant offsite consequences, it could have important onsite implications particularly for recovery actions. This type of release was not modeled in the study.

An issue related to the environmental conditions during accident progression which was also surfaced in the abridged study is related to the onsite dose predictions. Because containment performance is uncertain, the onsite "parking lot" dose rates are large. This finding highlights the importance of onsite evacuation schemes to limit the potential consequences to the exposed workers because there is a much larger population of onsite personnel performing maintenance duties, etc. during shutdown operation as compared with normal, full power operation.

11.4 Changes in Plant Configuration During Mid-Loop Operation

The impact of the ongoing risk study of mid-loop operation has had an impact on plant configuration and plant procedures during shutdown at Surry. The study has identified potential vulnerabilities over the last few years and the plant staff have responded, if they felt that a response was warranted, by making changes and improvements to plant configuration and procedures during shutdown (including mid-loop operation) to reduce these vulnerabilities. While these responses are encouraging and lead to improved plant safety, it has precluded an analysis based on a constant plant configuration and operations. In order to complete the study, some compromises had to be made therefore on how much new information could be incorporated within the time available.

11.5 Scope of the Study

Finally, this study is limited to internally initiated accidents during mid-loop operation, which reflects only two plant operating states (POS 6 and POS 10) for two types of outage (refueling and drained maintenance). The risk estimates therefore do not reflect the risk from all modes of operation that might occur during shutdown. If changes are made to reduce perceived vulnerabilities during mid-loop operation, they might have an adverse effect in other plant configurations and increase the risk. Changes to plant configuration and operation during low power and shutdown conditions should be made in light of a full PRA covering all POSs and outage types. In this way a determination can be made as to whether or not the reduction of risk in one POS could cause the risk in another POS or the overall risk of the plant to increase.

12 SUMMARY AND CONCLUSIONS

A systematic and integrated evaluation of risk has been performed for mid-loop operation at the Surry Unit 1 plant. The analysis includes accident frequency, accident progression, source terms, consequences, integrated risk and a determination of the uncertainty in each of the component analyses and in the final risk measures.

The integrated analysis takes into account the long time after shutdown that the accidents can occur and the impact of the consequent decay in power level and radionuclide inventory on the risk. The inclusion of this time factor in a risk analysis is a new development in PRA technology and represents a strength of the study from the methodological standpoint.

The results contained in the preceding Chapters of this report are based on the analysis of accident frequency during mid-loop operation (documented in Volume 2) where the accident sequences leading to core damage were binned into four plant damage state (PDS) groups: PDS 1 (station blackout events), PDS 2 (human errors, failure to diagnose or take proper actions on the part of the operators), PDS 3 (recirculation cooling failure) and PDS 4 (loss of 4 kV bus). These PDS served as the entry point for the further analysis of accident progression, the determination of potential recovery actions, and the evaluation of source terms, consequences and risk.

The main finding of the study is that during mid-loop operation the risk of consequence measures related to long-term health effects, latent cancer fatalities and population dose, are high, comparable to those at full power, despite the much lower level of the decay heat and the radionuclide inventory. The reason for this is that containment is likely to be unisolated for a significant fraction of the accidents initiated during mid-loop operation so the releases to the environment are large and the radionuclide species which mostly contribute to long-term health effects (such as cesium) have long half-lives. PDS 2 (diagnostic and corrective action failures) makes the largest contribution to the integrated risk. Another finding of the study is that the risk of early fatalities is low despite the unisolated containment due to the decay of the short-lived radionuclide species such as iodine and tellurium which contribute to early fatality risk. The integrated risk estimates have a range of uncertainty extending over approximately two orders of magnitude from the 5th to the 95th percentile of the distribution. The conclusions drawn from this finding are discussed below.

Containment Status

The major factor driving the risk is the status of containment during mid-loop operation. As discussed in more detail in Chapters 4 and 11, there is a high probability that the containment is either unisolated or that it would not have full pressure retaining capability during mid-loop operation. This is particularly the case for PDS 2. If the operators fail to diagnose the accident it was judged unlikely that they would take action to isolate containment or could succeed in doing so within the available time frame. For PDS 2, it was determined that the conditional probability (conditional on core damage) of the containment being unisolated ranged from 0.67 (5th percentile) to 0.99 (95th percentile) with a mean of 0.89. For other PDSs, the conditional probability of isolating the containment was judged to be higher. Overall, however, the conditional probability of the containment being unisolated ranged from 0.39 (5th percentile) to 0.88 (95th percentile) with a mean of 0.67. This factor played a significant role in influencing the risk estimates of mid-loop operation.

During the course of the study, Surry plant personnel made available new procedures for containment closure during mid-loop operation. While this response is encouraging in recognizing the need to reduce the vulnerability of the plant during mid-loop operation, it was difficult to assess the adequacy of these procedures in ensuring the pressure retaining capability of the containment within the time frame encompassed by this study. This feature contributed significantly to the uncertainty in containment status and the estimate of risk.

Availability of Containment Sprays

There is no requirement at Surry for the containment sprays to be available during shutdown. Plant records show that the spray systems could be inoperable because of maintenance. Spray availability was modeled as an uncertainty parameter in the integrated risk analysis. Since the sprays perform an important safety function in mitigating the effects of releases, spray unavailability contributed both to the risk and its uncertainty.

Possibility of Core Damage Arrest

The inclusion of the possibility of arresting the core degradation process before vessel failure is an important feature of this analysis as it was for the full power study. Termination of the accident in-vessel can significantly reduce some of the fission product releases and thus the risk. The potential for core recovery depends on the nature of the accident progression and is different for the various PDS Groups. For PDS Group 1 (SBO events) the conditional probability of core damage arrest (conditional on core damage) ranges from 0.45 (5th percentile) to 0.71 (95th percentile) with a mean of 0.55. The dominant factor affecting the arrest of core damage for this PDS Group is recovery of offsite power. For PDS Group 4 (loss of 4 kV bus) the conditional probability of arresting core damage ranges from 0.03 (5th percentile) to 0.86 (95th percentile) with a mean of 0.59. Recovery of the 4 kV bus is the major factor for this PDS. Accidents in PDS Group 2 are attributable to human error and the conditional probability of arresting core damage for this PDS Group ranges from 0.39 (5th percentile) to 0.45 (95th percentile) with a mean of 0.42. Recovery for PDS Group 2 depends on the operators making a correct diagnosis or taking proper action. Accidents in PDS Group 3 are initiated by recirculation failure due to sump plugging and recovery of the recirculation system and arresting further degradation of the core was assumed to be not possible after core damage occurs. Overall, the conditional probability of core damage arrest ranged from 0.23 (5th percentile) to 0.44 (95th percentile) with a mean of 0.35.

Comparison with Full Power Study

The results of the present study are compared in detail with the results of the full power study in Chapter 10. The comparison has shown that the mean core damage frequency for accidents during mid-loop operation is about an order of magnitude lower than the mean frequency of accidents caused by internal events at full power. However, the risk distributions obtained for comparable long term health consequences (measured by the population dose at 50 miles) are very similar in the two studies. In NUREG-1150, the 50 mile population dose ranged from about 5E-3 P-Sv/year (5th percentile) to 3E-1 P-Sv/year (95th percentile) with a mean of 6E-2 P-Sv/year. For mid-loop operation, the corresponding range is from 4E-3 P-Sv/year (5th percentile) to 2E-1 P-Sv/year (95th percentile) with a mean of 6E-2 P-Sv/year. What this finding implies is that the lower decay heat and lower radionuclide inventory of the mid-loop operating state, compared with full power, is offset by the likelihood of containment being unisolated. Finally, the mean risk of early health effects is over two orders of magnitude lower for accidents during mid-loop operation than for accidents during full power operation. This is due to the natural decay of those radionuclide species which have the greatest impact on early fatality risk because accidents during mid-loop operation occur a long time after shutdown.

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EVALUATION OF POTENTIAL SEVERE ACCIDENTS DURING LOW POWER AND SHUTDOWN OPERATIONS AT SURRY UNIT-1

Evaluation of Severe Accident Risks During Mid-loop Operations

Appendices

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APPENDIX A

SUPPORTING INFORMATION FOR THE PDS ANALYSIS

APPENDIX A

Appendix B presents additional information on how the 2186 core damage frequency cutsets were grouped together into a smaller number of plant damage states. The process is described in Sections 5.2 and 5.3 of Part 1 of this volume and that material will not be repeated here.

Initially seven characteristics were defined that largely determine the progression of the accident. These seven characteristics (refer to Table A.1) were then used to allocate each of the 2186 cutsets to a PDS. The PDS designators of the 82 highest frequency cutsets are given in Table A.2. Only cutsets with frequencies higher than 10^{-8} per year are included in the table. These 82 cutsets comprise about 70% of the core damage frequency. However, the remaining 2104 cutsets with frequencies lower than 10^{-8} per year were all allocated to PDS designators and included in the subsequent accident progression analysis.

Many of the cutsets in Table A.2 have similar PDS designators. When all of the cutsets with the same PDS designators were combined, 48 individual plant damage states resulted. The 48 PDSs are listed in Table A.3. An uncertainty analysis described in Chapter 12 of Volume 2 of this report was performed for the 48 PDSs. Four statistical measures of the distribution obtained for each PDS are also included in Table A.3.

The 48 PDSs listed in Table A.3 were regrouped into four PDS groups to be processed by the accident progression event tree (refer to Appendix B). Table A.4 indicates how the 48 PDSs were placed into four more general PDS groups.

An uncertainty analysis was performed for the four PDS groups using 100 samples. The PDS group frequencies for each of the 100 samples are given in Table A.5, which also provides the frequencies for the four time windows. The information in Table A.5 was used to obtain the uncertainty in the PDS and Time Window frequencies. Four statistical measures of the PDS and Time Window frequency distributions are presented in Table 5.5 of Part 1 of this volume.

Table A.1 PDS Definition

1. Time of Accident Initiation
   - 1: Window 1
   - 2: Window 2
   - 3: Window 3
   - 4: Window 4
2. AC Power
   - Y: Available
   - U: Unrecoverable blackout
   - B: Blackout (recoverable by recovery of off-site power)
   - F: Loss of 4KV Bus
3. Human Error
   - N: No human error or non-recoverable human error
   - D: Diagnosis error
   - A: Action error
4. RCS Status at Onset of Core Damage
   - L: Low pressure
   - G: Intermediate pressure
5. ECCS Status
   - U: Unrecoverable hardware failure
   - R: Recoverable if human error, LOSP, or 4 kV is recovered
   - C: Failure of recirculation
6. Recirculation Spray Status
   - R: Recoverable
   - U: Unrecoverable
7. RWST Status
   - Y: Injected
   - R: Not injected but recoverable
   - N: Not injected and not recoverable

Table A.2 Plant Damage State Assignment of the Dominant Cutsets

Cutset Number*	PDS Designator	Frequency per Reactor Year
1	2YDLRRR	2.20E-07
2	3YDLRRR	2.07E-07
3	2YDLRRR	1.70E-07
4	1YNGCUY	1.46E-07
5	3YDLRRR	1.46E-07
6	1YDGRRR	1.24E-07
7	2YDLRRR	1.22E-07
8	1YNGCUY	1.08E-07
9	3YDLRRR	9.52E-08
10	2YDLRRR	9.37E-08
11	1YDGRRR	9.04E-08
12	1YDGRRR	8.73E-08
13	2YDLRRR	7.74E-08
14	1BNGRRR	7.38E-08
15	3YDLRRR	6.70E-08
16	1YDGRRR	5.82E-08
17	1FNGRRY	5.41E-08
18	2YDLRRR	5.38E-08
19	3YDLRRR	5.29E-08
20	3YDLRRR	5.07E-08
21	3YDLRRR	4.86E-08
22	2BNLRRR	4.75E-08
23	1YNGCYY	4.39E-08
24	2YDLRRR	4.27E-08
25	4BNLRRR	4.07E-08
26	4YDLRRR	3.91E-08
27	1BNGRRR	3.63E-08

Table A.2 (continued)

Cutset Number*	PDS Designator	Frequency per Reactor Year
28	2YDLRRR	3.39E-08
29	2UDLRUR	3.38E-08
30	2YDLRRR	3.36E-08
31	2YDLRRR	3.30E-08
32	1BNGRRR	3.27E-08
33	1YNGCYY	3.23E-08
34	2YDLRRR	2.97E-08
35	1YNGCUY	2.95E-08
36	3YDLRRR	2.92E-08
37	2YDLRRR	2.91E-08
38	3UDLRUR	2.91E-08
39	3YDLRRR	2.89E-08
40	2BNLRRR	2.62E-08
41	1FNGRRY	2.52E-08
42	1YNGCUY	2.47E-08
43	3YDLRRR	2.33E-08
44	3YDLRRR	2.23E-08
45	1YNGCUY	2.17E-08
46	3BNLRRR	2.13E-08
47	2YALRUY	2.11E-08
48	2YDLRRR	2.01E-08
49	2YDLRRR	1.89E-08
50	2YDLRRR	1.87E-08
51	2UDLRUR	1.87E-08
52	2YDLRRR	1.86E-08
53	1YNGCUY	1.83E-08
54	3YDLRRR	1.83E-08
55	1YDGRRR	1.79E-08

Table A.2 (continued)

Cutset Number*	PDS Designator	Frequency per Reactor Year
56	1YDGRRR	1.75E-08
57	1YDGRRR	1.73E-08
58	3YDLRRR	1.73E-08
59	1BNGRRR	1.68E-08
60	4YDLRRR	1.68E-08
61	2YDLRRR	1.61E-08
62	1YDGRRR	1.60E-08
63	1YNGCYY	1.53E-08
64	1YAGCRY	1.46E-08
65	1YNGCUY	1.35E-08
66	3YDLRRR	1.34E-08
67	3UDLRUR	1.34E-08
68	3YDLRRR	1.33E-08
69	2YDLRRR	1.29E-08
70	2FALRRY	1.26E-08
71	1YNGCYY	1.19E-08
72	1YNGCYY	1.19E-08
73	3YDLRRR	1.19E-08
74	1FNGRRY	1.15E-08
75	1YNGCUY	1.13E-08
76	2YDLRRR	1.11E-08
77	1FNGRRY	1.09E-08
78	1YAGCRY	1.08E-08
79	3YDLRRR	1.07E-08
80	4YDLRRR	1.07E-08
81	3YDLRRR	1.06E-08
82	2YDLRRR	1.04E-08

*Defined in Table 10.50 of Volume 2 of this report.

Table A.3 Results of Plant Damage State Uncertainty Analysis (per reactor year)

PDS	MEAN	5th Percentile	50th Percentile	95th Percentile
1BNGCRY	2.95E-09	5.90E-11	7.48E-10	1.11E-08
1BNGRRR	1.71E-07	4.07E-09	4.40E-08	6.35E-07
1BNLCRY	1.43E-10	3.93E-13	1.40E-11	4.70E-10
1FAGRRY	9.07E-09	1.39E-10	2.12E-09	3.31E-08
1FNGRRR	4.78E-10	6.02E-12	9.57E-11	1.73E-09
1FNGRRY	1.25E-07	2.69E-09	3.43E-08	4.62E-07
1UAGCUY	1.89E-10	1.27E-12	2.92E-11	7.31E-10
1UDGUUR	8.29E-09	7.73E-12	4.40E-10	2.51E-08
1UDLCUY	6.08E-10	8.79E-13	4.23E-11	1.86E-09
1YAGCRY	8.11E-08	4.12E-09	2.84E-08	2.76E-07
1YAGCUY	2.12E-08	5.18E-10	5.48E-09	7.93E-08
1YAGRRR	2.75E-09	1.46E-10	1.04E-09	9.32E-09
1YDGRRR	4.64E-07	1.22E-08	1.19E-07	1.68E-06
1YNGCUY	5.41E-07	1.29E-08	1.28E-07	1.96E-06
1YNGCYY	3.15E-07	1.90E-08	1.08E-07	1.01E-06
1YNGUUR	1.49E-09	2.67E-11	3.54E-10	5.50E-09
1YNGUYR	8.84E-09	6.17E-10	3.93E-09	3.03E-08
1YNLCUY	2.22E-09	2.91E-12	1.30E-10	6.00E-09
1YNLCYY	7.02E-10	2.15E-12	7.03E-11	2.45E-09
2BNLCRY	3.34E-08	9.03E-10	9.76E-09	1.18E-07
2BNLCUY	2.99E-09	1.12E-11	4.46E-10	1.09E-08
2BNLRRR	1.05E-07	4.01E-09	3.57E-08	3.75E-07
2FALRRR	1.46E-08	2.44E-10	3.57E-09	5.48E-08
2FALRRY	4.31E-08	1.17E-09	1.41E-08	1.62E-07
2FNLRRR	2.75E-08	7.12E-10	8.11E-09	9.97E-08
2UALRUY	1.17E-10	1.26E-12	2.40E-11	4.19E-10
2UDLRUR	5.12E-08	1.08E-10	4.21E-09	1.68E-07

Table A.3 (continued)

PDS	MEAN	5th Percentile	50th Percentile	95th Percentile
2YALRRR	7.30E-09	4.48E-10	3.08E-09	2.58E-08
2YALRRY	8.72E-09	5.40E-10	3.78E-09	3.02E-08
2YALRUR	1.50E-08	8.43E-10	5.81E-09	5.20E-08
2YALRUY	4.93E-08	2.05E-09	1.61E-08	1.85E-07
2YALRYR	7.06E-08	1.14E-08	4.11E-08	2.06E-07
2YDLRRR	1.08E-06	3.10E-08	2.80E-07	3.62E-06
2YNLCUY	4.84E-08	1.12E-09	1.28E-08	1.59E-07
2YNLCYY	1.94E-08	1.39E-09	8.11E-09	6.61E-08
3BDLRRR	2.39E-10	2.31E-13	1.30E-11	6.68E-10
3BNLRRR	4.18E-08	1.65E-09	1.38E-08	1.40E-07
3UDLRUR	4.24E-08	9.48E-11	3.49E-09	1.34E-07
3YALRRR	5.31E-09	4.18E-10	2.41E-09	1.80E-08
3YALRUR	2.55E-08	1.53E-09	1.05E-08	8.64E-08
3YALRYR	4.38E-08	6.96E-09	2.57E-08	1.30E-07
3YDLRRR	9.15E-07	2.51E-08	2.42E-07	3.19E-06
4BNLRRR	5.81E-08	7.00E-10	8.98E-09	1.96E-07
4UDLRUR	1.16E-08	1.47E-11	6.35E-10	3.37E-08
4YALRRR	2.75E-09	1.37E-11	3.86E-10	9.32E-09
4YALRUR	2.28E-08	3.76E-10	4.85E-09	7.55E-08
4YALRYR	8.06E-08	3.34E-09	2.95E-08	2.54E-07
4YDLRRR	1.28E-07	1.42E-09	1.81E-08	3.85E-07

Table A.4 Grouping of 48 PDSs into Four PDS Groups

PDS Group Number	Group Description	PDS
1	Station Blackout	1BNGRRR
		2BNLRRR
		3BDLRRR
		3BNLRRR
		4BNLRRR
2	Human Error	1YAGRRR
		1YDGRRR
		2UALRUY
		2UDLRUR
		2YALRRR
		2YALRRY
		2YALRUR
		2YALRUY
		2YALRYR
		2YDLRRR
		3UDLRUR
		3YALRRR
		3YALRUR
		3YALRYR
		3YDLRRR
		4UDLRUR
		4YALRRR
		4YALRUR
		4YALRYR
		4YDLRRR
3	Recirculation Failure	1BNGCRY
		1BNLCRY
		1UAGCUY

Table A.4 (continued)

PDS Group Number	Group Description	PDS
		1UDGUUR
		1UDLCUY
		1YAGCRY
		1YAGCUY
		1YNGCUY
		1YNGCYY
		1YNGUUR
		1YNGUYR
		1YNLCUY
		1YNLCYY
		2BNLCRY
		2BNLCUY
		2YNLCUY
		2YNLCYY
4	Loss of 4 kV Bus	1FAGRRY
		1FNGRRR
		1FNGRRY
		2FALRRR
		2FALRRY
		2FNLRRR

Table A-5

Frequencies of PDS groups for Each Time Window for the 100 Observations

OBS 1	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.35938E-07	0.76949E-08	0.31299E-07	0.72024E-08	0.82134E-07	25.85
Window 2	0.12034E-07	0.12130E-06	0.60941E-08	0.10525E-07	0.14995E-06	47.19
Window 3	0.18983E-08	0.74006E-07	0.00000E+00	0.00000E+00	0.75904E-07	23.89
Window 4	0.61907E-10	0.96789E-08	0.00000E+00	0.00000E+00	0.97408E-08	3.07
PDS Tot	0.49933E-07	0.21268E-06	0.37393E-07	0.17728E-07	0.31773E-06	
%	15.72	66.94	11.77	5.58		
OBS 2	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.21056E-06	0.93224E-06	0.32112E-05	0.38126E-07	0.43921E-05	65.36
Window 2	0.28091E-07	0.99648E-06	0.57716E-07	0.50233E-07	0.11325E-05	16.85
Window 3	0.78191E-08	0.90320E-06	0.00000E+00	0.00000E+00	0.91102E-06	13.56
Window 4	0.25194E-08	0.28164E-06	0.00000E+00	0.00000E+00	0.28416E-06	4.23
PDS Tot	0.24899E-06	0.31136E-05	0.32689E-05	0.88359E-07	0.67198E-05	
%	3.71	46.33	48.65	1.31		
OBS 3	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.50705E-06	0.31922E-05	0.71610E-06	0.54813E-07	0.44702E-05	79.14
Window 2	0.28312E-06	0.33085E-06	0.35390E-07	0.68466E-07	0.71783E-06	12.71
Window 3	0.15038E-06	0.26066E-06	0.00000E+00	0.00000E+00	0.41105E-06	7.28
Window 4	0.12308E-07	0.37336E-07	0.00000E+00	0.00000E+00	0.49645E-07	0.88
PDS Tot	0.95286E-06	0.38210E-05	0.75149E-06	0.12328E-06	0.56487E-05	
%	16.87	67.64	13.30	2.18		
OBS 4	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.79522E-07	0.10539E-06	0.45254E-06	0.35306E-06	0.99052E-06	49.19
Window 2	0.15084E-06	0.28002E-06	0.83831E-07	0.20553E-06	0.72022E-06	35.77
Window 3	0.37971E-07	0.21401E-06	0.00000E+00	0.00000E+00	0.25198E-06	12.51
Window 4	0.43635E-08	0.46370E-07	0.00000E+00	0.00000E+00	0.5733E-07	2.52
PDS Tot	0.27270E-06	0.64579E-06	0.53637E-06	0.55860E-06	0.20135E-05	
%	13.54	32.07	26.64	27.74		
OBS 5	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.63809E-07	0.31048E-07	0.33797E-06	0.63118E-07	0.49594E-06	15.09
Window 2	0.29902E-07	0.67669E-06	0.29259E-07	0.45196E-07	0.78104E-06	23.77
Window 3	0.25323E-07	0.82070E-06	0.00000E+00	0.00000E+00	0.84602E-06	25.75
Window 4	0.32023E-06	0.84242E-06	0.00000E+00	0.00000E+00	0.11626E-05	35.39
PDS Tot	0.43926E-06	0.23708E-05	0.36723E-06	0.10831E-06	0.32856E-05	
%	13.37	72.16	11.18	3.30		
OBS 6	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.11972E-07	0.17025E-07	0.71249E-07	0.28420E-08	0.10309E-06	49.82
Window 2	0.13055E-07	0.37166E-07	0.41381E-08	0.10939E-07	0.65299E-07	31.56
Window 3	0.32327E-08	0.29396E-07	0.00000E+00	0.00000E+00	0.32628E-07	15.77
Window 4	0.11984E-08	0.47143E-08	0.00000E+00	0.00000E+00	0.59126E-08	2.86
PDS Tot	0.29458E-07	0.88301E-07	0.75387E-07	0.13781E-07	0.20693E-06	
%	14.24	42.67	36.43	6.66		
OBS 7	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.25172E-07	0.31924E-06	0.21624E-06	0.21064E-06	0.77128E-06	65.49
Window 2	0.81922E-08	0.25981E-06	0.77212E-08	0.12843E-07	0.28856E-06	24.50
Window 3	0.90263E-08	0.81781E-07	0.00000E+00	0.00000E+00	0.90808E-07	7.71
Window 4	0.88940E-09	0.26203E-07	0.00000E+00	0.00000E+00	0.27093E-07	2.30
PDS Tot	0.43280E-07	0.68703E-06	0.22396E-06	0.22348E-06	0.11777E-05	
%	3.67	58.33	19.02	18.98		
OBS 8	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.21966E-07	0.81915E-07	0.55854E-07	0.54081E-09	0.16028E-06	26.27
Window 2	0.63433E-08	0.14400E-06	0.99062E-08	0.15010E-08	0.16175E-06	26.51
Window 3	0.23911E-08	0.20266E-06	0.00000E+00	0.00000E+00	0.20505E-06	33.60
Window 4	0.32022E-07	0.51110E-07	0.00000E+00	0.00000E+00	0.83132E-07	13.62
PDS Tot	0.62722E-07	0.47968E-06	0.65760E-07	0.20418E-08	0.61020E-06	
%	10.28	78.61	10.78	0.33		
OBS 9	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%

Table A-5 (Cont.)

Window 1	0.83072E-07	0.12362E-06	0.15289E-05	0.28536E-07	0.17642E-05	37.03
Window 2	0.16202E-06	0.14201E-05	0.15041E-06	0.24041E-07	0.17566E-05	36.87
Window 3	0.49724E-07	0.11195E-05	0.00000E+00	0.00000E+00	0.11692E-05	24.54
Window 4	0.20709E-07	0.54141E-07	0.00000E+00	0.00000E+00	0.74849E-07	1.57
PDS Tot	0.31552E-06	0.27174E-05	0.16794E-05	0.52577E-07	0.47648E-05	
%	6.62	57.03	35.24	1.10		
OBS 10	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.46735E-07	0.22160E-06	0.99369E-06	0.21884E-06	0.14809E-05	58.50
Window 2	0.42169E-07	0.26825E-06	0.53158E-07	0.21655E-06	0.58013E-06	22.92
Window 3	0.68925E-08	0.40504E-06	0.00000E+00	0.00000E+00	0.41194E-06	16.27
Window 4	0.25548E-07	0.33062E-07	0.00000E+00	0.00000E+00	0.58609E-07	2.32
PDS Tot	0.12134E-06	0.92796E-06	0.10469E-05	0.43539E-06	0.25315E-05	
%	4.79	36.66	41.35	17.20		
OBS 11	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.37537E-07	0.14091E-07	0.41051E-06	0.82066E-08	0.47034E-06	23.97
Window 2	0.33626E-06	0.40611E-06	0.77910E-07	0.10314E-06	0.92342E-06	47.06
Window 3	0.37749E-07	0.31835E-06	0.00000E+00	0.00000E+00	0.35610E-06	18.15
Window 4	0.64349E-07	0.14786E-06	0.00000E+00	0.00000E+00	0.21221E-06	10.82
PDS Tot	0.47590E-06	0.88641E-06	0.48842E-06	0.11135E-06	0.19621E-05	
%	24.25	45.18	24.89	5.67		
OBS 12	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.36364E-07	0.13442E-07	0.29476E-06	0.57167E-07	0.40173E-06	30.62
Window 2	0.50758E-07	0.37000E-06	0.74044E-07	0.36845E-07	0.53165E-06	40.52
Window 3	0.26202E-07	0.30857E-06	0.00000E+00	0.00000E+00	0.33477E-06	25.52
Window 4	0.62415E-08	0.37592E-07	0.00000E+00	0.00000E+00	0.43833E-07	3.34
PDS Tot	0.11956E-06	0.72961E-06	0.36880E-06	0.94012E-07	0.13120E-05	
%	9.11	55.61	28.11	7.17		
OBS 13	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.47102E-07	0.63613E-07	0.15245E-06	0.18551E-06	0.44868E-06	45.28
Window 2	0.10588E-07	0.12628E-06	0.81105E-07	0.59365E-07	0.27734E-06	27.99
Window 3	0.23891E-07	0.78852E-07	0.00000E+00	0.00000E+00	0.10274E-06	10.37
Window 4	0.14112E-07	0.14802E-06	0.00000E+00	0.00000E+00	0.16213E-06	16.36
PDS Tot	0.95693E-07	0.41677E-06	0.23356E-06	0.24487E-06	0.99089E-06	
%	9.66	42.06	23.57	24.71		
OBS 14	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.10893E-06	0.90293E-07	0.39536E-06	0.44081E-07	0.63866E-06	46.82
Window 2	0.26616E-07	0.16602E-06	.12849E-06	0.45216E-07	0.36634E-06	26.86
Window 3	0.14086E-07	0.14014E-06	0.00000E+00	0.00000E+00	0.15423E-06	11.31
Window 4	0.15300E-07	0.18941E-06	0.00000E+00	0.00000E+00	0.20471E-06	15.01
PDS Tot	0.16493E-06	0.58587E-06	0.52384E-06	0.89297E-07	0.13639E-05	
%	12.09	42.95	38.41	6.55		
OBS 15	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.13345E-07	0.74152E-08	0.54946E-06	0.38349E-07	0.60856E-06	59.03
Window 2	0.78420E-08	0.20428E-06	0.71637E-07	0.11974E-07	0.29573E-06	28.69
Window 3	0.50995E-08	0.11308E-06	0.00000E+00	0.00000E+00	0.11818E-06	11.46
Window 4	0.59663E-09	0.77934E-08	0.00000E+00	0.00000E+00	0.83901E-08	0.81
PDS Tot	0.26883E-07	0.33256E-06	0.62109E-06	0.50323E-07	0.10309E-05	
%	2.61	32.26	60.25	4.88		
OBS 16	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.42251E-05	0.16963E-05	0.32839E-05	0.14038E-05	0.10609E-04	53.91
Window 2	0.11232E-05	0.38026E-05	0.20560E-05	0.55104E-06	0.75329E-05	38.27
Window 3	0.13018E-06	0.12824E-05	0.00000E+00	0.00000E+00	0.14126E-05	7.18
Window 4	0.39241E-07	0.87204E-07	0.00000E+00	0.00000E+00	0.12644E-06	0.64
PDS Tot	0.55177E-05	0.68685E-05	0.53399E-05	0.19548E-05	0.19681E-04	
%	28.04	34.90	27.13	9.93		
OBS 17	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.33350E-07	0.55349E-07	0.46075E-07	0.79535E-08	0.14273E-06	42.57
Window 2	0.17503E-07	0.62886E-07	0.96159E-08	0.29756E-08	0.92980E-07	27.73
Window 3	0.46633E-08	0.47898E-07	0.00000E+00	0.00000E+00	0.52562E-07	15.68
Window 4	0.60529E-08	0.40924E-07	0.00000E+00	0.00000E+00	0.46977E-07	14.01

Table A-5 (Cont.)

PDS Tot	0.61569E-07	0.20706E-06	0.55691E-07	0.10929E-07	0.33525E-06	
%	18.37	61.76	16.61	3.26		
OBS 18	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.24249E-08	0.80289E-08	0.73129E-07	0.17586E-07	0.10117E-06	22.35
Window 2	0.44791E-08	0.17859E-06	0.51859E-07	0.26759E-07	0.26169E-06	57.82
Window 3	0.26321E-08	0.75559E-07	0.00000E+00	0.00000E+00	0.78191E-07	17.28
Window 4	0.13179E-08	0.10196E-07	0.00000E+00	0.00000E+00	0.11513E-07	2.54
PDS Tot	0.10854E-07	0.27238E-06	0.12499E-06	0.44344E-07	0.45256E-06	
%	2.40	60.19	27.62	9.80		
OBS 19	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.82271E-07	0.14032E-05	0.16582E-05	0.10652E-05	0.42089E-05	33.76
Window 2	0.18008E-06	0.34601E-05	0.62249E-07	0.27486E-06	0.39772E-05	31.90
Window 3	0.30052E-07	0.41418E-05	0.00000E+00	0.00000E+00	0.41718E-05	33.46
Window 4	0.34287E-08	0.10630E-06	0.00000E+00	0.00000E+00	0.10973E-06	0.88
PDS Tot	0.29583E-06	0.91114E-05	0.17204E-05	0.13401E-05	0.12468E-04	
%	2.37	73.08	13.80	10.75		
OBS 20	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.32259E-07	0.54201E-06	0.61333E-06	0.10323E-06	0.12908E-05	46.94
Window 2	0.14641E-07	0.29599E-06	0.87619E-07	0.21570E-07	0.41982E-06	15.27
Window 3	0.11395E-07	0.34336E-06	0.00000E+00	0.00000E+00	0.35476E-06	12.90
Window 4	0.14688E-06	0.53761E-06	0.00000E+00	0.00000E+00	0.68448E-06	24.89
PDS Tot	0.20517E-06	0.17190E-05	0.70095E-06	0.12479E-06	0.27499E-05	
%	7.46	62.51	25.49	4.54		
OBS 21	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.77878E-08	0.10114E-06	0.15887E-06	0.67902E-07	0.33570E-06	35.43
Window 2	0.17497E-07	0.11154E-06	0.38468E-07	0.70508E-07	0.23802E-06	25.12
Window 3	0.76891E-08	0.89199E-07	0.00000E+00	0.00000E+00	0.96888E-07	10.22
Window 4	0.27799E-07	0.24919E-06	0.00000E+00	0.00000E+00	0.27699E-06	29.23
PDS Tot	0.60774E-07	0.55108E-06	0.19734E-06	0.13841E-06	0.94760E-06	
%	6.41	58.15	20.83	14.61		
OBS 22	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.34908E-07	0.51489E-06	0.80669E-06	0.13216E-06	0.14887E-05	22.87
Window 2	0.14466E-07	0.20647E-05	0.90815E-06	0.45105E-07	0.30324E-05	46.59
Window 3	0.45689E-08	0.19552E-05	0.00000E+00	0.00000E+00	0.19597E-05	30.11
Window 4	0.13385E-08	0.26343E-07	0.00000E+00	0.00000E+00	0.27682E-07	0.43
PDS Tot	0.55282E-07	0.45611E-05	0.17148E-05	0.17727E-06	0.65085E-05	
%	0.85	70.08	26.35	2.72		
OBS 23	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.11154E-05	0.51635E-06	0.10823E-05	0.24442E-06	0.29585E-05	33.75
Window 2	0.10560E-05	0.25170E-05	0.23429E-06	0.77115E-07	0.38844E-05	44.31
Window 3	0.14815E-06	0.16792E-05	0.00000E+00	0.00000E+00	0.18273E-05	20.84
Window 4	0.60064E-07	0.36527E-07	0.00000E+00	0.00000E+00	0.96591E-07	1.10
PDS Tot	0.23796E-05	0.47491E-05	0.13166E-05	0.32154E-06	0.87669E-05	
%	27.14	54.17	15.02	3.67		
OBS 24	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.46696E-07	0.18501E-06	0.23757E-06	0.86890E-07	0.55616E-06	42.08
Window 2	0.16853E-07	0.35502E-06	0.15536E-07	0.14312E-07	0.40172E-06	30.40
Window 3	0.11106E-07	0.27640E-06	0.00000E+00	0.00000E+00	0.28750E-06	21.76
Window 4	0.68140E-08	0.69350E-07	0.00000E+00	0.00000E+00	0.76164E-07	5.76
PDS Tot	0.81468E-07	0.88578E-06	0.25310E-06	0.10120E-06	0.13216E-05	
%	6.16	67.03	19.15	7.66		
OBS 25	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.14866E-07	0.57564E-07	0.91206E-07	0.88289E-08	0.17246E-06	26.46
Window 2	0.13227E-07	0.22690E-06	0.24158E-07	0.89934E-08	0.27328E-06	41.93
Window 3	0.19867E-08	0.19837E-06	0.00000E+00	0.00000E+00	0.20035E-06	30.74
Window 4	0.14140E-08	0.42417E-08	0.00000E+00	0.00000E+00	0.56557E-08	0.87
PDS Tot	0.31494E-07	0.48707E-06	0.11536E-06	0.17822E-07	0.65175E-06	
%	4.83	74.73	17.70	2.73		
OBS 26	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.16144E-08	0.14942E-07	0.96896E-07	0.21278E-08	0.11558E-06	10.62

Table A-5 (Cont.)

Window 2	0.52671E-08	0.55696E-06	0.10490E-07	0.66981E-08	0.57942E-06	53.24
Window 3	0.11524E-08	0.31763E-06	0.00000E+00	0.00000E+00	0.31878E-06	29.29
Window 4	0.72645E-08	0.67367E-07	0.00000E+00	0.00000E+00	0.74631E-07	6.86
PDS Tot	0.15298E-07	0.95690E-06	0.10739E-06	0.88259E-08	0.10884E-05	
%	1.41	87.92	9.87	0.81		
OBS 27	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.25138E-07	0.52702E-07	0.19857E-06	0.16185E-07	0.29259E-06	25.64
Window 2	0.57331E-07	0.37851E-06	0.73091E-07	0.45340E-07	0.55427E-06	48.57
Window 3	0.18226E-07	0.21518E-06	0.00000E+00	0.00000E+00	0.23341E-06	20.45
Window 4	0.30602E-07	0.30281E-07	0.00000E+00	0.00000E+00	0.60883E-07	5.34
PDS Tot	0.13130E-06	0.67668E-06	0.27166E-06	0.61526E-07	0.11412E-05	
%	11.51	59.30	23.81	5.39		
OBS 28	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.10236E-06	0.21537E-05	0.10008E-05	0.17265E-06	0.34296E-05	53.36
Window 2	0.27397E-07	0.15281E-05	0.64212E-07	0.14064E-07	0.16338F-05	25.42
Window 3	0.70735E-08	0.12454E-05	0.00000E+00	0.00000E+00	0.12525E-05	19.49
Window 4	0.47035E-07	0.64730E-07	0.00000E+00	0.00000E+00	0.11177E-06	1.74
PDS Tot	0.18387E-06	0.49920E-05	0.10650E-05	0.18672E-06	0.64276E-05	
%	2.86	77.66	16.57	2.90		
OBS 29	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.25486E-07	0.41389E-07	0.13592E-06	0.18062E-07	0.22086E-06	27.72
Window 2	0.30897E-07	0.22824E-06	0.35102E-07	0.36370E-08	0.29788E-06	37.39
Window 3	0.53659E-08	0.16423E-06	0.00000E+00	0.00000E+00	0.16959E-06	21.28
Window 4	0.20382E-07	0.88057E-07	0.00000E+00	0.00000E+00	0.10844E-06	13.61
PDS Tot	0.82131E-07	0.52192E-06	0.17102E-06	0.21699E-07	0.79677E-06	
%	10.31	65.50	21.46	2.72		
OBS 30	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.30604E-06	0.27878E-06	0.38808E-05	0.53605E-06	0.50017E-05	16.22
Window 2	0.78809E-07	0.13723E-04	0.20206E-07	0.34636E-07	0.13857E-04	44.93
Window 3	0.35369E-07	0.11800E-04	0.00000E+00	0.00000E+00	0.11836E-04	38.38
Window 4	0.27092E-07	0.12084E-06	0.00000E+00	0.00000E+00	0.14793E-06	0.48
PDS Tot	0.44731E-06	0.25923E-04	0.39010E-05	0.57068E-06	0.30842E-04	
%	1.45	84.05	12.65	1.85		
OBS 31	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.26224E-06	0.61781E-06	0.25586E-05	0.10504E-06	0.35437E-05	36.80
Window 2	0.52281E-07	0.19727E-05	0.14627E-06	0.18506E-06	0.23563E-05	24.47
Window 3	0.39184E-07	0.36200E-05	0.00000E+00	0.00000E+00	0.36592E-05	38.00
Window 4	0.51364E-08	0.65540E-07	0.00000E+00	0.00000E+00	0.70677E-07	0.73
PDS Tot	0.35884E-06	0.62761E-05	0.27049E-05	0.29011E-06	0.96299E-05	
%	3.73	65.17	28.09	3.01		
OBS 32	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.78507E-08	0.23845E-07	0.27573E-07	0.11404E-08	0.60409E-07	13.87
Window 2	0.34414E-07	0.13923E-06	0.25689E-07	0.10566E-07	0.20990E-06	48.19
Window 3	0.68439E-08	0.12210E-06	0.00000E+00	0.00000E+00	0.12894E-06	29.60
Window 4	0.47913E-08	0.31546E-07	0.00000E+00	0.00000E+00	0.36337E-07	8.34
PDS Tot	0.53900E-07	0.31672E-06	0.53263E-07	0.11706E-07	0.43558E-06	
%	12.37	72.71	12.23	2.69		
OBS 33	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.19401E-05	0.15312E-05	0.37487E-05	0.19180E-06	0.74118E-05	26.02
Window 2	0.77805E-06	0.17574E-04	0.44049E-06	0.12604E-05	0.20053E-04	70.40
Window 3	0.12626E-06	0.67061E-06	0.00000E+00	0.00000E+00	0.79687E-06	2.80
Window 4	0.56244E-07	0.16564E-06	0.00000E+00	0.00000E+00	0.22188E-06	0.78
PDS Tot	0.29006E-05	0.19941E-04	0.41892E-05	0.14522E-05	0.28483E-04	
%	10.18	70.01	14.71	5.10		
OBS 34	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.25391E-06	0.43575E-06	0.87454E-06	0.31713E-06	0.18813E-05	24.07
Window 2	0.12713E-06	0.16870E-05	0.74579E-07	0.14190E-06	0.20306E-05	25.98
Window 3	0.11929E-06	036592E-05	0.00000E+00	0.00000E+00	0.37785E-05	48.35
Window 4	0.13973E-07	0.11106E-06	0.00000E+00	0.00000E+00	0.12503E-06	1.60
PDS Tot	0.51430E-06	0.58930E-05	0.94912E-06	0.45903E-06	0.78154E-05	

Table A-5 (Cont.)

%	6.58	75.40	12.14	5.87		
OBS 35	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.12899E-07	0.16241E-06	0.12518E-06	0.19135E-07	0.31963E-06	33.89
Window 2	0.26843E-07	0.18572E-06	0.13226E-06	0.84921E-08	0.35332E-06	37.46
Window 3	0.14604E-07	0.70245E-07	0.00000E+00	0.00000E+00	0.84849E-07	9.00
Window 4	0.43283E-07	0.14210E-06	0.00000E+00	0.00000E+00	0.18539E-06	19.66
PDS Tot	0.97629E-07	0.56049E-06	0.25744E-06	0.27627E-07	0.94319E-06	
%	10.35	59.42	27.30	2.93		
OBS 36	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.93856E-06	0.36185E-06	0.16123E-05	0.38026E-06	0.32930E-05	45.76
Window 2	0.40865E-06	0.13970E-05	0.48794E-06	0.23517E-06	0.25288E-05	35.14
Window 3	0.15098E-06	0.11021E-05	0.00000E+00	0.00000E+00	0.12531E-05	17.42
Window 4	0.30663E-07	0.89998E-07	0.00000E+00	0.00000E+00	0.12066E-06	1.68
PDS Tot	0.15289E-05	0.29510E-05	0.21002E-05	0.61542E-06	0.71955E-05	
%	21.25	41.01	29.19	8.55		
OBS 37	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.11861E-06	0.65170E-06	0.31487E-06	0.72920E-07	0.11581E-05	53.64
Window 2	0.10718E-06	0.27052E-06	0.25563E-07	0.18220E-07	0.42148E-06	19.52
Window 3	0.20599E-06	0.28324E-05	0.00000E+00	0.00000E+00	0.48923E-06	22.66
Window 4	0.10203E-07	0.79904E-07	0.00000E+00	0.00000E+00	0.90107E-07	4.17
PDS Tot	0.44199E-06	0.12854E-05	0.34044E-06	0.91140E-07	0.21589E-05	
%	20.47	59.54	15.77	4.22		
OBS 38	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.74179E-07	0.14189E-06	0.42023E-06	0.49810E-08	0.64128E-06	19.64
Window 2	0.77507E-07	0.11006E-05	0.47376E-07	0.11992E-07	0.12375E-05	37.89
Window 3	0.15305E-07	0.10112E-05	0.00000E+00	0.00000E+00	0.10265E-05	31.43
Window 4	0.14399E-07	0.34603E-06	0.00000E+00	0.00000E+00	0.36043E-06	11.04
PDS Tot	0.18139E-06	0.25998E-05	0.46761E-06	0.16973E-07	0.32657E-05	
%	5.55	79.61	14.32	0.52		
OBS 39	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.21384E-07	0.28264E-07	0.25109E-06	0.13638E-07	0.31438E-06	14.08
Window 2	0.61960E-07	0.98408E-06	0.12448E-06	0.34987E-07	0.12055E-05	53.99
Window 3	0.73769E-08	0.66571E-06	0.00000E+00	0.00000E+00	0.67309E-06	30.14
Window 4	0.51139E-08	0.34946E-07	0.00000E+00	0.00000E+00	0.40060E-07	1.79
PDS Tot	0.95835E-07	0.17130E-05	0.37557E-06	0.48624E-07	0.22330E-05	
%	4.29	76.71	16.82	2.18		
OBS 40	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.32237E-08	0.18701E-07	0.59516E-07	0.14406E-07	0.95847E-07	14.72
Window 2	0.11946E-07	0.26851E-06	0.11577E-07	0.22890E-07	0.31492E-06	48.36
Window 3	0.59082E-08	0.19637E-06	0.00000E+00	0.00000E+00	0.20228E-06	31.06
Window 4	0.14975E-08	0.36617E-07	0.00000E+00	0.00000E+00	0.38114E-07	5.85
PDS Tot	0.22576E-07	0.52020E-06	0.71093E-07	0.37296E-07	0.65116E-06	
%	3.47	79.89	10.92	5.73		
OBS 41	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.58141E-07	0.61548E-07	0.12868E-06	0.24739E-07	0.27311E-06	45.35
Window 2	0.26049E-07	0.10154E-06	0.46036E-07	0.17861E-07	0.19149E-06	31.80
Window 3	0.52189E-08	0.77478E-07	0.00000E+00	0.00000E+00	0.82697E-07	13.73
Window 4	0.46007E-08	0.50274E-07	0.00000E+00	0.00000E+00	0.54875E-07	9.11
PDS Tot	0.94010E-07	0.29084E-06	0.17472E-06	0.42600E-07	0.60217E-06	
%	15.61	48.30	29.01	7.07		
OBS 42	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.15083E-07	0.95170E-07	0.15610E-06	0.12427E-07	0.27878E-06	30.25
Window 2	0.15098E-07	0.25858E-06	0.89119E-07	0.29653E-07	0.39245E-06	42.59
Window 3	0.62891E-08	0.15304E-06	0.00000E+00	0.00000E+00	0.15933E-06	17.29
Window 4	0.26731E-07	0.64229E-07	0.00000E+00	0.00000E+00	0.90961E-07	9.87
PDS Tot	0.63202E-07	0.57102E-06	0.24522E-06	0.42081E-07	0.92152E-06	
%	6.86	61.96	26.61	4.57		
OBS 43	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.41934E-08	0.33763E-07	0.34780E-06	0.16917E-07	0.40267E-06	7.22
Window 2	0.10930E-07	0.21486E-05	0.61672E-07	0.13541E-07	0.22348E-05	40.06

Table A-5 (Cont.)

Window 3	0.11913E-07	0.16576E-05	0.00000E+00	0.00000E+00	0.16696E-05	29.92
Window 4	0.41914E-07	0.12303E-05	0.00000E+00	0.00000E+00	0.12722E-05	2[illegible].80
PDS Tot	0.68951E-07	0.50703E-05	0.40947E-06	0.30459E-07	0.55792E-05	
%	1.24	90.88	7.34	0.55		
OBS 44	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.14481E-06	0.21372E-06	0.24253E-06	0.11590E-06	0.71695E-06	52.52
Window 2	0.18705E-07	0.24634E-06	0.49173E-07	0.70813E-08	0.32129E-06	23.53
Window 3	0.15829E-07	0.29390E-06	0.00000E+00	0.00000E+00	0.30972E-06	22.69
Window 4	0.36783E-08	0.13559E-07	0.00000E+00	0.00000E+00	0.17237E-07	1.26
PDS Tot	0.18302E-06	0.76751E-06	0.29170E-06	0.12298E-06	0.13652E-05	
%	13.41	56.22	21.37	9.01		
OBS 45	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.93795E-06	0.16155E-06	0.30532E-06	0.41719E-07	0.14465E-05	57.30
Window 2	0.12886E-06	0.32266E-06	0.16283E-06	0.50692E-07	0.66504E-06	26.34
Window 3	0.10038E-06	0.24228E-06	0.00000E+00	0.00000E+00	0.34266E-06	13.57
Window 4	0.12931E-07	0.57250E-07	0.00000E+00	0.00000E+00	0.70182E-07	2.78
PDS Tot	0.11801E-05	0.78374E-06	0.46814E-06	0.92411E-07	0.25244E-05	
%	46.75	31.05	18.54	3.66		
OBS 46	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.58814E-07	0.69448E-06	0.62466E-06	0.79770E-06	0.21757E-05	22.80
Window 2	0.27648E-06	0.29718E-05	0.24096E-06	0.11709E-05	0.46602E-05	48.84
Window 3	0.65370E-07	0.24768E-05	0.00000E+00	0.00000E+00	0.25422E-05	26.64
Window 4	0.30451E-07	0.13324E-06	0.00000E+00	0.00000E+00	0.16369E-06	1.72
PDS Tot	0.43112E-06	0.62764E-05	0.86563E-06	0.19686E-05	0.95417E-05	
%	4.52	65.78	9.07	20.63		
OBS 47	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.15554E-07	0.13802E-06	0.12609E-06	0.37916E-07	0.31758E-06	38.22
Window 2	0.24935E-08	0.27703E-06	0.58859E-08	0.83436E-08	0.29375E-06	35.35
Window 3	0.90438E-09	0.20914E-06	0.00000E+00	0.00000E+00	0.21005E-06	25.28
Window 4	0.50584E-09	0.90858E-08	0.00000E+00	0.00000E+00	0.95916E-08	1.15
PDS Tot	0.19458E-07	0.63328E-06	0.13198E-06	0.46260E-07	0.83097E-06	
%	2.34	76.21	15.88	5.57		
OBS 48	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.16865E-07	0.76815E-07	0.18108E-06	0.38658E-08	0.27863E-06	16.34
Window 2	0.44769E-08	0.11555E-06	0.24122E-07	0.84533E-08	0.15261E-06	8.95
Window 3	0.13447E-07	0.87637E-07	0.00000E+00	0.00000E+00	0.10108E-06	5.93
Window 4	0.18435E-06	0.98825E-06	0.00000E+00	0.00000E+00	0.11726E-05	68.78
PDS Tot	0.21914E-06	0.12683E-05	0.20520E-06	0.12319E-07	0.17049E-05	
%	12.85	74.39	12.04	0.72		
OBS 49	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.76929E-08	0.28022E-07	0.79731E-07	0.57275E-08	0.12117E-06	11.44
Window 2	0.16730E-08	0.38054E-07	0.10420E-07	0.89391E-08	0.59086E-07	5.58
Window 3	0.26159E-08	0.46905E-07	0.00000E+00	0.00000E+00	0.49521E-07	4.67
Window 4	0.84576E-07	0.74512E-06	0.00000E+00	0.00000E+00	0.82970E-06	78.31
PDS Tot	0.96558E-07	0.85810E-06	0.90151E-07	0.14667E-07	0.10595E-05	
%	9.11	80.99	8.51	1.38		
OBS 50	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.63547E-07	0.28974E-07	0.21947E-06	0.21673E-07	0.33367E-06	41.76
Window 2	0.38289E-07	0.20731E-06	0.88753E-08	0.13671E-07	0.26814E-06	33.56
Window 3	0.10276E-07	0.16490E-06	0.00000E+00	0.00000E+00	0.17518E-06	21.92
Window 4	0.38599E-08	0.18172E-07	0.00000E+00	0.00000E+00	0.22032E-07	2.76
PDS Tot	0.11597E-06	0.41935E-06	0.22835E-06	0.35344E-07	0.79902E-06	
%	14.51	52.48	28.58	4.42		
OBS 51	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.12539E-06	0.48745E-07	0.24002E-06	0.28015E-07	0.44217E-06	13.23
Window 2	0.56009E-07	0.12033E-05	0.21601E-07	0.99594E-08	0.12909E-05	38.62
Window 3	0.22133E-07	0.15024E-05	0.00000E+00	0.00000E+00	0.15246E-05	45.61
Window 4	0.46039E-08	0.80266E-07	0.00000E+00	0.00000E+00	0.84870E-07	2.54
PDS Tot	0.20813E-06	0.28348E-05	0.26162E-06	0.37974E-07	0.33425E-05	
%	6.23	84.81	7.83	1.14		

Table A-5 (Cont.)

OBS 52	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.14761E-07	0.15837E-07	0.11946E-06	0.23745E-07	0.17380E-06	26.49
Window 2	0.30749E-07	0.17133E-06	0.42234E-07	0.30507E-07	0.27482E-06	41.88
Window 3	0.96681E-08	0.11686E-06	0.00000E+00	0.00000E+00	0.12652E-06	19.28
Window 4	0.32757E-07	0.48284E-07	0.00000E+00	0.00000E+00	0.81041E-07	12.35
PDS Tot	0.87935E-07	0.35230E-06	0.16169E-06	0.54252E-07	0.65618E-06	
%	13.40	53.69	24.64	8.27		
OBS 53	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.17478E-07	0.19359E-06	0.55767E-06	0.45395E-07	0.81413E-06	21.68
Window 2	0.34511E-07	0.15485E-05	0.58945E-07	0.51193E-07	0.16931E-05	45.09
Window 3	0.83719E-08	0.10665E-05	0.00000E+00	0.00000E+00	0.10749E-05	28.63
Window 4	0.12033E-08	0.17157E-06	0.00000E+00	0.00000E+00	0.17277E-06	4.60
PDS Tot	0.61564E-07	0.29801E-05	0.61662E-06	0.96588E-07	0.37549E-05	
%	1.64	79.37	16.42	2.57		
OBS 54	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.60314E-07	0.91655E-07	0.62512E-06	0.41118E-07	0.81821E-06	20.47
Window 2	0.94891E-07	0.15896E-05	0.10250E-06	0.11519E-06	0.19021E-05	47.58
Window 3	0.32138E-07	0.12219E-05	0.00000E+00	0.00000E+00	0.12541E-05	31.37
Window 4	0.72031E-08	0.16457E-07	0.00000E+00	0.00000E+00	0.23660E-07	0.59
PDS Tot	0.19455E-06	0.29196E-05	0.72762E-06	0.15631E-06	0.39981E-05	
%	4.87	73.02	18.20	3.91		
OBS 55	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.41413E-06	0.23124E-05	0.96024E-06	0.54511E-06	0.42319E-05	73.34
Window 2	0.26056E-06	0.35535E-06	0.86437E-07	0.12894E-06	0.83129E-06	14.41
Window 3	0.16033E-06	0.27444E-06	0.00000E+00	0.00000E+00	0.43477E-06	7.53
Window 4	0.69169E-07	0.20339E-06	0.00000E+00	0.00000E+00	0.27256E-06	4.72
PDS Tot	0.90419E-06	0.31456E-05	0.10467E-05	0.67405E-06	0.57705E-05	
%	15.67	54.51	18.14	11.68		
OBS 56	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.72928E-08	0.16151E-07	0.87974E-07	0.16956E-07	0.12837E-06	15.16
Window 2	0.95997E-08	0.30526E-06	0.18945E-07	0.15689E-07	0.34949E-06	41.27
Window 3	0.29723E-08	0.28319E-06	0.00000E+00	0.00000E+00	0.28616E-06	33.79
Window 4	0.14344E-07	0.68554E-07	0.00000E+00	0.00000E+00	0.82897E-07	9.79
PDS Tot	0.34209E-07	0.67315E-06	0.10692E-06	0.32646E-07	0.84692E-06	
%	4.04	79.48	12.62	3.85		
OBS 57	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.23469E-07	0.63326E-08	0.11031E-06	0.10218E-07	0.15033E-06	46.51
Window 2	0.48986E-08	0.60541E-07	0.18432E-07	0.38819E-08	0.87753E-07	27.15
Window 3	0.15825E-08	0.50037E-07	0.00000E+00	0.00000E+00	0.51619E-07	15.97
Window 4	0.14789E-08	0.32056E-07	0.00000E+00	0.00000E+00	0.33535E-07	10.37
PDS Tot	0.31429E-07	0.14897E-06	0.12874E-06	0.14100E-07	0.32323E-06	
%	9.72	46.09	39.83	4.36		
OBS 58	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.12735E-05	0.83973E-05	0.15441E-05	0.22111E-05	0.13426E-04	67.46
Window 2	0.17221E-05	0.21541E-05	0.15878E-06	0.39329E-07	0.40743E-05	20.47
Window 3	0.84233E-07	0.20102E-05	0.00000E+00	0.00000E+00	0.20944E-05	10.52
Window 4	0.16775E-07	0.29060E-06	0.00000E+00	0.00000E+00	0.30738E-06	1.54
PDS Tot	0.30966E-05	0.12852E-04	0.17028E-05	0.22505E-05	0.19902E-04	
%	15.56	64.58	8.56	11.31		
OBS 59	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.29626E-07	0.18118E-06	0.36889E-06	0.29453E-07	0.60914E-06	51.53
Window 2	0.66405E-07	0.21055E-06	0.36993E-07	0.93745E-08	0.32332E-06	27.35
Window 3	0.31837E-08	0.22064E-06	0.00000E+00	0.00000E+00	0.22383E-06	18.94
Window 4	0.48332E-08	0.20947E-07	0.00000E+00	0.00000E+00	0.25780E-07	2.18
PDS Tot	0.10405E-06	0.63332E-06	0.40588E-06	0.38827E-07	0.11821E-05	
%	8.80	53.58	34.34	3.28		
OBS 60	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.91393E-07	0.37484E-06	0.13631E-06	0.16246E-07	0.61879E-06	44.55
Window 2	0.38049E-07	0.27913E-06	0.36746E-07	0.11688E-07	0.36561E-06	26.32
Window 3	0.23407E-07	0.20379E-06	0.00000E+00	0.00000E+00	0.22719E-06	16.36

Table A-5 (Cont.)

Window 4	0.54646E-08	0.17195E-06	0.00000E+00	0.00000E+00	0.17742E-06	12.77
PDS Tot	0.15831E-06	0.10297E-05	0.17305E-06	0.27935E-07	0.13890E-05	
%	11.40	74.13	12.46	2.01		
OBS 61	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.25778E-07	0.15020E-06	0.53609E-07	0.57093E-07	0.28668E-06	20.77
Window 2	0.54616E-07	0.44837E-06	0.43292E-07	0.11423E-06	0.66051E-06	47.85
Window 3	0.14462E-07	0.35532E-06	0.00000E+00	0.00000E+00	0.36979E-06	26.79
Window 4	0.96452E-08	0.53835E-07	0.00000E+00	0.00000E+00	0.63480E-07	4.60
PDS Tot	0.10450E-06	0.10077E-05	0.96900E-07	0.17132E-06	0.13804E-05	
%	7.57	73.00	7.02	12.41		
OBS 62	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.34340E-08	0.49196E-07	0.91266E-07	0.15536E-07	0.15943E-06	20.80
Window 2	0.17813E-08	0.23976E-06	0.73553E-08	0.91610E-08	0.25806E-06	33.67
Window 3	0.10466E-08	0.27030E-06	0.00000E+00	0.00000E+00	0.27134E-06	35.41
Window 4	0.41908E-08	0.73316E-07	0.00000E+00	0.00000E+00	0.77507E-07	10.11
PDS Tot	0.10453E-07	0.63257E-06	0.98622E-07	0.24697E-07	0.76634E-06	
%	1.36	82.54	12.87	3.22		
OBS 63	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.13869E-06	0.64465E-06	0.47450E-06	0.23176E-06	0.14896E-05	34.14
Window 2	0.60832E-06	0.11052E-05	0.81815E-07	0.20980E-07	0.18163E-05	41.63
Window 3	0.10900E-06	0.92734E-06	0.00000E+00	0.00000E+00	0.10363E-05	23.75
Window 4	0.10476E-07	0.10479E-07	0.00000E+00	0.00000E+00	0.20955E-07	0.48
PDS Tot	0.86648E-06	0.26876E-05	0.55632E-06	0.25274E-06	0.43632E-05	
%	19.86	61.60	12.75	5.79		
OBS 64	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.30405E-07	0.53977E-07	0.50448E-07	0.10630E-07	0.14546E-06	18.10
Window 2	0.32524E-07	0.24804E-06	0.43441E-07	0.42697E-09	0.32443E-06	40.36
Window 3	0.12239E-07	0.20909E-06	0.00000E+00	0.00000E+00	0.22133E-06	27.53
Window 4	0.30397E-07	0.82221E-07	0.00000E+00	0.00000E+00	0.11262E-06	14.01
PDS Tot	0.10557E-06	0.59333E-06	0.93889E-07	0.11057E-07	0.80384E-06	
%	13.13	73.81	11.68	1.38		
OBS 65	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.55681E-08	0.94198E-07	0.28449E-06	0.17233E-07	0.40149E-06	19.71
Window 2	0.90457E-08	0.77170E-06	0.85981E-08	0.54350E-07	0.84370E-06	41.42
Window 3	0.24745E-08	0.78120E-06	0.00000E+00	0.00000E+00	0.78368E-06	38.47
Window 4	0.15251E-08	0.67634E-08	0.00000E+00	0.00000E+00	0.82885E-08	0.41
PDS Tot	0.18613E-07	0.16539E-05	0.29309E-06	0.71583E-07	0.20372E-05	
%	0.91	81.19	14.39	3.51		
OBS 66	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.10183E-07	0.16570E-07	0.14319E-06	0.12818E-07	0.18276E-06	28.80
Window 2	0.13742E-07	0.19404E-06	0.12826E-07	0.67652E-08	0.22737E-06	35.83
Window 3	0.90819E-08	0.13573E-06	0.00000E+00	0.00000E+00	0.14481E-06	22.82
Window 4	0.51716E-08	0.74533E-07	0.00000E+00	0.00000E+00	0.79704E-07	12.56
PDS Tot	0.38179E-07	0.42087E-06	0.15601E-06	0.19583E-07	0.63465E-06	
%	6.02	66.32	24.58	3.09		
OBS 67	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.62776E-08	0.30465E-07	0.13110E-07	0.13356E-08	0.51189E-07	20.69
Window 2	0.30364E-08	0.98555E-07	0.80035E-08	0.13503E-08	0.11095E-06	44.83
Window 3	0.30182E-09	0.69959E-07	0.00000E+00	0.00000E+00	0.70261E-07	28.39
Window 4	0.29393E-08	0.12121E-07	0.00000E+00	0.00000E+00	0.15061E-07	6.09
PDS Tot	0.12555E-07	0.21110E-06	0.21114E-07	0.26858E-08	0.24746E-06	
%	5.07	85.31	8.53	1.09		
OBS 68	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.52499E-07	0.13354E-06	0.42533E-06	0.33938E-07	0.64530E-06	40.34
Window 2	0.13503E-06	0.17300E-06	0.37197E-07	0.11866E-07	0.35709E-06	22.32
Window 3	0.11327E-07	0.37293E-06	0.00000E+00	0.00000E+00	0.38426E-06	24.02
Window 4	0.93679E-07	0.11930E-06	0.00000E+00	0.00000E+00	0.21298E-06	13.31
PDS Tot	0.29253E-06	0.79877E-06	0.46252E-06	0.45804E-07	0.15996E-05	
%	18.29	49.93	28.91	2.86		
OBS 69	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%

Table A-5 (Cont.)

Window 1	0.72885E-07	0.14164E-06	0.75266E-06	0.55938E-07	0.10231E-05	48.95
Window 2	0.50696E-07	0.47515E-06	0.15503E-07	0.91195E-08	0.55047E-06	26.34
Window 3	0.21768E-07	0.45087E-06	0.00000E+00	0.00000E+00	0.47264E-06	22.61
Window 4	0.61882E-08	0.37683E-07	0.00000E+00	0.00000E+00	0.43871E-07	2.10
PDS Tot	0.15154E-06	0.11053E-05	0.76817E-06	0.65057E-07	0.20901E-05	
%	7.25	52.88	36.75	3.11		
OBS 70	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.11033E-07	0.11215E-07	0.22889E-06	0.86305E-09	0.25200E-06	18.87
Window 2	0.10109E-07	0.39810E-06	0.40611E-07	0.12634E-08	0.45009E-06	33.70
Window 3	0.60063E-08	0.24491E-06	0.00000E+00	0.00000E+00	0.25092E-06	18.79
Window 4	0.24814E-07	0.35791E-06	0.00000E+00	0.00000E+00	0.38272E-06	28.65
PDS Tot	0.51963E-07	0.10121E-05	0.26950E-06	0.21264E-08	0.13357E-05	
%	3.89	75.77	20.18	0.16		
OBS 71	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.22206E-06	0.74781E-07	0.70380E-06	0.73215E-07	0.10739E-05	35.71
Window 2	0.15917E-06	0.40150E-06	0.25370E-06	0.12002E-06	0.93440E-06	31.08
Window 3	0.18641E-06	0.54071E-06	0.00000E+00	0.00000E+00	0.72712E-06	24.18
Window 4	0.12955E-07	0.25851E-06	0.00000E+00	0.00000E+00	0.27147E-06	9.03
PDS Tot	0.58059E-06	0.12755E-05	0.95751E-06	0.19323E-06	0.30068E-05	
%	19.31	42.42	31.84	6.43		
OBS 72	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.15533E-06	0.35136E-05	0.59382E-06	0.65710E-07	0.43285E-05	64.33
Window 2	0.18610E-06	0.12715E-05	0.23267E-07	0.19074E-07	0.14999E-05	22.29
Window 3	0.37826E-07	0.83411E-06	0.00000E+00	0.00000E+00	0.87194E-06	12.96
Window 4	0.11300E-07	0.17070E-07	0.00000E+00	0.00000E+00	0.28370E-07	0.42
PDS Tot	0.39055E-06	0.56363E-05	0.61708E-06	0.84783E-07	0.67287E-05	
%	5.80	83.76	9.17	1.26		
OBS 73	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.62543E-06	0.14577E-05	0.29281E-05	0.13862E-06	0.51498E-05	38.63
Window 2	0.34170E-06	0.10427E-05	0.81971E-07	0.78150E-07	0.15446E-05	11.59
Window 3	0.18949E-06	0.91578E-06	0.00000E+00	0.00000E+00	0.11053E-05	8.29
Window 4	0.40077E-07	0.54911E-05	0.00000E+00	0.00000E+00	0.55312E-05	41.49
PDS Tot	0.11967E-05	0.89073E-05	0.30101E-05	0.21677E-06	0.13331E-04	
%	8.98	66.82	22.58	1.63		
OBS 74	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.51235E-07	0.90600E-07	0.25250E-07	0.33269E-08	0.17041E-06	57.96
Window 2	0.10931E-07	0.51893E-07	0.14477E-07	0.21960E-08	0.79497E-07	27.04
Window 3	0.23571E-08	0.34379E-07	0.00000E+00	0.00000E+00	0.36736E-07	12.49
Window 4	0.69209E-09	0.66797E-08	0.00000E+00	0.00000E+00	0.73718E-08	2.51
PDS Tot	0.65215E-07	0.18355E-06	0.39728E-07	0.55228E-08	0.29402E-06	
%	22.18	62.43	13.51	1.88		
OBS 75	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.64968E-07	0.57830E-06	0.12629E-05	0.10575E-06	0.20119E-05	18.47
Window 2	0.26932E-07	0.48284E-05	0.23726E-06	0.16900E-06	0.52616E-05	48.31
Window 3	0.25384E-07	0.35604E-05	0.00000E+00	0.00000E+00	0.35858E-05	32.92
Window 4	0.49639E-08	0.27937E-07	0.00000E+00	0.00000E+00	0.32901E-07	0.30
PDS Tot	0.12225E-06	0.89951E-05	0.15002E-05	0.27475E-06	0.10892E-04	
%	1.12	82.58	13.77	2.52		
OBS 76	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.84614E-07	0.51575E-07	0.62421E-06	0.77025E-07	0.83742E-06	22.27
Window 2	0.60877E-07	0.10540E-05	0.14611E-06	0.35698E-06	0.16179E-05	43.02
Window 3	0.26175E-07	0.80670E-06	0.00000E+00	0.00000E+00	0.83288E-06	22.15
Window 4	0.36514E-08	0.46877E-06	0.00000E+00	0.00000E+00	0.47243E-06	12.56
PDS Tot	0.17532E-06	0.23810E-05	0.77032E-06	0.43400E-06	0.37606E-05	
%	4.66	63.31	20.48	11.54		
OBS 77	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.15036E-06	0.26844E-06	0.33068E-06	0.69341E-07	0.81882E-06	53.85
Window 2	0.17404E-06	0.16091E-06	0.14094E-06	0.31758E-07	0.50764E-06	33.39
Window 3	0.40077E-07	0.12088E-06	0.00000E+00	0.00000E+00	0.16095E-06	10.59
Window 4	0.13014E-07	0.20136E-07	0.00000E+00	0.00000E+00	0.33149E-07	2.18

Table A-5 (Cont.)

PDS Tot	0.37749E-06	0.57036E-06	0.47161E-06	0.10110E-06	0.15206E-05	
%	24.83	37.51	31.02	6.65		
OBS 78	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.90264E-07	0.16212E-06	0.11842E-05	0.30755E-07	0.14673E-05	9.81
Window 2	0.61020E-07	0.73810E-05	0.11686E-06	0.21408E-07	0.75803E-05	50.70
Window 3	0.26578E-07	0.58066E-05	0.00000E+00	0.00000E+00	0.58332E-05	39.01
Window 4	0.44657E-08	0.67027E-07	0.00000E+00	0.00000E+00	0.71492E-07	0.48
PDS Tot	0.18233E-06	0.13417E-04	0.13010E-05	0.52163E-07	0.14952E-04	
%	1.22	89.73	8.70	0.35		
OBS 79	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.82739E-07	0.55237E-06	0.21718E-05	0.43239E-06	0.32393E-05	16.14
Window 2	0.21596E-07	0.90277E-05	0.45098E-07	0.34314E-07	0.91287E-05	45.49
Window 3	0.13704E-07	0.76259E-05	0.00000E+00	0.00000E+00	0.76396E-05	38.07
Window 4	0.11180E-08	0.57830E-07	0.00000E+00	0.00000E+00	0.58949E-07	0.29
PDS Tot	0.11916E-06	0.17264E-04	0.22169E-05	0.46671E-06	0.20067E-04	
%	0.59	86.03	11.05	2.33		
OBS 80	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.69809E-07	0.67363E-07	0.70820E-06	0.18601E-06	0.10314E-05	29.67
Window 2	0.64455E-07	0.91717E-06	0.11100E-06	0.47643E-07	0.11403E-05	32.80
Window 3	0.15446E-07	0.12421E-05	0.00000E+00	0.00000E+00	0.12576E-05	36.18
Window 4	0.32608E-08	0.43407E-07	0.00000E+00	0.00000E+00	0.46668E-07	1.34
PDS Tot	0.15297E-06	0.22701E-05	0.81920E-06	0.23365E-06	0.34759E-05	
%	4.40	65.31	23.57	6.72		
OBS 81	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.43633E-07	0.36242E-07	0.43530E-06	0.77243E-08	0.52290E-06	50.24
Window 2	0.46900E-07	0.23190E-06	0.15489E-07	0.24745E-07	0.31903E-06	30.66
Window 3	0.14107E-07	0.16257E-06	0.00000E+00	0.00000E+00	0.17667E-06	16.98
Window 4	0.68063E-08	0.15299E-07	0.00000E+00	0.00000E+00	0.22105E-07	2.12
PDS Tot	0.11145E-06	0.44600E-06	0.45079E-06	0.32469E-07	0.10407E-05	
%	10.71	42.86	43.32	3.12		
OBS 82	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.54438E-06	0.28379E-06	0.18070E-05	0.21789E-06	0.28530E-05	39.39
Window 2	0.22426E-06	0.23692E-05	0.84734E-07	0.45994E-07	0.27241E-05	37.61
Window 3	0.20015E-06	0.13217E-05	0.00000E+00	0.00000E+00	0.15219E-05	21.01
Window 4	0.24860E-07	0.11889E-06	0.00000E+00	0.00000E+00	0.14375E-06	1.98
PDS Tot	0.99365E-06	0.40936E-05	0.18917E-05	0.26388E-06	0.72428E-05	
%	13.72	56.52	26.12	3.64		
OBS 83	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.80377E-08	0.78094E-06	0.30604E-07	0.56060E-08	0.82519E-06	29.69
Window 2	0.80940E-08	0.10496E-05	0.34420E-07	0.98912E-07	0.11910E-05	42.85
Window 3	0.18458E-08	0.74273E-06	0.00000E+00	0.00000E+00	0.74457E-06	26.79
Window 4	0.14334E-08	0.17290E-07	0.00000E+00	0.00000E+00	0.18723E-07	0.67
PDS Tot	0.19411E-07	0.25906E-05	0.65024E-07	0.10452E-06	0.27795E-05	
%	0.70	93.20	2.34	3.76		
OBS 84	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.75585E-08	0.49385E-06	0.12416E-06	0.65013E-07	0.69057E-06	36.71
Window 2	0.28588E-07	0.51749E-06	0.33993E-07	0.69116E-07	0.64919E-06	34.51
Window 3	0.10019E-07	0.52259E-06	0.00000E+00	0.00000E+00	0.53261E-06	28.31
Window 4	0.15288E-08	0.74786E-08	0.00000E+00	0.00000E+00	0.90074E-08	0.48
PDS Tot	0.47693E-07	0.15414E-05	0.15815E-06	0.13413E-06	0.18814E-05	
%	2.54	81.93	8.41	7.13		
OBS 85	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.22045E-07	0.70328E-07	0.17291E-06	0.43878E-07	0.30916E-06	26.76
Window 2	0.28284E-07	0.49845E-06	0.10147E-07	0.30852E-08	0.53997E-06	46.73
Window 3	0.80781E-08	0.25912E-06	0.00000E+00	0.00000E+00	0.26720E-06	23.13
Window 4	0.10066E-07	0.29011E-07	0.00000E+00	0.00000E+00	0.39077E-07	3.38
PDS Tot	0.68473E-07	0.85691E-06	0.18306E-06	0.46963E-07	0.11554E-05	
%	5.93	74.17	15.84	4.06		
OBS 86	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.18292E-06	0.21635E-06	0.59112E-06	0.72247E-07	0.10626E-05	48.44

Table A-5 (Cont.)

Window 2	0.53354E-07	0.33909E-06	0.20017E-06	0.41007E-07	0.63362E-06	28.88
Window 3	0.76944E-07	0.24734E-06	0.00000E+00	0.00000E+00	0.32428E-06	14.78
Window 4	0.37848E-07	0.13528E-06	0.00000E+00	0.00000E+00	0.17313E-06	7.89
PDS Tot	0.35106E-06	0.93805E-06	0.79129E-06	0.11325E-06	0.21937E-05	
%	16.00	42.76	36.07	5.16		
OBS 87	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.79965E-08	0.23201E-05	0.84755E-06	0.22672E-07	0.31983E-05	87.00
Window 2	0.93829E-08	0.13161E-06	0.27806E-07	0.32056E-07	0.20085E-06	5.46
Window 3	0.34871E-08	0.73856E-07	0.00000E+00	0.00000E+00	0.77343E-07	2.10
Window 4	0.22886E-07	0.17697E-06	0.00000E+00	0.00000E+00	0.19986E-06	5.44
PDS Tot	0.43753E-07	0.27025E-05	0.87536E-06	0.54728E-07	0.36764E-05	
%	1.19	73.51	23.81	1.49		
OBS 88	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.90728E-08	0.14437E-07	0.10373E-06	0.25153E-07	0.15239E-06	20.16
Window 2	0.24435E-07	0.27771E-06	0.30815E-07	0.26642E-08	0.33562E-06	44.41
Window 3	0.79787E-08	0.19206E-06	0.00000E+00	0.00000E+00	0.2004E-06	26.47
Window 4	0.14371E-07	0.53342E-07	0.00000E+00	0.00000E+00	0.67713E-07	8.96
PDS Tot	0.55857E-07	0.53754E-06	0.13454E-06	0.27817E-07	0.75576E-06	
%	7.39	71.13	17.80	3.68		
OBS 89	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.60471E-06	0.26523E-05	0.14790E-06	0.52810E-07	0.34578E-05	69.52
Window 2	0.16228E-06	0.39619E-06	0.76690E-07	0.42465E-06	0.10598E-05	21.31
Window 3	0.37468E-07	0.27508E-06	0.00000E+00	0.00000E+00	0.31255E-06	6.28
Window 4	0.10836E-07	0.13291E-06	0.00000E+00	0.00000E+00	0.14375E-06	2.89
PDS Tot	0.81529E-06	0.34565E-05	0.22459E-06	0.47746E-06	0.49739E-05	
%	16.39	69.49	4.52	9.60		
OBS 90	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.52302E-06	0.10148E-05	0.14635E-05	0.16188E-06	0.31631E-05	32.68
Window 2	0.78096E-06	0.27277E-05	0.40089E-07	0.82071E-07	0.36308E-05	37.52
Window 3	0.32763E-06	0.24902E-05	0.00000E+00	0.00000E+00	0.28179E-05	29.12
Window 4	0.17598E-07	0.48744E-07	0.00000E+00	0.00000E+00	0.66342E-07	0.69
PDS Tot	0.16492E-05	0.62815E-05	0.15036E-05	0.24395E-06	0.96782E-05	
%	17.04	64.90	15.54	2.52		
OBS 91	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.36988E-07	0.11059E-06	0.17646E-05	0.93936E-07	0.20061E-05	54.42
Window 2	0.15049E-06	0.75491E-06	0.13649E-06	0.49630E-07	0.10915E-05	29.61
Window 3	0.27207E-07	0.45563E-06	0.00000E+00	0.00000E+00	0.48284E-06	13.10
Window 4	0.10390E-07	0.95197E-07	0.00000E+00	0.00000E+00	0.10559E-06	2.86
PDS Tot	0.22507E-06	0.14163E-05	0.19011E-05	0.14357E-06	0.36861E-05	
%	6.11	38.42	51.58	3.89		
OBS 92	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.40497E-07	0.70165E-06	0.10256E-05	0.28733E-07	0.17965E-05	63.41
Window 2	0.20574E-07	0.23511E-06	0.33443E-06	0.63874E-07	0.65399E-06	23.08
Window 3	0.16998E-07	0.33085E-06	0.00000E+00	0.00000E+00	0.34785E-06	12.28
Window 4	0.31661E-08	0.31796E-07	0.00000E+00	0.00000E+00	0.34962E-07	1.23
PDS Tot	0.81235E-07	0.12994E-05	0.13600E-05	0.92607E-07	0.28333E-05	
%	2.87	45.86	48.00	3.27		
OBS 93	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.16905E-06	0.12956E-06	0.52774E-06	0.38803E-07	0.86515E-06	60.49
Window 2	0.57265E-07	0.15367E-06	0.28422E-07	0.12486E-07	0.25184E-06	17.61
Window 3	0.20805E-07	0.16568E-06	0.00000E+00	0.00000E+00	0.18649E-06	13.04
Window 4	0.17991E-07	0.10873E-06	0.00000E+00	0.00000E+00	0.12673E-06	8.86
PDS Tot	0.26511E-06	0.55765E-06	0.55616E-06	0.51289E-07	0.14302E-05	
%	18.54	38.99	38.89	3.59		
OBS 94	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.12508E-06	0.37938E-06	0.41910E-06	0.31583E-07	0.95514E-06	46.32
Window 2	0.41346E-07	0.40101E-06	0.46777E-07	0.20724E-06	0.69637E-06	33.77
Window 3	0.10495E-07	0.27475E-06	0.00000E+00	0.00000E+00	0.28524E-06	13.83
Window 4	0.32798E-08	0.12221E-06	0.00000E+00	0.00000E+00	0.12549E-06	6.09
PDS Tot	0.18020E-06	0.11773E-05	0.46587E-06	0.23882E-06	0.20622E-05	

Table A-5 (Cont.)

%	8.74	57.09	22.59	11.58		
OBS 95	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.33419E-07	0.11146E-06	0.28315E-06	0.22544E-07	0.45057E-06	23.34
Window 2	0.11732E-07	0.74892E-06	0.34095E-07	0.17881E-07	0.81263E-06	42.10
Window 3	0.61220E-08	0.60094E-06	0.00000E+00	0.00000E+00	0.60706E-06	31.45
Window 4	0.85457E-09	0.59202E-07	0.00000E+00	0.00000E+00	0.60056E-07	3.11
PDS Tot	0.52127E-07	0.15205E-05	0.31724E-06	0.40425E-07	0.19303E-05	
%	2.70	78.77	16.43	2.09		
OBS 96	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.91322E-06	0.12728E-05	0.33461E-05	0.45621E-07	0.55778E-05	74.10
Window 2	0.61304E-07	0.52757E-06	0.39511E-06	0.50319E-07	0.10343E-05	13.74
Window 3	0.20782E-06	0.46883E-06	0.00000E+00	0.00000E+00	0.67665E-06	8.99
Window 4	0.12334E-07	0.22637E-06	0.00000E+00	0.00000E+00	0.23871E-06	3.17
PDS Tot	0.11947E-05	0.24956E-05	0.37412E-05	0.95940E-07	0.75274E-05	
%	15.87	33.15	49.70	1.27		
OBS 97	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.33112E-08	0.26705E-08	0.92293E-08	0.21753E-08	0.17386E-07	7.10
Window 2	0.14972E-07	0.36432E-07	0.28933E-08	0.74878E-08	0.61785E-07	25.24
Window 3	0.28184E-08	0.22234E-07	0.00000E+00	0.00000E+00	0.25053E-07	10.24
Window 4	0.57910E-07	0.82642E-07	0.00000E+00	0.00000E+00	0.14055E-06	57.42
PDS Tot	0.79012E-07	0.14398E-06	0.12123E-07	0.96631E-08	0.24478E-06	
%	32.28	58.82	4.95	3.95		
OBS 98	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.13051E-06	0.18870E-06	0.47275E-06	0.35666E-06	0.11486E-05	7.93
Window 2	0.12056E-05	0.61325E-05	0.37643E-06	0.66732E-06	0.83818E-05	57.89
Window 3	0.20899E-06	0.46179E-05	0.00000E+00	0.00000E+00	0.48269E-05	33.34
Windo 4	0.88871E-07	0.33032E-07	0.00000E+00	0.00000E+00	0.12190E-06	0.84
PDS Tot	0.16340E-05	0.10972E-04	0.84918E-06	0.10240E-05	0.14479E-04	
%	11.28	75.78	5.86	7.07		
OBS 99	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.11782E-08	0.51819E-07	0.14185E-06	0.32448E-07	0.22730E-06	17.12
Window 2	0.24906E-08	0.15408E-06	0.40274E-07	0.21677E-07	0.21852E-06	16.46
Window 3	0.91693E-08	0.11616E-06	0.00000E+00	0.00000E+00	0.12533E-06	9.44
Window 4	0.57074E-07	0.69969E-06	0.00000E+00	0.00000E+00	0.75676E-06	56.99
PDS Tot	0.69912E-07	0.10217E-05	0.18213E-06	0.54125E-07	0.13279E-05	
%	5.26	76.94	13.72	4.08		
OBS 100	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.32671E-07	0.53852E-07	0.16999E-06	0.18437E-07	0.27495E-06	34.37
Window 2	0.16401E-07	0.21093E-06	0.57429E-07	0.79370E-08	0.29269E-06	36.59
Window 3	0.72367E-08	0.13707E-06	0.00000E+00	0.00000E+00	0.14431E-06	18.04
Window 4	0.12722E-07	0.75196E-07	0.00000E+00	0.00000E+00	0.87918E-07	10.99
PDS Tot	0.69032E-07	0.47705E-06	0.22742E-06	0.26374E-07	0.79987E-06	
%	8.63	59.64	28.43	3.30		
Mean	PDS Group 1	PDS Group 2	PDS Group 3	PDS Group 4	Window Total	%
Window 1	0.19791E-06	0.49014E-06	0.66870E-06	0.13326E-06	0.14900E-05	35.31
Window 2	0.13327E-06	0.12362E-05	0.10835E-06	0.88147E-07	0.15660E-05	37.11
Window 3	0.40498E-07	0.91445E-06	0.00000E+00	0.00000E+00	0.95495E-06	22.63
Window 4	0.23813E-07	0.18529E-06	0.00000E+00	0.00000E+00	0.20911E-06	4.96
PDS Tot	0.39548E-06	0.28261E-05	0.77705E-06	0.22140E-06	0.42201E-05	
%	9.37	66.97	18.41	5.25		

APPENDIX B

SUPPORTING INFORMATION FOR THE ACCIDENT PROGRESSION ANALYSIS

APPENDIX B

INTRODUCTION TO APPENDIX B

Appendix B contains a detailed description and listing of the Accident Progression Event Tree (APET) and the binner that groups the outcomes of evaluating the APET.

A brief description of the Surry Low Power APET is given in Section 6.1, and the binner is treated in Section 6.3. The material in these sections is not repeated here. The 40 questions in the APET are listed concisely in Table 6.1. This appendix consists of four subsections. Subsection B.1 contains a discussion of each question in the APET. The event tree itself is too large to be depicted graphically and exists only in computer input format, which appears in subsection B.2. Subsection B.3 is a detailed discussion of the binner, and subsection B.4 contains a listing of the binner.

B.1 Description of the Accident Progression Event Tree

Question 1. Time Window
4 Branches, Type 1, 1 Case

The Branches for this question are:

1. Win-1 The core damage accident was initiated at the time window 1.

2. Win-2 The core damage accident was initiated at the time window 2.

3. Win-3 The core damage accident was initiated at the time window 3.

4. Win-4 The core damage accident was initiated at the time window 4.

The branch taken in this question depends solely upon the first PDS characteristic. For each PDS group, fractions of frequencies of PDSs belonging to each time window are assigned into each time window.

Question 2. Size of the RCS Break when the Core Uncovers?
2 Branches, Type 2

The branches for this question are:

1. Large Sufficient break area is available to maintain the RCS pressure below 500 psia.

2. Small The break area is too small to maintain the RCS pressure below 500 psia.

The branch taken in this question depends on the time window of the first question and fourth PDS characteristic. The level 1 analysis on success criteria indicates that the RCS pressure will remain below 500 psi for all time windows except Time Window 1 if at least one PORV is available. For Time Window 1, the RCS pressure will continue to rise if only one PORV is open until it reaches 650 psi, the rupture pressure of the RHR vent valve.

Case 1: For Time Window 1, only one PORV is available. The RCS pressure may not reach the rupture pressure of RHR vent valve before vessel failure. This branch probability was determined internally by BNL staff.

Branch 1: 0.95
Branch 2: 0.05

Case 2: For all other time windows, 1 PORV is always open and the pressure does not rise above 500 psi.

Branch 1: 1.00
Branch 2: 0.00

Question 3. Is the RCS depressurized before Breach by Opening the Pressurizer PORVs?
2 Branches, Type 2, 2 Cases

The branches for this question are:

1. VDep The operators open the pressurizer PORVs and depressurize the RCS successfully before vessel breach.

2. noVDep The operators either do not open the pressurizer PORVs or they open the pressurizer PORVs so late that there is not enough time to depressurize the RCS before vessel breach.

This question was quantified internally. The branch taken at this question depends upon the branch previously taken at Questions 1 and 2.

The pressure in the RCS may be reduced directly if the operators open the PORVs on the pressurizer in Time Window 1, and if there is sufficient time to blow down the RCS through the PORVs before core melt. As opening the PORVs is a last resort action, it is not clear that the operators will reach this step before core melt is well advanced, and, even if they do reach this step and open the PORVs, it is not clear that depressurization of the RCS will have been accomplished before vessel breach. Further, operator depressurization is not possible here if the operators have already failed to open the PORVs.

Case 1: At Time Window 1, only one PORV was open at core damage.

Branch 1: VDep - 0.80
Branch 2: noVDep - 0.20

Case 2: Other time windows or more than PORV is already available.
Branch 1: VDep - 1.0
Branch 2: noVDep - 0.0

Question 4. Status of AC Power?
3 Branches, Type 1

The branches for this question are:

1. AC AC electrical power is available from offsite or from the DGs throughout the accident.

2. No-AC AC electrical power is not available, but may be recovered.

3. No4KV AC electrical power is not available to the injection pumps because of loss of a 4 kv bus.

The branch taken depends upon the second PDS characteristic.

Loss of offsite power and failure of the diesel generators to start (SBO) leads to the second branch since offsite power may always be restored. AC power available when the ECCS and sprays are failed means that an ignition source is likely to be present in the containment when a significant amount of hydrogen has accumulated after VB.

Question 5. Is the core damage accident due to human errors?
3 Branches, Type 2

The branches for this question are:

1. No-HX Accident is not due to human errors.

2. HXA Operators fail to diagnose loss of core cooling. AC electrical power is available throughout.

3. HXD Operators fail to take correct actions after loss of RHR cooling. AC electrical power is available throughout.

The branch taken depends upon the third PDS characteristic.

For internal initiators, accidents begin with loss of RHR cooling during shutdown. Operators either fail to recognize the accident, make a wrong diagnosis, or take a wrong action. Consequently, the core cooling is not restored and the accident progresses to core damage. AC power is available throughout the accident; recovery of core cooling and termination of the accident depends solely on the recovery from the human error.

Question 6. Status of ECCS?
5 Branches, Type 1

The branches for this question are:

1. ECCSf4KV The ECCS are available and can operate when the 4 kv bus to the injection pumps is restored.

2. ECCSfAC The ECCS are available and can operate when electric power is restored.

3. ECCSfHX The ECCS are available and can operate when human erros are corrected.

4. ECCSfHW The ECCS is failed, and is not recoverable.

5. ECCSfREC The ECCS have worked in the injection mode from RWST, but is failed in the recirculation mode.

The branch taken depends upon the fifth PDS characteristic.

The first branch is chosen in situations where the ECCS are available, but not operating because of the loss of the 4 kv bus; if or when the 4 kv bus is restored, the ECCS will function. The second branch is chosen in blackout situations with no ECCS failures; if or when power is recovered, the ECCS will function. The third branch is chosen when core damage occurs because of human errors but ECCS is available; if or when the human error is corrected, the ECCS will function. The fourth branch is selected when the failures are in the ECCS themselves, and there is no recovery within the time frame of this analysis. Since the period in which the ECCS operate in the injection mode occurs before the uncovering of the core, this branch is chosen for those PDS's in which the ECCS never operate. For those PDS's in which the ECCS operate in the injection mode and fail in the recirculation mode, the fifth branch is chosen.

Question 7. Status of Sprays?
6 Branches, Type 2, 2 Cases.

The branches for this question are:

1. SP The containment sprays are operating or are operable in the recirculation mode prior to vessel failure.

2. SPfAC The containment sprays are available and can operate when electric power is restored.

3. SPfHX The containment sprays are available and can operate when human error is corrected.

4. SPfHW The sprays themselves are not operable, and not recoverable.

5. SPfREC The containment sprays are failed in the recirculation mode and are not recoverable.

6. SPf4KV The containment sprays are available and can operate when the 4 kv bus is restored.

The branch taken depends upon the sixth PDS characteristic, and upon the branches taken at Question 1.

This question concerns the sprays during the period of core degradation, and has impact on the source term calculations. The second and sixth branches are chosen in situations where the sprays are available, but not operating because of SBO and the loss of the 4 kv bus respectively; if or when power is recovered or the 4 kv bus is restored, the spray will function. The third branch is chosen when core damage occurs because of human errors but spray is available; if or when the human error is corrected, the spray will function. The fourth branch is selected when the failures are in the sprays themselves, and there is no recovery within the time frame of this analysis. For those PDS's in which the ECCS operate in the injection mode and fail in the recirculation mode, the fifth branch is chosen.

Question 8. RWST Injected into Containment?
4 Branches, Type 2, 2 Cases.

The branches for this question are:

1. RWST-In The contents of the refueling water storage tank have been injected into the containment.

2. RWSTfAC The contents of the RWST have not been injected into the containment, but can be injected if AC power is recovered.

3. RWSTfHX The contents of the RWST have not been injected into the containment, but can be injected if human error is corrected.

4. RWSTfIn The contents of the RWST have not been injected into the containment, and cannot be injected even if power is recovered or human error is corrected.

The branch taken depends upon the seventh PDS characteristic and upon the branch taken at Questions 6 and 7.

The branch taken in this question is used to determine whether the reactor cavity is full of water.

Question 9. Initial Containment Isolation Failure?

2 Branches, Type 1

The branches for this questions are:

1. CLsdCI Prior to the accident, the containment is isolated.

2. noCLsdCI At the time of accident initiation, the containment is open.

This question addresses whether the containment is closed at the time of accident initiation. This question was included in this APET because the Surry personnel indicated in a discussion that they may consider to close containment before entering the Mid-loop operation. However, since subsequent discussions did not provide any further information on this subject, the second branch is taken for all PDS's in this phase of analysis.

Question 10. Is containment closed before core damage?
2 Branches, Type 2, 6 Cases.

The branches for this questions are:

1. CLsdCD The containment is successfully isolated before core damage.

2. noCLsdCD The containment is not isolated before core damage.

This question addresses whether the containment is successfully closed before core damage; it does not determine whether the containment leaks even if it is closed. The branch taken in this question depends on the first, fifth and ninth questions. The split fractions in this questions were sampled and their distributions are determined internally by BNL staff.

Case 1: The containment is closed at the initiation of accident. The split fraction of the first branch is 1.0.

Case 2: The core damage accident occurs because operators fail to recognize the accident or fail to make correct diagnosis. under this circumstance, it is not very likely for operators to close the containment in time. The split fraction of the second branch is 1.0.

Case 3 through 6: Operators recognize a potential core damage accident and attempt to close the containment. The success probability of containment closure depends on the available time before onset of core damage; it increases with increasing time window. Since Surry has prepared a detailed procedure to perform this closure, the probability that the containment is successfully isolated is considered high. The mean split fractions for the first branch are 0.8, 0.9, 0.95 and 1.0 for time windows 1, 2, 3 and 4 respectively.

Question 11. Containment Pressure Capability.
3 Branches, Type 2, 3 Cases

The branches for this question are:

1. CP126p The mean containment failure pressure is 126 psig.

2. CP45p The mean containment failure pressure is 45 psig.

1. CP2p The mean containment failure pressure is 2 psig.

This question determines the mean containment failure pressure. The Surry containment closure procedure during POS 6 would provide a barrier capable of withstanding 45 psig, which is the design pressure of the containment during the full power operation. The mean failure pressure was estimated to be 126 psig in the NUREG-1150 study. However, it is not clear whether the containment during POS 6 can also provide 126 psig of failure pressure. Therefore, it was assumed in the base case analysis that the mean failure pressure is 45 psig when the containment is successfully isolated. A sensitivity analysis also performed where the failure pressure is 126 psig. The third branch is taken when the containment is not successfully isolated or leaks even after isolated. This question does not provide the distribution of the failure pressure itself; it is addressed in question 23.

Case 1: The containment is closed when the mid-loop operation begins. The probability that the containment is successfully isolated and does not leak is very high. The probability of containment leak after isolation was 0.0002 in the NUREG-1150. This value is increased to 0.01 in this study. The rest (0.99) is assigned to Branch 2 (CP45p) for the base case and to Branch 1 (CP126p) for the sensitivity case respectively.

Case 2: The containment is open when the mid-loop operation begins, but successfully closed before core damage. The probability that the containment may leak is higher than Case 1. This value is increased to 0.1. The rest (0.9) is assigned to Branch 2 (CP45p) for the base case and to Branch 1 (CP126p) for the sensitivity case respectively.

Case 3: The containment fails to close. The split fraction for Branch 3 is 1.00.

Question 12. Is AC Power Available Early?
2 Branches, Type 2, 9 Cases

The branches for this question are:

1. ERAC AC power is available in this time period.

2. NoERAC AC power is not available in this time period.

This question addresses the recovery of electric power to the injection pumps before the vessel breach for the cases where either off-site power or 4 kv bus was not available. Cases 1 through 8 of this question are sampled; the distributions were obtained from an analysis of data on offsite power recovery and restoration of the 4 kv bus for the Surry plant. These data are available in Vol. 1 of this report. The branching at this question depends upon the branches taken at Questions 1 and 4.

Probability of power recovery means the probability that offsite electrical power is recovered, or the 4 kv bus is restored in a specified period given that power was not recovered prior to the start of the period. These time periods available to recover power before vessel breach vary depending on the time window. The time periods used in Cases 1 through 8 are listed in Section 6.2. These time periods are derived from the results of MELCOR calculations which are presented in Appendix F of this report.

> Case 1 through 4: Offsite power was not available at the start of the accident which occurred in Time Windows 1, 2, 3 and 4 respectively. The probability of power recovery before vessel breach is calculated based on the recovery distribution and is assigned to the first branch.
>
> Case 5 through 8: Injection pumps were not operating because of loss of 4 kv bus to the available injection pumps at the start of the accident which occurred in Time Windows 1, 2, 3 and 4 respectively. The probability of restoring the 4 kv bus before vessel breach is assigned to the first branch.
>
> Case 9: Power was available at the start of the accident and remains available. The quantification for this case is:
> Branch 1: ERAC - 1.0
> Branch 2: NoERAC - 0.0

Question 13. Recovered from human errors early?
2 Branches, Type 2, 9 Cases

The branches for this question are:

1. ERHX Operators recover from previous error.
2. NoERHX Operators do not recover from the previous error.

This question addresses the recovery from operator errors for the accidents where the core damage occurred because of inadequate operator actions following the loss of RHR cooling. Cases 1 through 8 of this question are sampled; the distributions were obtained from "Handbook of HRA," (Reference 6.6 of this report) The branching at this question depends upon the branches taken at Questions 1 and 5.

The meaning of probability of recovery in this question is similar to that of Question 12, except that it concerns recovery from erroneous operator decisions in this question. The time periods in this question is identical to those of Question 12.

Question 14. Is Core Damage Arrested? No Vessel Breach?
2 Branches, Type 2, 4 Cases

The branches for this question are:

1. noVB The process of core degradation is arrested and a safe stable state is reached with the vessel intact.

2. VB Core degradation continues, resulting in core melt and vessel breach.

The branching at this question depends upon the branches previously taken at Questions 6, 12 and 13.

Case 1: ECCS were not available because of hardware error of injection pumps, or they failed during recirculation. In both cases, the ECCS are not recoverable and accident progress to vessel breach. The quantification for this case is:

Branch 1: 0.0
Branch 2: 1.0

Case 2: ECCS were not available due to loss of either offsite power or 4kv bus.They are not recovered during this time period. The quantification for this case is:

Branch 1: 0.0
Branch 2: 1.0

Case 3: ECCS were not available due to operator errors. They are not recovered during this time period. The quantification for this case is:

Branch 1: 0.0
Branch 2: 1.0

Case 4: ECCS are recovered either due to recovery of power or recovery from operator errors during this time period. The quantification for this case is:

Branch 1: 1.0
Branch 2: 0.0

Question 15. Vessel Pressure just before Breach?
2 Branches, Type 2, 2 Cases

The branches for this question are:

1. I-ImPr The vessel is at intermediate pressure before breach, about 500 to 1000 psia.

2. I-LoPr The vessel is at low pressure before breach, about 500 psia or less.

During mid-loop operation, at least one PORV is kept open. One PORV is sufficient to keep the RCS pressure below 500 psia at all time windows except Window 1. For Window 1, the RCS pressure can increase to 650 psia, which is the vent pressure of the RHR system. In any window, the RCS pressure would not increase to "high Pressure" range which is above 1000 psia.

Case 1: Accidents occur either in Windows 2, 3 or 4; more than 1 PORV is open; or operator successfully depressurize by opening additional PORV's.

Branch 1: ImPr - 0.00
Branch 2: LoPr - 1.00

Case 2: Accidents occur in Window 1, only one PORV is open and operator fails to open additional PORVs.

Branch 1: ImPr - 1.00
Branch 2: LoPr - 0.00

Question 16. Does an Alpha Mode Event Fail both the Vessel and the Containment?
2 Branches, Type 2, 2 Cases

The branches for this question are:

1. Alpha — A very energetic molten fuel-coolant interaction (steam explosion) in the vessel fails the vessel and generates a missile which fails the containment as well.

2. noAlpha — The vessel does not fail in this manner.

This question is sampled; the distribution used was taken from NUREG-1150, which was developed from opinions expressed by the Steam Explosion Review Group (SERG) and can be found in Volume 2, Part 6 of NUREG/CR-4551. The branch taken at this question depends upon the branches previously taken at Questions 14 and 15.

Case 1: There is vessel breach and the RCS was at low pressure. Steam explosions are more likely when the RCS is at low pressure than when the RCS is at some higher pressure. The aggregate distribution developed from distribution in the SERG was used for this case. This distribution covers many orders of magnitude. Based on the mean value of the distribution, the quantification for this case is:

Branch 1: Alpha - 0.008

Branch 2: noAlpha - 0.992

Case 2: This case includes two different groups of accidents. In the first group, the core degradation process has been arrested and there is no vessel breach. In the second group, there is vessel breach and the RCS was not at low pressure. In the latter group, steam explosions are possible but less likely when the RCS is not at low pressure. In NUREG-1150, this probability was set at 1/10 of the low pressure case, i.e., 0.0008. Since the fractions of accidents at intermediate pressure or higher is very small in this study, this probability is ignored. In both groups, the quantification is:

Branch 1: Alpha - 0.0
Branch 2: noAlpha - 1.0

Question 17. Type of Vessel Breach?
5 Branches, Type 2, 4 Cases

The branches for this question are:

1. PrEj The molten core material is ejected under considerable pressure from a hole in the bottom of the vessel.

2. Pour The molten core material pours slowly from the vessel, primarily driven by gravity.

3. BtmHd A large portion of the bottom head fails, perhaps due to a circumferential failure.

4. noVBoA There is no failure of the reactor pressure vessel.

5. Alpha An alpha mode failure has occurred.

Cases 3 is sampled. The type of vessel breach was taken from NUREG-1150 which was determined by the In-Vessel Expert Panel. The conclusions of the Experts and their aggregate distributions are presented in Volume 2, Part 1, of NUREG/CR-4551. The branch taken at this question depends upon the branches previously taken at Questions 14, 15 and 16.

The pressurized ejection failure mode requires that the RCS be at high pressure when the vessel fails. Although the pour failure mode is often considered to occur only with the RCS at low pressure, at least one Expert concluded that the probability of the failure mode with the RCS at high pressure at VB was non-zero. Since there could be a small driving force due to the gas pressure in the RCS, the Pour failure mode is distinguished by the fact that gravity is the primary force causing the molten core debris to leave the vessel.

The bottom head failure mode can occur at any RCS pressure; the failure could be a circumferential failure in which the whole bottom head falls into the cavity or some other failure in which a substantial portion of the bottom head fails. Bottom head failure at high pressure has effects similar to HPME; bottom head failure at low pressure has effects similar to a pour failure. The fourth branch is used when there is no vessel breach. The fifth branch is specified when the vessel failed in the Alpha mode.

Case 1: The core degradation process has been arrested and there is no vessel breach. The quantification for this case is:

Branch 1: PrEj - 0.0
Branch 2: Pour - 0.0
Branch 3: BtmHd - 0.0
Branch 4: noVB - 1.0
Branch 5: Alpha - 0.0

Case 2: An alpha mode failure of both the vessel and the containment has occurred. The quantification for this case is:

Branch 1: PrEj - 0.0
Branch 2: Pour - 0.0
Branch 3: BtmHd - 0.0
Branch 4: noVB - 0.0
Branch 5: Alpha - 1.0

Case 3: The vessel fails when the RCS is at intermediate pressure. The most likely failure mode is penetration failure leading to HPME. Based on the mean value of the distribution provided by the Experts, the quantification is:

Branch 1: PrEj - 0.60
Branch 2: Pour - 0.27
Branch 3: BtmHd - 0.13
Branch 4: noVB - 0.0
Branch 5: Alpha - 0.0

Case 4: The vessel fails when the RCS is at low pressure. The failure mode is gravity pour. The quantification for this case is:

Branch 1: PrEj - 0.0
Branch 2: Pour - 1.0
Branch 3: BtmHd - 0.0
Branch 4: noVB - 0.0
Branch 5: Alpha - 0.0

Question 18. Early Sprays?
4 Branches, Type 2, 5 Cases

The branches for this question are:

1. E-Sp The containment sprays are operating.

2. ESPfAC The containment sprays are available to operate if power is recovered.

3. ESPfHX The containment sprays are available to operate if operators recover from previous error.

4. ESPf The containment sprays are failed and cannot be recovered.

This question is not sampled; the branch chosen depends directly upon the branches taken at previous questions. The branch chosen for this question depends upon the branches taken at Questions 7, 12 and 13.

If the sprays were initially in the "available" state, the sprays will operate in this period, when power has been recovered, the 4 kv bus has been restored, or operators have recovered from previous errors, depending on what led to the core damage. If power is recovered and the sprays operate, the contents of the RWST will be transferred to the containment and the cavity will fill up with water.

Case 1: The sprays were available at the start of the accident. The quantification for this case is:

Branch 1: E-Sp - 1.0
Branch 2: ESPfAC - 0.0
Branch 3: ESPfHX - 0.0
Branch 4: ESPf - 0.0

Case 2: The sprays were failed at the start of the accident, and no recovery is possible, so the sprays remain failed. The quantification for this case is:

Branch 1: E-Sp - 0.0
Branch 2: ESpfAC - 0.0
Branch 3: ESPfHX - 0.0
Branch 4: ESPf - 1.0

Case 3: The sprays were available to operate at the start of the accident, but power or the 4 kv bus has not been recovered so the sprays remain available to operate in the future when power or 4 kv bus is recovered. The quantification for this case is:

Branch 1: E-Sp - 0.0
Branch 2: ESPfAC - 1.0
Branch 3: ESPfHX - 0.0
Branch 4: ESPf - 0.0

Case 4: The sprays were available to operate at the start of the accident, but operators have not recovered from previous errors so the sprays remain available to operate in the future when the operator error is corrected. The quantification for this case is:

Branch 1: E-Sp - 0.0
Branch 2: ESPfAC - 0.0
Branch 3: ESPfHX - 1.0
Branch 4: ESPf - 0.0

Case 5: The sprays were available to operate at the start of the accident. The sprays now operate because power has been recovered, the 4 kv bus has been restored, or operators have recovered from previous errors, depending on what led to the core damage. The quantification for this case is:

Branch 1: E-Sp - 1.0
Branch 2: ESpfAC - 0.0
Branch 3: ESPfHX - 0.0
Branch 4: ESPf - 0.0

Question 19. Amount of Water in the Reactor Cavity at Vessel Breach?
2 Branches, Type 2, 4 Cases

The branches for this question are:

1. RC-Wet The reactor cavity is full or nearly full of water.

2. RC-Dry The reactor cavity contains little or no water.

This question is not sampled; the amount of water in the reactor cavity may be reliably deduced from the information available about the injection of the RWST water into the containment and the operation of the sprays. The branch taken at this question depends upon the branches previously taken at Questions 8, 12 and 13.

What is of interest here is the presence of water for the direct containment heating (DCH) and ex-vessel steam explosion (EVSE) events. The magnitude of the pressure rise due to DCH depends upon whether there is water in the cavity. Whether an EVSE occurs also depends upon whether there is water in the cavity.

Case 1: The RWST was injected into the containment before breach or the sprays operated before breach; the reactor cavity is full of water at breach. The quantification for this case is:

Branch 1: RC-Wet - 1.0
Branch 2: RC-Dry - 0.0

Case 2: The RWST was not injected into the containment before breach, but the sprays operate before breach because of power recovery; the reactor cavity full of water at breach. The quantification for this case is:

Branch 1: RC-Wet - 1.0
Branch 2: RC-Dry - 0.0

Case 3: The RWST was not injected into the containment before breach, but the sprays operate before breach because of recovery from operator errors; the reactor cavity full of water at breach. The quantification for this case is:

Branch 1: RC-Wet - 1.0
Branch 2: RC-Dry - 0.0

Case 4: The RWST was not injected into the containment before breach, and the sprays never operate before breach; the reactor cavity contains little or no water at breach. The quantification for this case is:

Branch 1: RC-Wet - 0.0
Branch 2: RC-Dry - 1.0

Question 20. Baseline Containment Pressure just before Vessel Breach?
1 Branch, Type 4, 3 Cases

The single branch has the same name as the parameter read in at this question:

P1 IPBase The baseline pressure in the containment is read in as Parameter 1.

This question is not sampled; the baseline pressure before VB is a direct function of whether there is blowdown to the containment, whether there is containment heat removal, and whether the containment is successfully isolated. The available codes are in reasonable agreement about the value of the pressure in the containment before vessel breach for the full power cases, and it is believed that they similarly provide reasonable value of pressure for the lower power cases. Several calculations have been performed using the MELCOR codes to obtain the containment pressure. Results of these calculations are reported in Appendix F of this volume. The cases for this question depend upon the branches taken at Questions 11 and 17.

Case 1: The containment is not successfully isolated and does not have pressure retaining capability. The containment is near normal operating pressure. The value of IPBase is 17 psia.

Case 2: The containment is successfully isolated. There is no blowdown to containment, or the core damage has been arrested. The containment is near normal operating pressure. The value of IPBase is 17 psia.

Case 3: The containment is successfully isolated. There is no containment heat removal, and there is blowdown to the containment from PORV's and/or SRV's. The MELCOR results show containment pressures at about 19 psia.

Question 21. Pressure Rise at Vessel Breach?
1 Branch, Type 4, 5 Cases

A parameter is read in at this question:

P4. DP-VB The total containment pressure rise due to all the events that occur at vessel breach is read in as Parameter 4.

Cases 3 through 5 are sampled. Distributions for the pressure rise at vessel breach are taken from NUREG-1150 which was provided by the containment loads review group. The NUREG-1150 results on this subject were developed for full power cases. However, the pressure increase at vessel breach is primarily a function of the RCS pressure. Therefore, the NUREG-1150 results at low and intermediate RCS pressure are also applicable to the low power cases. The branch taken at this question depends upon the branches previously taken at Questions 15, 17 and 19. The experts provided distributions for pressure rise at VB that included the effects of all the events that accompany vessel failure. These include EVSE, vessel blowdown, hydrogen combustion, and DCH. The effects of the various events are not separable, so there is no way to extract, for example, the contribution of DCH or hydrogen combustion to the total pressure rise. Statistical tests on the aggregate distributions provided by the experts showed that their distributions for several cases are not distinguishable from their distributions for other cases. Thus, several cases are grouped together. Detailed information on the determination of the aggregate distributions for pressure rise at VB by the NUREG-1150 Containment Loads Expert Panel may be found in Volume 2, Part 2, of NUREG/CR-4551.

Case 1: There is no vessel breach, or alpha mode failure of the containment. The pressure rise is set to zero.

Case 2: There is an Alpha mode failure of the vessel and the containment. The pressure rise at vessel breach is set to an arbitrary high value to ensure that containment failure occurs in Question 24.

Case 3: At breach, the RCS is at low pressure, or the molten core debris pours out of the vessel under the influence primarily of gravity alone. The mean value of the aggregate distribution of the pressure rise for this case is 5.0 psi.

Case 4: The vessel fails with intermediate pressure in the RCS and there is water in the reactor cavity. The fraction of the core ejected is medium. The mean value of the aggregate distribution of the pressure rise for this case is 57.7 psi.

Case 5: The vessel fails with intermediate pressure in the RCS and there is little or no water in the reactor cavity. The mean value of the aggregate distribution of the pressure rise for this case is 64.7 psi.

Question 22. Does a Significant Ex-Vessel Steam Explosion occur?
2 Branches, Type 2, 2 Cases

The branches for this question are:

1. EVSE An energetic molten fuel-coolant interaction occurs in the reactor cavity upon vessel breach.

2. noEVSE No energetic molten fuel-coolant interaction occurs in the reactor cavity upon vessel breach.

This question is not sampled. The branch fractions were taken from NUREG-1150 which were quantified internally. The branch taken at this question depends upon the branches previously taken at Questions 17 and 19.

The dropping of hot metal into water has been observed to cause energetic and violent reactions which are commonly known as steam explosions. They appear to be more likely when the water is considerably below the saturation temperature. At Sandia National Laboratories, steam explosions were observed in 86% of the tests where hot metal was dropped into water. Some of these explosions were extremely energetic, others were not very energetic. In a severe reactor accident, a steam explosion may occur when the core slumps into the lower head of the vessel, known as an in-vessel steam explosion (IVSE), or when the lower head of the vessel fails and the core falls or is expelled into water in the reactor cavity beneath the vessel. This latter event is known as an ex-vessel steam explosion (EVSE). While IVSEs were explicitly considered for the BWR APETs, the probability of a PWR vessel failure by an IVSE was judged to be negligible. Thus IVSEs are not considered in this analysis for Surry.

The effects of EVSEs are considered in two places in this APET. If the RCS is at intermediate pressure at VB, the effects of an EVSE at VB are considered in Question 21. The experts who considered pressure rise at VB included the pressure rise due to EVSEs in their distributions for total pressure rise. The other effects of an EVSE are considered to be small compared with the effects of HPME. This question considers the effects of EVSEs when the vessel fails at low pressure or the molten debris pours from the vessel due to gravity alone. Whether an EVSE occurs following a low pressure VB also determines whether the debris bed in the reactor cavity after VB is in a coolable configuration and the amount of core involved in CCI.

Case 1: The vessel failure resulted in the melt pouring out, driven primarily by gravity, and there was water in the cavity when this occurred. This is the only situation in which an ex-vessel steam explosion is of interest. Not all steam explosions are "significant" in context used here. The fraction of the time that a pour of hot metal into water results in a significant EVSE is thought to be between 0.1 and 0.9. The state of knowledge in this area is such that, at this time, it is not possible to do a great deal better than assigning a probability of 0.50 to the probability for a significant EVSE. Thus, the quantification for this case is:

Branch 1: EVSE - 0.50
Branch 2: noEVSE - 0.50

Case 2: There was no vessel breach, or the cavity was dry at breach, or the vessel failed by an alpha mode event or by pressurized ejection of the melt. An ex-vessel steam explosion is not of interest or is not credible. The quantification for this case is:

Branch 1: EVSE - 0.0
Branch 2: noEVSE - 1.0

Question 23. Containment Failure Pressure?
1 Branch, Type 4, 3 Cases

A parameter is read in at this question:

P6. CF-Pr The containment failure pressure is read in as Parameter 6.

This question reads in the failure pressure of the containment. The comparison of the failure pressure with the load pressure, and the determination of the mode of failure, take place in the next question. Two distributions of the failure pressure are read in this question, depending on the nominal containment pressure which was discussed in Question 11. The distribution when the mean value is 126 psig is taken from NUREG-1150 (which was provided by the Structural Expert Panel.) The distribution when the mean value is 45 psig is obtained by scaling the above distribution proportionally.

Question 24. Containment Failure at VB?
2 Branches, Type 5, 1 Case

The branches for this question are:

1. ICF-Rupt The containment fails by rupture.

2. no-ICF The containment does not fail at vessel breach.

The result in this question depends upon the branches previously taken at Questions 20, 21 and 23.

This question adds the pressure rise at vessel breach (Parameter 4, DP-VB) to the base pressure in the containment before breach (Parameter 1, IPBase) to obtain the load pressure. This is then compared to the aggregate distribution of containment failure pressure (Parameter 6, CF-Pr) to determine whether the containment will fail or not.

Case 1: The containment is failed by an alpha mode event or a rocket
event. Dummy values are used with the comparison capability of EVNTRE

so that rupture, Branch 2, is selected.

Question 25. Containment Status at VB?
3 Branches, Type 2, 5 Case

The branches for this question are:

1.ICF-Rupt The containment is not successfully isolated at core damage, or fails at VB either by Alpha mode or rupture.

2.ICF-Leak The containment is isolated, but leaks.

3.no_ICF The containment is successfully isolated and does not fail at vessel breach.

This question summarizes the containment status based on the results of previous questions (Questions 10, 11, 17 and 24)

Question 26. Is AC Power Available Late?
2 Branches, Type 2, 9 Cases

The branches for this question are:

1. LRAC AC power is available during the initial portion of CCI.

2. noLRAC AC power is not available for this time period, but may be recovered in the future.

The time period of interest here is between vessel breach and the initial portion of CCI. This time period for each time window is given Section 6.1. This question addresses the recovery of electric power to the injection pumps during this time period for the cases where either off-site power or 4 kv bus was not available. The meaning of recovery probability and their distributions are the same as discussed in Question 12. The branching at this question depends upon the branches taken at Questions 1, 6 and 12. Cases 1 through 8 of this question are sampled.

Case 1 through 4: Offsite power was not available at the VB. The probability of power recovery during this period is calculated based on the recovery distribution and is assigned to the first branch.

Case 5 through 8: Injection pumps were not operating because of loss of 4 kv bus to the available injection pumps at VB. The probability of restoring the 4 kv bus during this period is assigned to the first branch.

Case 9: Power was available at VB and remains available. The quantification for this case is:

Branch 1: ERAC - 1.0

Branch 2: NoERAC - 0.0

Question 27. Recovered from human errors late?
2 Branches, Type 2, 5 Cases

The branches for this question are:

1. LRHX Operators recover from previous error.

2. NoLRHX Operators do not recover from the previous error.

This question addresses the recovery from operator errors for the accidents where the VB occurred because of inadequate operator actions following the core damage. The time period of interest is the same as Question 26. Cases 1 through 4 of this question are sampled. The meaning of recovery probability and their distributions are the same as discussed in Question 13. The branching at this question depends upon the branches taken at Questions 1 and 13.

Question 28. Late Sprays?
4 Branches, Type 2, 5 Cases

The branches for this question are:
The branches for this question are:

1. L-Sp The containment sprays are operating during this period.

2. LSPfAC The containment sprays are available to operate and will operate when power is recovered.

3. LSPfHX The containment sprays are available to operate if operators recover from previous error.

4. LSPf The containment sprays are failed and cannot be recovered.

This question is not sampled. The branch chosen for this question depends directly upon the branches taken at Questions 18, 26 and 27. The time period of interest is the same as in the preceding question. If sprays are recovered during this period, the release from CCI will be considerably reduced. If the debris bed is coolable and water was present, but was not being replenished, spray recovery will prevent dryout and the start of CCI. If the sprays were in the "available" state before, the sprays will operate in this period, when power has been recovered, the 4 kv bus has been restored, or operators have recovered from previous errors, depending on what led to the core damage. If power is recovered and the sprays operate, the contents of the RWST will be transferred to the containment and the cavity will fill up with water.

Case 1: The sprays were available at the start of the accident. The quantification for this case is:

Branch 1: L-Sp - 1.0
Branch 2: LSPfAC - 0.0
Branch 3: LSPfHX - 0.0
Branch 4: LSPf - 0.0

Case 2: The sprays were failed at the start of the accident, and no recovery is possible, so the sprays remain failed. The quantification for this case is:

Branch 1: L-Sp - 0.0
Branch 2: LSPfAC - 0.0
Branch 3: LSPfHX - 0.0
Branch 4: LSPf - 1.0

Case 3: The sprays are available to operate during this time period, but power or the 4 kv bus has not been recovered so the sprays remain available to operate in the future when power or 4 kv bus is recovered. The quantification for this case is:

Branch 1: L-Sp - 0.0
Branch 2: LSPfAC - 1.0
Branch 3: LSPfHX - 0.0
Branch 4: LSPf - 0.0

Case 4: The sprays are available to operate at the start of the accident, but operators have not recovered from previous errors so the sprays remain available to operate in the future when the operator error is corrected. The quantification for this case is:

Branch 1: L-Sp - 0.0
Branch 2: LSPfAC - 0.0
Branch 3: LSPfHX - 1.0
Branch 4: LSPf - 0.0

Case 5: The sprays were available to operate during the previous time period. The sprays now operate because power has been recovered, the 4 kv bus has been restored, or operators have recovered from previous errors, depending on what led to the core damage. The quantification for this case is:

Branch 1: L-Sp - 1.0
Branch 2: LSPfAC - 0.0
Branch 3: LSPfHX - 0.0
Branch 4: LSPf - 0.0

Question 29. Late Ignition
2 Branches, Type 2, 3 Cases

The branches for this question are:

1. L-Ign Ignition of the hydrogen in the containment will

occur during this period if the concentration is flammable.

2. noL-Ign Ignition of the hydrogen in the containment will not occur during this period even if the concentration is flammable.

This question determines whether conditions exist to ignite the hydrogen in the containment between VB and early part of CCI. The conditions that make hydrogen combustion capable of threatening the Surry containment in the late period are no prior failure, little or no combustion at VB, and absence of continuous electrical power and sprays. If the sprays do not operate in this period, the containment will be steam inert through this period and combustion is not possible. If power is recovered during this period and the sprays operate, then ignition is very likely.

Case 1: The containment is already failed. Ignition and burn at this time is irrelevant. The quantification for this case is:

Branch 1: L-Ign - 0.0
Branch 2: noL-Ign - 1.0

Case 2: Electrical power and spray operation were recovered during this period. Ignition is highly likely. The quantification for this case is:

Branch 1: L-Ign - 0.99
Branch 2: noL2-Ign - 0.01

Case 3: Electric power is not recovered during this time period. The sprays do not operate, so the containment will remain inerted by the high steam concentration. The quantification for this case is:

Branch 1: L-Ign - 0.00
Branch 2: noL-Ign - 1.00

Question 30. Number of Holes in vessel?
2 Branches, Type 1, 1 Case

The branches for this question are:

1. 1-Hole There is only one large hole in the RCS following VB.

2. 2-Hole There are two large holes in the RCS.

This question was intended to provide the information on the number of holes in the RCS to the source term code. However, source term analysis showed that this parameter does not make a significant contribution to the source term release fractions. Therefore, the this question is dummied out and take the first branch always.

Question 31. Late Containment Failure due to hydrogen burning?
3 Branches, Type 2, 6 Cases

The branches for this question are:

1. LCF-Rupt The containment already failed, or ruptures due to hydrogen burning during this period.

2. LCF-Leak The containment already leaks. No hydrogen burning is possible, since all hydrogen escaped the containment.

3. no-LCF The containment does not fail during this period.

The time period of interest of this question is same as the previous question. This question determines the status of the containment at the end of this time period, including rupture by hydrogen burning during this period. This question is not sampled. The branch chosen for this question depends directly upon the branches taken at Questions 11, 17, 18, 25, and 29.

The pressure generated by hydrogen burning when ignited is not calculated in this analysis. Calculations show that the pressure generated by hydrogen burning is generally much higher than 45 psig, but substantially below 126 psig. The probability of the containment failure by hydrogen burning is relatively small in this analysis since the dominating containment failure mode is initial isolation failure. Therefore, it has been assumed in this analysis that the containment would fail by hydrogen burning if the containment pressure capability is 45 psig, but would not if it is 126 psig.

Case 1: The containment already failed. The quantification for this case is:
Branch 1: LCF-Rupt - 1.0
Branch 2: LCF-Leak - 0.0
Branch 3: no-LCF - 0.0

Case 2: The containment already leaked. The quantification for this case is:
Branch 1: LCF-Rupt - 0.0
Branch 2: LCF-Leak - 1.0
Branch 3: no-LCF - 0.0

Case 3 : Conditions do not exist for hydrogen ignition. The containment does not fail. The quantification for this case is:
Branch 1: LCF-Rupt - 0.0
Branch 2: LCF-Leak - 0.0
Branch 3: no-LCF - 1.0

Case 4: The containment failure pressure is 45 psig. Conditions exist for hydrogen ignition. HPME occurred at VB, and most of hydrogen released at VB did not burn. The quantification for this case is:
Branch 1: LCF-Rupt - 1.0
Branch 2: LCF-Leak - 0.0
Branch 3: no-LCF - 0.0

Case 5: The containment failure pressure is 45 psig. Conditions exist for hydrogen ignition. The core debris poured out of the vessel at VB. None of hydrogen released at VB burned. The quantification for this case is:

Branch 1: LCF-Rupt - 1.0
Branch 2: LCF-Leak - 0.0
Branch 3: no-LCF - 0.0

Case 6: The containment does not fail either because the containment failure pressure is 126 psig, or sufficient amount of hydrogen does not exist in the containment. The quantification for this case is:

Branch 1: LCF-Rupt - 0.0
Branch 2: LCF-Leak - 0.0
Branch 3: no-LCF - 1.0

Question 32. Is the Debris Bed in a Coolable Configuration?
2 Branches, Type 2, 5 Cases

The branches for this question are:

1. CDB The debris bed is coolable; no CCI takes place as long as the debris remains covered with water.

2. noCDB The debris bed is not coolable. CCI will begin as soon as the melt reheats whether water is present or not.

This question is not sampled and was quantified internally. The branch taken at this question depends upon the branches previously taken at Questions 17 and 22.

Core-concrete interactions will not occur if the debris bed is in a coolable configuration, and if there is water present to cool it. This question determines whether the debris bed is coolable assuming that water is present when the core debris enters the cavity and is continuously replenished thereafter. Whether water is present is determined in the next question. The portion of the molten core that participates in DCH is unavailable for CCI. Thus the core debris considered in this question is the debris expelled at VB that remains in the cavity and the debris that leaves the vessel some time after VB. More discussion of debris coolability topic may be found in Volume 2, Part 6, of NUREG/CR-4551.

An Alpha event is likely to scatter at least some corium around the containment. If this corium comes to rest in a thin uniform layer, air cooling will suffice. However, it is possible that drifts of corium particles might accumulate in corners, in the wall-floor angle, and so on, that would be large enough to reheat and start CCI. Debris coolability is very uncertain in scenarios such as these.

Case 1: The vessel has failed by an Alpha mode event which also fails the containment. These events are so energetic that a substantial portion of the core debris is likely to be widely scattered throughout the containment. However, very little is known about these events or the expected corium distribution. Since the Alpha mode failure of the containment also fail the sprays, there is no supply of water to the cavity and CCI will occur eventually even if the debris bed is initially coolable and some water is present. Thus the quantification for this case is largely irrelevant. The quantification for this case is:

Branch 1: CDB - 0.85
Branch 2: noCDB - 0.15

Case 2: There was no vessel failure; CCI does not occur. The quantification of this case is:

Branch 1: CDB - 1.0
Branch 2: noCDB - 0.0

Case 3: The vessel failure resulted in HPME or gross bottom head failure at high pressure. The core debris involved in HPME but which does not participate in DCH is likely to be widely distributed throughout the containment. The state of the debris ejected at vessel failure which remains in the cavity is not well known. The debris that pours out of the vessel some time after vessel blowdown may assume a coolable form if it fragments into pieces that are neither too small nor too large. The quantification for this case is:

Branch 1: CDB - 0.80
Branch 2: noCDB - 0.20

Case 4: A gravity pour of the core debris resulted at vessel breach, and an EVSE occurred. The EVSE may spread a portion of the debris throughout the containment where it would be coolable. On the other hand, the EVSE may create fine particles that remain in the cavity and make the bulk of the core debris in the cavity noncoolable. The quantification for this case is:

Branch 1: CDB - 0.80
Branch 2: noCDB - 0.20

Case 5: The vessel has failed with a gravity pour resulting. No EVSE occurred. All of the core which exits the vessel should remain on the cavity floor. To form a coolable debris bed, the debris must fragment when it hits the water, the resulting particles must quench while falling through the water, and the size of the bulk of the particles must fall within a certain size range. Further, if a portion of the debris bed is noncoolable, the available evidence is that this portion of the bed will grow in size until essentially the entire bed has become noncoolable. The quantification for this case is:

Branch 1: CDB - 0.35
Branch 2: noCDB - 0.65

Question 33. Does Prompt CCI Occur?
2 Branches, Type 2, 2 Cases

The branches for this question are:

1. PrmptCCI CCI occurs promptly following vessel breach.

2. noPrmCCI CCI does not occur promptly after vessel breach.

This question is not sampled; whether prompt CCI occurs follows logically from the information available about the coolability of the core debris and the presence of water in the reactor cavity. The branch taken at this question depends upon the branches previously taken at Questions 17, 19, and 32.

Case 1: The debris is coolable and there is water in the cavity, or there has been no vessel breach. In either case, CCI does not begin promptly. The situation where the containment sprays operate only in the late period is not considered a sufficient water supply to prevent CCI from starting. Electrical power may be recovered any time in the period, so the sprays may not start until several hours after VB. Water must be continuously present from VB to prevent prompt CCI. The quantification for this case is:
Branch 1: PrmptCCI - 0.0
Branch 2: noPrmCCI - 1.0

Case 2: There is no water in the cavity, or the debris is not coolable. In either case, CCI begins promptly, either at once, if the debris is hot when it leaves the vessel, or as soon as the debris has reheated. The quantification for this case is:
Branch 1: PrmptCCI - 1.0
Branch 2: noPrmCCI - 0.0

Question 34. Is AC Power Available Very Late?
2 Branches, Type 2, 9 Cases

The branches for this question are:

1. L2RAC AC power is available Very Late?

2. noL2RAC AC power is not available for this time period, but may be recovered in the future.

The time period of interest here is from the initial portion of CCI to 24 hours. This time period for each time window is given Section 6.1. This question addresses the recovery of electric power to the injection pumps during this time period for the cases where either off-site power or 4 kv bus was not available. The meaning of recovery probability and their distributions are the same as discussed in Question 12. The branching at this question depends upon the branches taken at Questions 1, 6 and 26. Cases 1 through 8 of this question are sampled.

Case 1 through 4: Offsite power was not available at the end of the previous period. The probability of power recovery during this period is calculated based on the recovery distribution and is assigned to the first branch.

Case 5 through 8: Injection pumps were not operating because of loss of 4 kv bus to the available injection pumps at the end of the previous period. The probability of restoring the 4 kv bus during this period is assigned to the first branch.

Case 9: Power was available at the end of the previous period and remains available. The quantification for this case is:

Branch 1: ERAC - 1.0
Branch 2: NoERAC - 0.0

Question 35. Recovered from human errors very late?
2 Branches, Type 2, 5 Cases

The branches for this question are:

1. L2RHX Operators recover from previous error during this time period.

2. NoL2RHX Operators do not recover from the previous error during this time period.

This question addresses the recovery from operator errors for the accidents where the VB occurred because of inadequate operator actions following the core damage. The time period of interest is the same as Question 34. Cases 1 through 4 of this question are sampled. The meaning of recovery probability and their distributions are the same as discussed in Question 27. The branching at this question depends upon the branches taken at Questions 1 and 27.

Question 36. Very Late Sprays?
2 Branches, Type 2, 5 Cases

The branches for this question are:

1. L2-Sp The containment sprays are operating during this period.

2. L2SPf The containment sprays are not recovered.

This question is not sampled. The branch chosen for this question depends directly upon the branches taken at Questions 28, 34 and 35. The time period of interest is the same as in the preceding question. If sprays are recovered during this time period, steam condensation will de-inert the containment, making a hydrogen burn possible. If the debris bed is coolable, spray operation during this period is required to prevent dryout and concrete attack.

Case 1: The sprays were available in the previous period. The quantification for this case is:
Branch 1: L2-Sp - 1.0
Branch 2: L2SPf - 0.0

Case 2: The sprays were failed at the start of the accident, and no recovery is possible, so the sprays remain failed. The quantification for this case is:
Branch 1: L2-Sp - 0.0
Branch 2: L2Spf - 1.0

Case 3: The sprays are available to operate during this time period, but power or the 4 kv bus has not been recovered so the sprays remain available to operate in the future when power or 4 kv bus is recovered. The quantification for this case is:
Branch 1: L2-Sp - 0.0
Branch 2: L2SPf - 1.0

Case 4: The sprays are available to operate at the start of the accident, but operators have not recovered from previous errors so the sprays remain available to operate in the future when the operator error is corrected. The quantification for this case is:
Branch 1: L2-Sp - 0.0
Branch 2: L2SPf - 1.0

Case 5: The sprays were available to operate during the previous time period. The sprays now operate because power has been recovered, the 4 kv bus has been restored, or operators have recovered from previous errors, depending on what led to the core damage. The quantification for this case is:
Branch 1: L2-Sp - 1.0
Branch 2: L2Spf - 0.0

Question 37. Does Delayed CCI Occur?
2 Branches, Type 2, 2 Cases

The branches for this question are:

1. DldCCI CCI occurs after a delay to boil off the water in the cavity.

2. noDldCCI CCI does not occur after a delay to boil off the water in the cavity.

This question is not sampled; whether delayed CCI occurs follows logically from the information available about the coolability of the core debris, whether prompt CCI has occurred, and whether the sprays are operating. The branch taken at this question depends upon the branches previously taken at Questions 17, 33, and 36.

Case 1: Prompt CCI did not occur and the sprays are now operating, or prompt CCI occurred, or there was no vessel breach. If prompt CCI occurred, delayed CCI is not possible. If prompt CCI did not occur, the debris bed must have been coolable with water available. Since the sprays are now operating, the water cooling the debris bed is being replenished and delayed CCI will not take place. The quantification for this case is:

Branch 1: DelydCCI - 0.0
Branch 2: noDldCCI - 1.0

Case 2: Prompt CCI did not occur, and the sprays are not operating. The debris bed must have been coolable, and there must have been some water present, or prompt CCI would have resulted. As the water being boiled off is not being replenished, delayed CCI will begin when the water is all boiled off. The quantification for this case is:

Branch 1: DelydCCI - 1.0
Branch 2: noDldCCI - 0.0

Question 38. Does Very Late Ignition Occur?
2 Branches, Type 2, 5 Cases

The branches for this question are:

1. L2-Ign — Ignition of the hydrogen in the containment will occur during this period if the concentration is flammable.

2. noL2-Ign — Ignition of the hydrogen in the containment will not occur during this period even if the concentration is flammable.

This question is not sampled and was quantified internally. The applicable case depends upon the branches taken at Questions 26, 29, 31, 34 and 36.

This question determines whether conditions exist to ignite the hydrogen in the containment during the latter part of CCI. In the very late period, if no burns, containment failures, or bypasses have occurred, the hydrogen available is that produced in-vessel or at VB, the hydrogen produced by oxidizing all the remaining unoxidized Zr during the initial part of CCI, and the hydrogen produced in CCI in addition to that from oxidizing the rest of the zirconium. Significant combustion events during this period are negligible if electric power and containment sprays have been continuously available since the start of the accident. They may occur in this period when electric power and the sprays are recovered.

Case 1: The containment is already failed. Ignition and burn at this time is irrelevant. The quantification for this case is:

Branch 1: L2-Ign - 0.0
Branch 2: noL2-Ign - 1.0

Case 2: Ignition occurred in the previous time period, and most of hydrogen already burned. The quantification for this case is:

Branch 1: L2-Ign - 0.0
Branch 2: noL2-Ign - 1.0

Case 3: Electrical power and spray operation were recovered during this period. Ignition is highly likely. The quantification for this case is:

Branch 1: L2-Ign - 0.99
Branch 2: noL2-Ign - 0.01

Case 4: Electric power is not recovered during the time frame of interest for this analysis. The sprays do not operate, so the containment will remain inerted by the high steam concentration for some time. Eventually the steam concentration in the containment may drop to about 55%, and then ignition is possible if enough hydrogen is present. When no electrical power is available ignition appears to be a stochastic phenomenon. A similar case was considered by the experts considering hydrogen combustion events at Grand Gulf. They gave distributions for ignition probability which depended on the hydrogen concentration. This issue is summarized in Volume 2, Part 2, of NUREG/CR-4551. The mean values of their ignition probability distributions for concentrations of interest are about 0.30. This value (0.30) for the ignition probability also includes implicitly the probability that heat loss through the containment wall alone eventually causes enough wall condensation to reduce the steam concentration to about 55%. The quantification for this case is:

Branch 1: L2-Ign - 0.30
Branch 2: noL2-Ign - 0.70

Case 5: The concentration is not flammable or is steam-inert. Very late ignition cannot take place. The quantification for this case is:

Branch 1: L2-Ign - 0.0
Branch 2: noL2-Ign - 1.0

Question 39. Very Late Containment Failure due to hydrogen burning?
3 Branches, Type 2, 5 Cases

The branches for this question are:

1. L2CF-Rupt The containment already failed, or ruptures due to hydrogen burning during this period.

2. L2CF-Leak The containment already leaks. No hydrogen burning is possible, since all hydrogen escaped the containment.

3. no-L2CF The containment does not fail during this period.

The time period of interest of this question is same as the previous question. This question determines the status of the containment at the end of this time period, including rupture by hydrogen burning during this period. This question is not sampled. The branch chosen for this question depends directly upon the branches taken at Questions 11, 31, 33, 37, and 38.

As in Question 31, the pressure generated by hydrogen burning when ignited is not calculated in this question and it is assumed that the containment would fail by hydrogen burning if the containment pressure capability is 45 psig, but would not if it is 126 psig.

Case 1: The containment already failed. The quantification for this case is:
Branch 1: L2CF-Rupt - 1.0
Branch 2: L2CF-Leak - 0.0
Branch 3: no-L2CF - 0.0

Case 2: The containment already leaked. The quantification for this case is:
Branch 1: L2CF-Rupt - 0.0
Branch 2: L2CF-Leak - 1.0
Branch 3: no-L2CF - 0.0

Case 3 : Conditions do not exist for hydrogen ignition. The containment does not fail. The quantification for this case is:
Branch 1: L2CF-Rupt - 0.0
Branch 2: L2CF-Leak - 0.0
Branch 3: no-L2CF - 1.0

Case 4: The containment failure pressure is 45 psig. Conditions exist for hydrogen ignition. Prompt or delayed CCI occurred. The quantification for this case is:
Branch 1: L2CF-Rupt - 1.0
Branch 2: L2CF-Leak - 0.0
Branch 3: no-L2CF - 0.0

Case 5: The containment does not fail either because the containment failure pressure is 126 psig, or sufficient amount of hydrogen does not exist in the containment. The quantification for this case is:
Branch 1: L2CF-Rupt - 0.0
Branch 2: L2CF-Leak - 0.0
Branch 3: no-L2CF - 1.0

Question 40. Final Containment Condition and Failure Time?
6 Branches, Type 2, 7 Cases

The branches for this question are:

1. Leak-I — The containment was initially isolated, but leaks.

2. Rupt-I — The containment was not successfully isolated before the core damage.

3. Rupt-VB — The containment failed at VB either because of Alpha mode failure, DCH or EVSE.

4. Rupt-L — The containment failed late due to hydrogen burning.

5. Rupt-L2 — The containment failed very late due to hydrogen burning.

6. No-CF — The containment remains intact in 24 hours.

This question is not sampled. This question utilizes the results of many previous questions to summarize the state of the containment at the end of this event tree analysis. Only the most important condition in determining the releases is considered. The branches in this question depend upon the branches previously taken at Questions 11, 25, 31, and 39.

B.2 Listing of the Accident Event Tree

```
  SURRY LPSD APET, Rev 10, Oct 21, 93 - Adapted from N1150
   40
  This input represent SPDS 1 / SBO (5 PDS).
    1                    1.                                              $$ SBO
                        SBO
1 Window                                                     $ PDS1
    4     Win-1       Win-2       Win-3       Win-4
    1         1           2           3           4
          1.000       0.000       0.000       0.000
2 Vessel Break Size                                          $ PDS 4
    2     Large       Small
    2         1           2
    2 Cases
    1         1
              1
             W1
           0.95        0.05
      Otherwise
          1.000       0.000
3 Depressurization before vessel breach?
    2      Vdep     No-Vdep
    2         1           2
    2 Cases
    2         1           2
              1 *         2
             W1 &     1PORV
          0.800       0.200           $ Depends on HRA?
      Otherwise
          1.000       0.000
4 Status of AC at CD?                                        $ PDS 2
    3        AC       No-AC      No-4KV
    1         1           2           3
          0.000       1.000       0.000
5 CD due to HX?                                         $ PDS 3
    3     No-HX         HXA         HXD
    2         1           2           3
    5  Cases
    1         4
              2
          No-AC
          1.000       0.000       0.000
    1         1
              1
             W1
          1.000       0.000       0.000
    1         1
              2
             W2
          1.000       0.000       0.000
    1         1
              3
             W3
          1.000       0.000       0.000
       Otherwise      $(1/4 W4)
          1.000       0.000       0.000     $ Depends on Level 1
6 Status of ECCS at CD?                                            $ PDS 5
    5  ECCSf4KV    ECCSfAC     ECCSfHX     ECCSfHW    ECCSfREC
```

```
    1         1            2            3            4            5
          0.000        0.000        0.000        0.000        1.000  $Level 1
  7 Spray at CD?                                           $ PDS 6
    6        SP     SPfAC     SPfHX     SPfHW    SPfREC   SPf4KV
    2         1         2         3         4         5        6
    2 Cases
    1         1
              1
             W1
          0.000     0.000     0.000     0.000     0.000    1.000
      Otherwise
          0.000     0.000     0.000     0.000     0.000    1.000
  8 RWST at CD                                             $ PDS 8
    4   RWST-In       RWSTfAC     RWSTfHX     RWSTfIn
    2         1            2            3           4
    2 Cases
    2         6            7
              4  *         4
        ECCSfHW &    SPfHW
          0.000        0.000        0.000       1.000
       Otherwise
          1.000        0.000        0.000       0.000
  9 Is containment closed at beginning of accident?      $ PDS0
    2   ClsdCI   No-ClsdCI
    1         1            2
          0.000        1.000
 10 Is containment closed before core damage?
    2   ClsdCD   No-ClsdCD
    2         1            2
    6  Cases
    1         9
              1
        ClsdCI
          1.000        0.000
    1         5
              3
            HXD
          0.000        1.000
    1         1
              1
             W1
          0.800        0.200             $ Based on Human Error Curve
    1         1
              2
             W2
          0.900        0.100
    1         1
              3
             W3
          0.950        0.050
      Otherwise                                      $ W4
          1.000        0.000
 11 Containment Pressure Capability
    3   CP126p         CP45p         CP2p
    2         1            2            3
    3 Cases
    1         9
              1
        ClsdCI
        0.9998         0.000       0.0002
```

```
      2         9           10
                2 *          1
     No-ClsdCI &      ClsdCD
        0.900         0.000         0.100
      Otherwise
        0.000         0.000         1.000
12 AC recovered before VB?   $ Conditional..Power/4KV Recovery Curve
      2       ERAC     No-ERAC
      2         1           2
      9 Cases
      2         1           4
                1 *         2
               W1 &     No-AC
        0.800         0.200
      2         1           4
                2 *         2
               W2 &     No-AC
        0.850         0.150
      2         1           4
                3 *         2
               W3 &     No-AC
        0.900         0.100
      2         1           4
                4 *         2
               W4 &     No-AC
        0.950         0.050
      2         1           4
                1 *         3
               W1 &     No-4KV
        0.800         0.200
      2         1           4
                2 *         3
               W2 &     No-4KV
        0.850         0.150
      2         1           4
                3 *         3
              W3 &      No-4KV
        0.900         0.100
      2         1           4
                4 *         3
               W4 &     No-4KV
        0.950         0.050
       Otherwise
        1.000         0.000
13 Recovered from HX before VB?    $ Conditional .. HRA curve(Chu?)
      2       ERHX      No-ERHX
      2         1            2
      9 Cases
      2         1            5
                1 *          2
               w1 &        HXA
        0.500          0.500
      2         1            5
                1 *          3
               W1 &        HXD
        0.850          0.150
      2         1            5
                2 *          2
               w2 &        HXA
        0.600          0.400
```

```
      2           1              5
                  2 *            3
                 W2 &          HXD
              0.900          0.100
      2           1              5
                  3 *            2
                 w3 &          HXA
              0.700          0.300
      2           1              5
                  3 *            3
                 W3 &          HXD
              0.930          0.070
      2           1              5
                  4 *            2
                 w4 &          HXA
              0.800          0.200
      2           1              5
                  4 *            3
                 W4 &          HXD
              0.950          0.050
          Otherwise
              1.000          0.000
14 VB?
      2     No-VB               VB
      2           1              2
      4 Cases
      2           6              6
                  4   +          5
          ECCSfHW   or     Recf
              0.000          1.000
      3           6              6            12
        (         1   +          2 )   *        2
         (ECCSf4KV or  ECCSfAC)   &   No-ERAC
              0.000          1.000
      2           6             13
                  3   *          2
          ECCSfHX   &   No-ERHX
              0.000          1.000
          Otherwise
              1.000          0.000
15 Vessel Pressure just before Breach?
      2    I-ImPr        I-LoPr
      2           1              2
      2 Cases
      1           3
                  1
               VDEP
              0.000          1.000
          Otherwise
              1.000          0.000
16 ALPHA Mode Failure?
      2     Alpha        noAlpha
      2           1              2
      2 Cases
      2          14             15
                  2   *          2
                 VB   &     I-Lopr
             0.0080         0.9920
          Otherwise
             0.0000         1.0000
```

```
17 Type of Vessel Breach?   $ Summary .. Cumulative.
    5      PrEj          Pour        BtmHd          noVB       ALPHA
    2         1             2            3             4           5
    4 Cases
    1        14
              1
           noVB
          0.000         0.000        0.000         1.000       0.000
    1        16
              1
          ALPHA
          0.000         0.000        0.000         0.000       1.000
    2        14            15
              2  *          1
             VB  &    I-ImPr
         1.0000        0.0000       0.0000        0.0000       0.000
       Otherwise - I-LoPr
         0.0000        1.0000       0.0000        0.0000       0.000
18 Early Sprays before vb?  $ Cumulative
    4      E-Sp         ESPfAC      ESPfHX          ESPf
    2         1             2            3             4
    5 Cases
    1         7
              1
             Sp
          1.000         0.000        0.000         0.000
    2         7             7
              4  +          5
          SPfHW  or      Recf
          0.000         0.000        0.000         1.000
    3         7             7           12
       (      2  +          6 )  *       2
       ( SPfAC   or   SPf4KV )   & No-ERAC
          0.000         1.000        0.000         0.000
    2         7            13
              3     *       2
          SPfHX     & No-ERHX
          0.000         0.000        1.000         0.000
       Otherwise
          1.000         0.000        0.000         0.000
19 Amount of Water in the Reactor Cavity at Vessel Breach?
    2    RC-Wet        RC-Dry
    2         1             2
    4 Cases
    1         8
              1
         RWSTIn
          1.000         0.000
    2         8            12
              2  *          1
       RWSTfAC   &       ERAC
          1.000         0.000
    2         8            13
              3  *          1
       RWSTfHX   &       ERHX
          1.000         0.000
       Otherwise
          0.000         1.000
20 Baseline Containment Pressure just before VB?
    1    IPBase
```

```
      4          1
      3 Cases
      1         11
                 3
              CP2p
             1.000
      1
      1     17.00
      1         17
                 4
              noVB
             1.000
      1
      1     17.00
          Otherwise
             1.000
      1
      1     19.00
21 Pressure Rise at Vessel Breach?
      1     DP-VB
      4          1
      5 Cases
      1         17
                 4
              noVB
             1.000
      1
      4      0.00
      1         17
                 5
             Alpha
             1.000
      1
      4    777.00
      2         15           17
                 2     +      2
           I-LoPr    or    Pour
             1.000
      1
      4      5.00
      2         15           19
                 1     *      1
           I-ImPr    &   RC-Wet
             1.000
      1
      4     57.70
          Otherwise
             1.000
      1
      4     64.70
22 Does a Significant Ex-Vessel Steam Explosion Occur?
      2      EVSE        noEVSE
      2         1             2
      2 Cases
      2        19            17
                1    *        2
           RC-Wet    &     Pour
            0.500         0.500
          Otherwise -- No EVSE
            0.000         1.000
```

```
23 Containment Failure Pressure?
     1     CF-Pr
     4         1
     3 Case
     1        .1
               1
         CP126p
          1.000
     1
     6    125.70
     1        11
               2
          CP45p
          1.000
     1
     6     59.70
        Otherwise
          1.000
     1
     6     16.70
24 Containment Rupture at VB?
     2   ICF-Rupt      no-ICF
     5         1           2
     3         1           4          -6
         IPBase        DP-VB       CF-Pr
            AND
        GETHRESH  1        0
         ICF rupture if IPBase + (DP-VB) > CF-Pr
25 Containment Status at VB?
     3   ICF-Rupt     ICF-Leak      no-ICF
     2         1           2           3
     5 Cases
     2        10         11
               1  *        3
          ClsdCD &     CP2p
          0.000        1.000       0.000
     2        10         11
               2  *        3
        NoClsdCD &     CP2p
          1.000        0.000       0.000
     1        17
               5
          Alpha
          1.000        0.000       0.000
     1        24
               1
          ICF-R
          1.000        0.000       0.000
        Otherwise
          0.000        0.000       1.000
26 AC recovered Late?  $ Conditional
     2      LRAC      No-LRAC
     2         1           2
     9 Cases
     3        12            1           4
               2  *         1  *        2
        NO-ERAC &          W1  &    No-AC
          0.800         0.200
     3        12            1           4
               2  *         2  *        2
```

```
      NO-ERAC &          W2  &    No-AC
        0.850        0.150
    3       12            1            4
             2 *          3  *         2
      NO-ERAC &          W3  &    No-AC
        0.900        0.100
    3       12            1            4
             2 *          4  *         2
      NO-ERAC &          W4  &    No-AC
        0.950        0.050
    3       12            1            4
             2 *          1  *         3
      NO-ERAC &          W1  &    No-4KV
        0.800        0.200
    3       12            1            4
             2 *          2  *         3
      NO-ERAC &          W2  &    No-4KV
        0.850        0.150
    3       12            1            4
             2 *          3  *         3
      NO-ERAC &          W3  &    No-4KV
        0.900        0.100
    3       12            1            4
             2 *          4  *         3
      NO-ERAC &          W4  &    No-4KV
        0.950        0.050
      Otherwise               $ 12/1
        1.000        0.000
27 Recovered from HX Late?  $Conditional
    2     LRHX     No-LRHX
    2        1           2
    5 Cases
    2       13            1
             2  *         1
      NO-ERHX  &         W1
        0.800        0.200
    2       13            1
             2  *         2
      NO-ERHX  &         W2
        0.850        0.150
    2       13            1
             2  *         3
      NO-ERHX  &         W3
        0.900        0.100
    2       13            1
             2  *         4
      NO-ERHX  &         W4
        0.950        0.050
      Otherwise
        1.000        0.000
28 Late Sprays (After VB but before major CCI)?  $ Cumulative
    4     L-Sp         LSPfAC     LSPfHX          LSPf
    2        1            2            3             4
    5 Cases
    1       18
             1
          E-Sp
        1.000        0.000        0.000         0.000
    1       18
             4
```

```
         ESPf
        0.000        0.000        0.000        1.000
    2       18           26
             2    *       2
       ESPfAC     & No-LRAC
        0.000        1.000        0.000        0.000
    2       18           27
             3    *       2
       ESPfHX     & No-LRHX
        0.000        0.000        1.000        0.000
      Otherwise
        1.000        0.000        0.000        0.000
29 Does Late Ignition Occur?
    2   L-Ign       noL-Ign
    2       1            2
    3 Cases
    2      25           25
            1    +       2
      C-Rupt   or C-Leak
        0.000        1.000
    3      12           26           28
            2    *       1    *       1
      noE-RAC  &   L-RAC    &    L-SP
        0.990        0.010
      Otherwise
        0.000        1.000
30 DUMMY
    2  1-Hole        2-Hole
    1       1              2
      1.000         0.000
31 Late Containment Failure due to H2 Burning?  $(After VB but before CCI)
    3  LCF-Rupt     LCF-Leak     no-LCF
    2       1            2            3
    6 Cases
    1      25
            1
     ICF-Rupt
        1.000         0.000        0.000
    1      25
            2
     ICF-Leak
        0.000         1.000        0.000
    1      29
            2
     NoL-Ign
        0.000         0.000       1.000
    4      11            18          17          17
            2  *         -1  *    (  1  +        3  )
        C60p  &      NoE-Sp  &    PrEj        BtmHd
        1.000         0.000        0.000
    2      11            17
            2  *          2
        C60p  &        Pour
        1.000         0.000        0.000
     Otherwise
        0.000         0.000        1.000
32 Is the Debris Bed in a Coolable Configuration?
    2      CDB        noCDB
    2       1            2
    5 Cases
```

```
      1       17
               5
            Alpha
            0.850        0.150
      1       17
               4
             noVB
            1.000        0.000
      2       17           17
               1 +          3
             PrEj or    BtmHd
            0.800        0.200
      1       22
               1
             EVSE
            0.800        0.200
          Otherwise - Pour & noEVSE
            0.350        0.650
33 Does Prompt CCI Occur?
      2  PrmptCCI     noPrmCCI
      2        1            2
      3 Cases
      1       17
               4
             noVB
            0.000        1.000
      2       32           19
               1 *          1
              CDB &    RC-Wet
            0.000        1.000
          Otherwise -- Not coolable or no water
            1.000        0.000
34 AC rec before 24 hours?  $ Conditional
      2     L2RAC     No-L2RAC
      2        1            2
      9 Cases
      3       26            1            4
               2 *          1    *       2
          NO-LRAC &        W1    &   No-AC
            0.800        0.200
      3       26            1            4
               2 *          2    *       2
          NO-LRAC &        W2    &   No-AC
            0.850        0.150
      3       26            1            4
               2 *          3    *       2
          NO-LRAC &        W3    &   No-AC
            0.900        0.100
      3       26            1            4
               2 *          4    *       2
          NO-LRAC &        W4    &   No-AC
            0.950        0.050
      3       26            1            4
               2 *          1    *       3
          NO-LRAC &        W1    &   No-4KV
            0.800        0.200
      3       26            1            4
               2 *          2    *       3
          NO-LRAC &        W2    &   No-4KV
            0.850        0.150
```

```
     3        26            1           4
               2 *          3  *        3
        NO-LRAC &          W3  &     No-4KV
          0.900        0.100
     3        26            1           4
               2 *          4  *        3
        NO-LRAC &        W4  &       No-4KV
          0.950        0.100
        Otherwise                $ 12/1
          1.000        0.000
35 Recovered from HX before 24 HOURS?  $Conditional
     2      L2RHX     No-L2RHX
     2        1             2
     5 Cases
     2        27            1
               2  *         1
        NO-LRHX  &         W1
          0.800        0.200
     2        27            1
               2  *         2
        NO-LRHX  &         W2
          0.850        0.150
     2        27            1
               2  *         3
        NO-LRHX  &         W3
          0.900        0.100
     2        27            1
               2  *         4
        NO-LRHX  &         W4
          0.950        0.050
        Otherwise
          1.000        0.000
36 Very Late Sprays (BEFORE  24 hours)?  $Cumulative
     2      L2-Sp          L2SPf
     2        1              2
     5 Cases
     1        28
               1
            L-SP
          1.000          0.000
     1        28
               4
            LSPf
          0.000          1.000
     2        28            34
               2  *          2
          LSPfAC  &    No-L2RAC
          0.000          1.000
     2        28            35
               3  *          2
          LSPfHX  &    No-L2RHX
          0.000          1.000
        Otherwise
          1.000          0.000
37 Does Delayed CCI Occur?
     2      DldCCI      noDldCCI
     2        1             2
     4 Cases
     1        17
               4
```

```
         noVB
        0.000         1.000
    1       33
             1
      PrmptCCI
        0.000         1.000
    2       33           36
             2            1
      noPrmCCI          LSp
        0.000         1.000
      Otherwise
        1.000         0.000
38 Does Very Late Ignition Occur (BEFORE 24 HRS)?
    2   L2-Ign     noL2-Ign
    2        1            2
    5 Cases
    2       31           31
             1   +        2
     LCF-Rupt  or LCF-Leak
        0.000         1.000
    1       29
             1
        L-Ign
        0.000         1.000
    3       26           34          36
             2   *        1  *        1
       noL-RAC  &    L2RAC  &    L2-SP
        0.990         0.010
    2       34           36
             2   *        2
       noL2RAC  & NoL2-SP
        0.300         0.700
      Otherwise
        0.000         1.000
39 Very Late Containment Failure due to H2 Burning (Before 24Hrs)?
    3 L2CF-Rupt    L2CF-Leak     no-L2CF
    2        1            2           3
    5 Cases
    1       31
             1
      LCF-Rupt
        1.000          0.000       0.000
    1       31
             2
      LCF-Leak
        0.000          1.000       0.000
    1       38
             2
      NoL2-Ign
        0.000          0.000       1.000
    3       11            33          37
             2  *     (    1  +        1   )
          C60p &      ( P-CCI or    D-CCI )
        1.000          0.000       0.000
      Otherwise
        0.000          0.000       1.000
40 Final Containment Sondition and Failure Time?
    6   Leak-I     Rupt-I     Rupt-VB    Rupt-L   Rupt-L2    No-CF
    2        1          2           3         4         5        6
    7 Cases
```

```
1        39
          3
   No-L2CF
     0.000      0.000      0.000       0.000      0.000    1.000
1        25
          2
   ICF-Leak
     1.000      0.000      0.000       0.000      0.000    0.000
2        11          25
          3   *       1
      CP2p   &  ICF-Rupt
     0.000      1.000      0.000       0.000      0.000    0.000
2        11          25
         -3   *       1
    NoCP2p   &  ICF-Rupt
     0.000      0.000      1.000       0.000      0.000    0.000
2        25          31
          3   *       1
    No-ICF   &  LCF-Rupt
     0.000      0.000     0.000        1.000      0.000    0.000
2        31          39
          3   *       1
    No-LCF   & L2CF-Rupt
     0.000      0.000      0.000       0.000      1.000    0.000
   Otherwise
     0.000      0.000      0.000       0.000      0.000    1.000
```

B.3 <u>Characteristics of the Surry Binner at LPSD</u>

The binner is the computer input which instructs EVNTRE how to group the outcomes from evaluating the APET.

These outcomes constitute the interface with the subsequent source term analysis. There are too many outcomes for them all to be saved for analysis afterwards, so as each unique path through the event tree is evaluated, the probability of that path is added to the probability for the appropriate accident progression bin. The term "binner" refers to the set of computer input that defines these bins.

Section 6.3 of this volume gives a general description of the accident progression bins and defines each attribute of each characteristic. That material is not repeated here. The binner itself, a computer input file read by EVNTRE, defines the accident progression bins and is listed in Section B.4. This section of Appendix B contains a case by case description of the binner. Since the format of this binner is designed to match the format requirement of the SURSOR code, some characteristics and attributes which are not specifically applicable to this study is included in the binning. The twelfth characteristic, "Time Window" is not required by the SURSOR code, but added to pass this information to the consequence analysis.

Characteristic 1. CF-Time (Time of Containment Failure)
7 Attributes, 7 Cases

The attributes for this characteristic are:

A. V-Dry — Check valve failures resulted in a pipe break in an interfacing low pressure system. The break location was not underwater at the start of core degradation.

B. V-Wet — Check valve failures resulted in a pipe break in an interfacing low pressure system. The break location was underwater at the start of core degradation.

C. Early-CF — The containment failed before vessel breach. This characteristic represents isolation failures not followed by CF-at-VB.

D. CF-at-VB — The containment failed at the time of vessel breach.

E. L/VL-CF — The containment failed in the late or very late period, including CCI.

F. Final-CF The containment failed during the final period.

G. No-CF The containment did not fail.

This characteristic primarily concerns the time of containment failure. In addition to four time periods in which the containment may fail, there is an attribute for no containment failure.

Case 1: This case defines the conditions for Attribute G, No-CF. For this characteristic, no containment failure is interpreted to mean no failure of the containment pressure boundary itself and no bypass by Event V. If an SGTR had occurred, it would have been included in this case, but no V or SGTR events were observed in the level analysis of this study. The size or type of containment failure is treated in Characteristic 10.

Case 2: This case defines the conditions for Attribute A, V-Dry. This attribute is not applicable to this analysis.

Case 3: This case defines the conditions for Attribute B, V-Wet. This attribute is not applicable to this analysis.

Case 4: This case defines the conditions for Attribute C, Early-CF. Early containment failure here means failure before vessel breach, which includes failure of initial isolation of the containment.

Case 5: This case defines the conditions for Attribute D, CF-at-VB. This is containment failure within a few minutes of vessel breach due to the events accompanying vessel failure (catastrophic rupture, rupture, or leak).

Case 6: This case defines the conditions for Attribute E, Late or Very Late CF. This is containment failure which occurs after VB. It could occur anywhere from a few tens of minutes after VB to several hours after VB. The major cause of this failure is hydrogen burning.

Case 7: This case defines the conditions for Attribute F, Final-CF. This mode of failure do not occur in this study as well as in the full power analysis.

Characteristic 2. Sprays (Operation of Containment Sprays)
8 Attributes, 8 Cases

The attributes for this characteristic are:

A. Sp-Early The sprays operate only in the Early period, that is, before vessel breach.

B. Sp-E+I The sprays operate only in the Early and

Intermediate periods, that is, before vessel breach and immediately after vessel breach.

C. Sp-E+I+L — The sprays operate only in the Early, Intermediate, and Late periods, that is, from UTAF through the initial part of CCI.

D. SpAlways — The sprays Always operate during the periods of interest for fission product removal, that is, for at least 24 hours starting at UTAF.

E. Sp-Late — The sprays operate only in the Late period, that is, during the initial part of CCI.

F. Sp-L+VL — The sprays operate only in the Late and Very Late periods, that is, from the start of CCI through the release of almost all the fission products from CCI.

G. Sp-VL — The sprays operate only in the Very Late period, that is, during the latter part of CCI.

H. Sp-Never — The sprays Never operate during the accident.

This characteristic concerns the operation of the containment sprays. Spray operation implies containment heat removal and radionuclide scrubbing.

Case 1: This case defines the conditions for Attribute A, Sp-Early. In this case, the sprays operate only in the period before vessel breach.

Case 2: This case defines the conditions for Attribute B, Sp-E+I. In this case, the sprays operate only before and at vessel breach.

Case 3: This case defines the conditions for Attribute C, Sp-E+I+L. In this case, the sprays operate only from the start of the accident through the initial part of CCI.

Case 4: This case defines the conditions for Attribute D, SpAlways. In this case, the sprays operate continuously from UTAF for at least 24 hours.

Case 5: This case defines the conditions for Attribute E, Sp-Late. In this case, the sprays operate only during the initial part of CCI.

Case 6: This case defines the conditions for Attribute F, Sp-L+VL. In this case, the sprays operate only in the Late and Very Late periods, that is, from the start of CCI through the release of almost all the fission products from CCI.

Case 7: This case defines the conditions for Attribute G, Sp-VL. In this case, the sprays operate only in during the latter part of CCI, which follows a hydrogen burn (if any).

Case 8: This case defines the conditions for Attribute H, Sp-Never. In this case, the containment sprays do not operate at all when they could contribute to fission product removal.

Characteristic 3. CCI (Core-Concrete Interactions)
6 Attributes, 6 Cases

The attributes for this characteristic are:

A.	Prom-Dry	CCI takes place promptly following vessel breach in a dry cavity. There is no overlying water pool to scrub the releases.
B.	PromShlw	CCI takes place promptly following vessel breach. The accumulators dump at vessel breach, so when CCI starts there is about 4.5 feet of water in the cavity.
C.	No-CCI	CCI does not take place.
D.	PromDeep	CCI takes place promptly following vessel breach. The cavity is full of water at this time; the pool is about 14 feet deep.
E.	SDlyd-Dry	CCI takes place after a short delay, in a dry cavity. The debris bed is coolable, but the water in the cavity is not replenished.
F.	LDlyd-Dry	CCI takes place after a long delay, in a dry cavity. The debris bed is coolable, but the water in the cavity is not replenished. The delay is the time needed to boil off the water in a full cavity.

This characteristic concerns the core-concrete interaction; if it takes place, when it takes place, and whether there is overlying pool of water to scrub the fission products released from the CCI.

Case 1: This case defines the conditions for Attribute A, Prom-Dry. CCI takes place promptly following vessel breach in a dry cavity. As there is no water in the cavity after VB, whether the debris bed is coolable is not relevant.

Case 2: This case defines the conditions for Attribute B, PromShlw. CCI takes place promptly following vessel breach. The cavity was dry just before vessel failure, but the accumulators discharge at vessel breach. Since there is water, the debris bed must be non-coolable.

Case 3: This case defines the conditions for Attribute C, No-CCI. If neither prompt CCI nor delayed CCI takes place, there is no CCI. Either there was no vessel breach, or the debris is coolable, water was present at VB, and the water supply is continuously replenished by the containment sprays.

Case 4: This case defines the conditions for Attribute D, PromDeep. CCI takes place promptly following vessel breach, and the cavity is full of water when CCI commences.

Case 5: This case defines the conditions for Attribute E, SDlyd-Dry. CCI takes place after a short delay. The debris bed is initially coolable, and the cavity contains the accumulator water (only). This caseir not applicable to this analysis.

Case 6: This case defines the conditions for Attribute F, LDlyd-Dry. CCI takes place after a long delay. The debris bed is initially coolable, and the cavity is full of water at vessel breach. After all the water is boiled away, CCI commences in a dry cavity.

Characteristic 4. RCS-Pres (RCS Pressure before Vessel Breach)
4 Attributes, 4 Cases

The attributes for this characteristic are:

A. SSPr Just before vessel breach, the RCS is at system setpoint pressure, about 2500 psia. This pressure is determined by the setpoint of the PORVs. This attribute is not applicable to this analysis.

B. HiPr Just before vessel breach, the RCS is in the range denoted high pressure. The hole in the RCS pressure boundary is small enough that the pressure spike that follows core slump decays away relatively slowly. The pressure at vessel breach can range from 1000 to 2000 psia. This attribute is not applicable to this analysis.

C. ImPr Just before vessel breach, the RCS is in the range denoted intermediate pressure. The hole in the RCS is larger than for Attribute B, so the pressure at breach is within the range of 500 to 1000 psia.

D. LoPr Just before vessel breach, the RCS is at low pressure, less than 500 psia.

This characteristic determines the pressure in the reactor coolant systemjust before the failure of the vessel. This pressure, together with the mode of vessel breach, Characteristic

5, largely determines the events thattake place in the containment immediately following vessel breach. In mostdetailed, mechanistic analyses of core degradation, vessel failure followsthe relocation or slumping of many tons of molten core material into thelower head of the vessel. The lower head usually contains some water atthis time, so the core slump generates a large amount of steam. This willincrease the vessel pressure, at least temporarily, if the RCS was belowthe PORV setpoint pressure at the time of the slump. The pressure at VBdepends upon how fast the RCS pressure decreases after core slump and thedelay between core slump and vessel failure.

Case 1: This case defines the conditions for Attribute A, SSPr. The RCS is at system setpoint pressure, about 2500 psia, when the vessel fails.

Case 2: This case defines the conditions for Attribute B, HiPr. The RCS is in the range denoted high pressure, 1000 to 2000 psia, when the vessel fails.

Case 3: This case defines the conditions for Attribute C, ImPr. The RCS is in the range denoted intermediate pressure, 500 to 1000 psia, when the vessel fails.

Case 4: This case defines the conditions for Attribute D, LoPr. The RCS is at low pressure, less than 500 psia, when the vessel fails.

Characteristic 5. VB-Mode (Mode of Vessel Breach)
6 Attributes, 6 Cases

The attributes for this characteristic are:

A. VB-HPME — Vessel breach occurs when one or more penetration(s) fails and the vessel is above 500 psia. These conditions ensure High Pressure Melt Ejection.

B. VB-Pour — Molten core material Pours out of the vessel at breach, driven primarily by the effects of gravity.

C. VB-BtmHd — Either there is a circumferential failure of the Bottom Head, or a large portion of the Bottom Head of the vessel fails.

D. Alpha — An Alpha mode failure occurs - resulting in containment failure as well as vessel failure.

E. Rocket — A Rocket mode failure occurs - resulting in containment failure as well as vessel failure.

F. No-VB No Vessel Breach occurs.

This characteristic determines the mode of vessel failure. The mode ofvessel failure and the pressure in the reactor coolant system just beforethe failure of the vessel, Characteristic 4, largely determine the eventsthat take place in the containment immediately following vessel breach. In two of the failure modes, the failure of the vessel directly causes thefailure of the containment as well.

Case 1: This case defines the conditions for Attribute A, VB-HPME. High Pressure Melt Ejection results when one or more penetration(s) fails and the vessel is above 200 psia.

Case 2: This case defines the conditions for Attribute B, VB-Pour. The molten core Pours out of the vessel, driven primarily by the effects of gravity. This mode of vessel failure always occurs if the vessel is at low pressure when it fails. It can also occur when the vessel is at higher pressures if the gases in the vessel escape before an appreciable amount of molten core material leaves the vessel.

Case 3: This case defines the conditions for Attribute E, Rocket. If the bottom head of the vessel fails and the vessel is at very high pressure, it is conceivable that the entire vessel could be propelled upward and somehow fail the containment. As the Rocket failure mode requires that the bottom head failure mode occur, either this case has to be placed here, before the BtmHd case, or the BtmHd case has to specify that no Rocket failure occurs.

Case 4: This case defines the conditions for Attribute C, VB-BtmHd. The vessel failure involves a substantial part of the Bottom Head.

Case 5: This case defines the conditions for Attribute D, Alpha. Alpha mode failure is defined to be a steam explosion in the vessel that fails the vessel and also results in containment failure.

Case 6: This case defines the conditions for Attribute F, No-VB. Core damage was arrested before vessel breach.

Characteristic 6. SGTR (Steam Generator Tube Rupture)
3 Attributes, 3 Cases

This characteristic is not applicable to this analysis and is included only to match the SURSOR code requirement. A dummy attribute is taken for all cases.

Characteristic 7. Amt-CCI (Amount of Core not in HPME available for CCI)
4 Attributes, 4 Cases

The attributes for this characteristic are:

A. Lrg-CCI A Large amount of the Core (70-100%) not in HPME participates in the Core-Concrete Interaction.

B. Med-CCI A Medium amount of the Core (30-70%) not in HPME participates in the Core-Concrete Interaction.

C. Sml-CCI A Small amount of the Core (0-30%) not in HPME participates in the Core-Concrete Interaction.

D. No-CCI There is no Core-Concrete Interaction.

This characteristic determines how much of the core that is not in HPME that participates in the core-concrete interaction. Whether the CCI occurs at all, and the timing and the conditions of the CCI, are determined in Characteristic 3. The selection of one of the first three attributes in this characteristic implies that CCI occurs. The definition of this binning characteristic is different from the definition used in the APET itself. In the APET, the amount of core in CCI was the amount of the total core available to participate in CCI, without respect to whether HPME had occurred. This value was used in determining the amount of hydrogen pro- duced during CCI and the likelihood of basemat melt-through. The primary use of this binning characteristic is to pass information on to ZISOR for the source term analysis. SURSOR internally subtracts out the amount of core involved in HPME from the amount passed to it in this characteristic. (The fraction of the core involved in HPME is determined by Characteristic 9.) Therefore, in the binner it is necessary to define this characteristic as the amount of the core not involved in HPME that takes part in the core- concrete interaction. Otherwise, the amount of the core participating in CCI would be subtracted twice.

Case 1: This case defines the conditions for Attribute D, No-CCI. If there is no prompt CCI, and there is no delayed CCI, there is no Core-Concrete Interaction.

Case 2: This case defines the conditions for Attribute A, Lrg-CCI. Either a Large amount of the Core (70-100%) was determined to be available for CCI in the APET, or HPME occurred. In SURSOR, the fraction of the core involved in HPME will be subtracted from the total amount of core material. Setting Characteristic 7 to Large here ensures that a large fraction of the core not involved in HPME is available for CCI. HPME is meant to include all the events in which core material leaves the vessel first under high gas pressure, followed by blowdown of the gas. The PrEj case in the APET includes only those cases where the hole in the vessel involves only a small fraction of the area of the bottom head. Thus the situation where the bottom head fails at any pressure above a few hundred psia has to be specifically included.

Case 3: This case defines the conditions for Attribute B, Med-CCI. A Medium amount of the Core (30-70%) was determined to be available for CCI in the APET.

Case 4: This case defines the conditions for Attribute C, Sml-CCI. A Small amount of the Core (0-30%) was determined to be available for CCI in the APET.

Characteristic 8. Zr-Ox (Zr Oxidation in-vessel)
2 Attributes, 2 Cases

This characteristic is not applicable to this analysis and is included only to match the SURSOR code requirement. A dummy attribute is taken for all cases.

Characteristic 9. HPME (High Pressure Melt Ejection)
4 Attributes, 4 Cases

This characteristic is considered to be not important for this analysis and is included only to match the SURSOR code requirement. A dummy attribute is taken for all cases.

Characteristic 10. CF-Size (Containment Failure Size or Type)
4 Attributes, 4 Cases

The attributes for this characteristic are:

A. Cat-Rupt — The containment failed by catastrophic rupture, resulting in a very large hole and gross structural failure.

B. Rupture — The containment failed by the development of a large hole or rupture; nominal hole size is 7 square feet.

C. Leak — The containment failed by the development of a small hole or a leak; nominal hole size is 0.10 square foot.

D. No-CF — The containment did not fail.

This characteristic determines how the containment failed. The first three attributes define the hole size if the containment pressure boundary failed above ground. The fourth attribute indicates that the pressure boundary did not fail.

Case 1: This case defines the conditions for Attribute A, Cat-Rupt. The containment failed by catastrophic rupture or major structural failure. This can occur at vessel breach or due to a hydrogen burn after VB.

Case 2: This case defines the conditions for Attribute B, Rupture. The containment failed by the development of a large hole, denoted rupture in this analysis. This can occur at vessel breach or due to a hydrogen burn after VB.

Case 3: This case defines the conditions for Attribute C, Leak. The containment failed by the development of a small hole, denoted a leak in this analysis. This case includes situations with an isolation failure and core damage arrest before vessel breach.

Case 4: This case defines the conditions for Attribute D, No-CF. The containment did not fail above ground or below ground, and it was not bypassed.

Characteristic 11. RCS-Hole (Number of large holes in the RCS)
? Attributes, 2 Cases

This characteristic is considered to be not important for this analysis and is included only to match the SURSOR code requirement. A dummy attribute is taken for all cases.

Characteristic 12. Time Window
4 Attribute, 4 Cases

The attributes for this characteristic are:

1. Win-1: The core damage accident was initiated in Time Window 1.

1. Win-2: The core damage accident was initiated in Time Window 2.

1. Win-3: The core damage accident was initiated in Time Window 3.

1. Win-4: The core damage accident was initiated in Time Window 4.

B.4 <u>Lising of the Binner for the Surry Shutdown Risk Study</u>

```
Surry LP Binning - Rev. 1 - Sep 23, 93 - 12 Characteristics
   12 CF-Time  Sprays  CCI  RCS-Pres  VB-Mode  SGTR  Amt-CCI
      Zr-Ox  HPME  CF-Size  RCS-Hole  WINDOW
    7   7  V-Dry V-Wet Early-CF CF-at-VB L/VLate-CF Final-CF No-CF
    1   7        40
                  6
               noCF
    1   1        30
                  2
               zero
    1   2        30
                  2
               zero
    2   3        40           40
                  1 +          2
             Leak-I or    Rupt-I
    1   4        40
                  3
            Rupt-VB
    2   5        40           40
                  4 +          5
             Rupt-L       Rupt-VL
    1   6        30
                  2
               zero
    8   8  Sp-Early  Sp-E+I  Sp-E+I+L  SpAlways  Sp-Late  Sp-L+VL
            Sp-VL  Sp-Never
    3   1        18           28           36
                  1   *      -1 *         -1
               E-Sp   &   noLSP &     noL2Sp
    1   2        30
                  2
               zero
    3   3        18           28           36
                  1   *       1 *         -1
               E-Sp   &     LSp &     noL2Sp
    3   4        18           28           36
                  1   *       1  *         1
               E-Sp   &     LSp  &      L2SP
    3   5        18           28           36
                 -1   *       1  *        -1
             noE-Sp   &     LSp  &    noL2SP
    3   6        18           28           36
                 -1   *       1  *         1
             noE-SP   &     LSP  &      L2SP
    3   7        18           28           36
                 -1   *      -1    *       1
             noE-SP   &  noL-SP    &     L2-SP
    1   8        36
                  2
```

```
          No-L2SP
6    6   Promt-Dry   PromtShlw   No-CCI PromtDeep   SDlyd-Dry
         LDlyd-Dry
2    1        33            19
               1   *         2
         PrmptCCI      RC-Dry
1    2        30
               2
            zero
2    3        33            37
               2    *        2
         noPrmCCI   & noDldCCI
2    4        33            19
               1    *        1
         PrmptCCI   &  RC-Wet
1    6        30
               2
            zero
1    5        37
               1
         DelydCCI
4    4   SSPr  HiPr  ImPr  LoPr
1    1        30
               2
            zero/I-SSPr
1    2        30
               2
            zero/I-HiPr
1    3        15
               1
           I-ImPr
1    4        15
               2
           I-LoPr
6    6   VB-HPME  VB-Pour  VB-BtmHd  Alpha  Rocket  No-VB
1    1        17
               1
            PrEj
1    2        17
               2
            Pour
1    3        17
               3
           BtmHd
1    4        17
               5
           Alpha
1    5        30
               2
            zero/Rocket
1    6        17
               4
            noVB
3    3   SGTR  SGTR-SRVO  No-SGTR
```

```
1    1        30
               2
            zero/B-SGTR
1    2        30
               2
            zero/B-SGTR
1    3        30
               1
             one/noB-SGTR
4    4    Lrg-CCI  Med-CCI  Sml-CCI  No-CCI
1    3        30
               2
            zero
3    4        17               33           37
               4    +    (      2    *        2 )
          NO-VB    or   (NoPrmCCI  &  noDldCCI)
3    2        17               17           17
               5    +           1    +        3
            ALPHA  or        PrEj  or     BtmHd
1    1        17
               2
            pour
2    2    Lo-ZrOx  Hi-ZrOx
1    1        30
               2
            zero
1    2        30
               1
             one
4    4    Hi-HPME  Md-HPME  Lo-HPME  No-HPME
1    1        30
               2
            zero
2    2        17          17
               1 +         3
            PrEj or    BtmHd
1    3        30
               2
            zero
3    4        17          17              17
               2 +         4 +             5
            Pour or     noVB or       ALPHA
4    4    Cat-Rupt  Rupture  Leak   No-CF
3    1        40          40           40
               3   +       4   +        5
          Rupt-VB  or  Rupt-L  or Rupt-L2
1    2        40
               2
          Rupt-I
1    3        40
               1
          Leak-I
1    4        40
               6
```

```
          noCF
2   2  1-Hole  2-Holes
1   1       30
             1
           one
1   2       30
             2
          zero
4   4     WIN1       WIN2      WIN3       WIN4
1   1        1
             1
        WIN-1
1   2        1
             2
        WIN-2
1   3        1
             3
        WIN-3
1   4        1
             4
        WIN-4
```

APPENDIX C

SUPPORTING INFORMATION FOR THE

SOURCE TERM ANALYSIS

CONTENTS

LIST OF FIGURES

Appendix C: Source Term Analysis

C.1 Introduction

Source term estimation for a low power/shutdown (LP/SD) PRA was first investigated in an earlier abridged low power/shutdown PRA study for Surry[C1,C2]. It was decided in the abridged study that the source terms should address uncertainty and wherever possible, the NUREG-1150 distributions for ST definition would be used to calculate the source terms from an accident during mid-loop operation. The parametric code, SURSOR[C3], that was developed in NUREG-1150 for Surry, was used to define the source terms in the abridged study. Two measures were taken to assure the adequacy of the source terms: The first involved comparing the calculations from MELCOR with the data used in and the results obtained from SURSOR. Second, an advisory group called the Source Term Advisory Group was established to provide guidance, and any additional information on modifying the SURSOR code for the LP/SD study.

Based on the results of the previous abridged study, the SURSOR code was used in the present study to predict the source terms for the accident progression bins obtained in the accident progression (Level 2) analysis. Since it is not practical to perform consequence calculations (Level 3 analysis) for all source terms obtained in the present analysis (about 15,000), source terms were grouped according to their health effects. This source term partitioning process reduces the number of MACCS calculations required for the Level 3 analysis. The methods used for source term definition and partitioning are described in the following sections. Some of the source term calculational results are also provided in this Appendix for information.

C.2 Source Term Definition

C.2.1 Description of Parametric Model

The SURSOR code, together with its associated distributions from NUREG-1150, was selected as the basis for ST definition. This section provides a brief discussion of the SURSOR code, its evaluation (for modification, if required), and the final parametric model used.

SURSOR is a parametric computer code used in NUREG-1150 to predict source terms for full power operation. Table C.1 lists the parameters used in the SURSOR code, which were defined in NUREG-1150 by expert elicitation. A distribution, instead of a single value, was assigned to each parameter to address uncertainty. Considering the differences between full power and shutdown operations, the Source Term Advisory Group, identified two parameters in SURSOR as important and possibly different than the values used in NUREG-1150. The first parameter is the fraction of the fission products in the core that are released to the vessel before vessel breach (i.e., FCOR). The second parameter is the fraction of the fission products released to the vessel that are subsequently released to the containment (i.e., FVES). The distributions of these two parameters as defined in NUREG-1150 were compared with MELCOR calculations to establish their values to be used in the low power/shutdown study.

The MELCOR model for Surry, which is a three-loop Westinghouse design, includes eight control volumes for the primary reactor coolant system and seven control volumes for the containment. The core is nodalized into 39 cells. The three radial rings are selected according to the arrangement for fuel enrichment. The axial nodalization is divided into 13 sections, with 10 sections for the core region, one section for the lower core plate, and two sections for the lower plenum. There are 126 heat structures in the model, which provide heat sinks as well as deposition areas for fission products. The flat-bottom cylindrical geometry with a limestone/common sand concrete type is used in the cavity package. The number of rays used in the CORCON system is 75: 10 at the bottom, 10 at the corner, and 55 at the cavity wall. The fuel dispersal interaction package, which is used to model low pressure molten fuel ejection from the reactor vessel to the cavity, is included in the model.

Since the primary system pressure is most likely to be low during a low power/shutdown accident sequence, the primary system pressure is assumed in the MELCOR calculation to be at 1 atmosphere with the water level at the top of the reactor core. The containment condition at the beginning of the accident is based on the most significant containment failure mode predicted by the accident progression analysis. The containment is assumed to have a 1 ft^2 leakage area to the environment at an elevation of 16.4 ft above the lower compartment floor. The RHR system is assumed to be failed at the beginning of the accident, and primary coolant injection and secondary heat removal are assumed not available. The accident initiation time from reactor shutdown, one of the most important parameters distinguishing full power and shutdown conditions, varies from 24 hours to 240 hours in the MELCOR calculations. The decay heat power for the various accident initiation time is based on the ANS standard for light water reactor (ANS-5.1-1979) with a two-year reactor operation period and 80% capacity factor.

Figures C.1(a) and C.1(b) give the distributions (the range and the median value) of FCOR (fraction of fuel inventory released to vessel) and FVES (material leaving the RCS as a fraction of those which are entering the RCS) used in SURSOR, and the calculated values from MELCOR for three cases. As shown in Figure C.1, the difference between the MELCOR calculated values for the three cases with different accident initiation times (24, 72, and 240 hours after reactor shutdown) is not significant. Figure C.2(a) and C.2(b) present the results of two MELCOR calculations with same accident initiation time but with different core damage status. In one calculation the core continues to melt until the reactor vessel is breached. In another calculation core injection is recovered at 175 minutes, when 45% of the core has been relocated to the lower plenum. Core melt is arrested after core injection recovery and the vessel integrity is maintained. As shown in Figure C.2 the values predicted for the core recovery case are less than those predicted for the MELCOR cases that proceeds to vessel breach. Certainly, the release fractions for the core-recovery case would depend on the time of recovery.

In addition to the above comparison, the environmental release of fission products obtained from SURSOR and MELCOR are also compared. In SURSOR, a source term is uniquely defined by the Accident Progression Bin (APB) using eleven characteristics. Table C.2 presents the APB characteristics used in SURSOR and their attributes. Table C.3 shows the APBs selected for comparison and the attributes for the 11 characteristics. These are the APBs obtained from the accident progression event tree (APET) analysis in the abridged study and they remain important in the present study. The attributes of the characteristics are defined based on APET analysis, the LP/S conditions or are not important for LP/S conditions.

In the SURSOR calculations, two hundred s... (or observations) of release fractions were produced for each of the five bins presented in Table C.3 to address source term uncertainty. Figures C.3(a) through (d) present the ranges (5 percentile to 95 percentile) of the release fractions of the nine radionuclide categories for APBs 1 through 4, respectively. These figures show the median (50 percentile) and mean values of the release fractions from the 200 observations, and the calculations from the MELCOR cases that are related to the individual APBs.

The above figures show that generally, the MEL values fall within the ranges of SURSOR predictions. Although, for some radionuclide categories, the MELCOR calculated values are closer to the upper ranges of the SURSOR predictions, there are no apparent phenomenological reasons to modify the SURSOR distributions. Consequently, the Source Term Advisory Group did not recommend any change to the SURSOR code for ST predictions.

C.2.2 Source Term Results

In the present study, the SURSOR code was used to predict the source terms for all the APBs in each of the 100 observations. There are about 150 APBs in each observation. Although the total number of APBs (for the 100 observations) is 15,443, the number of APBs with different characteristics is only 360. Although same APBs (i.e., APBs with the same characteristics) occur in different observations, the source terms of the APBs in different observations are different because the LHS values vary from observation to observation. Table C.4 lists the 360 APBs, their characteristics (represented by their attribute strings), the frequency weighted mean release fractions (of the 100 observations) for the nine radionuclide groups, and their mean frequencies. The exceedance frequencies for the release fractions of eight of the nine radionuclide groups are presented in Figure C.4.

C.3 Source Term Partition

C.3 Source Term Partition

The accident progression and source term analyses resulted in a total of 15,443 source terms for internally initiated accidents during mid-loop operation. It is computationally impractical to carry out a consequence calculation for each source term to obtain the integrated risk for the selected consequence measures. To create an interface between the source term analysis and the consequence calculation, the total number of source terms are grouped into a much smaller number of source term groups. The groups are created such that the source terms within each group have similar properties with respect to consequences, i.e., their potential for causing early fatalities and latent cancer fatalities is similar. A frequency weighted mean source term is determined for each group and the consequence calculations are performed for the mean source term in four time windows. This Appendix contains some details pertinent to the source term partitioning procedures described in Chapter 7 of this report.

The four core radionuclide inventories which were used in the partitioning consequence calculations for the four time windows are shown in Table ~~7.4old(C.?...)~~ C.5. These inventories were obtained by interpolation from the data published in Reference ~~4~~ C5 on the core inventories at Surry as a function of time after reactor shutdown.

Table ~~C....(7.4old)~~ C.5 Isotope~~s~~ Inventories for Four Time Windows (Bq)

Radionuclide	Window 1 48 hr	Window 2 120 hr	Window 3 288 hr	Window 4 768 hr	MACCS Group
KR-88	1.6E+13	3.75E+05	0.00E+00	0.00E+00	1
KR-87	6.2E+06	0.00E+00	0.00E+00	0.00E+00	1
XE-133	4.8E+18	3.36E+18	1.76E+18	9.68E+16	1
XE-135	2.9E+17	1.48E+15	0.00E+00	0.00E+00	1
KR-85	1.8E+16	1.73E+16	1.73E+16	1.72E+16	1
KR-85M	4.4E+14	6.41E+09	0.00E+00	0.00E+00	1
I-132	2.5E+18	1.33E+18	4.60E+17	4.27E+15	2
I-131	2.3E+18	1.76E+18	1.14E+18	1.72E+17	2
I-133	1.1E+18	1.02E+17	1.86E+15	4.26E+07	2
I-134	7.7E+02	0.00E+00	0.00E+00	0.00E+00	2
I-135	3.3E+16	1.73E+13	0.00E+00	0.00E+00	2
CS-136	8.1E+16	6.90E+16	5.29E+16	1.65E+16	3
CS-137	1.8E+17	1.81E+17	1.81E+17	1.80E+17	3
CS-134	2.1E+17	2.12E+17	2.11E+17	2.07E+17	3
RB-86	3.1E+15	2.82E+15	2.34E+15	1.03E+15	3
TE-127	2.2E+17	1.42E+17	7.86E+16	2.78E+16	4
TE-127M	3.2E+16	3.21E+16	3.14E+16	2.75E+16	4
SB-127	2.0E+17	1.14E+17	4.63E+16	8.81E+14	4

Radionuclide	Window 1 48 hr	Window 2 120 hr	Window 3 288 hr	Window 4 768 hr	MACCS Group
SB-129	3.8E+14	3.68E+09	0.00E+00	0.00E+00	4
TE-131M	1.3E+17	2.37E+16	1.48E+15	7.47E+09	4
TE-132	2.4E+18	1.29E+18	4.47E+17	4.14E+15	4
TE-129	7.7E+16	7.23E+16	6.52E+16	4.14E+16	4
TE-129M	1.2E+17	1.11E+17	1.00E+17	6.36E+16	4
SR-90	1.3E+17	1.34E+17	1.34E+17	1.34E+17	5
SR-91	1.0E+17	5.26E+14	0.00E+00	0.00E+00	5
SR-92	1.7E+13	1.68E+05	0.00E+00	0.00E+00	5
SR-89	2.6E+18	2.54E+18	2.37E+18	1.75E+18	5
CO-58	2.26E+16	2.19E+16	2.06E+16	1.69E+16	6
CO-60	1.76E+16	1.76E+16	1.75E+16	1.74E+16	6
RU-103	3.8E+18	3.57E+18	3.27E+18	2.22E+18	6
TC-99M	2.8E+18	1.32E+18	3.74E+17	1.46E+15	6
MO-99	2.9E+18	1.37E+18	3.88E+17	1.52E+15	6
RU-105	1.5E+15	1.94E+10	0.00E+00	0.00E+00	6
RH-105	1.1E+18	2.60E+17	2.48E+16	7.92E+11	6
RU-106	9.2E+17	9.13E+17	9.04E+17	8.68E+17	6
LA-140	4.6E+18	4.03E+18	3.12E+18	9.53E+17	7
AM-241	2.90E+14	2.90E+14	2.90E+14	2.90E+14	7
CM-242	3.14E+16	3.11E+16	3.01E+16	2.77E+16	7
CM-244	1.86E+15	1.86E+15	1.85E+15	1.85E+15	7
ND-147	1.6E+18	1.29E+18	9.44E+17	2.38E+17	7
LA-141	1.0E+15	3.11E+09	0.00E+00	0.00E+00	7
LA-142	2.2E+09	0.00E+00	0.00E+00	0.00E+00	7
PR-143	4.0E+18	3.50E+18	2.74E+18	8.94E+17	7
Y-93	1.5E+17	1.11E+15	0.00E+00	0.00E+00	7
Y-92	1.2E+15	9.48E+08	0.00E+00	0.00E+00	7
ZR-95	4.4E+18	4.25E+18	4.02E+18	3.17E+18	7
NB-95	4.4E+18	4.36E+18	4.33E+18	4.06E+18	7
ZR-97	6.2E+17	3.26E+16	2.37E+14	9.34E+04	7
Y-91	3.4E+18	3.27E+18	3.08E+18	2.37E+18	7
Y-90	1.4E+17	1.36E+17	1.35E+17	1.34E+17	7

Radionuclide	Window 1 48 hr	Window 2 120 hr	Window 3 288 hr	Window 4 768 hr	MACCS Group
PU-239	7.1E+14	7.11E+14	7.14E+14	7.15E+14	8
PU-241	2.0E+17	2.04E+17	2.04E+17	2.04E+17	8
PU-240	8.9E+14	8.94E+14	8.94E+14	8.94E+14	8
PU-238	2.8E+15	2.77E+15	2.78E+15	2.81E+15	8
CE-144	2.6E+18	2.56E+18	2.53E+18	2.39E+18	8
CE-143	1.5E+18	3.36E+17	2.70E+16	4.13E+11	8
NP-239	2.9E+19	1.20E+19	2.76E+18	4.27E+15	8
CE-141	4.3E+18	4.07E+18	3.66E+18	2.29E+18	8
BA-140	4.2E+18	3.58E+18	2.73E+18	8.28E+17	9
BA-139	1.8E+08	0.00E+00	0.00E+00	0.00E+00	9

The partitioning procedures described below consist of defining an early health effect weight, EH, and a latent health effect weight, LH, for each source term and grouping the source terms based on these weights.

Early Fatality Health Weight,

C.3.1 Calculation of EH ~~Weight e~~

The early health effect weight was calculated by converting the radionuclide releases associated with a range of source terms into equivalent I-131 releases. Surry site-specific consequence calculations of the early health effects were performed in each of the time windows used in the analysis and the results were presented as a function of equivalent I-131 release. This correlation of the estimated number of early health effects is the EH weight which was used in partitioning of all source terms.

The relationship between the early health effect weight and the equivalent I-131 release is shown in Figure C.8. This relationship was produced by following the methodology used in the PARTITION code, Reference 6. Table C.6 shows the equivalent I-131 inventories in four time windows for a range of source terms used to predict the EH weight.

Table C.6 Equivalent I-131 Inventory for Four Time Windows (Bq)

Window 1	Window 2	Window 3	Window 4
3.55e+15	1.85e+15	9.50e+14	6.59e+13
1.85e+15	1.02e+17	5.27e+16	3.48e+15
9.50e+14	5.27e+16	5.85e+16	4.19e+15
6.59e+13	3.48e+15	4.19e+15	4.96e+15
2.59e+17	1.35e+17	6.90e+16	5.94e+15
3.08e+17	1.61e+17	8.23e+16	8.24e+15

Window 1	Window 2	Window 3	Window 4
2.90e+17	1.53e+17	7.86e+16	9.51e+15
5.08e+17	2.66e+17	1.34e+17	1.53e+16
4.36e+17	2.33e+17	1.21e+17	1.96e+16
2.64e+17	1.46e+17	8.06e+16	2.15e+16
3.59e+17	1.99e+17	1.11e+17	3.04e+16
5.27e+17	2.97e+17	1.68e+17	5.27e+16
6.23e+17	3.55e+17	2.03e+17	6.84e+16
9.92e+17	5.73e+17	3.26e+17	1.15e+17
1.22e+18	7.07e+17	4.06e+17	1.50e+17
1.75e+18	1.02e+18	5.94e+17	2.33e+17
2.88e+18	1.71e+18	9.87e+17	3.98e+17
3.69e+18	2.32e+18	1.46e+18	6.68e+17
5.62e+18	3.44e+18	2.08e+18	9.05e+17
7.34e+18	4.48e+18	2.72e+18	1.25e+18
1.17e+19	7.53e+18	4.82e+18	2.31e+18
1.76e+19	1.21e+19	8.25e+18	4.14e+18
2.62e+19	1.88e+19	1.35e+19	7.03e+18
2.91e+19	2.09e+19	1.49e+19	7.80e+18
4.53e+19	3.04e+19	2.05e+19	1.04e+19

Using the data displayed in Fig. C.~~19~~ 8 for the early health effects, 213 source terms with EH > 0 and LH > 0 have been identified and grouped into four cells as shown in section C.3.3 below.

C.3.2 Calculation of Latent Cancer Fatality Health Weight, LH ~~Weight~~

The latent health effect weight, LH, was calculated by assuming a linear relationship between the number of latent cancer fatalities due to a particular radionuclide and the amount of release of that radionuclide.

The average release fractions for the nine MACCS groups (averaged over all 15,433 source terms generated by SURSOR) were calculated to be used in the LH weight consequence calculations. Thirty six consequence calculations were then performed, nine for each of the four time windows. Each of the nine calculations was run with only one of the nine radionuclide groups present in the source term while the release fractions of the remaining eight groups were set to zero. The results of the thirty six calculations for number of latent health effects are presented in Table ...~~7.5old~~ C.7 (LH_{iw}, where i = 1,9 is the number of radionuclide groups and w = 1,4 is the number of time windows). The last column in Table ...~~7.5old~~ C.7 contains the release fractions of the nine MACCS radionuclide groups averaged over all 15,433 source terms, RF^s_i.

C.7

Table ...~~7.5old~~ Numbers of Latent Health Effects Predicted in the LH Weight Calculations

MACCS Radionuclide Group	Window 1	Window 2	Window 3	Window 4	Average Source Term
Xe	3.98E-01	2.23E-01	1.17E-01	6.53E-03	0.77
I	3.99E+01	2.97E+01	2.02E+01	3.74E+00	0.165
Cs	6.04E+02	6.03E+02	6.02E+02	6.02E+02	0.133
Te	4.37E+01	2.44E+01	9.84E+00	1.34E+00	0.075
Sr	3.57E+01	3.53E+01	3.46E+01	3.30E+01	0.02
Ru	2.96E+01	2.88E+01	2.79E+01	2.53E+01	0.004
La	3.01E+01	2.92E+01	2.78E+01	2.31E+01	0.002
Ce	1.21E+02	1.19E+02	1.17E+02	1.14E+02	0.006
Ba	5.64E+01	4.91E+01	3.89E+01	1.46E+01	0.02

C.7

The four corresponding values of latent health effects (LH) predicted by using the complete source terms ("complete" in the present context means that all radionuclide groups are non-zero) for four windows are (LH^c_w): 6.79E+02, 6.56E+02, 6.36E+02, and 6.25E+02, respectively. Summation of $LH_{i,w}$ over nine groups for four windows in Table ...~~7.5old~~ produces LH^s_w for four windows (superscript "s" stands for "separate', i.e., calculation with only one non-zero radionuclide group): 9.61E+02, 9.19E+02, 8.78E+02, and 8.17E+02, respectively. An adjustment factor AF_w is calculated as LH^c_w/LH^s_w (7.07E-01, 7.14E-01, 7.24E-01, and 7.65E-01) to be used in the final steps of the partitioning procedure.

The adjustment factor AF_w is needed to take into account the effects of the counter-measures: population is relocated based on the projected individual dose level. These dose projections are made using combined effect of all radionuclide groups. Therefore, the collective dose (and latent health effects) predicted for a "complete" source term will be lower than a sum of doses predicted in "separate" radionuclide group calculations because of a more extensive relocation of the population.

In addition, in order to account for the dependency of the health consequences on the decay of the core inventory, a window factor, WF_w defined as a ratio of the inventory of a particular MACCS isotope group in any window to the corresponding inventory for window 1 was also calculated and it is shown in Table ...7.6old.

C.8

Table ~~7.6old~~ Windows Inventories Relative to Inventory of Window 1

	Window 1	Window 2	Window 3	Window 4
Xe	1	0.664	0.349	0.022
I	1	0.538	0.271	0.030
Cs	1	0.973	0.936	0.848
Te	1	0.556	0.239	0.051
Sr	1	0.928	0.870	0.655
Ru	1	0.648	0.432	0.269
La	1	0.901	0.794	0.510
Ce	1	0.509	0.243	0.130
Ba	1	0.851	0.648	0.197

Using the information presented in Tables C.7 ~~7.5old~~ and C.8 ~~7.6old~~, the number of latent health effects for all 15,443 source terms (i.e., LH_j, where j = 1 to 15,443) can be estimated using the following correlation:

$$LH_j = \sum_i^9 \left(\frac{LF_{iw}}{RF_i^a} \right) \times RF_i \times WF_w \times AF_w \tag{7.1}$$

where RF_i is the release fraction of the MACCS isotope group i and LH_w is the corresponding latent health effect corresponding to the average source term RF_i^a for the same radionuclide group i. Use of the window factor, WF_w is based on the assumption that the number of latent health effects is proportional to the magnitude of release.

Finally, 15,230 source terms with LH > 0 and EH = 0 were grouped into twenty one cells assuming a Max/Min ratio of 1.5 for each cell as discussed in section C.3.3 below.

C.3.3 Results of Source Terms Partitioning

Using the data displayed in Fig. C.8 ~~10~~ for early fatalities and the correlation of section C.3.2 between the release fractions and time windows and the number of latent health effects, values of EHs and LHs for all 15,443 source terms were calculated and the source terms subdivided into three groups:

a) EH > 0 and LH > 0 (total of 213 source terms),

b) EH = 0 and LH > 0 (total of 15,220 source terms), and

c) EH = 0, LH = 0 (no source terms).

Each of the above categories was treated separately for partitioning.

The logarithms of minimum and maximum predictions for the early and latent fatalities for group (a) are as follows:

Log_{10}(Min EH) = -2.0; Log_{10}(Max EH) = 0.6,
Log_{10}(Min LH) = 3.6; Log_{10}(Max LH) = 4.0.

The logarithms of minimum and maximum predictions for the early and latent fatalities for group (b) are as follows:

Log_{10}(Min LH) = -0.9; Log_{10}(Max LH) = 3.5.

Group (a). In the process of partitioning for the latent fatalities, all source terms in group (a) were placed into four groups defined such that the ratio of the maximum LH to minimum LH within the same group was 1.5. Similar ratio for the four groups used to partition source terms based on the number of the early health effects was 10. Partitioning of the source terms was performed in two steps. First, the source terms were assigned to the corresponding cells according to the health effect weights. Results of this preliminary partitioning for group (a) are shown in Table C.~~8~~ 9

Table C.~~6~~ 9. Preliminary Partitioning of Source Terms with Non-Zero Early Fatalities and Non-Zero Latent Fatalities

					EH Weight
Number of Source Terms				33	0.74
	24	27	24	18	-0.26
	60	17	4	6	-1.26
LH Weight (mid-point)	3.64	3.82	4.00	4.17	
ST Group	22	23	24	25	

Secondly, based on the number of source terms in each cell, the source terms were combined in the cells carrying the highest number of source terms. Results of this final partitioning for group (a) are shown in Table C.~~7~~ 10

10

Table C.~~7~~. Final Partitioning of Source Terms with Non-Zero Early Fatalities and Non-Zero Latent Fatalities

					EH Weight
Number of Source Terms				57	0.74
		44	28		-0.26
	84				-1.26
LH Weight (mid-point)	3 64	3.82	4.00	4.17	
ST Group	22	23	24	25	

Group (b). The one dimensional partitioning of the source terms in group (b) which contains source terms with zero early health effects and non-zero latent health effects is shown in Table C.~~8~~ 11

As a result of partitioning, 25 source term groups were formed for further consequence calculations. A frequency weighted mean source term was then calculated by frequency averaging the source terms in each of these 25 partition groups. The resulting 25 mean source terms are shown in Table C.~~9~~.
12

C.4 References

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C3. "XSOR Codes User's Manual," NUREG/CR-5360, Sandia National Laboratories, December 1989.

C4. "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants, Final Summary Report" NUREG-1150, U. S. Nuclear Regulatory Commission (USNRC), December 1990.

C5. R. S. Denning, et al., "Source Term Calculations for assessing Radiation Dose to Equipment," NUREG/CR-4949, July 1989.

C6. R. J. Breeding, et al., "Evaluation of Severe Accident Risks: Surry Unit 1," Sandia National Laboratories, NUREG/CR-4551, Vol. 3, Rev. 1, October 1990.

C7. R. L. Iman, et al., "PARTITION: A Program for Defining the Source Term/Consequence Analysis Interface in the NUREG-1150 Probabilistic Risk Assessments, User's Guide," Sandia National Laboratories, NUREG/CR-5253, May 1990.

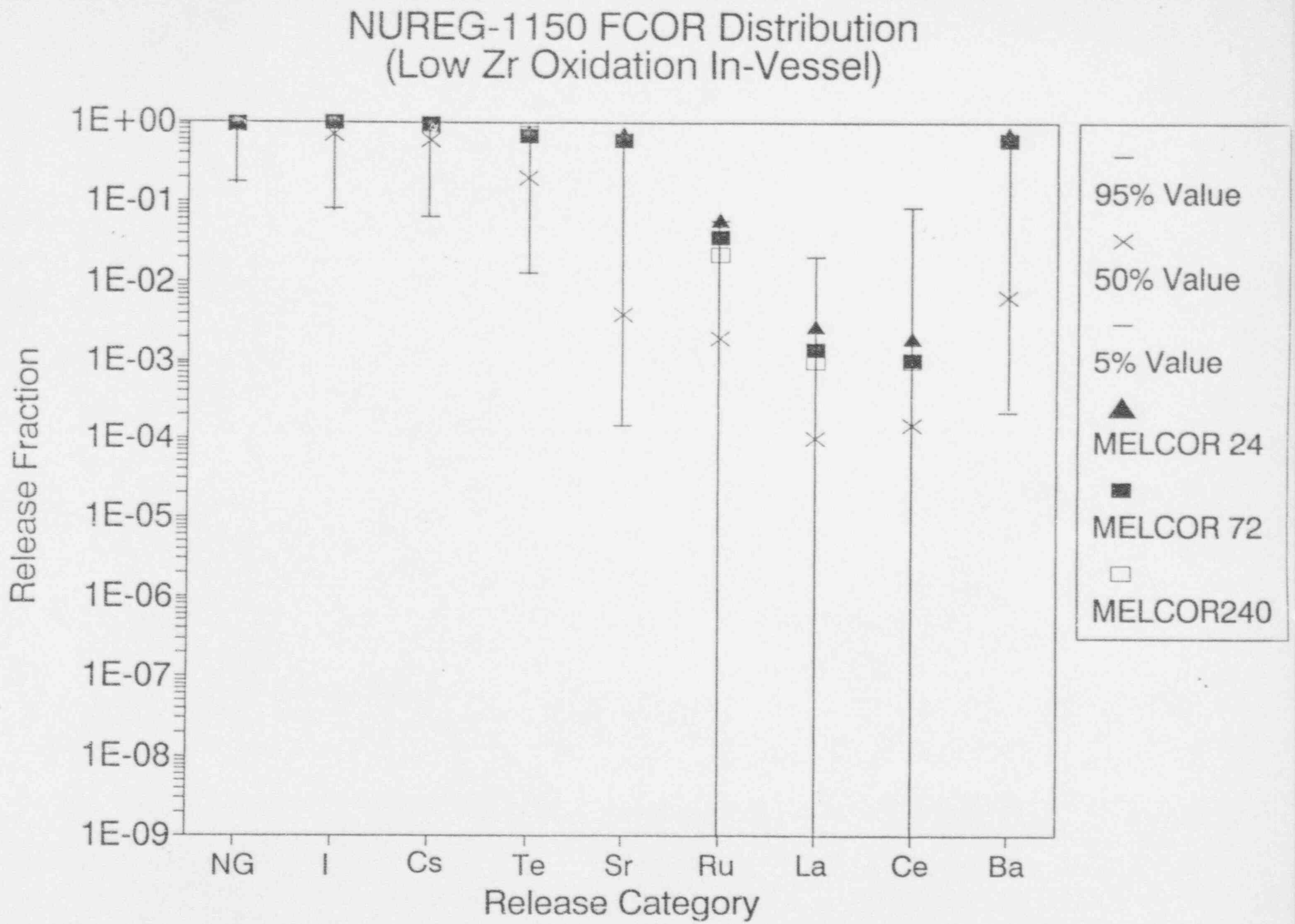
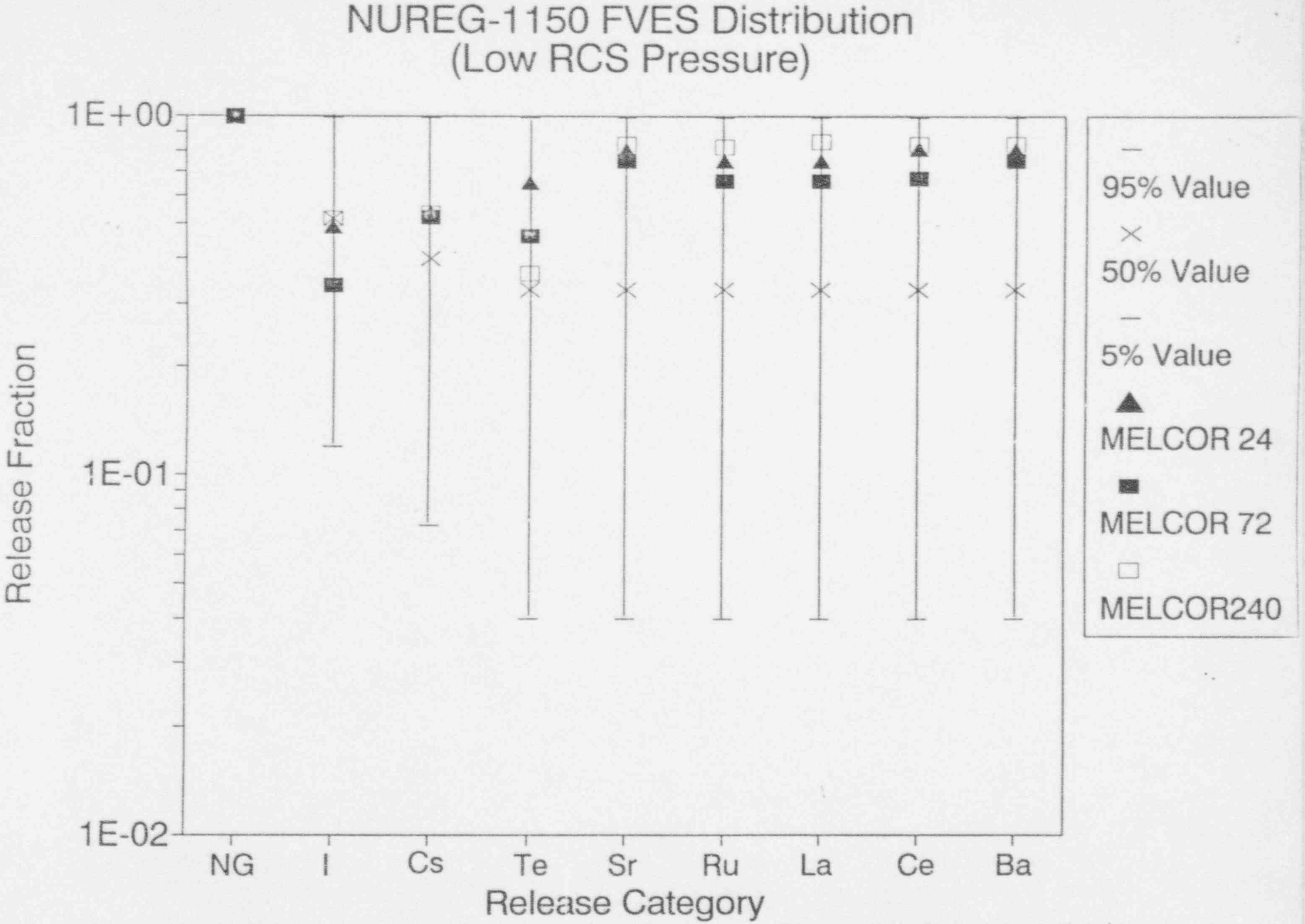


Figure C.1 (a) Fraction of fission products in the core that are released to the vessel



C.1 (b) Fraction of fission products in the vessel that are released into the containment

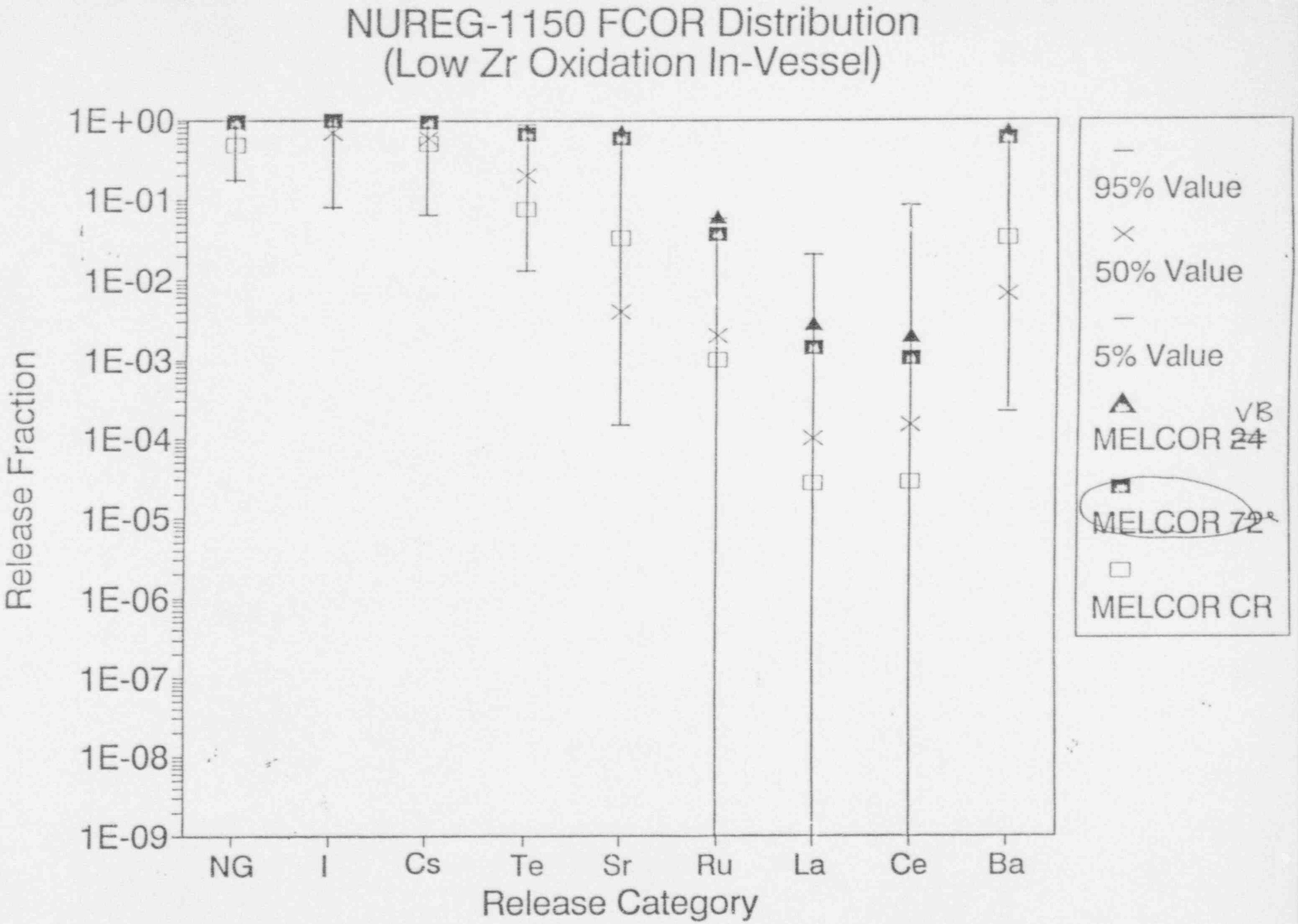


Figure C.2 (a) Fraction of fission products in the core that are released to the vessel

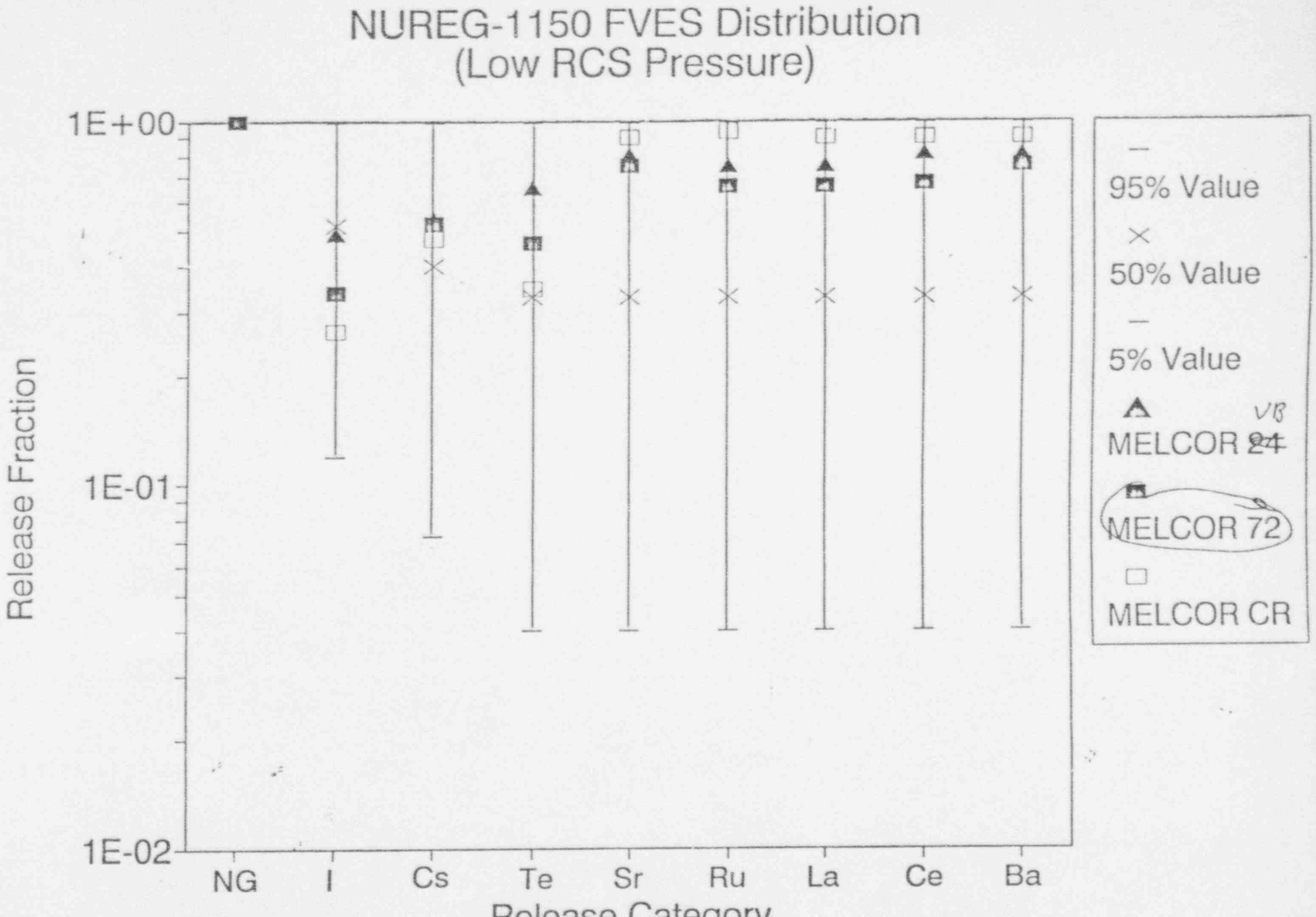
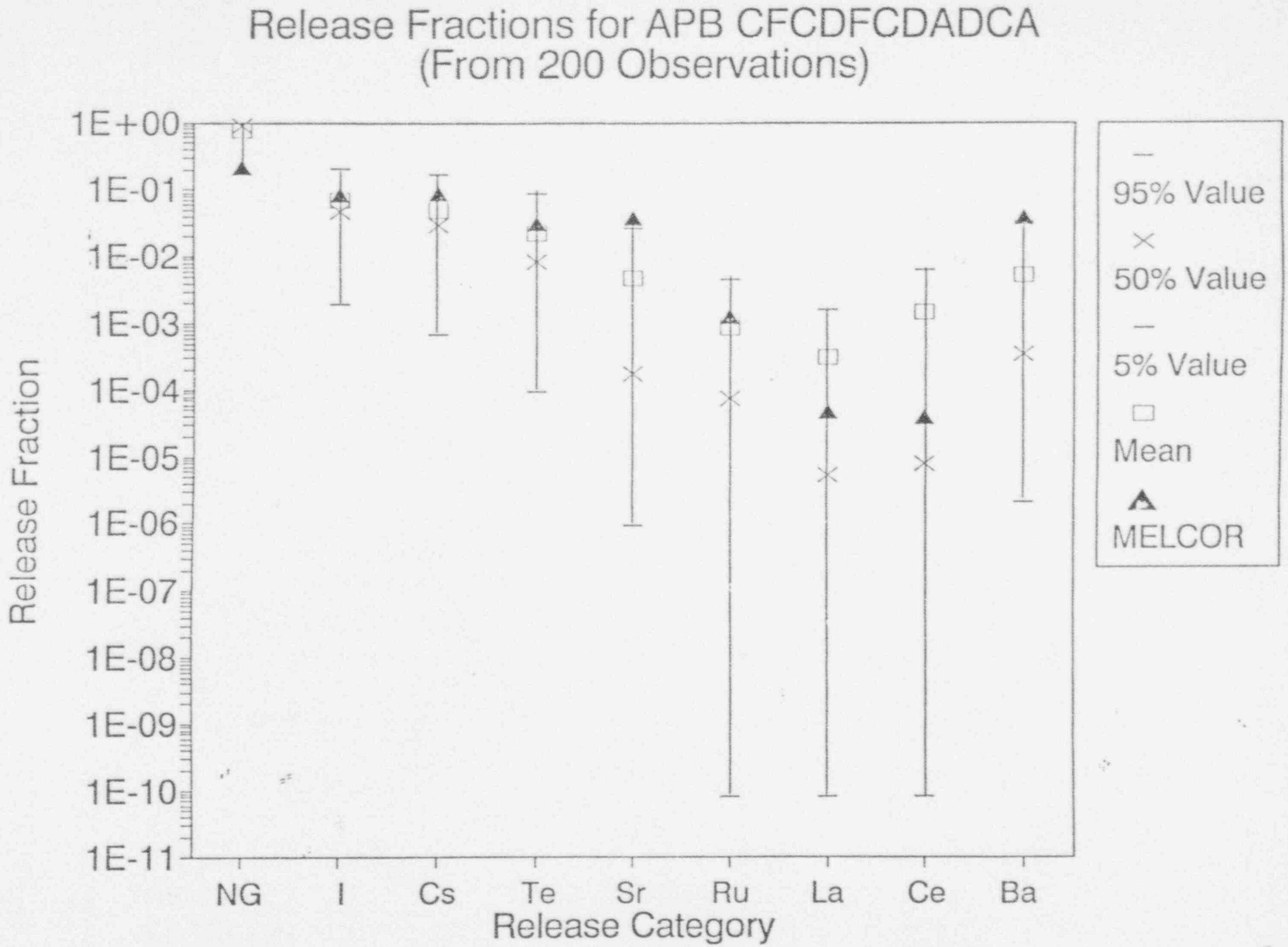
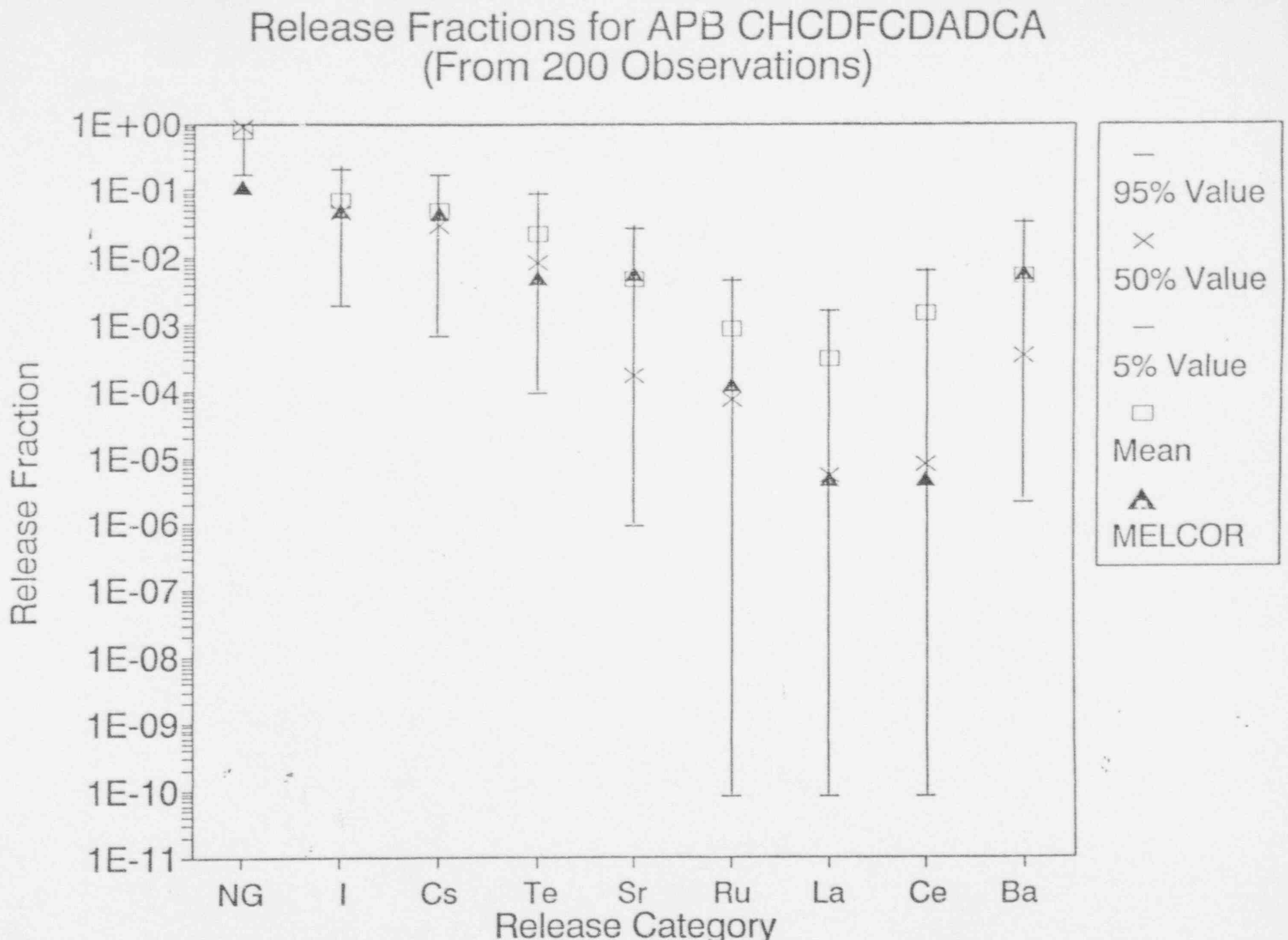


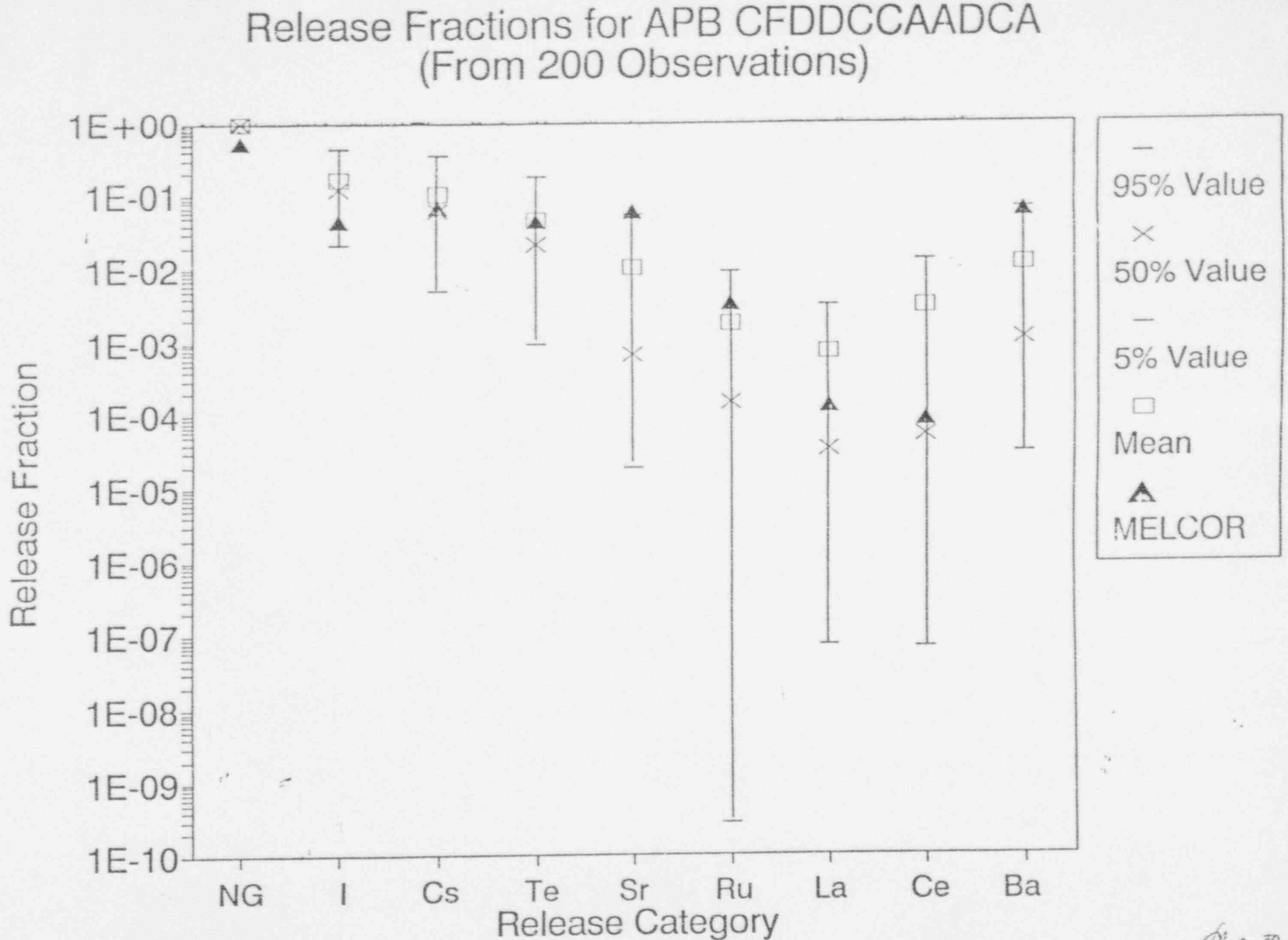
Figure C.2 (b) Fraction of fission products in the vessel that are released into the containment



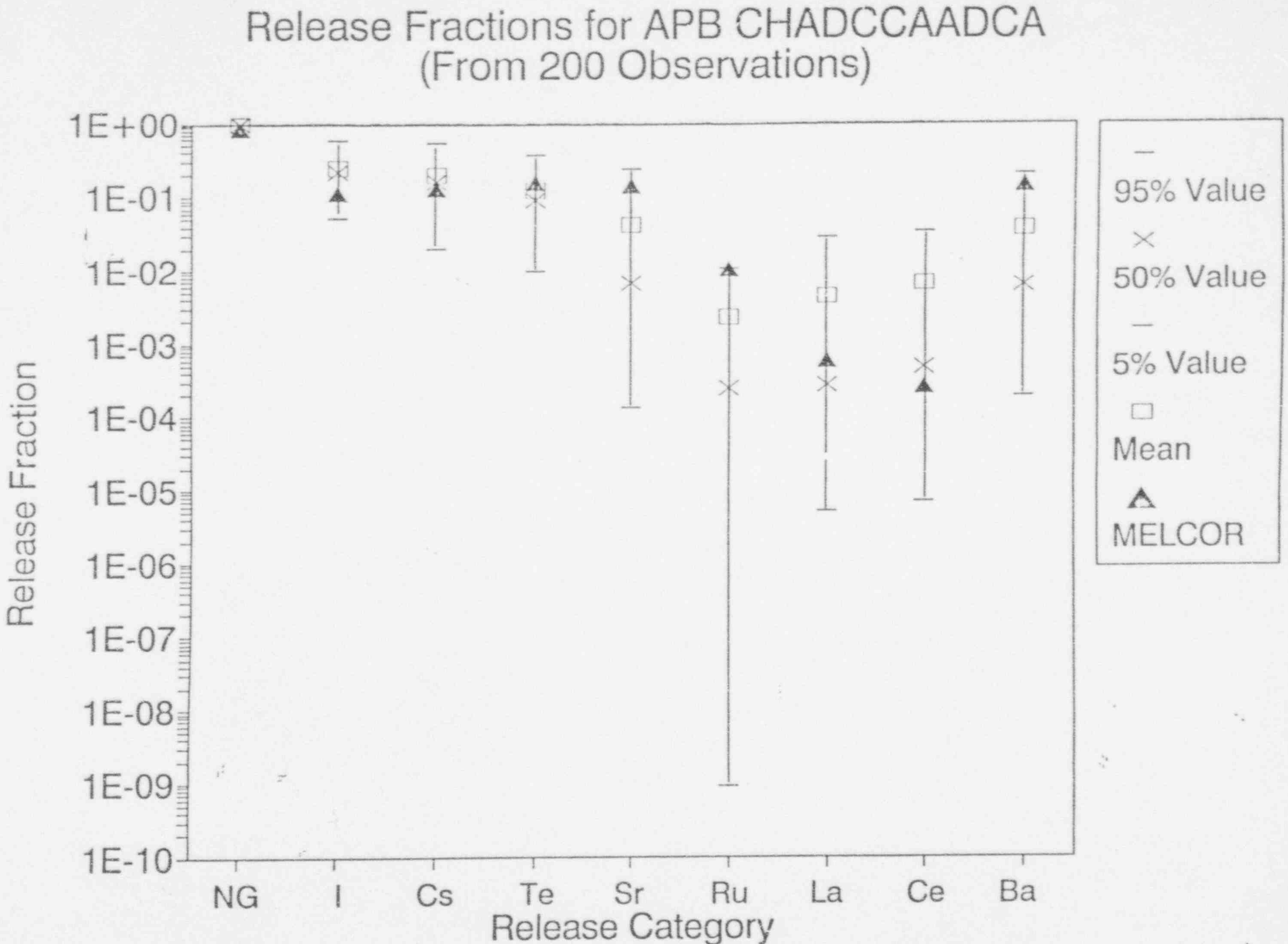
C.3.(a) No Vessel Breach, Early Containment Failure, Containment Spray



C.3.(b) No Vessel Breach, Early Containment Failure, No Containment Spray

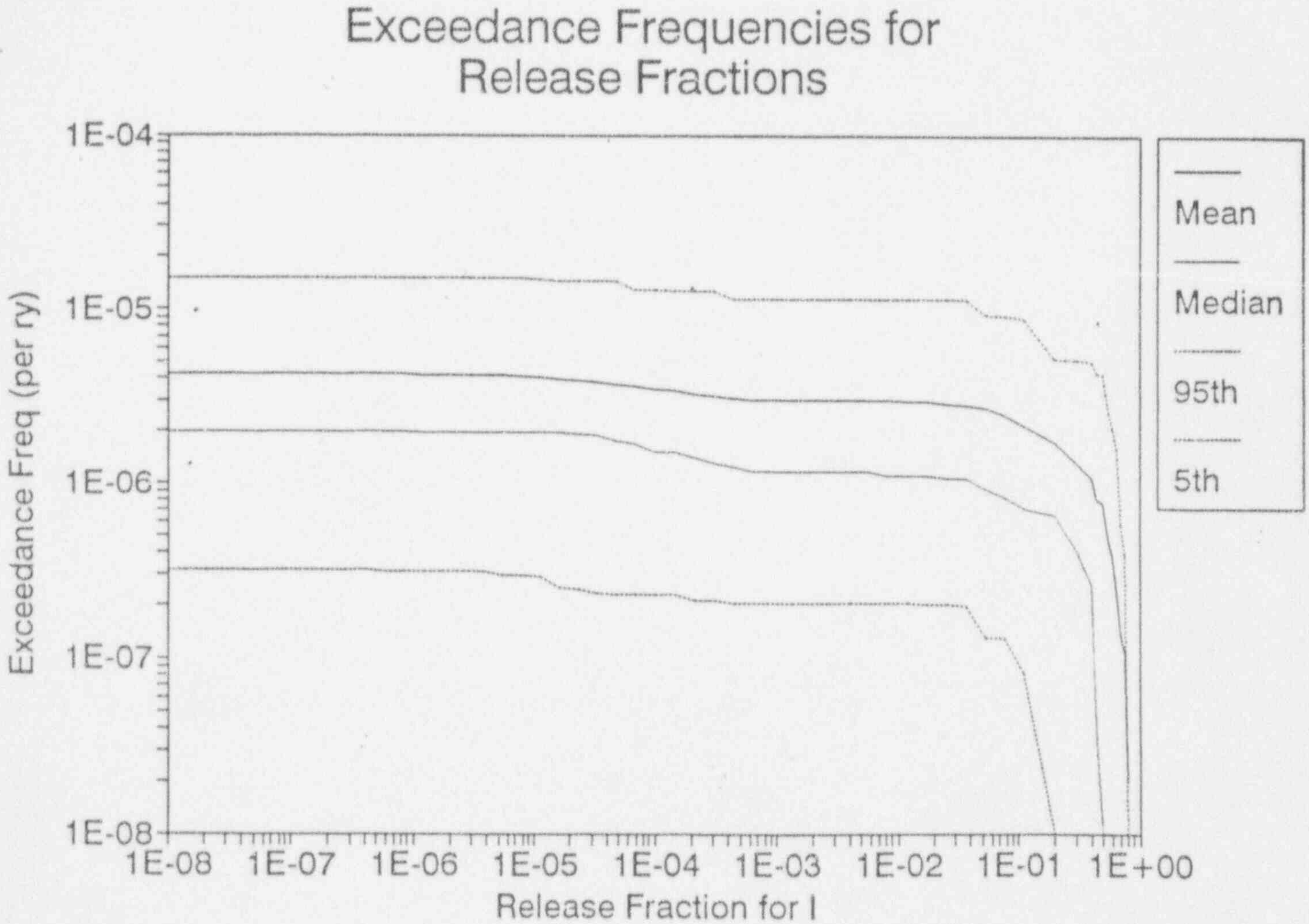


C.3.(c) Vessel Breach, Early Containment Failure, Containment Spray and Deep Wet Cavity

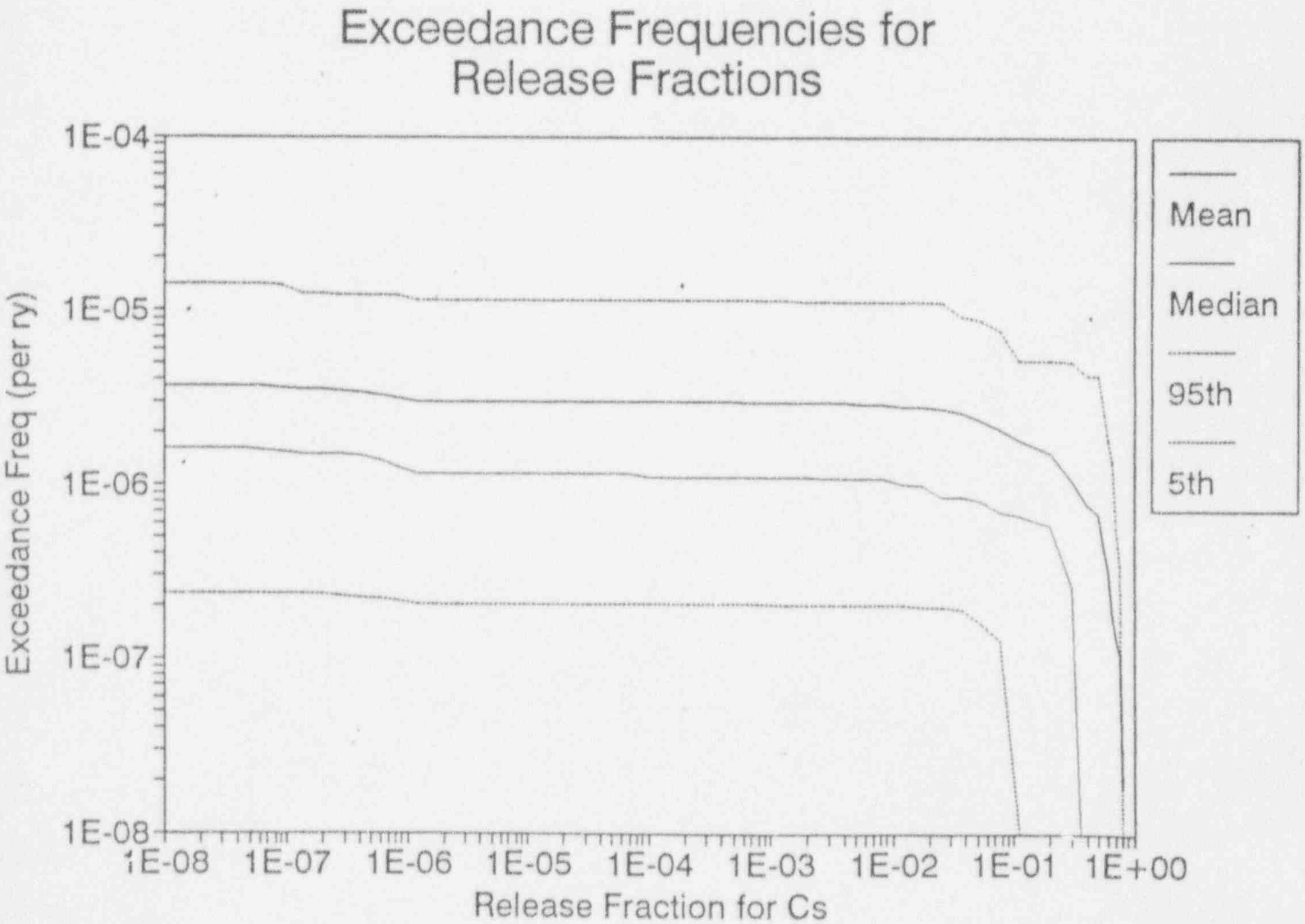


C.3.(d) Vessel Breach, Early Containment Failure, No Containment Spray, and Dry Cavity

~~Figure 7.1(a) to 7.1(h)~~

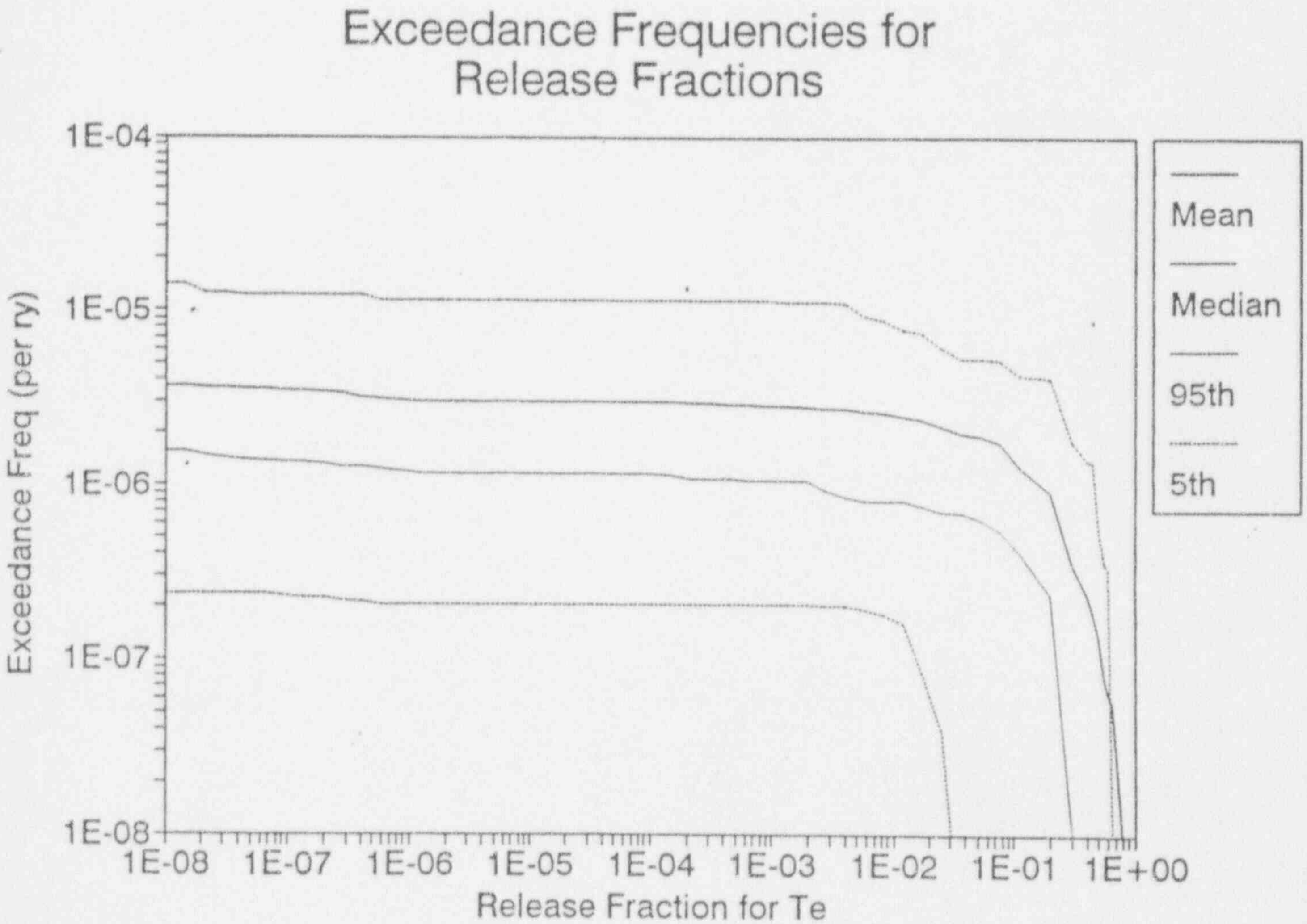


(a)

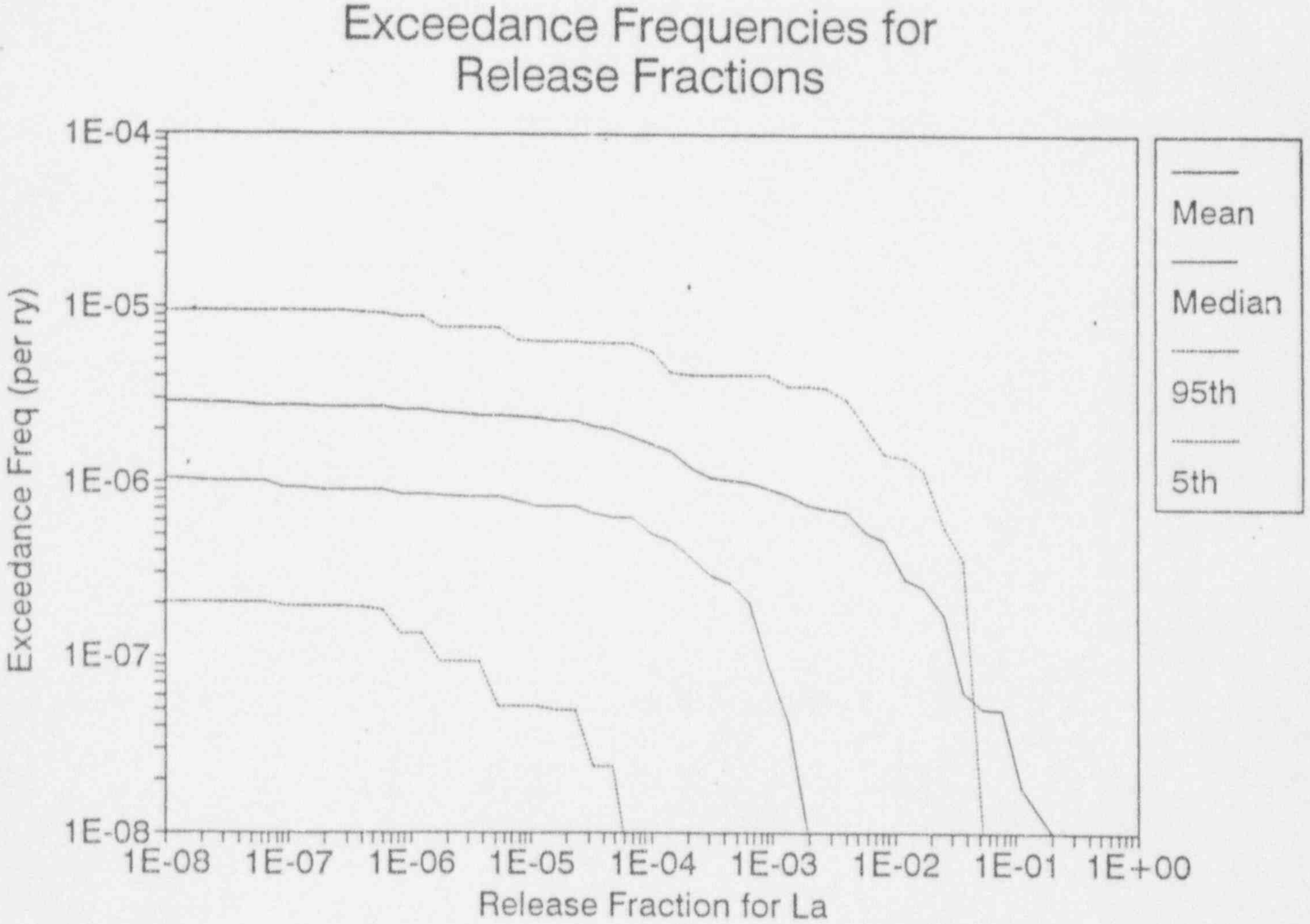


Exceedance Frequencies for
Release Fractions
1E-04
1E-05
1E-06
1E-07
1E-08
Exceedance Freq (per ry)
1E-08
1E-07
1E-06
1E-05
1E-04
1E-03
1E-02
1E-01
1E+00
Release Fraction for Cs
Mean
Median
95th
5th

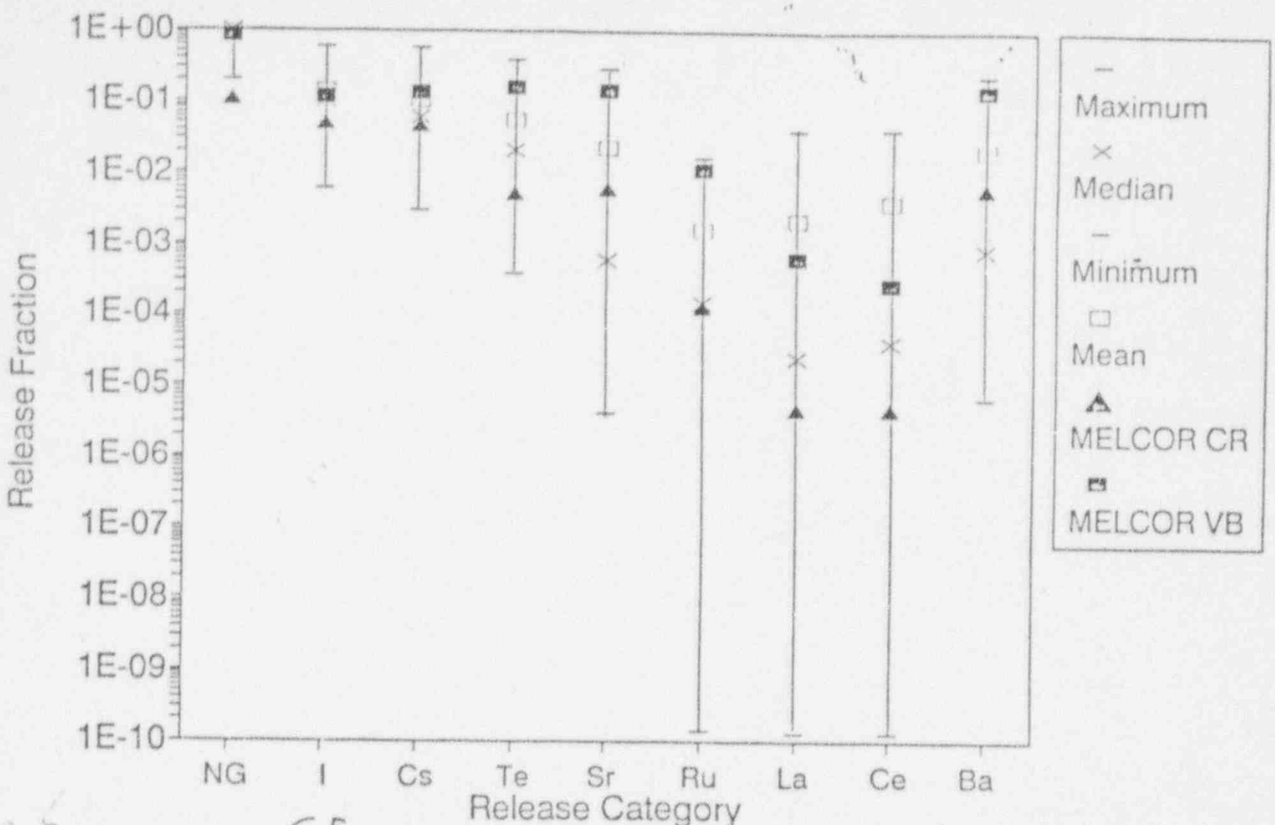
C.4 (b)


Exceedance Frequencies for
Release Fractions
1E-04
1E-05
1E-06
1E-07
1E-08
Exceedance Freq (per ry)
1E-08
1E-07
1E-06
1E-05
1E-04
1E-03
1E-02
1E-01
1E+00
Release Fraction for Te
Mean
Median
95th
5th

C.4 (c)

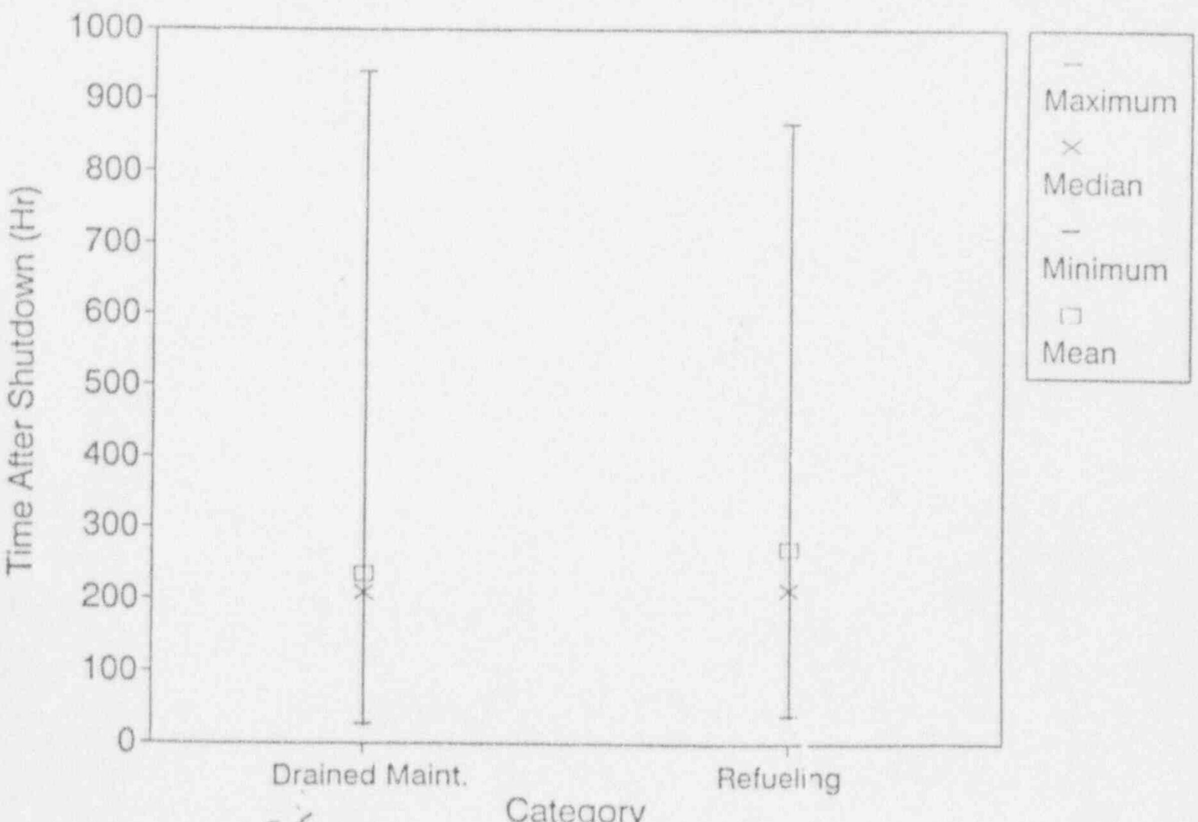

Exceedance Frequencies for
Release Fractions
Exceedance Freq (per ry)
1E-04
1E-05
1E-06
1E-07
1E-08
1E-08
1E-07
1E-06
1E-05
1E-04
1E-03
1E-02
1E-01
1E+00
Release Fraction for La
Mean
Median
95th
5th

C.4.(d)



C.5

Figure ~~4~~ C.5 Release Fractions of the Nine Radionuclide Categories for the LP/S Source Terms



C.6

Figure ~~4~~ C.6 Time Between Reactor Shutdown and Accident Initiation for LP/S Risk Analyses

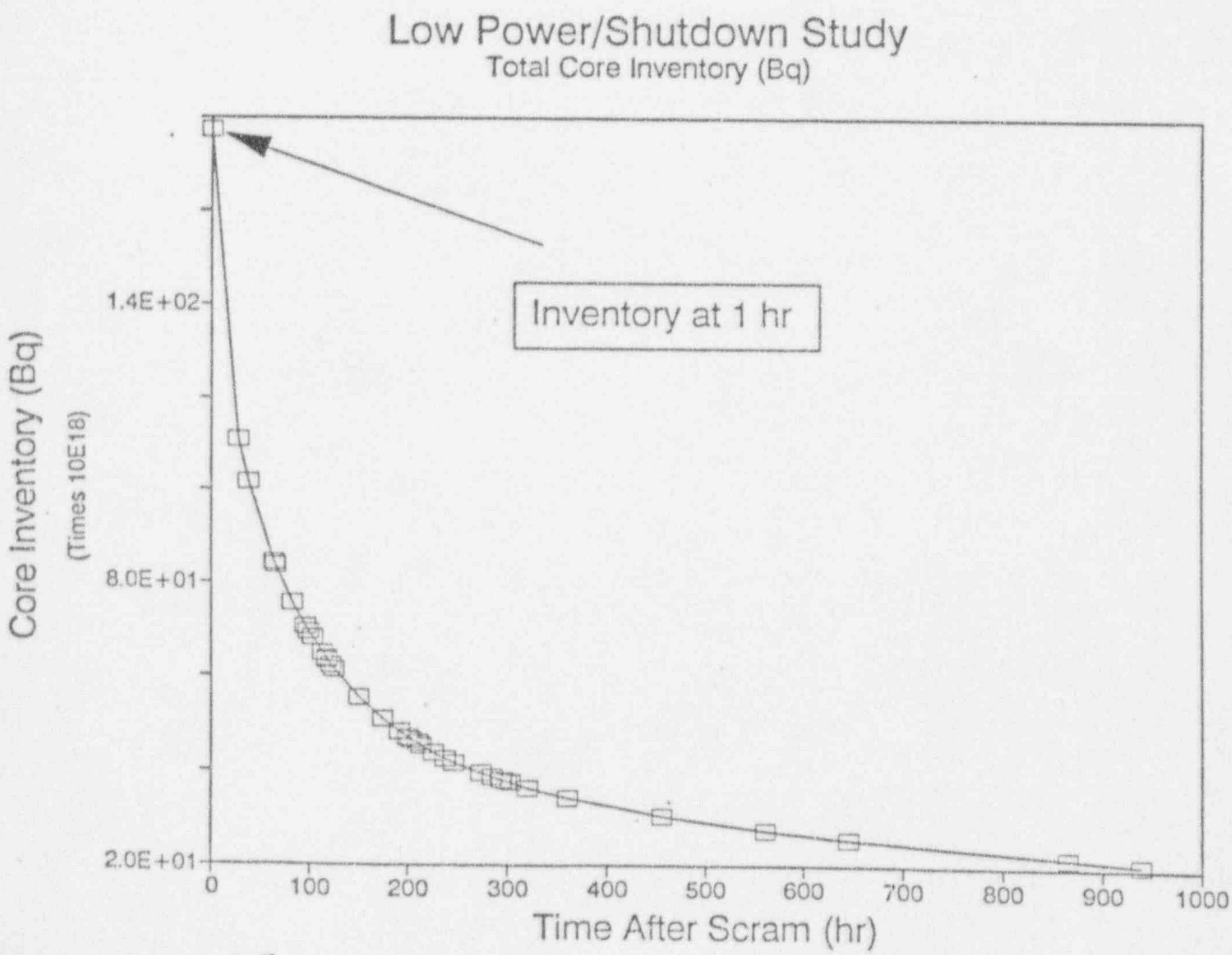


Figure C.7 Total Core Inventory as a Function of Time After Reactor Shutdown

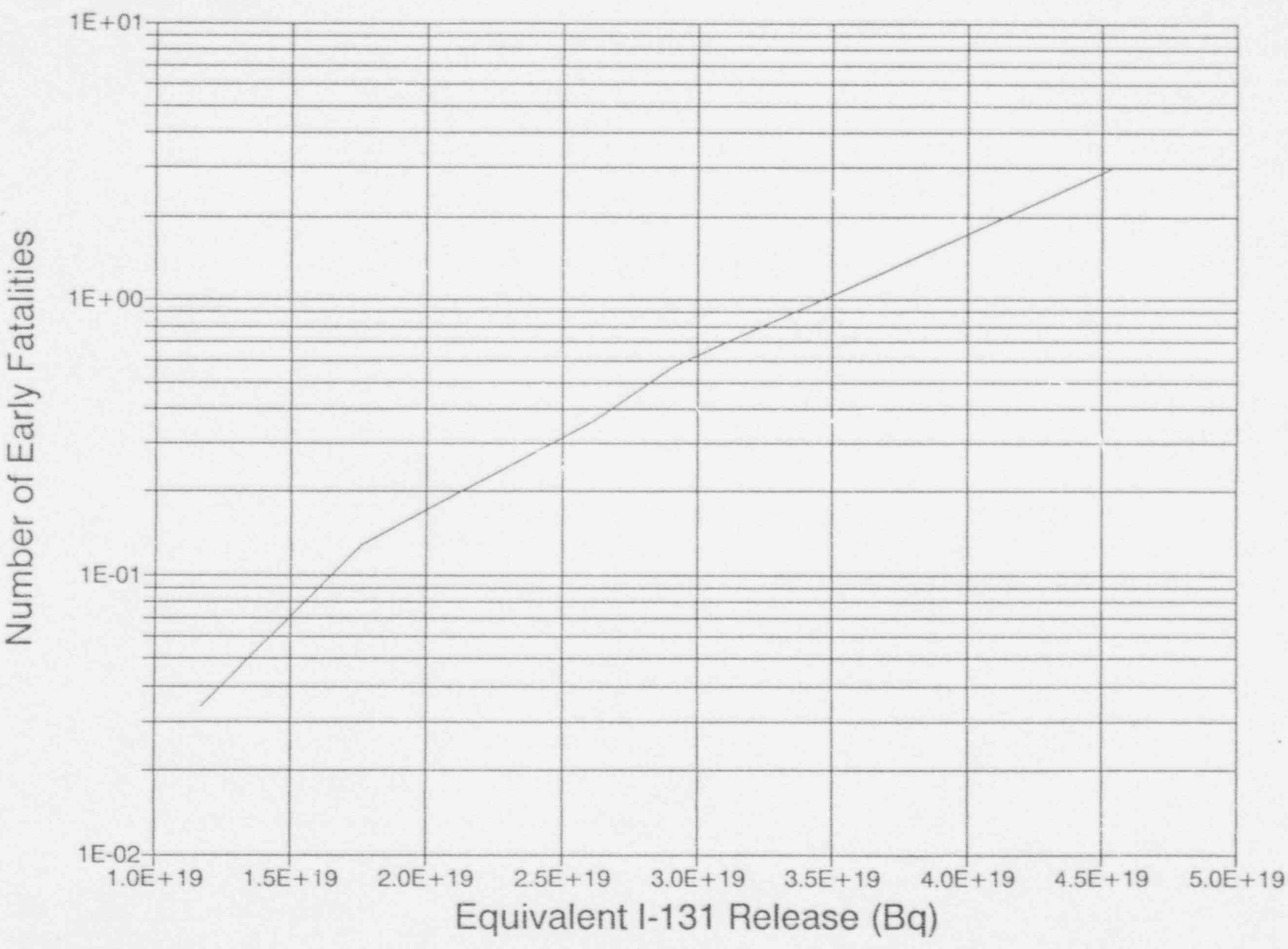


Fig. C.~~9~~8 Prediction of Early Fatalities vs. Equivalent I-131 Release

Table C.1 Parameters Used in the SURSOR Code

FCOR	Fraction of the radionuclide in the core released to the vessel before or at vessel breach (VB)
FVES	Fraction of the radionuclide released from the vessel to the containment before or at VB
VDF	Decontamination factor for pool scrubbing for Event V (not used)
FCONV	Fraction of the radionuclide in the containment from RCS release that is released from the containment in the absence of any mitigating effects
FCCI	Fraction release of radionuclide from corium during CCI
FCONC	Containment transport fraction for ex-vessel release
SPRDF	Decontamination factor for containment sprays
LATEI	Fraction of the iodine deposited in the containment which is revolatilized and released to the environment late in the accident
FLATE	Fractional release of material deposited in the RCS due to revaporization
DST	Fraction of core radionuclide released to the containment due to DCH at VB
FISGFOSG	Fraction of radionuclide released from the RCS to the steam generator, and from the steam generator to the environment (not used)
POOL-DF	Decontamination factor for a pool of water overlying the core debris during CCI

Table C.2 Low Power/Shutdown APBs for Source Term Calculations

	APB ID[1]	1	2	3	4	5	6	7	8	9	10	11
		CF Time	Spray	CCI Mode	RCS Pres	VB Mode	SGTP	CCI Size	Zr Oxide	HPME	CF Size	RCS Hol
1	CFCDFCDADCA	Early	Lt-to-VL	No	Low	No	No	No	Low	No	Leak	One
2	CHCDFCDADCA	Early	No	No	Low	No	No	No	Low	No	Leak	One
3	CFDDCCAADCA	Early	Lt-to-VL	Prompt Deep	Low	BtmHd	No	Large	Low	No	Leak	One
4	CHADCCAADCA	Early	No	Prompt Dry	Low	BtmHd	No	Large	Low	No	Leak	One

Note: (1) According to ABP identification used in NUREG-1150.

Table C.3 Frequency-Weighted Release Fractions and Frequencies

1	CHADBCABDBAB	1.000E+00	4.310E-01	3.884E-01	2.167E-01	6.362E-02	6.039E-03	7.747E-03	1.149E-02	5.585E-02	4.174E-07
2	CHADBCABDBAC	1.000E+00	4.128E-01	3.649E-01	2.232E-01	7.863E-02	6.400E-03	9.742E-03	1.348E-02	6.780E-02	2.994E-07
3	CDCDFCDBDBAB	8.433E-01	1.407E-01	1.136E-01	5.439E-02	1.392E-02	2.890E-03	8.251E-04	3.671E-03	1.484E-02	2.785E-07
4	CHCDFCDBDBAB	8.352E-01	1.334E-01	1.091E-01	4.728E-02	7.374E-03	1.965E-03	4.435E-04	2.038E-03	8.273E-03	2.280E-07
5	CDCDFCDBDBAC	8.122E-01	1.186E-01	8.812E-02	4.487E-02	1.519E-02	2.874E-03	8.964E-04	3.953E-03	1.600E-02	2.165E-07
6	GHEDBCABDDAA	5.000E-03	1.330E-04	6.603E-07	3.889E-07	1.400E-07	1.050E-08	1.562E-08	2.077E-08	1.175E-07	1.959E-07
7	CHCDFCDBDBAC	8.099E-01	1.144E-01	8.780E-02	3.851E-02	7.339E-03	1.790E-03	4.282E-04	1.888E-03	8.150E-03	1.741E-07
8	CHADBCABDBAA	1.000E+00	5.015E-01	4.621E-01	2.631E-01	6.994E-02	7.239E-03	8.911E-03	1.245E-02	6.410E-02	1.595E-07
9	CGADBCABDBAB	1.000E+00	4.319E-01	3.849E-01	2.072E-01	7.240E-02	6.794E-03	8.520E-03	1.391E-02	6.372E-02	1.580E-07
10	GDCDBCDBDDAA	5.000E-03	9.194E-05	2.754E-09	1.472E-09	4.288E-10	8.350E-11	1.826E-11	7.210E-11	4.550E-10	1.484E-07
11	GHDDBCABDDAA	5.000E-03	1.327E-04	3.668E-07	2.067E-07	4.025E-08	9.186E-09	2.334E-09	7.632E-09	4.386E-08	1.448E-07
12	CDCDFCDBDBAA	8.244E-01	1.514E-01	1.281E-01	6.185E-02	1.093E-02	2.871E-03	5.880E-04	2.593E-03	1.228E-02	1.240E-07
13	GDDDBCABDDAA	5.000E-03	1.589E-04	3.685E-09	2.085E-09	6.088E-10	8.499E-11	4.091E-11	9.523E-11	5.813E-10	1.097E-07
14	CGADBCABDBAC	1.000E+00	4.157E-01	3.645E-01	2.078E-01	8.569E-02	6.916E-03	1.016E-02	1.574E-02	7.392E-02	1.090E-07
15	CHCDFCDBDBAA	8.543E-01	1.556E-01	1.3?0E-01	6.381E-02	9.515E-03	2.657E-03	5.294E-04	2.443E-03	1.081E-02	1.069E-07
16	CFADBCABDBAB	1.000E+00	4.335E-01	3.8[illegible]-01	2.099E-01	7.333E-02	6.926E-03	8.636E-03	1.414E-02	6.461E-02	1.058E-07
17	CFADBCABDBAC	1.000E+00	4.178E-01	3.657E-01	2.115E-01	8.768E-02	7.140E-03	1.040E-02	1.622E-02	7.576E-02	6.643E-08
18	GDCDFCDBDDAA	4.122E-03	4.994E-05	1.249E-09	6.411E-10	1.802E-10	3.583E-11	7.617E-12	2.963E-11	1.925E-10	6.250E-08
19	CGADBCABDBAA	1.000E+00	5.024E-01	4.626E-01	2.584E-01	7.959E-02	6.887E-03	9.255E-03	1.302E-02	6.948E-02	5.947E-08
20	GDCDFCDBDDAB	4.106E-03	4.407E-05	9.361E-10	4.273E-10	5.702E-11	1.650E-11	2.495E-12	9.185E-12	6.519E-11	5.818E-08
21	GHCDFCDBDDAA	4.075E-03	6.148E-05	2.292E-07	9.753E-08	9.855E-09	3.339E-09	4.605E-10	1.654E-09	1.190E-08	5.234E-08
22	CFADBCABDBAA	1.000E+00	5.069E-01	4.690E-01	2.647E-01	8.139E-02	7.085E-03	9.495E-03	1.328E-02	7.110E-02	5.118E-08
23	GHCDFCDBDDAB	3.972E-03	5.410E-05	1.977E-07	8.265E-08	8.523E-09	2.859E-09	4.253E-10	1.652E-09	1.028E-08	5.063E-08
24	CHADBCABDBAD	1.000E+00	5.149E-01	4.855E-01	2.020E-01	5.555E-02	6.260E-03	5.606E-03	1.483E-02	4.965E-02	5.026E-08
25	GHADBCABDDAB	5.000E-03	2.019E-04	7.505E-07	4.117E-07	7.400E-08	7.269E-09	6.816E-09	1.056E-08	6.303E-08	4.666E-08
26	GDCDBCDBDDAB	5.000E-03	6.374E-05	2.128E-09	1.217E-09	4.166E-10	7.851E-11	1.850E-11	7.662E-11	4.391E-10	4.191E-08
27	CDCDFCDBDBAD	7.881E-01	1.223E-01	8.641E-02	3.634E-02	1.100E-02	2.246E-03	8.266E-04	4.491E-03	1.168E-02	3.792E-08
28	GHADBCABDDAA	5.000E-03	1.535E-04	8.205E-07	3.689E-07	5.473E-08	6.945E-09	4.858E-09	7.129E-09	4.865E-08	3.695E-08
29	GDDDBCABDDAB	5.000E-03	1.460E-04	2.755E-09	1.580E-09	5.167E-10	7.929E-11	3.219E-11	9.052E-11	5.112E-10	3.096E-08
30	GHADBCABDDAD	5.000E-03	1.541E-04	6.680E-07	4.291E-07	1.264E-07	9.041E-09	[illegible]348E-08	2.235E-08	1.046E-07	2.205E-08
31	GHEDBCABDDAB	5.000E-03	1.580E-04	6.771E-07	3.987E-07	1.016E-07	8.457E-09	1.172E-08	1.610E-08	8.779E-08	2.194E-08
32	CHCDFCDBDBAD	8.242E-01	1.474E-01	1.181E-01	5.610E-02	1.670E-02	3.682E-03	1.366E-03	7.948E-03	1.771E-02	2.138E-08
33	EFADBCABDAAA	1.000E+00	2.723E-02	9.914E-04	6.034E-04	2.375E-04	1.074E-05	3.227E-05	3.671E-05	1.811E-04	2.005E-08
34	CHEDBCABDCAA	1.000E+00	2.662E-01	2.292E-01	1.157E-01	3.909E-02	5.046E-03	4.340E-03	6.879E-03	3.739E-02	1.975E-08
35	GHADBCABDDAC	5.000E-03	1.567E-04	7.054E-07	4.003E-07	8.568E-08	8.461E-09	7.645E-09	1.217E-08	7.313E-08	1.941E-08
36	CGADBCABDBAD	1.000E+00	5.164E-01	4.905E-01	1.504E-01	4.385E-02	4.374E-03	4.590E-03	1.080E-02	3.902E-02	1.900E-08
37	GDCDFCDBDDAC	4.152E-03	4.318E-05	1.405E-09	7.582E-10	2.264E-10	4.498E-11	9.630E-12	3.754E-11	2.416E-10	1.774E-08
38	CDCDBCDBDCAA	1.000E+00	3.017E-02	1.137E-02	6.805E-03	2.638E-03	4.737E-04	1.116E-04	4.438E-04	2.765E-03	1.695E-08
39	GHDDBCABDDAB	5.000E-03	1.577E-04	3.604E-07	1.981E-07	3.065E-08	6.619E-09	2.581E-09	6.946E-09	3.244E-08	1.622E-08
40	GDCDFCDBDDAD	4.398E-03	4.556E-05	1.280E-09	6.287E-10	1.605E-10	3.295E-11	6.879E-12	2.706E-11	1.718E-10	1.548E-08
41	CHDDBCABDCAA	1.000E+00	1.921E-01	1.392E-01	7.758E-02	1.951E-02	4.053E-03	1.158E-03	3.726E-03	2.092E-02	1.460E-08
42	EGADBCABDAAA	1.000E+00	9.420E-02	7.201E-02	4.092E-02	5.399E-03	7.792E-04	4.693E-04	8.176E-04	5.009E-03	1.437E-08
43	GHCDFCDBDDAC	4.078E-03	4.091E-05	1.694E-07	8.107E-08	1.361E-08	3.762E-09	6.857E-10	2.703E-09	1.575E-08	1.418E-08
44	GHCDFCDBDDAD	3.982E-03	3.677E-05	1.339E-07	5.869E-08	1.408E-08	3.012E-09	8.546E-10	4.024E-09	1.527E-08	1.388E-08
45	CDDDBCABDCAA	1.000E+00	4.722E-02	1.441E-02	8.087E-03	3.269E-03	4.843E-04	2.102E-04	5.421E-04	3.210E-03	1.253E-08
46	EFADBCABDAAB	1.000E+00	3.493E-02	9.017E-[illegible]	6.967E-04	1.509E-04	7.240E-06	1.814E-05	1.876E-05	1.105E-04	9.497E-09
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52	GGCDBCDBDDAA	5.000E-03	7.050E-05	1.879E-09	8.524E-10	1.284E-10	3.096E-11	5.530E-12	2.059E-11	1.441E-10	6.684E-09
53	EGADBCABDAAB	1.000E+00	7.786E-02	4.262E-02	2.859E-02	3.651E-03	3.816E-04	3.507E-04	4.863E-04	3.094E-03	6.622E-09
54	CDCCFCDBDBAA	8.193E-01	1.320E-01	1.137E-01	5.658E-02	1.127E-02	2.798E-03	6.793E-04	3.323E-03	1.240E-02	6.481E-09
55	CHACACBBBBAA	1.000E+00	3.938E-01	3.563E-01	1.647E-01	3.329E-02	7.338E-03	5.373E-03	5.979E-03	3.235E-02	6.393E-09
56	CHCDFCDBDCAA	7.602E-01	9.325E-02	7.160E-02	2.960E-02	4.206E-03	1.248E-03	2.005E-04	7.359E-04	4.997E-03	6.109E-09
57	CDCDBCDBDBAC	1.000E+00	3.090E-01	2.316E-01	1.254E-01	5.163E-02	8.680E-03	2.966E-03	1.301E-02	5.331E-02	6.085E-09
58	CHCDFCDBDCAB	7.534E-01	7.972E-02	5.865E-02	2.399E-02	3.356E-03	9.984E-04	1.713E-04	6.734E-04	3.993E-03	6.047E-09
59	CHCCFCDBDBAA	8.668E-01	1.496E-01	1.374E-01	6.302E-02	9.936E-03	2.638E-03	6.403E-04	3.335E-03	1.102E-02	5.752E-09
60	CHADBCABDCAB	1.000E+00	2.740E-01	2.278E-01	1.034E-01	1.950E-02	2.771E-03	1.855E-03	3.324E-03	1.817E-02	5.744E-09
61	CDCDBCDBDCAB	1.000E+00	2.035E-02	8.124E-03	5.165E-03	2.152E-03	3.854E-04	9.350E-05	3.821E-04	2.250E-03	5.673E-09
62	GFADBCABDDAB	5.000E-03	1.607[illegible]	3.656E-09	2.009E-09	5.377E-10	5.582E-11	5.083E-11	7.618E-11	4.690E-10	5.611E-09
63	DHECACBBBAAA	1.000E+00	4.018[illegible]	3.881E-01	1.686E-01	3.645E-0[illegible]	1.822E-02	4.869E-03	7.092E-03	3.960E-02	5.572E-09
64	CDDDBCABDBAB	1.000E+00	3.621E-01	2.858E-01	1.508E-01	4.947E-02	8.599E-03	3.028E-03	1.224E-02	5.099E-02	5.566E-09
65	GGADBCABDDAD	5.000E-03	1.401E-04	4.051E-09	2.693E-09	9.892E-10	8.741E-11	8.136E-11	1.358E-10	8.398E-10	5.219E-09

66	EGDDBCABDAAA	1.000E+00	4.534E-02	1.508E-02	1.005E-02	1.633E-03	3.012E-04	9.543E-05	2.570E-04	1.681E-03	4.891E-09
67	CHADBCABDCAA	1.000E+00	3.763E-01	3.596E-01	1.157E-01	1.436E-02	2.640E-03	1.174E-03	2.182E-03	1.462E-02	4.873E-09
68	GGADBCABDDAC	5.000E-03	1.443E-04	3.976E-09	2.265E-09	6.089E-10	6.475E-11	5.224E-11	8.386E-11	5.305E-10	4.483E-09
69	CDDDBCABDBAC	1.000E+00	3.317E-01	2.462E-01	1.385E-01	[illegible]	8.800E-03	3.325E-03	1.343E-02	5.529E-02	4.470E-09
70	GGCDBCDBDDAB	5.000E-03	3.995E-05	1.456E-09	7.294E-10	1.621E-10	3.430E-11	6.898E-12	2.661E-11	1.745E-10	4.341E-09
71	DDCCACDBBAAA	1.000E+00	1.061E-01	8.857E-02	3.200E-02	5.481E-03	1.003E-02	2.195E-03	2.232E-03	7.930E-03	4.308E-09
72	CDDDBCABDCAB	1.000E+00	4.066E-02	9.763E-03	5.975E-03	2.552E-03	3.896E-04	1.520E-04	4.412E-04	2.517E-03	4.182E-09
73	EFADBCABDAAC	1.000E+00	3.231E-02	1.364E-03	8.969E-04	2.948E-04	1.378E-05	3.911E-05	4.416E-05	2.226E-04	3.617E-09
74	CDCDBCDBDBAA	1.000E+00	3.393E-01	2.854E-01	1.282E-01	1.923E-02	5.015E-03	9.914E-04	4.153E-03	2.159E-02	3.532E-09
75	CHEDBCABDBAA	1.000E+00	4.889E-01	4.501E-01	3.005E-01	9.791E-02	8.326E-03	1.103E-02	1.562E-02	8.456E-02	3.514E-09
76	EFCDBCDBDAAA	1.000E+00	2.172E-02	7.518E-05	3.996E-05	2.836E-06	7.853E-07	1.288E-07	4.704E-07	3.246E-06	3.327E-09
77	CHADDCBBDBAB	1.000E+00	3.898E-01	3.174E-01	2.420E-01	1.406E-01	1.549E-02	1.239E-02	3.173E-02	1.295E-01	3.229E-09
78	EGDDBCABDAAB	1.000E+00	3.010E-02	1.104E-02	6.830E-03	1.345E-03	2.615E-04	6.819E-05	2.186E-04	1.409E-03	3.179E-09
79	GDCCFCDBDDAA	4.117E-03	4.568E-05	1.051E-09	5.478E-10	1.500E-10	2.965E-11	6.345E-12	2.474E-11	1.599E-10	3.165E-09
80	CHEDBCABDCAB	1.000E+00	[illegible]	1.975E-01	1.050E-01	2.709E-02	3.544E-03	3.278E-03	4.879E-03	2.565E-02	2.926E-09
81	GHCCFCDBDDAA	4.155E-03	6.595E-05	1.986E-07	8.246E-08	7.450E-09	2.483E-09	3.480E-10	1.301E-09	8.850E-09	2.923E-09
82	GFADBCABDDAC	5.000E-03	1.433E-04	3.987E-09	2.294E-09	6.273E-10	6.571E-11	5.428E-11	8.642E-11	5.455E-10	2.684E-09
83	CFADBCABDCAA	1.000E+00	2.579E-01	2.034E-01	9.584E-02	1.229E-02	3.863E-03	6.227E-04	2.057E-03	1.426E-02	2.630E-09
84	CDDDBCABDBAA	1.000E+00	3.658E-01	3.015E-01	1.536E-01	2.495E-02	5.259E-03	1.884E-03	5.061E-03	2.619E-02	2.602E-09
85	CHDDBCABDBAA	1.000E+00	3.177E-01	2.503E-01	1.514E-01	2.642E-02	6.038E-03	1.815E-03	6.346E-03	2.823E-02	2.595E-09
86	CHADDCBBDBAC	1.000E+00	3.958E-01	3.214E-01	2.462E-01	1.480E-01	1.667E-02	1.304E-02	3.376E-02	1.372E-01	2.560E-09
87	GHECACBBBDAA	5.000E-03	4.836E-05	3.977E-07	2.385E-07	9.937E-08	1.046E-08	1.156E-08	1.663E-08	9.097E-08	2.559E-09
88	GFADBCABDDAD	5.000E-03	1.369E-04	3.986E-09	2.668E-09	9.878E-10	8.638E-11	8.184E-11	1.357E-10	8.377E-10	2.520E-09
89	CHADBCABDCAD	1.000E+00	2.645E-01	2.121E-01	1.316E-01	3.432E-02	3.294E-03	3.004E-03	6.856E-03	2.964E-02	2.486E-09
90	EFDDBCABDAAA	1.000E+00	3.426E-02	4.274E-04	7.600E-04	1.693E-04	7.215E-06	2.411E-05	2.466E-05	1.263E-04	2.458E-09
91	GDCCACDBBDAA	5.000E-03	5.277E-05	5.402E-09	3.537E-09	1.463E-09	2.568E-10	6.141E-11	2.432E-10	1.526E-09	2.259E-09
92	EGADBCABDAAC	1.000E+00	8.071E-02	4.865E-02	3.108E-02	6.696E-03	8.038E-04	5.839E-04	1.020E-03	6.027E-03	2.202E-09
93	CHADBCABDCAC	1.000E+00	2.519E-01	2.056E-01	1.073E-01	2.333E-02	3.647E-03	1.975E-03	3.985E-03	2.239E-02	2.185E-09
94	GHECBCABDDAA	5.000E-03	1.387E-04	7.097E-07	3.690E-07	8.469E-08	7.257E-09	7.472E-09	1.010E-08	7.088E-08	2.176E-09
95	CHDDBCABDCAB	1.000E+00	1.786E-01	1.232E-01	6.187E-02	1.021E-02	2.573E-03	6.784E-04	2.282E-03	1.128E-02	2.157E-09
96	CDCDFCDBDCAC	8.115E-01	1.537E-02	5.446E-03	3.239E-03	1.252E-03	2.253E-04	5.274E-05	2.081E-04	1.314E-03	2.132E-09
97	CGCDBCDBDBAA	1.000E+00	2.495E-01	2.042E-01	9.985E-02	1.595E-02	4.413E-03	8.931E-04	3.949E-03	1.851E-02	2.007E-09
98	CGADBCABDCAB	1.000E+00	2.583E-01	2.094E-01	9.191E-02	1.904E-02	2.958E-03	1.927E-03	3.670E-03	1.831E-02	1.932E-09
99	CFADBCABDCAB	1.000E+00	1.845E-01	1.249E-01	5.643E-02	9.303E-03	2.486E-03	5.506E-04	2.037E-03	1.055E-02	1.901E-09
100	CGADBCABDCAA	1.000E+00	3.886E-01	3.661E-01	1.360E-01	1.918E-02	3.694E-03	1.557E-03	2.982E-03	1.925E-02	1.879E-09
101	GDCCBCDBDDAA	5.000E-03	4.602E-05	8.853E-10	3.599E-10	2.181E-11	7.959E-12	9.557E-13	3.161E-12	2.778E-11	1.837E-09
102	CHACBCABDBAA	1.000E+00	5.385E-01	5.294E-01	2.654E-01	5.045E-02	9.368E-03	3.256E-03	7.045E-03	5.068E-02	1.831E-09
103	CGACACBBBBAA	1.000E+00	3.666E-01	3.291E-01	1.538E-01	4.383E-02	1.099E-02	6.671E-03	7.964E-03	4.194E-02	1.809E-09
104	CHEDBCABDBAB	1.000E+00	4.300E-01	3.966E-01	2.339E-01	6.810E-02	5.036E-03	7.204E-03	1.086E-02	5.686E-02	1.770E-09
105	EFCDBCDBDAAB	1.000E+00	1.451E-02	5.924E-05	2.924E-05	9.029E-07	5.883E-07	5.080E-08	1.486E-07	1.314E-06	1.759E-09
106	CDCDFCDBDCAD	8.777E-01	1.384E-02	4.560E-03	2.510E-03	8.674E-04	1.610E-04	3.703E-05	1.472E-04	9.142E-04	1.678E-09
107	CGADDCBBDBAB	1.000E+00	4.015E-01	3.110E-01	2.438E-01	1.752E-01	2.339E-02	1.541E-02	4.466E-02	1.707E-01	1.636E-09
108	CHCDFCDBDCAD	7.686E-01	7.074E-02	4.800E-02	2.000E-02	5.311E-03	1.076E-03	3.400E-04	1.615E-03	5.746E-03	1.636E-09
109	GHDCBCABDDAA	5.000E-03	1.384E-04	3.086E-07	1.739E-07	2.484E-08	7.133E-09	1.348E-09	3.925E-09	2.885E-08	1.607E-09
110	EHADBCABDAAA	1.000E+00	8.855E-02	6.007E-02	3.377E-02	4.323E-03	5.344E-04	3.752E-04	6.472E-04	3.931E-03	1.595E-09
111	CFACACBBBBAA	1.000E+00	3.679E-01	3.305E-01	1.550E-01	4.383E-02	1.104E-02	6.704E-03	7.966E-03	4.190E-02	1.577E-09
112	EHADBCABDAAB	1.000E+00	8.326E-02	3.499E-02	2.628E-02	2.810E-03	2.322E-04	2.492E-04	3.351E-04	2.272E-03	1.576E-09
113	CHCDFCDBDCAC	7.908E-01	8.159E-02	6.237E-02	3.155E-02	6.305E-03	1.638E-03	3.169E-04	1.254E-03	7.202E-03	1.551E-09
114	CGDDBCABDBAA	1.000E+00	2.954E-01	2.291E-01	1.281E-01	2.130E-02	4.442E-03	1.548E-03	4.623E-03	2.223E-02	1.482E-09
115	CGCDBCDBDBAB	1.000E+00	1.826E-01	1.303E-01	5.596E-02	8.207E-03	2.360E-03	5.060E-04	2.391E-03	9.456E-03	1.402E-09
116	DHDCACBBBAAA	1.000E+00	3.146E-01	2.809E-01	1.375E-01	2.761E-02	1.821E-02	4.010E-03	6.228E-03	3.344E-02	1.392E-09
117	GDDCBCABDDAA	5.000E-03	1.233E-04	1.729E-09	1.084E-09	2.278E-10	8.350E-12	2.396E-11	2.690E-11	1.668E-10	1.356E-09
118	CHDDBCABDBAB	1.000E+00	2.617E-01	1.947E-01	1.065E-01	1.862E-02	3.991E-03	1.418E-03	5.024E-03	1.942E-02	1.306E-09
119	EFDDBCABDAAB	1.000E+00	2.720E-02	2.366E-04	3.584E-04	3.112E-05	1.405E-06	3.336E-06	3.530E-06	2.141E-05	1.300E-09
120	EHEDBCABDAAB	1.000E+00	4.280E-02	2.479E-02	2.183E-02	3.403E-03	2.236E-04	2.794E-04	4.087E-04	2.785E-03	1.264E-09
121	CGADDCBBDBAC	1.000E+00	4.072E-01	3.155E-01	2.489E-01	1.815E-01	2.419E-02	1.598E-02	4.632E-02	1.768E-01	1.217E-09
122	DHEDDCBBDAAA	1.000E+00	4.530E-01	3.669E-01	2.775E-01	1.716E-01	1.928E-02	1.719E-02	4.031E-02	1.591E-01	1.205E-09
123	EHEDBCABDAAA	1.000E+00	6.515E-02	3.501E-02	3.325E-02	6.351E-03	2.855E-04	5.711E-04	7.270E-04	4.979E-03	1.176E-09
124	CFADDCBBDBAB	1.000E+00	4.057E-01	3.141E-01	2.471E-01	1.778E-01	2.378E-02	1.560E-02	4.539E-02	1.732E-01	1.035E-09
125	DDDCACBBBAAA	1.000E+00	1.155E-01	9.277E-02	3.381E-02	6.012E-03	1.004E-02	2.245E-03	2.285E-03	8.263E-03	1.074E-09
126	DDCDDCDBDAAA	1.000E+00	2.422E-02	3.969E-03	1.977E-03	4.676E-04	9.798E-05	2.550E-05	1.086E-04	5.077E-04	1.040E-09
127	CGDDBCABDBAB	1.000E+00	2.256E-01	1.570E-01	8.602E-02	1.354E-02	2.383E-03	1.275E-03	3.172E-03	1.337E-02	1.035E-09
128	CFCDBCDBDBAA	1.000E+00	3.216E-01	2.735E-01	1.671E-01	3.551E-02	9.843E-03	2.117E-03	9.758E-03	4.066E-02	1.007E-09
129	CGACBCABDBAA	1.000E+00	5.043E-01	4.874E-01	2.310E-01	3.643E-02	6.776E-03	2.493E-03	4.871E-03	3.592E-02	9.853E-10
130	EHDDBCABDAAB	1.000E+00	3.137E-02	1.076E-02	6.964E-03	1.183E-03	2.203E-04	5.833E-05	1.862E-04	1.237E-03	9.324E-10
131	GDCCCCDBBDAA	5.000E-03	4.457E-05	2.295E-10	1.693E-10	9.476E-11	1.485E-11	4.328E-12	1.807E-11	9.649E-11	9.106E-10
132	EHDDBCABDAAA	1.000E+00	4.851E-02	1.553E-02	1.033E-02	1.527E-03	2.714E-04	8.415E-05	2.350E-04	1.575E-03	8.680E-10
133	DHACACBBBAAA	1.000E+00	5.054E-01	4.858E-01	1.963E-01	1.127E-02	7.695E-03	1.589E-03	1.956E-03	1.298E-02	8.489E-10

134	EGADBCABDAAD	1.000E+00	5.900E-02	2.997E-02	3.576E-02	8.874E-03	4.880E-04	9.788E-04	1.206E-03	6.779E-03	8.277E-10
135	CFACBCABDBAA	1.000E+00	5.174E-01	5.044E-01	2.423E-01	3.856E-02	7.286E-03	2.625E-03	5.198E-03	3.820E-02	8.251E-10
136	CFCDBCDBDBAB	1.000E+00	2.208E-01	1.703E-01	7.435E-02	1.123E-02	3.369E-03	7.375E-04	3.684E-03	1.290E-02	8.246E-10
137	CHECACBBBCAA	1.000E+00	1.843E-01	1.590E-01	8.108E-02	3.037E-02	9.238E-03	4.132E-03	5.948E-03	3.110E-02	8.208E-10
138	GHECCCBBBDAA	5.000E-03	7.330E-05	3.000E-07	2.320E-07	1.721E-07	2.053E-08	1.893E-08	4.411E-08	1.626E-07	8.155E-10
139	EFADBCABDAAD	1.000E+00	2.774E-02	1.020E-03	1.878E-03	5.024E-04	1.382E-05	5.069E-05	5.190E-05	3.449E-04	7.917E-10
140	GHACBCABDDAA	5.000E-03	4.254E-05	8.175E-07	2.742E-07	2.800E-08	5.603E-09	1.701E-09	3.395E-09	2.804E-08	7.742E-10
141	CGADBCABDCAC	1.000E+00	2.462E-01	1.966E-01	9.486E-02	2.157E-0[illegible]	3.514E-03	1.817E-03	3.897E-03	2.097E-02	7.628E-10
142	CFDDBCABDBAA	1.000E+00	3.553E-01	2.937E-01	1.904E-01	4.165E-02	1.003E-02	3.050E-03	1.070E-02	4.541E-02	7.434E-10
143	CFADDCBBDBAC	1.000E+00	4.139E-01	3.209E-01	2.536E-01	1.859E-01	[illegible].482E-02	1.634E-02	4.757E-02	1.812E-01	7.401E-10
144	CFADBCABDCAC	1.000E+00	2.044E-01	1.483E-01	7.616E-02	1.493E-02	3.6[illegible]E-03	8.091E-04	2.940E-03	1.654E-02	7.11[illegible]E-10
145	CGCDBCDBDCAA	1.000E+00	1.454E-01	1.111E-01	5.372E-02	8.951E-03	2.399E-03	4.109E-04	1.507E-03	1.035E-02	6.906E-10
146	CDCCACDBBCAA	1.000E+00	3.102E-02	1.317E-02	8.642E-03	3.488E-03	6.196E-04	1.467E-04	5.799E-04	3.646E-03	6.600E-10
147	DDCCCCDBBAAA	1.000E+00	1.096E-01	9.759E-02	3.582E-02	3.850E-03	7.845E-03	1.769E-03	2.126E-03	5.234E-03	6.397E-10
148	GHDCACBBBDAA	5.000E-03	4.822E-05	3.083E-07	1.664E-07	4.681E-08	8.348E-09	2.516E-09	7.747E-09	4.868E-08	6.388E-10
149	CFDDBCABDBAB	1.000E+00	2.565E-01	1.947E-01	1.065E-01	1.684E-02	3.429E-03	1.624E-03	4.583E-03	1.720E-02	6.082E-10
150	CDCDBCDBDBAD	1.000E+00	2.753E-01	1.950E-01	8.087E-02	1.590E-02	3.538E-03	1.233E-03	6.742E-03	1.715E-02	6.060E-10
151	CGADBCABDCAD	1.000E+00	2.347E-01	1.702E-01	1.086E-01	3.955E-02	4.574E-03	3.574E-03	9.396E-03	3.595E-02	5.885E-10
152	GDDCACBBBDAA	5.000E-03	6.992E-05	5.617E-09	3.729E-09	1.540E-09	2.606E-10	7.337E-11	2.557E-10	1.586E-09	5.628E-10
153	EHADBCABDAAC	1.000E+00	7.919E-02	4.248E-02	2.868E-02	5.292E-03	5.819E-04	4.505E-04	7.759E-04	4.673E-03	5.617E-10
154	CGCDBCDBDCAB	1.000E+00	1.159E-01	8.293E-02	4.106E-02	6.233E-03	1.844E-03	2.901E-04	1.051E-03	7.259E-03	5.507E-10
155	GDCDBCDBDDAC	5.000E-03	8.655E-05	2.848E-09	1.552E-09	4.193E-10	8.160E-11	1.770E-11	6.898E-11	4.460E-10	5.339E-10
156	CGDDBCABDCAA	1.000E+00	1.618E-01	1.151E-01	5.797E-02	9.715E-03	2.425E-03	5.164E-04	1.599E-03	1.090E-02	5.088E-10
157	CHACCCBBBBAA	1.000E+00	3.990E-01	3.548E-01	2.386E-01	1.578E-01	3.292E-02	1.767E-02	8.514E-02	1.565E-01	4.531E-10
158	DHECCCBBBAAA	1.000E+00	4.881E-01	4.281E-01	2.502E-01	1.241E-01	2.542E-02	1.765E-02	5.049E-02	1.211E-01	4.495E-10
159	CDDDBCABDBAD	1.000E+00	3.013E-01	2.055E-01	8.628E-02	1.809E-02	3.409E-03	1.542E-03	7.239E-03	1.847E-02	4.249E-10
160	CGDDBCABDCAB	1.000E+00	1.263E-01	8.651E-02	4.310E-02	6.541E-03	1.846E-03	3.278E-04	1.091E-03	7.466E-03	4.053E-10
161	GDCDBCDBDDAD	5.000E-03	9.119E-05	3.423E-09	1.567E-09	2.458E-10	5.918E-11	1.038E-11	3.949E-11	2.696E-10	4.047E-10
162	EHADBCABDAAD	1.000E+00	6.395E-02	3.408E-02	3.929E-02	6.943E-03	3.576E-04	7.210E-04	9.021E-04	5.246E-03	3.985E-10
163	EFACBCABDAAA	1.000E+00	8.582E-03	1.605E-04	1.617E-04	2.585E-05	1.015E-07	2.397E-06	2.434E-06	1.788E-05	3.823E-10
164	CDCCFCDBDCAA	7.874E-01	1.217E-02	3.670E-03	2.263E-03	8.225E-04	1.508E-04	3.477E-05	1.370E-04	8.647E-04	3.794E-10
165	GDDDBCABDDAC	5.000E-03	1.503E-04	4.038E-09	2.263E-09	6.027E-10	8.512E-11	3.287E-11	8.680E-11	5.642E-10	3.790E-10
166	CHADDCBBDBAA	1.000E+00	3.193E-01	2.590E-01	1.958E-01	8.169E-02	8.711E-03	7.140E-03	1.717E-02	7.338E-02	3.619E-10
167	CFCDBCDBDCAA	1.000E+00	1.716E-01	1.287E-01	5.384E-02	8.653E-03	2.438E-03	4.434E-04	1.696E-03	1.019E-02	3.526E-10
168	CFADBCABDCAD	1.000E+00	1.610E-01	1.086E-01	5.901E-02	2.327E-02	3.997E-03	1.633E-03	7.065E-03	2.392E-02	3.127E-10
169	CHCCFCDBDCAA	7.678E-01	7.683E-02	5.932E-02	2.484E-02	3.560E-03	1.008E-03	1.706E-04	6.413E-04	4.152E-03	2.966E-10
170	GDDDBCABDDAD	5.000E-03	1.455E-04	4.110E-09	2.812E-09	6.343E-10	5.731E-11	4.897E-11	7.728E-11	5.075E-10	2.907E-10
171	EGACBCABDAAA	1.000E+00	9.355E-02	9.642E-02	4.576E-02	9.123E-04	9.819E-05	7.536E-05	8.276E-05	8.112E-04	2.856E-10
172	CGACCCBBBBAA	1.000E+00	3.511E-01	2.891E-01	1.912E-01	1.110E-01	2.217E-02	1.176E-02	5.012E-02	1.096E-01	2.793E-10
173	DFACACBBBAAA	1.000E+00	3.184E-01	2.901E-01	1.005E-01	1.311E-02	1.218E-02	2.730E-03	3.373E-03	1.649E-02	2.742E-10
174	CFDDBCABDCAA	1.000E+00	1.852E-01	1.307E-01	5.479E-02	9.369E-03	2.468E-03	5.688E-04	1.822E-03	1.076E-02	2.595E-10
175	CHECBCABDCAA	1.000E+00	2.948E-01	2.754E-01	1.127E-01	1.592E-02	3.205E-03	9.723E-04	2.194E-03	1.642E-02	2.554E-10
176	CDCCBCDBDCAA	1.000E+00	1.165E-02	2.306E-03	1.009E-03	8.826E-05	2.750E-05	3.889E-06	1.330E-05	1.080E-04	2.501E-10
177	EFACACBBBAAA	1.000E+00	2.634E-02	6.347E-04	5.910E-04	4.294E-04	3.350E-05	5.599E-05	7.234E-05	3.510E-04	2.492E-10
178	CFACCCBBBBAA	1.000E+00	3.537E-01	2.918E-01	1.937E-01	1.133E-01	2.253E-02	1.217E-02	5.054E-02	1.119E-01	2.389E-10
179	GDDCCCBBBDAA	5.000E-03	6.634E-05	3.317E-10	2.294E-10	1.045E-10	1.488E-11	5.576E-12	1.927E-11	1.023E-10	2.273E-10
180	CDCDDCDBDBAB	1.000E+00	4.368E-01	3.392E-01	2.616E-01	1.817E-01	2.751E-02	1.051E-02	4.647E-02	1.840E-01	2.078E-10
181	DHDDDCBBDAAA	1.000E+00	4.111E-01	3.123E-01	2.225E-01	1.266E-01	1.945E-02	7.496E-03	3.180E-02	1.281E-01	2.064E-10
182	GHACACBBBDAA	5.000E-03	5.751E-05	4.700E-07	2.796E-07	9.451E-08	1.206E-08	9.326E-09	1.542E-08	8.764E-08	2.062E-10
183	GHDCCCBBBDAA	5.000E-03	7.329E-05	2.661E-07	2.097E-07	1.353E-07	2.036E-08	7.964E-09	3.451E-08	1.366E-07	2.038E-10
184	CFCDBCDBDCAB	1.000E+00	1.723E-01	1.315E-01	5.039E-02	3.708E-03	1.612E-03	1.850E-04	6.221E-04	4.894E-03	2.010E-10
185	CHDCACBBBCAA	1.000E+00	1.563E-01	1.238E-01	6.738E-02	1.935E-02	8.648E-03	2.146E-03	4.016E-03	2.162E-02	2.010E-10
186	GFADBCABDDAA	5.000E-03	1.307E-04	4.002E-09	2.275E-09	9.200E-10	1.102E-10	7.447E-11	1.450E-10	8.544E-10	2.009E-10
187	CGADDCBBDBAA	1.000E+00	2.947E-01	2.312E-01	1.898E-01	1.021E-01	1.172E-02	9.248E-03	2.235E-02	9.513E-02	2.007E-10
188	DGACACBBBAAA	1.000E+00	4.465E-01	4.221E-01	1.478E-01	1.907E-02	1.146E-02	3.159E-03	3.823E-03	2.026E-02	1.967E-10
189	CHDCBCABDCAA	1.000E+00	1.741E-01	1.312E-01	7.020E-02	1.100E-02	3.208E-03	5.389E-04	1.773E-03	1.303E-02	1.883E-10
190	CDCDDCDBDBAC	1.000E+00	4.384E-01	3.406E-01	2.628E-01	1.827E-01	2.766E-02	1.057E-02	4.673E-02	1.850E-01	1.881E-10
191	CDCCCCDBBCAA	1.000E+00	1.608E-02	2.916E-03	1.550E-03	7.128E-04	1.214E-04	3.894E-05	1.928E-04	7.319E-04	1.880E-10
192	DHADDCBBDAAB	1.000E+00	4.237E-01	3.717E-01	2.512E-01	7.497E-02	1.305E-03	6.525E-03	7.347E-03	5.331E-02	1.843E-10
193	CFADDCBBDBAA	1.000E+00	2.917E-01	2.283E-01	1.864E-01	9.862E-02	1.133E-02	8.921E-03	2.153E-02	9.183E-02	1.804E-10
194	DDDDDCBBDAAA	1.000E+00	2.910E-02	5.161E-03	2.915E-03	8.379E-04	9.481E-05	7.168E-05	1.552E-04	7.450E-04	1.780E-10
195	CDDCBCABDCAA	1.000E+00	2.549E-02	4.580E-03	2.067E-03	3.099E-04	2.871E-05	2.755E-05	3.828E-05	2.582E-04	1.778E-10
196	GGCCBCDBDDAA	5.000E-03	2.470E-05	7.491E-10	3.020E-10	1.359E-11	5.678E-12	6.089E-13	1.935E-12	1.795E-11	1.695E-10
197	DHEDDCBBDAAB	1.000E+00	4.429E-01	3.922E-01	2.485E-01	8.863E-02	3.499E-03	8.421E-03	1.278E-02	6.826E-02	1.656E-10
198	CDCCACDBBBAA	1.000E+00	3.051E-01	2.740E-01	1.029E-01	8.714E-03	7.904E-03	1.504E-03	2.074E-03	1.109E-02	1.621E-10
199	CDDCACBBBCAA	1.000E+00	3.885E-02	1.473E-02	9.356E-03	4.041E-03	6.495E-04	2.309E-04	6.761E-04	4.084E-03	1.602E-10
200	EGACACBBBAAA	1.000E+00	1.062E-01	7.803E-02	5.503E-02	1.332E-02	2.543E-03	1.001E-03	2.207E-03	1.330E-02	1.598E-10
201	DDDCCCBBBAAA	1.000E+00	1.174E-01	1.002E-01	3.803E-02	4.927E-03	7.857E-03	1.947E-03	2.305E-03	6.074E-03	1.596E-10

202	CHECCCBBBCAA	1.000E+00	1.927E-01	1.467E-01	9.258E-02	5.305E-02	9.881E-03	5.439E-03	1.597E-02	5.282E-02	1.552E-10
203	CFDDBCABDCAB	1.000E+00	1.893E-01	1.339E-01	5.079E-02	3.611E-03	1.585E-03	1.886E-04	5.779E-04	4.760E-03	1.474E-10
204	CHEDDCBBDCAA	1.000E+00	2.019E-01	1.511E-01	1.144E-01	6.464E-02	8.239E-03	4.645E-03	1.486E-02	6.174E-02	1.445E-10
205	GGADBCABDDAA	5.000E-03	1.276E-04	4.076E-09	2.364E-09	9.660E-10	1.171E-10	7.696E-11	1.524E-10	8.985E-10	1.431E-10
206	CDCDDCDBDCAA	1.000E+00	2.142E-02	1.007E-03	4.761E-04	1.377E-04	2.633E-05	7.674E-06	3.270E-05	1.457E-04	1.378E-10
207	CHACBCABDCAA	1.000E+00	4.591E-01	4.825E-01	1.178E-01	6.053E-03	1.596E-03	3.495E-04	8.354E-04	7.063E-03	1.330E-10
208	EGDCBCABDAAA	1.000E+00	2.538E-02	6.978E-03	5.502E-03	4.797E-04	4.553E-05	4.519E-05	5.895E-05	4.023E-04	1.244E-10
209	CHADDCBBDBAD	1.000E+00	3.959E-01	3.574E-01	2.492E-01	8.797E-02	3.169E-03	8.180E-03	1.549E-02	6.539E-02	1.154E-10
210	DDCDDCDBDAAB	1.000E+00	2.949E-02	7.836E-03	5.074E-03	1.035E-03	2.415E-04	6.006E-05	2.863E-04	1.138E-03	1.151E-10
211	DHDCCCBBBAAA	1.000E+00	4.091E-01	3.412E-01	1.696E-01	6.982E-02	2.180E-02	8.536E-03	4.143E-02	7.307E-02	1.121E-10
212	DGCCACDBBAAA	1.000E+00	3.778E-01	3.632E-01	1.826E-01	4.108E-02	1.694E-02	3.585E-03	7.972E-03	4.683E-02	1.091E-10
213	DFCCACDBBAAA	1.000E+00	4.084E-01	4.015E-01	1.459E-01	6.450E-03	6.517E-03	1.114E-03	1.372E-03	8.682E-03	9.156E-11
214	CHECBCABDBAA	1.000E+00	4.921E-01	4.805E-01	2.997E-01	8.440E-02	7.969E-03	6.678E-03	1.038E-02	7.294E-02	8.772E-11
215	DHADDCBBDAAA	1.000E+00	3.213E-01	2.577E-01	1.824E-01	6.705E-02	6.964E-03	6.161E-03	1.337E-02	5.953E-02	7.537E-11
216	CHACACBBBCAA	1.000E+00	2.165E-01	1.732E-01	8.910E-02	1.928E-02	5.499E-03	2.111E-03	3.468E-03	1.977E-02	7.406E-11
217	DHADDCBBDAAC	1.000E+00	4.814E-01	4.379E-01	2.873E-01	9.628E-02	1.146E-03	8.592E-03	9.754E-03	6.816E-02	7.385E-11
218	CHECACBBBBAA	1.000E+00	3.348E-01	2.935E-01	1.488E-01	3.024E-02	9.088E-03	5.111E-03	5.355E-03	2.767E-02	7.363E-11
219	GHACCCBBBDAA	5.000E-03	4.989E-05	2.495E-07	7.072E-09	1.638E-11	4.282E-14	1.326E-13	[illegible]	3.679E-11	6.633E-11
220	CHDCBCABDBAA	1.000E+00	2.839E-01	2.338E-01	1.521E-01	3.368E-02	7.818E-03	1.785E-03	[illegible]	3.719E-02	6.385E-11
221	CGCCBCDBDBAA	1.000E+00	1.695E-01	1.415E-01	8.234E-02	1.761E-02	4.596E-03	8.115E-04	[illegible]	2.040E-02	6.274E-11
222	EFCCBCDBDAAA	1.000E+00	4.301E-03	3.453E-05	1.770E-05	1.473E-07	6.021E-08	6.465E-09	[illegible]	1.851E-07	6.113E-11
223	DHACCCBBBAAA	1.000E+00	4.371E-01	3.786E-01	2.072E-01	7.719E-02	1.733E-02	9.746E-03	[illegible]	7.320E-02	5.877E-11
224	DHADDCBBDAAD	1.000E+00	5.108E-01	4.530E-01	3.082E-01	9.995E-02	9.042E-04	8.839E-03	9.448E-03	7.026E-02	5.788E-11
225	CFACBCABDCAA	1.000E+00	1.512E-01	1.096E-01	3.600E-02	4.011E-03	1.369E-03	1.906E-04	6.373E-04	5.172E-03	5.590E-11
226	CFACACBBBCAA	1.000E+00	3.088E-01	2.691E-01	1.453E-01	1.927E-02	8.165E-03	1.421E-03	3.534E-03	2.224E-02	5.584E-11
227	DFADDCBBDAAA	1.000E+00	3.345E-01	2.545E-01	1.534E-01	4.465E-02	9.472E-03	2.588E-03	1.099E-02	4.724E-02	5.395E-11
228	CGCCACDBBBAA	1.000E+00	2.352E-01	1.977E-01	7.868E-02	5.750E-03	1.070E-02	2.114E-03	2.565E-03	8.486E-03	4.692E-11
229	CDDCCCBBBCAA	1.000E+00	2.187E-02	3.810E-03	1.988E-03	8.207E-04	1.243E-04	5.351E-05	2.081E-04	8.154E-04	4.678E-11
230	CGDCBCABDBAA	1.000E+00	2.332E-01	1.776E-01	1.066E-01	2.149E-02	4.608E-03	1.242E-03	3.359E-03	2.304E-02	4.623E-11
231	EFDCBCABDAAA	1.000E+00	1.383E-02	4.684E-04	7.934E-04	1.589E-04	1.638E-07	1.759E-05	1.820E-05	1.065E-04	4.501E-11
232	GGDDBCABDDAA	5.000E-03	1.269E-04	3.719E-09	2.360E-09	6.868E-10	3.281E-11	6.940E-11	8.737E-11	5.234E-10	4.369E-11
233	DGADDCBBDAAA	1.000E+00	3.472E-01	2.731E-01	2.094E-01	6.793E-02	9.647E-03	6.257E-03	1.520E-02	6.426E-02	4.116E-11
234	CGACBCABDCAA	1.000E+00	4.531E-01	4.757E-01	1.126E-01	5.045E-03	1.305E-03	2.969E-04	6.864E-04	5.905E-03	4.072E-11
235	CDCDBCDBDCAC	1.000E+00	3.422E-02	1.224E-02	7.491E-03	3.114E-03	5.463E-04	1.308E-04	5.170E-04	3.259E-03	3.924E-11
236	CDCCBCDBDBAA	1.000E+00	3.960E-01	3.481E-01	1.701E-01	2.147E-02	7.073E-03	9.986E-04	3.453E-03	2.576E-02	3.907E-11
237	CHDCCCBBBCAA	1.000E+00	1.680E-01	1.213E-01	7.783E-02	4.352E-02	9.193E-03	3.324E-03	1.429E-02	4.431E-02	3.744E-11
238	CDDDDCBBDBAB	1.000E+00	4.387E-01	3.415E-01	2.638E-01	1.841E-01	2.781E-02	1.076E-02	4.713E-02	1.863E-01	3.621E-11
239	DFACCCBBBAAA	1.000E+00	4.238E-01	3.411E-01	1.718E-01	7.268E-02	2.554E-02	8.240E-03	1.969E-02	7.770E-02	3.592E-11
240	DFADDCBBDAAB	1.000E+00	2.214E-01	1.427E-01	7.593E-02	1.923E-02	4.336E-03	1.074E-03	4.395E-03	2.072E-02	3.522E-11
241	DGADDCBBDAAB	1.000E+00	2.820E-01	1.990E-01	1.454E-01	2.870E-02	3.984E-03	2.139E-03	4.965E-03	2.631E-02	3.522E-11
242	CHEDDCBBDBAA	1.000E+00	3.239E-01	2.611E-01	1.642E-01	3.628E-02	2.168E-03	3.001E-03	4.368E-03	2.787E-02	3.463E-11
243	CHADDCBBDCAB	1.000E+00	1.668E-01	1.298E-01	1.041E-01	3.450E-02	4.320E-04	3.043E-03	3.272E-03	2.402E-02	3.427E-11
244	CDDDDCBBDBAC	1.000E+00	4.391E-01	3.419E-01	2.641E-01	1.844E-01	2.785E-02	1.078E-02	4.719E-02	1.865E-01	3.293E-11
245	CFCCBCDBDBAA	1.000E+00	2.651E-01	2.366E-01	1.571E-01	3.786E-02	9.678E-03	1.749E-03	6.290E-03	4.358E-02	3.189E-11
246	CFCCACDBBBAA	1.000E+00	3.097E-01	2.673E-01	1.243E-01	8.608E-03	1.495E-02	2.703E-03	3.307E-03	1.330E-02	3.072E-11
247	DFCCCCDBBAAA	1.000E+00	4.445E-01	3.766E-01	1.798E-01	2.920E-02	1.692E-02	3.541E-03	7.851E-03	3.690E-02	3.070E-11
248	EHECBCABDAAA	1.000E+00	4.835E-02	2.654E-02	3.059E-02	5.538E-03	4.101E-05	5.412E-04	5.564E-04	3.959E-03	3.054E-11
249	GFCDBCDBDDAA	5.000E-03	1.120E-04	2.312E-09	9.876E-10	7.100E-11	2.811E-11	3.272E-12	1.063E-11	9.144E-11	2.937E-11
250	EHACBCABDAAA	1.000E+00	9.711E-02	1.024E-01	4.782E-02	6.114E-04	9.904E-05	4.683E-05	5.489E-05	6.098E-04	2.898E-11
251	CGACACBBBCAA	1.000E+00	3.437E-01	3.055E-01	1.717E-01	3.422E-02	9.548E-03	3.172E-03	6.052E-03	3.550E-02	2.831E-11
252	DHDDDCBBDAAB	1.000E+00	2.913E-01	2.019E-01	1.044E-01	2.552E-02	3.016E-03	1.818E-03	5.437E-03	2.417E-02	2.704E-11
253	DGDCACBBBAAA	1.000E+00	3.986E-01	3.757E-01	1.888E-01	4.192E-02	1.701E-02	3.623E-03	8.088E-03	4.765E-02	2.677E-11
254	CHEDDCBBDCAB	1.000E+00	1.598E-01	1.259E-01	9.839E-02	3.324E-02	1.845E-04	2.959E-03	3.037E-03	2.281E-02	2.676E-11
255	CDDDBCABDCAC	1.000E+00	5.424E-02	1.564E-02	8.764E-03	3.948E-03	6.106E-04	2.297E-04	6.596E-04	3.954E-03	2.584E-11
256	CHEDDCBBDBAB	1.000E+00	3.262E-01	2.883E-01	1.905E-01	6.448E-02	2.104E-04	5.959E-03	6.072E-03	4.472E-02	2.552E-11
257	DGACCCBBBAAA	1.000E+00	4.671E-01	3.945E-01	2.549E-01	1.407E-01	2.612E-02	1.914E-02	3.193E-02	1.322E-01	2.551E-11
258	GGDDBCABDDAB	5.000E-03	6.899E-05	2.773E-09	1.463E-09	4.042E-10	3.886E-11	2.937E-11	5.423E-11	3.427E-10	2.519E-11
259	CDCDBCDBDCAD	1.000E+00	2.824E-02	8.698E-03	3.308E-03	1.458E-04	7.523E-05	5.903E-06	1.808E-05	1.930E-04	2.449E-11
260	DGCCCCDBBAAA	1.000E+00	4.970E-01	4.246E-01	2.200E-01	3.581E-02	1.848E-02	3.632E-03	8.429E-03	4.527E-02	2.449E-11
261	CHDDDCBBDCAA	1.000E+00	1.862E-01	1.278E-01	9.414E-02	5.662E-02	8.636E-03	3.317E-03	1.431E-02	5.717E-02	2.370E-11
262	CFDCBCABDBAA	1.000E+00	3.208E-01	2.681E-01	1.693E-01	3.925E-02	9.708E-03	1.893E-03	6.455E-03	4.457E-02	2.350E-11
263	CDDCBCABDBAA	1.000E+00	4.539E-01	4.131E-01	2.141E-01	2.699E-02	8.896E-03	1.257E-03	4.344E-03	3.223E-02	2.296E-11
264	EFCCACDBBAAA	1.000E+00	1.392E-02	7.406E-05	4.014E-05	1.434E-05	2.579E-06	6.041E-07	2.386E-06	1.500E-05	2.276E-11
265	CDDDDCBBDCAA	1.000E+00	2.673E-02	1.352E-03	8.449E-04	2.937E-04	2.193E-05	2.475E-05	4.877E-05	2.378E-04	2.266E-11
266	EHDCBCABDAAA	1.000E+00	3.138E-02	7.382E-03	5.785E-03	3.942E-04	3.184E-05	3.816E-05	4.598E-05	3.221E-04	2.209E-11
267	DFDCACBBBAAA	1.000E+00	4.240E-01	4.125E-01	1.497E-01	6.391E-03	6.045E-03	1.013E-03	1.251E-03	8.437E-03	2.198E-11
268	CDDCACBBBBAA	1.000E+00	3.950E-01	3.687E-01	1.409E-01	3.921E-03	3.839E-03	5.749E-04	6.063E-04	5.212E-03	2.027E-11
269	GFDDBCABDDAA	5.000E-03	1.734E-04	3.409E-09	1.895E-09	4.176E-10	3.184E-11	5.538E-11	6.258E-11	3.449E-10	1.914E-11

270	CGCCBCDBDCAA	1.000E+00	8.685E-02	6.661E-02	3.159E-02	4.070E-03	1.385E-03	1.955E-04	6.623E-04	5.034E-03	1.891E-11
271	DDDDDCBBDAAB	1.000E+00	3.413E-02	8.573E-03	5.582E-03	1.192E-03	2.464E-04	7.388E-05	3.160E-04	1.254E-03	1.854E-11
272	CHDCACBBBBAA	1.000E+00	2.900E-01	2.416E-01	1.099E-01	6.477E-03	8.170E-03	1.832E-03	2.003E-03	8.330E-03	1.747E-11
273	CDDDBCABDCAD	1.000E+00	3.748E-02	1.161E-02	8.513E-03	2.600E-03	8.404E-05	2.377E-04	2.598E-04	1.730E-03	1.571E-11
274	GGCCACDB3DAA	5.000E-03	4.329E-05	8.486E-09	5.711E-09	2.389E-09	4.198E-10	1.004E-10	3.975E-10	2.491E-09	1.482E-11
275	DFCDDCDBDAAA	1.000E+00	4.733E-01	3.774E-01	2.696E-01	1.467E-01	2.453E-02	8.415E-03	3.662E-02	1.510E-01	1.432E-11
276	CGDCBCABDCAA	1.000E+00	8.700E-02	6.100E-02	2.843E-02	3.696E-03	1.004E-03	2.089E-04	5.705E-04	4.201E-03	1.330E-11
277	CDCDDCDBDCAB	1.000E+00	2.264E-02	2.114E-03	1.124E-03	1.452E-04	4.477E-05	6.735E-06	2.375E-05	1.709E-04	1.305E-11
278	CHADDCBBDCAA	1.000E+00	1.532E-01	1.007E-01	6.982E-02	2.121E-02	2.623E-03	1.566E-03	3.873E-03	1.906E-02	1.290E-11
279	CHADDCBBDCAC	1.000E+00	1.834E-01	1.525E-01	1.302E-01	4.753E-02	2.030E-04	4.232E-03	4.540E-03	3.274E-02	1.289E-11
280	CGCCACDBBCAA	1.000E+00	1.691E-01	1.463E-01	7.789E-02	2.169E-02	8.534E-03	1.859E-03	4.320E-03	2.429E-02	1.257E-11
281	CGDCACBBBBAA	1.000E+00	2.511E-01	2.024E-01	8.469E-02	6.269E-03	1.043E-02	2.270E-03	2.626E-03	8.513E-03	1.104E-11
282	EFCCCCDBBAAA	1.000E+00	1.275E-02	8.967E-06	8.382E-06	5.433E-07	8.223E-08	3.143E-08	1.390E-07	5.500E-07	1.053E-11
283	CDCCCCDBBBAA	1.000E+00	3.517E-01	2.887E-01	1.574E-01	3.598E-02	1.070E-02	2.738E-03	1.207E-02	4.018E-02	1.036E-11
284	CHADDCBBDCAD	1.000E+00	1.829E-01	1.523E-01	1.302E-01	4.693E-02	2.832E-05	4.203E-03	4.286E-03	3.195E-02	1.030E-11
285	CHACCCBBBCAA	1.000E+00	1.649E-01	1.464E-01	5.462E-02	1.634E-02	6.079E-03	1.874E-03	4.183E-03	1.652E-02	1.013E-11
286	GFCDBCDBDDAB	5.000E-03	8.537E-05	2.093E-09	8.975E-10	3.384E-11	2.466E-11	1.633E-12	4.340E-12	5.013E-11	9.997E-12
287	DGCDDCDBDAAA	1.000E+00	4.526E-01	3.935E-01	1.875E-01	3.248E-02	8.438E-03	1.717E-03	6.640E-03	3.743E-02	9.910E-12
288	CFADDCBBDCAA	1.000E+00	2.167E-01	1.585E-01	8.988E-02	1.417E-02	4.286E-03	8.367E-04	3.394E-03	1.588E-02	9.738E-12
289	CGCDDCDBDBAB	1.000E+00	1.065E-01	6.064E-02	1.014E-02	2.068E-04	1.260E-04	5.653E-06	1.095E-05	3.632E-04	9.717E-12
290	CFCCACDBBCAA	1.000E+00	1.451E-01	1.178E-01	4.569E-02	2.388E-03	3.960E-03	7.306E-04	8.350E-04	3.613E-03	9.704E-12
291	CHECCCBBBBAA	1.000E+00	5.092E-01	4.933E-01	4.098E-01	3.604E-01	6.751E-02	4.998E-02	1.830E-01	3.582E-01	9.342E-12
292	CGADDCBBDBAD	1.000E+00	2.529E-01	1.694E-01	1.084E-01	4.184E-02	1.794E-03	5.048E-03	5.900E-03	3.520E-02	9.197E-12
293	EHACACBBBAAA	1.000E+00	1.059E-01	8.057E-02	5.878E-02	2.094E-02	3.445E-03	1.536E-03	3.517E-03	2.059E-02	8.228E-12
294	EFACCCBBBAAA	1.000E+00	9.740E-03	3.473E-05	1.484E-05	1.106E-08	1.043E-11	6.109E-09	4.663E-09	1.102E-08	7.549E-12
295	CFCCBCDBDCAA	1.000E+00	1.440E-01	1.073E-01	3.816E-02	7.614E-03	1.907E-03	3.461E-04	1.246E-03	8.992E-03	7.301E-12
296	CGCDDCDBDBAA	1.000E+00	1.757E-01	1.219E-01	4.083E-02	9.176E-04	5.995E-04	2.393E-05	4.413E-05	1.420E-03	7.298E-12
297	DFDCCCBBBAAA	1.000E+00	4.576E-01	3.874E-01	1.969E-01	3.544E-02	1.779E-02	4.432E-03	8.814E-03	4.241E-02	7.083E-12
298	CGADDCBBDCAA	1.000E+00	1.491E-01	9.274E-02	6.002E-02	1.753E-02	3.203E-03	1.159E-03	4.033E-03	1.766E-02	6.911E-12
299	CFCCCCDBBBAA	1.000E+00	3.544E-01	3.162E-01	2.059E-01	1.547E-01	3.600E-02	1.717E-02	1.034E-01	1.563E-01	6.840E-12
300	GFDDBCABDDAB	5.000E-03	1.392E-04	2.884E-09	1.233E-09	6.712E-11	2.570E-11	4.431E-12	7.465E-12	7.199E-11	6.829E-12
301	DFADDCBBDAAC	1.000E+00	2.744E-01	1.876E-01	1.211E-01	5.490E-02	9.733E-03	3.110E-03	1.335E-02	5.698E-02	6.705E-12
302	CFDCACBBBBAA	1.000E+00	3.096E-01	2.616E-01	1.403E-01	1.451E-02	1.687E-02	3.918E-03	4.577E-03	1.875E-02	6.554E-12
303	CFADDCBBDCAB	1.000E+00	1.288E-01	7.262E-02	3.577E-02	5.399E-03	1.751E-03	3.177E-04	1.256E-03	6.198E-03	6.328E-12
304	EGACCCBBBAAA	1.000E+00	2.356E-02	1.464E-02	1.003E-03	1.479E-06	7.500E-09	1.350E-07	1.156E-07	3.025E-06	6.207E-12
305	DGDCCCBBBAAA	1.000E+00	5.054E-01	4.311E-01	2.279E-01	3.792E-02	1.800E-02	4.019E-03	7.033E-03	4.620E-02	5.936E-12
306	DGADDCBBDAAC	1.000E+00	3.674E-01	2.478E-01	1.984E-01	6.929E-02	9.674E-03	6.104E-03	1.555E-02	6.667E-02	5.568E-12
307	DGCDDCDBDAAB	1.000E+00	2.301E-01	1.780E-01	5.091E-02	1.337E-03	3.990E-04	3.197E-05	8.194E-05	1.969E-03	5.491E-12
308	CHDDDCBBDBAA	1.000E+00	2.479E-01	1.630E-01	8.515E-02	4.338E-03	1.059E-03	3.131E-04	3.636E-04	4.008E-03	5.393E-12
309	CDCDDCDBDBAA	1.000E+00	1.352E-01	9.284E-02	5.451E-02	3.193E-02	4.841E-03	1.840E-03	8.127E-03	3.243E-02	5.390E-12
310	CFDCBCABDCAA	1.000E+00	1.462E-01	1.061E-01	3.802E-02	7.715E-03	1.935E-03	3.511E-04	1.264E-03	9.101E-03	5.315E-12
311	CGADDCBBDCAB	1.000E+00	1.334E-01	7.993E-02	4.643E-02	7.238E-03	1.715E-03	4.467E-04	1.392E-03	7.394E-03	5.152E-12
312	EFDCACBBBAAA	1.000E+00	1.826E-02	1.198E-04	2.977E-04	4.495E-04	5.988E-05	9.855E-05	9.854E-05	4.156E-04	5.017E-12
313	CFCDDCDBDBAB	1.000E+00	1.170E-01	6.703E-02	9.288E-03	2.094E-04	2.032E-04	8.894E-06	1.898E-05	3.795E-04	4.722E-12
314	CFADDCBBDBAD	1.000E+00	2.516E-01	1.687E-01	1.115E-01	4.416E-02	1.793E-03	5.349E-03	6.197E-03	3.696E-02	4.505E-12
315	CHDDDCBBDCAB	1.000E+00	9.674E-02	4.548E-02	2.859E-02	3.252E-03	2.008E-05	3.126E-04	3.263E-04	2.283E-03	4.161E-12
316	CFACCCBBBCAA	1.000E+00	1.655E-01	1.142E-01	5.995E-02	2.668E-02	8.488E-03	2.526E-03	6.963E-03	2.838E-02	3.792E-12
317	DGADDCBBDAAD	1.000E+00	5.438E-01	3.272E-01	2.890E-01	6.197E-02	6.532E-03	4.773E-03	8.314E-03	5.596E-02	3.599E-12
318	CHDDDCBBDBAB	1.000E+00	2.009E-01	1.252E-01	6.217E-02	6.704E-03	2.811E-05	6.452E-04	6.717E-04	4.895E-03	3.593E-12
319	EGDCACBBBAAA	1.000E+00	6.357E-02	4.396E-02	2.955E-02	1.122E-02	2.114E-03	5.576E-04	1.892E-03	1.170E-02	3.371E-12
320	CFCCCCDBBCAA	1.000E+00	1.653E-01	1.178E-01	6.757E-02	2.587E-02	7.214E-03	2.031E-03	6.534E-03	2.765E-02	3.367E-12
321	CGCCCCDBBBAA	1.000E+00	4.612E-01	4.217E-01	3.181E-01	2.523E-01	5.677E-02	2.747E-02	1.680E-01	2.550E-01	2.868E-12
322	CFCDDCDBDBAA	1.000E+00	1.369E-01	8.080E-02	3.115E-02	1.005E-02	2.526E-03	1.007E-03	6.362E-03	1.060E-02	2.813E-12
323	DFCDDCDBDAAB	1.000E+00	2.[illegible]31E-01	2.200E-01	1.311E-01	5.823E-02	1.013E-02	3.360E-03	1.466E-02	5.996E-02	2.796E-12
324	EHACCCBBBAAA	1.000E+00	2.332E-02	1.462E-02	7.223E-04	1.253E-06	9.336E-10	1.908E-09	4.764E-10	2.865E-06	2.595E-12
325	CGDCACBBBCAA	1.000E+00	1.925E-01	1.559E-01	8.373E-02	2.500E-02	9.275E-03	2.141E-03	4.893E-03	2.773E-02	2.592E-12
326	EFDCCCBBBAAA	1.000E+00	1.405E-02	1.065E-05	8.741E-06	1.447E-06	8.596E-08	3.490E-07	4.225E-07	1.171E-06	2.571E-12
327	CGACCCBBBCAA	1.000E+00	1.856E-01	1.299E-01	7.715E-02	4.148E-02	9.345E-03	4.064E-03	1.017E-02	4.179E-02	2.314E-12
328	CHDCCCBBBBAA	1.000E+00	4.157E-01	3.824E-01	3.003E-01	2.656E-01	5.922E-02	2.907E-02	1.761E-01	2.666E-01	2.093E-12
329	CDDDDCBBDCAB	1.000E+00	2.417E-02	1.780E-03	7.662E-04	9.895E-05	1.460E-05	8.581E-06	1.057E-05	8.050E-05	2.068E-12
330	DFDDDCBBDAAA	1.000E+00	5.298E-01	4.289E-01	3.145E-01	1.801E-01	2.951E-02	1.035E-02	4.498E-02	1.847E-01	2.033E-12
331	DFADDCBBDAAD	1.000E+00	2.946E-01	1.807E-01	1.354E-01	3.841E-02	8.545E-03	1.745E-03	6.371E-03	4.261E-02	1.580E-12
332	GGCCCCDBBDAA	5.000E-03	3.510E-05	8.834E-11	7.473E-11	1.935E-11	2.926E-12	1.119E-12	4.951E-12	1.960E-11	1.553E-12
333	CGDDDCBBDBAB	1.000E+00	1.329E-01	8.145E-02	3.547E-02	4.782E-03	1.795E-04	7.239E-04	7.377E-04	3.870E-03	1.427E-12
334	CFDCCCBBBBAA	1.000E+00	3.569E-01	3.199E-01	2.215E-01	1.815E-01	3.995E-02	2.004E-02	1.234E-01	1.820E-01	1.421E-12
335	CGCCCCDBBCAA	1.000E+00	2.104E-01	1.582E-01	7.730E-02	1.155E-02	6.882E-03	1.320E-03	2.284E-03	1.499E-02	1.336E-12
336	CDDCCCBBBBAA	1.000E+00	4.745E-01	4.052E-01	2.402E-01	3.489E-02	1.234E-02	1.795E-03	5.685E-03	4.295E-02	1.118E-12
337	EHECACBBBAAA	1.000E+00	7.944E-02	6.096E-02	4.612E-02	2.349E-02	3.298E-03	1.519E-03	3.873E-03	2.281E-02	1.061E-12

338	CFCDDCDBDCAA	1.000E+00	2.386E-01	1.787E-01	1.376E-01	9.622E-02	1.456E-02	5.565E-03	2.461E-02	9.738E-02	1.056E-12
339	CFDCACBBBCAA	1.000E+00	1.380E-01	1.113E-01	5.136E-02	4.402E-03	6.350E-03	1.542E-03	1.615E-03	6.152E-03	9.184E-13
340	DGDDDCBBDAAA	1.000E+00	4.332E-01	3.684E-01	2.149E-01	3.237E-02	1.107E-02	1.723E-03	5.204E-03	3.938E-02	8.275E-13
341	CFDDDCBBDBAB	1.000E+00	1.293E-01	7.713E-02	4.005E-02	6.317E-03	1.590E-04	1.007E-03	1.022E-03	5.016E-03	8.170E-13
342	CGDDDCBBDBAA	1.000E+00	1.887E-01	1.218E-01	5.509E-02	2.245E-03	7.899E-04	1.657E-04	1.970E-04	2.360E-03	7.984E-13
343	DDCDDCDBDAAC	1.000E+00	2.043E-02	1.520E-03	8.613E-04	5.092E-04	7.743E-05	2.933E-05	1.296E-04	5.165E-04	6.573E-13
344	CGCDDCDBDCAB	1.000E+00	6.826E-02	3.783E-02	9.678E-03	2.037E-04	1.08[illegible]E-04	4.898E-06	9.255E-06	3.254E-04	5.895E-13
345	DGDDDCBBDAAB	1.000E+00	1.691E-01	1.052E-01	3.593E-02	3.370E-03	9.76[illegible] 05	3.675E-04	3.790E-04	2.709E-03	5.582E-13
346	CFDCCCBBBCAA	1.000E+00	1.254E-01	7.911E-02	2.801E-02	4.220E-03	4.517E-03	1.133E-03	1.135E-03	5.568E-03	3.970E-13
347	EHECCCBBBAAA	1.000E+00	6.520E-02	4.328E-02	5.347E-02	3.567E-03	3.926E-08	1.710E-04	1.838E-04	2.225E-03	3.948E-13
348	CGCDDCDBDCAA	1.000E+00	6.748E-02	3.820E-02	8.975E-03	1.716E-04	4.700E-05	2.215E-06	2.982E-06	2.701E-04	3.511E-13
349	DDCDDCDBDAAD	1.000E+00	1.050E-01	5.475E-02	4.773E-02	9.739E-03	2.183E-03	4.366E-04	1.622E-03	1.083E-02	3.224E-13
350	EHDCACBBBAAA	1.000E+00	8.544E-02	6.572E-02	4.622E-02	2.089E-02	3.591E-03	8.774E-04	3.482E-03	2.173E-02	2.426E-13
351	CGDCCCBBBBAA	1.000E+00	8.349E-01	8.329E-01	8.284E-01	8.224E-01	1.698E-01	8.838E-02	5.605E-01	8.224E-01	2.124E-13
352	CFADDCBBDCAC	1.000E+00	2.190E-01	1.624E-01	1.227E-01	8.496E-02	1.286E-02	4.914E-03	2.173E-02	8.598E-02	1.957E-13
353	CFCDDCDBDCAB	1.000E+00	6.807E-02	3.705E-02	1.223E-02	3.266E-04	3.191F-04	1.419E-05	3.052E-05	5.108E-04	1.603E-13
354	CGADDCBBDCAC	1.000E+00	2.386E-01	1.787E-01	1.396E-01	1.008E-01	1.458E-02	6.972E-03	2.585E-02	1.006E-01	1.304E-13
355	DFDDDCBBDAAB	1.000E+00	1.156E-01	6.479E-02	2.142E-02	5.729E-04	5.577E-04	2.489E-05	5.344E-05	8.940E-04	1.106E-13
356	EHDCCCBBBAAA	1.000E+00	4.377E-02	1.583E-02	2.311E-02	6.845E-04	5.077E-09	2.971E-05	3.325E-05	4.073E-04	9.869E-14
357	CGDDDCBBDCAB	1.000E+00	7.405E-02	4.046E-02	9.144E-03	1.821E-04	5.307E-05	2.496E-06	3.362E-06	2.721E-04	6.606E-14
358	CDCDDCDBDBAD	1.000E+00	8.056E-02	4.654E-02	1.191E-02	2.405E-04	5.101E-05	3.137E-06	4.061E-06	3.740E-04	6.564E-14
359	CGDDDCBBDCAA	1.000E+00	7.405E-02	4.046E-02	9.144E-03	1.821E-04	5.307E-05	2.496E-06	3.362E-06	2.721E-04	5.074E-14
360	EGDCCCBBBAAA	1.000E+00	3.595E-02	1.560E-02	1.543E-02	2.629E-06	3.522E-07	2.556E-07	9.770E-07	2.566E-06	3.691E-14

Table C.4

	Fraction of STs with Non-Zero Frequency Among 100 Observations	Source Term Group Representation Among 100 Observations																									
		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	Frequency
CDCCACDBBBAA	0.28	0	0	0	0	0	0	0	0	0	0	0	0	0	3	0	2	5	7	7	3	1	0	0	0	0	1.62E-10
CDCCACDBBCAA	0.57	6	2	8	4	4	4	4	5	5	3	3	4	1	1	1	1	0	1	0	0	0	0	0	0	0	6.60E-10
CDCCBCDBDBAA	0.08	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	3	0	2	1	1	0	0	0	0	3.91E-11
CDCCBCDBDCAA	0.24	2	1	0	1	3	3	3	2	3	1	3	0	0	2	0	0	0	0	0	0	0	0	0	0	0	2.50E-10
CDCCCCDBBBAA	0.06	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	2	0	0	2	0	0	0	0	1	1.04E-11
CDCCCCDBBCAA	0.13	2	1	0	0	0	3	2	1	0	0	0	0	3	0	0	1	0	0	0	0	0	0	0	0	0	1.88E-10
CDCCFCDBDBAA	0.94	3	1	1	1	0	2	0	2	6	3	5	5	10	8	10	8	9	13	4	1	1	0	1	0	0	6.48E-09
CDCCFCDBDCAA	0.70	15	4	4	5	10	5	8	3	6	4	2	3	0	0	0	1	0	0	0	0	0	0	0	0	0	3.79E-10
CDCDBCDBDBAA	0.91	0	0	1	0	0	0	0	0	1	0	2	4	1	6	10	10	11	13	8	13	9	0	1	0	1	3.53E-09
CDCDBCDBDBAB	0.96	0	0	1	0	0	0	0	0	1	0	2	4	2	7	9	16	7	14	8	15	8	1	0	0	1	7.54E-09
CDCDBCDBDBAC	0.94	0	0	1	0	0	0	0	0	1	0	3	4	1	7	9	16	7	14	7	14	8	1	0	0	1	6.08E-09
CDCDBCDBDBAD	0.59	0	1	0	0	0	0	0	0	0	0	1	1	0	7	4	11	4	7	6	10	6	0	0	0	1	6.06E-10
CDCDBCDBDCAA	0.99	5	5	1	9	9	12	9	8	8	6	9	4	5	4	2	1	1	0	1	0	0	0	0	0	0	1.70E-08
CDCDBCDBDCAB	0.98	8	1	5	7	10	11	8	9	8	4	10	3	5	4	2	1	1	1	0	0	0	0	0	0	0	5.67E-09
CDCDBCDBDCAC	0.33	1	1	2	2	3	4	6	1	4	1	3	1	1	2	0	0	0	1	0	0	0	0	0	0	0	3.92E-11
CDCDBCDBDCAD	0.22	4	1	1	1	1	0	6	1	1	0	3	2	1	0	0	0	0	0	0	0	0	0	0	0	0	2.45E-11
CDCDDCDBDBAA	0.04	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	0	0	0	0	0	1	0	0	0	0	5.39E-12
CDCDDCDBDBAB	0.05	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	0	0	0	1	0	1	0	0	0	0	2.08E-10
CDCDDCDBDBAC	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	1	0	1	0	0	0	0	1.88E-10
CDCDDCDBDBAD	0.02	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	0	0	0	0	0	0	0	0	0	0	6.56E-14
CDCDDCDBDCAA	0.16	2	1	0	1	1	4	0	1	1	2	2	0	1	0	0	0	0	0	0	0	0	0	0	0	0	1.38E-10
CDCDDCDBDCAB	0.08	0	0	1	0	0	2	0	1	1	1	1	0	1	0	0	0	0	0	0	0	0	0	0	0	0	1.31E-11
CDCDFCDBDBAA	1.00	1	0	0	0	0	0	1	0	1	5	3	7	9	14	13	10	9	11	10	4	1	0	0	1	0	1.24E-07
CDCDFCDBDBAB	1.00	1	0	0	0	0	0	1	0	1	5	3	7	11	14	12	10	8	15	9	1	1	0	0	1	0	2.78E-07
CDCDFCDBDBAC	1.00	1	0	0	0	0	0	1	0	1	5	3	8	11	14	11	12	7	14	10	0	1	0	1	0	0	2.16E-07
CDCDFCDBDBAD	0.98	1	0	0	0	0	0	1	0	1	5	4	6	11	13	11	10	8	14	11	1	0	0	1	0	0	3.79E-08
CDCDFCDBDCAA	0.94	9	9	8	10	9	8	7	7	5	6	5	5	2	2	0	1	1	0	0	0	0	0	0	0	0	7.70E-09
CDCDFCDBDCAB	0.93	12	10	10	7	7	7	7	6	8	3	5	5	2	2	1	0	1	0	0	0	0	0	0	0	0	7.10E-09
CDCDFCDBDCAC	0.92	17	7	8	7	7	8	6	5	9	2	6	5	1	2	1	0	1	0	0	0	0	0	0	0	0	2.13E-09
CDCDFCDBDCAD	0.90	22	5	10	5	8	6	5	3	9	2	5	5	1	2	1	0	1	0	0	0	0	0	0	0	0	1.68E-09
CDDCACBBBBAA	0.09	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	1	1	3	1	2	0	0	0	0	0	2.03E-11
CDDCACBBBCAA	0.45	1	1	4	3	4	2	3	6	4	4	6	4	0	1	0	1	0	1	0	0	0	0	0	0	0	1.50E-10
CDDCBCABDBAA	0.07	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	1	1	0	3	1	0	0	0	0	0	2.30E-11
CDDCBCABDCAA	0.24	0	1	0	1	1	3	1	2	4	2	2	3	1	2	1	0	0	0	0	0	0	0	0	0	0	1.78E-10
CDDCCCBBBBAA	0.01	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	1.12E-12
CDDCCCBBBCAA	0.12	0	1	0	1	0	2	1	1	1	1	0	0	1	2	0	1	0	0	0	0	0	0	0	0	0	4.68E-11
CDDDBCABDBAA	0.90	0	0	0	0	0	0	0	0	0	0	1	0	1	4	7	11	17	13	11	14	9	0	1	0	1	2.60E-09
CDDDBCABDBAB	0.94	0	0	0	0	0	0	0	0	0	0	1	0	1	4	9	14	17	11	12	14	9	1	0	0	1	5.57E-09
CDDDBCABDBAC	0.92	0	0	0	0	0	0	0	0	0	0	1	0	2	3	10	18	13	13	8	13	9	1	0	0	1	4.47E-09
CDDDBCABDBAD	0.54	0	0	0	0	0	0	0	0	0	0	0	0	1	2	4	10	9	8	5	9	5	0	0	0	1	4.25E-10
CDDDBCABDCAA	0.99	1	2	0	2	3	17	8	9	9	9	4	10	12	4	5	1	2	0	1	0	0	0	0	0	0	1.25E-08
CDDDBCABDCAB	0.98	0	2	2	1	8	14	6	11	9	6	6	9	11	6	3	1	2	0	1	0	0	0	0	0	0	4.18E-09
CDDDBCABDCAC	0.26	0	0	2	0	3	3	4	1	2	1	1	2	2	4	0	0	0	1	0	0	0	0	0	0	0	2.58E-11
CDDDBCABDCAD	0.20	0	2	0	1	1	1	4	1	0	1	3	2	2	2	0	0	0	0	0	0	0	0	0	0	0	1.57E-11
CDDDDCBBDBAB	0.02	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	1	0	0	0	0	3.62E-11
CDDDDCBBDBAC	0.01	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	3.29E-11
CDDDDCBBDCAA	0.07	0	1	0	1	0	2	0	0	1	1	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.27E-11
CDDDDCBBDCAB	0.04	0	0	0	0	0	1	0	0	1	1	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.07E-12

	Fraction of STs with Non-Zero Frequency Among 100 Observations	Source Term Group Representation Among 100 Observations																									
		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	Frequency
CFACACBBBBAA	0.53	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	1	4	9	15	15	4	2	0	0	0	1.58E-09
CFACACBBBCAA	0.25	0	0	0	0	0	0	0	0	0	2	1	0	0	4	2	5	5	2	1	2	1	0	0	0	0	5.58E-11
CFACBCABDBAA	0.24	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	6	3	5	7	1	0	0	0	8.25E-10
CFACBCABDCAA	0.14	0	0	0	0	0	0	0	0	0	0	1	1	2	0	2	2	1	3	2	0	0	0	0	0	0	5.59E-11
CFACCCBBBBAA	0.10	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	1	0	2	0	3	1	1	0	0	1	2.39E-10
CFACCCBBBCAA	0.06	0	0	0	0	0	0	0	0	0	0	0	1	0	0	1	0	2	1	1	0	0	0	0	0	0	3.79E-12
CFADBCABDBAA	0.99	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	8	12	21	21	26	2	4	1	1	5.12E-08
CFADBCABDBAB	1.00	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	4	6	15	20	21	25	4	2	1	1	1.06E-07
CFADBCABDBAC	1.00	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	4	7	14	23	19	24	5	2	0	1	6.64E-08
CFADBCABDBAD	0.92	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	4	7	14	19	18	23	4	2	0	1	9.47E-09
CFADBCABDCAA	0.91	0	0	0	0	0	0	1	2	0	2	3	4	10	11	13	14	8	9	8	2	2	1	0	1	0	2.63E-09
CFADBCABDCAB	0.91	0	0	0	0	0	0	1	1	1	1	3	6	8	12	13	14	9	7	9	2	3	0	0	1	0	1.90E-09
CFADBCABDCAC	0.83	0	0	0	0	0	0	1	1	1	1	4	4	9	11	13	11	7	7	7	3	2	0	0	1	0	7.12E-10
CFADBCABDCAD	0.60	0	0	0	0	0	1	1	0	0	2	5	1	5	12	10	7	4	4	3	2	2	0	0	1	0	3.13E-10
CFADDCBBDBAA	0.14	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	2	3	2	4	0	0	0	0	1.80E-10
CFADDCBBDBAB	0.17	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	3	3	3	4	3	0	0	0	0	1.08E-09
CFADDCBBDBAC	0.12	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	3	1	2	4	1	0	0	0	0	7.40E-10
CFADDCBBDBAD	0.04	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	1	0	0	0	0	0	0	4.50E-12
CFADDCBBDCAA	0.06	0	0	0	0	0	0	0	0	0	0	0	0	0	2	1	1	0	0	1	0	1	0	0	0	0	9.74E-12
CFADDCBBDCAB	0.05	0	0	0	0	0	0	0	0	0	0	0	0	1	2	1	0	0	0	1	0	0	0	0	0	0	6.33E-12
CFADDCBBDCAC	0.02	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	1	0	0	0	0	0	0	1.96E-13
CFCCACDBBBAA	0.26	0	0	0	0	0	0	0	0	0	0	0	1	0	3	0	1	6	6	6	2	1	0	0	0	0	3.07E-11
CFCCACDBBCAA	0.19	0	0	0	0	0	0	0	0	0	2	1	1	0	1	2	5	3	2	1	1	0	0	0	0	0	9.70E-12
CFCCBCDBDBAA	0.08	0	0	0	0	0	0	0	0	0	0	1	1	0	0	0	1	1	1	1	1	1	0	0	0	0	3.19E-11
CFCCBCDBDCAA	0.07	0	0	0	0	0	0	0	0	0	0	0	0	1	0	2	1	1	1	1	0	0	0	0	0	0	7.30E-12
CFCCCCDBBBAA	0.07	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	0	1	2	0	0	0	0	1	6.84E-12
CFCCCCDBBCAA	0.05	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	2	1	1	0	0	0	0	0	0	3.37E-12
CFCDBCDBDBAA	0.85	0	0	1	0	0	0	0	0	1	0	2	3	2	4	9	10	10	13	8	12	9	0	0	0	1	1.01E-09
CFCDBCDBDBAB	0.74	0	0	1	0	0	0	0	0	1	0	2	1	2	4	8	14	6	12	7	10	5	0	0	0	1	8.25E-10
CFCDBCDBDCAA	0.75	0	0	0	0	0	1	0	1	3	4	2	3	5	8	11	10	10	5	8	1	2	0	0	1	0	3.53E-10
CFCDBCDBDCAB	0.62	0	0	0	0	0	1	1	0	3	3	1	4	3	6	10	11	6	5	7	0	1	0	0	0	0	2.01E-10
CFCDDCDBDBAA	0.05	0	0	0	0	0	0	0	0	0	0	0	1	0	0	1	1	0	0	0	1	0	0	0	0	1	2.81E-12
CFCDDCDBDBAB	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	0	0	0	0	1	0	0	0	0	0	4.72E-12
CFCDDCDBDCAA	0.01	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	1.06E-12
CFCDDCDBDCAB	0.01	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0	0	1.60E-13
CFDCACBBBBAA	0.13	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	4	2	3	1	0	0	0	0	6.55E-12
CFDCACBBBCAA	0.04	0	0	0	0	0	0	0	0	0	1	0	0	0	1	0	0	1	0	0	1	0	0	0	0	0	9.18E-13
CFDCBCABDBAA	0.08	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	0	2	0	2	1	0	0	0	0	2.35E-11
CF[illegible]BCABDCAA	0.05	0	0	0	0	0	0	0	0	0	0	0	0	1	0	1	1	1	0	1	0	0	0	0	0	0	5.32E-12
CFDCCCBBBBAA	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	0	0	0	0	0	0	1	1.42E-12
CFDCCCBBBCAA	0.02	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	1	0	0	0	0	0	0	0	0	3.97E-13
CFDDBCABDBAA	0.84	0	0	0	0	0	0	0	0	0	0	1	0	1	3	6	10	17	12	11	13	9	0	0	0	1	7.43E-10
CFDDBCABDBAB	0.72	0	0	0	0	0	0	0	0	0	0	1	0	1	3	7	9	15	8	11	11	5	0	0	0	1	6.08E-10
CFDDBCABDCAA	0.72	0	0	0	0	0	0	1	1	0	1	2	3	7	6	13	12	8	7	7	1	2	0	0	1	0	2.59E-10
CFDDBCABDCAB	0.59	0	0	0	0	0	0	1	1	1	1	2	4	1	8	12	10	5	6	6	0	1	0	0	0	0	1.47E-10
CFDDDCBBDBAB	0.02	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	0	0	0	0	0	0	0	0	0	8.17E-13
CGACACBBBBAA	0.53	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	1	5	9	15	14	4	2	0	0	0	1.81E-09
CGACACBBBCAA	0.20	0	0	0	0	0	0	0	0	0	0	2	0	1	1	0	1	7	1	5	2	0	0	0	0	0	2.83E-11

	Fraction of STs with Non-Zero Frequency Among 100 Observations	Source Term Group Representation Among 100 Observations																									
		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	Frequency
CGACBCABDBAA	0.24	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	6	2	6	7	1	0	0	0	9.85E-10
CGACBCABDCAA	0.11	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	1	2	2	3	1	0	0	0	0	0	4.07E-11
CGACCCBBBBAA	0.10	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	1	0	2	0	3	1	1	0	0	1	2.79E-10
CGACCCBBBCAA	0.05	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	1	0	1	2	0	0	0	0	0	0	2.31E-12
CGADBCABDBAA	0.99	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	8	12	21	21	26	2	4	1	1	5.95E-08
CGADBCABDBAB	1.00	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	4	6	15	20	21	25	4	2	1	1	1.58E-07
CGADBCABDBAC	1.00	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	4	7	14	23	19	24	5	2	0	1	1.09E-07
CGADBCABDBAD	0.93	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	4	7	14	19	19	23	4	2	0	1	1.90E-08
CGADBCABDCAA	0.88	0	0	0	0	0	0	0	0	0	0	1	2	3	4	9	15	11	13	13	9	7	1	0	0	0	1.88E-09
CGADBCABDCAB	0.89	0	0	0	0	0	0	0	0	0	0	1	2	3	4	9	16	11	13	16	5	8	0	0	1	0	1.93E-09
CGADBCABDCAC	0.78	0	0	0	0	0	0	0	0	0	0	1	2	2	5	8	15	8	11	14	5	7	0	0	0	0	7.63E-10
CGADBCABDCAD	0.63	0	0	0	0	0	0	0	0	0	0	1	2	2	5	8	12	6	8	7	6	5	0	0	1	0	5.89E-10
CGADDCBBDBAA	0.14	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	2	3	2	4	0	0	0	0	2.01E-10
CGADDCBBDBAB	0.21	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	4	3	3	5	4	0	0	0	0	1.64E-09
CGADDCBBDBAC	0.18	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	4	4	2	4	3	0	0	0	0	1.22E-09
CGADDCBBDBAD	0.04	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	1	0	0	0	0	0	0	9.20E-12
CGADDCBBDCAA	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	0	1	0	0	0	0	0	0	6.91E-12
CGADDCBBDCAB	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	0	0	1	0	0	0	0	0	0	5.15E-12
CGADDCBBDCAC	0.01	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	1.30E-13
CGCCACDBBBAA	0.25	0	0	0	0	0	0	0	0	0	0	0	0	0	2	0	2	4	8	4	3	2	0	0	0	0	4.69E-11
CGCCACDBBCAA	0.20	0	0	0	0	0	0	0	0	0	1	2	1	0	3	1	3	3	2	1	2	1	0	0	0	0	1.26E-11
CGCCBCDBDBAA	0.10	0	0	0	0	0	0	0	0	0	0	1	1	0	0	1	1	1	1	1	2	1	0	0	0	0	6.27E-11
CGCCBCDBDCAA	0.11	0	0	0	0	0	0	0	0	1	1	0	0	2	0	1	1	1	2	2	0	0	0	0	0	0	1.89E-11
CGCCCCDBBBAA	0.06	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	0	1	1	0	0	0	0	1	2.87E-12
CGCCCCDBBCAA	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	1	1	0	0	0	0	0	0	0	1.34E-12
CGCDBCDBDBAA	0.83	0	0	1	0	0	0	0	0	1	0	2	4	2	5	8	9	11	11	6	13	8	0	1	0	1	2.01E-09
CGCDBCDBDBAB	0.83	0	0	1	0	0	0	0	0	1	0	2	3	2	6	8	15	7	11	5	13	7	1	0	0	1	1.40E-09
CGCDBCDBDCAA	0.77	0	0	0	0	0	1	0	1	3	3	3	2	2	12	11	11	7	7	7	3	2	1	0	1	0	6.91E-10
CGCDBCDBDCAB	0.72	0	0	0	0	0	1	0	1	3	4	3	2	2	11	11	11	5	6	8	1	3	0	0	0	0	5.51E-10
CGCDDCDBDBAA	0.08	0	0	0	0	0	0	0	0	0	0	0	1	0	1	2	1	1	0	0	2	0	0	0	0	0	7.30E-12
CGCDDCDBDBAB	0.07	0	0	0	0	0	0	0	0	0	0	0	1	0	1	3	1	0	0	0	1	0	0	0	0	0	9.72E-12
CGCDDCDBDCAA	0.02	0	0	0	0	0	0	0	0	0	0	0	0	1	1	0	0	0	0	0	0	0	0	0	0	0	3.51E-13
CGCDDCDBDCAB	0.03	0	0	0	0	0	0	0	0	0	0	0	0	1	2	0	0	0	0	0	0	0	0	0	0	0	5.89E-13
CGDCACBBBBAA	0.13	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	3	1	2	3	1	0	0	0	0	1.10E-11
CGDCACBBBCAA	0.09	0	0	0	0	0	0	0	0	0	0	1	0	0	2	0	1	2	0	1	1	1	0	0	0	0	2.59E-12
CGDCBCABDBAA	0.10	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	2	0	2	0	3	1	0	0	0	0	4.62E-11
CGDCBCABDCAA	0.10	0	0	0	0	0	0	0	0	0	0	0	1	3	0	0	2	1	2	1	0	0	0	0	0	0	1.33E-11
CGDCCCBBBBAA	0.01	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2.12E-13
CGDDBCABDBAA	0.78	0	0	0	0	0	0	0	0	0	0	1	0	1	4	6	9	16	11	7	13	8	0	1	0	1	1.48E-09
CGDDBCABDBAB	0.76	0	0	0	0	0	0	0	0	0	0	1	0	1	4	7	13	14	7	8	12	7	1	0	0	1	1.03E-09
CGDDBCABDCAA	0.73	0	0	0	0	0	0	0	0	2	0	1	3	6	9	10	14	7	8	7	3	2	1	0	0	0	5.09E-10
CGDDBCABDCAB	0.65	0	0	0	0	0	0	0	1	1	0	3	4	3	11	10	12	4	6	6	1	3	0	0	0	0	4.05E-10
CGDDDCBBDBAA	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	0	0	1	0	0	0	0	0	0	0	7.98E-13
CGDDDCBBDBAB	0.05	0	0	0	0	0	0	0	0	0	0	0	0	0	0	3	1	0	0	0	1	0	0	0	0	0	1.43E-12
CGDDDCBBDCAA	0.01	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0	0	5.07E-14
CGDDDCBBDCAB	0.01	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0	0	6.61E-14
CHACACBBBBAA	0.58	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	1	6	11	15	15	5	2	0	0	0	6.39E-09
CHACACBBBCAA	0.31	0	0	0	0	0	0	0	0	0	0	2	1	2	1	2	3	10	3	4	2	1	0	0	0	0	7.41E-11

	Fraction of STs with Non-Zero Frequency Among 100 Observations	Source Term Group Representation Among 100 Observations																									
		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	Frequency
CHACBCABDBAA	0.25	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	6	3	6	7	1	0	0	0	1.83E-09
CHACBCABDCAA	0.13	0	0	0	0	0	0	0	0	0	0	0	0	0	0	3	0	2	2	5	1	0	0	0	0	0	1.33E-10
CHACCCBBBBAA	0.12	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	3	0	5	1	1	0	0	1	4.53E-10
CHACCCBBBCAA	0.07	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	1	1	3	0	0	0	0	0	0	1.01E-11
CHADBCABDBAA	1.00	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	8	12	21	21	27	2	4	1	1	1.60E-07
CHADBCABDBAB	1.00	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	4	6	15	20	21	25	4	2	1	1	4.17E-07
CHADBCABDBAC	1.00	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	4	7	14	23	19	24	5	2	0	1	2.99E-07
CHADBCABDBAD	1.00	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	4	7	15	20	21	25	4	2	0	1	5.03E-08
CHADBCABDCAA	0.93	0	0	0	0	0	0	0	0	0	0	1	2	3	5	9	15	11	12	15	11	7	1	0	1	0	4.87E-09
CHADBCABDCAB	0.94	0	0	0	0	0	0	0	0	0	0	1	2	3	6	9	16	11	13	18	6	8	0	0	1	0	5.74E-09
CHADBCABDCAC	0.90	0	0	0	0	0	0	0	0	0	0	1	2	3	7	8	16	10	11	17	7	7	0	0	1	0	2.18E-09
CHADBCABDCAD	0.82	0	0	0	0	0	0	0	0	0	0	1	2	4	6	9	15	8	10	14	7	5	0	0	1	0	2[illegible]
CHADDCBBDBAA	0.21	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	2	4	3	5	4	0	0	0	1	3.[illegible]
CHADDCBBDBAB	0.26	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	4	5	3	6	4	0	0	0	1	3.2[illegible]E-09
CHADDCBBDBAC	0.24	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	1	4	6	2	6	3	0	0	0	0	2.56E-09
CHADDCBBDBAD	0.12	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	2	3	2	3	0	0	0	0	1	1.15E-10
CHADDCBBDCAA	0.05	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	0	1	1	0	0	0	0	0	0	1.29E-11
CHADDCBBDCAB	0.05	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	0	1	1	0	0	0	0	0	0	3.43E-11
CHADDCBBDCAC	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	1	1	0	0	0	0	0	0	1.29E-11
CHADDCBBDCAD	0.01	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0	1.03E-11
CHCCFCDBDBAA	0.95	3	1	1	1	0	1	0	3	6	3	6	4	10	10	10	8	8	13	4	1	1	0	1	0	0	5.75E-09
CHCCFCDBDCAA	0.64	4	1	0	1	0	0	3	2	3	2	5	9	8	6	4	6	6	2	0	1	0	1	0	0	0	2.97E-10
CHCDFCDBDBAA	1.00	1	0	0	0	0	0	1	0	1	5	3	7	9	14	13	10	9	11	10	4	[illegible]	0	0	1	0	1.07E-07
CHCDFCDBDBAB	1.00	1	0	0	0	0	0	1	0	1	5	3	7	11	14	12	10	8	15	9	1	1	0	0	1	0	2.28E-07
CHCDFCDBDBAC	1.00	1	0	0	0	0	0	1	0	1	5	3	8	11	14	11	12	7	14	10	0	1	0	1	0	0	1.74E-07
CHCDFCDBDBAD	0.98	1	0	0	0	0	0	1	0	1	5	4	7	11	13	12	9	8	14	10	1	0	0	1	0	0	2.14E-08
CHCDFCDBDCAA	0.93	1	0	0	1	0	1	2	4	2	4	8	13	10	14	6	11	3	9	2	1	0	0	1	0	0	6.11E-09
CHCDFCDBDCAB	0.95	1	0	0	1	0	1	2	4	4	3	10	10	11	15	7	10	5	7	2	1	0	0	1	0	0	6.05E-09
CHCDFCDBDCAC	0.88	1	0	0	1	0	1	3	3	2	4	9	11	10	12	7	8	7	5	2	1	0	1	0	0	0	1.55E-09
CHCDFCDBDCAD	0.84	1	0	0	1	0	1	3	3	3	4	9	11	10	10	8	8	4	4	3	0	0	1	0	0	0	1.64E-09
CHDCACBBBBAA	0.23	0	0	0	0	0	0	0	0	0	0	0	0	1	1	2	2	4	3	5	3	2	0	0	0	0	1.75E-11
CHDCACBBBCAA	0.46	0	0	0	0	0	0	0	0	0	1	3	2	0	5	3	9	10	6	2	4	1	0	0	0	0	2.01E-10
CHDCBCABDBAA	0.11	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	3	0	2	0	3	1	0	0	0	0	6.39E-11
CHDCBCABDCAA	0.24	0	0	0	0	0	0	0	0	0	0	0	2	5	2	3	4	1	4	2	1	0	0	0	0	0	1.88E-10
CHDCCCBBBBAA	0.05	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	1	1	0	0	0	0	1	2.09E-12
CHDCCCBBBCAA	0.11	0	0	0	0	0	0	0	0	0	0	0	1	0	0	2	1	2	2	2	0	0	0	0	1	0	3.74E-11
CHDDBCABDBAA	0.94	0	0	0	0	0	0	0	0	0	0	1	0	1	5	7	11	18	14	12	15	8	0	1	0	1	2.60E-09
CHDDBCABDBAB	0.86	0	0	0	0	0	0	0	0	0	0	1	0	1	5	8	13	15	12	9	14	6	1	0	0	1	1.31E-09
CHDDBCABDCAA	1.00	0	0	0	0	0	0	0	0	2	0	3	3	9	14	14	17	11	11	8	4	2	1	0	1	0	1.46E-08
CHDDBCABDCAB	0.96	0	0	0	0	0	0	0	1	1	0	4	4	6	14	15	16	10	10	8	3	3	0	0	1	0	2.16E-09
CHDDDCBBDBAA	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	1	1	0	0	0	0	0	0	0	5.39E-12
CHDDDCBBDBAB	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	0	1	0	0	0	0	0	0	0	0	3.59E-12
CHDDDCBBDCAA	0.05	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	0	1	0	1	0	0	0	0	0	0	2.37E-11
CHDDDCBBDCAB	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	3	0	0	0	0	0	0	0	0	0	0	0	4.16E-12
CHECACBBBBAA	0.34	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	1	3	7	7	9	4	1	0	0	0	7.36E-11
CHECACBBBCAA	0.56	0	0	0	0	0	0	0	0	0	0	2	1	2	1	7	6	11	14	6	5	1	0	0	0	0	8.21E-10
CHECBCABDBAA	0.13	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	2	3	5	0	0	0	0	8.77E-11
CHECBCABDCAA	0.24	0	0	0	0	0	0	0	0	0	0	0	0	1	2	4	3	3	2	6	2	1	0	0	0	0	2.55E-10

	Fraction of STs with Non-Zero Frequency Among 100 Observations	Source Term Group Representation Among 100 Observations																									
		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	Frequency
CHECCCBBBBAA	0.08	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	2	0	2	1	1	0	0	1	9.34E-12
CHECCCBBBCAA	0.13	0	0	0	0	0	0	0	0	0	0	0	1	0	1	0	2	2	1	4	1	0	0	0	1	0	1.55E-10
CHEDBCABDBAA	0.96	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	8	10	20	21	26	2	4	1	1	3.51E-09
CHEDBCABDBAB	0.89	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	4	6	14	17	19	21	3	2	1	1	1.77E-09
CHEDBCABDCAA	1.00	0	0	0	0	0	0	0	0	0	0	1	2	3	5	10	17	12	13	15	12	7	2	0	1	0	1.98E-08
CHEDBCABDCAB	0.98	0	0	0	0	0	0	0	0	0	0	1	2	3	5	9	18	12	13	18	7	8	1	0	1	0	2.93E-09
CHEDDCBBDBAA	0.11	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	2	1	2	2	0	0	0	1	3.46E-11
CHEDDCBBDBAB	0.09	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	1	3	1	2	0	0	0	0	0	2.55E-11
CHEDDCBBDCAA	0.16	0	0	0	0	0	0	0	0	0	0	0	0	1	2	2	2	2	3	4	0	0	0	0	0	0	1.45E-10
CHEDDCBBDCAB	0.05	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	2	0	1	0	0	0	0	0	0	0	2.68E-11
DDCCACDBBAAA	0.41	0	0	0	0	0	0	0	0	0	1	2	2	4	2	7	13	1	8	1	0	0	0	0	0	0	4.31E-09
DDCCCCDBBAAA	0.08	0	0	0	0	0	0	0	0	0	0	0	1	0	1	2	0	3	0	1	0	0	0	0	0	0	6.40E-10
DDCDDCDBDAAA	0.37	1	3	1	1	3	2	2	7	1	3	4	1	1	2	4	1	0	0	0	0	0	0	0	0	0	1.04E-09
DDCDDCDBDAAB	0.24	2	1	1	0	1	2	2	2	1	2	4	0	1	2	2	1	0	0	0	0	0	0	0	0	0	1.15E-10
DDCDDCDBDAAC	0.02	0	0	0	0	0	0	1	0	0	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	6.57E-13
DDCDDCDBDAAD	0.02	0	0	0	0	0	0	0	0	0	0	1	0	0	0	1	0	0	0	0	0	0	0	0	0	0	3.22E-13
DDDCACBBBAAA	0.41	0	0	0	0	0	0	0	0	0	1	2	1	5	2	6	13	2	7	2	0	0	0	0	0	0	1.07E-09
DDDCCCBBBAAA	0.08	0	0	0	0	0	0	0	0	0	0	0	1	0	1	2	0	3	0	1	0	0	0	0	0	0	1.60E-10
DDDDDCBBDAAA	0.24	0	0	1	0	2	0	1	2	2	3	3	1	1	4	2	2	0	0	0	0	0	0	0	0	0	1.78E-10
DDDDDCBBDAAB	0.13	0	0	0	1	1	0	2	0	0	3	1	1	1	2	1	0	0	0	0	0	0	0	0	0	0	1.85E-11
DFACACBBBAAA	0.37	0	0	0	0	0	0	0	0	0	0	0	0	0	2	3	3	3	10	7	7	0	2	0	0	0	2.74E-10
DFACCCBBBAAA	0.07	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	2	0	2	1	0	0	0	1	3.59E-11
DFADDCBBDAAA	0.19	0	0	0	0	0	0	0	0	0	0	0	1	0	0	3	3	3	1	1	3	4	0	0	0	0	5.40E-11
DFADDCBBDAAB	0.16	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	4	2	2	1	3	2	0	0	0	0	3.52E-11
DFADDCBBDAAC	0.09	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	3	2	1	0	0	2	0	0	0	0	6.70E-12
DFADDCBBDAAD	0.05	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	1	1	0	1	0	0	0	0	0	1.58E-12
DFCCACDBBAAA	0.31	0	0	0	0	0	0	0	0	0	0	0	0	0	2	2	1	4	9	6	6	0	1	0	0	0	9.16E-11
DFCCCCDBBAAA	0.06	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	1	2	0	0	0	0	1	3.07E-11
DFCDDCDBDAAA	0.12	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	3	1	1	2	3	0	0	0	0	1.43E-11
DFCDDCDBDAAB	0.13	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	4	1	1	1	2	2	0	0	0	0	2.80E-12
DFDCACBBBAAA	0.23	0	0	0	0	0	0	0	0	0	0	0	0	0	2	1	1	2	7	4	6	0	0	0	0	0	2.20E-11
DFDCCCBBBAAA	0.05	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	0	2	0	0	0	0	1	7.08E-12
DFDDDCBBDAAA	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	2	0	0	0	0	2.03E-12
DFDDDCBBDAAB	0.01	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0	1.11E-13
DGACACBBBAAA	0.36	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	1	7	11	13	1	2	0	0	0	1.97E-10
DGACCCBBBAAA	0.07	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	1	1	1	2	0	0	0	1	2.55E-11
DGADDCBBDAAA	0.17	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	1	2	5	6	1	0	0	0	4.12E-11
DGADDCBBDAAB	0.15	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	2	2	5	4	0	0	0	0	3.52E-11
DGADDCBBDAAC	0.06	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	1	1	2	0	0	0	0	5.57E-12
DGADDCBBDAAD	0.05	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	2	0	0	0	0	0	3.60E-12
DGCCACDBBAAA	0.30	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	2	4	8	6	7	0	1	0	0	0	1.09E-10
DGCCCCDBBAAA	0.05	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	0	2	0	0	0	0	1	2.45E-11
DGCDDCDBDAAA	0.13	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	1	2	0	2	4	2	0	0	0	0	9.91E-12
DGCDDCDBDAAB	0.11	0	0	0	0	0	0	0	0	0	0	0	0	0	0	4	3	1	0	0	3	0	0	0	0	0	5.49E-12
DGDCACBBBAAA	0.22	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	3	2	5	4	6	0	1	0	0	0	2.68E-11
DGDCCCBBBAAA	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	2	0	0	0	0	0	5.94E-12
DGDDDCBBDAAA	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	0	0	0	0	1	0	0	0	0	8.28E-13
DGDDDCBBDAAB	0.03	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	0	1	0	0	0	0	0	0	0	5.58E-13

	Fraction of STs with Non-Zero Frequency Among 100 Observations	Source Term Group Representation Among 100 Observations																									
		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	Frequency
DHACACBBBAAA	0.39	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	1	7	12	14	2	1	0	0	0	8.49E-10
DHACCCBBBAAA	0.10	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	1	1	4	2	0	0	0	1	5.88E-11
DHADDCBBDAAA	0.14	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	2	1	3	5	1	0	0	0	7.54E-11
DHADDCBBDAAB	0.17	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	4	2	3	5	0	0	0	0	1.84E-10
DHADDCBBDAAC	0.10	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	1	3	3	0	0	0	0	7.38E-11
DHADDCBBDAAD	0.09	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	3	4	1	0	0	0	0	5.79E-11
DHDCACBBBAAA	0.41	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	4	4	10	9	9	0	2	0	0	0	1.39E-09
DHDCCCBBBAAA	0.08	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	3	1	2	0	0	0	0	1	1.12E-10
DHDDDCBBDAAA	0.22	0	0	0	0	0	0	0	0	0	0	0	0	1	0	2	3	4	3	2	3	3	1	0	0	0	2.06E-10
DHDDDCBBDAAB	0.10	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	3	2	2	0	2	1	0	0	0	0	2.70E-11
DHECACBBBAAA	0.41	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	1	8	12	13	3	2	0	0	0	5.57E-09
DHECCCBBBAAA	0.08	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	1	1	2	1	1	0	0	1	4.50E-10
DHEDDCBBDAAA	0.36	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	2	3	9	9	8	2	0	0	1	1.21E-09
DHEDDCBBDAAB	0.20	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	2	4	2	6	4	0	0	0	1	1.66E-10
EFACACBBBAAA	0.12	1	1	2	2	1	0	1	3	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.49E-10
EFACBCABDAAA	0.27	3	2	3	0	3	3	5	4	2	1	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	3.82E-10
EFACCCBBBAAA	0.02	0	0	1	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	7.55E-12
EFADBCABDAAA	0.97	4	3	7	7	16	11	11	20	7	7	1	3	0	0	0	0	0	0	0	0	0	0	0	0	0	2.01E-08
EFADBCABDAAB	0.97	7	4	5	16	9	16	9	17	5	5	1	3	0	0	0	0	0	0	0	0	0	0	0	0	0	9.50E-09
EFADBCABDAAC	0.96	10	5	14	9	12	13	12	9	5	3	2	2	0	0	0	0	0	0	0	0	0	0	0	0	0	3.62E-09
EFADBCABDAAD	0.93	34	9	8	6	9	6	6	5	5	2	2	1	0	0	0	0	0	0	0	0	0	0	0	0	0	7.92E-10
EFCCACDBBAAA	0.14	6	0	3	2	1	0	2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.28E-11
EFCCBCDBDAAA	0.21	9	3	1	2	1	3	2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	6.11E-11
EFCCCCDBBAAA	0.03	1	0	0	2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.05E-11
EFCDBCDBDAAA	0.86	21	5	12	8	18	7	8	6	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	3.33E-09
EFCDBCDBDAAB	0.81	23	9	8	17	5	11	7	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.76E-09
EFDCACBBBAAA	0.09	1	1	2	0	2	0	2	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5.02E-12
EFDCBCABDAAA	0.20	3	1	3	0	3	3	3	2	1	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	4.50E-11
EFDCCCBBBAAA	0.03	0	0	1	2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.57E-12
EFDDBCABDAAA	0.86	4	2	9	8	16	10	14	17	2	3	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	2.46E-09
EFDDBCABDAAB	0.81	6	5	6	14	12	12	16	7	2	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.30E-09
EGACACBBBAAA	0.12	0	0	0	0	1	1	2	0	0	0	1	1	0	0	1	4	1	0	0	0	0	0	0	0	0	1.60E-10
EGACBCABDAAA	0.26	0	0	0	0	0	0	0	4	4	1	1	4	2	3	3	2	1	1	0	0	0	0	0	0	0	2.86E-10
EGACCCBBBAAA	0.02	0	0	0	0	0	0	0	0	0	0	1	1	0	0	0	0	0	0	0	0	0	0	0	0	0	6.21E-12
EGADBCABDAAA	0.95	0	0	0	0	0	0	2	6	8	7	5	10	13	15	7	7	7	5	2	1	0	0	0	0	0	1.44E-08
EGADBCABDAAB	0.96	0	0	0	0	0	0	3	7	10	4	6	9	15	12	8	7	7	6	2	0	0	0	0	0	0	6.62E-09
EGADBCABDAAC	0.94	0	0	0	0	0	0	7	6	8	3	7	8	16	10	10	6	6	6	1	0	0	0	0	0	0	2.20E-09
EGADBCABDAAD	0.92	0	0	0	0	1	4	4	6	5	3	9	8	15	10	9	7	4	6	1	0	0	0	0	0	0	8.28E-10
EGDCACBBBAAA	0.07	0	0	0	2	0	0	0	0	1	1	1	0	0	0	0	1	1	0	0	0	0	0	0	0	0	3.37E-12
EGDCBCABDAAA	0.19	0	1	0	0	0	1	5	3	1	3	1	2	1	0	0	1	0	0	0	0	0	0	0	0	0	1.24E-10
EGDCCCBBBAAA	0.01	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	3.69E-14
EGDDBCABDAAA	0.96	0	1	0	0	2	6	9	8	9	12	11	9	10	8	4	2	3	1	0	1	0	0	0	0	0	4.89E-09
EGDDBCABDAAB	0.94	0	1	0	0	5	5	9	10	9	10	13	6	9	6	4	2	3	1	1	0	0	0	0	0	0	3.18E-09
EHACACBBBAAA	0.06	0	0	0	0	1	1	0	0	0	0	0	0	0	0	0	3	1	0	0	0	0	0	0	0	0	8.23E-12
EHACBCABDAAA	0.14	0	0	0	0	0	0	0	2	2	1	0	2	2	2	2	0	1	0	0	0	0	0	0	0	0	2.90E-11
EHACCCBBBAAA	0.01	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.59E-12
EHADBCABDAAA	0.91	0	0	0	0	0	0	2	6	8	6	5	10	13	14	6	8	5	5	2	1	0	0	0	0	0	1.59E-09
EHADBCABDAAB	0.96	0	0	0	0	0	0	3	7	10	4	6	9	15	12	8	8	6	6	2	0	0	0	0	0	0	1.58E-09

	Fraction of STs with Non-Zero Frequency Among 100 Observations	Source Term Group Representation Among 100 Observations 1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	Frequency
EHADBCABDAAC	0.93	0	0	0	0	0	0	7	[illegible]	8	3	8	8	16	10	9	6	5	6	1	0	0	0	0	0	0	5.62E-10
EHADBCABDAAD	0.90	0	0	0	0	1	4	4	[illegible]	5	3	9	8	14	10	8	8	4	5	1	0	0	0	0	0	0	3.98E-10
EHDCACBBBAAA	0.02	0	0	0	1	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	2.43E-13
EHDCBCABDAAA	0.08	0	0	0	0	0	0	3	0	1	1	1	2	0	0	0	0	0	0	0	0	0	0	0	0	0	2.21E-11
EHDCCCBBBAAA	0.01	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	9.87E-14
EHDDBCABDAAA	0.70	0	1	0	0	2	5	6	6	7	9	10	5	4	4	4	2	3	1	0	1	0	0	0	0	0	8.68E-10
EHDDBCABDAAB	0.72	0	1	0	0	5	4	8	8	6	8	9	4	5	4	4	2	3	1	0	0	0	0	0	0	0	9.32E-10
EHECACBBBAAA	0.03	0	0	0	0	0	1	0	0	0	0	1	0	0	0	0	1	0	0	0	0	0	0	0	0	0	1.06E-12
EHECBCABDAAA	0.10	0	0	0	0	0	0	0	1	2	0	0	1	1	1	2	2	0	0	0	0	0	0	0	0	0	3.05E-11
EHECCCBBBAAA	0.01	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0	0	3.95E-13
EHEDBCABDAAA	0.72	0	0	0	0	0	0	2	5	5	4	5	7	9	10	5	7	6	5	1	1	0	0	0	0	0	1.18E-09
EHEDBCABDAAB	0.76	0	0	0	0	0	0	2	7	8	4	6	5	12	7	7	8	5	5	0	0	0	0	0	0	0	1.26E-09
GDCCACDBBDAA	0.19	19	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.26E-09
GDCCBCDBDDAA	0.27	27	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.84E-09
GDCCCCDBBDAA	0.05	5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	9.11E-10
GDCCFCDBDDAA	0.93	93	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	3.16E-09
GDCDBCDBDDAA	1.00	100	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.48E-07
GDCDBCDBDDAB	1.00	100	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	4.19E-08
GDCDBCDBDDAC	0.79	79	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5.34E-10
GDCDBCDBDDAD	0.77	77	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	4.05E-10
GDCDFCDBDDAA	0.97	97	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	6.25E-08
GDCDFCDBDDAB	0.97	97	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5.82E-08
GDCDFCDBDDAC	0.97	97	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.77E-08
GDCDFCDBDDAD	0.97	97	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.55E-08
GDDCACBBBDAA	0.19	19	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5.63E-10
GDDCBCABDDAA	0.27	27	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.36E-09
GDDCCCBBBDAA	0.05	5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.27E-10
GDDDBCABDDAA	1.00	100	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.10E-07
GDDDBCABDDAB	1.00	100	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	3.10E-08
GDDDBCABDDAC	0.77	77	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	3.79E-10
GDDDBCABDDAD	0.70	70	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.91E-10
GFADBCABDDAA	0.75	75	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.01E-10
GFADBCABDDAB	0.92	92	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5.61E-09
GFADBCABDDAC	0.84	84	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.68E-09
GFADBCABDDAD	0.78	78	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.52E-09
GFCDBCDBDDAA	0.47	47	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.94E-11
GFCDBCDBDDAB	0.24	24	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.00E-11
GFDDBCABDDAA	0.38	38	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.91E-11
GFDDBCABDDAB	0.18	18	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	6.83E-12
GGADBCABDDAA	0.70	70	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.43E-10
GGADBCABDDAB	0.94	94	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	8.61E-09
GGADBCABDDAC	0.90	90	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	4.48E-09
GGADBCABDDAD	0.84	84	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5.22E-09
GGCCACDBBDAA	0.10	10	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.48E-11
GGCCBCDBDDAA	0.23	23	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.69E-10
GGCCCCDBBDAA	0.04	4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.55E-12
GGCDBCDBDDAA	0.97	97	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	6.68E-09
GGCDBCDBDDAB	0.95	95	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	4.34E-09

	Fraction of STs with Non-Zero Frequency Among 100 Observations	Source Term Group Representation Among 100 Observations																									
		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	Frequency
GGDDBCABDDAA	0.35	35	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	4.37E-11
GGDDBCABDDAB	0.28	28	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.52E-11
GHACACBBBDAA	0.11	11	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.06E-10
GHACBCABDDAA	0.27	27	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	7.74E-10
GHACCCBBBDAA	0.02	2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	6.63E-11
GHADBCABDDAA	0.97	97	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	3.70E-08
GHADBCABDDAB	0.97	97	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	4.67E-08
GHADBCABDDAC	0.97	97	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.94E-08
GHADBCABDDAD	0.97	97	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.20E-08
GHCCFCDBDDAA	0.93	93	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.92E-09
GHCDFCDBDDAA	0.97	97	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5.23E-08
GHCDFCDBDDAB	0.97	97	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5.06E-08
GHCDFCDBDDAC	0.97	97	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.42E-08
GHCDFCDBDDAD	0.96	96	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.39E-08
GHDCACBBBDAA	0.19	19	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	6.39E-10
GHDCBCABDDAA	0.27	27	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.61E-09
GHDCCCBBBDAA	0.05	5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.04E-10
GHDDBCABDDAA	1.00	100	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.45E-07
GHDDBCABDDAB	1.00	100	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.62E-08
GHECACBBBDAA	0.19	19	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.56E-09
GHECBCABDDAA	0.27	27	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.18E-09
GHECCCBBBDAA	0.05	5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	8.15E-10
GHEDBCABDDAA	1.00	100	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1.96E-07
GHEDBCABDDAB	1.00	100	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2.19E-08

Total Frequency (1/yr): 4.22E-06

Table C.~~8~~11. Partitioning of Source Terms with Zero Early Fatalities and Non-Zero Latent Fatalities

4047	121	163	177	225	265	310	322	315	276	393	439	547	735	826	1036	946	1187	1109	1001	800	# of STs
-.30	0.10	0.27	0.45	0.63	0.81	0.98	1.16	1.34	1.52	1.69	1.87	2.05	2.22	2.40	2.58	2.76	2.93	3.11	3.29	3.47	LH Weight
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	ST

Table C.~~9~~12 Release Fractions for 25 Mean Source Term Partition Groups

Group	Xe	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	1.57E-02	1.36E-04	6.58E-07	3.20E-07	6.28E-08	6.77E-09	5.80E-09	1.26E-08	5.74E-08
2	9.91E-01	4.29E-03	8.92E-05	2.48E-05	1.86E-06	2.82E-07	1.54E-07	4.89E-07	1.99E-06
3	9.83E-01	7.77E-03	1.51E-04	4.45E-05	2.11E-06	3.55E-07	1.67E-07	4.21E-07	2.28E-06
4	9.49E-01	1.07E-02	2.98E-04	1.15E-04	9.12E-06	2.07E-06	7.48E-07	2.23E-06	9.91E-06
5	9.29E-01	1.46E-02	4.50E-04	1.98E-04	1.62E-05	3.76E-06	8.78E-07	2.57E-06	1.80E-05
6	9.62E-01	2.03E-02	8.04E-04	4.01E-04	7.87E-05	1.03E-05	8.06E-06	1.42E-05	7.33E-05
7	8.45E-01	2.03E-02	1.86E-03	5.11E-04	3.87E-05	9.81E-06	2.49E-06	4.71E-06	4.26E-05
8	9.82E-01	4.76E-02	1.47E-03	9.09E-04	2.01E-04	1.66E-05	2.01E-05	2.76E-05	1.68E-04
9	9.61E-01	3.42E-02	4.04E-03	2.54E-03	3.91E-04	3.69E-05	4.67E-05	6.00E-05	3.30E-04
10	4.28E-01	2.35E-02	8.49E-03	9.90E-04	8.05E-05	1.14E-05	6.12E-06	1.04E-05	9.77E-05
11	6.48E-01	2.90E-02	1.21E-02	2.17E-03	1.21E-04	4.14E-05	1.29E-05	1.67E-05	1.37E-04
12	7.61E-01	4.41E-02	2.22E-02	4.47E-03	2.22E-04	3.54E-05	1.30E-05	3.36E-05	2.70E-04
13	7.67E-01	5.20E-02	2.90E-02	6.79E-03	4.04E-04	1.23E-04	2.67E-05	4.90E-05	4.67E-04
14	9.22E-01	7.50E-02	4.44E-02	1.97E-02	1.61E-03	7.80E-04	8.72E-05	2.56E-04	2.20E-03
15	8.38E-01	1.08E-01	6.51E-02	1.91E-02	1.00E-03	1.82E-04	4.69E-05	6.20E-05	1.01E-03
16	8.57E-01	1.47E-01	1.00E-01	3.49E-02	2.13E-03	4.96E-04	1.36E-04	2.59E-04	2.33E-03

Group	Xe	I	Cs	Te	Sr	Ru	La	Ce	Ba
17	9.91E-01	1.95E-01	1.49E-01	9.38E-02	9.79E-03	1.96E-03	5.58E-04	1.00E-03	9.01E-03
18	9.78E-01	2.11E-01	1.82E-01	1.06E-01	3.34E-02	2.41E-03	3.67E-03	4.70E-03	2.68E-02
19	9.96E-01	3.74E-01	2.93E-01	1.63E-01	3.51E-02	4.67E-03	2.41E-03	6.14E-03	3.24E-02
20	1.00E+00	5.27E-01	5.07E-01	2.05E-01	2.46E-02	3.67E-03	1.91E-03	4.24E-03	2.24E-02
21	1.00E+00	6.16E-01	5.82E-01	3.48E-01	1.28E-01	1.31E-02	1.24E-02	2.41E-02	1.14E-01
22	1.00E+00	5.97E-01	5.58E-01	4.95E-01	3.22E-01	3.96E-02	3.98E-02	5.71E-02	2.98E-01
23	1.00E+00	5.82E-01	5.92E-01	6.66E-01	5.61E-01	4.40E-02	1.01E-01	1.10E-01	5.24E-01
24	1.00E+00	6.29E-01	6.44E-01	6.65E-01	6.05E-01	7.52E-02	1.10E-01	1.66E-01	5.92E-01
25	1.00E+00	8.87E-01	8.86E-01	8.81E-01	8.86E-01	1.70E-01	9.25E-02	6.03E-01	8.86E-01

APPENDIX D

SUPPORTING INFORMATION FOR THE

THE CONSEQUENCE ANALYSIS

Appendix D

This appendix contains the input data files used in the MACCS calculations. The original files taken from the NUREG-1150 Surry study were modified where needed to reflect the specifics of the LP/SD study.

D.1 ATMOS Input File

```
* GENERAL DESCRIPTIVE TITLE DESCRIBING THIS "ATMOS" INPUT
* ... used for Surry LP/SD study
RIATNAM1001  'SURRY ATMOS INPUT FOR FINAL NUREG-1150 CALCULATIONS'
* FLAG TO INDICATE THAT THIS IS THE LAST PROGRAM IN THE SERIES TO BE RUN
*
OCENDAT1001  .FALSE. (SET THIS VALUE TO .TRUE. TO SKIP EARLY AND CHRONC)
*****************************************************************************************
* GEOMETRY DATA BLOCK, LOADED BY INPGEO, STORED IN /GEOM/
*
* NUMBER OF RADIAL SPATIAL ELEMENTS
*
GENUMRAD001  26
*
* SPATIAL ENDPOINT DISTANCES IN MILES
*
*    END001      0.25      0.5       0.75      1.0       1.5
*    END002      2.0       2.5       3.0       3.5       5.0
*    END003      7.0      10        13        16        20
*    END004     25        30        40        50        70
*    END005    100       150       200       350       500
*    END006   1000
*
*  SURRY
*
GESPAENDO01      .16       .52      1.21      1.61      2.13
GESPAENDO02     3.22      4.02      4.83      5.63      8.05
GESPAENDO03    11.27     16.09     20.92     25.75     32.19
GESPAENDO04    40.23     48.28     64.37     80.47    112.65
GESPAENDO05   160.93    241.14    321.87    563.27    804.67
GESPAENDO06  1609.34
*****************************************************************************************
* NUCLIDE DATA BLOCK, LOADED BY INPISO, STORED IN /ISOGRP/, /ISONAM/
*
* NUMBER OF NUCLIDES
*
ISNUMISO001 60
*
* NUMBER OF NUCLIDE GROUPS
*
ISMAXGRP001  9
*
* WET AND DRY DEPOSITION FLAGS FOR EACH NUCLIDE GROUP
*
*              WETDEP     DRYDEP
*
ISDEPFLA001    .FALSE.   .FALSE.
ISDEPFLA002    .TRUE.    .TRUE.
```

```
ISDEPFLA003     .TRUE.     .TRUE.
ISDEPFLA004     .TRUE.     .TRUE.
ISDEPFLA005     .TRUE.     .TRUE.
ISDEPFLA006     .TRUE.     .TRUE.
ISDEPFLA007     .TRUE.     .TRUE.
ISDEPFLA008     .TRUE.     .TRUE.
ISDEPFLA009     .TRUE.     .TRUE.
*
* NUCLIDE GROUP DATA FOR 9 NUCLIDE GROUPS
*
*                NUCNAM      PARENT   IGROUP          HAFLIF
*
ISOTPGRP001      CO-58        NONE      6          6.160E+06
ISOTPGRP002      CO-60        NONE      6          1.660E+08
ISOTPGRP003      KR-85        NONE      1          3.386E+08
ISOTPGRP004      KR-85M       NONE      1          1.613E+04
ISOTPGRP005      KR-87        NONE      1          4.560E+03
ISOTPGRP006      KR-88        NONE      1          1.008E+04
ISOTPGRP007      RB-86        NONE      3          1.611E+06
ISOTPGRP008      SR-89        NONE      5          4.493E+06
ISOTPGRP009      SR-90        NONE      5          8.865E+08
ISOTPGRP010      SR-91        NONE      5          3.413E+04
ISOTPGRP011      SR-92        NONE      5          9.756E+03       NEW
ISOTPGRP012      Y-90        SR-90      7          2.307E+05
ISOTPGRP013      Y-91        SR-91      7          5.080E+06
ISOTPGRP014      Y-92         SR-92     7          1.274E+04       NEW
ISOTPGRP015      Y-93         NONE      7          3.636E+04       NEW
ISOTPGRP016      ZR-95        NONE      7          5.659E+06
ISOTPGRP017      ZR-97        NONE      7          6.048E+04
ISOTPGRP018      NB-95       ZR-95      7          3.033E+06
ISOTPGRP019      MO-99        NONE      6          2.377E+05
ISOTPGRP020      TC-99M      MO-99      6          2.167E+04
ISOTPGRP021      RU-103       NONE      6          3.421E+06
ISOTPGRP022      RU-105       NONE      6          1.598E+04
ISOTPGRP023      RU-106       NONE      6          3.188E+07
ISOTPGRP024      RH-105      RU-105     6          1.278E+05
ISOTPGRP025      SB-127       NONE      4          3.283E+05
ISOTPGRP026      SB-129       NONE      4          1.562E+04
ISOTPGRP027      TE-127      SB-127     4          3.366E+04
ISOTPGRP028      TE-127M      NONE      4          9.418E+06
ISOTPGRP029      TE-129      SB-129     4          4.200E+03
ISOTPGRP030      TE-129M      NONE      4          2.886E+06
ISOTPGRP031      TE-131M      NONE      4          1.080E+05
ISOTPGRP032      TE-132       NONE      4          2.808E+05
ISOTPGRP033      I-131       TE-131M    2          6.947E+05
ISOTPGRP034      I-132       TE-132     2          8.226E+03
ISOTPGRP035      I-133        NONE      2          7.488E+04
ISOTPGRP036      I-134        NONE      2          3.156E+03
ISOTPGRP037      I-135        NONE      2          2.371E+04
ISOTPGRP038      XE-133      I-133      1          4.571E+05
ISOTPGRP039      XE-135      I-135      1          3.301E+04
ISOTPGRP040      CS-134       NONE      3          6.501E+07
ISOTPGRP041      CS-136       NONE      3          1.123E+06
ISOTPGRP042      CS-137       NONE      3          9.495E+08
ISOTPGRP043      BA-139       NONE      9          4.986E+03       NEW
ISOTPGRP044      BA-140       NONE      9          1.105E+06
```

```
ISOTPGRP045     LA-140     BA-140      7        1.448E+05
ISOTPGRP046     LA-141       NONE      7        1.418E+04        NEW
ISOTPGRP047     LA-142       NONE      7        5.724E+03        NEW
ISOTPGRP048     CE-141     LA-141      8        2.811E+06  PARENT ADDED
ISOTPGRP049     CE-143       NONE      8        1.188E+05
ISOTPGRP050     CE-144       NONE      8        2.457E+07
ISOTPGRP051     PR-143     CE-143      7        1.173E+06
ISOTPGRP052     ND-147       NONE      7        9.495E+05
ISOTPGRP053     NP-239       NONE      8        2.030E+05
ISOTPGRP054     PU-238     CM-242      8        2.809E+09
ISOTPGRP055     PU-239     NP-239      8        7.700E+11
ISOTPGRP056     PU-240     CM-244      8        2.133E+11
ISOTPGRP057     PU-241       NONE      8        4.608E+08
ISOTPGRP058     AM-241     PU-241      7        1.366E+10
ISOTPGRP059     CM-242       NONE      7        1.408E+07
ISOTPGRP060     CM-244       NONE      7        5.712E+08
*******************************************************************************
* WET DEPOSITION DATA BLOCK, LOADED BY INPWET, STORED IN /WETCON/
*
* WASHOUT COEFFICIENT NUMBER ONE, LINEAR FACTOR
*
WDCWASH1001  9.5E-5   (HELTON AFTER JONES, 1986)
*
* WASHOUT COEFFICIENT NUMBER TWO, EXPONENTIAL FACTOR
*
WDCWASH2001  0.8      (HELTON AFTER JONES, 1986)
*******************************************************************************
* DRY DEPOSITION DATA BLOCK, LOADED BY INPDRY, STORED IN /DRYCON/
*
* NUMBER OF PARTICLE SIZE GROUPS
*
DDNPSGRP001    1
*
* DEPOSITION VELOCITY OF EACH PARTICLE SIZE GROUP (M/S)
*
DDVDEPOS001   0.01   (VALUE SELECTED BY S. ACHARYA, NRC)
*******************************************************************************
* DISPERSION PARAMETER DATA BLOCK, LOADED BY INPDIS, STORED IN /DISPY/, /DISPZ/
*
* SIGMA = A X ** B  WHERE A AND B VALUES ARE FROM TADMOR AND GUR (1969)
*
* LINEAR TERM OF THE EXPRESSION FOR SIGMA-Y, 6 STABILITY CLASSES
*
* STABILITY CLASS: A        B        C        D        E        F
*
DPCYSIGA001   0.3658   0.2751   0.2089   0.1474   0.1046   0.0722
*
* EXPONENTIAL TERM OF THE EXPRESSION FOR SIGMA-Y, 6 STABILITY CLASSES
*
* STABILITY CLASS: A        B        C        D        E        F
*
DPCYSIGB001    .9031    .9031    .9031    .9031    .9031    .9031
*
* LINEAR TERM OF THE EXPRESSION FOR SIGMA-Z, 6 STABILITY CLASSES
*
* STABILITY CLASS: A        B        C        D        E        F
```

```
*
DPCZSIGA001   2.5E-4   1.9E-3       .2       .3       .4       .2
*
* EXPONENTIAL TERM OF THE EXPRESSION FOR SIGMA-Z, 6 STABILITY CLASSES
*
* STABILITY CLASS; A        B        C        D        E        F
*
DPCZSIGB001    2.125   1.6021     .8543    .6532    .6021    .6020
*
*  LINEAR SCALING FACTOR FOR SIGMA-Y FUNCTION, NORMALLY 1
*
DPYSCALE001    1.
*
*  LINEAR SCALING FACTOR FOR SIGMA-Z FUNCTION,
*  NORMALLY USED FOR SURFACE ROUGHNESS LENGTH CORRECTION.
*  (Z1 / Z0) ** 0.2,   FROM CRAC2 WE HAVE  (10 CM / 3 CM) ** 0.2 = 1.27
*
DPZSCALE001    1.27
********************************************************************************
*
* EXPANSION FACTOR DATA BLOCK, LOADED BY INPEXP, STORED IN /EXPAND/
*
* TIME BASE FOR EXPANSION FACTOR (SECONDS)
*
PMTIMBAS001    600.   (10 MINUTES)
*
* BREAK POINT FOR FORMULA CHANGE (SECONDS)
*
PMBRKPNT001   3600.   (1 hour)
*
* EXPONENTIAL EXPANSION FACTOR NUMBER 1
*
PMXPFAC1001      0.2
*
* EXPONENTIAL EXPANSION FACTOR NUMBER 2
*
PMXPFAC2001      0.25
********************************************************************************
* PLUME RISE DATA BLOCK, LOADED BY INPLRS, STORED IN /PLUMRS/
*
* SCALING FACTOR FOR THE CRITICAL WIND SPEED FOR ENTRAINMENT OF A BOUYANT PLUME
* (USED BY FUNCTION CAUGHT)
*
PRSCLCRW001   1.
*
* SCALING FACTOR FOR THE A-D STABILITY PLUME RISE FORMULA
* (USED BY FUNCTION PLMRIS)
*
PRSCLADP001   1.
*
* SCALING FACTOR FOR THE E-F STABILITY PLUME RISE FORMULA
* (USED BY FUNCTION PLMRIS)
*
PRSCLEFP001   1.
********************************************************************************
* WAKE EFFECTS DATA BLOCK, LOADED BY INPWAK, STORED IN /BILWAK/
```

```
*
*    SITE         GG   PB   SEQ  SUR
*    WIDTH  (M)   40   50   40   40
*    HEIGHT (M)   60   50   40   50
*
* BUILDING WIDTH (METERS)
*
WEBUILDW001   40.  *  SURRY
*
* BUILDING HEIGHT (METERS)
*
WEBUILDH001   50.  *  SURRY
*
*
* 3412 MWTH PWR CORE INVENTORY, END-OF-CYCLE
*           SUPPLIED BY D.E. BENNETT, 5/14/86
*  r e p l a c e d  by  SURRY LP/SD specific data
* LPSD Window 1 inventory (48 hours) 12/7/93
*
*               NUCNAM        CORINV(BQ)
*
RDCORINV001     CO-58         0.000E+00
RDCORINV002     CO-60         0.000E+00
RDCORINV003     KR-85         1.762E+16
RDCORINV004     KR-85M        4.410E+14
RDCORINV005     KR-87         6.194E+06
RDCORINV006     KR-88         1.618E+13
RDCORINV007     RB-86         3.148E+15
RDCORINV008     SR-89         2.647E+18
RDCORINV009     SR-90         1.343E+17
RDCORINV010     SR-91         1.006E+17
RDCORINV011     SR-92         1.669E+13
RDCORINV012     Y-90          1.383E+17
RDCORINV013     Y-91          3.383E+18
RDCORINV014     Y-92          1.211E+15
RDCORINV015     Y-93          1.548E+17
RDCORINV016     ZR-95         4.385E+18
RDCORINV017     ZR-97         6.246E+17
RDCORINV018     NB-95         4.362E+18
RDCORINV019     MO-99         2.916E+18
RDCORINV020     TC-99M        2.807E+18
RDCORINV021     RU-103        3.770E+18
RDCORINV022     RU-105        1.480E+15
RDCORINV023     RU-106        9.183E+17
RDCORINV024     RH-105        1.067E+18
RDCORINV025     SB-127        1.955E+17
RDCORINV026     SB-129        3.818E+14
RDCORINV027     TE-127        2.179E+17
RDCORINV028     TE-127M       3.229E+16
RDCORINV029     TE-129        7.733E+16
RDCORINV030     TE-129M       1.181E+17
RDCORINV031     TE-131M       1.252E+17
RDCORINV032     TE-132        2.449E+18
RDCORINV033     I-131         2.258E+18
RDCORINV034     I-132         2.525E+18
RDCORINV035     I-133         1.118E+18
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RDCORINV036     I-134          7.748E+02
RDCORINV037     I-135          3.292E+16
RDCORINV038     XE-133         4.784E+18
RDCORINV039     XE-135         2.885E+17
RDCORINV040     CS-134         2.130E+17
RDCORINV041     CS-136         8.081E+16
RDCORINV042     CS-137         1.807E+17
RDCORINV043     BA-139         1.805E+08
RDCORINV044     BA-140         4.207E+18
RDCORINV045     LA-140         4.581E+18
RDCORINV046     LA-141         1.015E+15
RDCORINV047     LA-142         2.155E+09
RDCORINV048     CE-141         4.336E+18
RDCORINV049     CE-143         1.525E+18
RDCORINV050     CE-144         2.577E+18
RDCORINV051     PR-143         3.966E+18
RDCORINV052     ND-147         1.558E+18
RDCORINV053     NP-239         2.905E+19
RDCORINV054     PU-238         2.758E+15
RDCORINV055     PU-239         7.071E+14
RDCORINV056     PU-240         8.936E+14
RDCORINV057     PU-241         2.042E+17
RDCORINV058     AM-241         0.000E+00
RDCORINV059     CM-242         0.000E+00
RDCORINV060     CM-244         0.000E+00
*
RDCORSCA001   1.0
*
* PARTICLE SIZE DISTRIBUTION OF EACH NUCLIDE GROUP
* YOU MUST SPECIFY A COLUMN OF DATA FOR EACH OF THE PARTICLE SIZE GROUPS
*
RDPSDIST001      1.
RDPSDIST002      1.
RDPSDIST003      1.
RDPSDIST004      1.
RDPSDIST005      1.
RDPSDIST006      1.
RDPSDIST007      1.
RDPSDIST008      1.
RDPSDIST009      1.
****************************************************************************
* OUTPUT CONTROL DATA BLOCK, LOADED BY INPOPT, STORED IN /ATMOPT/
*
OCIDEBUG001  0
*
* NAME OF THE NUCLIDE TO BE LISTED ON THE DISPERSION LISTINGS
*
OCNUCOUT001  CS-137
****************************************************************************
* METEOROLOGICAL SAMPLING DATA BLOCK
*
* METEOROLOGICAL SAMPLING OPTION CODE:
*
* METCOD = 1, USER SPECIFIED DAY AND HOUR IN THE YEAR (FROM MET FILE),
*          2, WEATHER CATEGORY BIN SAMPLING,
*          3, 120 HOURS OF WEATHER SPECIFIED ON THE ATMOS USER INPUT FILE,
```

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

```
*          4, CONSTANT MET (BOUNDARY WEATHER USED FROM THE START),
*          5, STRATIFIED RANDOM SAMPLES FOR EACH DAY OF THE YEAR.
*
M1METCOD001  2
M3ISTRDY001  1
M3ISTRHR001  1
*
* LAST SPATIAL INTERVAL FOR MEASURED WEATHER
*
M2LIMSPA001  25
*
*     BOUNDARY WEATHER, NO RAIN, WIND SPEED = 0.5 M/S, A-STABILITY,
*     MIXING HEIGHT = 1000 M, APPLIES TO THE LAST SPATIAL INTERVAL
*     (500 - 1000 MILES)
*
* BOUNDARY WEATHER MIXING LAYER HEIGHT
*
M2BNDMXH001  1000.  (METERS)
*
* BOUNDARY WEATHER STABILITY CLASS INDEX
*
M2IBDSTB001  1       (A-STABILITY)
*
* BOUNDARY WEATHER RAIN RATE
*
M2BNDRAN001  0.      (0 MM / HOUR = NO RAIN)
*
* BOUNDARY WEATHER WIND SPEED
*
M2BNDWND001  0.5     (M / S)
*
* NUMBER OF SAMPLES PER BIN
*
M4NSMPLS001  4  (THIS NUMBER SHOULD BE SET TO 4 FOR RISK ASSESSMENT)
*
* NUMBER OF RAIN DISTANCE INTERVALS FOR BINNING
*
M4NRNINT001  6
*
* ENDPOINTS OF THE RAIN DISTANCE INTERVALS (KILOMETERS)
*
* NOTE: THESE MUST BE CHOSEN TO MATCH THE SPATIAL ENDPOINT DISTANCES
*       SPECIFIED FOR THE ARRAY SPAEND (10 % ERROR IS ALLOWED).
*
*            2.0   3.5    7.0   13.0   25.0   50.0   MILES
*
M4RNDSTS001  3.22  5.63  11.27  20.92  40.23  80.47  KM
*
* NUMBER OF RAIN INTENSITIY BREAKPOINTS
*
M4NRINTN001   3
*
* RAIN INTENSITY BREAKPOINTS FOR WEATHER BINNING (MILLIMETERS PER HOUR)
*
M4RNRATE001   1.  2.  3.
*
```

```
* INITIAL SEED FOR RANDOM NUMBER GENERATOR
*
M4IRSEED001   1

********************** RELEASE DATA BLOCK *************
*
* WARNING TIME
*
* 1 hr based on similarity to V-sequence in 1150
RDOALARM001 3.6E+3
*
* SELECTION OF RISK DOMINANT PLUME
*
RDMAXRIS001          1
*
* REFERENCE TIME FOR DISPERSION AND RADIOACTIVE DECAY
*
RDREFTIM001         0.     .5
*
* NUMBER OF PLUME SEGMENTS THAT ARE RELEASED
*
RDNUMREL001         1
*
* HEAT CONTENT OF THE RELEASE SEGMENTS (W)
* A VALUE SPECIFIED FOR EACH OF THE RELEASE SEGMENTS
*
* release power is zero...
RDPLHEAT001 0.0
*
* HEIGHT OF THE PLUME SEGMENTS AT RELEASE (M)
* A VALUE SPECIFIED FOR EACH OF THE RELEASE SEGMENTS
*
* 12/7/93  hatch elevation: 27'5" approx. 8.4 m
RDPLHITE001   8.4
*
* DURATION OF THE PLUME SEGMENTS (S)
* A VALUE SPECIFIED FOR EACH OF THE RELEASE SEGMENTS
*
* release duration 6 hr
RDPLUDUR001 21600.
*
* TIME OF RELEASE FOR EACH PLUME (S AFTER SCRAM)
* A VALUE SPECIFIED FOR EACH OF THE RELEASE SEGMENTS
*
* start of release 2hr based on MELCOR calculations
RDPDELAY001 7.2E+3
*
* RELEASE FRACTIONS FOR ISOTOPE GROUPS IN RELEASE
* 25 Source Term Groups from Partitioning
*
RDATNAM2001 'Group  1'
*        Xe      I      Cs      Te      Sr      Ru      La      Ce      Ba
RDRELFRC001 1.6E-2 1.4E-4 6.6E-7 3.2E-7 6.3E-8 6.8E-9 5.8E-9 1.3E-8 5.7E-8
*
RDATNAM2001 'Group  2'
*        Xe      I      Cs      Te      Sr      Ru      La      Ce      Ba
```

```
RDRELFRC001 9.9E-1 4.3E-3 8.9E-5 2.5E-5 1.9E-6 2.8E-7 1.5E-7 4.9E-7 2.0E-6
.
RDATNAM2001 'Group  3'
*        Xe       I       Cs       Te       Sr       Ru       La      Ce       Ba
RDRELFRC001 9.8E-1 7.8E-3 1.5E-4 4.4E-5 2.1E-6 3.5E-7 1.7E-7 4.2E-7 2.3E-6
.
RDATNAM2001 'Group  4'
*        Xe       I       Cs       Te       Sr       Ru       La      Ce       Ba
RDRELFRC001 9.5E-1 1.1E-2 3.0E-4 1.2E-4 9.1E-6 2.1E-6 7.5E-7 2.2E-6 9.9E-6
.
RDATNAM2001 'Group  5'
*        Xe       I       Cs       Te       Sr       Ru       La      Ce       Ba
RDRELFRC001 9.3E-1 1.5E-2 4.5E-4 2.0E-4 1.6E-5 3.8E-6 8.8E-7 2.6E-6 1.8E-5
.
RDATNAM2001 'Group  6'
*        Xe       I       Cs       Te       Sr       Ru       La      Ce       Ba
RDRELFRC001 9.6E-1 2.0E-2 8.0E-4 4.0E-4 7.9E-5 1.0E-5 8.1E-6 1.4E-5 7.3E-5
.
RDATNAM2001 'Group  7'
*        Xe       I       Cs       Te       Sr       Ru       La      Ce       Ba
RDRELFRC001 8.4E-1 2.0E-2 1.9E-3 5.1E-4 3.9E-5 9.8E-6 2.5E-6 4.7E-6 4.3E-5
.
RDATNAM2001 'Group  8'
*        Xe       I       Cs       Te       Sr       Ru       La      Ce       Ba
RDRELFRC001 9.8E-1 4.8E-2 1.5E-3 9.1E-4 2.0E-4 1.7E-5 2.0E-5 2.8E-5 1.7E-4
.
RDATNAM2001 'Group  9'
*        Xe       I       Cs       Te       Sr       Ru       La      Ce       Ba
RDRELFRC001 9.6E-1 3.4E-2 4.0E-3 2.5E-3 3.9E-4 3.7E-5 4.7E-5 6.0E-5 3.3E-4
.
RDATNAM2001 'Group 10'
*        Xe       I       Cs       Te       Sr       Ru       La      Ce       Ba
RDRELFRC001 4.3E-1 2.3E-2 8.5E-3 9.9E-4 8.1E-5 1.1E-5 6.1E-6 1.0E-5 9.8E-5
.
RDATNAM2001 'Group 11'
*        Xe       I       Cs       Te       Sr       Ru       La      Ce       Ba
RDRELFRC001 6.5E-1 2.9E-2 1.2E-2 2.2E-3 1.2E-4 4.1E-5 1.3E-5 1.7E-5 1.4E-4
.
RDATNAM2001 'Group 12'
*        Xe       I       Cs       Te       Sr       Ru       La      Ce       Ba
RDRELFRC001 7.6E-1 4.4E-2 2.2E-2 4.5E-3 2.2E-4 3.5E-5 1.3E-5 3.4E-5 2.7E-4
.
RDATNAM2001 'Group 13'
*        Xe       I       Cs       Te       Sr       Ru       La      Ce       Ba
RDRELFRC001 7.7E-1 5.2E-2 2.9E-2 6.8E-3 4.0E-4 1.2E-4 2.7E-5 4.9E-5 4.7E-4
.
RDATNAM2001 'Group 14'
*        Xe       I       Cs       Te       Sr       Ru       La      Ce       Ba
RDRELFRC001 9.2E-1 7.5E-2 4.4E-2 2.0E-2 1.6E-3 7.8E-4 8.7E-5 2.6E-4 2.2E-3
.
RDATNAM2001 'Group 15'
*        Xe       I       Cs       Te       Sr       Ru       La      Ce       Ba
RDRELFRC001 8.4E-1 1.1E-1 6.5E-2 1.9E-2 1.0E-3 1.8E-4 4.7E-5 6.2E-5 1.0E-3
.
RDATNAM2001 'Group 16'
*        Xe       I       Cs       Te       Sr       Ru       La      Ce       Ba
```

```
RDRELFRC001 8.6E-1 1.5E-1 1.0E-1 3.5E-2 2.1E-3 5.0E-4 1.4E-4 2.6E-4 2.3E-3
.
RDATNAM2001 'Group 17'
*        Xe       I        Cs       Te       Sr       Ru       La       Ce       Ba
RDRELFRC001 9.9E-1 1.9E-1 1.5E-1 9.4E-2 9.8E-3 2.0E-3 5.6E-4 1.0E-3 9.0E-3
.
RDATNAM2001 'Group 18'
*        Xe       I        Cs       Te       Sr       Ru       La       Ce       Ba
RDRELFRC001 9.8E-1 2.1E-1 1.8E-1 1.1E-1 3.3E-2 2.4E-3 3.7E-3 4.7E-3 2.7E-2
.
RDATNAM2001 'Group 19'
*        Xe       I        Cs       Te       Sr       Ru       La       Ce       Ba
RDRELFRC001 1.0E+00 3.7E-1 2.9E-1 1.6E-1 3.5E-2 4.7E-3 2.4E-3 6.1E-3 3.2E-2
.
RDATNAM2001 'Group 20'
*        Xe       I        Cs       Te       Sr       Ru       La       Ce       Ba
RDRELFRC001 1.0E+00 5.3E-1 5.1E-1 2.1E-1 2.5E-2 3.7E-3 1.9E-3 4.2E-3 2.2E-2
.
RDATNAM2001 'Group 21'
*        Xe       I        Cs       Te       Sr       Ru       La       Ce       Ba
RDRELFRC001 1.0E+00 6.2E-1 5.8E-1 3.5E-1 1.3E-1 1.3E-2 1.2E-2 2.4E-2 1.1E-1
.
RDATNAM2001 'Group 22'
*        Xe       I        Cs       Te       Sr       Ru       La       Ce       Ba
RDRELFRC001 1.0E+00 6.0E-1 5.6E-1 4.9E-1 3.2E-1 4.0E-2 4.0E-2 5.7E-2 3.0E-1
.
RDATNAM2001 'Group 23'
*        Xe       I        Cs       Te       Sr       Ru       La       Ce       Ba
RDRELFRC001 1.0E+00 5.8E-1 5.9E-1 6.7E-1 5.6E-1 4.4E-2 1.0E-1 1.1E-1 5.2E-1
.
RDATNAM2001 'Group 24'
*        Xe       I        Cs       Te       Sr       Ru       La       Ce       Ba
RDRELFRC001 1.0E+00 6.3E-1 6.4E-1 6.7E-1 6.0E-1 7.5E-2 1.1E-1 1.7E-1 5.9E-1
.
RDATNAM2001 'Group 25'
*        Xe       I        Cs       Te       Sr       Ru       La       Ce       Ba
RDRELFRC001 1.0E+00 8.9E-1 8.9E-1 8.8E-1 8.9E-1 1.7E-1 9.3E-2 6.0E-1 8.9E-1
.
```

D.2 EARLY Input File

```
* GENERAL DESCRIPTIVE TITLE DESCRIBING THIS "EARLY" INPUT FILE
* ... used for Surry LP/SD study; modified for MACCS 1.5.11.1
MIEANAM1001  'SURRY EARLY INPUT FOR FINAL NUREG-1150 CALCULATIONS'
*
* FLAG TO INDICATE THAT THIS IS THE LAST PROGRAM IN THE SERIES TO BE RUN
*
MIENDAT2001  .FALSE.    (SET THIS VALUE TO .TRUE. TO SKIP CHRONC)
*
* DISPERSION MODEL OPTION CODE:  1  *  STRAIGHT LINE
*                                2  *  WIND-SHIFT WITH ROTATION
*                                3  *  WIND-SHIFT WITHOUT ROTATION
*
MIIPLUME001  2
*
* NUMBER OF FINE GRID SUBDIVISIONS USED BY THE MODEL
*
MINUMFIN001  7    (3, 5 OR 7 ALLOWED)
*
* LEVEL OF DEBUG OUTPUT REQUIRED, NORMAL RUNS SHOULD SPECIFY ZERO
*
MIIPRINT001  0
*
* FLAG INDICATING IF WIND-ROSES FROM ATMOS ARE TO BE OVERRIDDEN
*
MIOVRRID001  .FALSE.  (USE THE WIND ROSE CALCULATED FOR EACH WEATHER BIN)
*
* LOGICAL FLAG SIGNIFYING THAT THE BREAKDOWN OF RISK BY WEATHER CATEGORY
* BIN ARE TO BE PRESENTED TO SHOW THEIR RELATIVE CONTRIBUTION TO THE MEAN
*
*             RISBIN
*
MIRISCAT001   .FALSE.
*********************************************************************************
* POPULATION DISTRIBUTION DATA BLOCK, LOADED BY INPOPU, STORED IN /POPDAT/
*
PDPOPFLG001  FILE
*
*PDPOPFLG001  UNIFORM
*PDIBEGIN001  1   (SPATIAL INTERVAL AT WHICH POPULATION BEGINS)
*PDPOPDEN001  50. (POPULATION DENSITY (PEOPLE PER SQUARE KILOMETER))
*********************************************************************************
* ORGAN DEFINITION DATA BLOCK, LOADED BY INORGA, STORED IN /EARDIM/ AND /ORGNAM/
*
* NUMBER OF ORGANS DEFINED FOR HEALTH EFFECTS
*
*********************************************************************************
* SHIELDING AND EXPOSURE FACTORS, LOADED BY INDFAC, STORED IN /EADFAC/
*
* THREE VALUES OF EACH PROTECTION FACTOR ARE SUPPLIED,
* ONE FOR EACH TYPE OF ACTIVTY:
*
* ACTIVITY TYPE:
*     1 - EVACUEES WHILE MOVING
```

```
*      2 - NORMAL ACTIVITY IN SHELTERING AND EVACUATION ZONE
*      3 - SHELTERED ACTIVITY
*
* CLOUD SHIELDING FACTOR
*
*     SITE         GG   PB   SEQ  SUR  ZION
*     SHELTERING  0.7  0.5  0.65 0.6  0.5
*
*               EVACUEES  NORMAL  SHELTER
*
SECSFACT001      1.       0.75     0.6   *  SURRY SHELTERING VALUE
*
* PROTECTION FACTOR FOR INHALATION
*
SEPROTIN001      1.       0.41      0.33  * VALUES FOR NORMAL ACTIVITY AND
*                                          SHELTERING SELECTED BY S. ACHARYA, NRC
*
* BREATHING RATE (CUBIC METERS PER SECOND)
*
SEBRRATE001  2.66E-4  2.66E-4  2.66E-4
*
* SKIN PROTECTION FACTOR
*
SESKPFAC001  1.0       0.41     0.33    * VALUES FOR NORMAL ACTIVITY AND
*                                          SHELTERING SELECTED BY S. ACHARYA, NRC
*
* GROUND SHIELDING FACTOR
*
*     SITE         GG   PB   SEQ  SUR  ZION
*     SHELTERING  0.25 0.1  0.2  0.2  0.1
*
SEGSHFAC001      0.5     0.33    0.2   * VALUE FOR NORMAL ACTIVITY SELECTED BY
*                                       S. ACHARYA, NRC;  SURRY SHELTERING VALUE
*
* RESUSPENSION INHALATION MODEL CONCENTRATION COEFFICIENT (/METER)
*
*   RESCON = 1.E-4 IS APPROPRIATE FOR MECHANICAL RESUSPENSION BY VEHICLES.
*   RESHAF = 2.11 DAYS CAUSES 1.E-4 TO DECAY IN ONE WEEK TO 1.E-5, THE VALUE
*   OF RESCON USED IN THE FIRST TERM OF THE LONG-TERM RESUSPENSION EQUATION
*   USED IN CHRONC.
*
SERESCON001   1.E-4     (RESUSPENSION IS TURNED ON)
*
* RESUSPENSION CONCENTRATION COEFFICIENT HALF-LIFE (SEC)
*
SERESHAF001   1.82E5    (2.11 DAYS)
*********************************************************************************
* EVACUATION ZONE DATA BLOCK, LOADED BY EVNETW, STORED IN /NETWOR/, /EOPTIO/
*
* THE TYPE OF WEIGHTING TO BE APPLIED TO THE EMERGENCY RESPONSE SCENARIOS
*   YOU MUST SUPPLY A VALUE OF 'TIME' OR 'PEOPLE'
*
EZWTNAME001  'TIME'
*
*      SITE          GG   PB   SEQ  SUR
*      CLDELAY (HR) 0.75 1.0  1.8  1.5
```

```
*       ESPEED (M/S) 3.7  4.8  1.4  1.8
*
*       EDELAY = CLDELAY + 0.5 HR
*
*       CLDELAY = DELAY BETWEEN WARNING OF PUBLIC TO BEGIN EVACUATION AND
*                 TIME EVACUATION ACTUALLY BEGINS; VALUES USED ARE DEVELOPED
*                 FROM SITE-SPECIFIC CLEAR TIME STUDIES
*
*       0.5 HR = MEAN (EXPECTED) TIME FROM GENERAL EMERGENCY
*                CONDITIONS TO WARNING OF PUBLIC (SIRENS, BROADCAST)
*
*
* RADIAL EVACUATION SPEED (M/S)
*
EZESPEED001    1.8    *  SURRY
*
* DURATION OF THE EMERGENCY PHASE (SECONDS FROM PLUME ARRIVAL)
*
SRENDEMP001  604800.  (ONE WEEK)
*
* CRITICAL ORGAN FOR RELOCATION DECISIONS
*
SRCRIORG001 'EDEWBODY'
*
* HOT SPOT RELOCATION TIME (SECONDS FROM PLUME ARRIVAL)
*
SRTIMHOT001  43200.     (ONE-HALF DAY)
*
* NORMAL RELOCATION TIME (SECONDS FROM PLUME ARRIVAL)
*
SRTIMNRM001  86400.     (ONE DAY)
*
* HOT SPOT RELOCATION DOSE CRITERION THRESHOLD (SIEVERTS)
*
SRDOSHOT001  0.5    (50 REM DOSE TO WHOLE BODY IN 1 WEEK TRIGGERS RELOCATION)
*
* NORMAL RELOCATION DOSE CRITERION THRESHOLD (SIEVERTS)
*
SRDOSNRM001  0.25   (25 REM DOSE TO WHOLE BODY IN 1 WEEK TRIGGERS RELOCATION)
*****************************************************************************
*****************************************************************************
* RESULT 1 OPTIONS BLOCK, LOADED BY INOUT1, STORED IN /INOUT1/
* TOTAL NUMBER OF A GIVEN EFFECT (LATENT CANCER, EARLY DEATH, EARLY INJURY)
*
* NUMBER OF DESIRED RESULTS OF THIS TYPE
*
TYPE1NUMBER    2
*
TYPE1OUT001    'ERL FAT/TOTAL'                 1  19 (50 MILES)
TYPE1OUT002    'CAN FAT/TOTAL'                 1  19  CCDF
*****************************************************************************
* RESULT 2 OPTIONS BLOCK, LOADED BY INOUT2, STORED IN /INOUT2/
* FURTHEST DISTANCE AT WHICH A GIVEN RISK OF EARLY DEATH IS EXCEEDED.
*
* NUMBER OF DESIRED RESULTS OF THIS TYPE
*
```

```
TYPE2NUMBER  0
*******************************************************************************
* RESULT 3 OPTIONS BLOCK, LOADED BY INOUT3, STORED IN /INOUT3/
* NUMBER OF PEOPLE WHOSE ACUTE DOSE TO A GIVEN ORGAN EXCEEDS A GIVEN THRESHOLD.
*
* NUMBER OF DESIRED RESULTS OF THIS TYPE
*
TYPE3NUMBER  0
*******************************************************************************
* RESULT 4 OPTIONS BLOCK, LOADED BY INOUT4, STORED IN /INOUT4/
* 360 DEGREE AVERAGE RISK OF A GIVEN EFFECT AT A GIVEN DISTANCE.
* NUMBER OF DESIRED RESULTS OF THIS TYPE
*
TYPE4NUMBER  0
*******************************************************************************
* RESULT 5 OPTIONS BLOCK, LOADED BY INOUT5, STORED IN /INOUT5/
*
* TOTAL POPULATION DOSE TO A GIVEN ORGAN BETWEEN TWO DISTANCES.
*
* NUMBER OF DESIRED RESULTS OF THIS TYPE
*
TYPE5NUMBER     1
TYPE5OUT001  'EDEWBODY'    1     19 CCDF (0-50 MILES)
*******************************************************************************
* RESULT 6 OPTIONS BLOCK, LOADED BY INOUT6, STORED IN /INOUT6/
*
* CENTERLINE DOSE TO AN ORGAN VS DIST BY PATHWAY, PATHWAY NAMES ARE AS FOLLOWS:
* NUMBER OF DESIRED RESULTS OF THIS TYPE
*
TYPE6NUMBER     0
*******************************************************************************
* RESULT 7 OPTIONS BLOCK, LOADED BY INOUT7, STORED IN /INOUT7/
*
* CENTERLINE RISK OF A GIVEN EFFECT VS DISTANCE
*
* NUMBER OF DESIRED RESULTS OF THIS TYPE
*
TYPE7NUMBER   0
*******************************************************************************
* RESULT 8 OPTIONS BLOCK, LOADED BY INOUT8, STORED IN /INOUT8/
*
* POPULATION WEIGHTED FATALITY RISK BETWEEN 2 DISTANCES
*
* NUMBER OF DESIRED RESULTS OF THIS TYPE
*
TYPE8NUMBER   0
*
*******************************************************************************
* copied from in2a.inp for MACCS Version 1.5.11.1
* EARLY FATALITY MODEL PARAMETERS, LOADED BY INEFAT, STORED IN /EFATAL/
*
* NUMBER OF EARLY FATALITY EFFECTS
*
EFNUMEFA001   3
*
*            ORGNAM        EFFACA   EFFACB   EFFTHR
```

```
*
EFATAGRP001 'RED MARR'      3.8      5.0    1.5
EFATAGRP002 'LUNGS'        10.0      7.0    5.0
EFATAGRP003 'LOWER LI'     15.0     10.0    8.0
*******************************************************************************
* EARLY INJURY MODEL PARAMETERS, LOADED BY INEINJ, STORED IN /EINJUR/
*
* NUMBER OF EARLY INJURY EFFECTS
*
EINUMEIN001  7
*
*           EINAME            ORGNAM   EISUSC EITHRE EIFACA EIFACB
*
EINJUGRP001 'PRODROMAL VOMIT' 'STOMACH'  1.     .5     2.     3.
EINJUGRP002 'DIARRHEA'        'STOMACH'  1.     1.     3.     2.5
EINJUGRP003 'PNEUMONITIS'     'LUNGS'    1.     5.    10.     7.
EINJUGRP004 'SKIN ERYTHEMA'   'SKIN'     1.     3.     6.     5.
EINJUGRP005 'TRANSEPIDERMAL'  'SKIN'     1.    10.    20.     5.
EINJUGRP006 'THYROIDITIS'     'THYROIDH' 1.    40.   240.     2.
EINJUGRP007 'HYPOTHYROIDISM'  'THYROIDH' 1.     2.    60.     1.3
*******************************************************************************
* ACUTE EXPOSURE CANCER PARAMETERS, LOADED BY INACAN STORED IN /ACANCR/.
*
* NUMBER OF ACUTE EXPOSURE CANCER EFFECTS
*
LCNUMACA001   7
*
* THRESHOLD DOSE FOR APPLYING THE DOSE DEPENDENT REDUCTION FACTOR
*
LCDDTHRE001   0.2  (LOWEST DOSE FOR WHICH DDREFA WILL BE APPLIED)
*
* DOSE THRESHOLD FOR LINEAR DOSE RESPONSE (SV)
*
LCACTHRE001   0.0  (LINEAR-QUADRATIC MODEL IS NOT BEING USED)
*
*           ACNAME     ORGNAM  ACSUSC DOSEFA DOSEFB CFRISK    CIRISK  DDREFA
*
LCANCERS001 'LEUKEMIA' 'RED MARR'  1.0   1.0   0.0   9.70E-3   9.70E-3  2.0
LCANCERS002 'BONE'     'BONE SUR'  1.0   1.0   0.0   9.00E-4   9.00E-4  2.0
LCANCERS003 'BREAST'   'BREAST'    1.0   1.0   0.0   5.40E-3   1.59E-2  1.0
LCANCERS004 'LUNG'     'LUNGS'     1.0   1.0   0.0   1.55E-2   1.73E-2  2.0
LCANCERS005 'THYROID'  'THYROIDH'  1.0   1.0   0.0   7.20E-4   7.20E-3  1.0
LCANCERS006 'GI'       'LOWER LI'  1.0   1.0   0.0   3.36E-2   5.75E-2  2.0
LCANCERS007 'OTHER'    'BLAD WAL'  1.0   1.0   0.0   2.76E-2   5.52E-2  2.0
ODNUMORG001   10
*
* NAMES OF THE ORGANS DEFINED FOR HEALTH EFFECTS
*
ODORGNAM001 'SKIN', 'EDEWBODY', 'LUNGS', 'RED MARR', 'LOWER LI', 'STOMACH',
ODORGNAM002 'THYROIDH', 'BONE SUR', 'BREAST', 'BLAD WAL'
*****************************************************
*        EMERGENCY RESPONSE SCENARIO
*
EZEANAM2001  'EVACUATION WITHIN 10 MILES'
*
* FRACTION OF THE TIME THIS SCENARIO AFFECTS
```

```
*
EZWTFRAC001    .995
*
* LAST RING IN THE MOVEMENT ZONE
*
EZLASMOV001      15
*
*  FIRST SPATIAL INTERVAL IN THE EVACUATION ZONE
*
EZINIEVA001    1    (NO INNER SHELTER ZONE)
*
*  DISTANCE INTERVALS OF THE THREE EVACUATION ZONES
*
EZLASEVA001  0    0      12
*
* EVAC DELAY TIMES FOR THE THREE EVAC DELAY RINGS:
* TIME FOR PEOPLE TO GET MOVING AFTER BEING WARNED
*
EZEDELAY001  0.  0.     7200.
****************************************************
*            SHELTER RESPONSE DEFINITION
*
* TIME TO TAKE SHELTER (INNER SHELTER ZONE) (S)
*
SRTTOSH1001          0.
*
* SHELTER DURATION (INNER SHELTER ZONE) (S)
*
SRSHELT1001          0.
*
* LAST RING (OUTER SHELTER ZONE)
*
SRLASHE2001           0
*
* TIME TO TAKE SHELTER (OUTER SHELTER ZONE) (S)
*
SRTTOSH2001          0.
*
* SHELTER DURATION (OUTER SHELTER ZONE) (S)
*
SRSHELT2001          0.
.
*********************************************************
*          EMERGENCY RESPONSE SCENARIO
*
EZEANAM2001  'NO EVACUATION'
*
* FRACTION OF THE TIME THIS SCENARIO AFFECTS
*
EZWTFRAC001  0.005
*
* LAST RING IN THE MOVEMENT ZONE
*
EZLASMOV001       0
*
*  FIRST SPATIAL INTERVAL IN THE EVACUATION ZONE
```

```
*
EZINIEVA001   1   (NO INNER SHELTER ZONE)
*
*  DISTANCE INTERVALS OF THE THREE EVACUATION ZONES
*
EZLASEVA001  0   0      0
*
* EVAC DELAY TIMES FOR THE THREE EVAC DELAY RINGS:
* TIME FOR PEOPLE TO GET MOVING AFTER BEING WARNED
*
EZEDELAY001  0.  0.        0.
***************************************************
*           SHELTER RESPONSE DEFINITION
*
* TIME TO TAKE SHELTER (INNER SHELTER ZONE) (S)
*
SRTTOSH1001          0.
*
* SHELTER DURATION (INNER SHELTER ZONE) (S)
*
SRSHELT1001          0.
*
* LAST RING (OUTER SHELTER ZONE)
*
SRLASHE2001           0
*
* TIME TO TAKE SHELTER (OUTER SHELTER ZONE) (S)
*
SRTTOSH2001          0.
*
* SHELTER DURATION (OUTER SHELTER ZONE) (S)
*
SRSHELT2001          0.
```

D.3 CHRONC Input File

```
* GENERAL DESCRIPTIVE TITLE DESCRIBING THIS "CHRONC" INPUT FILE
* a) used for Surry LP/SD study
* b) foodchain related data are copied from in3a.inp, Version 1.5.11.1
CHCHNAME001  'SURRY CHRONC INPUT FOR FINAL NUREG-1150 CALCULATIONS'
********************************************************************************
* EMERGENCY RESPONSE COST DATA BLOCK
*
* EVACUATION COST (DOLLARS/PERSON-DAY)
*
CHEVACST001  27.00
*
* RELOCATION DOST (DOLLARS/PERSON-DAY)
*
CHRELCST001  27.00
********************************************************************************
* LONG TERM PROTECTIVE ACTION DATA BLOCK
*
* END OF THE INTERMEDIATE PHASE PERIOD (SECONDS FROM ACCIDENT INITIATION)
*
CHTMIPND001  604800.             (7 DAYS, NO INTERMEDIATE PHASE)
*
* ACTION PERIOD (PROJECTION PERIOD) FROM THE START OF THE LONG TERM PHASE,
* THE POINT AT WHICH THE LONG TERM DOSE CRITERION IS EVALUATED (SECONDS)
*
CHTMPACT001   1.58E8             (5 YEARS)
*
* DOSE CRITERION FOR INTERMEDIATE PHASE RELOCATION (SV)
*
CHDSCRTI001   1.0E5              (NO INTERMEDIATE PHASE RELOCATION)
*
* DOSE CRITERION FOR LONG TERM PHASE RELOCATION (SV)
*
CHDSCRLT001   0.04   (2 REM IN FIRST YEAR, 0.5 REM PER YEAR FOR YRS 2 - 5)
*
* CRITICAL ORGAN NAME FOR LONG-TERM ACTIONS
*
CHCRTOCR001 'EDEWBODY'
*
********************************************************************************
* DECONTAMINATION PLAN DATA BLOCK
*
* NUMBER OF LEVELS OF DECONTAMINATION
*
CHLVLDEC001  2
*
* DECONTAMINATION TIMES CORRESPONDING TO THE LVLDEC LEVELS OF DECONTAMINATION
*  (SECONDS)
*
CHTIMDEC001   5.184E6  1.0368E7    (60, 120 DAYS)
*
* DOSE REDUCTION FACTORS CORRESPONDING TO THE LVLDEC LEVELS OF DECONTAMINATION
*
CHDSRFCT001   3.      15.
```

```
*
* COST OF FARM DECONTAMINATION PER UNIT AREA (DOLLARS/HECTARE)
*  FOR THE VARIOUS LEVELS OF DECONTAMINATION
*
CHCDFRM0001   562.5   1250.
*
* COST OF NONFARM DECONTAMINATION PER PERSON
*  FOR THE VARIOUS LEVELS OF DECONTAMINATION  (DOLLARS/PERSON)
*
CHCDNFRM001   3000.   8000.
*
* FRACTION OF FARMLAND DECONTAMINATION COST DUE TO LABOR
*  FOR THE VARIOUS DECONTAMINATION LEVELS
*
CHFRFDL0001  .3      .35
*
* FRACTION OF NON-FARM DECONTAMINATION COST DUE TO LABOR
*  FOR THE VARIOUS DECONTAMINATION LEVELS
*
CHFRNFDL001  .7      .5
*
* FRACTION OF TIME WORKERS IN FARM AREAS SPEND IN DECONTAMINATION WORK
*  FOR THE VARIOUS DECONTAMINATION LEVELS
*
CHTFWKF0001  .10     .33
*
* FRACTION OF TIME WORKERS IN NON-FARM AREAS SPEND IN DECONTAMINATION WORK
*  FOR THE VARIOUS DECONTAMINATION LEVELS
*
CHTFWKNF001  .33     .33
*
* AVERAGE COST OF DECONTAMINATION LABOR (DOLLARS/MAN-YEAR)
*
CHDLBCST001   35000.
*****************************************************************************
* INTERDICTION COST DATA BLOCK
*
* DEPRECIATION RATE DURING INTERDICTION PERIOD (PER YEAR)
*
CHDPRATE001    .20
*
* SOCIETAL DISCOUNT RATE DURING INTERDICTION PERIOD (PER YEAR)
*
CHDSRATE001    .12
*
* URBAN POPULATION REMOVAL COST (DOLLARS/PERSON)
*
CHPOPCST001   5000.
*****************************************************************************
* GROUNDSHINE WEATHERING DEFINITION DATA BLOCK
*
* NUMBER OF TERMS IN THE GROUNDSHINE WEATHERING RELATIONSHIP (EITHER 1 OR 2)
*
CHNGWTRM001   2
*
* GROUNDSHINE WEATHERING COEFFICIENTS
```

```
*
CHGWCOEF001   0.5    0.5            (GAYLE'S EQUATION)
*
* HALF LIVES CORRESPONDING TO THE GROUNDSHINE WEATHERING COEFFICIENTS (S)
*
CHTGWHLF001   1.6E7  2.8E9          (GAYLE'S EQUATION)
********************************************************************************
* RESUSPENSION WEATHERING DEFINITION DATA BLOCK
*
* NUMBER OF TERMS IN THE RESUSPENSION WEATHERING RELATIONSHIP
*
CHNRWTRM001    3
*
* RESUSPENSION CONCENTRATION COEFFICIENTS   (/ METER)
*  RELATIONSHIP BETWEEN GROUND CONCENTRATION AND INSTANTANEOUS AIR CONC.
*
CHRWCOEF001  1.0E-5   1.0E-7   1.0E-9
*
* HALF-LIVES CORRESPONDING TO THE RESUSPENSION CONCENTRATION COEFFICIENTS (S)
*
CHTRWHLF001  1.6E7    1.6E8    1.6E9  (6 MONTHS, 5 YEARS, 50 YEARS)
********************************************************************************
* SITE REGION DESCRIPTION DATA BLOCK
*
* FRACTION OF AREA THAT IS LAND IN THE REGION
*
CHFRACLD001   1.0E-35    (VALUE NOT USED SINCE SITE FILE PROVIDED)
*
* FRACTION OF LAND DEVOTED TO FARMING IN THE REGION
*
CHFRCFRM001   1.0E-35    (VALUE NOT USED SINCE SITE FILE PROVIDED)
*
* AVERAGE VALUE OF ANNUAL FARM PRODUCTION IN THE REGION (DOLLARS/HECTARE)
*  (CASH RECEIPTS FROM FARMING PLUS VALUE OF HOME CONSUMPTION)/(LAND IN FARMS)
*
CHFRMPRD001   0.         (VALUE NOT USED SINCE SITE FILE PROVIDED)
*
* FRACTION OF FARM PRODUCTION RESULTING FROM DAIRY PRODUCTION IN THE REGION
*  (VALUE OF MILK PRODUCED)/(CASH RECEIPTS FROM FARMING PLUS VALUE OF HOME...)
*
CHDPFRCT001   0.         (VALUE NOT USED SINCE SITE FILE PROVIDED)
*
* VALUE OF FARM WEALTH (DOLLARS/HECTARE)
*  (AVERAGE VALUE PER HECTARE OF FARM LAND AND BUILDINGS TO 100 MILES)
*
*  SITE                GG    LS    PB    SEQ   SUR   ZION
*  VALWF ($/HECTARE)   2561  3305  3421  1855  2613  2897
*
CHVALWF0001   3975.  *  SURRY
*
* FRACTION OF FARM WEALTH IN IMPROVEMENTS FOR THE REGION
*
*  SITE   GG    LS    PB    SEQ   SUR   ZION
*  FRFIM  0.3   0.19  0.25  0.27  0.25  0.49
*
CHFRFIM0001   0.25  *  SURRY
```

```
*
* NON-FARM WEALTH, PROPERTY AND IMPROVEMENTS FOR THE REGION (DOLLARS/PERSON)
*  THE VALUE OF ALL RESIDENTIAL, BUSINESS, AND PUBLIC ASSETS WHICH WOULD BE
*  LOST IN THE EVENT OF PERMANENT INTERDICTION OF THE AREA
*
*  SITE          GG   PB   SUR  SEQ  ZION
*  VALWNF ($K)   53   78   84   66   76
*
CHVALWNF001  123000.  *  SURRY  AGT BNL 8-26-93
*
* FRACTION OF NON-FARM WEALTH IN IMPROVEMENTS FOR THE REGION
*
CHFRNFIM001   0.8
*************************************************************************************
* SPECIAL OPTIONS DATA BLOCK
*
* DETAILED PRINT OPTION CONTROL SWITCHES, LOOK AT THE CODE BEFORE TURNING ON!!
*         (KCEPNT, KDFPNT, KDTPNT, KGCPNT, KLTPNT, KWTPNT, KSWRSK, KSWDSC)
*
CHKSWTCH001   0 0 0 0 0 0 0 0 0 0 0 0
*************************************************************************************
* WATER PATHWAY NUCLIDE DEFINITIONS FOR CHRONC
*
* NUMBER OF NUCLIDES IN THE WATER INGESTION PATHWAY MODEL
*
CHNUMWPI001    4
*
* TABLE OF NUCLIDE DEFINITIONS IN THE WATER INGESTION PATHWAY MODEL
*  WATER PATHWAY NUCLIDES MUST BE A SUBSET OF THE INGESTION MODEL NUCLIDES
*
* IF A SITE DATA FILE IS DEFINED, THE DATA DEFINING THE WATERSHED INGESTION
*  FACTOR IS SUPERSEDED BY THE CORRESPONDING DATA IN THE SITE DATA FILE
*
*       WINGF VALUES BY DRAINAGE SYSTEM
*  NUCLIDE     SR-89   SR-90   CS-134  CS-137
*  RIVER       5.0E-6  5.0E-6  5.0E-6  5.0E-6
*  GREAT LAKE  2.0E-7  2.0E-7  2.0E-6  4.0E-6
*  OCEAN       0.0     0.0     0.0     0.0
*
*  ALL NUREG-1150 SITES HAVE RIVER DRAINAGE SYSTEMS EXCEPT LASALLE AND ZION
*
*                          INITIAL     ANNUAL    INGESTION FACTOR
*              WATER       WASHOFF     WASHOFF    ((BQ INGESTED)/
*             NUCLIDE      FRACTION     RATE        (BQ IN WATER))
*
*              NAMWPI       WSHFRI      WSHRTA          WINGF
CHWTRIS0001    SR-89         0.01       0.004          5.0E-6
CHWTRIS0002    SR-90         0.01       0.004          5.0E-6
CHWTRIS0003    CS-134        0.005      0.001          5.0E-6
CHWTRIS0004    CS-137        0.005      0.001          5.0E-6
*************************************************************************************
* CROP PATHWAY DEFINITIONS FOR CHRONC
*
* MODIFIED 14 OCT 88, BY JLS, VALUES CHANGED TO THOSE DEVELOPED BY J. ROLLSTIN
*
* NUMBER OF DEFINED CROPS IN THE CHRONC FOOD INGESTION MODEL
```

```
*
CHNFICRP001     7    (UP TO 10 ALLOWED)
*
* NOTE TO USER: THE CODE MAKES SPECIAL TREATMENT OF CROP NAMES BEGINNING WITH
* 'PASTURE' DUE TO THE CONTINOUS NATURE OF THE HARVESTING PROCESS.
*
* IF THE USER WISHES TO DEFINE A NEW CROP CATEGORY FOR RANGELAND PASTURE,
*  IT SHOULD BE CALLED 'PASTURE-RANGE' OR 'PASTURE-DRY'
*
* TABLE OF CROP DEFINITIONS FOR THE CHRONC FOOD INGESTION MODEL
*
*                             FRACTION OF CROP CONSUMED BY
*
*                                          DAIRY    MEAT
*             CROP NAME             MAN   ANIMALS  ANIMALS
*
*              NAMCRP              FRCTCH  FRCTCM  FRCTCB
CHCRPTBL001 'PASTURE              ' 0.0     0.1     0.9
CHCRPTBL002 'STORED FORAGE        ' 0.0     0.13    0.87
CHCRPTBL003 'GRAINS               ' 0.35    0.040   0.61
CHCRPTBL004 'GRN LEAFY VEGETABLES' 1.0     0.0     0.0
CHCRPTBL005 'OTHER FOOD CROPS     ' 1.0     0.0     0.0
CHCRPTBL006 'LEGUMES AND SEEDS    ' 0.24    0.046   0.714
CHCRPTBL007 'ROOTS AND TUBERS     ' 1.0     0.0     0.0
*******************************************************************************
* CHRONC INGESTION PATHWAY NUCLIDE DEFINITIONS
*
* NUMBER OF NUCLIDES IN THE CHRONC FOOD INGESTION MODEL
*
CHNFIISO001  6  (UP TO 10 ALLOWED, BEWARE THAT DAUGHTER BUILDUP IS NOT TREATED)
*
* TABLE OF NUCLIDE DEFINITIONS IN THE CHRONC INGESTION PATHWAY MODEL
*
* NUCLIDES THAT WERE DEFINED IN THE WATER PATHWAY DATA ABOVE MUST BE
*  A SUBSET OF THE CHRONC INGESTION FOOD PATHWAY NUCLIDES.  THE WATER
*  PATHWAY NUCLIDES MUST BE LISTED FIRST IN THIS DATA BLOCK AND IN THE
*  SAME ORDER AS THEY WERE LISTED IN THE WATER PATHWAY DATA BLOCK
*
*                                                  TRANSFER FACTORS
*                         RETENTION FACTORS      [(BQ TRANSFERED)/
*           INGESTION   PROCESSING AND DECAY      (BQ INGESTED)]
*            NUCLIDE     MILK/MAN  MEAT/MAN        MILK      MEAT
*
*            NAMIPI       DCYPMH    DCYPBH         TFMLK     TFBF
CHISODEF001  SR-89        0.66      0.77           0.022     0.00022
CHISODEF002  SR-90        1.0       1.0            0.022     0.00022
CHISODEF003  CS-134       1.0       1.0            0.11      0.023
CHISODEF004  CS-137       1.0       1.0            0.11      0.024
CHISODEF005  I-131        0.28      0.18           0.13      0.0024
CHISODEF006  I-133        0.002     0.0            0.062     0.0011
*******************************************************************************
* TRANSFER FACTOR FROM SOIL TO PLANT BY ROOT-UPTAKE (AND BY SOIL INGESTION FOR
* GRAZING ON PASTURE) INTEGRATED OVER ALL TIME [(BQ TRANSFERED)/(BQ DEPOSITED)]
*
*                                               GREEN    OTHER    LEGUMES   ROOTS
*                           STORED              LEAFY    FOOD     AND       AND
```

```
*           NUCLIDE  PASTURE  FORAGE  GRAINS  VEG     CROPS   SEEDS    TUBERS
*
*           NAMISO   TCROOT   TCROOT  TCROOT  TCROOT  TCROOT  TCROOT   TCROOT
CHTCROOT001 SR-89    4.1E-4   1.3E-3  4.3E-5  1.7E-4  8.6E-6  3.7E-4   1.1E-4
CHTCROOT002 SR-90    2.6E-2   9.0E-2  3.3E-3  1.3E-2  6.6E-4  2.8E-2   8.4E-3
CHTCROOT003 CS-134   1.3E-3   7.1E-4  3.5E-5  1.4E-5  1.1E-4  9.3E-5   5.6E-5
CHTCROOT004 CS-137   6.9E-3   1.5E-3  7.6E-5  3.0E-5  2.3E-4  2.0E-4   1.2E-4
CHTCROOT005 I-131    1.6E-4   0.0     0.0     0.0     0.0     0.0      0.0
CHTCROOT006 I-133    1.7E-6   0.0     0.0     0.0     0.0     0.0      0.0
*****************************************************************************
*  RADIOACTIVE DECAY RETENTION FACTORS (I.E., 1 - F WHERE F = FRACTION OF
*  RADIOACTIVITY LOST BY DECAY) FOR NUCLIDES IN CROPS FROM TIME OF HARVEST
*  TO TIME OF CONSUMPTION BY HUMANS (FRACTION RETAINED)
*
*                                         GREEN   OTHER   LEGUMES  ROOTS
*                             STORED              LEAFY   FOOD    AND      AND
*           NUCLIDE  PASTURE  FORAGE  GRAINS  VEG     CROPS   SEEDS    TUBERS
*
*           NAMISO   DCYPCH   DCYPCH  DCYPCH  DCYPCH  DCYPCH  DCYPCH   DCYPCH
CHDCYPCH001 SR-89    0.0      0.0     0.18    0.67    0.21    0.18     0.18
CHDCYPCH002 SR-90    0.0      0.0     0.99    1.0     0.99    0.99     0.99
CHDCYPCH003 CS-134   0.0      0.0     0.84    0.96    0.85    0.84     0.84
CHDCYPCH004 CS-137   0.0      0.0     0.99    1.0     0.99    0.99     0.99
CHDCYPCH005 I-131    0.0      0.0     0.0099  0.21    0.024   0.0099   0.0099
CHDCYPCH006 I-133    0.0      0.0     0.0     0.0     0.0     0.0      0.0
*****************************************************************************
*  CROP PROCESSING AND PREPARATION RETENTION FACTORS FOR NUCLIDES IN FOOD
*  CROPS CONSUMED BY HUMANS (FRACTION RETAINED).  FACTORS REFLECT LOSS OF
*  NUCLIDES FROM FOODS DUE TO PROCESSING (E.G., WASHING OF FRUIT, PEELING
*  OF POTATOES, LOSSES DURING CANNING) AND FOOD PREPARATION (COOKING) FROM
*  THE TIME OF PROCESSING OF THE HARVESTED CROP TO THE TIME OF CONSUMPTION
*  BY HUMANS.  FACTORS DO NOT REFLECT LOSSES DUE TO RADIOACTIVE DECAY.
*
*                                         GREEN   OTHER   LEGUMES  ROOTS
*                             STORED              LEAFY   FOOD    AND      AND
*           NUCLIDE  PASTURE  FORAGE  GRAINS  VEG     CROPS   SEEDS    TUBERS
*
*           NAMISO   FPLSCH   FPLSCH  FPLSCH  FPLSCH  FPLSCH  FPLSCH   FPLSCH
CHFPLSCH001 SR-89    0.0      0.0     0.25    0.5     0.71    0.8      0.8
CHFPLSCH002 SR-90    0.0      0.0     0.25    0.5     0.71    0.8      0.8
CHFPLSCH003 CS-134   0.0      0.0     0.25    0.5     0.71    0.8      0.8
CHFPLSCH004 CS-137   0.0      0.0     0.25    0.5     0.71    0.8      0.8
CHFPLSCH005 I-131    0.0      0.0     0.33    0.5     0.71    0.8      0.8
CHFPLSCH006 I-133    0.0      0.0     0.33    0.5     0.71    0.8      0.8
*****************************************************************************
*  RETENTION FACTORS FOR NUCLIDES IN CROPS FROM TIME OF HARVEST TO TIME OF
*  CONSUMPTION BY MILK-PRODUCING ANIMALS (FRACTION RETAINED).  FACTOR REFLECTS
*  LOSSES DUE TO RADIOACTIVE DECAY.
*
*                                         GREEN   OTHER   LEGUMES  ROOTS
*                             STORED              LEAFY   FOOD    AND      AND
*           NUCLIDE  PASTURE  FORAGE  GRAINS  VEG     CROPS   SEEDS    TUBERS
*
*           NAMISO   DCYPCM   DCYPCM  DCYPCM  DCYPCM  DCYPCM  DCYPCM   DCYPCM
CHDCYPCM001 SR-89    1.0      0.37    0.20    0.0     0.0     0.20     0.0
CHDCYPCM002 SR-90    1.0      0.99    0.99    0.0     0.0     0.99     0.0
```

```
CHDCYPCM003  CS-134      1.0     0.92    0.85    0.0     0.0     0.85     0.0
CHDCYPCM004  CS-137      1.0     0.99    0.99    0.0     0.0     0.99     0.0
CHDCYPCM005  I-131       1.0     0.063   0.032   0.0     0.0     0.032    0.0
CHDCYPCM006  I-133       1.0     0.0068  0.0034  0.0     0.0     0.0034   0.0
*********************************************************************************
*  RETENTION FACTORS FOR NUCLIDES IN CROPS FROM TIME OF HARVEST TO TIME OF
*  CONSUMPTION BY MEAT-PRODUCING ANIMALS (FRACTION RETAINED).  FACTOR REFLECTS
*  LOSSES DUE TO RADIOACTIVE DECAY.
*
*                                                GREEN   OTHER   LEGUMES   ROOTS
*                                STORED          LEAFY   FOOD    AND       AND
*            NUCLIDE   PASTURE   FORAGE  GRAINS  VEG     CROPS   SEEDS     TUBERS
*
*            NAMISO    DCYPCB    DCYPCB  DCYPCB  DCYPCB  DCYPCB  DCYPCB    DCYPCB
CHDCYPCB001  SR-89       1.0     0.37    0.20    0.0     0.0     0.20     0.0
CHDCYPCB002  SR-90       1.0     0.99    0.99    0.0     0.0     0.99     0.0
CHDCYPCB003  CS-134      1.0     0.92    0.85    0.0     0.0     0.85     0.0
CHDCYPCB004  CS-137      1.0     0.99    0.99    0.0     0.0     0.99     0.0
CHDCYPCB005  I-131       1.0     0.063   0.032   0.0     0.0     0.032    0.0
CHDCYPCB006  I-133       1.0     0.0068  0.0034  0.0     0.0     0.0034   0.0
*********************************************************************************
* DEFINE THE DIRECT DEPOSITION TO CROPS TRANSFER FUNCTION
*
* NUMBER OF TERMS IN THE DIRECT DEPOSITION TO CROPS TRANSFER FUNCTION
*
CHNTRTRM001    2
*
* LOSSES DUE TO WEATHERING FROM PLANT SURFACES AND DURING TRANSLOCATION
* FROM PLANT SURFACES TO INTERIOR EDIBLE PORTIONS OF PLANTS ARE MODELLED
* USING THE FOLLOWING EQUATION:
*
* FRACTION RETAINED = CTCOEF1*EXP(-LN2/CTHALF1) + CTCOEF2*EXP(-LN2/CTHALF2)
*
* FOR PASTURE, STORED FORAGE, GREEN LEAFY VEGETABLES, AND OTHER FOOD CROPS,
* THIS EQUATION IS USED AS A TWO TERM WEATHERING EQUATION.  FOR GRAINS,
* LEGUMES AND SEEDS, AND ROOTS AND TUBERS WHERE RADIOACTIVITY IS CONSUMED
* ONLY IF TRANSLOCATED TO EDIBLE PORTIONS OF THE PLANT, THIS EQUATION IS
* REDUCED TO A TRANSLOCATION TRANSFER FACTOR BY SETTING CTCOEF2 TO ZERO,
* CTHALF2 TO ONE SECOND, AND CTHALF1 TO ABOUT ONE MILLION YEARS (1E13
* SECONDS).  WHEN USED TO MODEL TRANSLOCATION, THE VALUE OF THE TRANSLOCATION
* TRANSFER FACTOR IS DEVELOPED FROM FALLOUT DATA AND IS INPUT AS THE VALUE
* OF CTCOEF1.
*
* TWO TIME PERIODS ARE USED FOR WEATHERING, THE FIRST IS 14 DAYS LONG (1.21E6
* SECONDS) AND THE SECOND IS 50 DAYS LONG (4.32E6 SECONDS).
*
* DIRECT DEPOSITION TRANSFER COEFFICIENTS BY CHRONC INGESTION MODEL NUCLIDE
*  ((BQ TRANSFERED)/(BQ DEPOSITED))
*
*                                                GREEN   OTHER   LEGUMES   ROOTS
*                                STORED          LEAFY   FOOD    AND       AND
*  TERM 1    NUCLIDE   PASTURE   FORAGE  GRAINS  VEG     CROPS   SEEDS     TUBERS
CHCTCOEF101  SR-89       0.3       0.2   0.01    0.24      0.2     0.005   0.0006
CHCTCOEF102  SR-90       0.3       0.2   0.01    0.24      0.2     0.005   0.0006
CHCTCOEF103  CS-134      0.3       0.2   0.05    0.24      0.2     0.01    0.025
CHCTCOEF104  CS-137      0.3       0.2   0.05    0.24      0.2     0.01    0.025
CHCTCOEF105  I-131       0.3       0.2   0.0     0.24      0.2     0.0     0.0
```

```
CHCTCOEF106  I-133       0.3        0.2     0.0     0.24       0.2      0.0     0.0
*  TERM 2
CHCTCOEF201  SR-89       0.076      0.05    0.0     0.06       0.05     0.0     0.0
CHCTCOEF202  SR-90       0.076      0.05    0.0     0.06       0.05     0.0     0.0
CHCTCOEF203  CS-134      0.076      0.05    0.0     0.06       0.05     0.0     0.0
CHCTCOEF204  CS-137      0.076      0.05    0.0     0.06       0.05     0.0     0.0
CHCTCOEF205  I-131       0.076      0.05    0.0     0.06       0.05     0.0     0.0
CHCTCOEF206  I-133       0.076      0.05    0.0     0.06       0.05     0.0     0.0
*
*  CROP TRANSFER HALF-LIVES BY CHRONC INGESTION MODEL NUCLIDE (SECONDS)
*
*                                                 GREEN    OTHER    LEGUMES   ROOTS
*                                   STORED        LEAFY    FOOD     AND       AND
*  TERM 1    NUCLIDE   PASTURE     FORAGE   GRAINS  VEG      CROPS    SEEDS     TUBERS
CHCTHALF101  SR-89     1.21E6   1.21E6   1E13   1.21E6   1.21E6    1E13   1E13
CHCTHALF102  SR-90     1.21E6   1.21E6   1E13   1.21E6   1.21E6    1E13   1E13
CHCTHALF103  CS-134    1.21E6   1.21E6   1E13   1.21E6   1.21E6    1E13   1E13
CHCTHALF104  CS-137    1.21E6   1.21E6   1E13   1.21E6   1.21E6    1E13   1E13
CHCTHALF105  I-131     1.21E6   1.21E6      1.0  1.21E6   1.21E6        1.0       1.0
CHCTHALF106  I-133     1.21E6   1.21E6      1.0  1.21E6   1.21E6        1.0       1.0
*  TERM2
CHCTHALF201  SR-89     4.32E6   4.32E6      1.0  4.32E6   4.32E6        1.0       1.0
CHCTHALF202  SR-90     4.32E6   4.32E6      1.0  4.32E6   4.32E6        1.0       1.0
CHCTHALF203  CS-134    4.32E6   4.32E6      1.0  4.32E6   4.32E6        1.0       1.0
CHCTHALF204  CS-137    4.32E6   4.32E6      1.0  4.32E6   4.32E6        1.0       1.0
CHCTHALF205  I-131     4.32E6   4.32E6      1.0  4.32E6   4.32E6        1.0       1.0
CHCTHALF206  I-133     4.32E6   4.32E6      1.0  4.32E6   4.32E6        1.0       1.0
**********************************************************************************
* TABLE OF CROP DATA (GROWING SEASON AND FARMLAND SHARE) IN THE REGION.
*
* IF A SITE DATA FILE IS BEING USED (AS SPECIFIED ON THE EARLY USER INPUT FILE),
* THEN DATA FROM THE SITE FILE (AND NOT THE DATA BELOW) IS USED FOR THE
* CALCULATION OF DOSES AND COSTS FROM THE AGRICULTURE MODEL AND THE NUMBERS
* BELOW ARE IGNORED.
*
* IF A SITE DATA FILE IS NOT BEING USED, THE DATA BELOW IS USED IN ITS STEAD.
*
*        FARMLAND SHARE VALUES (FRCTFL) BY SITE AND CROP CATEGORY
*
*     SITE                   GG       LS       PB       SEQ      SUR      ZION
*
*     PASTURE                0.70     0.47     0.38     0.69     0.41     0.45
*     STORED FORAGE          0.05     0.10     0.13     0.006    0.13     0.11
*     GRAINS                 0.18     0.26     0.23     0.16     0.21     0.26
*     GRN LEAFY VEGETABLES   0.0005   0.0003   0.002    0.0007   0.002    0.0004
*     OTHER FOOD CROPS       0.004    0.001    0.004    0.005    0.004    0.001
*     LEGUMES AND SEEDS      0.13     0.13     0.16     0.15     0.15     0.13
*     ROOTS AND TUBERS       0.0008   0.002    0.004    0.001    0.003    0.002
*
*                                     GROWING
*                                  SEASON  (DAYS)  FARMLAND
*              CROP NAME             START   END      SHARE
*
*              NAMCRP               TGSBEG  TGSEND   FRCTFL
CHCRPRGN001 'PASTURE               '  90.   270.     0.41
CHCRPRGN002 'STORED FORAGE         ' 150.   240.     0.13
```

```
CHCRPRGN003 'GRAINS                 ' 150.  240.   0.21
CHCRPRGN004 'GRN LEAFY VEGETABLES' 150.  240.   0.002
CHCRPRGN005 'OTHER FOOD CROPS    ' 150.  240.   0.004
CHCRPRGN006 'LEGUMES AND SEEDS   ' 150.  240.   0.15
CHCRPRGN007 'ROOTS AND TUBERS    ' 150.  240.   0.003
******************************************************************************
* PROTECTIVE ACTION GUIDES FOR THE DIRECT DEPOSITION PATHWAY TO
*  MILK AND ITS PRODUCTS AND TO OTHER CROPS AND THEIR PRODUCTS
*  BY FOOD INGESTION MODEL NUCLIDE (PERMISSIBLE SURFACE
*  CONCENTRATION IN BECQUERELS PER SQUARE METER)
*
*  PERMISSIBLE SURFACE CONCENTRATIONS WERE DERIVED BY INVERTING
*  THE FOOD PATHWAY MODEL THEREBY MAKING THE DOSE TO AN ORGAN THE
*  INDEPENDENT VARIABLE AND GROUND CONCENTRATION THE DEPENDENT
*  VARIABLE.  PERMISSIBLE GROUND CONCENTRATIONS WERE CALCULATED
*  ASSUMING (1) ALLOWABLE FIRST YEAR (I.E., DIRECT DEPOSITION)
*  ORGAN DOSES OF 15 REM PER YEAR TO THYROID AND 5 REM PER YEAR
*  TO ANY OTHER ORGAN; AND (2) ALLOWABLE DOSES IN SUBSEQUENT YEARS
*  (I.E., ROOT UPTAKE PATH) OF 1.5 REM TO THYROID AND 0.5 REM TO
*  ANY OTHER ORGAN.
*
*                         MILK AND   OTHER CROPS
*            NUCLIDE      PRODUCTS   AND PRODUCTS
*
*             NAMIPI       PSCMLK       PSCOTH
******************************************************************************
* PROTECTIVE ACTION GUIDES FOR LONG-TERM TRANSFER TO FARM CROPS
*  FROM ROOT AND OTHER SOIL UPTAKE FROM SURFACE CONTAMINATION
*  BY CHRONC INGESTION MODEL NUCLIDE (PERMISSIBLE SURFACE
*  CONCENTRATION IN BEQUERELS PER SQUARE METER) AND THE ASSOCIATED
*  ANNUAL DEPLETION RATE FOR THE NUCLIDE IN THE SOIL.
*
*                        PERMISSIBLE      ANNUAL
*                          SURFACE       DEPLETION
*            NUCLIDE    CONCENTRATION      RATE
*
*             NAMIPI       GCMAXR         QROOT
******************************************************************************
* DEFINE THE TYPE 9 RESULTS
*
* LONG-TERM POPULATION DOSE IN A GIVEN REGION BROKEN DOWN BY THE 12 PATHWAYS
*
* NUMBER OF RESULTS OF THIS TYPE THAT ARE BEING REQUESTED
* FOR EACH RESULT YOU REQUEST, THE CODE WILL PRODUCE A SET OF 12
*
TYPE9NUMBER   1
*TYPE9OUT001   'EDEWBODY'  1  26  (0-1000 MILES)
TYPE9OUT001   'EDEWBODY'  1  19  (0-50 MILES)
******************************************************************************
* ECONOMIC COST RESULTS IN A REGION BROKEN DOWN BY 12 TYPES OF COSTS
*
* NUMBER OF RESULTS OF THIS TYPE THAT ARE BEING REQUESTED
* FOR EACH RESULT YOU REQUEST, THE CODE WILL PRODUCE A SET OF 12
*
TYP10NUMBER   0      (UP TO 10 ALLOWED)
*
```

```
*               INNER      OUTER
*
*TYP10OUT001      1          26    (0-1000 MILES)
*TYP10OUT002      1          21    (0-100 MILES)
*TYP10OUT003      1          19    (0-50 MILES)
*TYP10OUT004      1          12    (0-10 MILES)
*******************************************************************************
* DEFINE A FLAG THAT CONTROLS THE PRODUCTION OF THE ACTION DISTANCE RESULTS
*
* SPECIFYING A VALUE OF .TRUE. TURNS ON ALL 8 OF THE ACTION DISTANCE RESULTS,
* A VALUE OF .FALSE. WILL ELIMINATE THE ACTION DISTANCE RESULTS FROM THE OUTPUT.
*
TYP11FLAG11  .FALSE.
*******************************************************************************
* IMPACTED AREA/POPULATION RESULTS IN A REGION BROKEN DOWN BY 6 TYPES OF IMPACTS
*
* NUMBER OF RESULTS OF THIS TYPE THAT ARE BEING REQUESTED
* FOR EACH RESULT YOU REQUEST, THE CODE WILL PRODUCE A SET OF 8
*
TYP12NUMBER   0      (UP TO 10 ALLOWED)
***********************
* copied from in3a.inp for version 1.5.11.1
CHCOUPLD001  .FALSE.
*            NAMIPI      PSCMLK       PSCOTH
CHPAGMCP001  SR-89       2.2E07       2.2E07
CHPAGMCP002  SR-90       2.4E05       2.4E05
CHPAGMCP003  CS-134      2.2E05       2.2E05
CHPAGMCP004  CS-137      2.7E05       2.7E05
CHPAGMCP005  I-131       1.3E06       8.0E06
CHPAGMCP006  I-133       1.1E10       1.0E20
*******************************************************************************
*            NAMIPI      GCMAXR         QROOT
CHPAGLTS001  SR-89       1.8E8          4.9
CHPAGLTS002  SR-90       3.7E4          0.065
CHPAGLTS003  CS-134      4.1E6          0.59
CHPAGLTS004  CS-137      1.8E6          0.28
CHPAGLTS005  I-131       1.E20         32.0
CHPAGLTS006  I-133       1.E20        290.0
*
.
```

D.4 Site Input File

```
Note: NUREG-1150 input
 MACCS SITE DATA FILE FOR SURRY (JLS, 11/10/88)
 SECPOP POP DISTRIBUTION FROM 1980 CENSUS DATA ALTERED USING 0-10 MI NRC DATA
  26 SPATIAL INTERVALS
  16 WIND DIRECTIONS
   7 CROP CATEGORIES
   4 WATER PATHWAY ISOTOPES
   2 WATERSHEDS
  59 ECONOMIC REGIONS
 SPATIAL DISTANCES
     0.16      0.52      1.21      1.61       2.13      3.22      4.02      4.83
     5.63      8.05     11.27     16.09      20.92     25.75     32.19     40.23
    48.28     64.37     80.47    112.65     160.93    241.14    321.87    563.27
   804.67   1609.34
 POPULATION
       0.        0.        0.        0.        0.        0.        4.
5.
       6.       25.     3341.     7107.     2173.        0.     1305.      474.
    2252.     2945.     5403.    20169.   112004.  3431358.  1355700.  2742710.
  2487346.   104331.
       0.        0.        0.        0.        1.        2.        9.       13.
      15.       63.     1667.     3550.     1330.     1072.     3198.     2425.
     515.     9469.     5317.     7120.    13586.   198785.  1058744. 20508438.
  3290082.   830354.
       0.        0.        0.        0.        0.        0.        5.
6.
       8.       31.      822.     1752.     4543.     1713.     1597.     2296.
    6535.     1775.        0.     8555.    48596.   119411.   233382.  3003954.
  7620063.  1169436.
       0.        0.        0.        0.        0.        0.        1.
1.
       2.       11.      543.     1157.     3820.     1621.     3364.
0.
       0.      129.     6679.    11858.        0.        0.        0.
0.
       0.        0.
       0.        0.        0.        0.        0.        0.        0.
0.
       0.        0.     4798.    10202.    10348.    10480.     9570.
0.
       0.     2317.     1756.        0.        0.        0.        0.
0.
       0.        0.
       0.        0.        0.        0.        0.        0.        1.
1.
       1.        7.     8316.    17684.    16340.    30419.    39474.    74998.
   24195.    80412.    57477.        0.        0.        0.        0.
0.
       0.        0.
       0.        0.        0.        0.        0.        0.        0.
0.
       0.        0.        0.        0.     1722.     6433.    36763.    20632.
```

```
   126203.   372471.    68327.     8599.     6339.     1057.        0.
0.
       0.        0.
       0.        0.        0.        0.        0.        0.        2.
2.
       3.       13.      127.      273.     1649.     4571.     3441.     7838.
   11747.    19019.     3360.    36387.    10447.    12402.        0.
0.
       0.        0.
       0.        0.        5.        4.        8.       23.       14.       20.
      23.       93.      301.      650.        0.        0.     1264.     4065.
    1106.    14665.     4071.    18006.    37417.    89072.    81626.
0.
       0.        0.
       0.        0.        0.        0.        0.        0.       19.       25.
      29.      117.       45.      105.        0.      510.      951.     1521.
    1223.    17636.     4926.    30765.    53265.   289674.   216165.   479431.
  280809.  8801784.
       0.        0.        0.        0.        1.        2.       14.       20.
      23.       93.      155.      338.      125.     1079.        0.     1355.
    2765.      154.     5296.    21409.    62228.   523803.   479588.  1538059.
 1526840.  3099458.
       0.        0.        0.        0.        1.        2.       14.       20.
      23.       93.      110.      240.     1056.        0.       50.     1396.
     915.     3153.     4132.    16295.    35596.   239712.   709522.  2845970.
 3957581. 10560254.
       0.        0.        0.        0.        0.        0.       25.       33.
      38.      154.       30.       70.      450.        0.      980.      517.
     155.    66531.    40902.     9557.    44818.   194801.   376828.  1492286.
 2250273. 12145932.
       0.        0.        0.        0.        0.        0.        7.
9.
      12.       47.       31.       69.        0.      380.      281.      445.
    1986.    32459.   183133.   193630.    30369.   203275.    94113.  1328987.
 5086913. 19537940.
       0.        0.        0.        0.        0.        0.        0.
0.
       0.        0.      223.      477.        0.     1026.      609.     2575.
    2794.     6593.    96857.   107328.    47585.   156826.   101785.  4175263
 7535605.  9667977.
       0.        0.        0.        0.        0.        0.       15.       20.
      23.       92.     2503.     5326.     3508.     1826.     1884.      275.
    3965.     2084.     6270.    10765.   103787.   970659.   472558.  1396088.
 1969210.    73968.
 LAND FRACTION
 1.00 1.00 0.00 0.00 0.00 0.00 0.00 0.00 0.00 0.80 1.00 1.00 0.95 0.75 0.70 0.85
 1.00 0.85 0.70 0.75 0.55 0.70 0.60 1.00 1.00 0.95
 1.00 1.00 1.00 1.00 0.90 0.70 0.40 0.00 0.00 0.45 1.00 0.95 0.40 0.60 1.00 1.00
 0.90 0.45 0.60 0.20 0.50 0.50 0.30 0.25 0.50 0.60
 1.00 1.00 1.00 1.00 1.00 1.00 0.00 0.00 0.00 0.05 1.00 1.00 0.20 1.00 0.70 0.30
 0.85 0.20 0.00 0.20 0.00 0.00 0.00 0.00 0.00 0.00
 1.00 1.00 1.00 1.00 1.00 0.80 0.10 0.00 0.00 0.00 1.00 1.00 0.75 0.30 0.40 0.00
 0.15 0.00 0.45 0.30 0.00 0.00 0.00 0.00 0.00 0.00
 1.00 1.00 1.00 1.00 1.00 0.60 0.00 0.00 0.00 0.00 0.95 1.00 1.00 0.70 0.40 0.10
 0.00 0.50 0.00 0.00 0.00 0.00 0.00 0.00 0.00 0.00
 1.00 1.00 1.00 1.00 1.00 0.70 0.00 0.00 0.00 0.80 0.90 0.75 0.85 1.00 1.00 0.70
```

```
 0.15 0.25 0.25 0.00 0.00 0.00 0.00 0.00 0.00 0.00
 1.00 1.00 1.00 1.00 1.00 1.00 0.55 0.50 0.25 0.15 0.00 0.10 0.00 0.10 0.50 0.60
 0.85 1.00 1.00 0.40 0.40 0.05 0.00 0.00 0.00 0.00
 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 0.90 0.35 0.40 0.80 1.00 1.00 1.00
 1.00 1.00 1.00 1.00 0.95 0.20 0.00 0.00 0.00 0.00
 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00
 1.00 1.00 1.00 1.00 1.00 0.80 0.40 0.00 0.00 0.20
 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00
 1.00 1.00 1.00 1.00 1.00 1.00 1.00 0.75 0.50 0.40
 1.00 1.00 0.70 0.40 0.20 0.75 0.95 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00
 1.00 1.00 1.00 1.00 0.95 1.00 1.00 1.00 1.00 0.80
 1.00 1.00 0.00 0.00 0.00 0.00 0.05 0.70 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00
 1.00 1.00 1.00 1.00 0.95 0.90 1.00 1.00 1.00 1.00
 1.00 1.00 0.00 0.00 0.00 0.00 0.00 0.15 0.70 0.90 1.00 1.00 1.00 1.00 1.00 1.00
 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00
 1.00 1.00 0.00 0.00 0.00 0.00 0.00 0.00 0.20 0.50 0.40 0.30 0.25 0.75 0.80 0.85
 0.85 0.95 1.00 1.00 1.00 1.00 1.00 1.00 0.90 0.85
 1.00 1.00 0.00 0.00 0.00 0.00 0.00 0.15 0.80 0.75 0.80 0.85 0.95 1.00 1.00 1.00
 1.00 1.00 1.00 1.00 1.00 1.00 1.00 1.00 0.50 0.45
 1.00 1.00 0.00 0.00 0.00 0.00 0.00 0.00 0.00 0.90 1.00 1.00 1.00 1.00 1.00 1.00
 1.00 1.00 1.00 0.95 0.80 0.85 1.00 1.00 0.55 0.95
 REGION INDEX
4444505050505050504444444444444444444444441818 7283054
444444444444445050444444444444444444444418 728301917
4444444444445050504444444444444444445044185050505050
4444444444444450505044444444445044504444505050505050
4444444444445050505044444444444450445050505050505050
4444444444445050504444444444444444444450505050505050
4444444444444444444450445044444444444444313150505050
4444444444444444444444444444444444444431313150505050
444444444444444444444444444444444444313131313150 8
444444444444444444444444444444444444443131313138 9 8
44444444444444444444444444444444444444443131313 8 9 1
4444505050504444444444444444444444444444444444314040
4444505050505044444444444444444444444444444444461511
4444505050505050444444444444444444444444444444463347
4444505050505044444444444444444444444444444444365320
4444505050505050504444444444444444444444441818363053
 WATERSHED INDEX
 1 1 2 2 2 2 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
 1 1 1 1 1 1 2 2 2 2 1 1 2 1 1 1 1 1 1 2 2 2 2 2 2 1
 1 1 1 1 1 1 2 2 2 2 1 1 2 1 1 2 1 2 2 2 2 2 2 2 2 2
 1 1 1 1 1 1 2 2 2 2 1 1 1 2 2 2 2 2 2 2 2 2 2 2 2 2
 1 1 1 1 1 1 2 2 2 2 1 1 1 1 2 2 2 2 2 2 2 2 2 2 2 2
 1 1 1 1 1 1 2 2 2 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 2 2
 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 1 1 1 1 2 2 2 2 2 2 2
 1 1 1 1 1 1 1 1 1 1 2 2 1 1 1 1 1 1 1 1 1 2 2 2 2 2
 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2
 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 2 2
 1 1 1 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
 1 1 2 2 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
 1 1 2 2 2 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
 1 1 2 2 2 2 2 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
 1 1 2 2 2 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
 1 1 2 2 2 2 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
 CROP SEASON AND SHARE
```

```
  1 PASTURE                 90. 270.   0.41
  2 STORED FORAGE          150. 240.   0.13
  3 GRAINS                 150. 240.   0.21
  4 GRN LEAFY VEGETABLES   150. 240.   0.002
  5 OTHER FOOD CROPS       150. 240.   0.004
  6 LEGUMES AND SEEDS      150. 240.   0.15
  7 ROOTS AND TUBERS       150. 240.   0.003
WATERSHED DEFINITION  --  INITIAL AND ANNUAL WASHOFF AND INGESTION FACTORS
  1 SR-89                              5.0E-6    0.0
  2 SR-90                              5.0E-6    0.0
  3 CS-134                             5.0E-6    0.0
  4 CS-137                             5.0E-6    0.0
REGIONAL ECONOMIC DATA
  1   AL            .308 .027      676.       2302.     93617.
  2   AZ            .495 .127      128.        879.     99909.
  3   AR            .480 .026      657.       2082.     88486.
  4   CA            .311 .139     1502.       4438.    129140.
  5   CO            .497 .047      315.       1114.    117723.
  6   CT            .129 .173     2754.      11140.    158515.
  7   DE            .480 .031     2651.       5809.    125432.
  8   FL            .318 .070     1281.       5380.    115720.
  9   GA            .351 .060      730.       2729.    106394.
 10   ID            .264 .128      517.       1862.     95190.
 11   IL            .815 .045      676.       3660.    127604.
 12   IN            .697 .062      761.       3302.    105620.
 13   IA            .951 .054      749.       2951.    107992.
 14   KS            .917 .022      360.       1371.    113004.
 15   KY            .551 .095      546.       2653.     93579.
 16   LA            .323 .072      527.       2490.     90683.
 17   ME            .071 .207      811.       2746.    107255.
 18   MD            .352 .150     1510.       6207.    136430.
 19   MA            .140 .181     1474.       9524.    140787.
 20   MI            .303 .226      714.       2712.    114715.
 21   MN            .589 .181      577.       2218.    116918.
 22   MS            .433 .044      462.       2028.     80084.
 23   MO            .680 .098      324.       1907.    109103.
 24   MT            .655 .027       65.        817.     95527.
 25   NE            .955 .018      464.       1588.    109172.
 26   NV            .128 .131       91.        709.    118903.
 27   NH            .087 .349      662.       5755.    129664.
 28   NJ            .168 .079     1997.      11676.    155306.
 29   NM            .579 .177       83.        714.     88973.
 30   NY            .265 .518      928.       2635.    138128.
 31   NC            .321 .045     1202.       3349.    101532.
 32   ND            .929 .048      152.       1069.     95845.
 33   OH            .610 .151      644.       3203.    109659.
 34   OK            .751 .047      266.       1457.     96444.
 35   OR            .293 .099      317.       1640.    107249.
 36   PA            .279 .398     1163.       4693.    116593.
 37   RI            .150 .066     1753.      12649.    117405.
 38   SC            .259 .051      580.       2475.     94509.
 39   SD            .906 .063      188.       1040.     99185.
 40   TN            .455 .150      419.       2690.     99047.
 41   TX            .787 .064      224.       1452.    104347.
 42   UT            .209 .229      169.       1190.     87294.
 43   VT            .237 .777      788.       3169.    109272.
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44	VA	.355	.139	581.	3974.	122973.
45	WA	.375	.170	589.	2154.	117205.
46	WV	.259	.110	208.	1744.	85789.
47	WI	.518	.556	783.	2213.	109796.
48	WY	.563	.017	54.	598.	101638.
49	BRIT COL	.377	.154	476.	1948.	60000.
50	OCEAN	.0	.0	0.	0.	0.
51	SASKAT	.657	.030	61.	563.	60000.
52	MANITOBA	.924	.048	164.	948.	60000.
53	ONTARIO	.597	.223	516.	2111.	60000.
54	QUEBEC	.310	.589	711.	1378.	60000.
55	NOVA SCOT	.079	.260	662.	1133.	60000.
56	BAJA CAL	.330	.144	1022.	4394.	10000.
57	SONORA	.516	.104	110.	682.	10000.
58	CHIHUAHUA	.590	.144	53.	473.	10000.
59	COAHUILA	.816	.064	164.	1492.	10000.

END

APPENDIX E

SUPPORTING INFORMATION FOR THE

THE RISK EVALUATION

Appendix E Risk Evaluation

Figure E1 and Table E1 below provide details of the risk evaluation data presented for 360 APBs analyzed in the current study. For convenience, two arrangements of APBs are shown: alphabetical and by magnitude of the mean risk of latent fatalities (LF). Figure E2 shows the first ... dominant risk contributing APBs which collectively represent about 97% of total risk of latent fatalities.

Table E1: Risk of Latent Fatalities by APBs

APB ID	APB #	Mean Risk of LF (1/yr)	APB ID	APB #	Mean Risk of LF (1/yr)
(In Alphabetical Order)			(By Magnitude of LF Risk)		
CDCCACDBBBAA	1	3.55E-08	CHADBCABDBAB	150	5.07E-04
CDCCACDBBCAA	2	3.90E-08	CHADBCABDBAC	151	3.55E-04
CDCCBCDBDBAA	3	2.53E-09	CHADBCABDBAA	149	2.00E-04
CDCCBCDBDCAA	4	5.56E-09	CGADBCABDBAB	102	1.92E-04
CDCCCCDBBBAA	5	1.41E-09	CDCDFCDBDBAB	24	1.54E-04
CDCCCCDBBCAA	6	3.40E-09	CGADBCABDBAC	103	1.30E-04
CDCCFCDBDBAA	7	2.88E-06	CFADBCABDBAB	56	1.29E-04
CDCCFCDBDCAA	8	1.50E-08	CHCDFCDBDBAB	168	1.27E-04
CDCDBCDBDBAA	9	2.77E-06	CDCDFCDBDBAC	25	1.17E-04
CDCDBCDBDBAB	10	6.03E-06	CHCDFCDBDBAC	169	9.39E-05
CDCDBCDBDBAC	11	4.74E-06	CFADBCABDBAC	57	7.89E-05
CDCDBCDBDBAD	12	3.25E-07	CGADBCABDBAA	101	7.37E-05
CDCDBCDBDCAA	13	2.10E-06	CDCDFCDBDBAA	23	7.07E-05
CDCDBCDBDCAB	14	6.69E-07	CFADBCABDBAA	55	6.34E-05
CDCDBCDBDCAC	15	1.51E-09	CHCDFCDBDBAA	167	6.10E-05
CDCDBCDBDCAD	16	4.77E-10	CHADBCABDBAD	152	5.95E-05
CDCDDCDBDBAA	17	1.49E-10	CGADBCABDBAD	104	2.11E-05
CDCDDCDBDBAB	18	7.61E-09	CDCDFCDBDBAD	26	2.02E-05
CDCDDCDBDBAC	19	5.23E-09	CHEDBCABDCAA	197	1.59E-05
CDCDDCDBDBAD	20	5.75E-13	CHCDFCDBDBAD	170	1.12E-05
CDCDDCDBDCAA	21	1.80E-09	CFADBCABDBAD	58	1.04E-05
CDCDDCDBDCAB	22	1.15E-10	CHDDBCABDCAA	183	9.16E-06
CDCDFCDBDBAA	23	7.07E-05	CDCDBCDBDBAB	10	6.03E-06
CDCDFCDBDBAB	24	1.54E-04	EGADBCABDAAA	275	5.04E-06
CDCDFCDBDBAC	25	1.17E-04	CDCDBCDBDBAC	11	4.74E-06
CDCDFCDBDBAD	26	2.02E-05	CDDDBCABDBAB	38	4.70E-06
CDCDFCDBDCAA	27	6.00E-07	CHEDBCABDBAA	195	4.26E-06
CDCDFCDBDCAB	28	5.23E-07	CHADBCABDCAB	154	4.17E-06
CDCDFCDBDCAC	29	1.51E-07	CDDDBCABDBAC	39	3.63E-06
CDCDFCDBDCAD	30	1.12E-07	CHADBCABDCAA	153	3.61E-06
CDDCACBBBBAA	31	1.41E-09	CHACACBBBBAA	143	3.49E-06

CDDCACBBBCAA	32	9.00E-09	CDCCFCDBDBAA	7	2.88E-06
CDDCBCABDBAA	33	1.48E-09	CDCDBCDBDBAA	9	2.77E-06
CDDCBCABDCAA	34	6.71E-09	CHCCFCDBDBAA	165	2.56E-06
CDDCCCBBBBAA	35	1.16E-11	CHCDFCDBDCAA	171	2.40E-06
CDDCCCBBBCAA	36	1.00E-09	CHCDFCDBDCAB	172	2.37E-06
CDDDBCABDBAA	37	2.17E-06	EGADBCABDAAB	276	2.28E-06
CDDDBCABDBAB	38	4.70E-06	CHEDBCABDCAB	198	2.26E-06
CDDDBCABDBAC	39	3.63E-06	DHECACBBBAAA	251	2.26E-06
CDDDBCABDBAD	40	2.20E-07	CDDDBCABDCAA	41	2.24E-06
CDDDBCABDCAA	41	2.24E-06	CHDDBCABDBAA	181	2.23E-06
CDDDBCABDCAB	42	7.16E-07	CDDDBCABDBAA	37	2.17E-06
CDDDBCABDCAC	43	1.17E-09	CDCDBCDBDCAA	13	2.10E-06
CDDDBCABDCAD	44	4.82E-10	CHEDBCABDBAB	196	1.92E-06
CDDDDCBBDBAB	45	6.81E-10	CHADBCABDCAD	156	1.53E-06
CDDDDCBBDBAC	46	4.64E-10	CHADBCABDCAC	155	1.51E-06
CDDDDCBBDCAA	47	1.16E-10	CGCDBCDBDBAA	122	1.45E-06
CDDDDCBBDCAB	48	9.55E-12	CFADBCABDCAA	59	1.43E-06
CFACACBBBBAA	49	7.96E-07	CGADBCABDCAB	106	1.33E-06
CFACACBBBCAA	50	8.52E-09	CHDDBCABDCAB	184	1.27E-06
CFACBCABDBAA	51	2.13E-07	CGADBCABDCAA	105	1.25E-06
CFACBCABDCAA	52	4.46E-09	CGDDBCABDBAA	135	1.09E-06
CFACCCBBBBAA	53	4.53E-08	EGDDBCABDAAA	282	1.06E-06
CFACCCBBBCAA	54	1.43E-10	CHADDCBBDBAB	158	1.05E-06
CFADBCABDBAA	55	6.34E-05	CFADBCABDCAB	60	1.03E-06
CFADBCABDBAB	56	1.29E-04	EFADBCABDAAA	258	1.02E-06
CFADBCABDBAC	57	7.89E-05	CHDDBCABDBAB	182	9.98E-07
CFADBCABDBAD	58	1.04E-05	CGCDBCDBDBAB	123	9.77E-07
CFADBCABDCAA	59	1.43E-06	DDCCACDBBAAA	203	9.35E-07
CFADBCABDCAB	60	1.03E-06	CGACACBBBBAA	95	9.06E-07
CFADBCABDCAC	61	3.45E-07	CFACACBBBBAA	49	7.96E-07
CFADBCABDCAD	62	1.07E-07	CFCDBCDBDBAA	76	7.32E-07
CFADDCBBDBAA	63	2.50E-08	CDDDBCABDCAB	42	7.16E-07
CFADDCBBDBAB	64	1.73E-07	CGDDBCABDBAB	136	7.13E-07
CFADDCBBDBAC	65	7.96E-08	EGADBCABDAAC	277	7.13E-07
CFADDCBBDBAD	66	1.40E-10	CDCDBCDBDCAB	14	6.69E-07
CFADDCBBDCAA	67	4.06E-10	EGDDBCABDAAB	283	6.29E-07
CFADDCBBDCAB	68	1.59E-10	CDCDFCDBDCAA	27	6.00E-07
CFADDCBBDCAC	69	2.56E-12	DHEDDCBBDAAA	253	5.75E-07
CFCCACDBBBAA	70	6.02E-09	CFDDBCABDBAA	90	5.72E-07
CFCCACDBBCAA	71	1.02E-09	CHCDFCDBDCAC	173	5.50E-07
CFCCBCDBDBAA	72	1.86E-09	EHADBCABDAAB	288	5.42E-07
CFCCBCDBDCAA	73	3.06E-10	CHADDCBBDBAC	159	5.40E-07
CFCCCCDBBBAA	74	1.01E-09	CHCDFCDBDCAD	174	5.29E-07
CFCCCCDBBCAA	75	1.16E-10	EHADBCABDAAA	287	5.28E-07
CFCDBCDBDBAA	76	7.32E-07	CDCDFCDBDCAB	28	5.23E-07
CFCDBCDBDBAB	77	5.09E-07	CFCDBCDBDBAB	77	5.09E-07
CFCDBCDBDCAA	78	1.54E-07	DHDCACBBBAAA	247	4.95E-07

CFCDBCDBDCAB	79	6.47E-08	CHACBCABDBAA	145	4.91E-07
CFCDDCDBDBAA	80	3.43E-10	CGADBCABDCAC	107	4.38E-07
CFCDDCDBDBAB	81	9.35E-11	EFADBCABDAAB	259	4.24E-07
CFCDDCDBDCAA	82	9.54E-12	CFDDBCABDBAB	91	3.94E-07
CFCDDCDBDCAB	83	6.51E-13	CFADBCABDCAC	61	3.45E-07
CFDCACBBBBAA	84	7.41E-10	EHEDBCABDAAB	300	3.30E-07
CFDCACBBBCAA	85	2.09E-11	CGCDBCDBDCAA	124	3.27E-07
CFDCBCABDBAA	86	1.56E-09	CDCDBCDBDBAD	12	3.25E-07
CFDCBCABDCAA	87	1.57E-10	EHEDBCABDAAA	299	3.25E-07
CFDCCCBBBBAA	88	1.60E-10	CGADDCBBDBAB	110	3.23E-07
CFDCCCBBBCAA	89	4.57E-12	CHECACBBBCAA	190	3.17E-07
CFDDBCABDBAA	90	5.72E-07	DHACACBBBAAA	241	3.17E-07
CFDDBCABDBAB	91	3.94E-07	CGADBCABDCAD	108	2.82E-07
CFDDBCABDCAA	92	1.14E-07	EGADBCABDAAD	278	2.56E-07
CFDDBCABDCAB	93	4.65E-08	CGACBCABDBAA	97	2.55E-07
CFDDDCBBDBAB	94	8.56E-12	DDDCACBBBAAA	209	2.36E-07
CGACACBBBBAA	95	9.06E-07	CGDDBCABDCAA	137	2.28E-07
CGACACBBBCAA	96	3.91E-09	CDDDBCABDBAD	40	2.20E-07
CGACBCABDBAA	97	2.55E-07	CGCDBCDBDCAB	125	2.16E-07
CGACBCABDCAA	98	3.29E-09	CFACBCABDBAA	51	2.13E-07
CGACCCBBBBAA	99	5.29E-08	CGADDCBBDBAC	111	2.02E-07
CGACCCBBBCAA	100	7.89E-11	EHADBCABDAAC	289	1.77E-07
CGADBCABDBAA	101	7.37E-05	CFADDCBBDBAB	64	1.73E-07
CGADBCABDBAB	102	1.92E-04	CFCDBCDBDCAA	78	1.54E-07
CGADBCABDBAC	103	1.30E-04	CDCDFCDBDCAC	29	1.51E-07
CGADBCABDBAD	104	2.11E-05	CGDDBCABDCAB	138	1.49E-07
CGADBCABDCAA	105	1.25E-06	EHDDBCABDAAA	294	1.38E-07
CGADBCABDCAB	106	1.33E-06	EHDDBCABDAAB	295	1.37E-07
CGADBCABDCAC	107	4.38E-07	EFADBCABDAAC	260	1.37E-07
CGADBCABDCAD	108	2.82E-07	EHADBCABDAAD	290	1.19E-07
CGADDCBBDBAA	109	2.79E-08	CFDDBCABDCAA	92	1.14E-07
CGADDCBBDBAB	110	3.23E-07	CDCDFCDBDCAD	30	1.12E-07
CGADDCBBDBAC	111	2.02E-07	CFADBCABDCAD	62	1.07E-07
CGADDCBBDBAD	112	2.85E-10	CHADDCBBDBAA	157	1.05E-07
CGADDCBBDCAA	113	1.49E-10	CHACCCBBBBAA	147	9.61E-08
CGADDCBBDCAB	114	1.00E-10	DFACACBBBAAA	213	8.59E-08
CGADDCBBDCAC	115	1.17E-12	EFDDBCABDAAA	270	8.02E-08
CGCCACDBBBAA	116	9.53E-09	CFADDCBBDBAC	65	7.96E-08
CGCCACDBBCAA	117	1.54E-09	DHECCCBBBAAA	252	7.89E-08
CGCCBCDBDBAA	118	4.61E-09	DGACACBBBAAA	227	6.99E-08
CGCCBCDBDCAA	119	1.12E-09	CHCCFCDBDCAA	166	6.96E-08
CGCCCCDBBBAA	120	3.93E-10	CFCDBCDBDCAB	79	6.47E-08
CGCCCCDBBCAA	121	2.56E-11	DDCDDCDBDAAA	205	6.01E-08
CGCDBCDBDBAA	122	1.45E-06	CHDCACBBBCAA	176	5.76E-08
CGCDBCDBDBAB	123	9.77E-07	CGACCCBBBBAA	99	5.29E-08
CGCDBCDBDCAA	124	3.27E-07	EFCDBCDBDAAA	265	5.26E-08
CGCDBCDBDCAB	125	2.16E-07	DHEDDCBBDAAB	254	4.73E-08

CGCDDCDBDBAA	126	3.62E-10	CFDDBCABDCAB	93	4.65E-08
CGCDDCDBDBAB	127	3.61E-10	CFACCCBBBBAA	53	4.53E-08
CGCDDCDBDCAA	128	2.58E-12	CHECBCABDCAA	192	4.39E-08
CGCDDCDBDCAB	129	6.73E-12	DHDDDCBBDAAA	249	3.96E-08
CGDCACBBBBAA	130	1.18E-09	CDCCACDBBCAA	2	3.90E-08
CGDCACBBBCAA	131	1.63E-10	CDCCACDBBBAA	1	3.55E-08
CGDCBCABDBAA	132	3.76E-09	DHADDCBBDAAB	244	3.12E-08
CGDCBCABDCAA	133	7.34E-10	EFDDBCABDAAB	271	3.10E-08
CGDCCCBBBBAA	134	2.08E-11	DDCCCCDBBAAA	204	2.92E-08
CGDDBCABDBAA	135	1.09E-06	DGCCACDBBAAA	233	2.80E-08
CGDDBCABDBAB	136	7.13E-07	CGADDCBBDBAA	109	2.79E-08
CGDDBCABDCAA	137	2.28E-07	CHDCBCABDCAA	178	2.50E-08
CGDDBCABDCAB	138	1.49E-07	CFADDCBBDBAA	63	2.50E-08
CGDDDCBBDBAA	139	1.36E-11	GHEDBCABDDAA	359	2.41E-08
CGDDDCBBDBAB	140	4.33E-11	CHECACBBBBAA	189	2.37E-08
CGDDDCBBDCAA	141	2.06E-13	DFCCACDBBAAA	219	2.35E-08
CGDDDCBBDCAB	142	2.69E-13	EGACBCABDAAA	273	2.26E-08
CHACACBBBBAA	143	3.49E-06	CHADDCBBDBAD	160	2.17E-08
CHACACBBBCAA	144	1.52E-08	EFCDBCDBDAAB	266	2.07E-08
CHACBCABDBAA	145	4.91E-07	EFADBCABDAAD	261	1.99E-08
CHACBCABDCAA	146	1.31E-08	CHECCCBBBCAA	194	1.95E-08
CHACCCBBBBAA	147	9.61E-08	GDCDBCDBDDAA	305	1.82E-08
CHACCCBBBCAA	148	5.35E-10	GHDDBCABDDAA	354	1.78E-08
CHADBCABDBAA	149	2.00E-04	DHDCCCBBBAAA	248	1.74E-08
CHADBCABDBAB	150	5.07E-04	CHACACBBBCAA	144	1.52E-08
CHADBCABDBAC	151	3.55E-04	CHEDDCBBDCAA	201	1.52E-08
CHADBCABDBAD	152	5.95E-05	CDCCFCDBDCAA	8	1.50E-08
CHADBCABDCAA	153	3.61E-06	GDDDBCABDDAA	316	1.35E-08
CHADBCABDCAB	154	4.17E-06	CHACBCABDCAA	146	1.31E-08
CHADBCABDCAC	155	1.51E-06	CHECBCABDBAA	191	1.24E-08
CHADBCABDCAD	156	1.53E-06	DHADDCBBDAAA	243	1.20E-08
CHADDCBBDBAA	157	1.05E-07	DHACCCBBBAAA	242	1.11E-08
CHADDCBBDBAB	158	1.05E-06	DDDDDCBBDAAA	211	1.02E-08
CHADDCBBDBAC	159	5.40E-07	CGCCACDBBBAA	116	9.53E-09
CHADDCBBDBAD	160	2.17E-08	CDDCACBBBCAA	32	9.00E-09
CHADDCBBDCAA	161	4.24E-10	DFADDCBBDAAA	215	8.58E-09
CHADDCBBDCAB	162	1.13E-09	CFACACBBBCAA	50	8.52E-09
CHADDCBBDCAC	163	2.75E-10	DGADDCBBDAAA	229	8.06E-09
CHADDCBBDCAD	164	7.84E-11	DHADDCBBDAAC	245	7.71E-09
CHCCFCDBDBAA	165	2.56E-06	CDCDDCDBDBAB	18	7.61E-09
CHCCFCDBDCAA	166	6.96E-08	GDCDFCDBDDAA	309	7.46E-09
CHCDFCDBDBAA	167	6.10E-05	DDDCCCBBBAAA	210	7.31E-09
CHCDFCDBDBAB	168	1.27E-04	GDCDFCDBDDAB	310	6.94E-09
CHCDFCDBDBAC	169	9.39E-05	CDDCBCABDCAA	34	6.71E-09
CHCDFCDBDBAD	170	1.12E-05	CHEDDCBBDBAA	199	6.61E-09
CHCDFCDBDCAA	171	2.40E-06	EGACACBBBAAA	272	6.53E-09
CHCDFCDBDCAB	172	2.37E-06	GHCDFCDBDDAA	347	6.24E-09

CHCDFCDBDCAC	173	5.50E-07	GHCDFCDBDDAB	348	6.04E-09
CHCDFCDBDCAD	174	5.29E-07	CFCCACDBBBAA	70	6.02E-09
CHDCACBBBBAA	175	3.20E-09	CHDCBCABDBAA	177	5.58E-09
CHDCACBBBCAA	176	5.76E-08	GHADBCABDDAB	343	5.57E-09
CHDCBCABDBAA	177	5.58E-09	CDCCBCDBDCAA	4	5.56E-09
CHDCBCABDCAA	178	2.50E-08	DFACCCBBBAAA	214	5.45E-09
CHDCCCBBBBAA	179	2.76E-10	DGADDCBBDAAB	230	5.43E-09
CHDCCCBBBCAA	180	3.91E-09	DHADDCBBDAAD	246	5.23E-09
CHDDBCABDBAA	181	2.23E-06	CDCDDCDBDBAC	19	5.23E-09
CHDDBCABDBAB	182	9.98E-07	DGDCACBBBAAA	237	5.17E-09
CHDDBCABDCAA	183	9.16E-06	GDCDBCDBDDAB	306	5.15E-09
CHDDBCABDCAB	184	1.27E-06	EFACBCABDAAA	256	4.77E-09
CHDDDCBBDBAA	185	1.03E-10	DDCDDCDBDAAB	206	4.72E-09
CHDDDCBBDBAB	186	5.82E-11	CGCCBCDBDBAA	118	4.61E-09
CHDDDCBBDCAA	187	6.94E-10	DFADDCBBDAAB	216	4.57E-09
CHDDDCBBDCAB	188	5.08E-11	CFACBCABDCAA	52	4.46E-09
CHECACBBBBAA	189	2.37E-08	GHADBCABDDAA	342	4.41E-09
CHECACBBBCAA	190	3.17E-07	DFCCCCDBBAAA	220	4.40E-09
CHECBCABDBAA	191	1.24E-08	DFDCACBBBAAA	223	4.05E-09
CHECBCABDCAA	192	4.39E-08	DGACCCBBBAAA	228	4.03E-09
CHECCCBBBBAA	193	1.63E-09	CHDCCCBBBCAA	180	3.91E-09
CHECCCBBBCAA	194	1.95E-08	CGACACBBBCAA	96	3.91E-09
CHEDBCABDBAA	195	4.26E-06	GDDDBCABDDAB	317	3.81E-09
CHEDBCABDBAB	196	1.92E-06	CGDCBCABDBAA	132	3.76E-09
CHEDBCABDCAA	197	1.59E-05	CDCCCCDBBCAA	6	3.40E-09
CHEDBCABDCAB	198	2.26E-06	DGCCCCDBBAAA	234	3.29E-09
CHEDDCBBDBAA	199	6.61E-09	CGACBCABDCAA	98	3.29E-09
CHEDDCBBDBAB	200	1.78E-09	EGDCBCABDAAA	280	3.27E-09
CHEDDCBBDCAA	201	1.52E-08	CHDCACBBBBAA	175	3.20E-09
CHEDDCBBDCAB	202	7.49E-10	GHEDBCABDDAB	360	2.69E-09
DDCCACDBBAAA	203	9.35E-07	GHADBCABDDAD	345	2.62E-09
DDCCCCDBBAAA	204	2.92E-08	CDCCBCDBDBAA	3	2.53E-09
DDCDDCDBDAAA	205	6.01E-08	GHADBCABDDAC	344	2.31E-09
DDCDDCDBDAAB	206	4.72E-09	DHDDDCBBDAAB	250	2.19E-09
DDCDDCDBDAAC	207	1.58E-12	GDCDFCDBDDAC	311	2.11E-09
DDCDDCDBDAAD	208	2.12E-12	GHDDBCABDDAB	355	1.99E-09
DDDCACBBBAAA	209	2.36E-07	CFCCBCDBDBAA	72	1.86E-09
DDDCCCBBBAAA	210	7.31E-09	GDCDFCDBDDAD	312	1.85E-09
DDDDDCBBDAAA	211	1.02E-08	CDCDDCDBDCAA	21	1.80E-09
DDDDDCBBDAAB	212	4.93E-10	CHEDDCBBDBAB	200	1.78E-09
DFACACBBBAAA	213	8.59E-08	GHCDFCDBDDAC	349	1.69E-09
DFACCCBBBAAA	214	5.45E-09	GHCDFCDBDDAD	350	1.64E-09
DFADDCBBDAAA	215	8.58E-09	CHECCCBBBBAA	193	1.63E-09
DFADDCBBDAAB	216	4.57E-09	DFCDDCDBDAAA	221	1.60E-09
DFADDCBBDAAC	217	4.79E-10	CFDCBCABDBAA	86	1.56E-09
DFADDCBBDAAD	218	5.58E-11	CGCCACDBBCAA	117	1.54E-09
DFCCACDBBAAA	219	2.35E-08	CDCDBCDBDCAC	15	1.51E-09

DFCCCCDBBAAA	220	4.40E-09	CDDCBCABDBAA	33	1.48E-09
DFCDDCDBDAAA	221	1.60E-09	CDCCCCDBBBAA	5	1.41E-09
DFCDDCDBDAAB	222	2.94E-10	CDDCACBBBBAA	31	1.41E-09
DFDCACBBBAAA	223	4.05E-09	CGDCACBBBBAA	130	1.18E-09
DFDCCCBBBAAA	224	9.51E-10	EHACBCABDAAA	285	1.18E-09
DFDDDCBBDAAA	225	7.27E-11	CDDDBCABDCAC	43	1.17E-09
DFDDDCBBDAAB	226	5.22E-13	DGCDDCDBDAAA	235	1.16E-09
DGACACBBBAAA	227	6.99E-08	CHADDCBBDCAB	162	1.13E-09
DGACCCBBBAAA	228	4.03E-09	CGCCBCDBDCAA	119	1.12E-09
DGADDCBBDAAA	229	8.06E-09	EHECBCABDAAA	297	1.02E-09
DGADDCBBDAAB	230	5.43E-09	CFCCACDBBCAA	71	1.02E-09
DGADDCBBDAAC	231	3.45E-10	CFCCCCDBBBAA	74	1.01E-09
DGADDCBBDAAD	232	1.67E-10	CDDCCCBBBCAA	36	1.00E-09
DGCCACDBBAAA	233	2.80E-08	GGADBCABDDAB	329	9.95E-10
DGCCCCDBBAAA	234	3.29E-09	DFDCCCBBBAAA	224	9.51E-10
DGCDDCDBDAAA	235	1.16E-09	EFACACBBBAAA	255	8.95E-10
DGCDDCDBDAAB	236	4.07E-10	GGCDBCDBDDAA	335	7.97E-10
DGDCACBBBAAA	237	5.17E-09	CHEDDCBBDCAB	202	7.49E-10
DGDCCCBBBAAA	238	1.69E-10	CFDCACBBBBAA	84	7.41E-10
DGDDDCBBDAAA	239	2.12E-11	CGDCBCABDCAA	133	7.34E-10
DGDDDCBBDAAB	240	1.07E-11	CHDDDCBBDCAA	187	6.94E-10
DHACACBBBAAA	241	3.17E-07	CDDDDCBBDBAB	45	6.81E-10
DHACCCBBBAAA	242	1.11E-08	GFADBCABDDAB	321	6.35E-10
DHADDCBBDAAA	243	1.20E-08	GGADBCABDDAD	331	5.39E-10
DHADDCBBDAAB	244	3.12E-08	CHACCCBBBCAA	148	5.35E-10
DHADDCBBDAAC	245	7.71E-09	GGCDBCDBDDAB	336	5.07E-10
DHADDCBBDAAD	246	5.23E-09	GGADBCABDDAC	330	4.96E-10
DHDCACBBBAAA	247	4.95E-07	DDDDDCBBDAAB	212	4.93E-10
DHDCCCBBBAAA	248	1.74E-08	CDDDBCABDCAD	44	4.82E-10
DHDDDCBBDAAA	249	3.96E-08	DFADDCBBDAAC	217	4.79E-10
DHDDDCBBDAAB	250	2.19E-09	CDCDBCDBDCAD	16	4.77E-10
DHECACBBBAAA	251	2.26E-06	CDDDDCBBDBAC	46	4.64E-10
DHECCCBBBAAA	252	7.89E-08	CHADDCBBDCAA	161	4.24E-10
DHEDDCBBDAAA	253	5.75E-07	DGCDDCDBDAAB	236	4.07E-10
DHEDDCBBDAAB	254	4.73E-08	CFADDCBBDCAA	67	4.06E-10
EFACACBBBAAA	255	8.95E-10	CGCCCCDBBBAA	120	3.93E-10
EFACBCABDAAA	256	4.77E-09	CGCDDCDBDBAA	126	3.62E-10
EFACCCBBBAAA	257	1.64E-12	GDCCFCDBDDAA	304	3.61E-10
EFADBCABDAAA	258	1.02E-06	CGCDDCDBDBAB	127	3.61E-10
EFADBCABDAAB	259	4.24E-07	DGADDCBBDAAC	231	3.45E-10
EFADBCABDAAC	260	1.37E-07	CFCDDCDBDBAA	80	3.43E-10
EFADBCABDAAD	261	1.99E-08	GHCCFCDBDDAA	346	3.34E-10
EFCCACDBBAAA	262	3.96E-11	CFCCBCDBDCAA	73	3.06E-10
EFCCBCDBDAAA	263	1.61E-10	EFDCBCABDAAA	268	3.02E-10
EFCCCCDBBAAA	264	2.87E-12	DFCDDCDBDAAB	222	2.94E-10
EFCDBCDBDAAA	265	5.26E-08	CGADDCBBDBAD	112	2.85E-10
EFCDBCDBDAAB	266	2.07E-08	GFADBCABDDAC	322	2.77E-10

EFDCACBBBAAA	267	1.28E-11	CHDCCCBBBBAA	179	2.76E-10
EFDCBCABDAAA	268	3.02E-10	CHADDCBBDCAC	163	2.75E-10
EFDCCCBBBAAA	269	9.08E-13	EHDCBCABDAAA	292	2.52E-10
EFDDBCABDAAA	270	8.02E-08	GFADBCABDDAD	323	2.42E-10
EFDDBCABDAAB	271	3.10E-08	EHACACBBBAAA	284	2.03E-10
EGACACBBBAAA	272	6.53E-09	DGDCCCBBBAAA	238	1.69E-10
EGACBCABDAAA	273	2.26E-08	DGADDCBBDAAD	232	1.67E-10
EGACCCBBBAAA	274	2.90E-11	CGDCACBBBCAA	131	1.63E-10
EGADBCABDAAA	275	5.04E-06	EFCCBCDBDAAA	263	1.61E-10
EGADBCABDAAB	276	2.28E-06	CFDCCCBBBBAA	88	1.60E-10
EGADBCABDAAC	277	7.13E-07	CFADDCBBDCAB	68	1.59E-10
EGADBCABDAAD	278	2.56E-07	CFDCBCABDCAA	87	1.57E-10
EGDCACBBBAAA	279	5.79E-11	CGADDCBBDCAA	113	1.49E-10
EGDCBCABDAAA	280	3.27E-09	CDCDDCDBDBAA	17	1.49E-10
EGDCCCBBBAAA	281	1.03E-13	CFACCCBBBCAA	54	1.43E-10
EGDDBCABDAAA	282	1.06E-06	CFADDCBBDBAD	66	1.40E-10
EGDDBCABDAAB	283	6.29E-07	CDDDDCBBDCAA	47	1.16E-10
EHACACBBBAAA	284	2.03E-10	CFCCCCDBBCAA	75	1.16E-10
EHACBCABDAAA	285	1.18E-09	CDCDDCDBDCAB	22	1.15E-10
EHACCCBBBAAA	286	4.90E-12	CHDDDCBBDBAA	185	1.03E-10
EHADBCABDAAA	287	5.28E-07	CGADDCBBDCAB	114	1.00E-10
EHADBCABDAAB	288	5.42E-07	CFCDDCDBDBAB	81	9.35E-11
EHADBCABDAAC	289	1.77E-07	CGACCCBBBCAA	100	7.89E-11
EHADBCABDAAD	290	1.19E-07	CHADDCBBDCAD	164	7.84E-11
EHDCACBBBAAA	291	1.44E-12	DFDDDCBBDAAA	225	7.27E-11
EHDCBCABDAAA	292	2.52E-10	GHECBCABDDAA	357	7.24E-11
EHDCCCBBBAAA	293	2.74E-13	GDCCBCDBDDAA	302	6.11E-11
EHDDBCABDAAA	294	1.38E-07	GHECACBBBDAA	356	5.98E-11
EHDDBCABDAAB	295	1.37E-07	CHDDDCBBDBAB	186	5.82E-11
EHECACBBBAAA	296	8.45E-12	EGDCACBBBAAA	279	5.79E-11
EHECBCABDAAA	297	1.02E-09	DFADDCBBDAAD	218	5.58E-11
EHECCCBBBAAA	298	1.61E-12	GHDCBCABDDAA	352	5.35E-11
EHEDBCABDAAA	299	3.25E-07	GDCCACDBBDAA	301	5.28E-11
EHEDBCABDAAB	300	3.30E-07	GDCDBCDBDDAC	307	5.19E-11
GDCCACDBBDAA	301	5.28E-11	CHDDDCBBDCAB	188	5.08E-11
GDCCBCDBDDAA	302	6.11E-11	GDDCBCABDDAA	314	4.52E-11
GDCCCCDBBDAA	303	5.60E-12	CGDDDCBBDBAB	140	4.33E-11
GDCCFCDBDDAA	304	3.61E-10	EFCCACDBBAAA	262	3.96E-11
GDCDBCDBDDAA	305	1.82E-08	GDCDBCDBDDAD	308	3.84E-11
GDCDBCDBDDAB	306	5.15E-09	GDDDBCABDDAC	318	3.59E-11
GDCDBCDBDDAC	307	5.19E-11	EGACCCBBBAAA	274	2.90E-11
GDCDBCDBDDAD	308	3.84E-11	GHACBCABDDAA	340	2.57E-11
GDCDFCDBDDAA	309	7.46E-09	CGCCCCDBBCAA	121	2.56E-11
GDCDFCDBDDAB	310	6.94E-09	GDDDBCABDDAD	319	2.51E-11
GDCDFCDBDDAC	311	2.11E-09	DGDDDCBBDAAA	239	2.12E-11
GDCDFCDBDDAD	312	1.85E-09	CFDCACBBBCAA	85	2.09E-11
GDDCACBBBDAA	313	1.32E-11	CGDCCCBBBBAA	134	2.08E-11

GDDCBCABDDAA	314	4.52E-11	GFADBCABDDAA	320	1.85E-11
GDDCCCBBBDAA	315	1.40E-12	GHDCACBBBDAA	351	1.49E-11
GDDDBCABDDAA	316	1.35E-08	CGDDDCBBDBAA	139	1.36E-11
GDDDBCABDDAB	317	3.81E-09	GDDCACBBBDAA	313	1.32E-11
GDDDBCABDDAC	318	3.59E-11	EFDCACBBBAAA	267	1.28E-11
GDDDBCABDDAD	319	2.51E-11	GGADBCABDDAA	328	1.23E-11
GFADBCABDDAA	320	1.85E-11	CDDCCCBBBBAA	35	1.16E-11
GFADBCABDDAB	321	6.35E-10	DGDDDCBBDAAB	240	1.07E-11
GFADBCABDDAC	322	2.77E-10	CDDDDCBBDCAB	48	9.55E-12
GFADBCABDDAD	323	2.42E-10	CFCDDCDBDCAA	82	9.54E-12
GFCDBCDBDDAA	324	1.70E-12	CFDDDCBBDBAB	94	8.56E-12
GFCDBCDBDDAB	325	2.95E-13	EHECACBBBAAA	296	8.45E-12
GFDDBCABDDAA	3[illegible]	8.93E-13	CGCDDCDBDCAB	129	6.73E-12
GFDDBCABDDAB	3[illegible]7	1.51E-13	GDCCCCDBBDAA	303	5.60E-12
GGADBCABDDAA	328	1.23E-11	GHECCCBBBDAA	358	5.01E-12
GGADBCABDDAB	329	9.95E-10	EHACCCBBBAAA	286	4.90E-12
GGADBCABDDAC	330	4.96E-10	GGCCBCDBDDAA	333	4.78E-12
GGADBCABDDAD	331	5.39E-10	CFDCCCBBBCAA	89	4.57E-12
GGCCACDBBDAA	332	1.82E-13	EFCCCCDBBAAA	264	2.87E-12
GGCCBCDBDDAA	333	4.78E-12	GHACACBBBDAA	339	2.79E-12
GGCCCCDBBDAA	334	7.63E-15	CGCDDCDBDCAA	128	2.58E-12
GGCDBCDBDDAA	335	7.97E-10	CFADDCBBDCAC	69	2.56E-12
GGCDBCDBDDAB	336	5.07E-10	DDCDDCDBDAAD	208	2.12E-12
GGDDBCABDDAA	337	1.88E-12	GGDDBCABDDAA	337	1.88E-12
GGDDBCABDDAB	338	8.68E-13	GFCDBCDBDDAA	324	1.70E-12
GHACACBBBDAA	339	2.79E-12	EFACCCBBBAAA	257	1.64E-12
GHACBCABDDAA	340	2.57E-11	EHECCCBBBAAA	298	1.61E-12
GHACCCBBBDAA	341	1.63E-13	DDCDDCDBDAAC	207	1.58E-12
GHADBCABDDAA	342	4.41E-09	EHDCACBBBAAA	291	1.44E-12
GHADBCABDDAB	343	5.57E-09	GDDCCCBBBDAA	315	1.40E-12
GHADBCABDDAC	344	2.31E-09	GHDCCCBBBDAA	353	1.25E-12
GHADBCABDDAD	345	2.62E-09	CGADDCBBDCAC	115	1.17E-12
GHCCFCDBDDAA	346	3.34E-10	EFDCCCBBBAAA	269	9.08E-13
GHCDFCDBDDAA	347	6.24E-09	GFDDBCABDDAA	326	8.93E-13
GHCDFCDBDDAB	348	6.04E-09	GGDDBCABDDAB	338	8.68E-13
GHCDFCDBDDAC	349	1.69E-09	CFCDDCDBDCAB	83	6.51E-13
GHCDFCDBDDAD	350	1.64E-09	CDCDDCDBDBAD	20	5.75E-13
GHDCACBBBDAA	351	1.49E-11	DFDDDCBBDAAB	226	5.22E-13
GHDCBCABDDAA	352	5.35E-11	GFCDBCDBDDAB	325	2.95E-13
GHDCCCBBBDAA	353	1.25E-12	EHDCCCBBBAAA	293	2.74E-13
GHDDBCABDDAA	354	1.78E-08	CGDDDCBBDCAB	142	2.69E-13
GHDDBCABDDAB	355	1.99E-09	CGDDDCBBDCAA	141	2.06E-13
GHECACBBBDAA	356	5.98E-11	GGCCACDBBDAA	332	1.82E-13
GHECBCABDDAA	357	7.24E-11	GHACCCBBBDAA	341	1.63E-13
GHECCCBBBDAA	358	5.01E-12	GFDDBCABDDAB	327	1.51E-13
GHEDBCABDDAA	359	2.41E-08	EGDCCCBBBAAA	281	1.03E-13
GHEDBCABDDAB	360	2.69E-09	GGCCCCDBBDAA	334	7.63E-15

APPENDIX F

SUPPORTING INFORMATION FOR THE MELCOR ANALYSIS

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APPENDIX F MELCOR CODE CALCULATIONS

F.1 Plant Model

The MELCOR nodalization scheme for the Surry plant is shown in Figure F.1. The reactor vessel is divided into six nodes representing the downcomer, lower plenum, core, core bypass, upper plenum and the upper head regions. In the core and lower plenum regions, fuel and structures are represented by 39 cells. There are three radial rings and 13 axial sections in each ring. Among the 13 axial sections, 10 are fuel elements and 3 are lower core plate and lower plenum nodes. Penetration tubes located on the lower head wall are modeled for each radial ring. The failure of the penetration tube indicates vessel breach.

Surry is a 3-loop PWR plant. Each loop is represented by a hot-leg and a cold-leg. The pressurizer, pressurizer surge tank, RHR and RWST are modeled as separate volumes in the MELCOR nodalization. The RHR was used to establish the steady-state condition for the MELCOR analysis.

The Surry containment is divided into seven nodes representing the basement, cavity, cubicles of the three steam generators, pressurizer room and the dome area. Environment is modeled as an additional node.

The 24 nodes are inter-connected by 52 flow paths. The descriptions of the control volumes and flow paths are given in Tables F.1 and F.2, respectively. It is believed that the MELCOR nodalizarion provides a reasonable representation of the Surry piant.

F.2 Sequence Description

A total of 6 calculations were performed in support of this study as summarized in Table F.3. The 'time window' approach was based on a set of representative decay heat levels. The first three MELCOR sequences assumed decay power levels of 13.2, 7 and 5 MW, respectively. These power levels correspond to that defined in windows 1, 3 and 4, respectively, as describeo in the main report. In these sequences, it was assumed that the ECCS is not available and the containment is closed during the entire transient.

Sensitivity studies were performed to evaluate the effects of containment leakage, actuation of containment sprays and the restoration ECCS. The sensitivity studies were performed using the decay power level corresponding to that defined in window 1.

F.3 MELCOR Analysis

The analysis was based on mid-loop operation with the primary system open to the atmosphere. All six loop isolation valves were closed to minimize the primary system inventory and to preclude the use of reflux cooling as a recovery procedure. A Tygon tube connects the upper head vent to the pressurizer relief tank, and at least one pressurizer SRV is assumed to have been removed which provides a vent path to the containment.

All calculations commenced with a period of 5000 seconds running in quasi-steady state conditions with the actuation of RHR. This method was sufficient for the MELCOR model to approach a close approximation to a steady state condition prior to initiating the transient. The initial conditions in each control volume including the atmospheric pressure, temperature, composition and water pool mass and temperature are included in Table F.1.

The transient was initiated by closing off the RHR flow paths to simulate a loss of RHR. All timings referred to in this discussion are measured relative to this time in the calculation. Transient analyses were focused on the occurrence of major events, such as the timing of core uncovery, gap release, failure of the core support plate and vessel breach. Core uncovery is defined as the depletion of water in the upper plenum of the reactor vessel. The gap release occurs when the clad reaches 1173 K, at which temperature all inventories of fission products in the gap region are released instantaneously to the core channel control volume. The failure of core support plate and penetration tubes attached to the lower head wall are determined by the user specified failure temperatures (default values are 1273 K). The failure of penetration tubes indicates the breach of the reactor vessel.

The standard ANS decay power curve, as programmed into MELCOR, was used throughout these calculations to provide a best-estimate heat generation rate. The selection of the initial decay power levels was determined by the 'time window' approach described in the main volume of this report.

The latest release of MELCOR, version 1.8.2, was used throughout this analysis. This version includes several major improvements and corrections, particularly in the areas of core melting, relocation and interaction in the lower plenum.

F.4 Results

Case 1

This is the case with the highest decay power level (13.2 MW) corresponding to that defined in window 1. The failure of ECCS was assumed and the containment was closed. The loss of RHR leads to core uncovery at about 5280 seconds. The boil-off of water causes an increase of pressure and temperature in the RCS as shown in Figures F.2 to F.3. The fuel clad temperatures reach the criterion of gap release between about 5600 to 7500 seconds for fuels in the three radial rings. Continued loss of coolant and core heating eventually cause the failure of core support plate at about 12900 to 13900 seconds. The relocation of corium into the lower plenum and the rapid heating of the penetration tubes (Figure F.4) result the failure of the reactor vessel.

Large quantities of water and corium are discharged into the cavity following the vessel breach. The corium/water interaction in the cavity gradually vaporizes all the water remained in the cavity as shown in Figure F.5. This interaction has two effects on containment performance: containment pressurization due to steam addition and reduction of corium/concrete interaction due to the cooling of core debris. A limestone/common sand concrete was assumed for the analysis. Figure F.6 shows the radial and axial concrete erosion in the cavity. The axial erosion distance is about 0.75 meters at the end of 120,000 seconds. This is about 25% of the concrete floor thickness. It appears that thermal attack of the cavity concrete floor is not a severe challenge to the Surry containment for accidents during mid-loop operation.

The MELCOR predicted containment pressures and temperatures are illustrated in Figures F.7 and F.8, respectively. There is a continuous pressure increase in the containment. The two pressure spikes at about 40,000 and 80,000 seconds are caused by hydrogen burn in the containment dome and basement area. Within the time period of 120,000 seconds, the pressure does not threaten the containment integrity. However, high temperatures occur in the cavity region over a very long period of time. This severe thermal condition could cause damage in that area.

The in-vessel hydrogen generation and hydrogen distribution in containment are shown in Figures F.9 to F.10. The sudden reduction of hydrogen mass at about 40,000 seconds in the dome and basement region, and at about 80,000 seconds in the dome area indicates deflagrations in the containment. They are reflected as pressure and temperature spikes in Figures F.7 and F.8.

MELCOR calculates radionuclides in two forms: vapor and aerosol. The distributions of active aerosols and vapor in various containment regions are given in Figures F.11 and F.12. Most of the active aerosols and vapor are accumulated in the dome and basement regions. The distributions of Cs and I elements at the end of 24 hours are summarized in Table F.5. Because the containment is assumed to be closed, there is no environmental release.

Case 2

This case is similar to Case 1, but with the decay power level reduced to 7 MW corresponding to that defined in window 2. With a lower decay power the occurrence of major events are delayed by 2 to 3 hours as shown in Table F.4. The results of the transient calculation are given in Figures F.13 to F.23. Similar to Case 1, hydrogen generation, hydrogen burn, concrete erosion and containment pressurization do not threaten the containment integrity within a time period of 24 hours after the initiation of accident. The high temperatures in the cavity still present a severe challenge to the containment integrity. The distributions of Cs and I elements are about the same as in Case 1 as shown in Table F.5.

Case 3

The decay power level is further reduced to 5 MW in this case. The transient behavior shown in Figures F.24 to F.34 are similar to that of Cases 1 and 2. However, at such a low level of decay power, the occurrence of major events are delayed considerably in comparison with that of Case 1 as summarized in Table F.4.

Case 4

This case is similar to Case 1 but with the assumption of containment leakage initiated at the beginning of the accident. The leak area is assumed to be 1 square ft and is located in the dome area. The MELCOR predicted transient behavior and results are given in Figures F.35 to F.45, and in Tables F.4 and F.5.

The leakage in containment does not have any major impact on transient behavior in the reactor vessel. Only the in-vessel hydrogen production is reduced by about 10% in comparison with Case 1. In the containment, the pressure is generally at atmospheric level. The gas temperature in the cavity region is still at an elevated level of about 1700 K (Figure F.41). It is noted that both the radial and axial concrete erosion in the cavity region (Figure F.39) are much stronger than that in the base case (i.e. Case 1). This strong erosion of concrete would release more gases from the concrete, which could become a driving force to discharge gases and aerosols into the environment. Figures F.44 and F.45 show that large quantities of active vapor and aerosols are released into the environment. The distributions of Cs, I elements and other species, summarized in Table F.5 indicate that about 20%, 87% and 53% of the total releases of Cs, I and all species are released to the environment, respectively.

Case 5

This case is similar to Case 4 but with the actuation of the containment sprays after vessel failure. The flow rate of the sprays is about 0.19 Kg/s as illustrated in Figure F.46. The spray heads are located in the dome area. Actuation of sprays after vessel failure has no effect on transient behavior in the reactor vessel as shown in Figures F.47 to F.49 and in Table F.4.

Figure F.50 shows that spray water is collected in the cavity and basement area. The cavity is completely filled with water in about 9.5 hours after the actuation of sprays. The presence of such a large water pool in the cavity reduced the concrete erosion rate as shown in Figures F.51. According to the MELCOR model, core debris in the cavity never reached a coolable configuration which could terminate the concrete erosion. As one expected, sprays eliminated high temperatures in containment (Figure F.53) and greatly reduced the releases of active aerosols and vapor to the environment (Figures F.56 and F.57). Table F.5 indicates that releases of radioactive species to the environment at the end of 73,515 seconds are negligible.

Case 6

The restoration of ECCS is studied in this case. It was assumed that the ECCS is restored about one hour after core damage. In the present analysis, core damage was conservatively assumed to occur when clad temperature exceeds 1000 K (1340 F). The choice was based on observations that clad oxidation becomes significant above this temperature. Since the oxidation is an exothermic reaction which accelerates with the increase of temperature, there exists a significant potential for fuel distortion, clad rupture, and its subsequent release of radioactive species at temperatures in excess of 1000 K.

MELCOR calculated results given in Figures F.58 to F.68 and in Tables F.4 to F.5 show that restoration of ECCS at about one hour after core damage effectively terminated the accident. No vessel failure was predicted and the environmental release is negligible.

It should be pointed out that the MELCOR predicted results for this case involve a large degree of uncertainty. MELCOR is not adequate for predicting thermal-hydraulic behavior after core damage. The code does not have an adequate model to accurately predict core reflood after core damage has occurred.

Table F-1 MELCOR 24-Node Compartment Description

(to be provided)

Table F-2 MELCOR 52 Inter-Compartment Flow Paths Description

(to be provided)

Table F-3 Summary of Accident Sequences Analyzed by MELCOR

Sequence No.	Decay Heat (MW)	Containment	ECCS	Vessel Failure
1	13.2	Closed	None	Yes
2	7.0	Closed	None	Yes
3	5.0	Closed	None	Yes
4	13.2	Leak*	None	Yes
5	13.2	Leak	Spray after VB	Yes
6	13.2	Leak	ECCS 1 hr after CD	No

* Leak area = 1.0 ft^2, Leak location = dome

Table F-4 Sequence of Events

Sequence No.	Core Uncovery* (s)	Gap Release			Core Support Plate Failure (s)			Vessel Breach (s)		
		Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3
1	5280	5598	6142	7504	12900	13018	13941	12949	13358	14129
2	10210	13769	14330	15734	22978	23153	25330	23030	23458	24905
3	14220	19803	20506	22276	31495	31588	31724	31550	31860	32056
4	5470	5424	5922	7285	11971	12215	12479	12016	12479	13362
5	5470	5424	5922	7285	11971	12215	12479	12016	12479	13362
6	5470	5424	5922	7285	—	—	—	—	—	—

* Defined as Dry-out of Upper Plenum

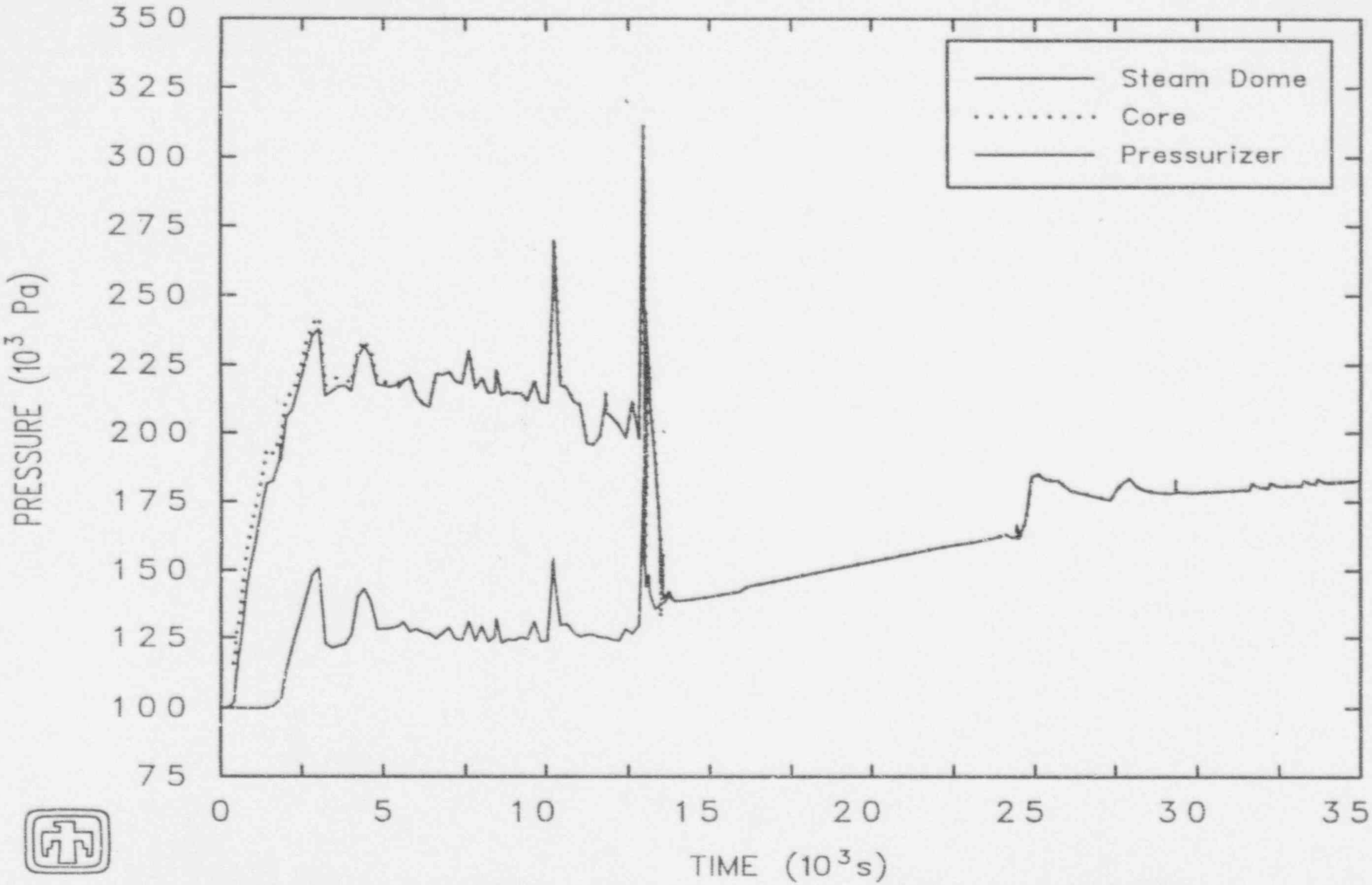
Table F-5 Distribution of Radioactive Species at 24 Hours

Case Number	Total Releases (Kg)			Containment (Kg)			Environment (Kg)		
	CS	I	All Species	CS	I	All Species	CS	I	All Species
1	138.2	9.14	493.4	121.1	9.12	462.2	0.0	0.0	0.0
2	138.9	8.47	445.0	126.3	8.45	427.9	0.0	0.0	0.0
3	138.9	8.47	449.6	124.7	8.45	430.2	0.0	0.0	0.0
4	138.9	8.48	491.1	95.0	1.09	204.9	28.6	7.39	261.0
5	4.3	0.333	12.86C	4.1	0.31	24.56C	7.9×10^{-3}	2.3×10^{-2}	0.569
6	4.3	0.333	12.86L	4.1	0.31	11.98C	7.9×10^{-3}	2.3×10^{-2}	0.571

* Results at 73515 s.

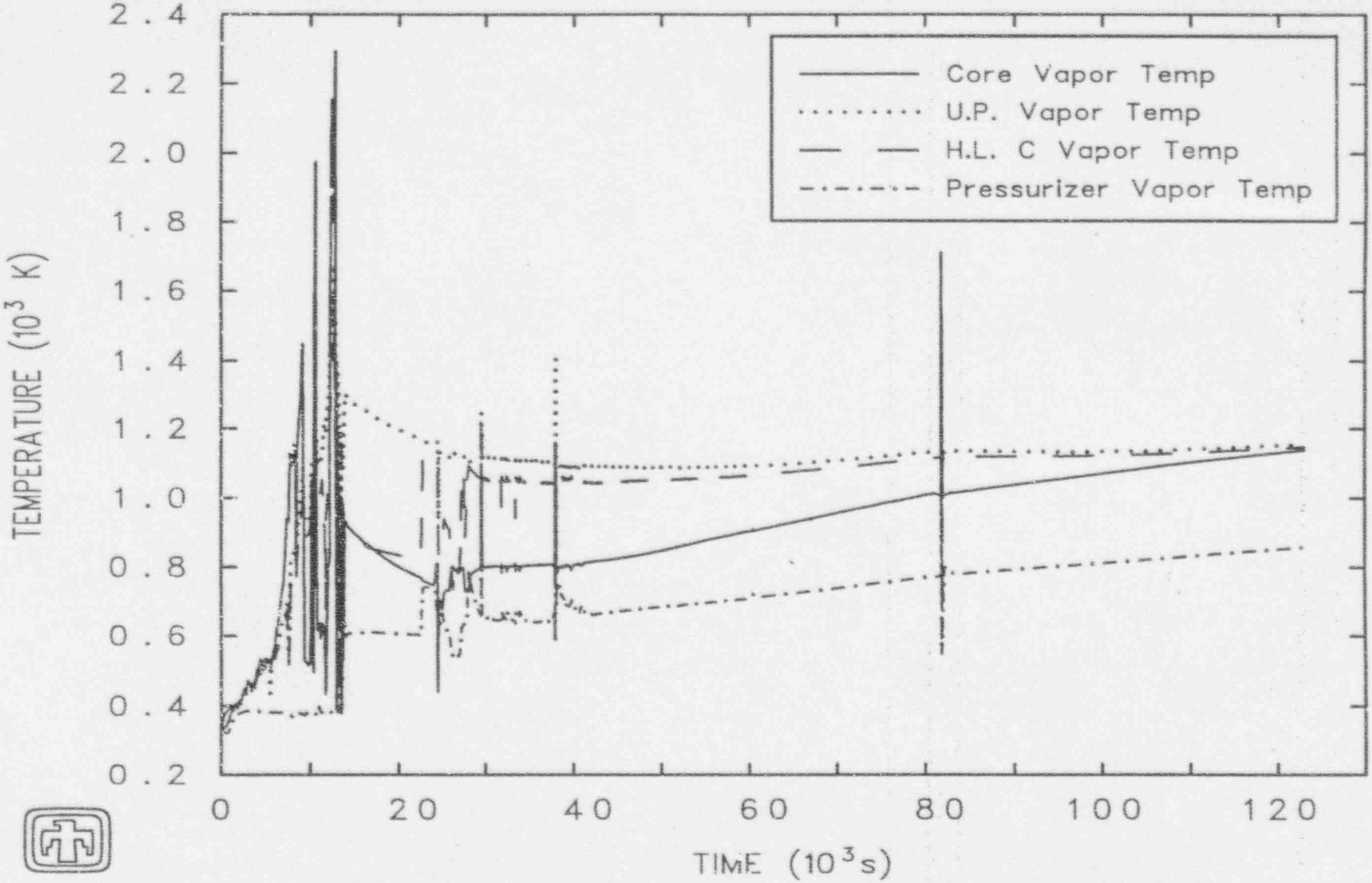
Figure F-1 MELCOR Nodalization

(to be provided)

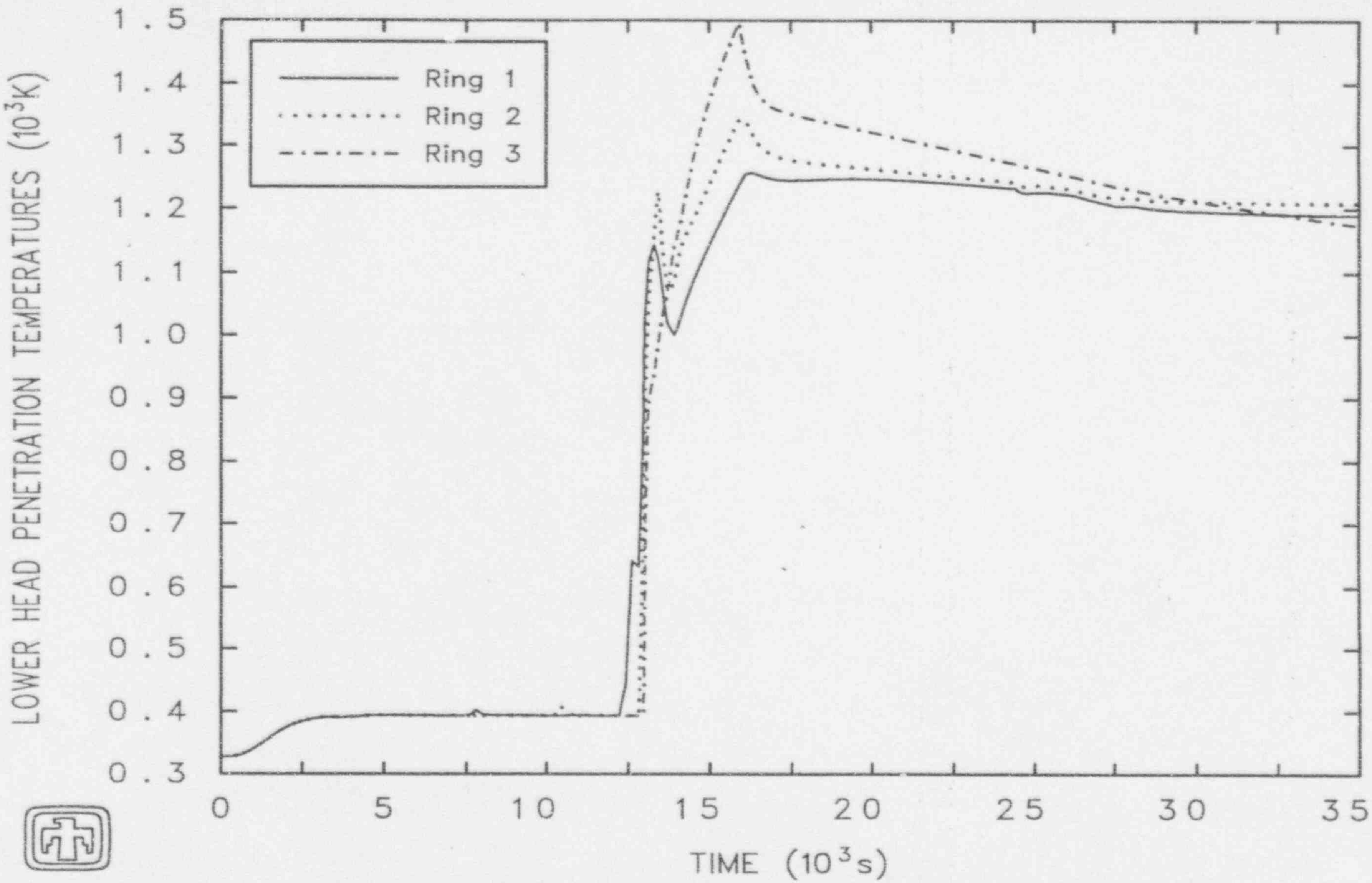


SRRYP6 - L2CASE2 - SHUTDOWN + 1.13 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 20:47:23 MELCOR

Fig 2-2

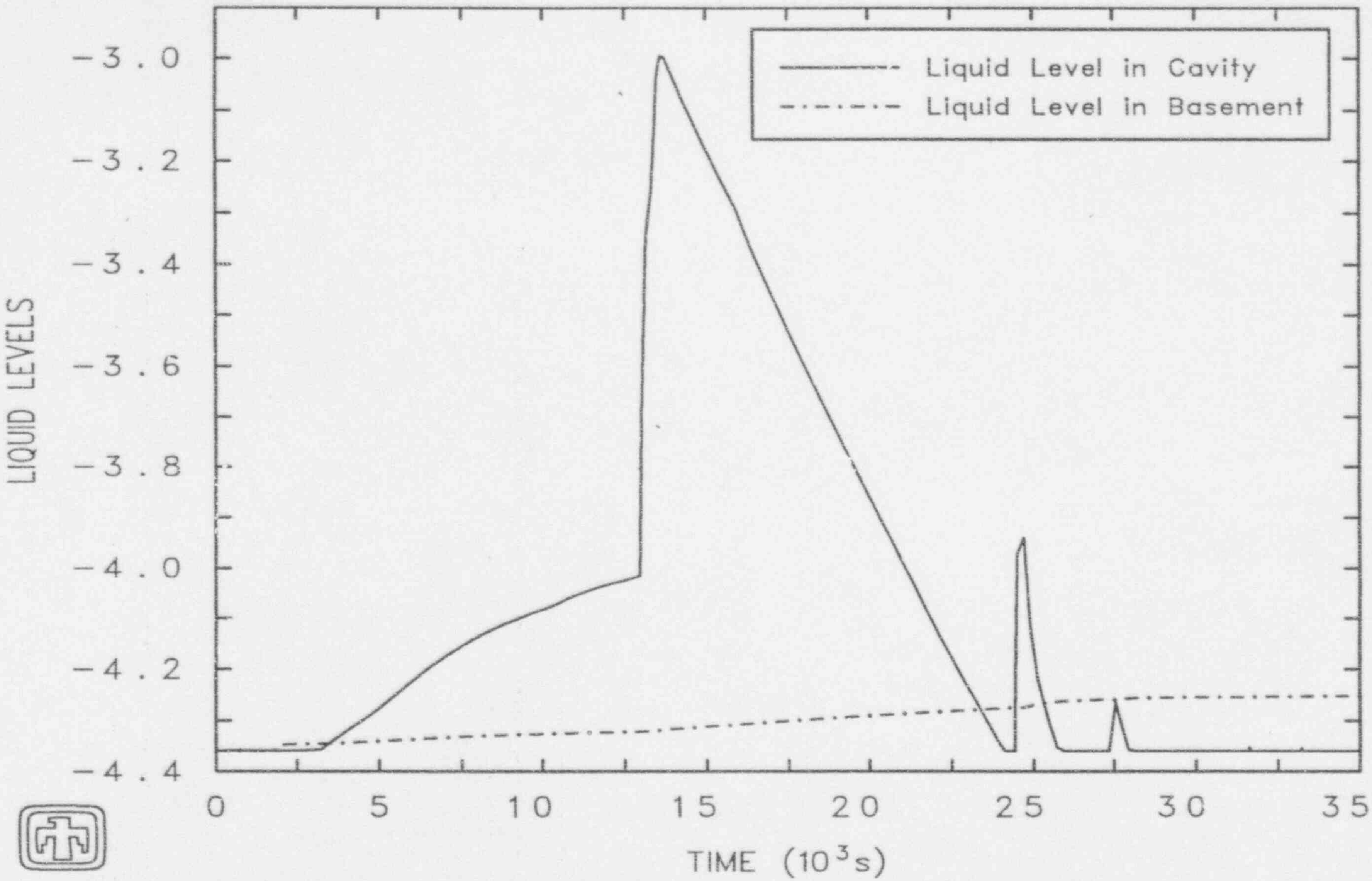


SRRYP6 - L2CASE2 - SHUTDOWN + 1.13 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 20:47:23 MELCOR



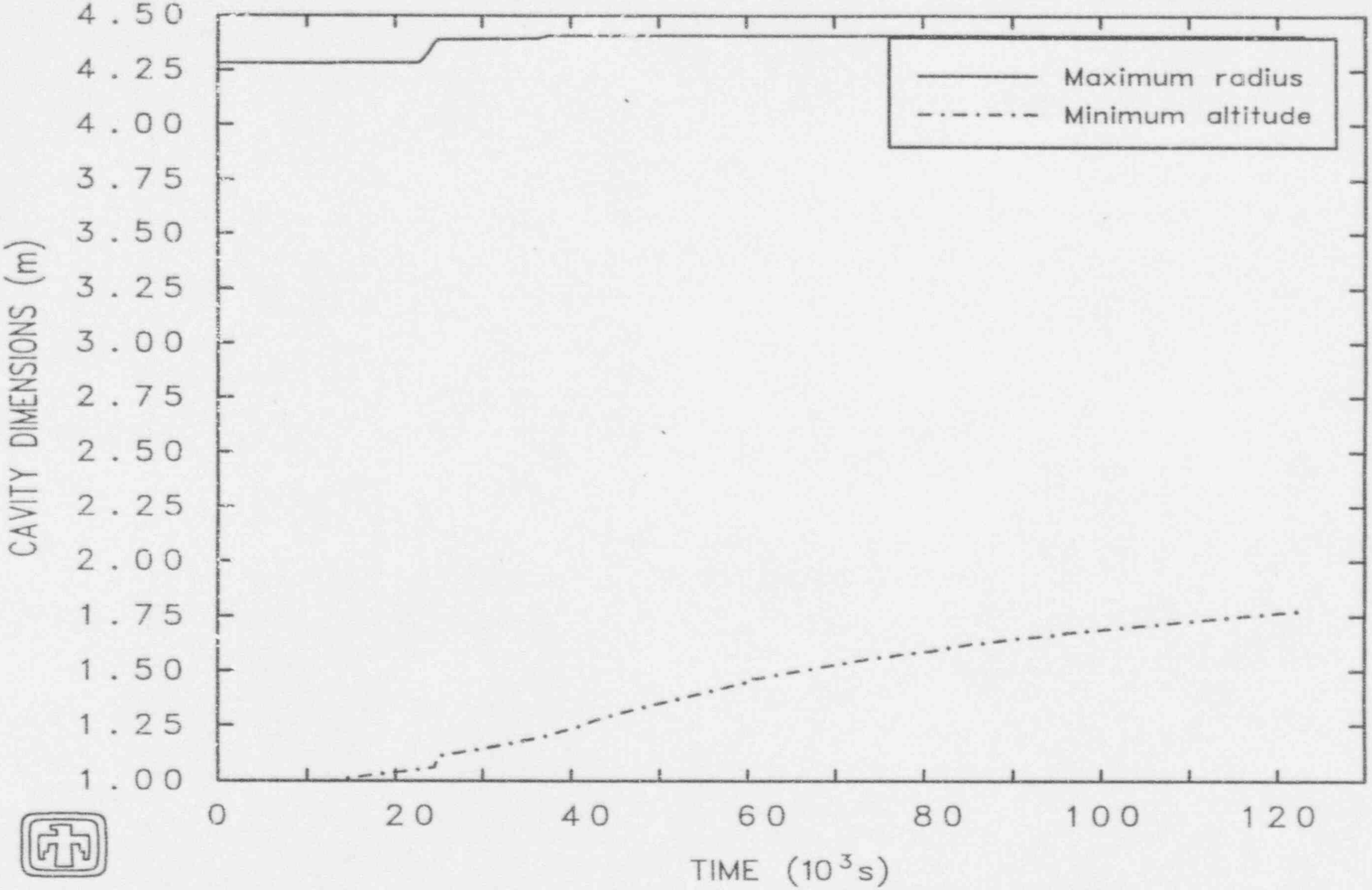
SRRYP6 – L2CASE2 – SHUTDOWN + 1.13 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 20:47:23 MELCOR

4



SRRYP6 - L2CASE2 - SHUTDOWN + 1.13 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 20:47:23 MELCOR

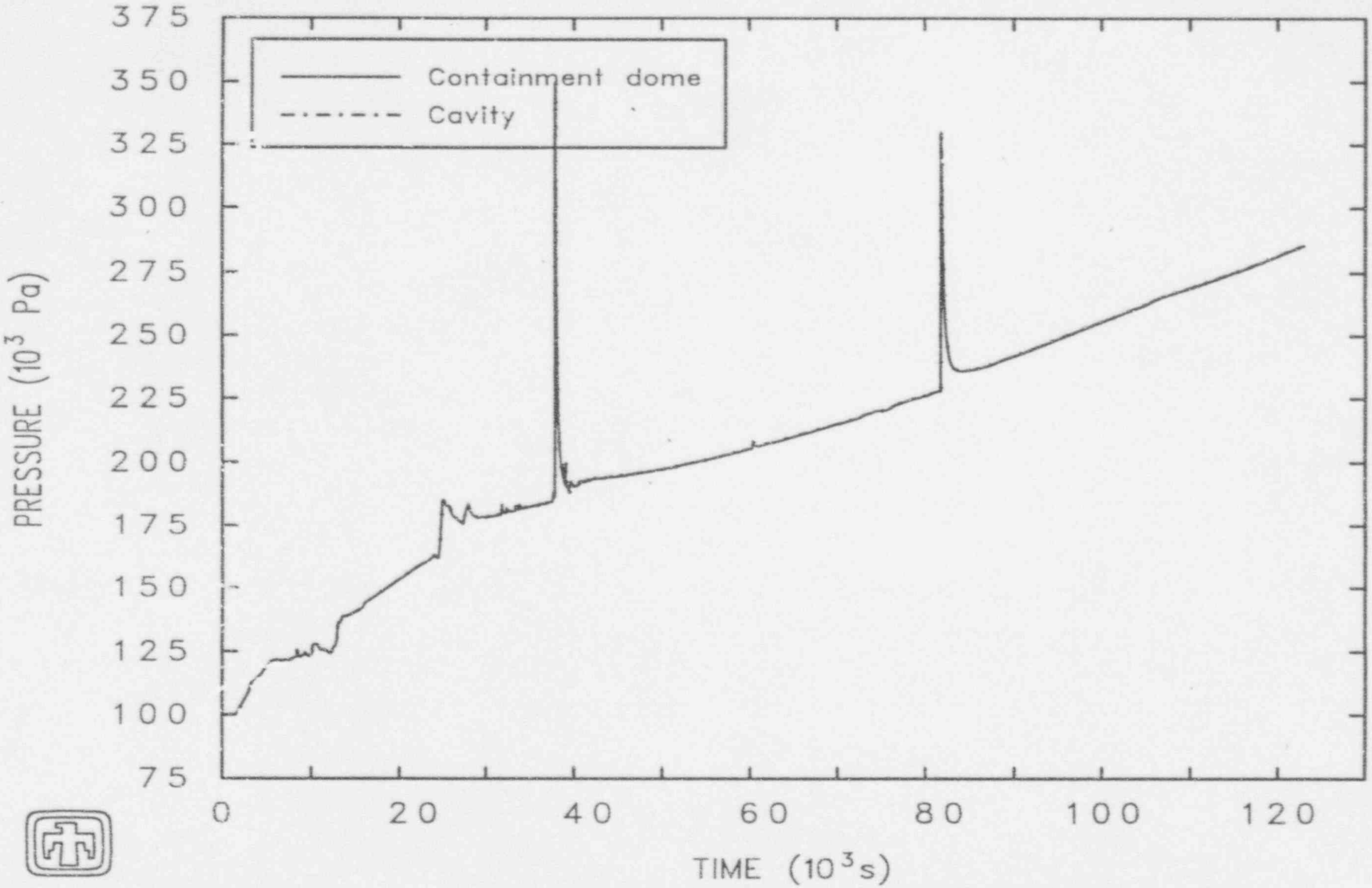
FIG 2-3



SRRYP6 – L2CASE2 – SHUTDOWN + 1.13 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 20:47:23 MELCOR

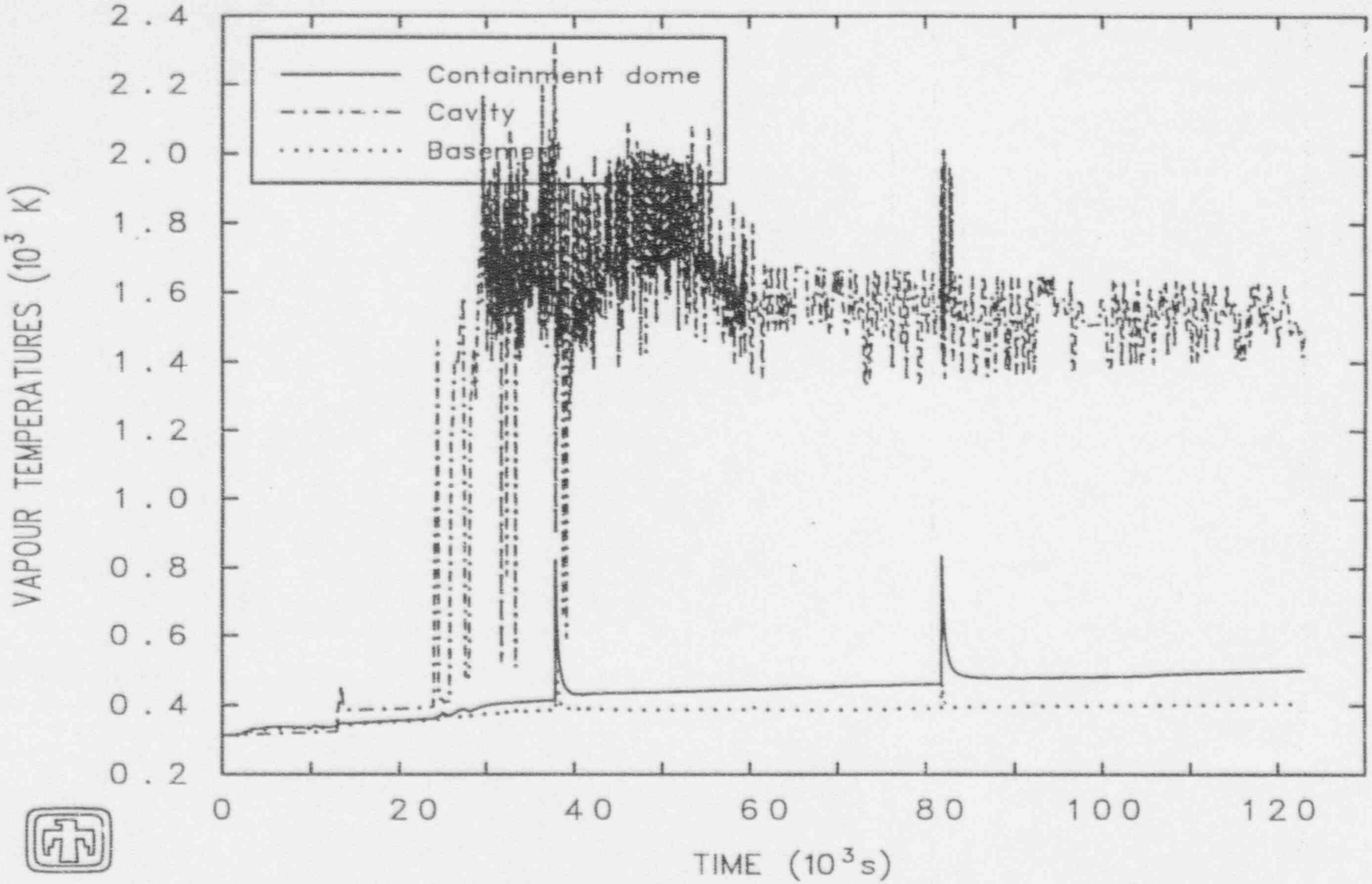
Fig 2-8

6

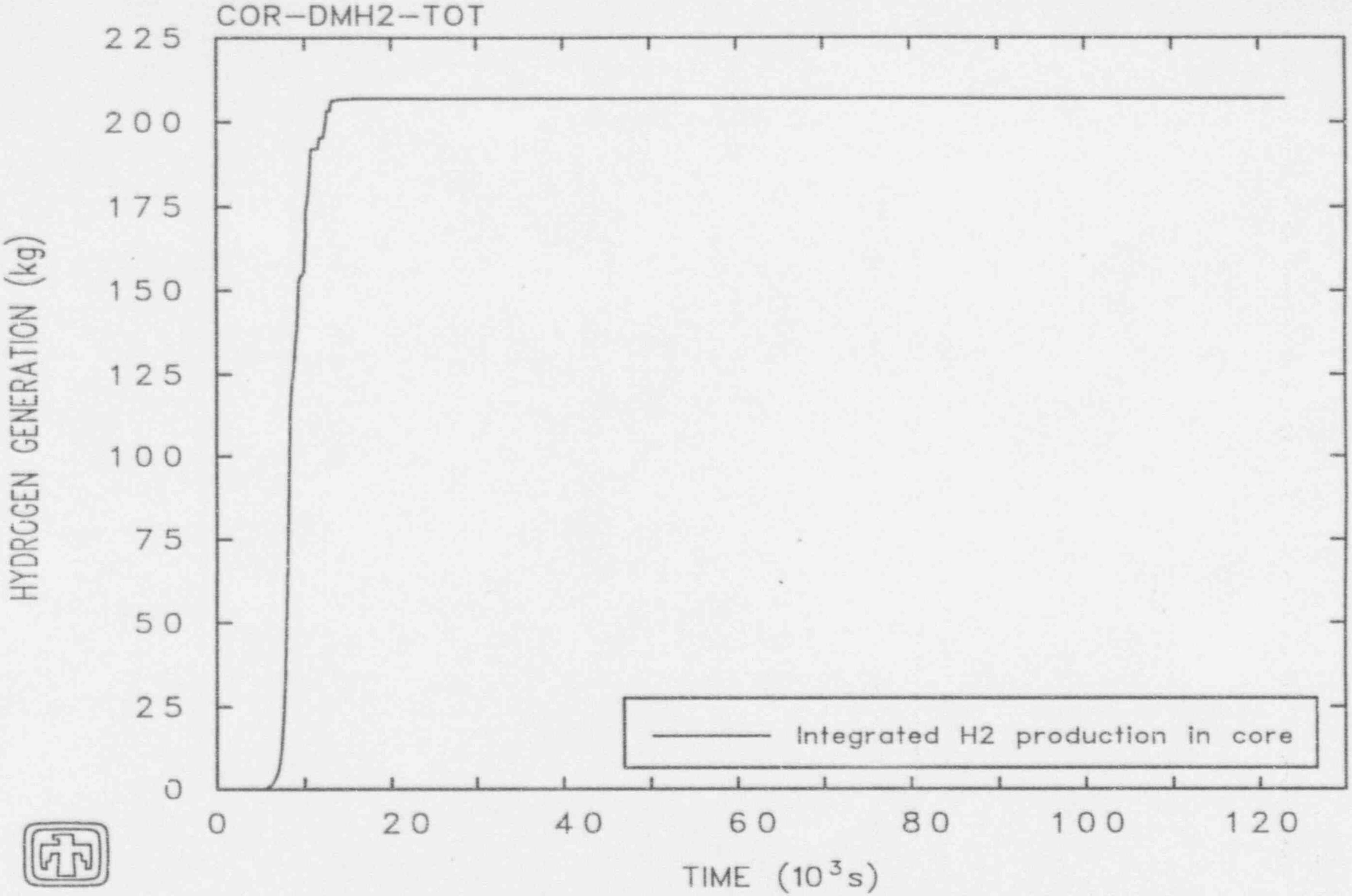


SRRYP6 - L2CASE2 - SHUTDOWN + 1.13 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 20:47:23 MELCOR

Fig 2-7

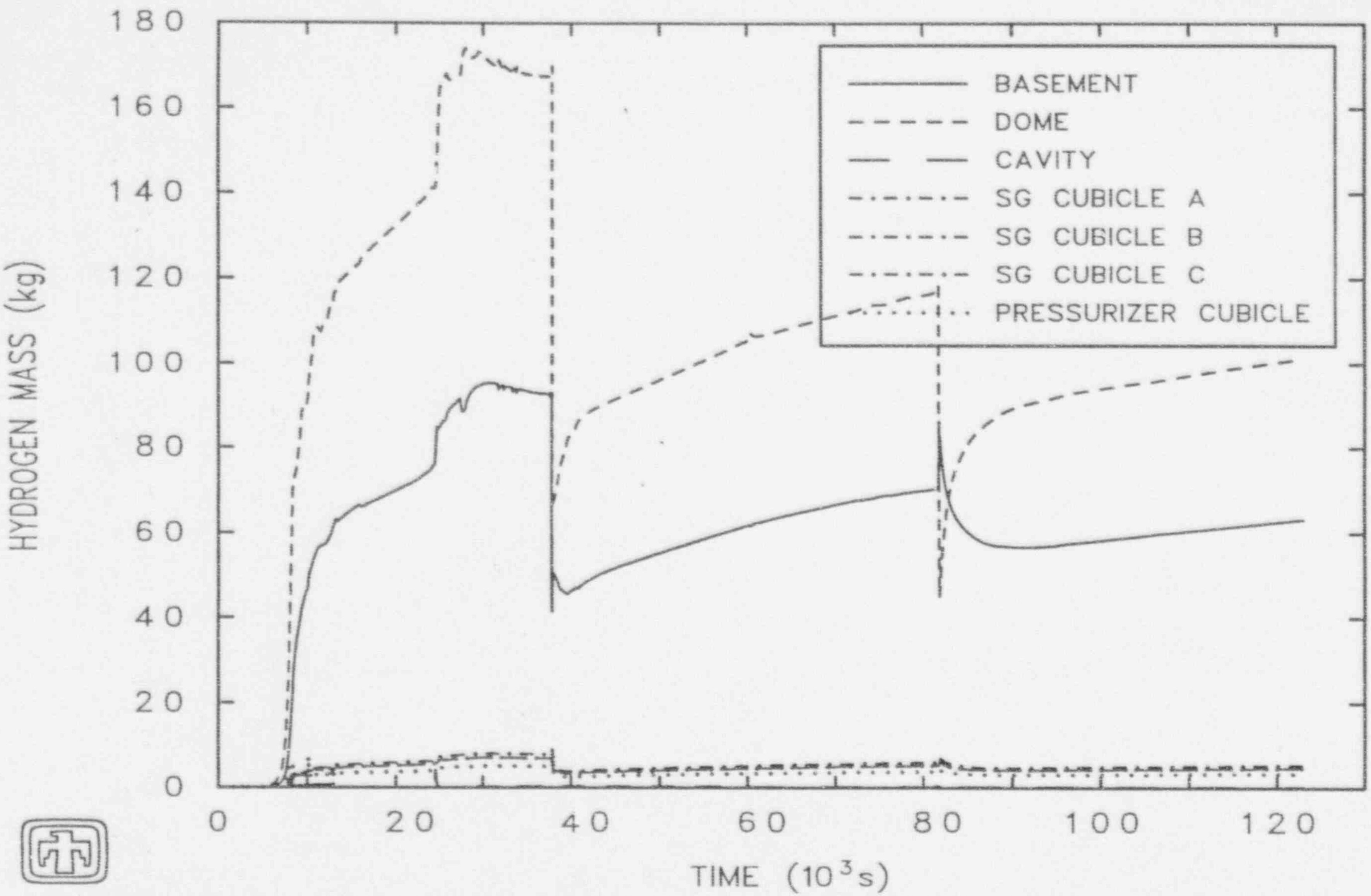


SRRYP6 - L2CASE2 - SHUTDOWN + 1.13 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 20:47:23 MELCOR



SRRYP6 - L2CASE2 - SHUTDOWN + 1.13 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 20:47:23 MELCOR

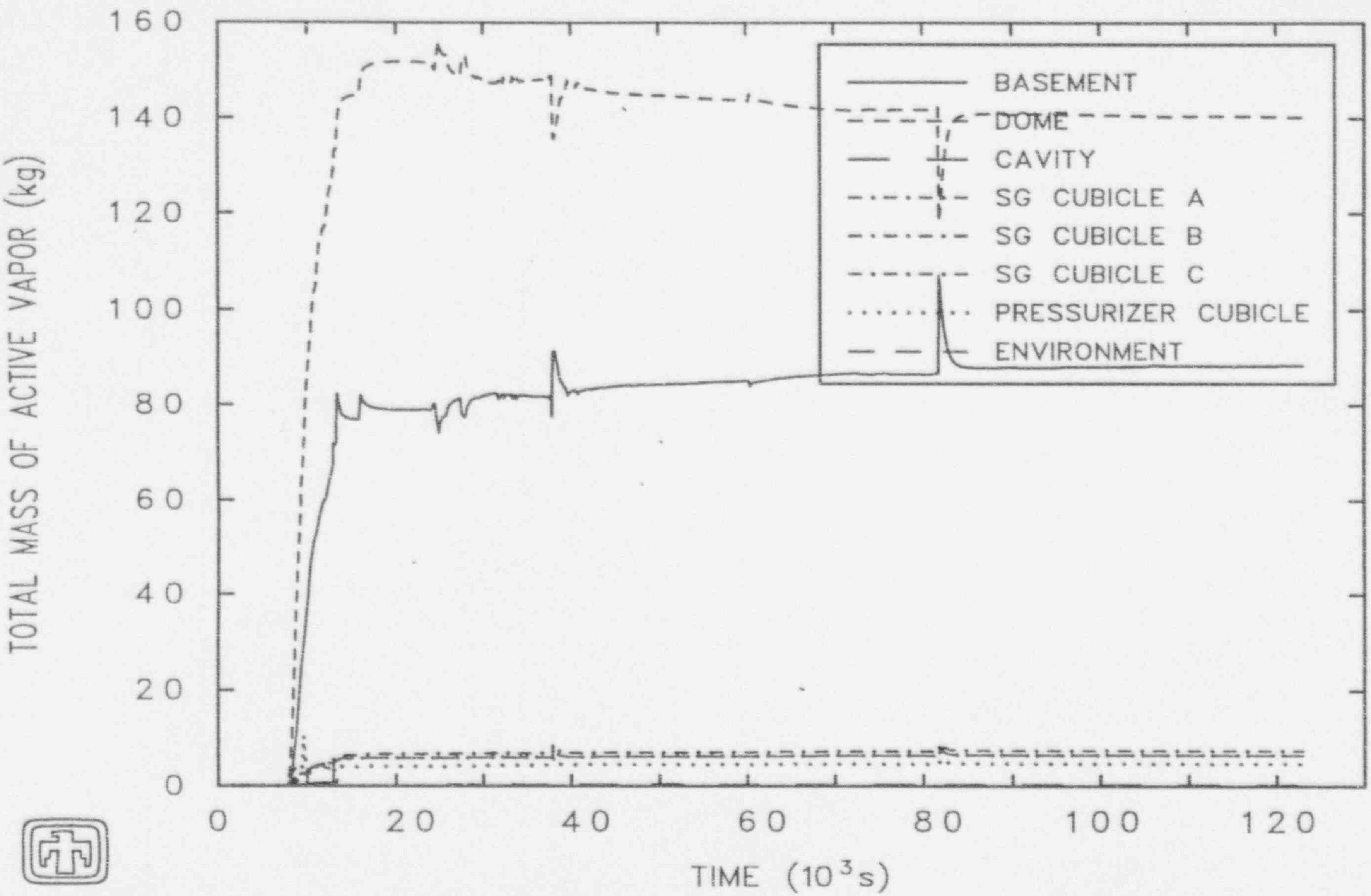
9



SRRYP6 - L2CASE2 - SHUTDOWN + 1.13 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 20:47:23 MELCOR

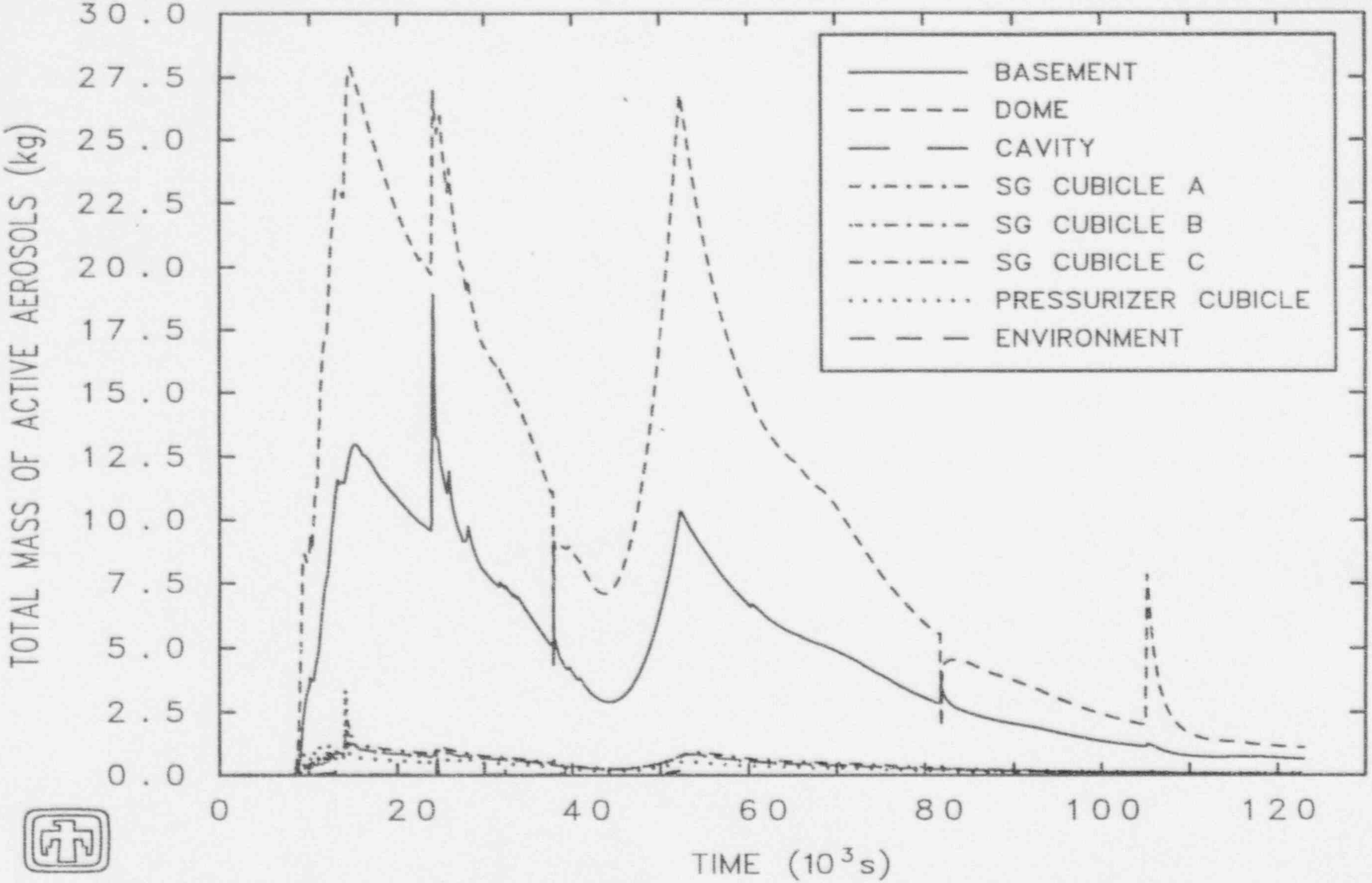
Fig 2-6

10



SRRYP6 - L2CASE2 - SHUTDOWN + 1.13 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 20:47:23 MELCOR

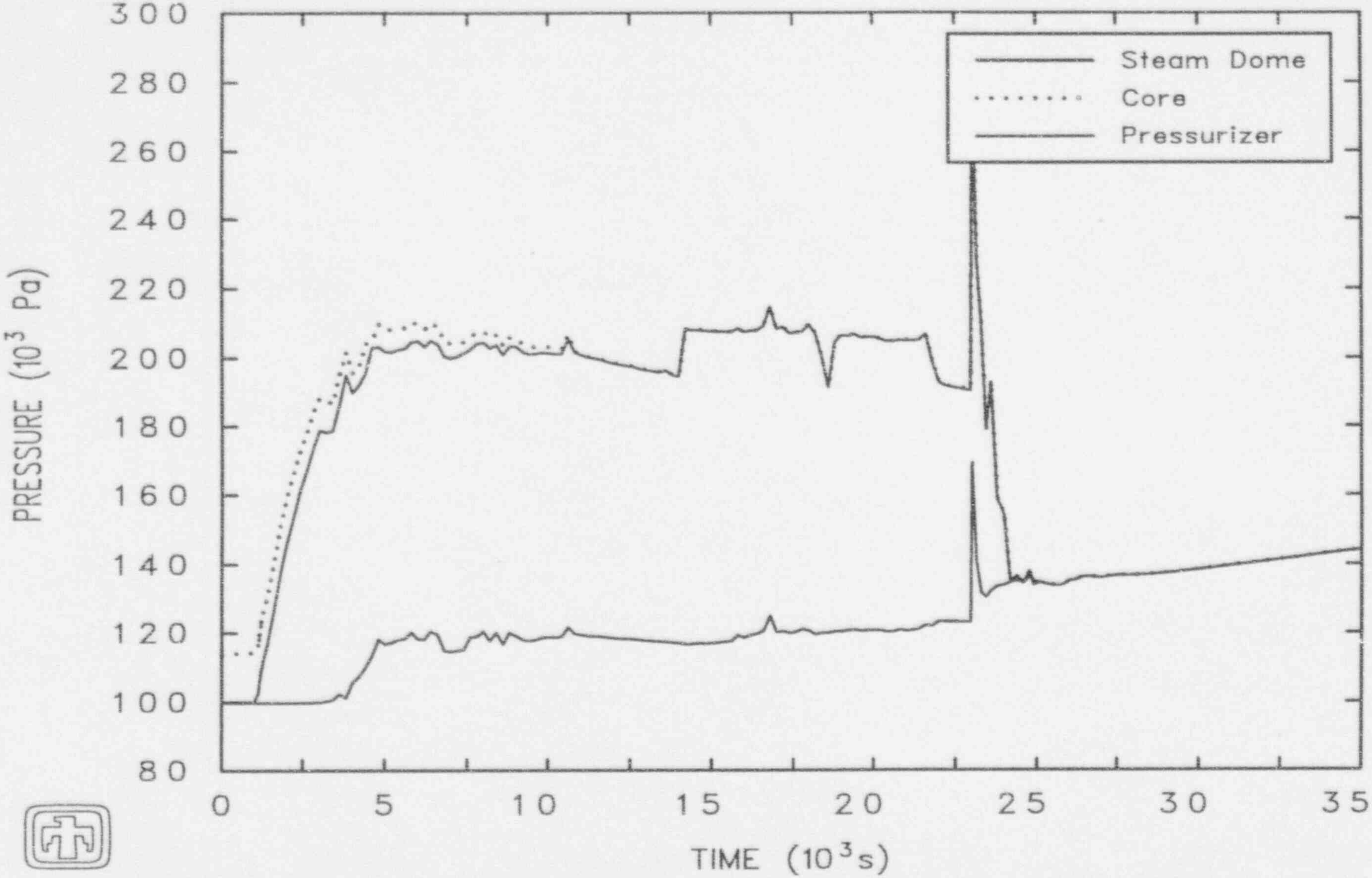
Fig 2-9



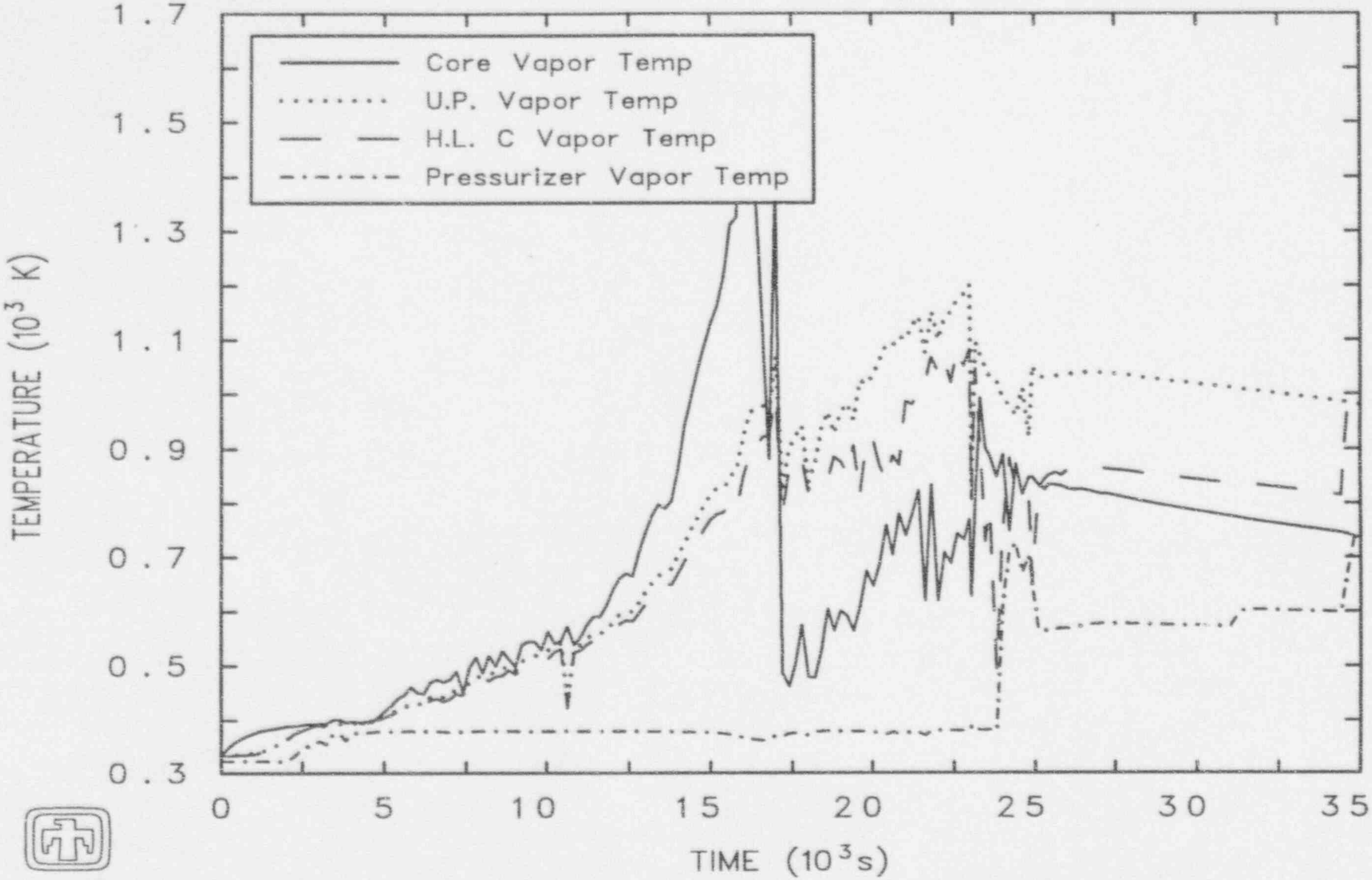
SRRYP6 - L2CASE2 - SHUTDOWN + 1.13 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 20:47:23 MELCOR

Fig 2-10

17

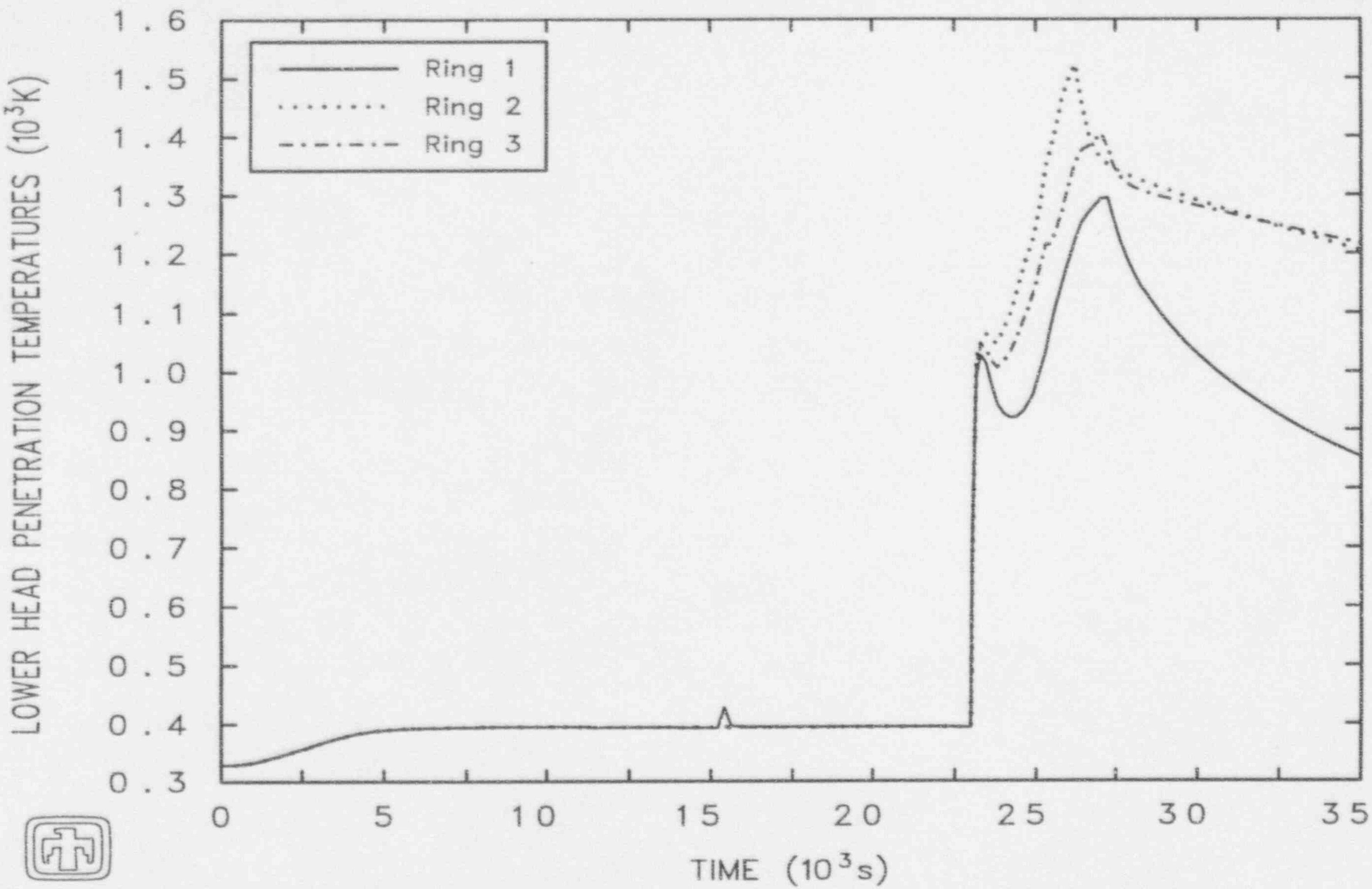


SRRYP6 - L2CASE3 - SHUTDOWN + 6.89 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 15:26:35 MELCOR

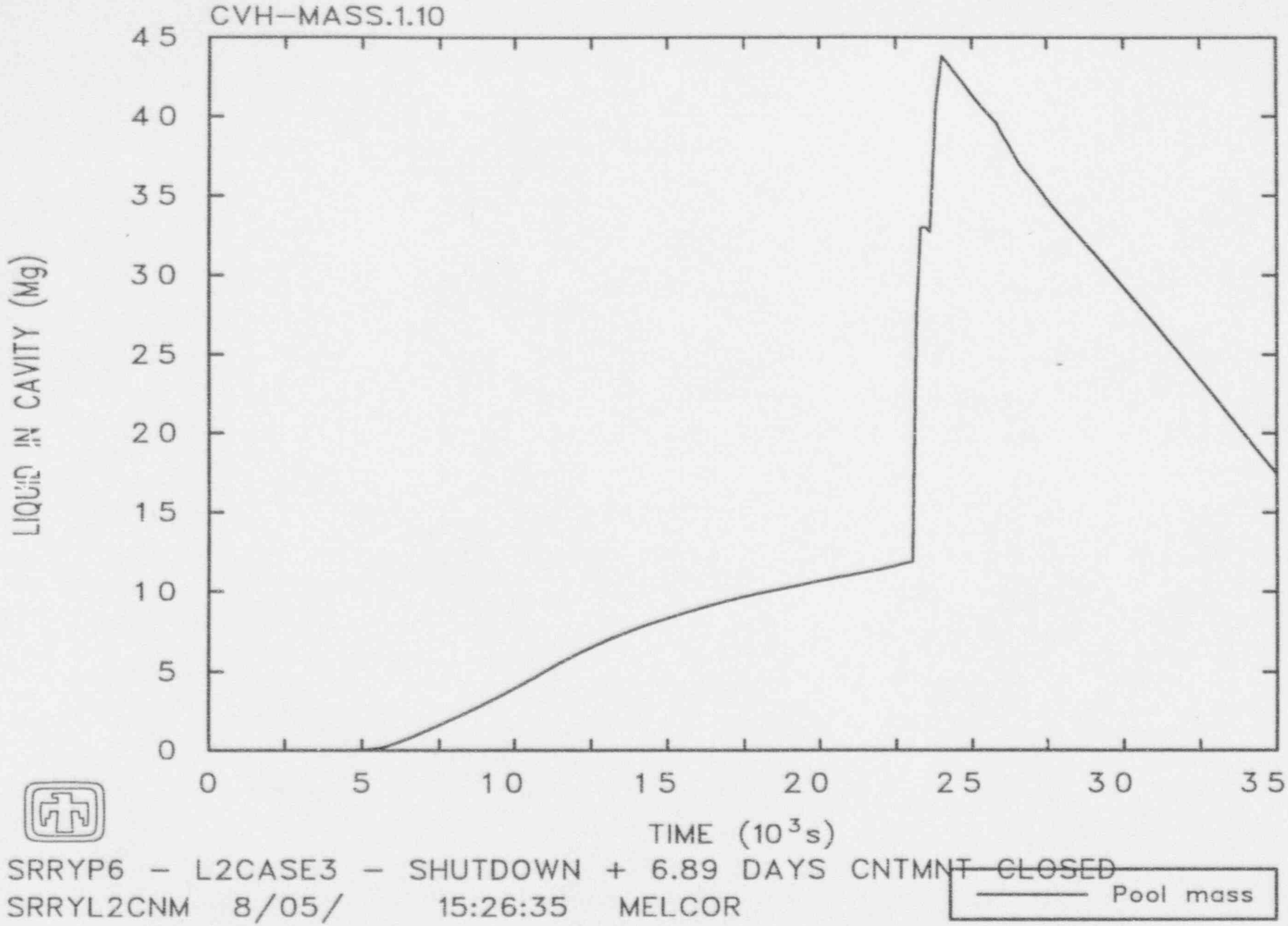


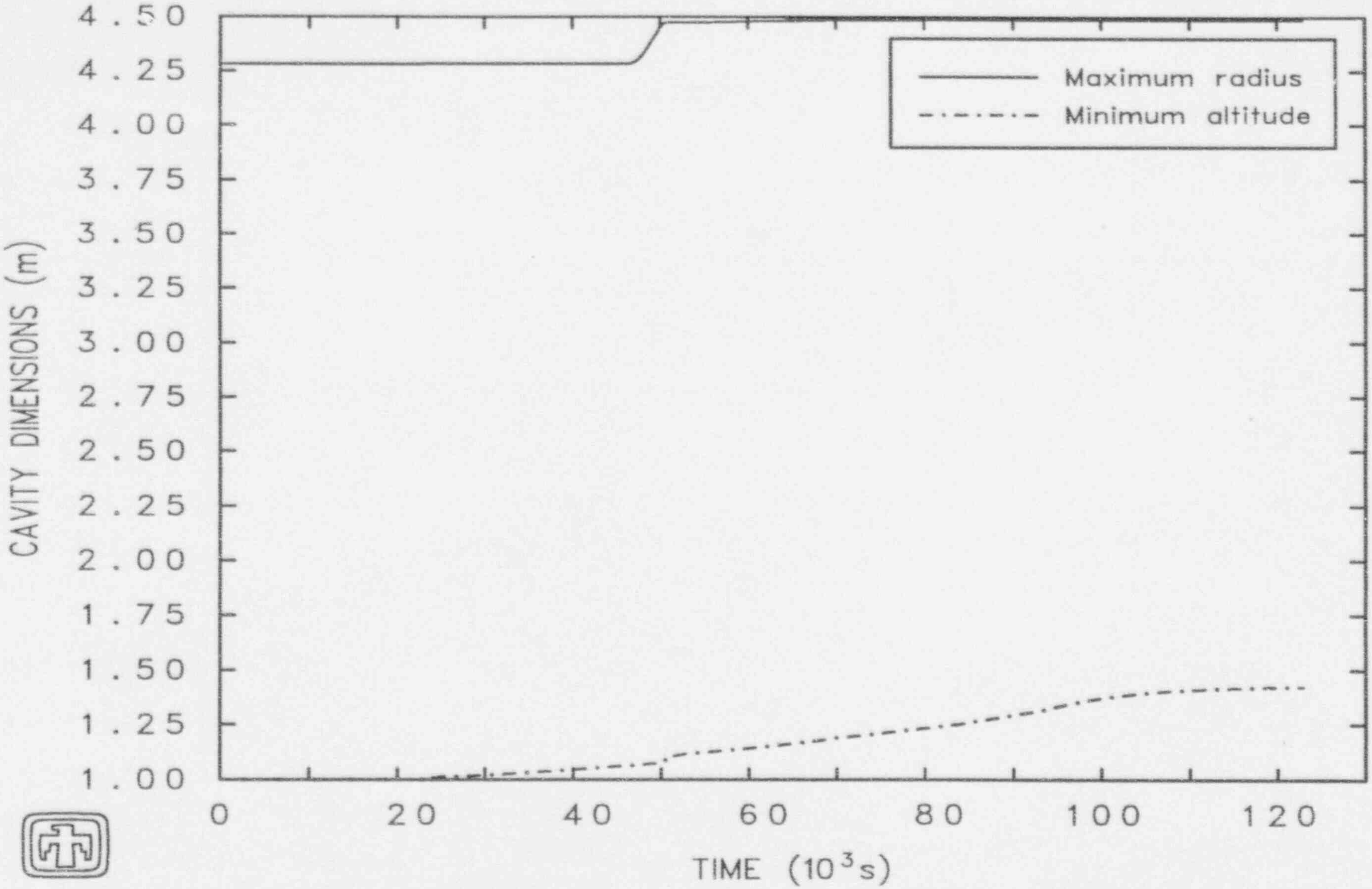
SRRYP6 - L2CASE3 - SHUTDOWN + 6.89 DAYS CNTMNT CLOSED

SRRYL2CNM 8/05/ 15:26:35 MELCOR

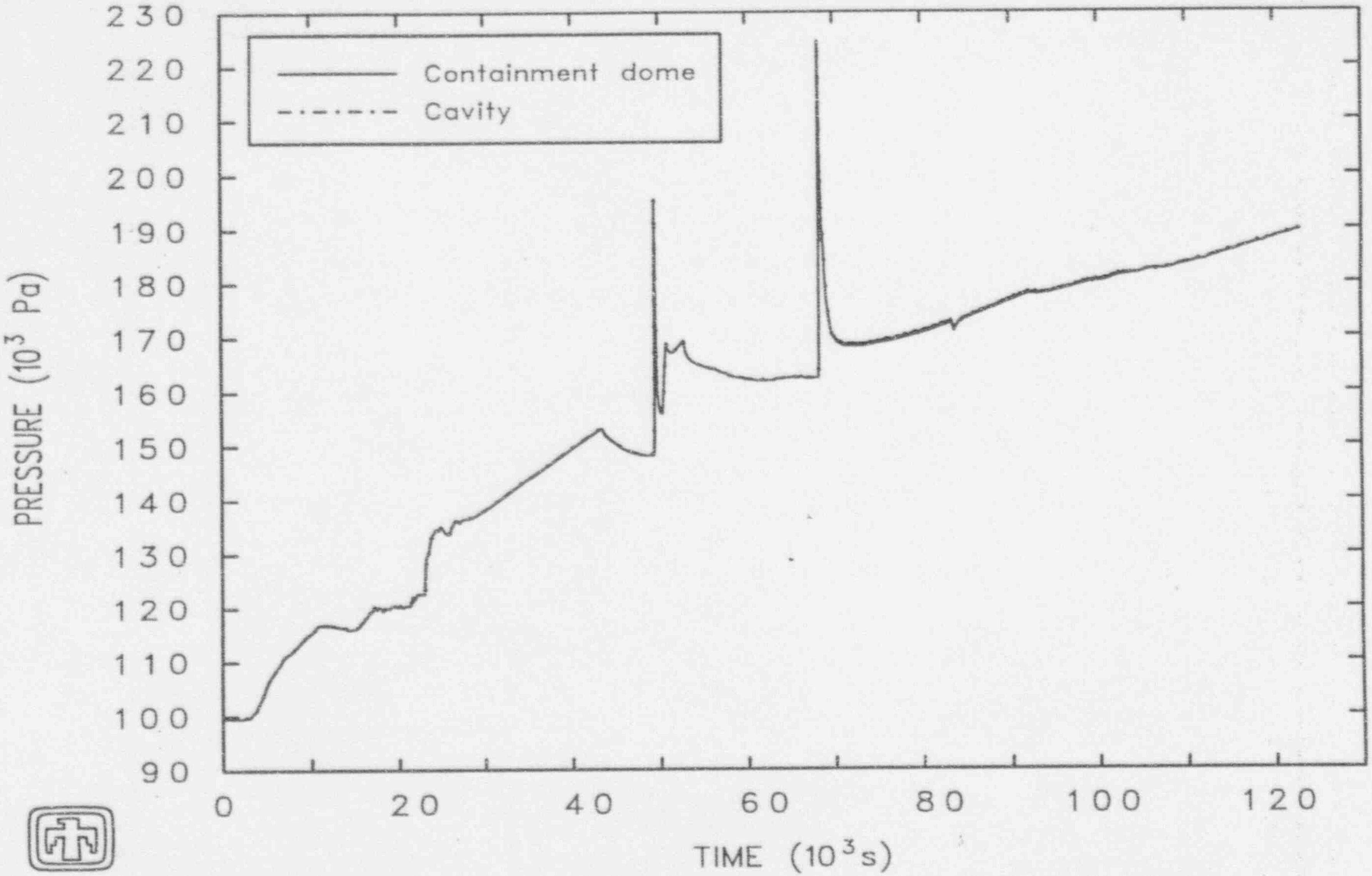


SRRYP6 – L2CASE3 – SHUTDOWN + 6.89 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 15:26:35 MELCOR


CVH-MASS.1.10
LIQUID IN CAVITY (Mg)
TIME (10^3s)
0
5
10
15
20
25
30
35
40
45
Pool mass
SRRYP6 - L2CASE3 - SHUTDOWN + 6.89 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 15:26:35 MELCOR



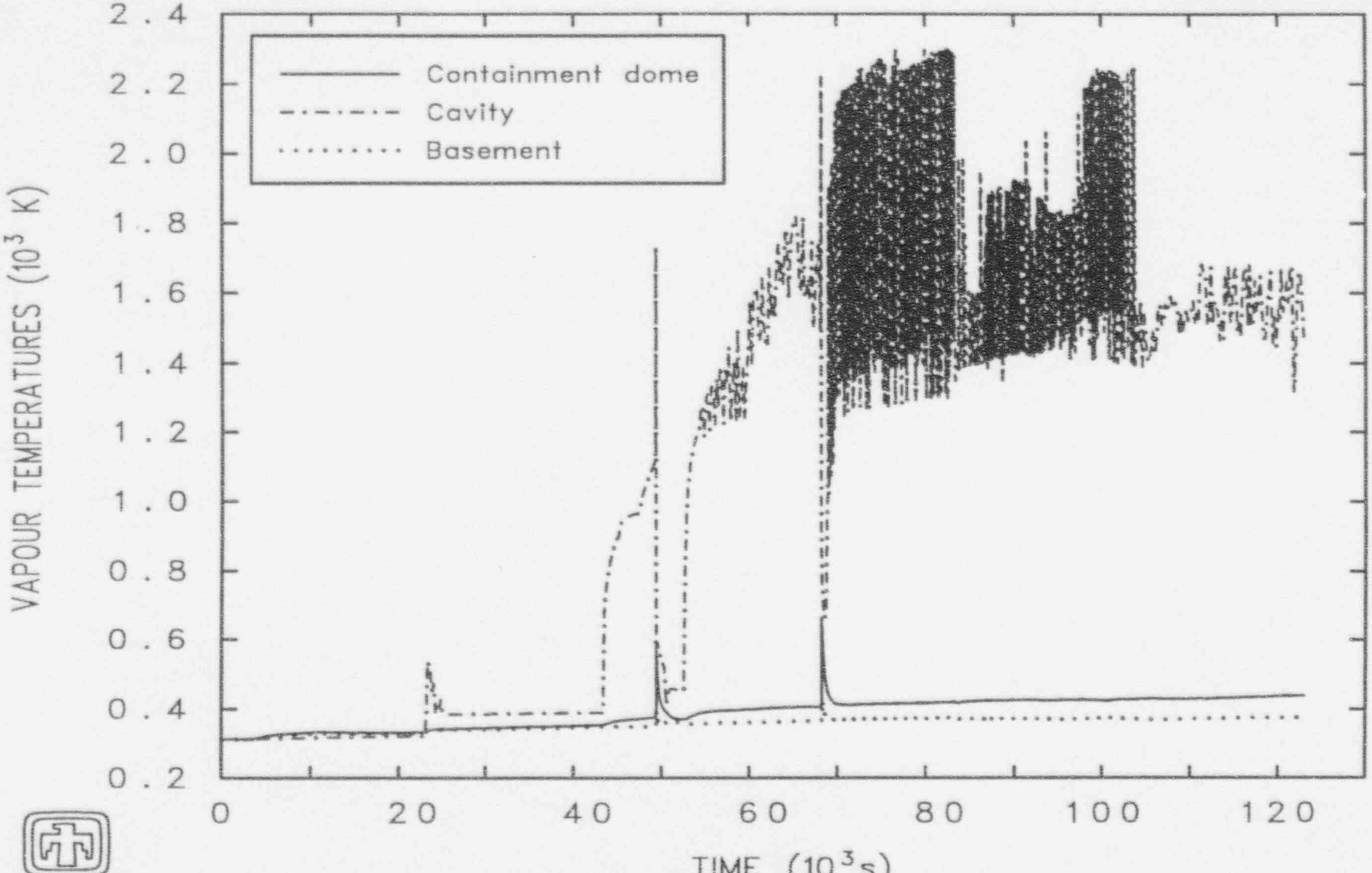
SRRYP6 - L2CASE3 - SHUTDOWN + 6.89 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 15:26:35 MELCOR



SRRYP6 - L2CASE3 - SHUTDOWN + 6.89 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 15:26:35 MELCOR

Fig. 2-17

18



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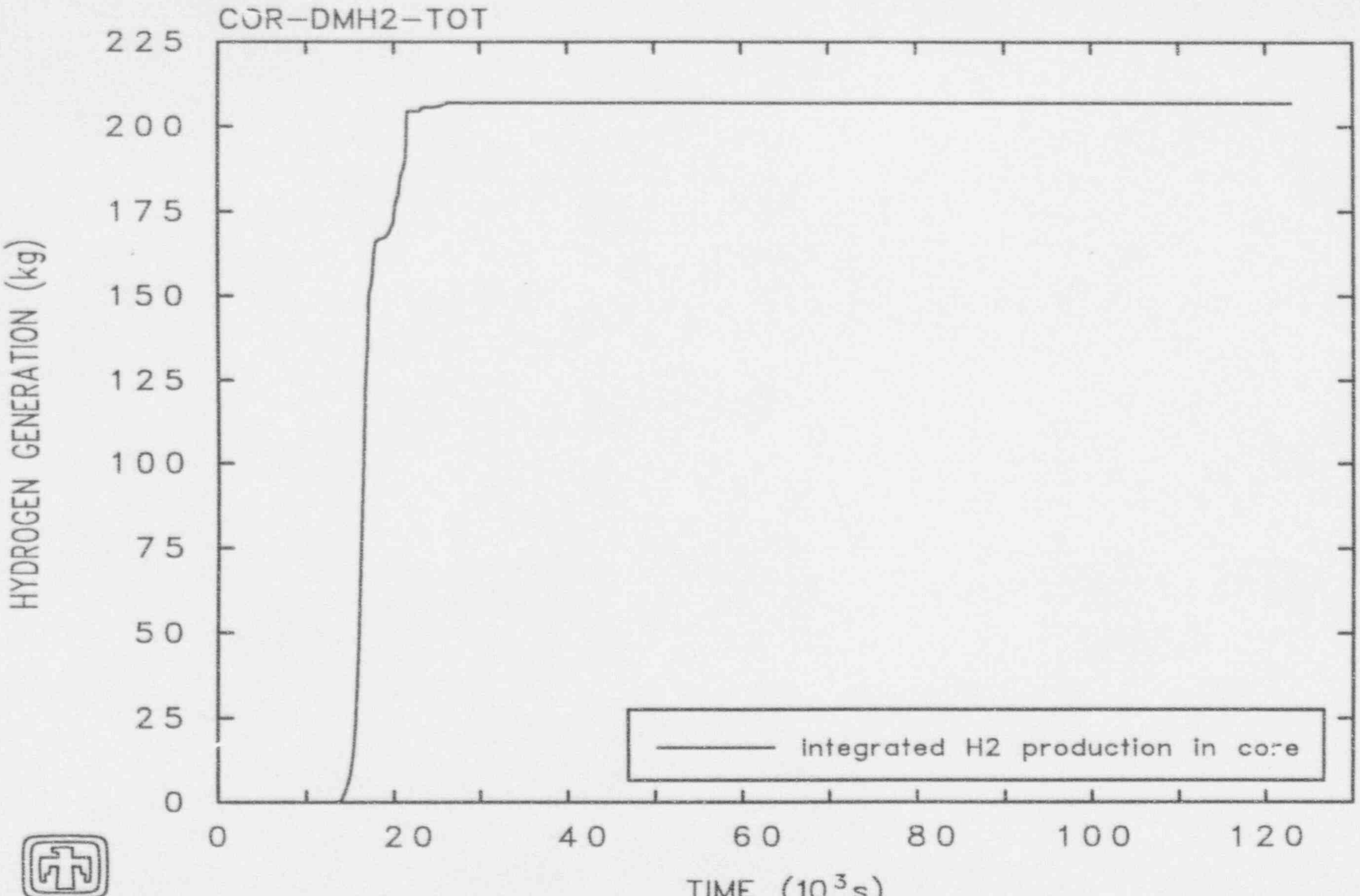
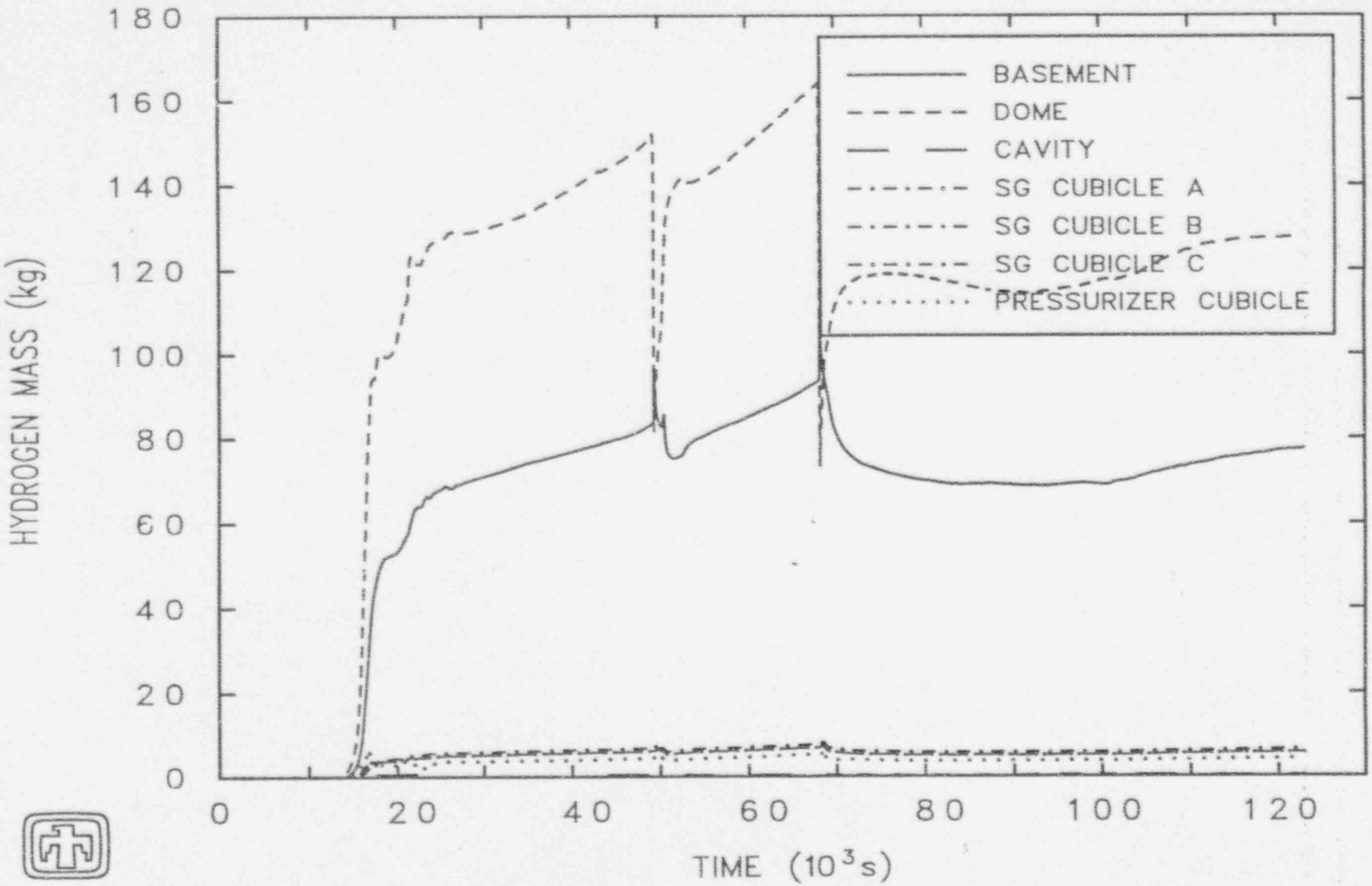
SRRYL2CNM 8/05/ 15:26:35 MELCOR

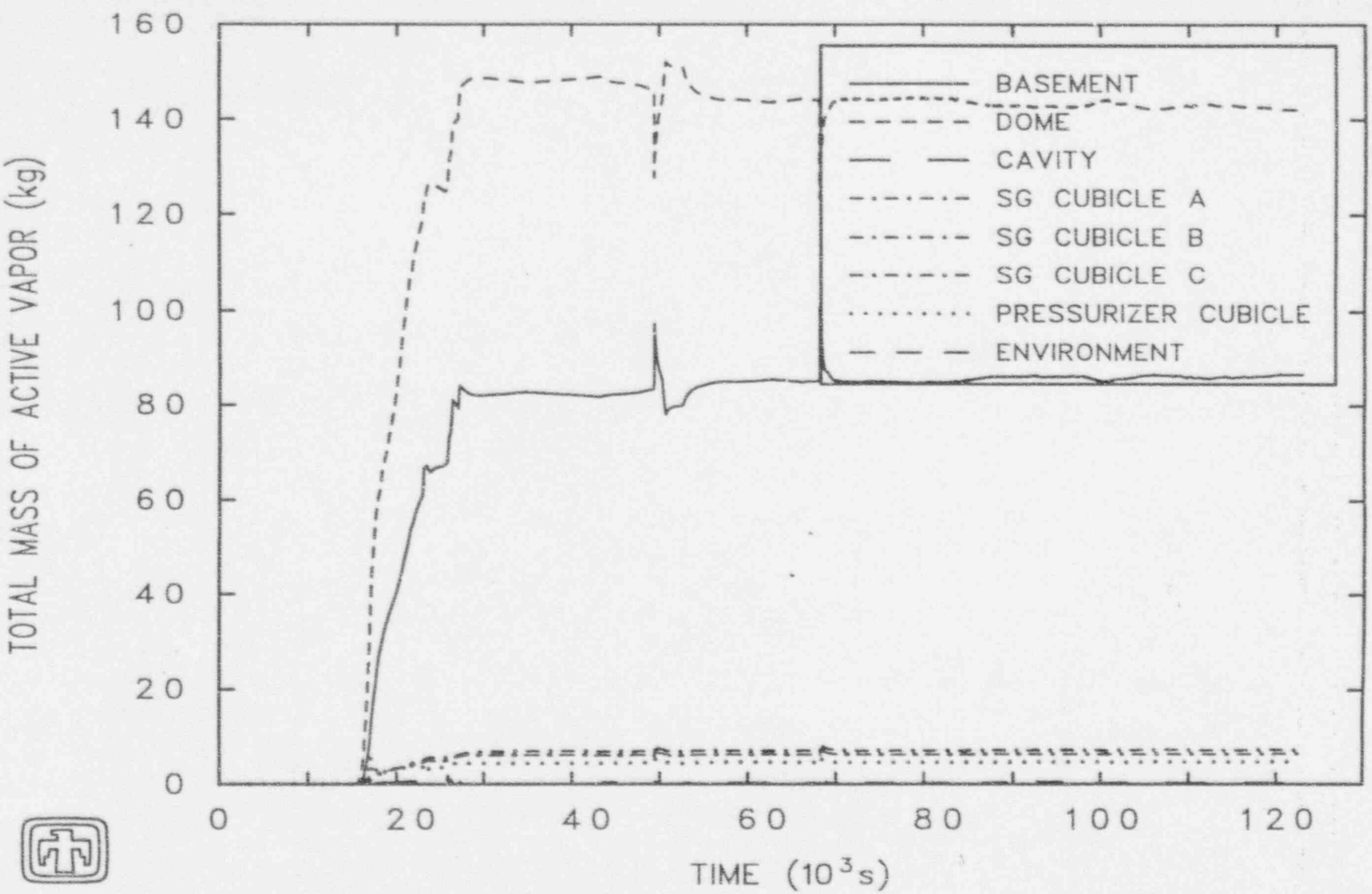


SRRYP6 - L2CASE3 - SHUTDOWN + 6.89 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 15:26:35 MELCOR



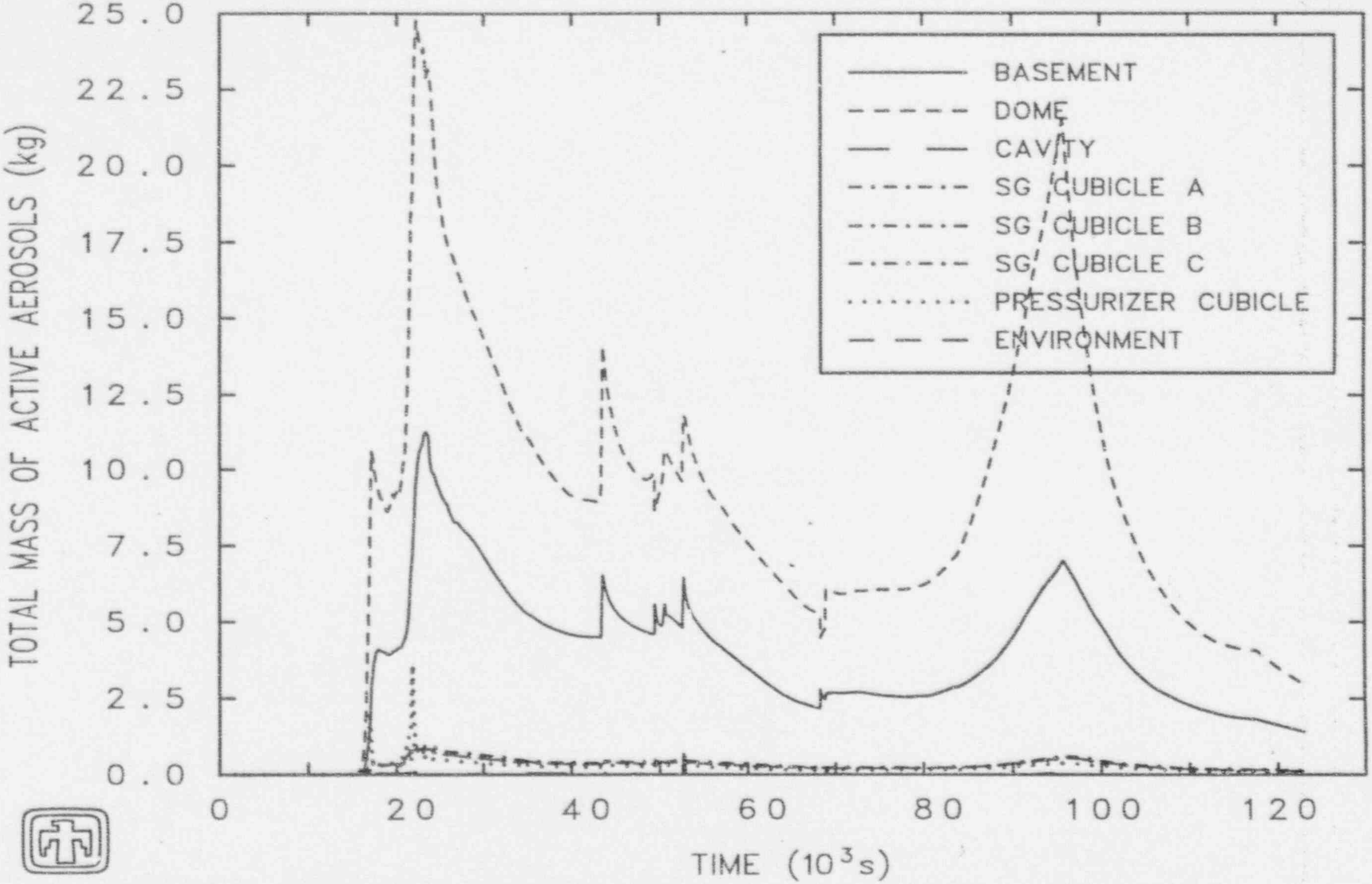
SRRYP6 – L2CASE3 – SHUTDOWN + 6.89 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 15:26:35 MELCOR

FIG 2-15



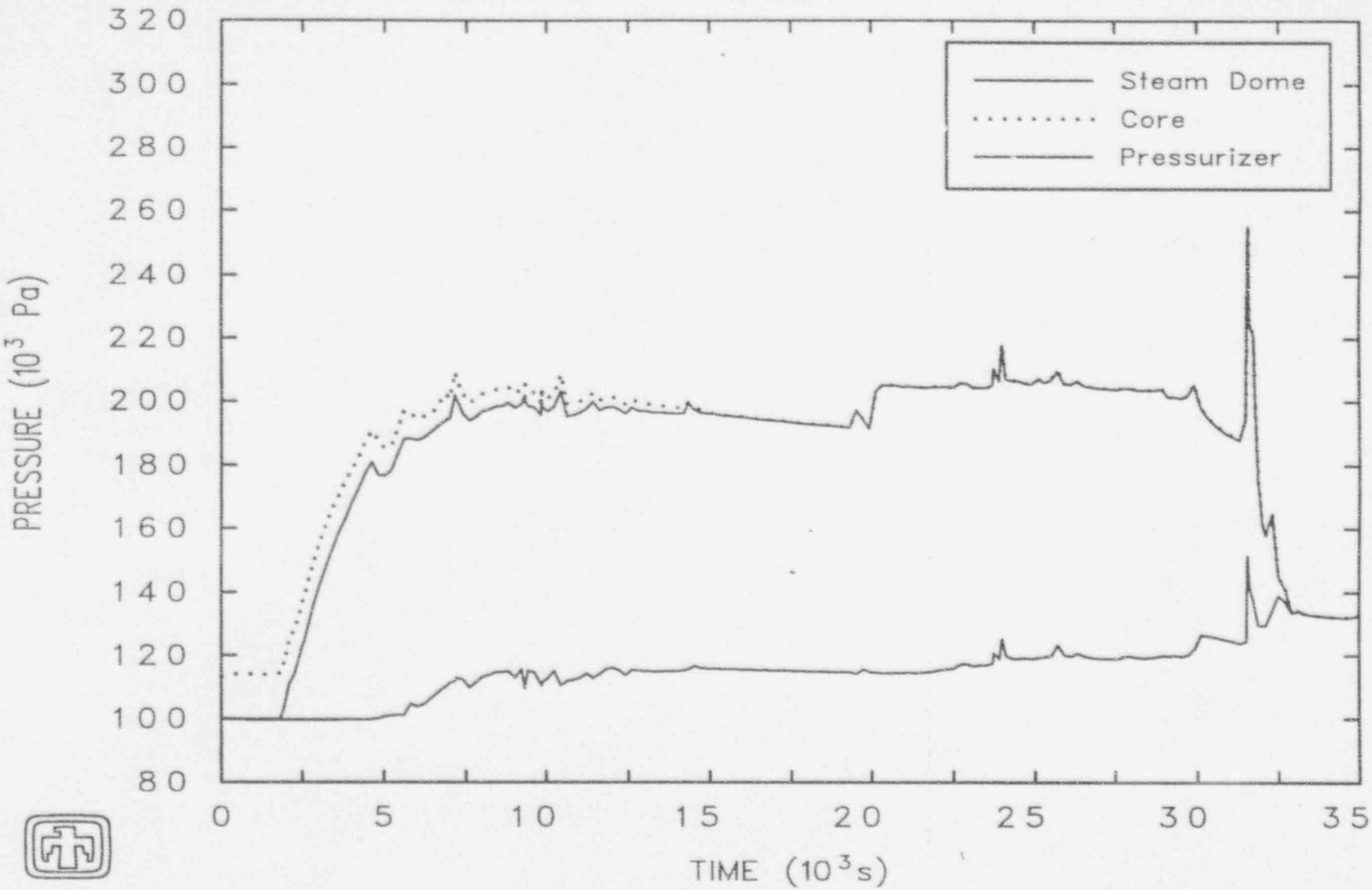
SRRYP6 - L2CASE3 - SHUTDOWN + 6.89 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 15:26:35 MELCOR

FIG 2-20

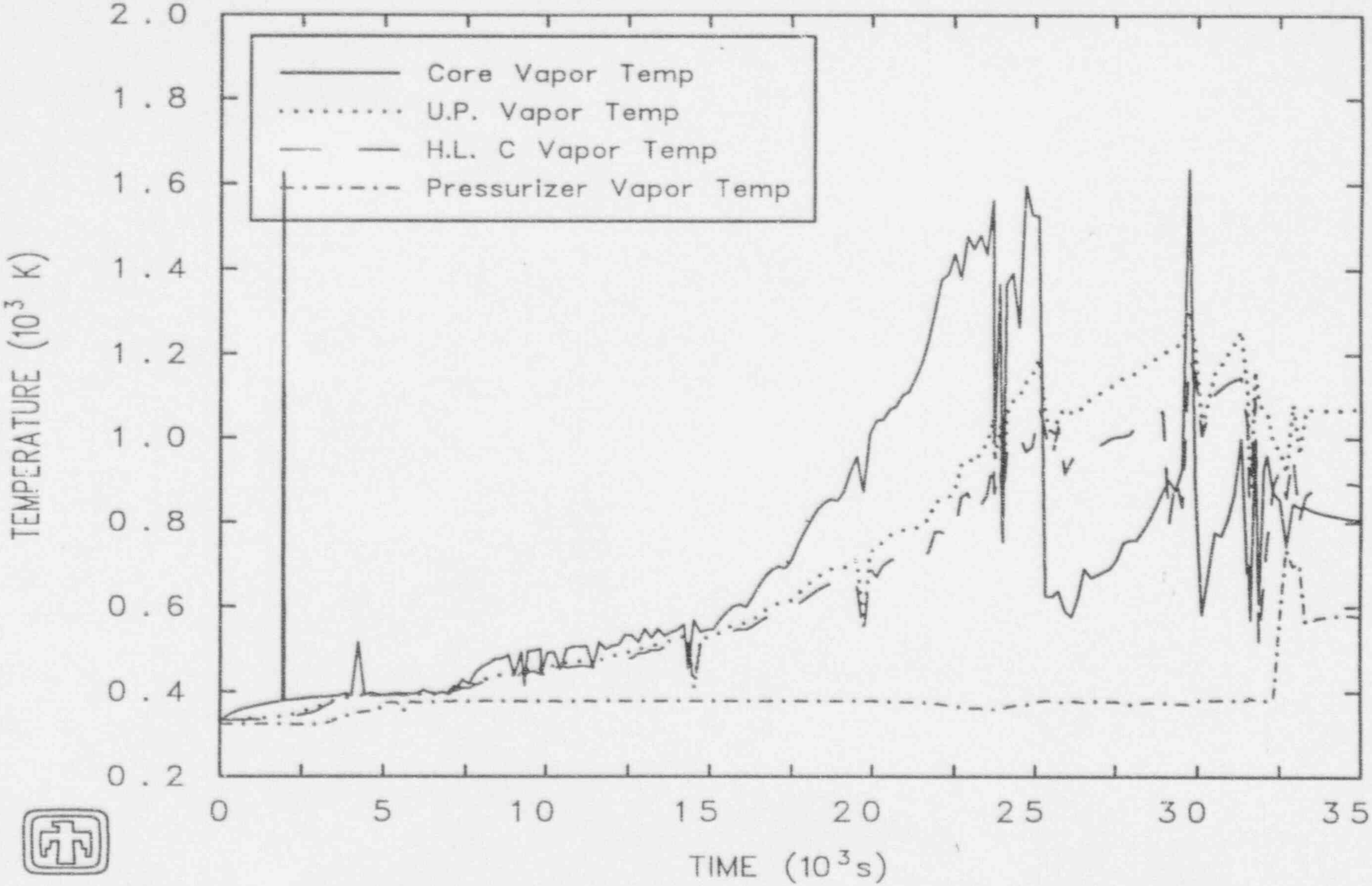


SRRYP6 – L2CASE3 – SHUTDOWN + 6.89 DAYS CNTMNT CLOSED
SRRYL2CNM 8/05/ 15:26:35 MELCOR

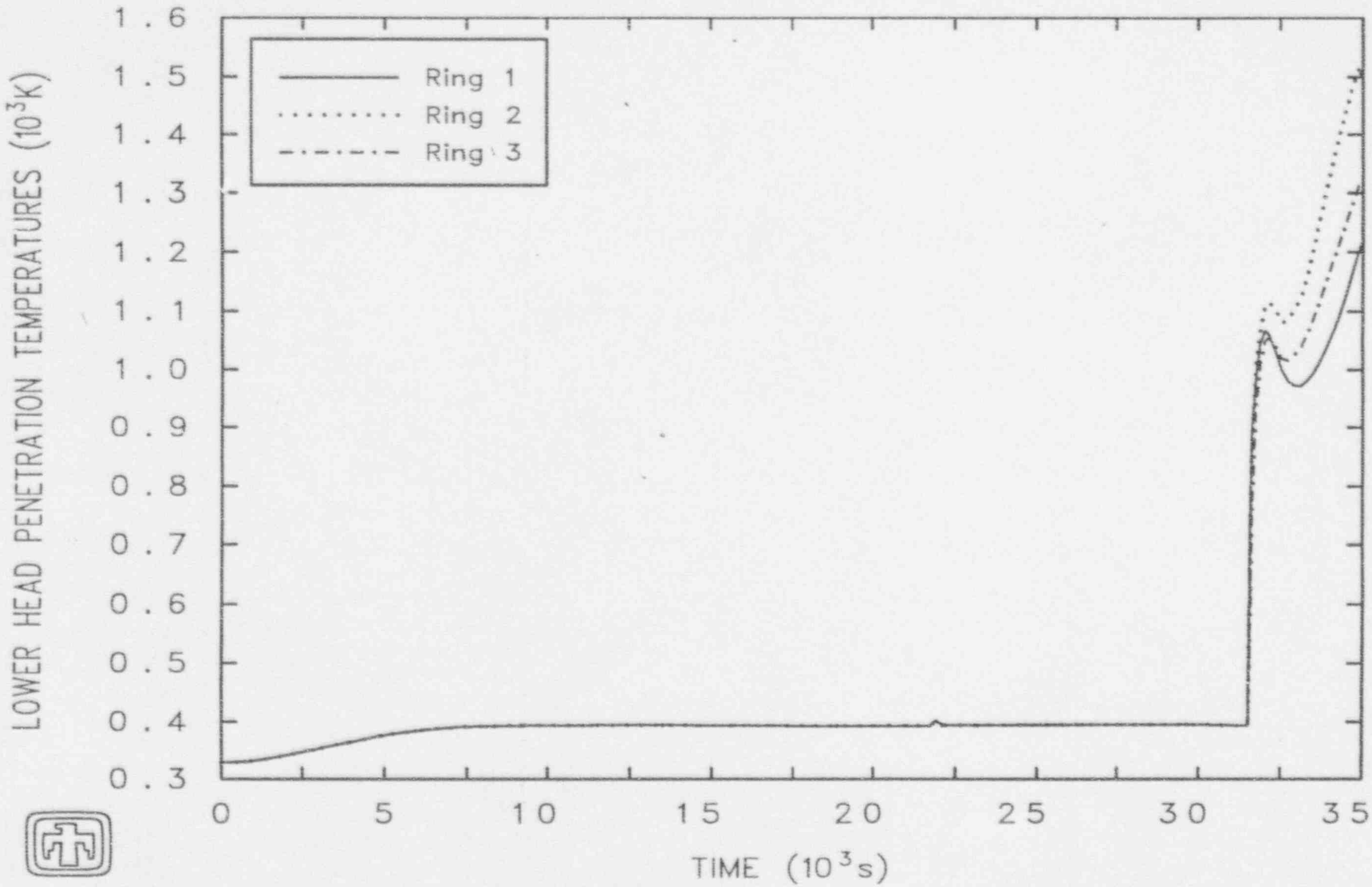
FIG 2-14



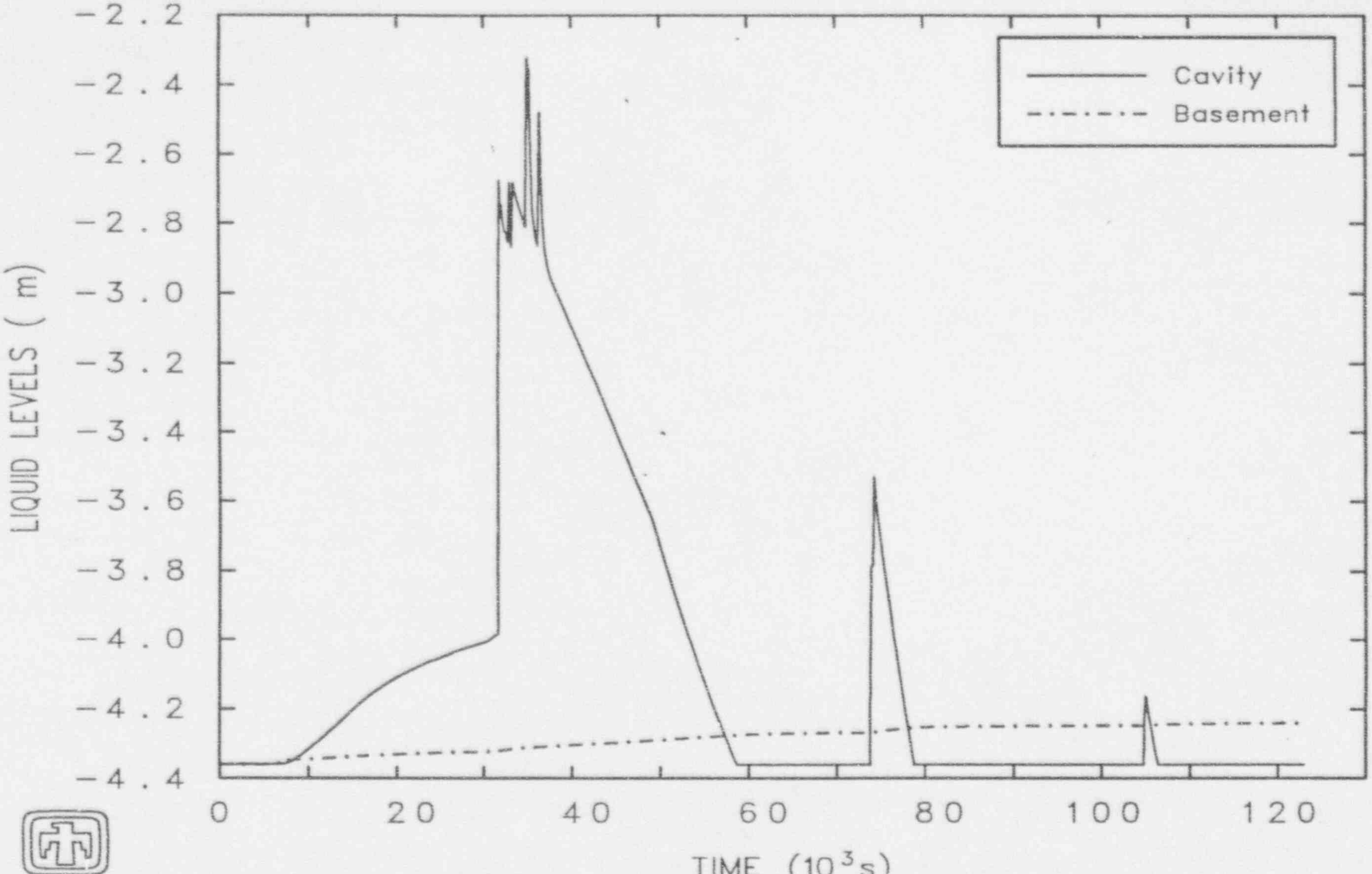
SRRYP6 - L2CASE4 - SHUTDOWN + 15 DAYS CNTMNT CLOSED
SRRYL2CNM 8/09/ 13:31:32 MELCOR



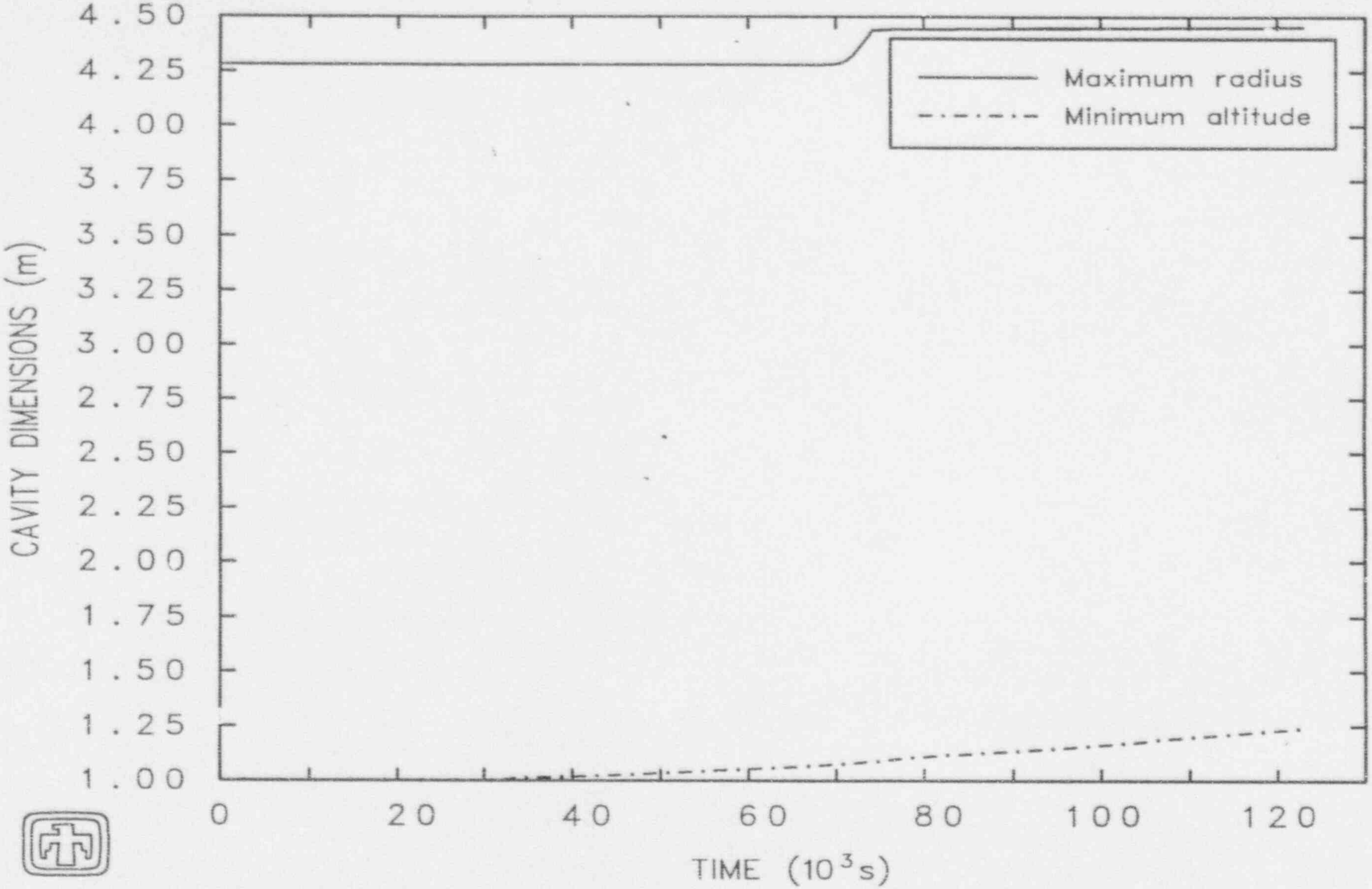
SRRYP6 - L2CASE4 - SHUTDOWN + 15 DAYS CNTMNT CLOSED
SRRYL2CNM 8/09/ 13:31:32 MELCOR



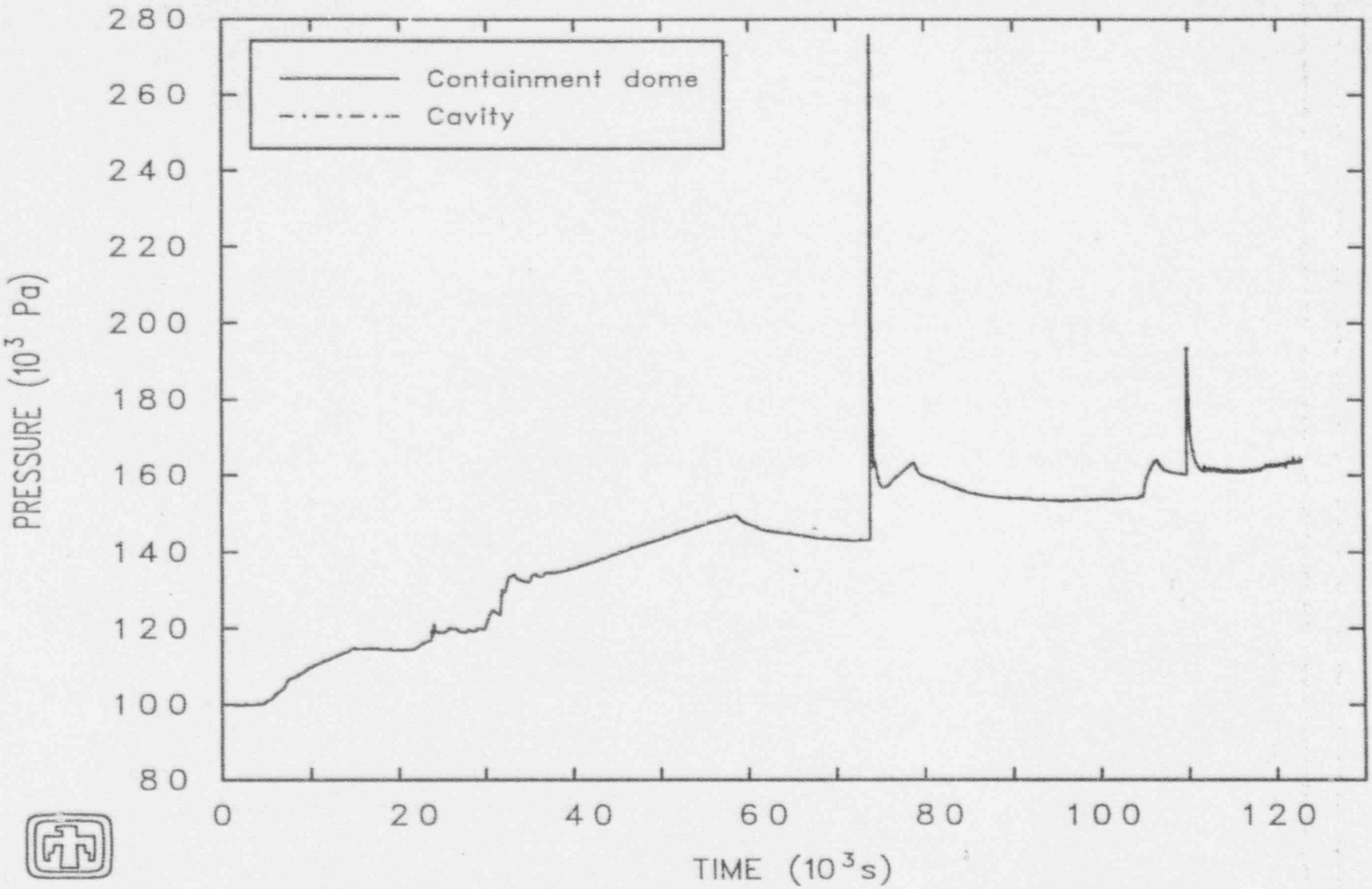
SRRYP6 - L2CASE4 - SHUTDOWN + 15 DAYS CNTMNT CLOSED
SRRYL2CNM 8/09/ 13:31:32 MELCOR



SRRYP6 - L2CASE4 - SHUTDOWN + 15 DAYS CNTMNT CLOSED
SRRYL2CNM 8/09/ 13:31:32 MELCOR

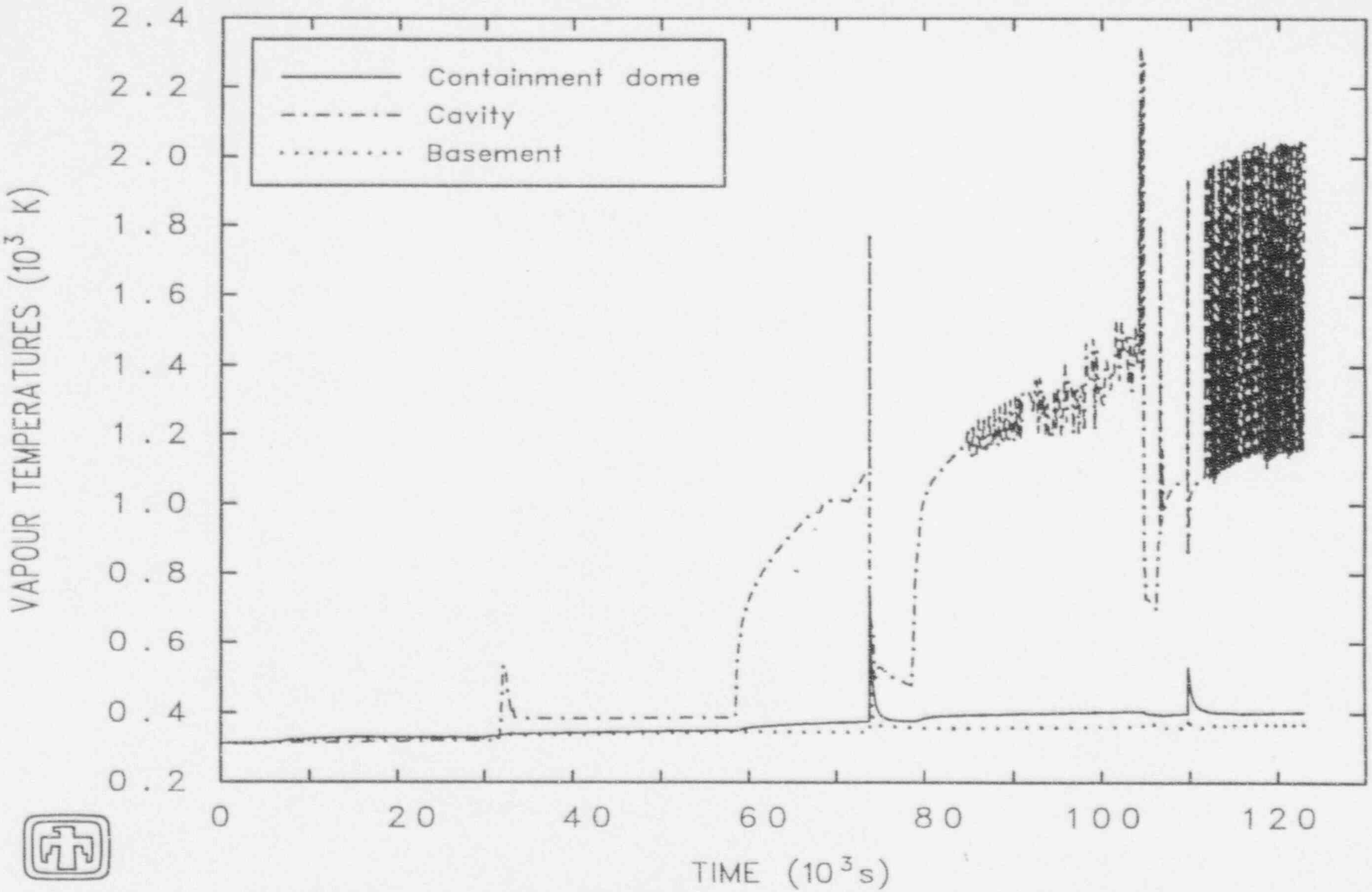


SRRYP6 - L2CASE4 - SHUTDOWN + 15 DAYS CNTMNT CLOSED
SRRYL2CNM 8/09/ 13:31:32 MELCOR

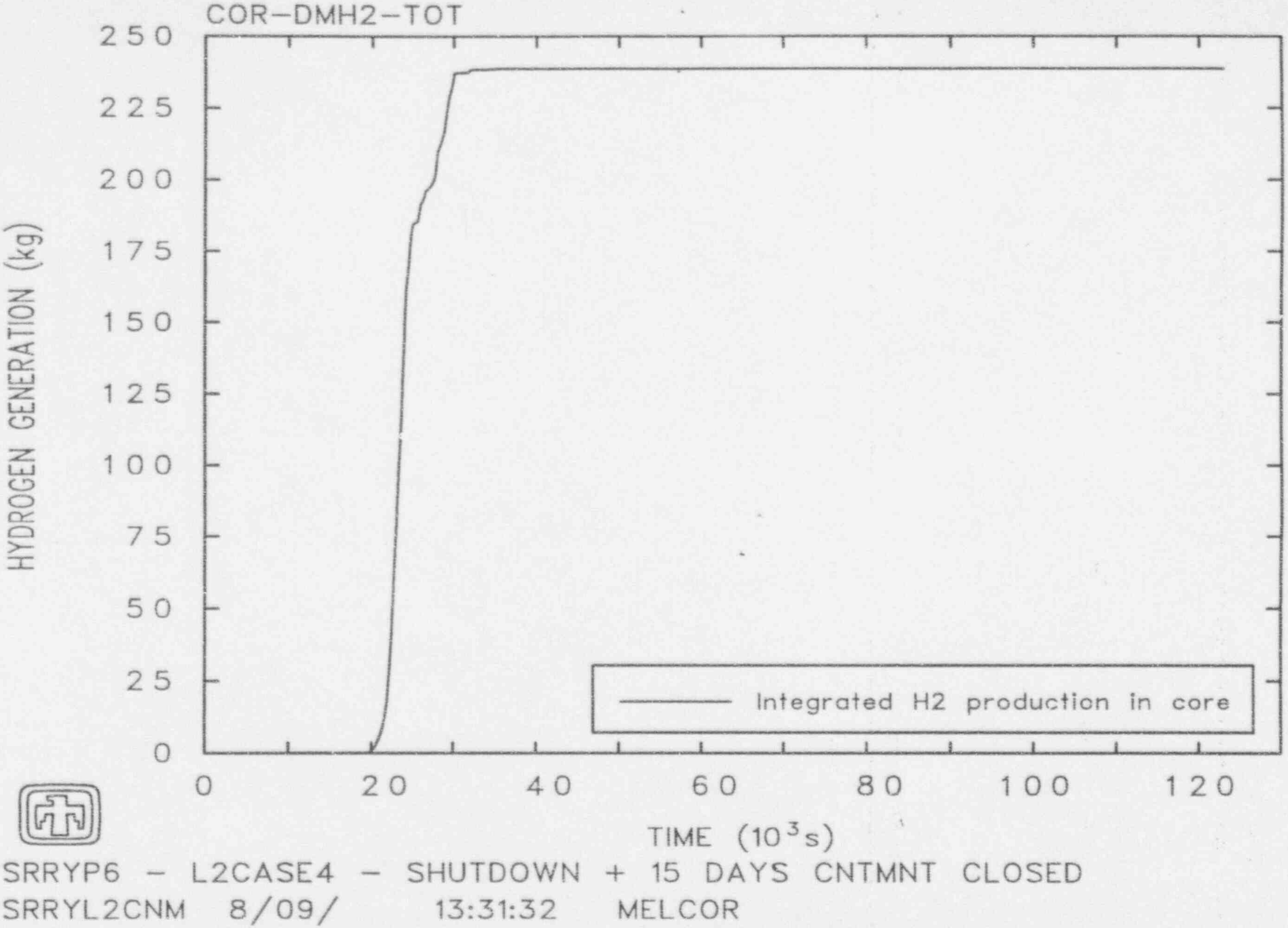


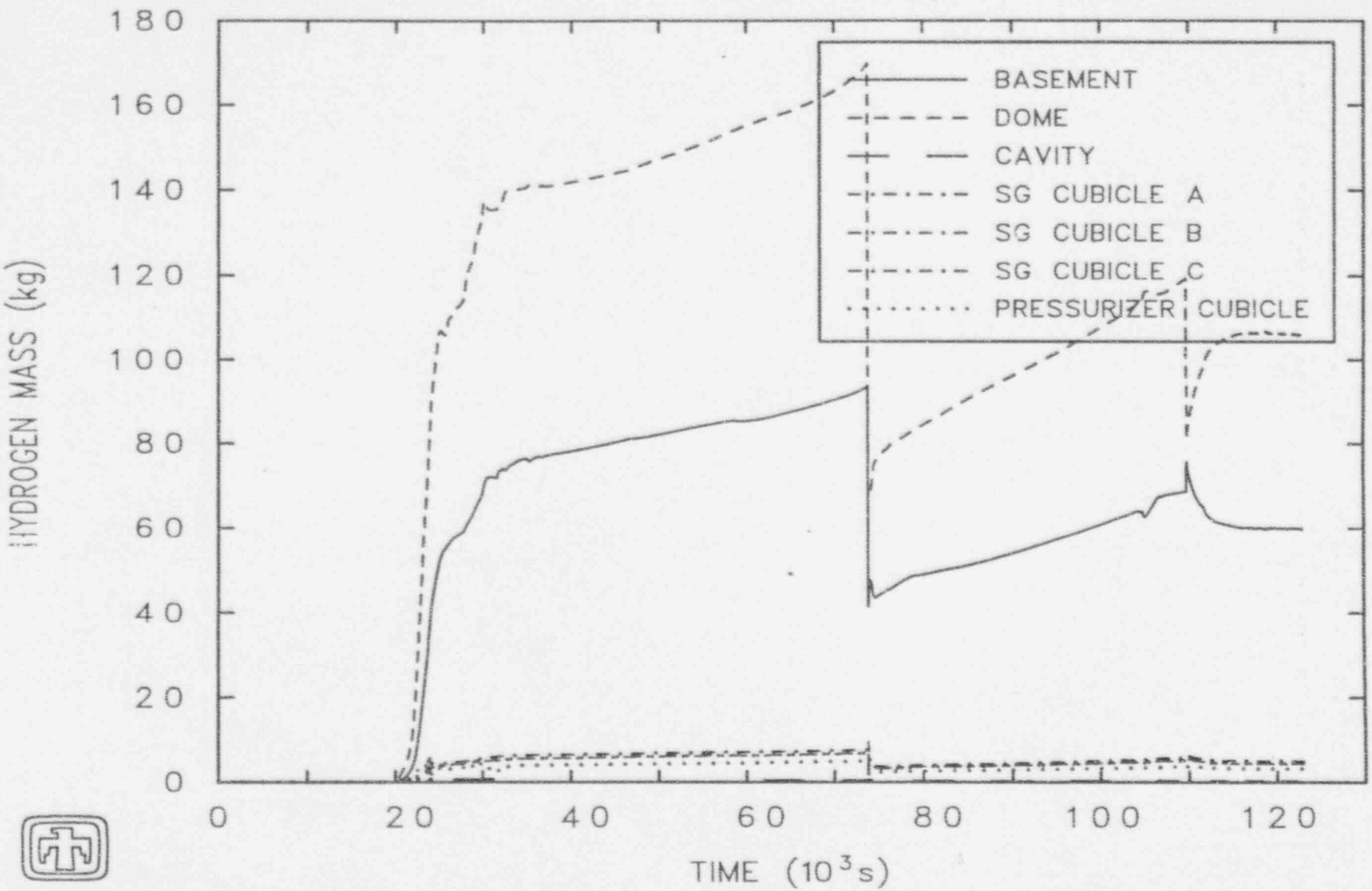
SRRYP6 - L2CASE4 - SHUTDOWN + 15 DAYS CNTMNT CLOSED
SRRYL2CNM 8/09/ 13:31:32 MELCOR

Fig 2.18



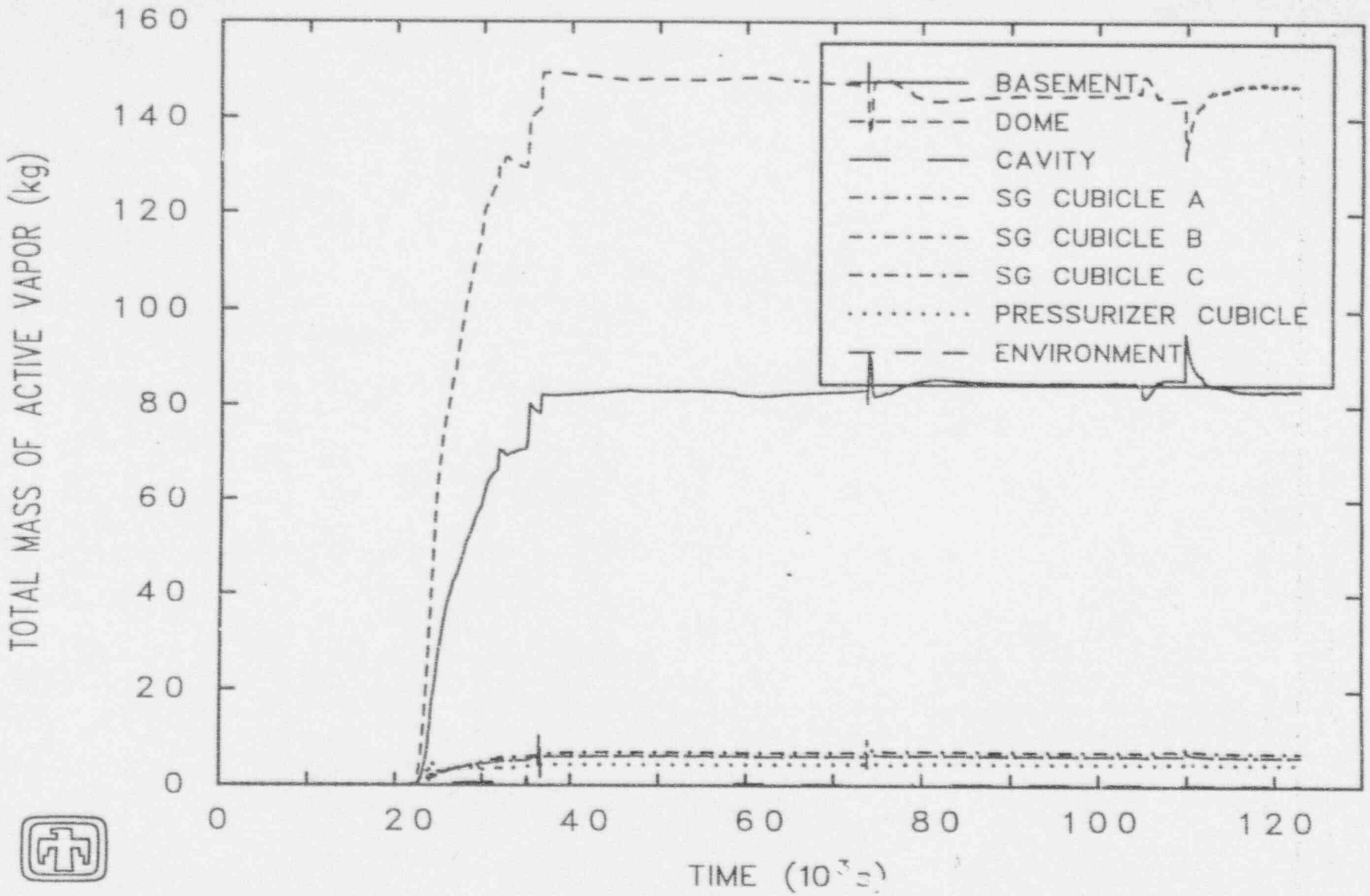
SRRYP6 - L2CASE4 - SHUTDOWN + 15 DAYS CNTMNT CLOSED
SRRYL2CNM 8/09/ 13:31:32 MELCOR


COR-DMH2-TOT
HYDROGEN GENERATION (kg)
250
225
200
175
150
125
100
75
50
25
0
0
20
40
60
80
100
120
TIME (10³s)
Integrated H2 production in core
SRRYP6 - L2CASE4 - SHUTDOWN + 15 DAYS CNTMNT CLOSED
SRRYL2CNM 8/09/ 13:31:32 MELCOR



SRRYP6 – L2CASE4 – SHUTDOWN + 15 DAYS CNTMNT CLOSED
SRRYL2CNM 8/09/ 13:31:32 MELCOR

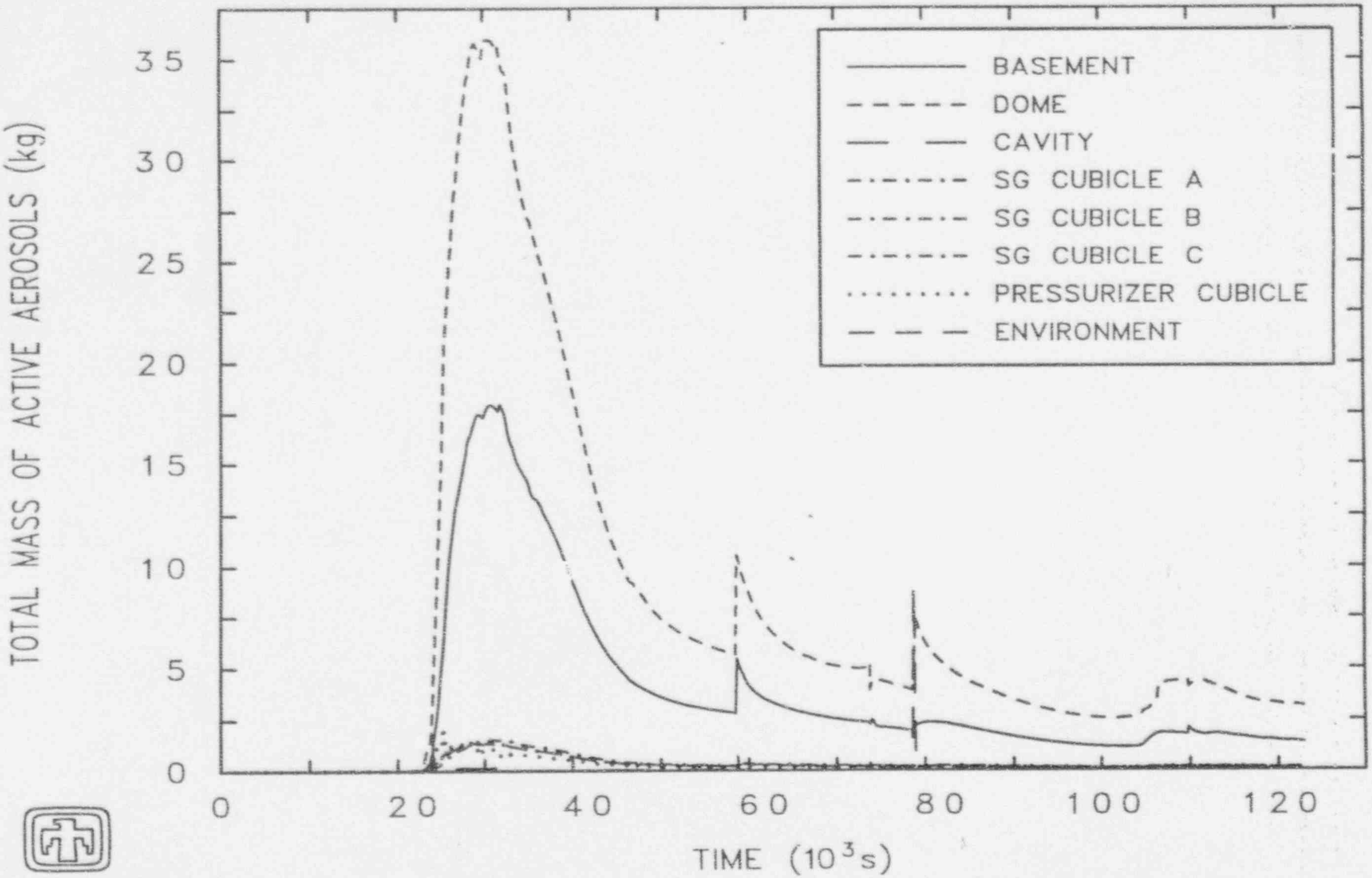
FIG 2-16



SRRYP6 – L2CASE4 – SHUTDOWN + 15 DAYS CNTMNT CLOSED
SRRYL2CNM 8/09/ 13:31:32 MELCOR

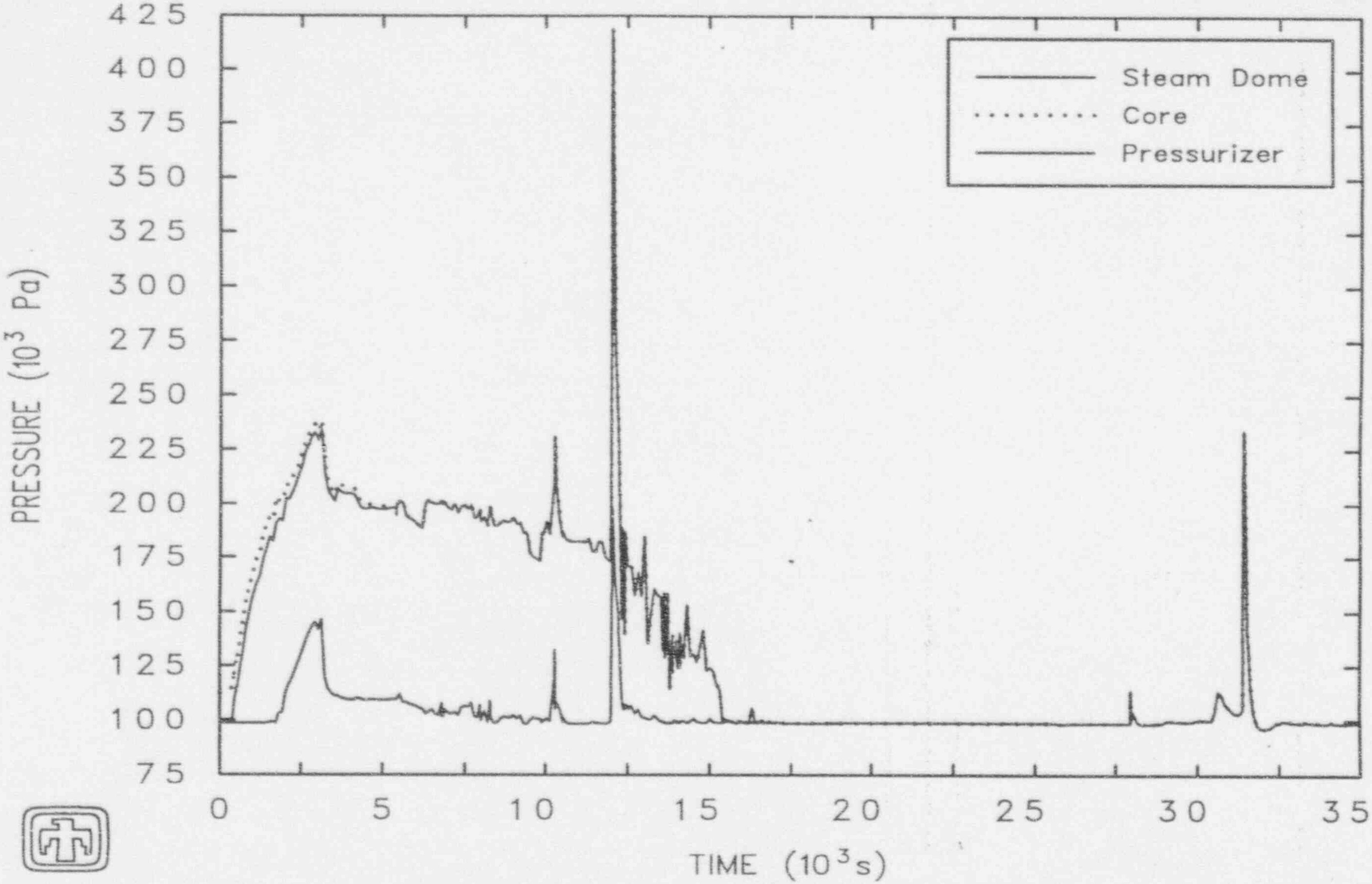
Fig 2-22

37

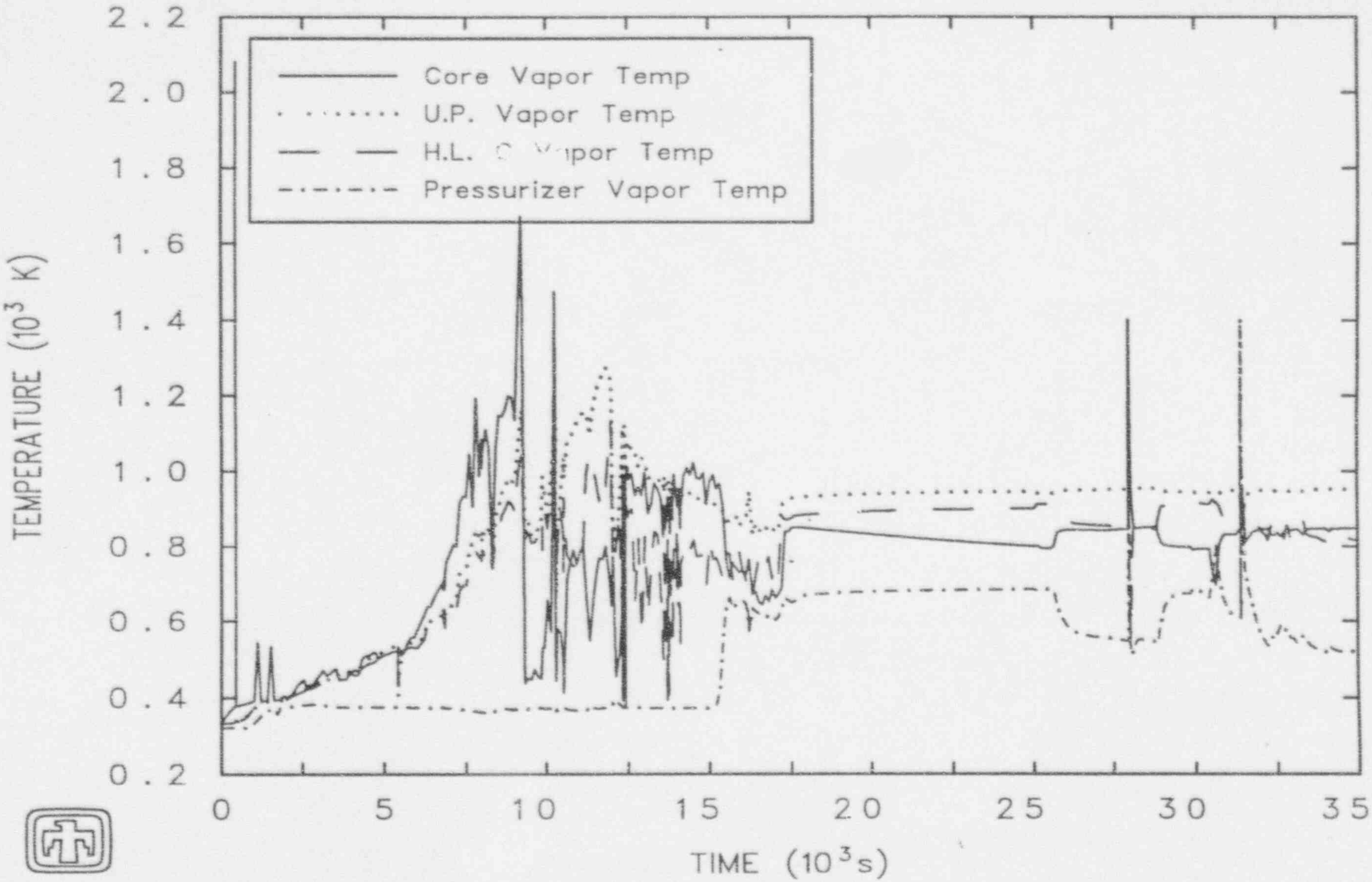


SRRYP6 – L2CASE4 – SHUTDOWN + 15 DAYS CNTMNT CLOSED
SRRYL2CNM 8/09/ 13:31:32 MELCOR

FIG 2-21

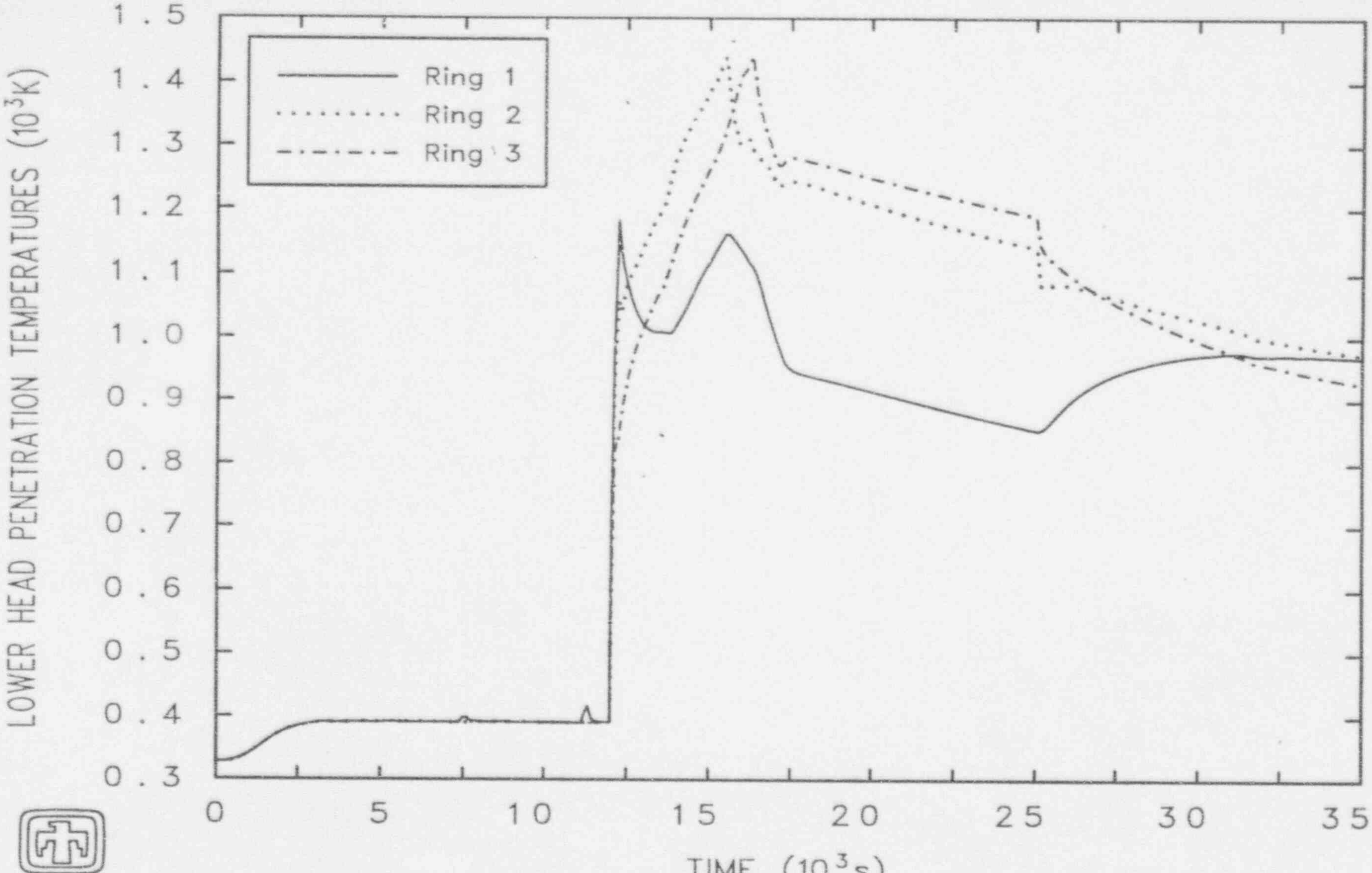


SRRYP6 - L2CASE5 - SHUTDOWN + 1.13 DAYS CNTMNT LEAKS
SRRYL2CNM 8/16/93 09:30:08 MELCOR

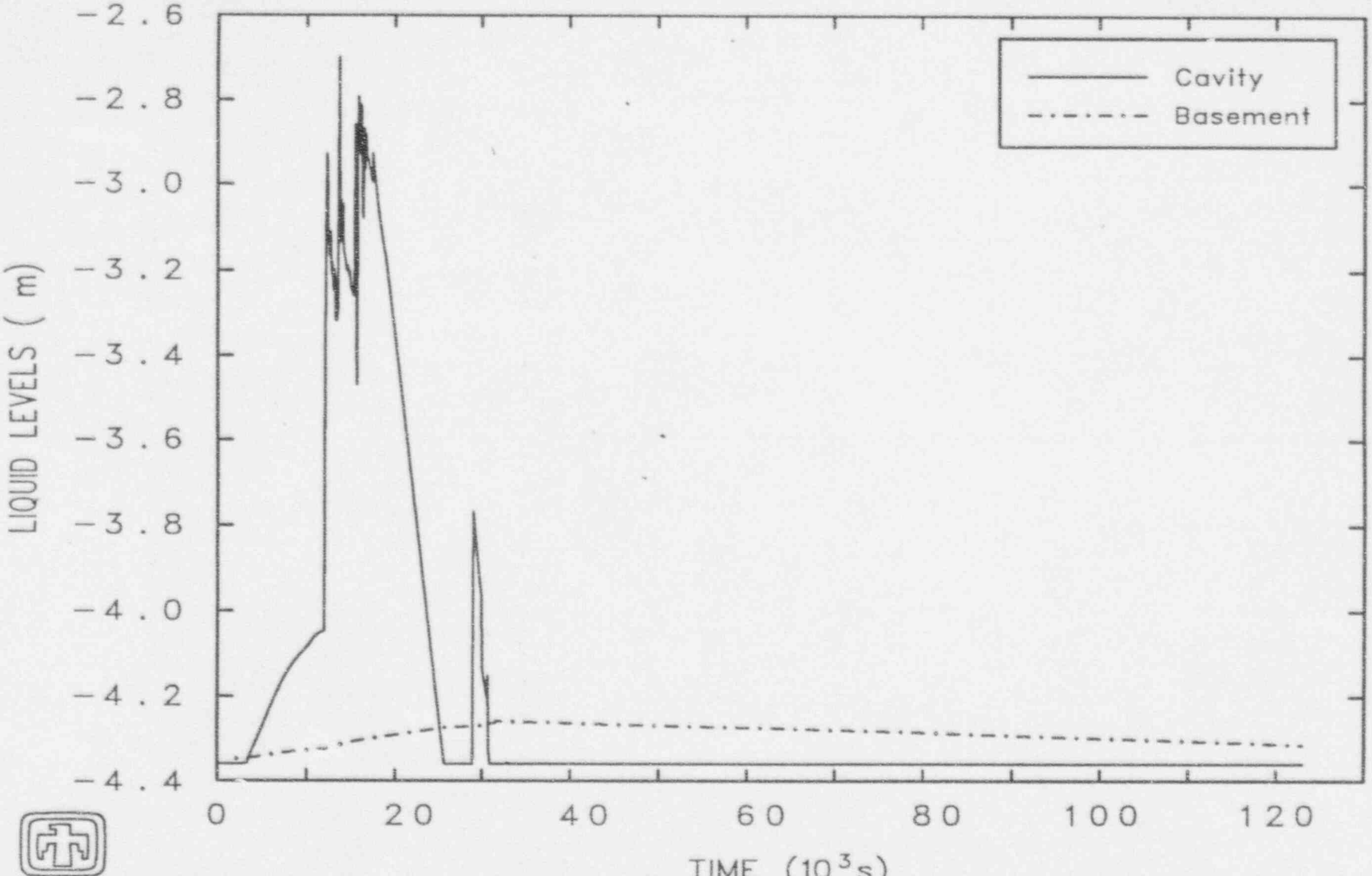


SRRYP6 – L2CASE5 – SHUTDOWN + 1.13 DAYS CNTMNT LEAKS
SRRYL2CNM 8/16/93 09:30:08 MELCOR

56



SRRYP6 – L2CASE5 – SHUTDOWN + 1.13 DAYS CNTMNT LEAKS
SRRYL2CNM 8/16/93 09:30:08 MELCOR



SRRYP6 - L2CASE5 - SHUTDOWN + 1.13 DAYS CNTMNT LEAKS
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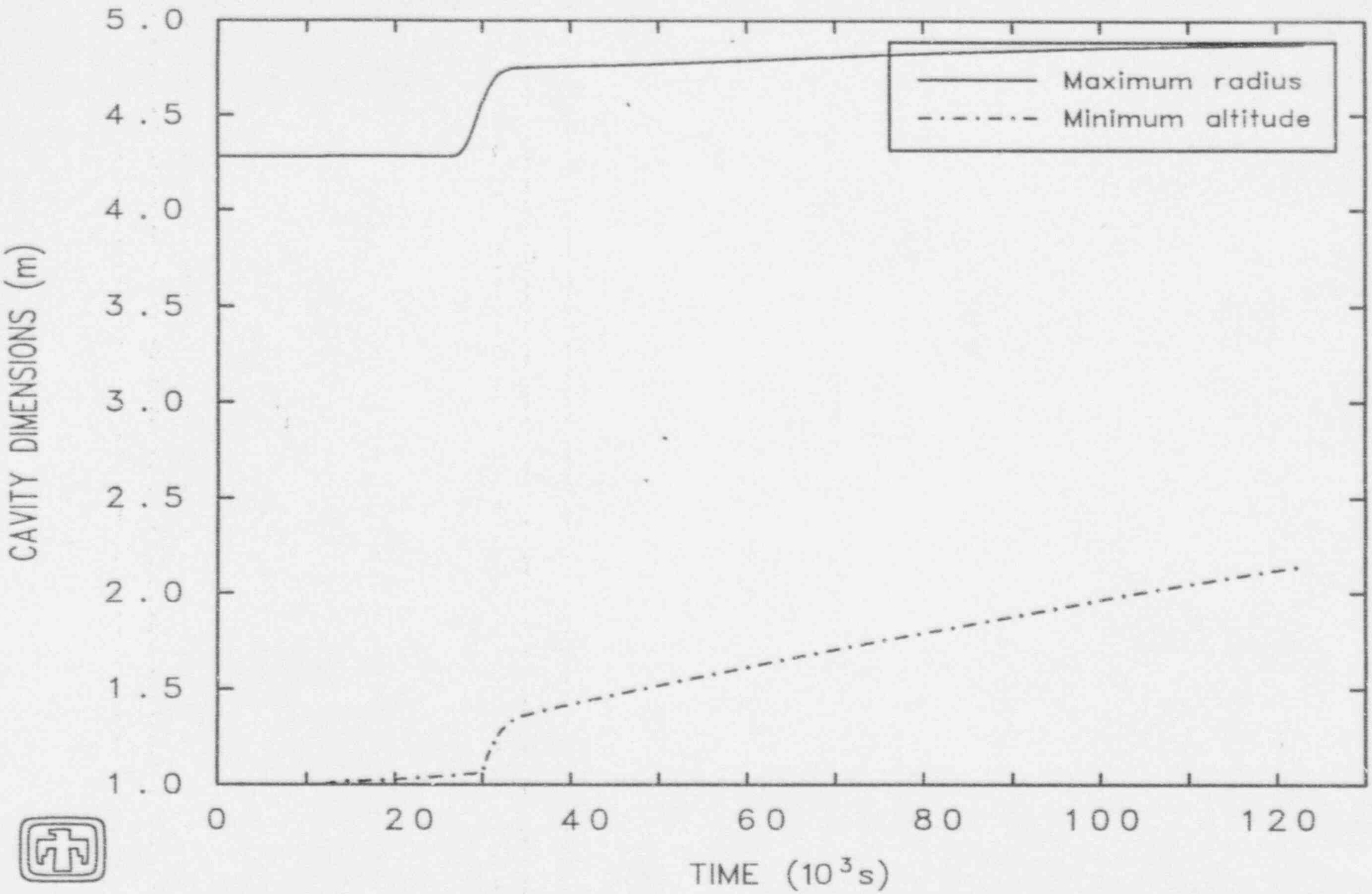
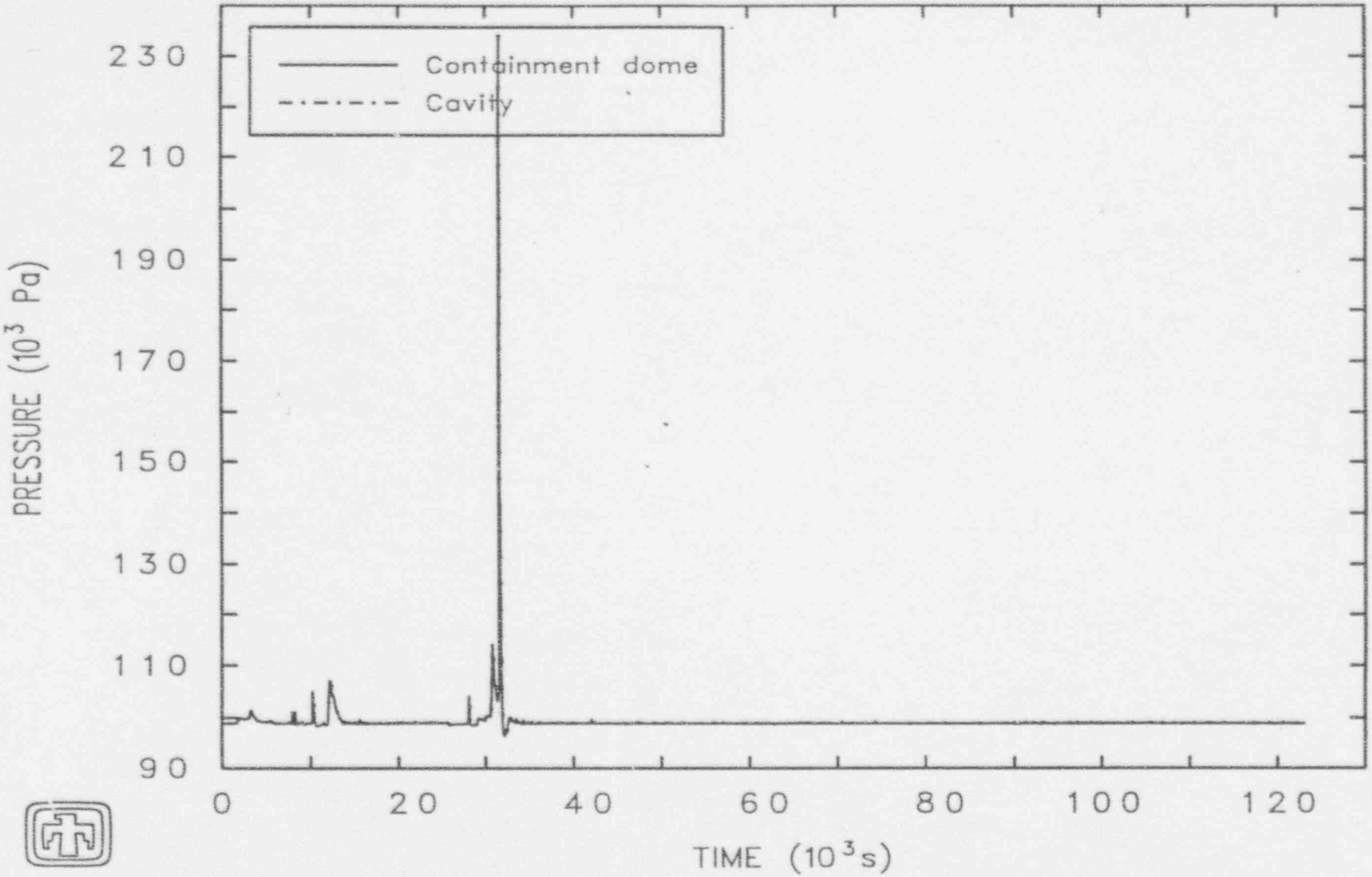


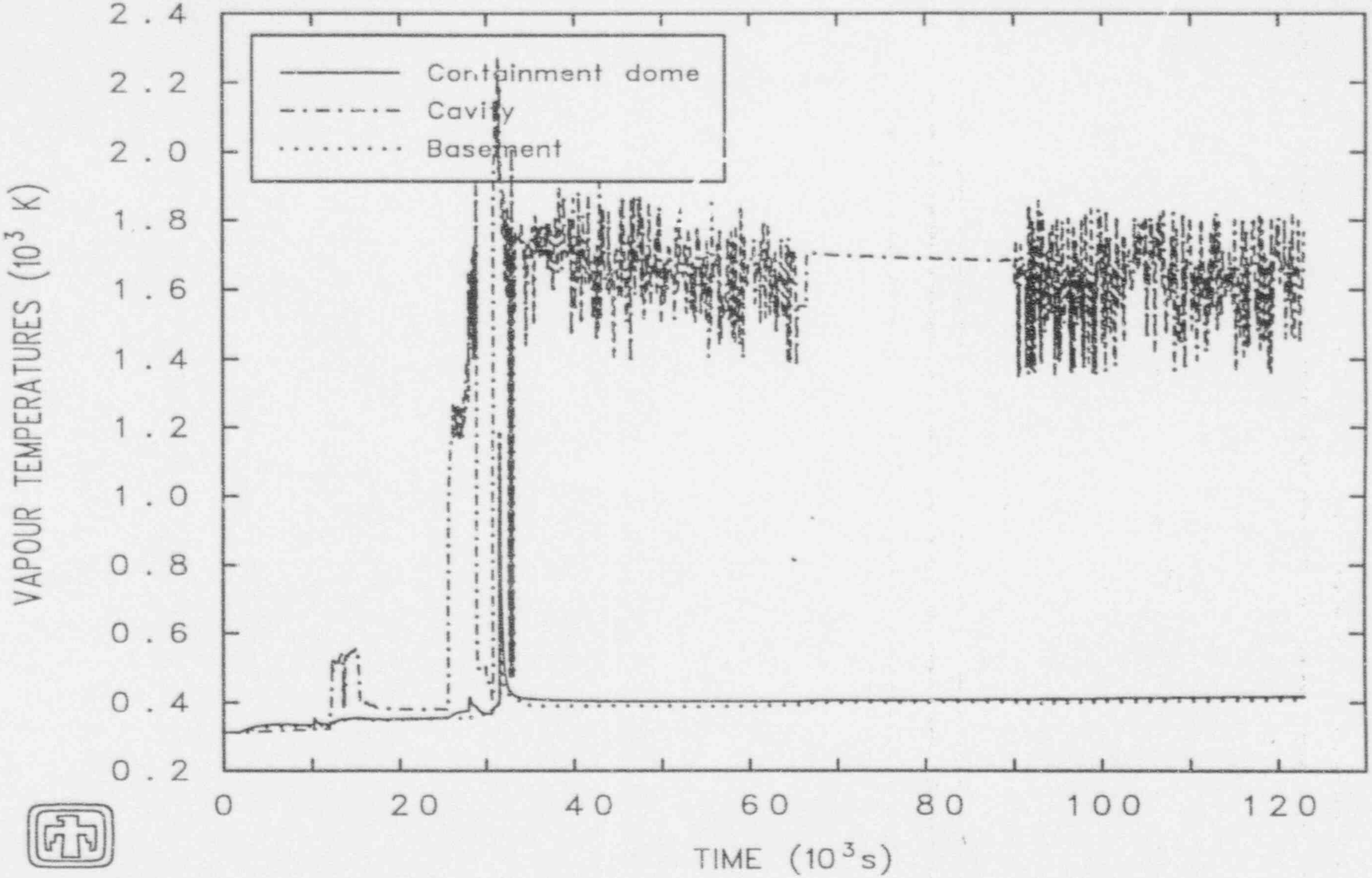
SRRYP6 - L2CASE5 - SHUTDOWN + 1.13 DAYS CNTMNT LEAKS
SRRYL2CNM 8/16/93 09:30:08 MELCOR

FIG 7-20

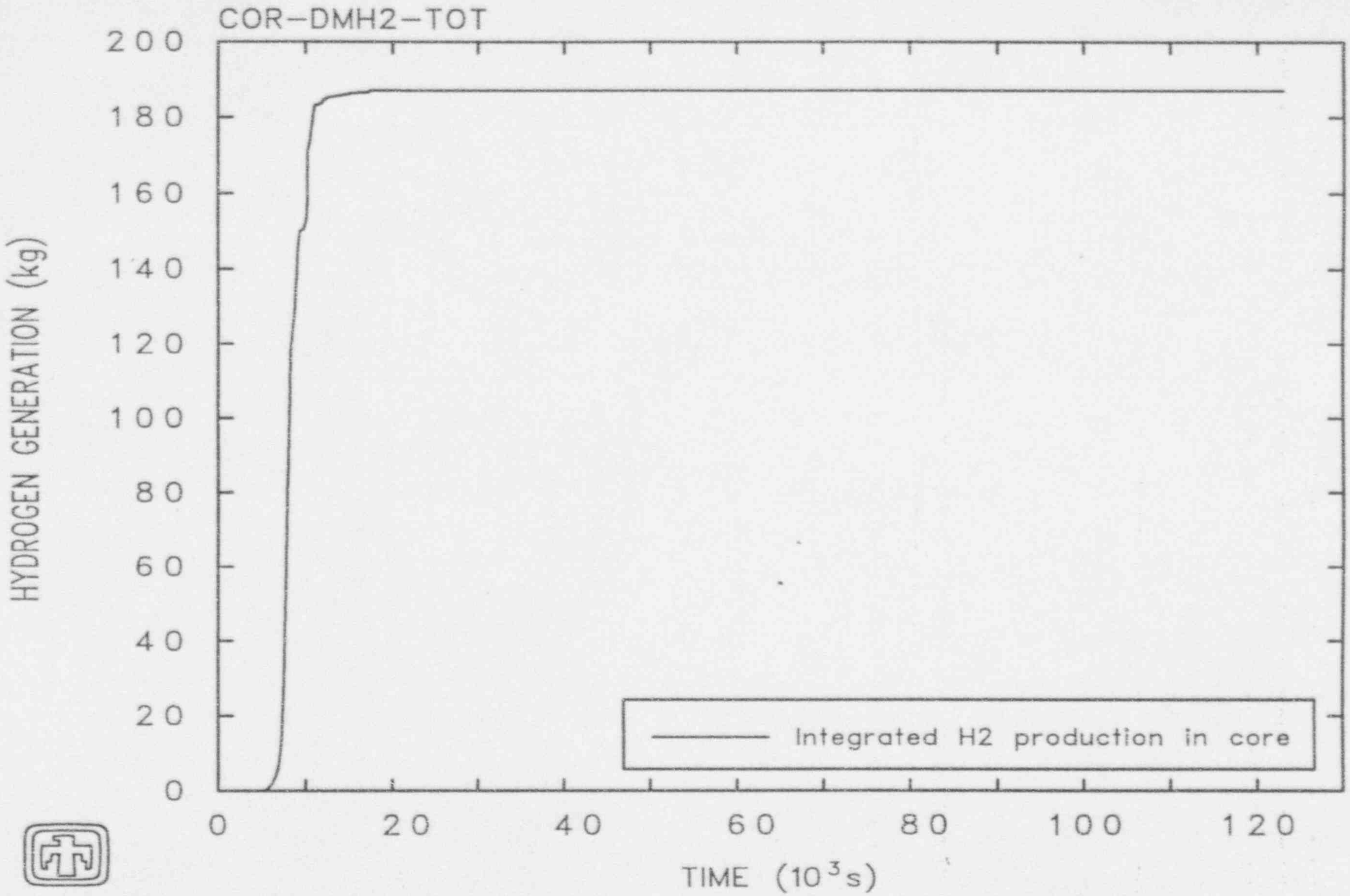


SRRYP6 – L2CASE5 – SHUTDOWN + 1.13 DAYS CNTMNT LEAKS
SRRYL2CNM 8/16/93 09:30:08 MELCOR

Fig 2-27



SRRYP6 - L2CASE5 - SHUTDOWN + 1.13 DAYS CNTMNT LEAKS
SRRYL2CNM 8/16/93 09:30:08 MELCOR



SRRYP6 - L2CASE5 - SHUTDOWN + 1.13 DAYS CNTMNT LEAKS
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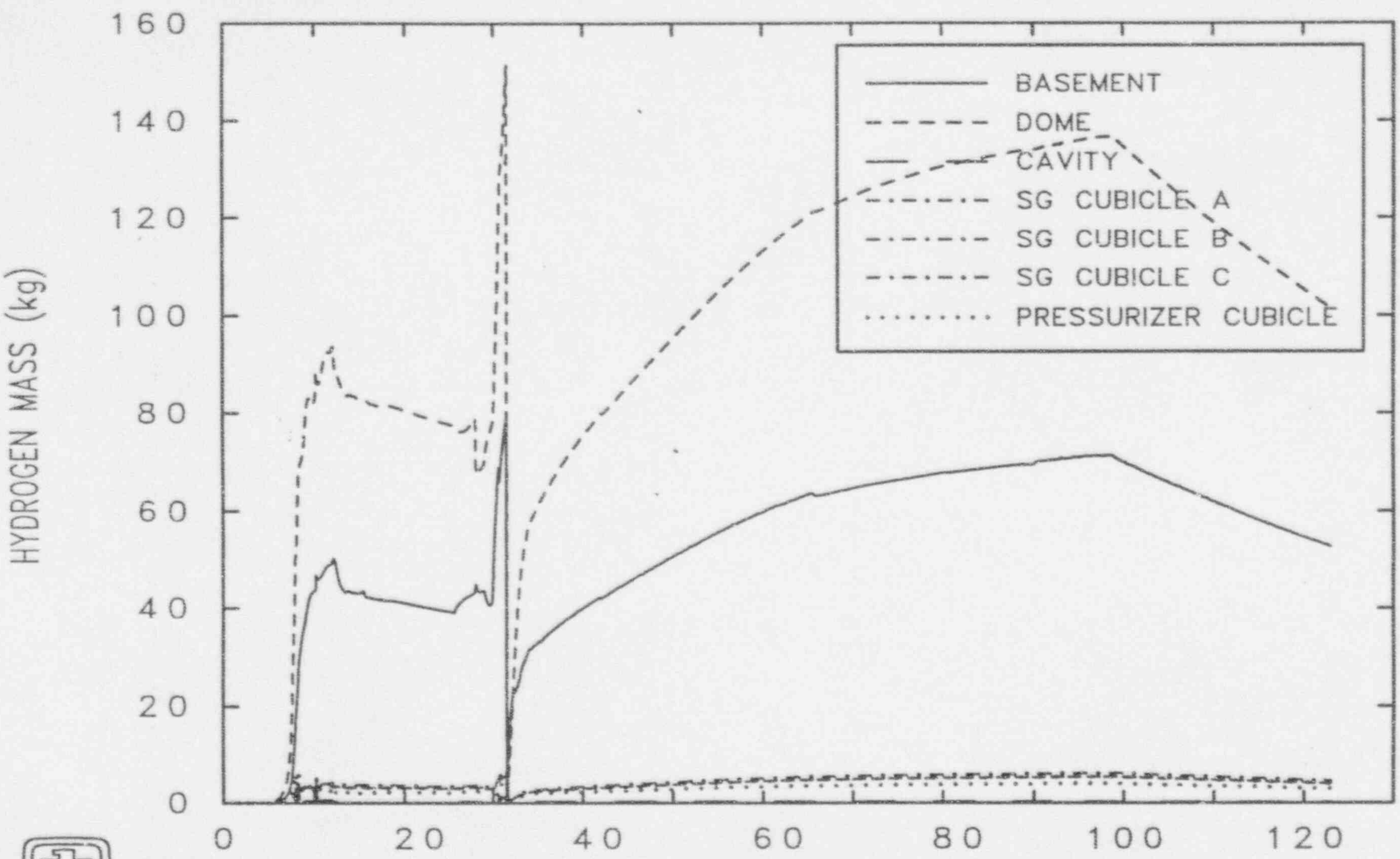
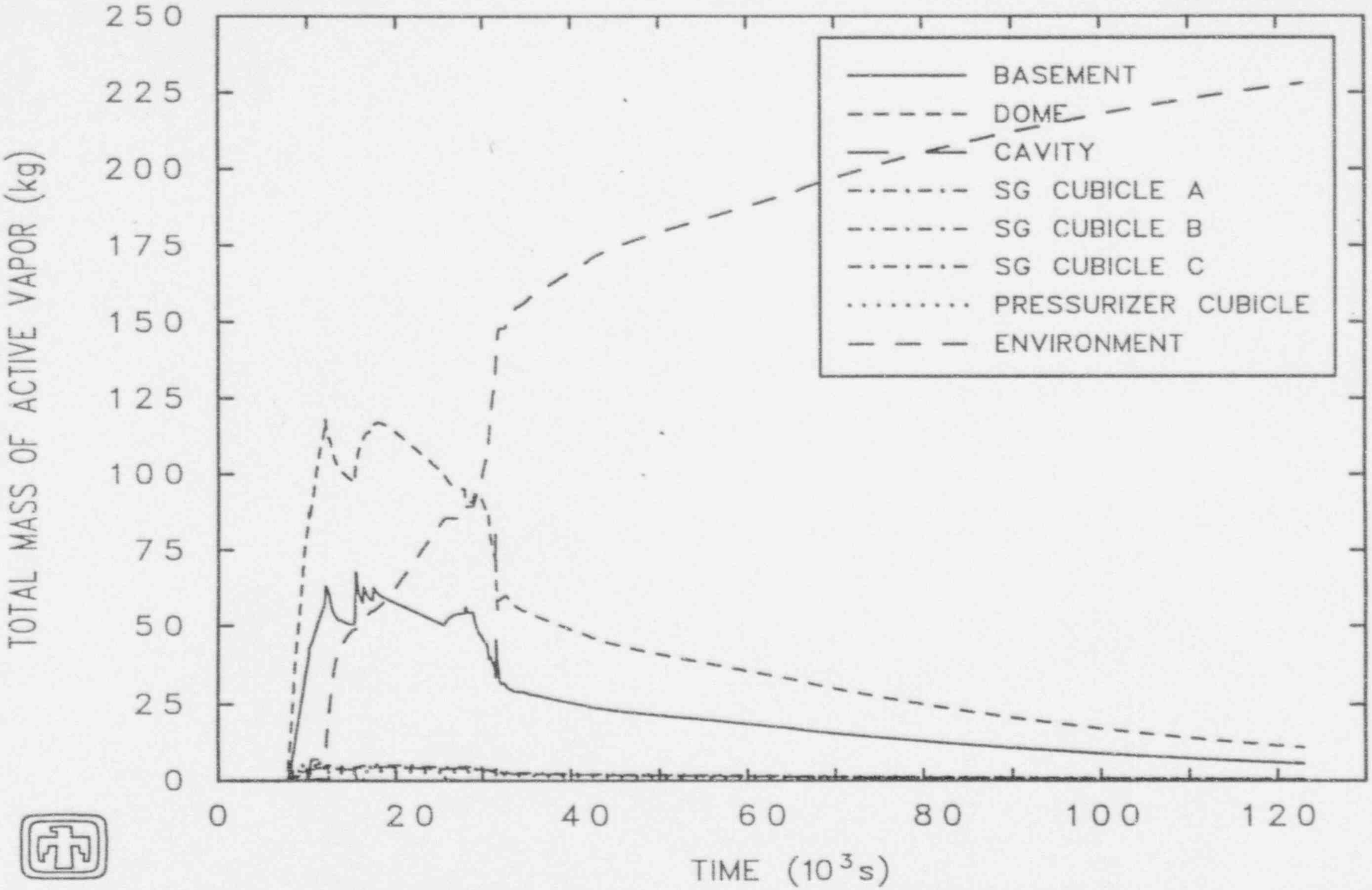


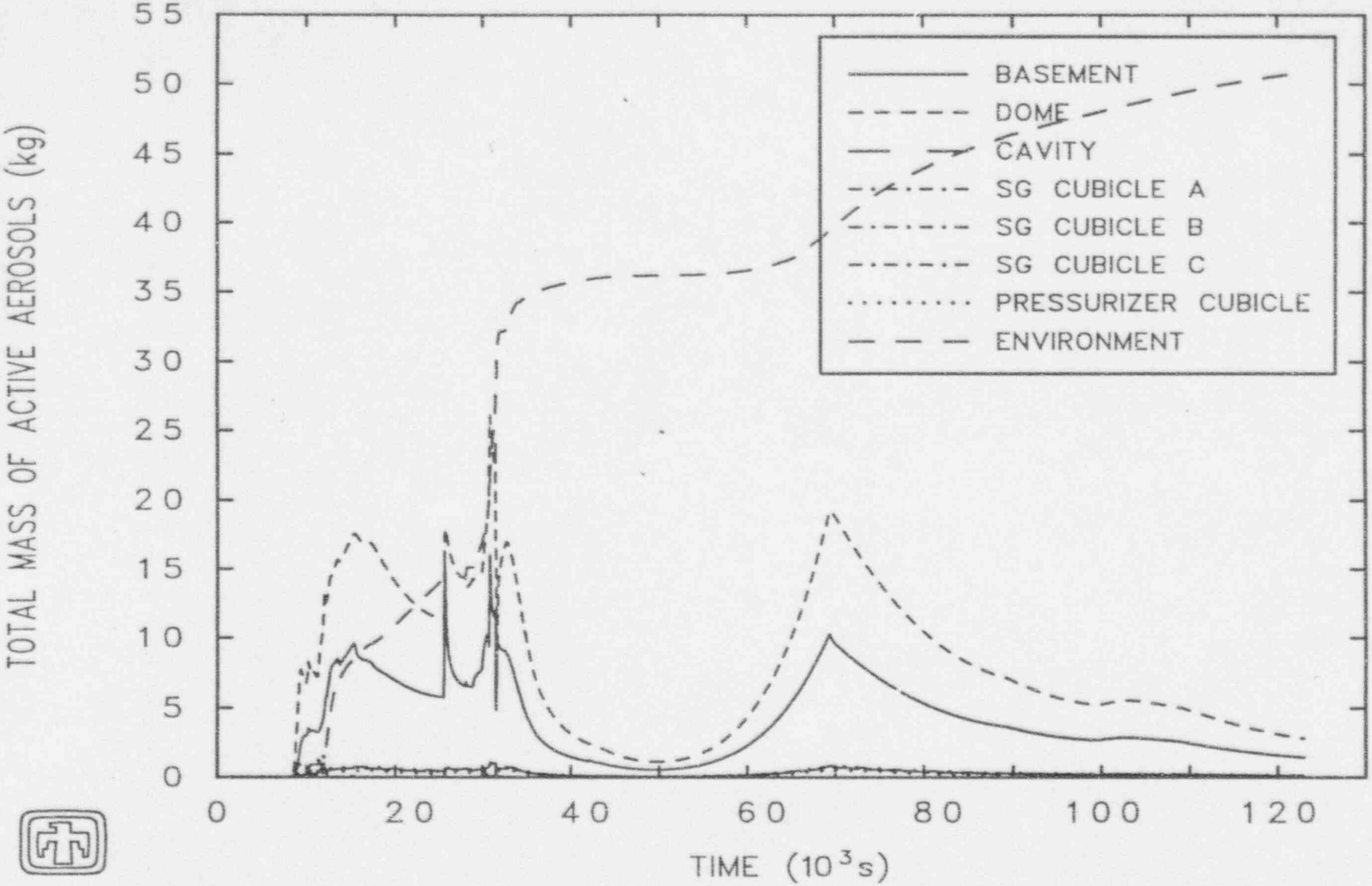
SRRYP6 - L2CASE5 - SHUTDOWN + 1.13 DAYS CNTMNT LEAKS
SRRYL2CNM 8/16/93 09:30:08 MELCOR

Fig 2-26



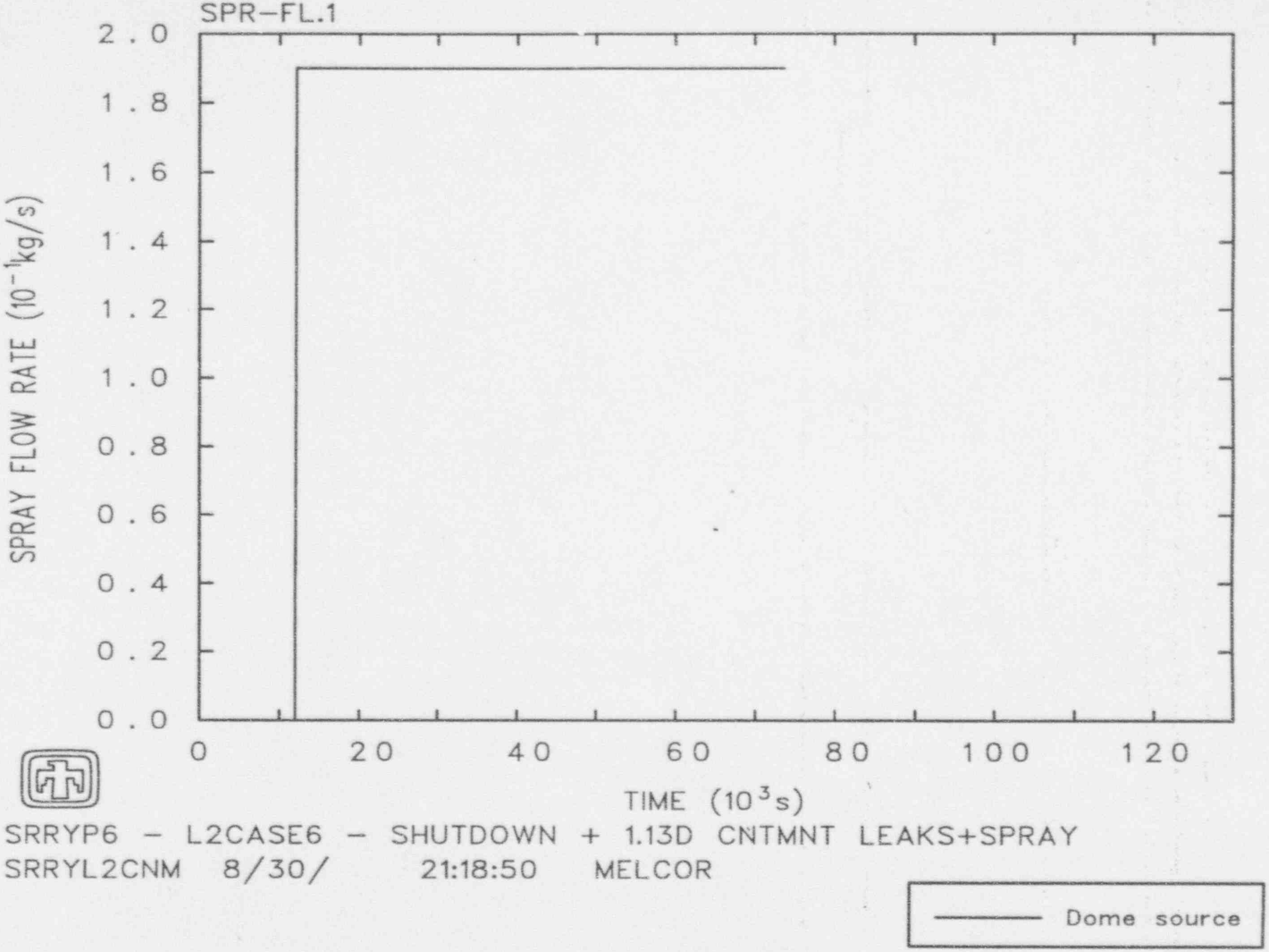
SRRYP6 – L2CASE5 – SHUTDOWN + 1.13 DAYS CNTMNT LEAKS
SRRYL2CNM 8/16/93 09:30:08 MELCOR

FIG 2-29

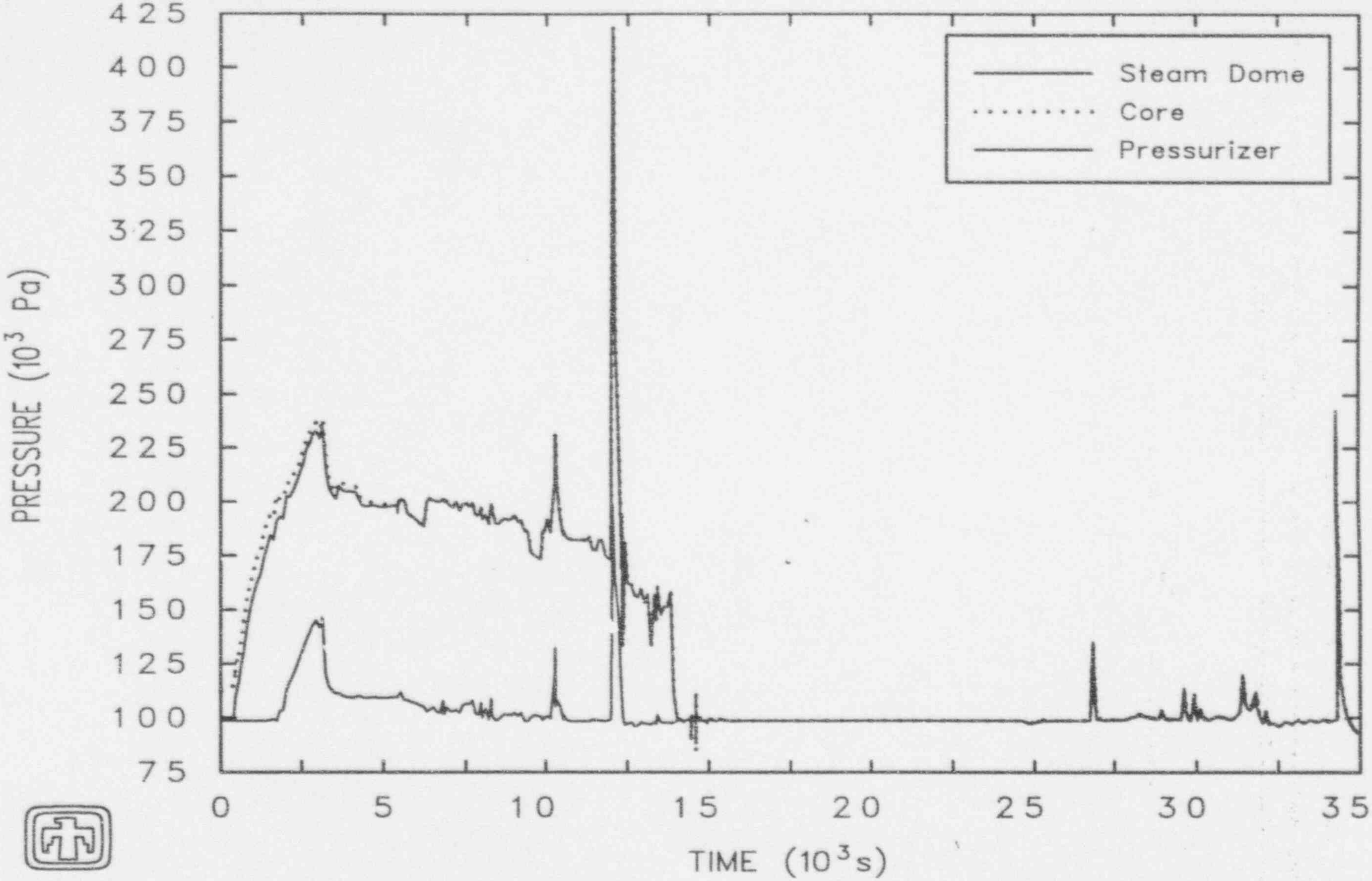


SRRYP6 – L2CASE5 – SHUTDOWN + 1.13 DAYS CNTMNT LEAKS
SRRYL2CNM 8/16/93 09:30:08 MELCOR

FiG 2-28



SPR-FL.1
2.0
1.8
1.6
1.4
1.2
1.0
0.8
0.6
0.4
0.2
0.0
SPRAY FLOW RATE (10^{-1}kg/s)
0
20
40
60
80
100
120
TIME (10^3s)
SRRYP6 - L2CASE6 - SHUTDOWN + 1.13D CNTMNT LEAKS+SPRAY
SRRYL2CNM 8/30/ 21:18:50 MELCOR
Dome source



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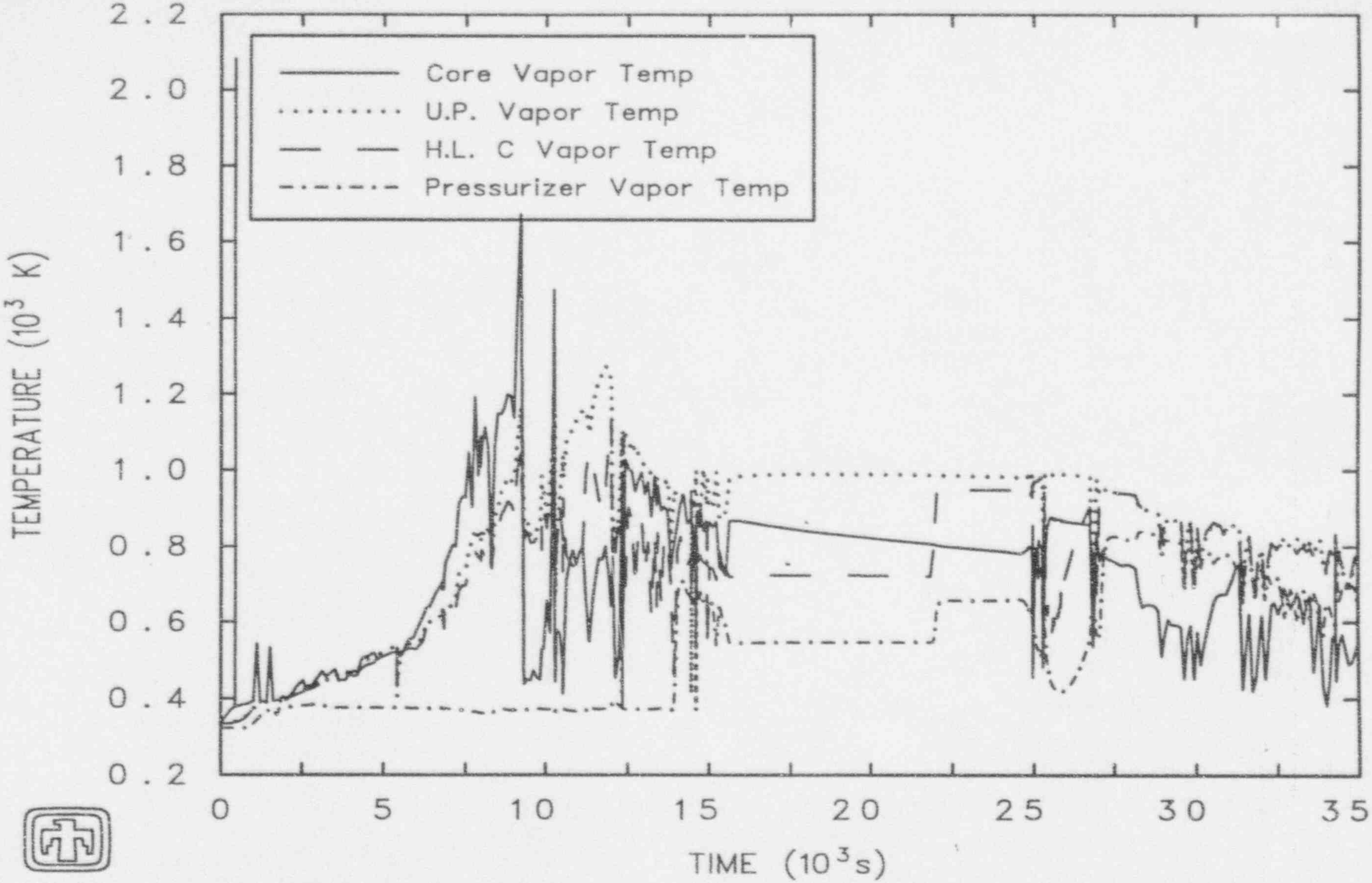

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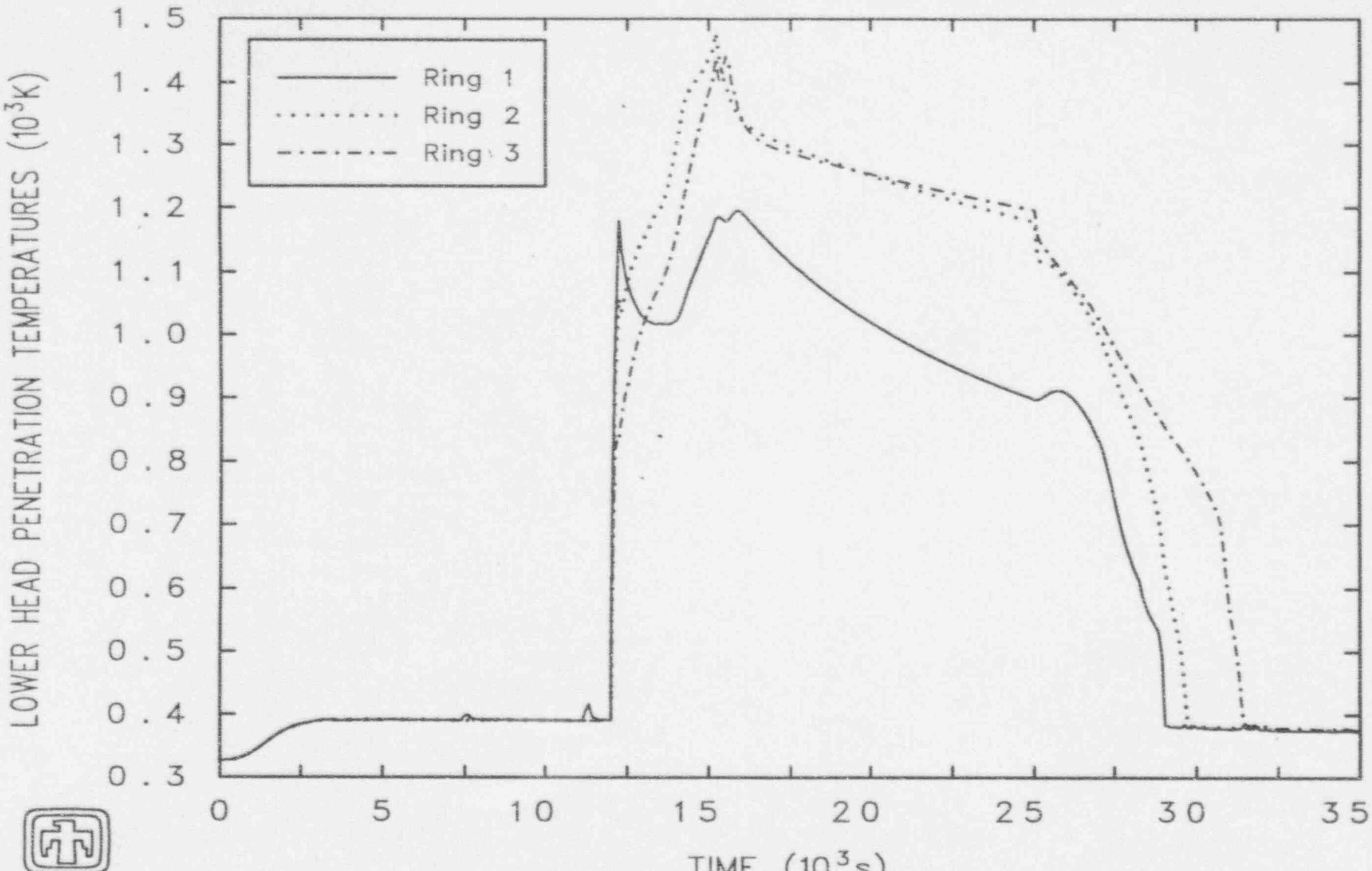

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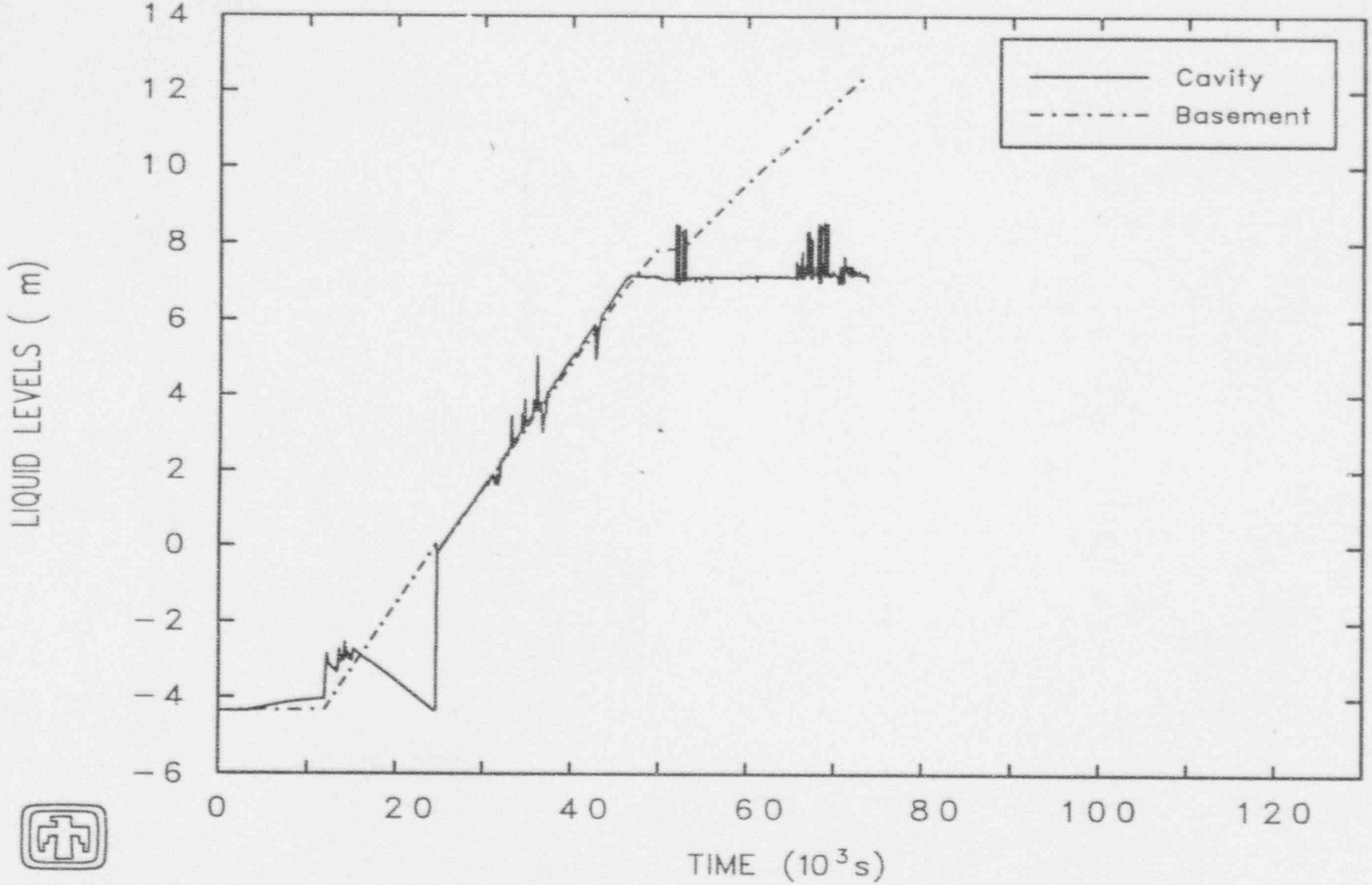

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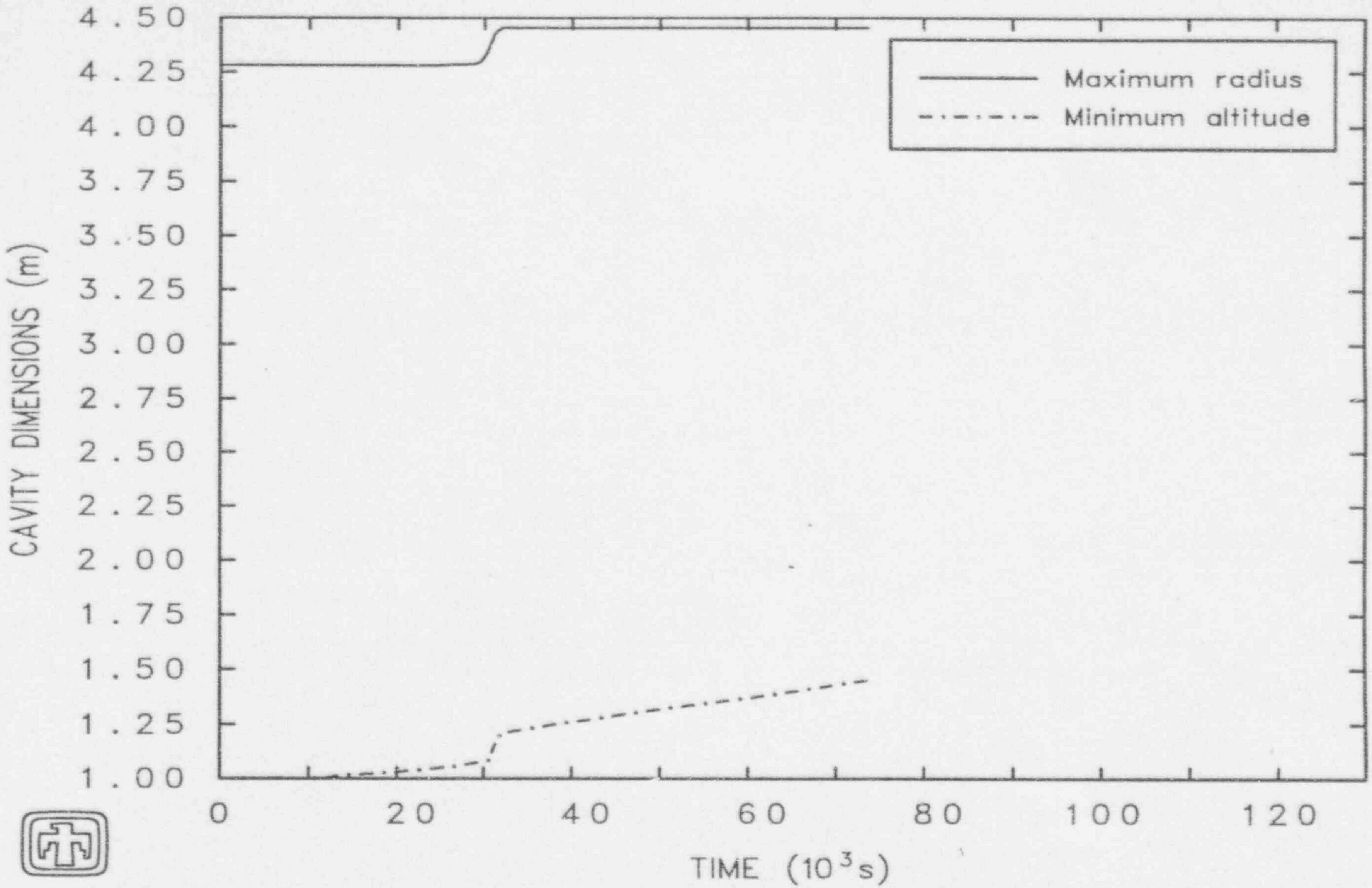

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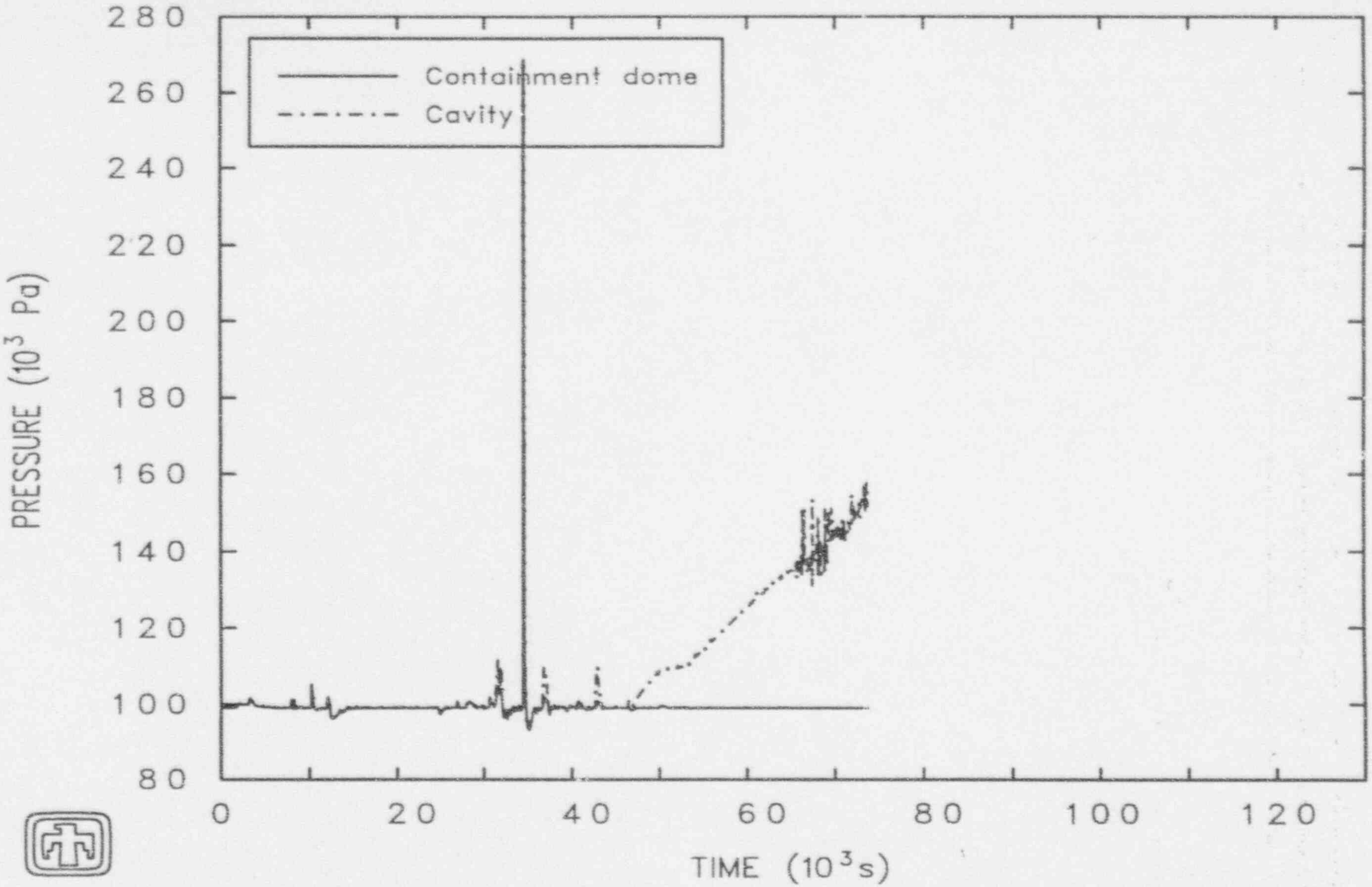
SRRYL2CNM 8/30/ 21:18:50 MELCOR



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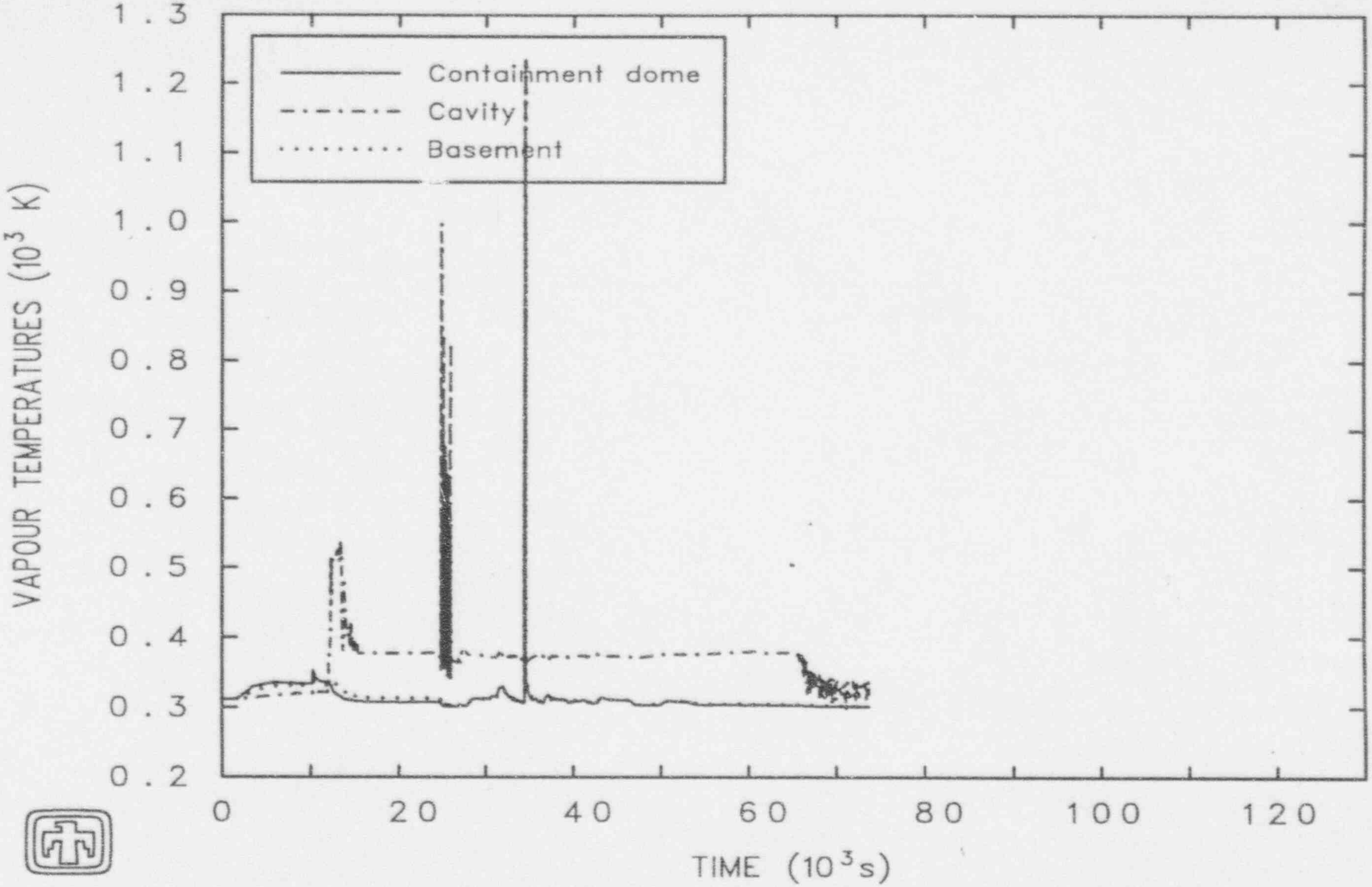

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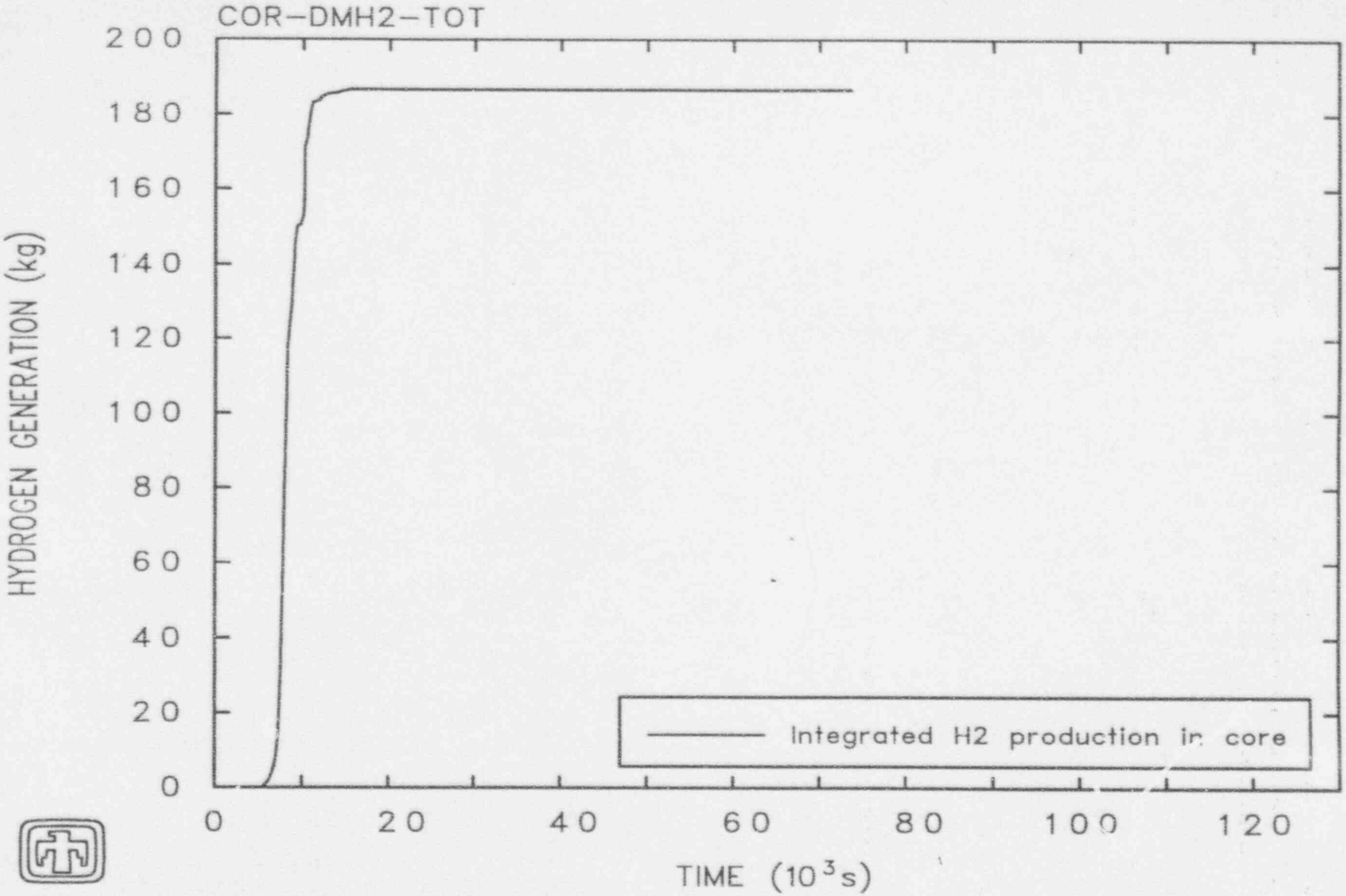

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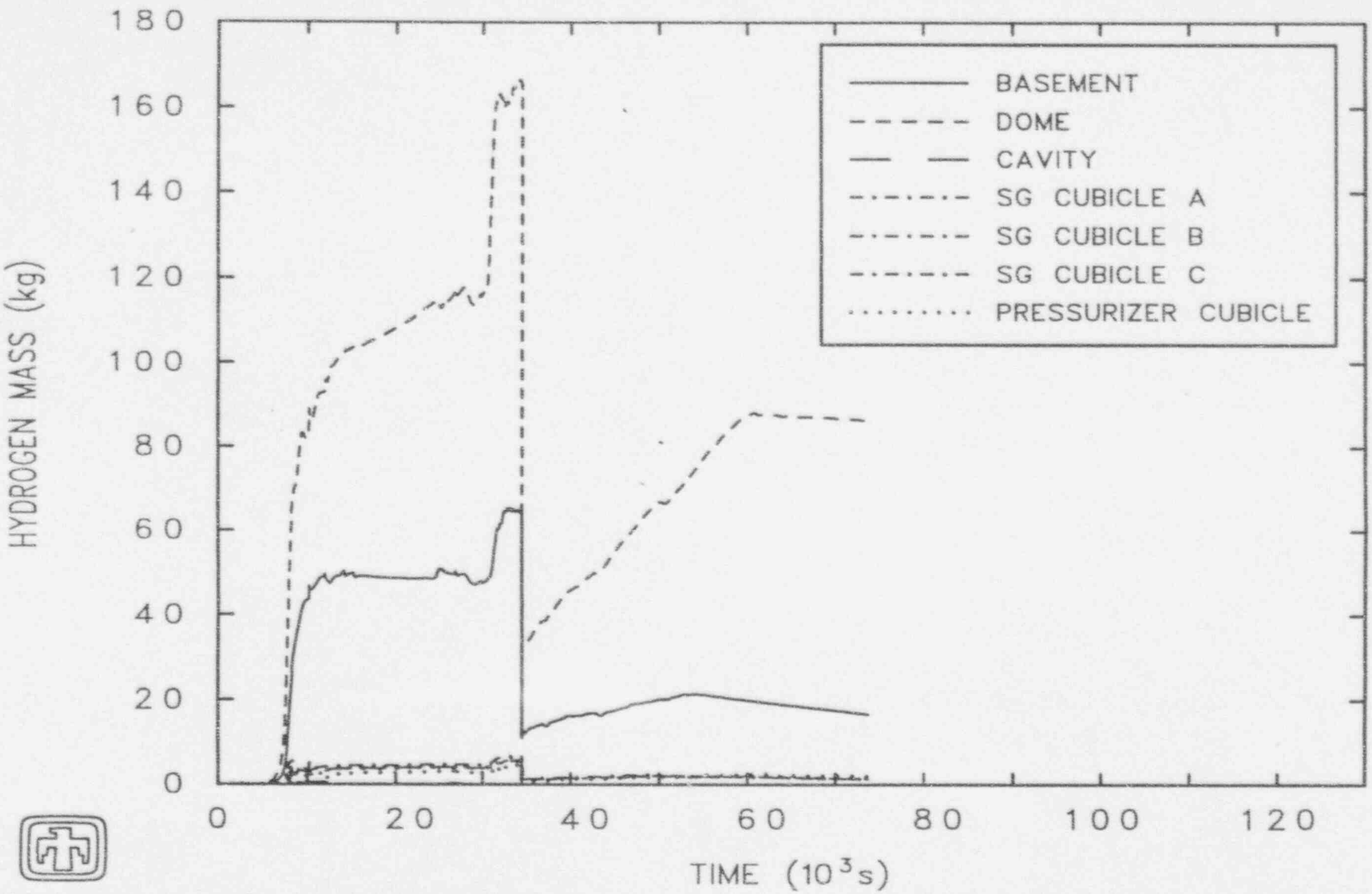
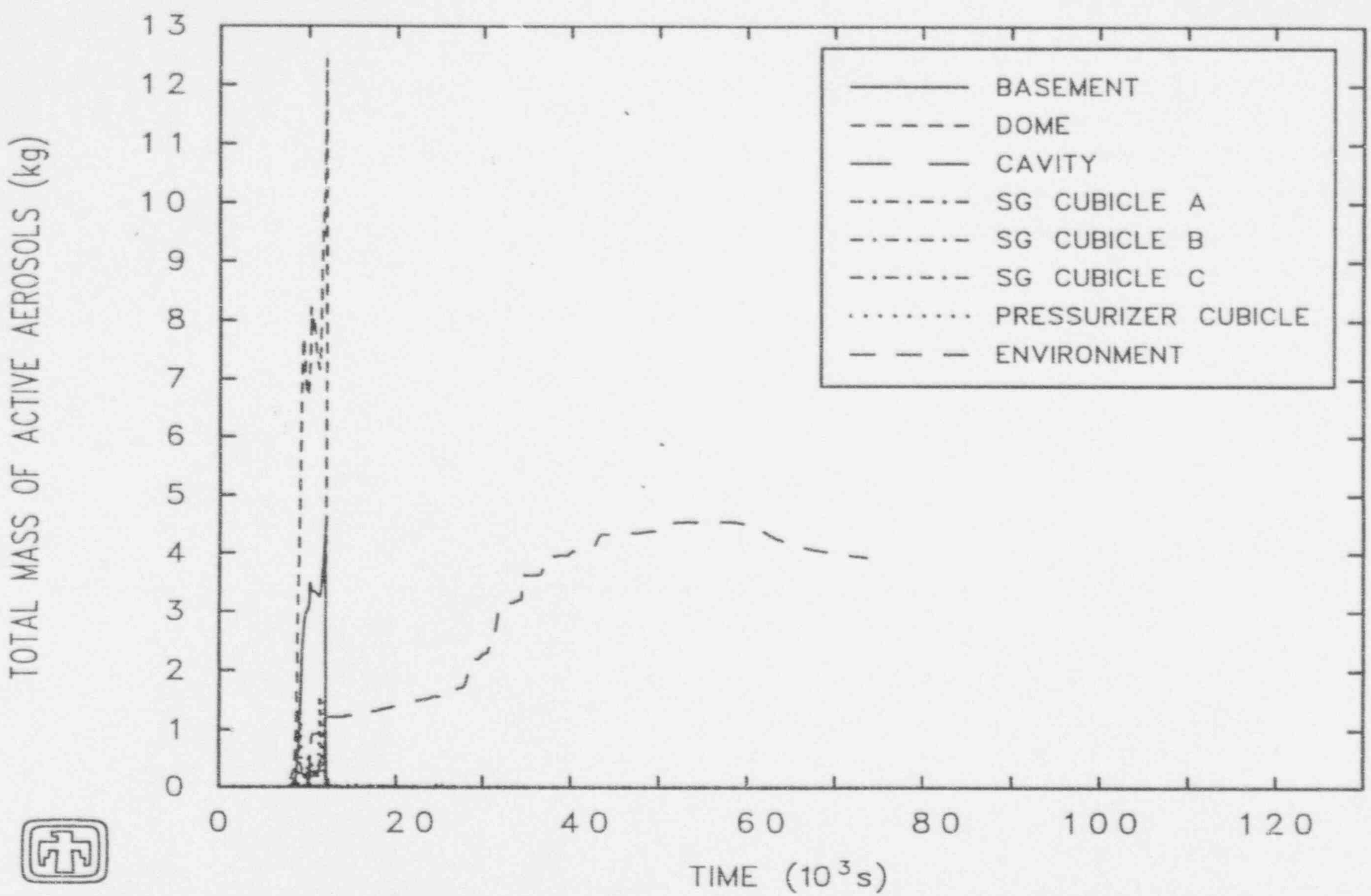
SRRYL2CNM 8/30/ 21:18:50 MELCOR



SRRYP6 – L2CASE6 – SHUTDOWN + 1.13D CNTMNT LEAKS+SPRAY
SRRYL2CNM 8/30/ 21:18:50 MELCOR

Fig 2-34



SRRYP6 – L2CASE6 – SHUTDOWN + 1.13D CNTMNT LEAKS+SPRAY
SRRYL2CNM 8/30/ 21:18:50 MELCOR

Fig 2-35

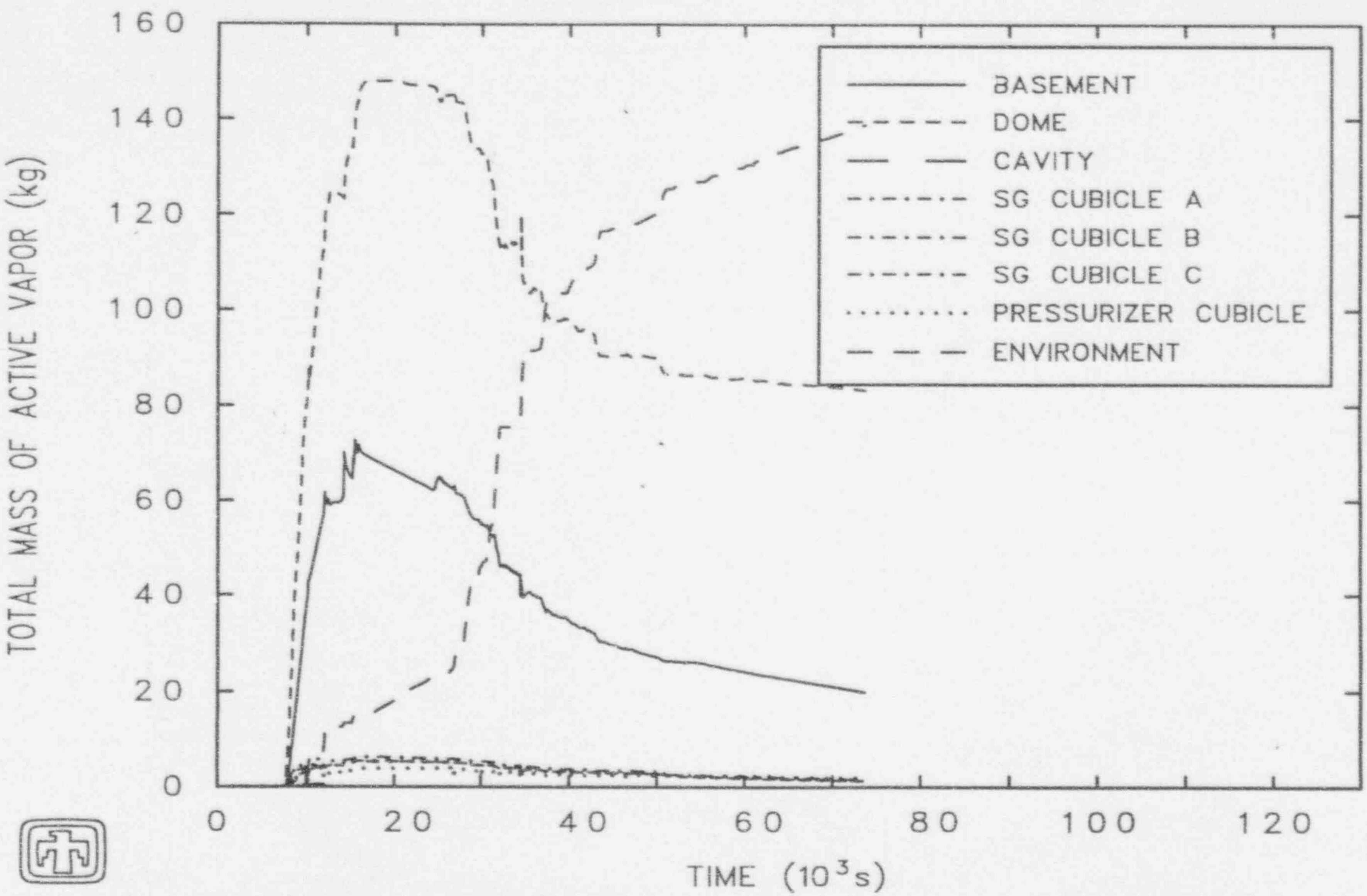


Fig 2-36

SRRYP6 – L2CASE6 – SHUTDOWN + 1.13D CNTMNT LEAKS+SPRAY
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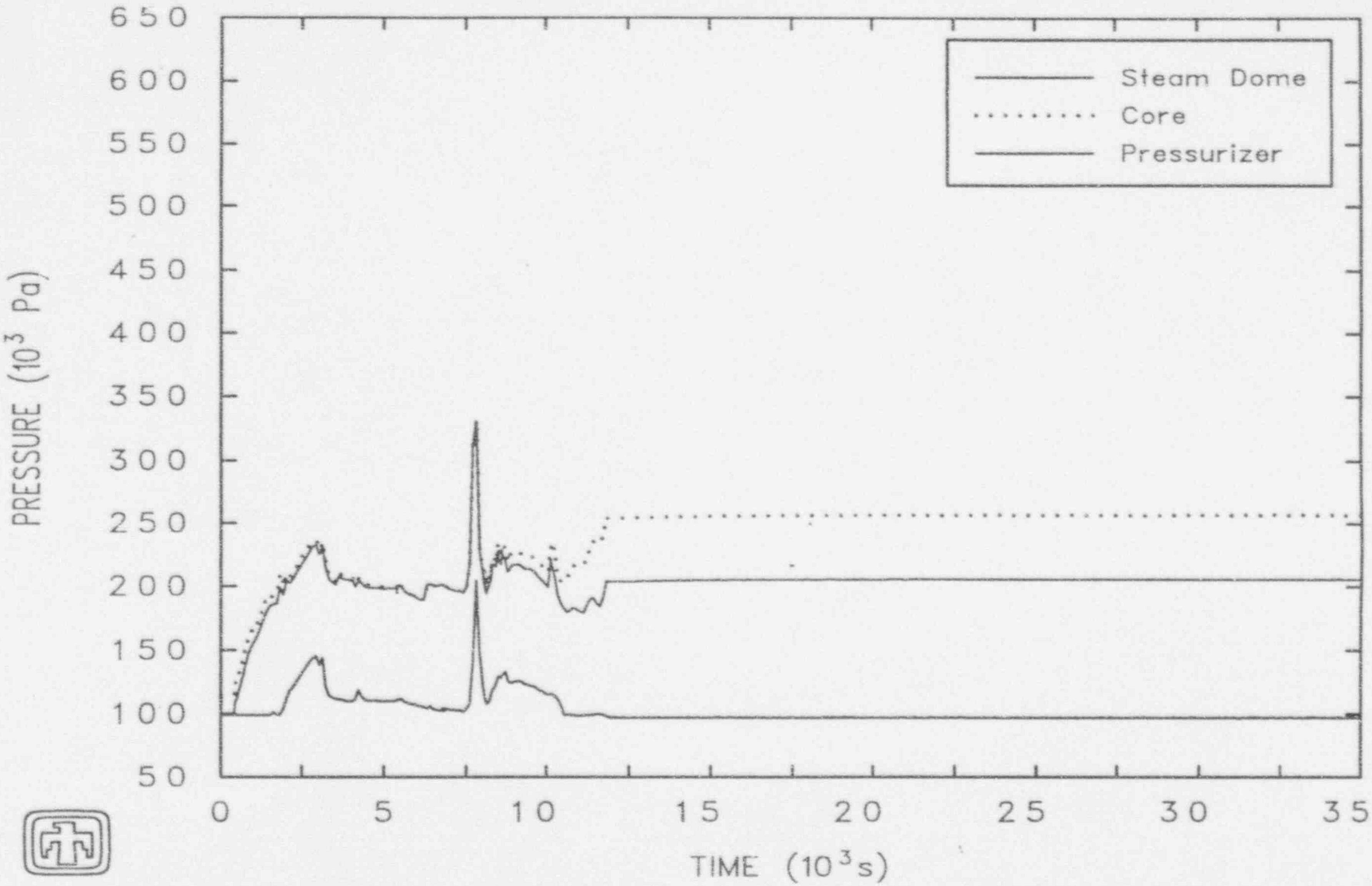
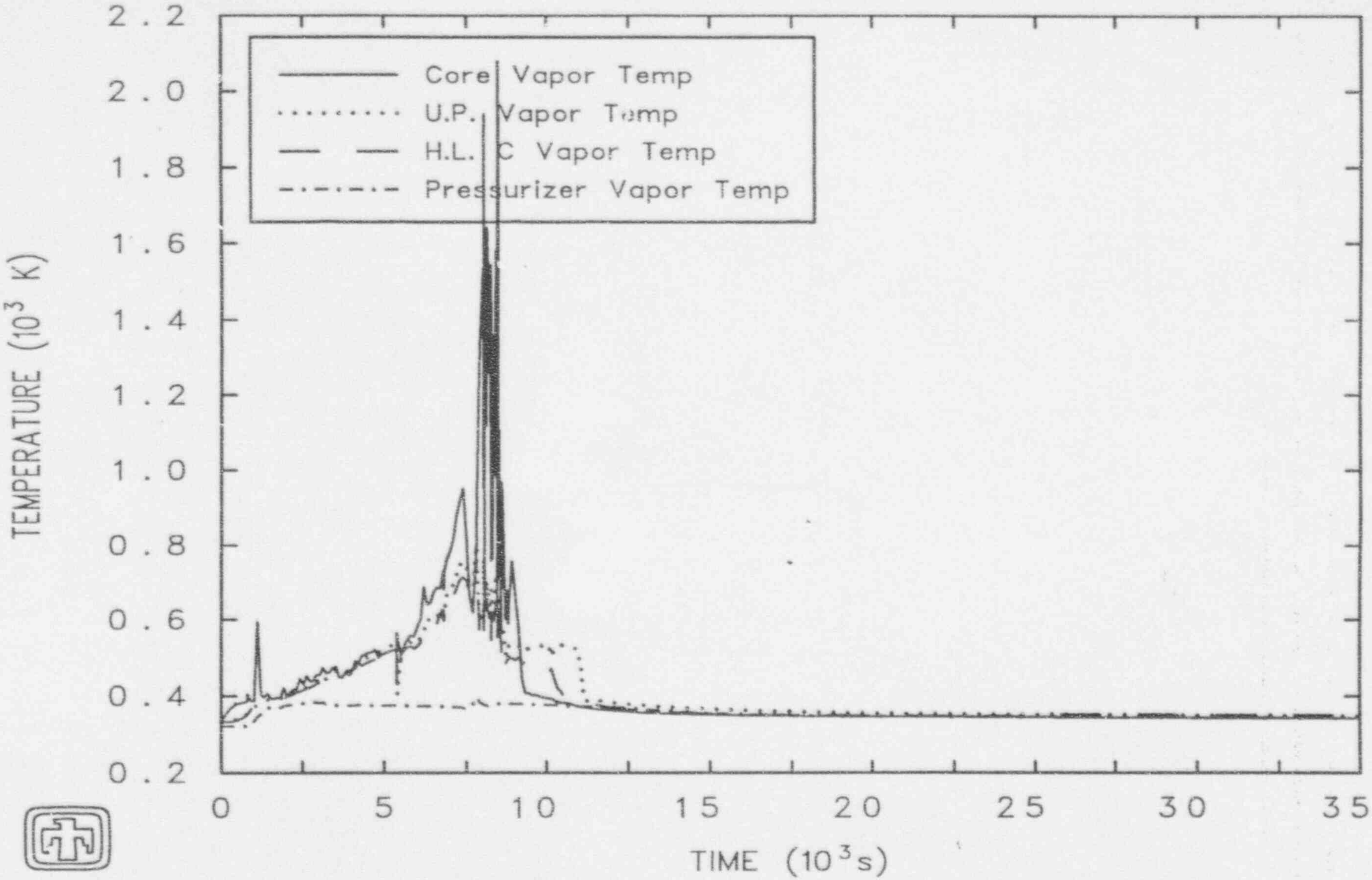


SRRYP6 – L2CASE7 – SHUTDOWN + 1.13D CNTMNT LEAKS+ECCS
SRRYL2CNM 8/26/93 18:01:36 MELCOR



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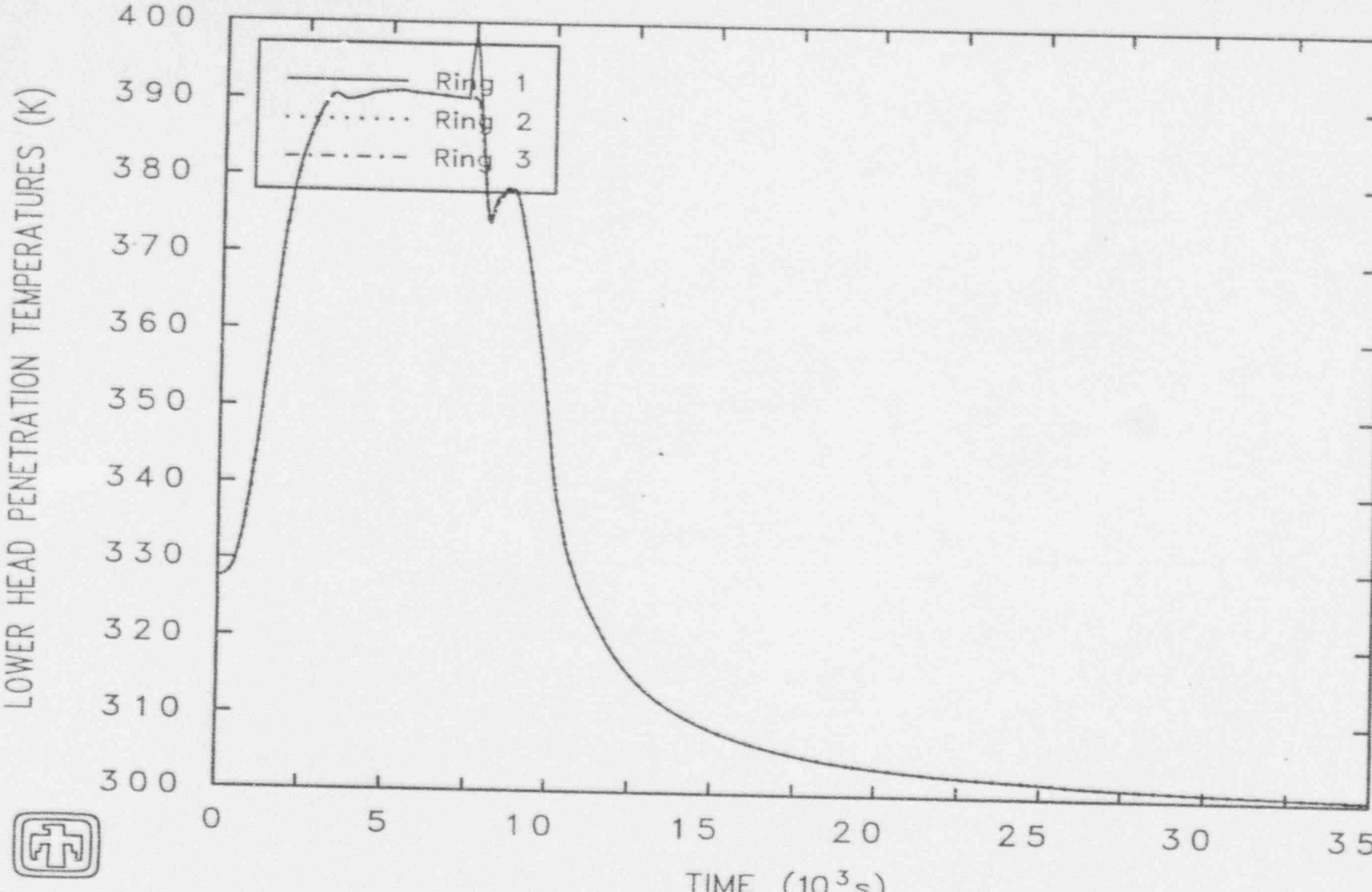
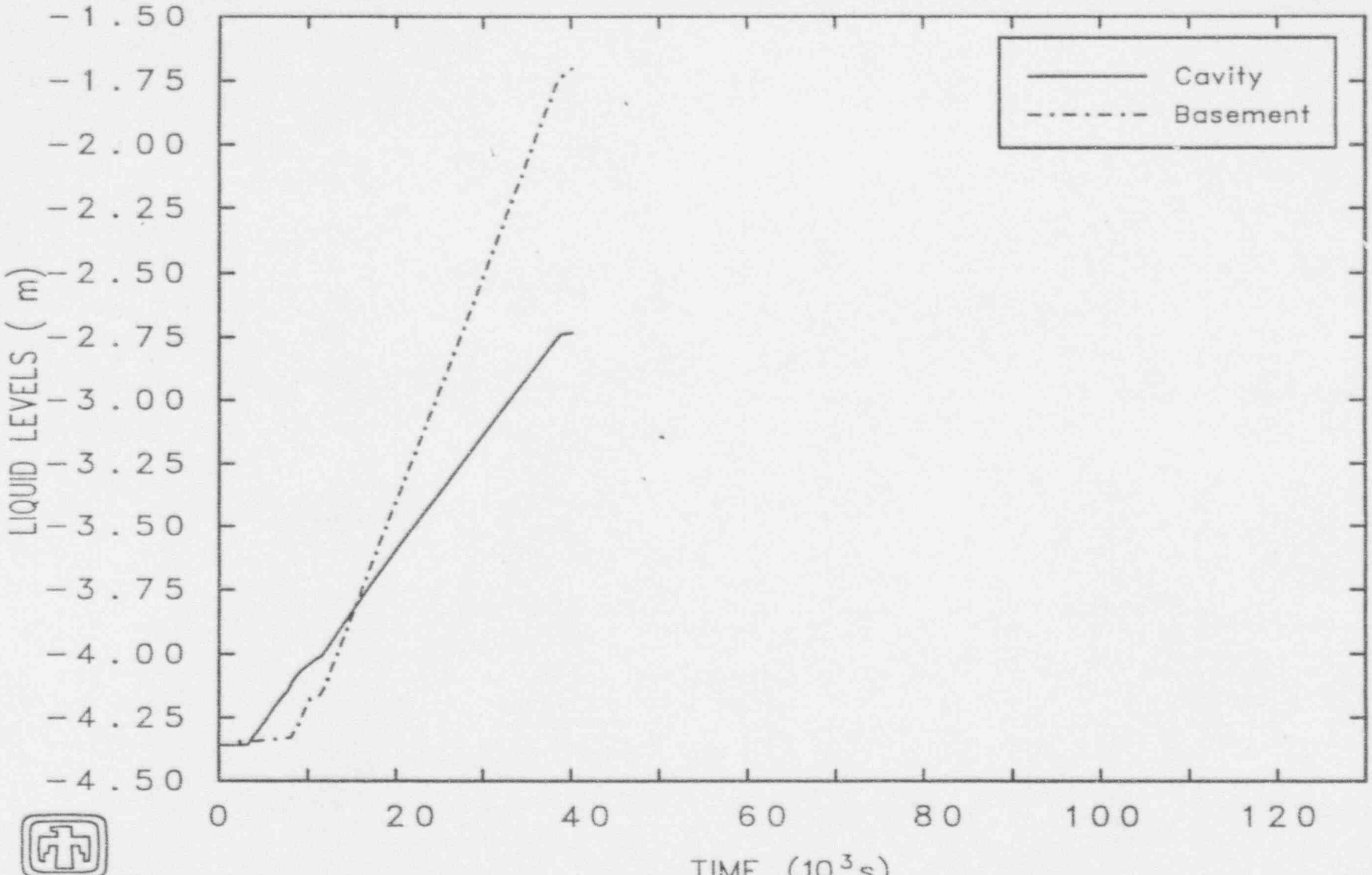
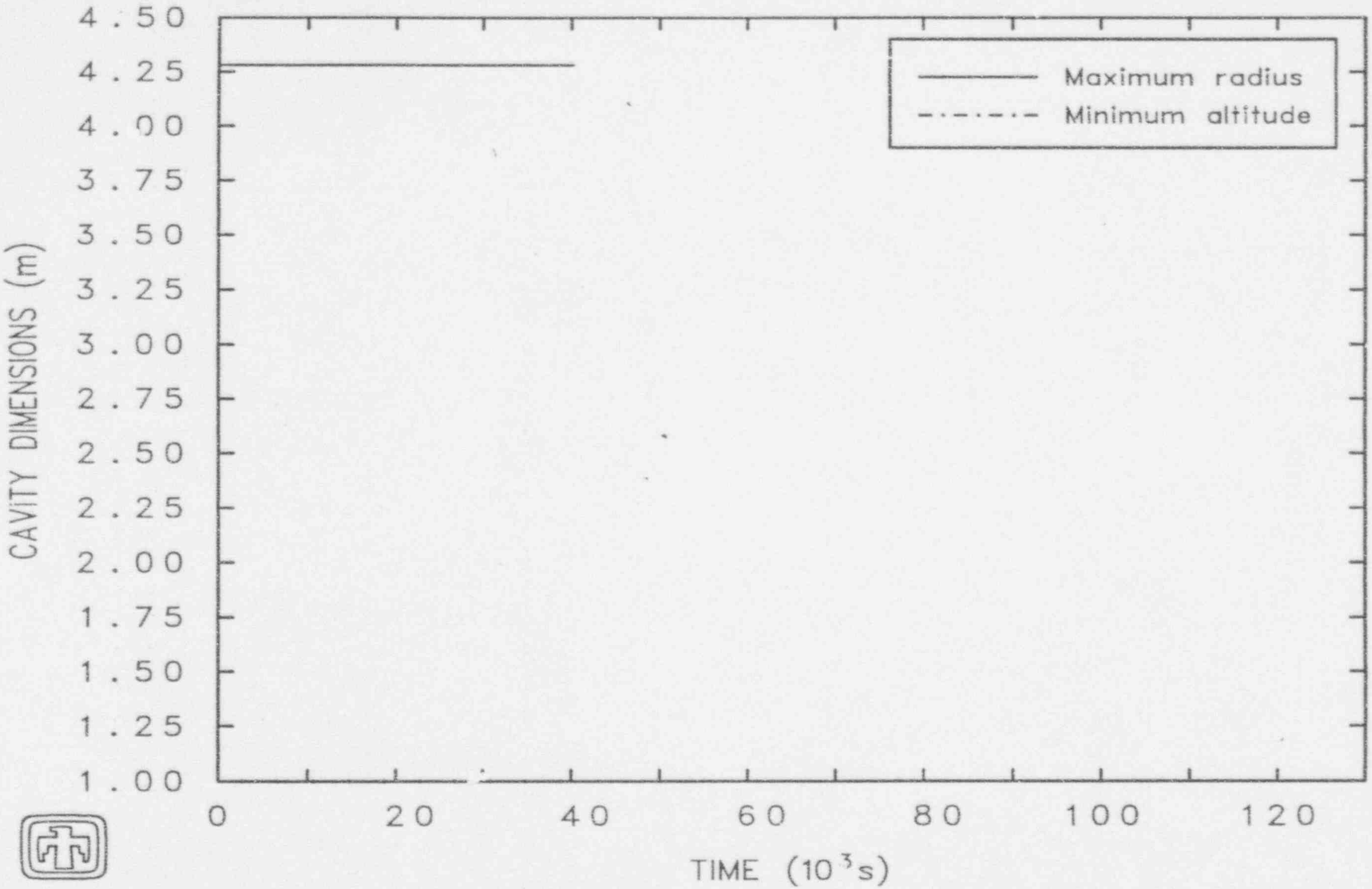


SRRYP6 – L2CASE7 – SHUTDOWN + 1.13D CNTMNT LEAKS+ECCS
SRRYL2CNM 8/26/93 18:01:36 MELCOR

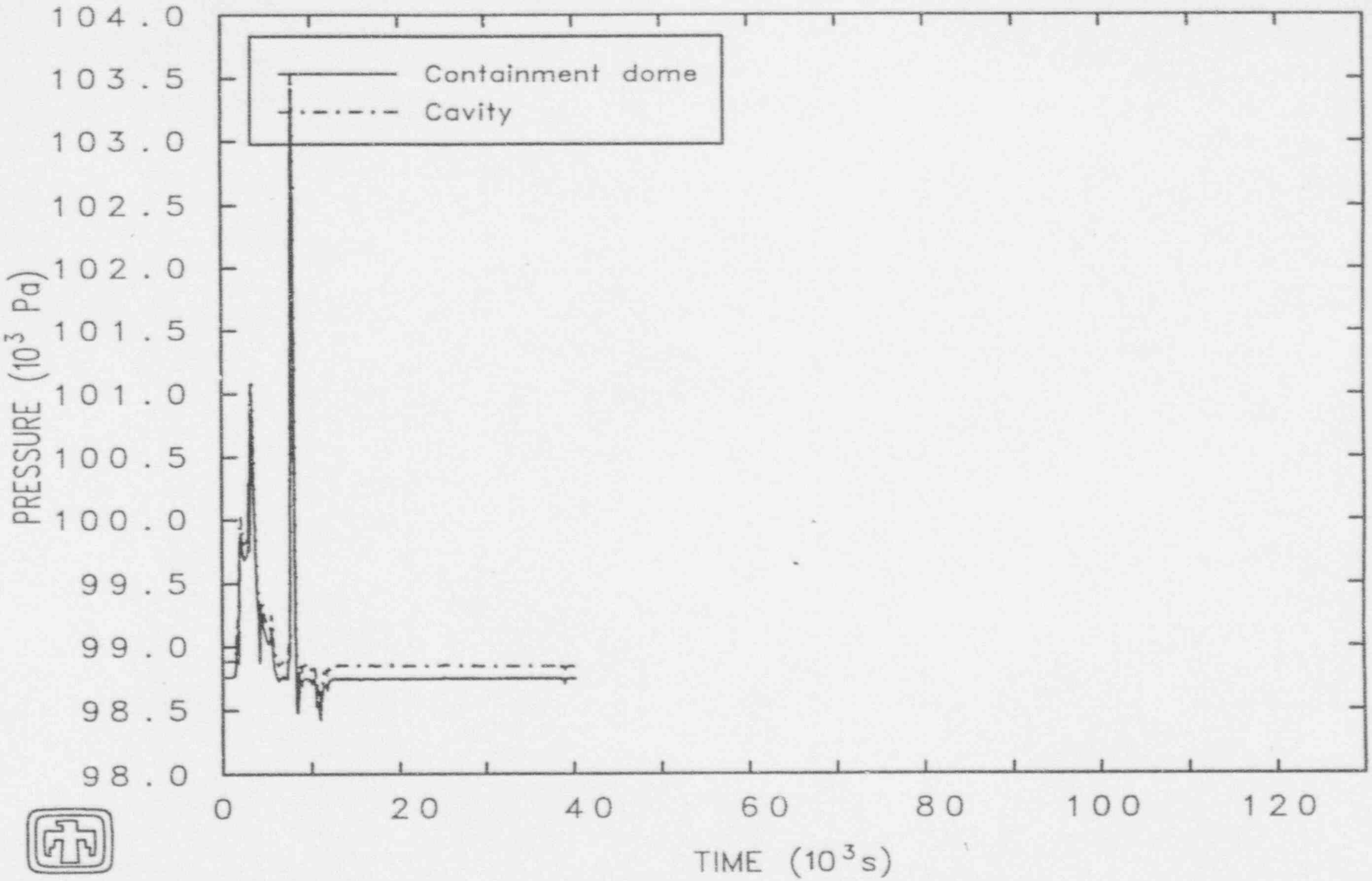


SRRYP6 - L2CASE7 - SHUTDOWN + 1.13D CNTMNT LEAKS+ECCS
SRRYL2CNM 8/26/93 18:01:36 MELCOR

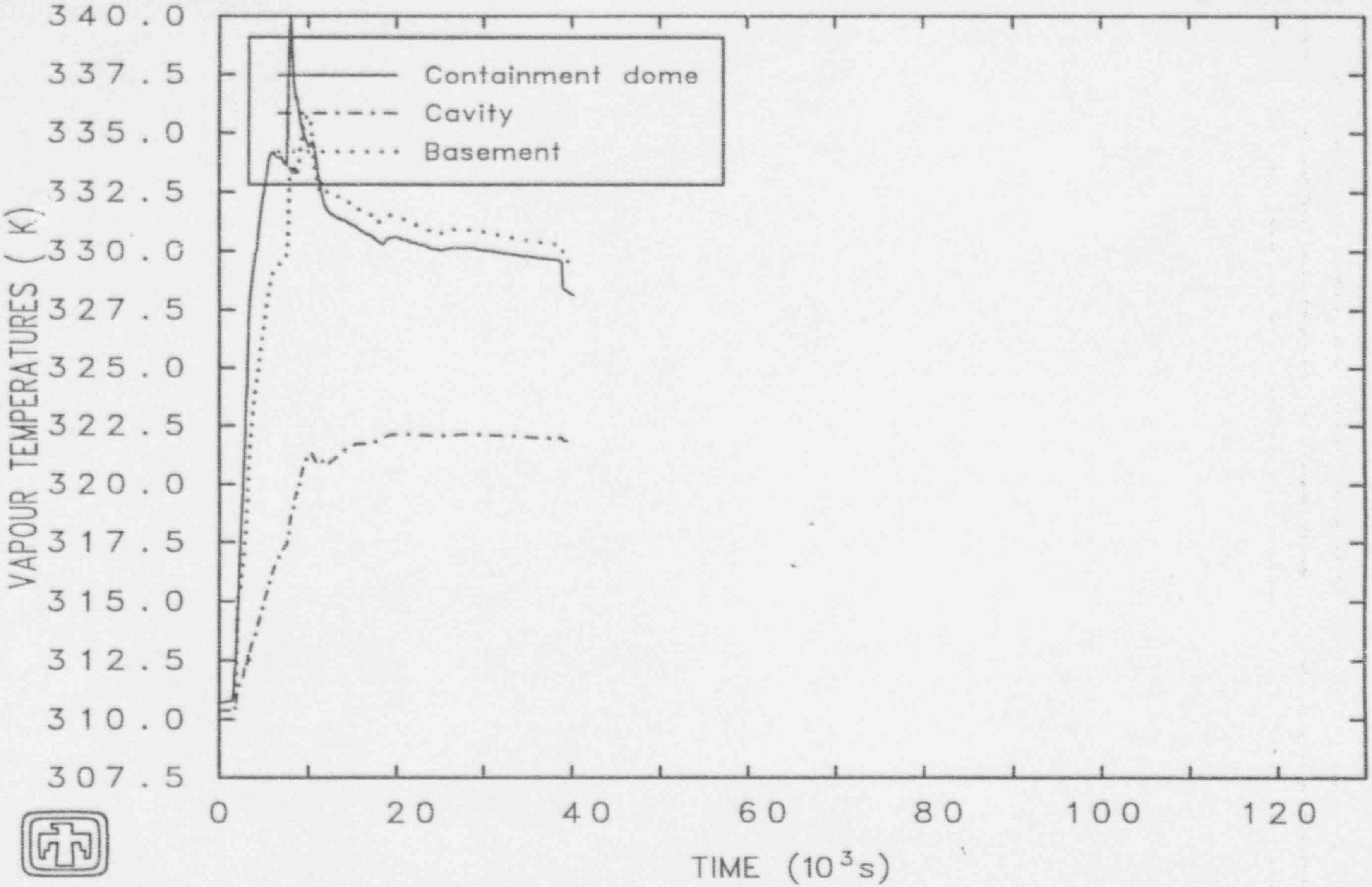


SRRYP6 - L2CASE7 - SHUTDOWN + 1.13D CNTMNT LEAKS+ECCS

SRRYL2CNM 8/26/93 18:01:36 MELCOR



SRRYP6 – L2CASE7 – SHUTDOWN + 1.13D CNTMNT LEAKS+ECCS
SRRYL2CNM 8/26/93 18:01:36 MELCOR



SRRYP6 – L2CASE7 – SHUTDOWN + 1.13D CNTMNT LEAKS+ECCS
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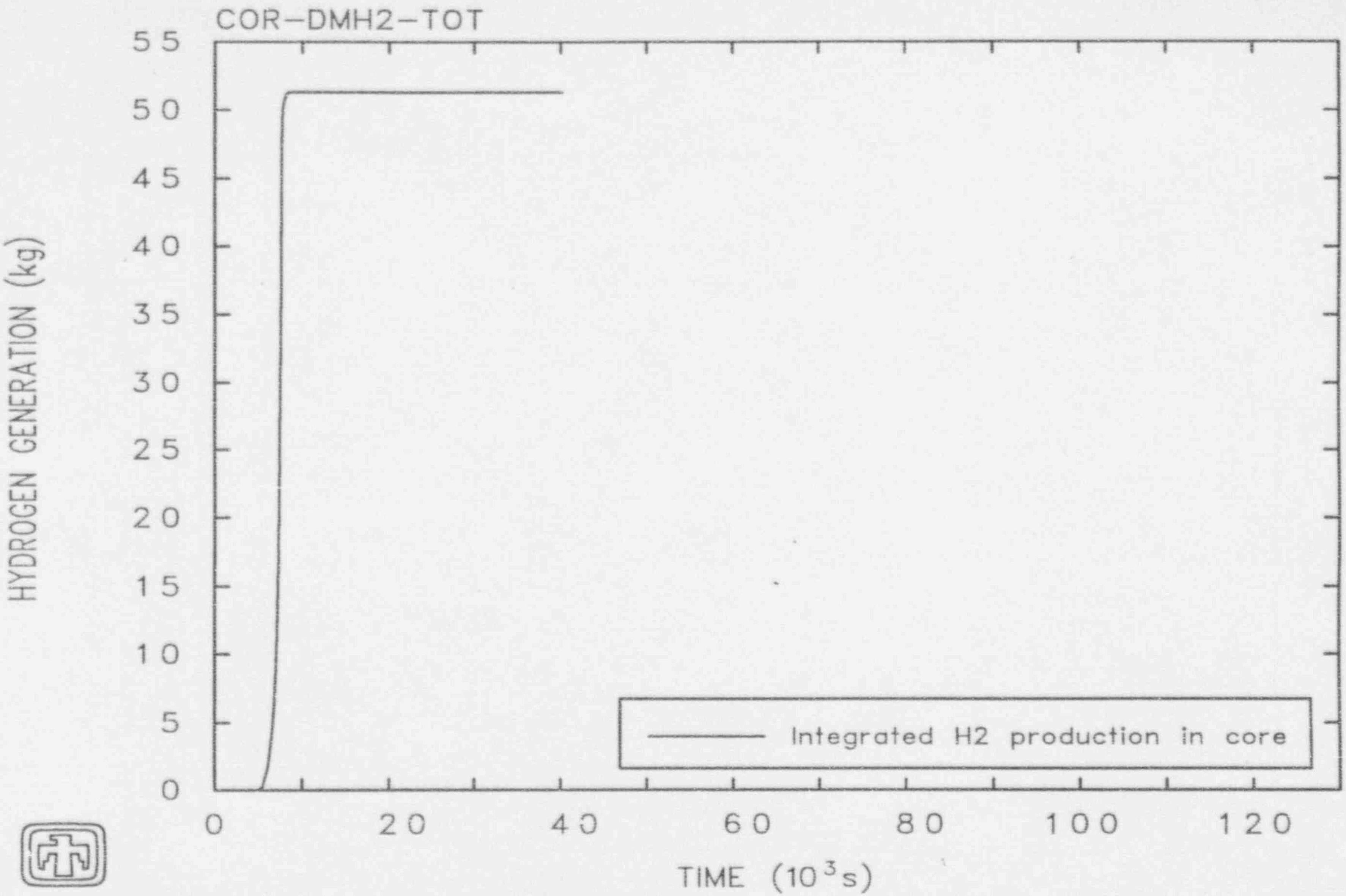

SRRYP6 - L2CASE7 - SHUTDOWN + 1.13D CNTMNT LEAKS+ECCS
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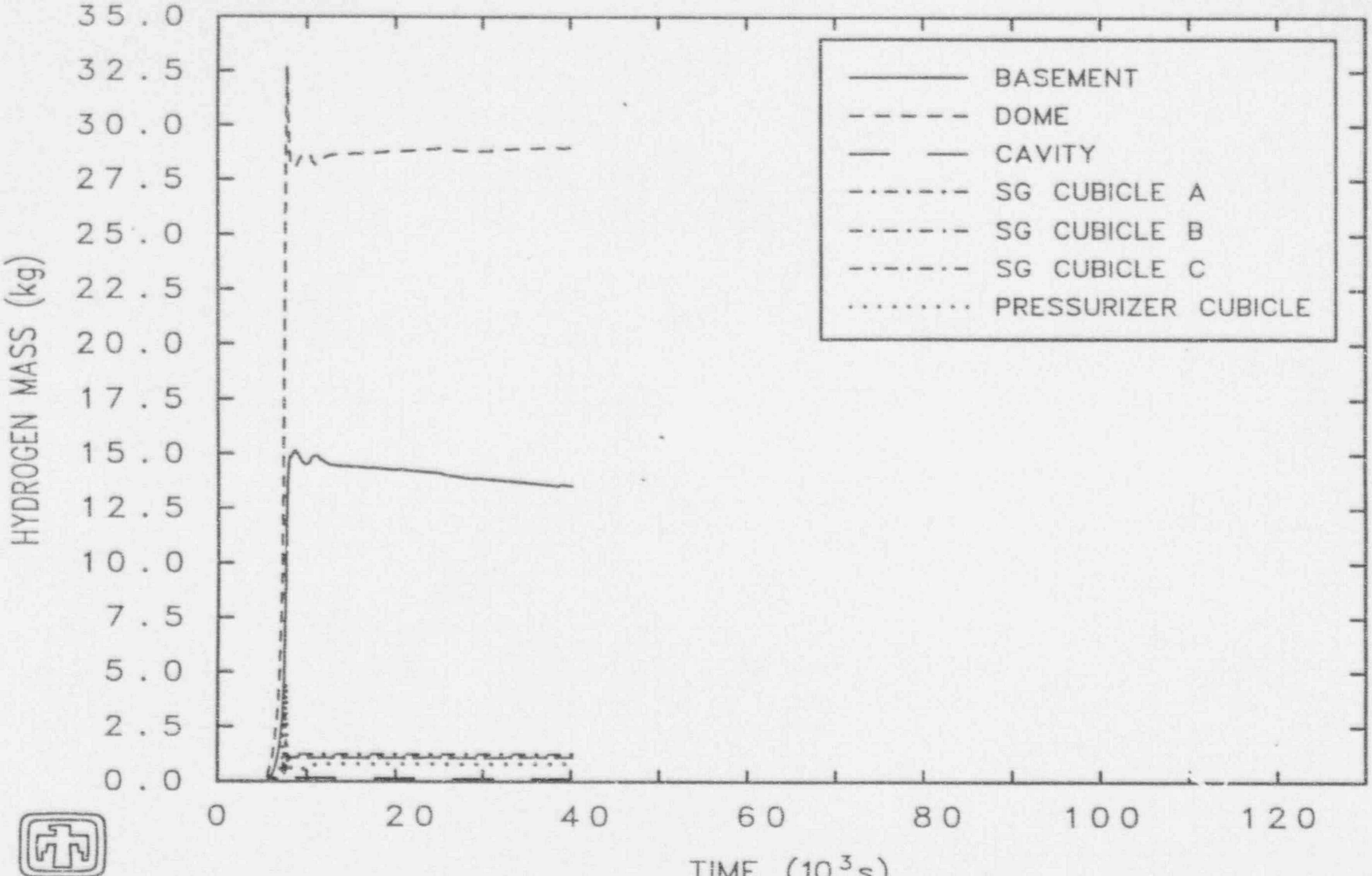

SRRYP6 – L2CASE7 – SHUTDOWN + 1.13D CNTMNT LEAKS+ECCS
SRRYL2CNM 8/26/93 18:01:36 MELCOR

Fig 2-37

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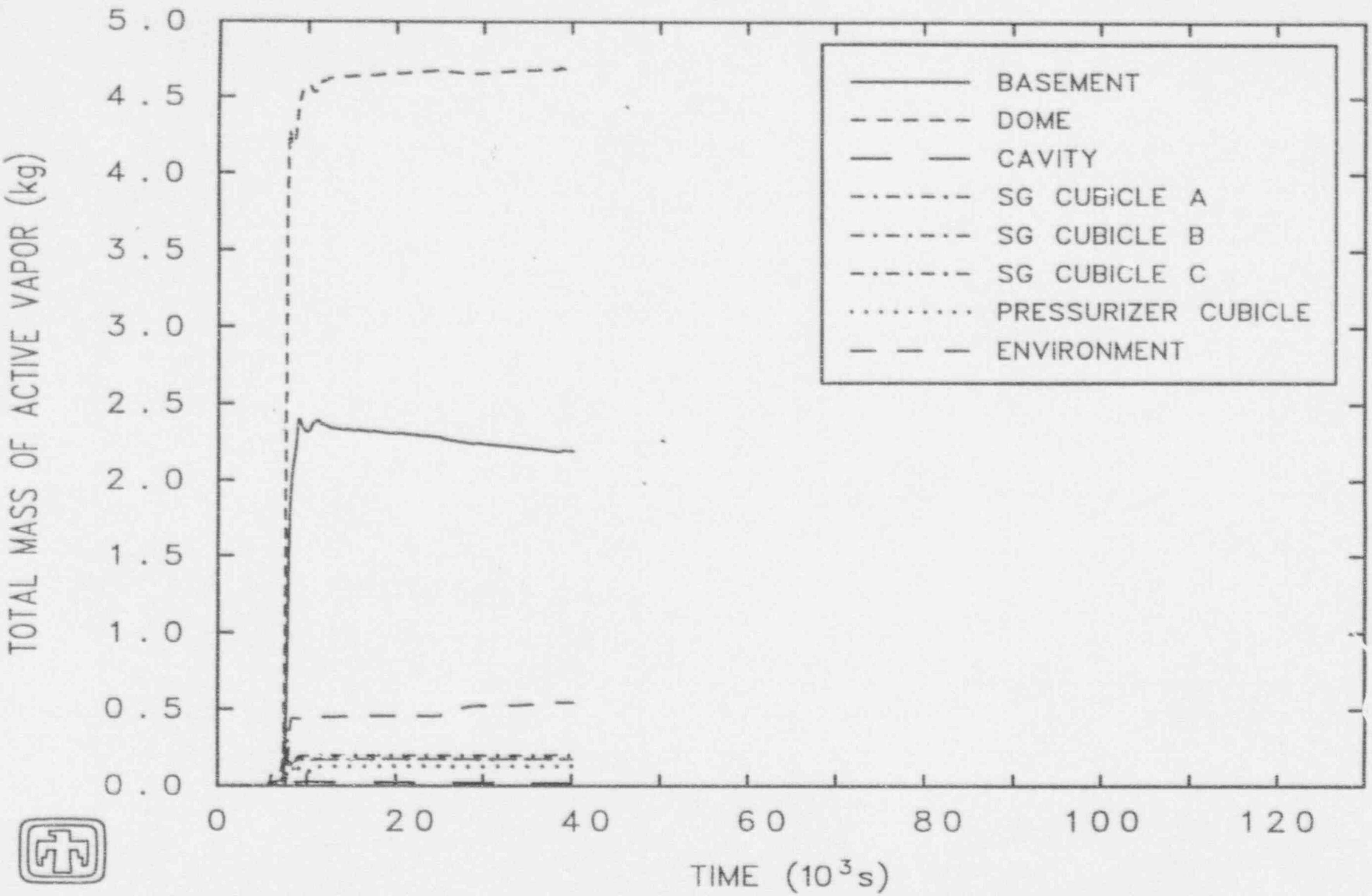
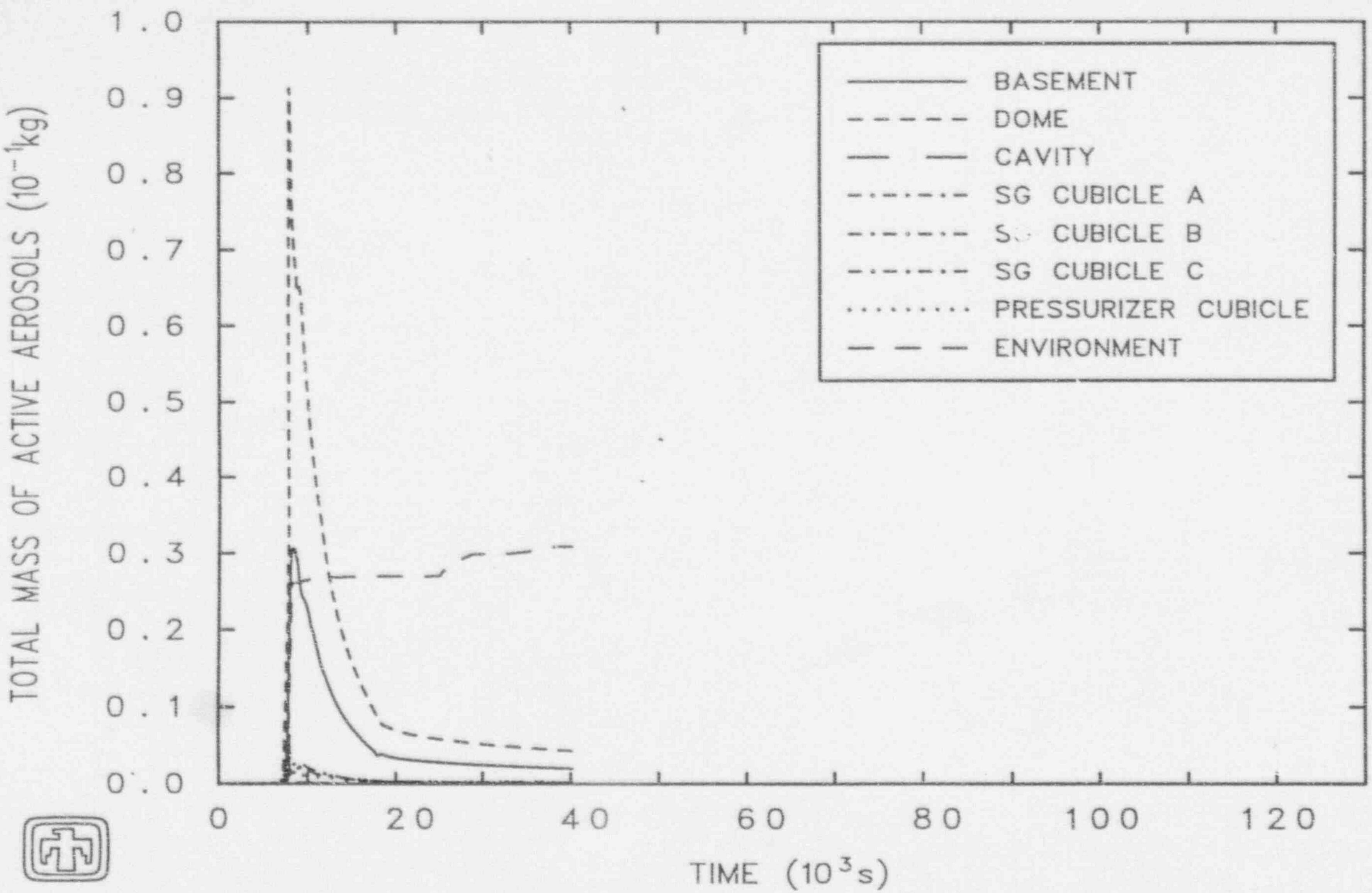


SRRYP6 - L2CASE7 - SHUTDOWN + 1.13D CNTMNT LEAKS+ECCS
SRRYL2CNM 8/26/93 18:01:36 MELCOR

F142-35



SRRYP6 – L2CASE7 – SHUTDOWN + 1.13D CNTMNT LEAKS+ECCS
SRRYL2CNM 8/26/93 18:01:36 MELCOR

Fig 2-38