AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.013

Question:

For post ADS long-term cooling, Section A3.5 (and Section A5.1 Long-Term Cooling Cases No. 5 and 6) provides your judgement, based on AP1000's increased power level, ADS4 flow capacity, IRWST injection, and containment recirculation over the AP600 design. The conclusion provided is that the AP1000 ADS4 vent capacity is sufficient for justification of the long-term core cooling success criteria assumed in the AP1000 PRA. Section A5.2 states that Section A5.5.2 documents the long-term cooling analyses performed with the WCOBRA/TRAC code with the details of the analysis methodologies used provided within each subsection. However, there is no Section A5.5.2.

Provide the WCOBRA/TRAC long-term cooling analysis as part of AP1000 PRA to support your conclusion.

Westinghouse Response:

Attached to this RAI response is analysis of two thermal hydraulic long-term cooling (LTC) analysis cases. These cases were determined to be the low margin, risk important cases that should be analyzed to bound their T/H uncertainty (refer to response to RAI 720.012). This WCOBRA-TRAC analysis shows that adequate core cooling is provided for these cases.

Design Control Document (DCD) Revision:

None

PRA Revision:

The attached analysis will be appropriately inserted into Appendix A of the PRA.

1. Objective

The objective of these analyses is to analyze the AP1000 long-term core cooling (LTCC) behavior following a guillotine double-ended direct vessel injection (DEDVI) line break to support the PRA thermal hydraulic (T/H) uncertainty evaluations. In order to bound the T&H uncertainty, this analysis is performed using the DCD code and conservative methods.

Two cases of LTCC following a DEDVI line break are analyzed. These cases were determined by T/H uncertainty evaluations performed for AP1000 (refer to RAI 720.012). One of these cases considers that the containment is isolated (case F), and the other case considers that the containment isolation has failed (case G). It is conservatively assumed that the DEDVI line break occurs in the PXS-B room. Since the size of this room is bigger than PXS-A, it reduces the containment water level during recirculation. It also takes more time for the water to fill it to the DVI nozzle elevation, where water can start flowing into the downcomer through the broken DVI line. In both cases, the general assumptions and methodology of the calculations are essentially the same. Conservative boundary and initial conditions are applied consistent with these multiple failure PRA based scenarios to ensure that the thermal/hydraulic uncertainties contained within the success criteria are bounded.

Below is a short summary of the two T/H uncertainty cases being described herein.

Case F:

- DEDVI LOCA in line B
- Available equipment - 1/1 CMT (A), both IRWST injection lines open with 1/2 valves open in each, only 1 recirculation line available with both valves open and this is the line attached to DVI-B, **3/4 ADS-4**, PCS water drain with 1/3 valves open
- Unavailable equipment - no ADS 1/2/3, PRHR, RNS injection / spill, IRWST gutter
- **Containment isolation** is assumed to have **worked.**

Case G:

- DEDVI LOCA in line B
- Available equipment - 1/1 CMT (A), both IRWST injection lines open with 1/2 valves open in each, 1/2 recirculation lines open with both valves open (line B), **4/4 ADS-4**, PCS water drain with 1/3 valves open
- Unavailable equipment - no ADS 1/2/3, PRHR, RNS injection / spill, IRWST gutter
- **Containment isolation** is assumed to have **failed** (18" HVAC line remains open).

2. WCOBRA/TRAC Modeling Methodology

The simulation methodology used in the current analyses is essentially the same as the one used for the AP600 design certification process (Ref.1).

- The T/H uncertainty analyses are performed using the WCOBRA/TRAC thermal hydraulic computer code (Ref.2).
- The WCOBRA/TRAC AP1000 model is the same as the one used in the AP1000 Post-LOCA Long-term Cooling analysis (Ref.3)
- The AP1000 LTCC simulations are performed using WCOBRA/TRAC in a transient mode. The transient mode approach has been validated by the Oregon State University Tests and was used in the AP600 Design Certification (Ref. 1).
- For each case, the AP1000 initial and boundary conditions are provided by a MAAP4 calculation. MAAP4 is capable of simulating the behavior and the interaction between the AP1000 primary system, the passive safety systems, containment, and the containment systems – a feature that is not present in WCOBRA/TRAC. The response to RAI 720.021 discusses the ability of MAAP4 to model the AP1000 in cases where there is a failure to isolate the containment.
- Like the MAAP4, the WCOBRA/TRAC simulation is performed with the following conservative general assumptions:
  - 102 percent core power
  - Appendix K decay heat
  - Maximum hydraulic resistance of the passive safety systems

3. Methodology Implementation

The transient mode calculation using WCOBRA/TRAC allows simulation of long transients with reasonable computer resources. As was shown in the validation of methods used in the DCD analysis (Ref.3), the calculation may be initiated from an arbitrary set of initial conditions. After an initial period of 500 to 1000 seconds the plant reaches a quasi-steady state that depends mostly on the system boundary conditions. During this "steady state" period, the boundary conditions are kept constant. After that, they are set as a function of time depending on the time window being simulated.

For the AP1000 T/H uncertainty analysis, a transient mode calculation was performed for Case F and Case G within the time period that was covered by the MAAP4 calculations for those cases. It was observed that WCOBRA/TRAC predicts higher ADS Stage 4 flows resulting in better depressurization of the primary system. Consequently, the predicted IRWST injection rates were higher when using WCOBRA/TRAC. Because of the faster IRWST draining it was estimated that the IRWST would reach its lowest level about 2 hours earlier than as predicted by MAAP4.

For each of the cases analyzed here (Case F and Case G), the IRWST level calculated by MAAP4 was adjusted to account for the more rapid draining predicted by WCOBRA/TRAC. The adjusted IRWST levels were then used as boundary conditions for each of the cases F and G.

The containment pressure, PXS-B level, IRWST and PXS-B temperatures calculated by MAAP4, together with the adjusted IRWST level, were used to define the limiting conditions that were used to assess the performance of the AP1000 passive safety system.

The following two sections document the results of the WCOBRA/TRAC simulations for these limiting windows performed for Cases F and G.

4. Results

4.1 Case F - DEDVI Line Break in the PXS-B Room with 3 of 4 ADS Stage 4, Containment Isolated

This subsection presents the simulation results of T/H uncertainty Case F – DEDVI line break located in the PXS-B room with 3 out of 4 ADS Stage 4 valves opened and the containment isolated. The initial conditions are based on the MAAP4 calculation results of the same accident scenario. They are selected such that the WCOBRA/TRAC simulation begins 3992 sec (~1 hour 6 min) after the break - shortly after IRWST injection begins.

For this transient, the initial IRWST level is 126.4 feet and its temperature is 121 deg F. The initial level in the PXS-B room is 95.8 feet. The available ADS Stage 4 paths are opened and the containment pressure is set to its initial value of 42.9 psia. Under these conditions, a 1000 second calculation is performed to ensure that the initial steady state conditions are achieved in the system. After that, the transient calculation is initiated with time-dependent boundary conditions taken from the MAAP4 calculation, but with adjusted IRWST level decrease, as discussed earlier.

Initially, the only injection comes from the IRWST into the reactor vessel through the intact DVI injection line (Figure RAI 720.013-14). Since at the beginning of the analysis, the level in the PXS-B room is below the DVI injection nozzle elevation, only steam from downcomer is vented out through the break (Figure RAI 720.023-13). Water starts to flow back into the downcomer through the broken DVI line about 2 hours into the transient. This is the time when the level in the PXS-B room becomes high enough to provide sufficient driving head. At the onset of this event the additional amount of water supplied into the downcomer through the DVI break supplement's the IRWST injection. This leads to enhanced core cooling and momentarily, faster depressurization occurs at about 2.05 hours into the transient (see Figure RAI 720.013-11). Consequently, the IRWST injection is increased even further and as a result, the levels in the downcomer (Figures RAI 720.013-1), the reactor core (Figures RAI 720.013-2) and the upper plenum (Figures RAI 720.013-8) are also increased. The effect of this injection flow increase can also be seen on Figure RAI 720.013-4, which shows a sharp void fraction decrease in the upper half of the fuel region.

The available 3 out of 4 ADS Stage 4 valves provide enough venting capacity to assure adequate depressurization and successful performance of the passive safety systems (Figures RAI 720.013-9 and RAI 720.013-10). The fuel remains covered throughout the transient and adequate core cooling is provided to remove the decay heat. The Hot Rod cladding temperature is about 20 deg F above saturation (Figure RAI 720.013-12) and is steadily decreasing.

As the transient proceeds, the IRWST drains to a minimum of 107 feet at about 3.9 hours after the break. After that time, the level is kept constant at 107 feet, as predicted by MAAP4. The transient is terminated at about 4.2 hours after the break with the system in a continuing depressurization phase with stable DVI injection flows, and decreasing decay heat.

4.2 Case G - DEDVI Line Break in the PXS-B Room with 4 of 4 ADS Stage 4, Containment Isolation Failed

This subsection presents the simulation results of T/H uncertainty Case G – DEDVI line break located in the PXS-B room with all ADS Stage 4 valves available and with containment isolation failure. The initial conditions are based on the MAAP4 calculation results of the same accident scenario. They are selected such that the WCOBRA/TRAC simulation begins 3298 sec (~55 min) after the break - shortly after IRWST injection begins.

For this transient, the initial IRWST level is 127.9 feet and its temperature is 120.5 deg F. The initial level in the PXS-B room is 93.1 feet. All the ADS Stage 4 paths are opened and the containment pressure is set to its initial value of 17.08 psia, as calculated by MAAP4. Under these conditions, first a 1000 seconds calculation is performed so that the initial steady state is achieved in the system. After that, the transient calculation is initiated with time-dependent boundary conditions taken from the MAAP4 calculation, but with the adjusted IRWST level decrease.

Initially, the only injection comes from the IRWST into the reactor vessel through the intact DVI injection line (Figure RAI 720.013-28). Since at the beginning of the analysis, the level in the PXS-B room is below the DVI injection nozzle elevation, only steam from downcomer is vented out through the break (Figure RAI 720.023-27). Water starts to flow back into the downcomer through the broken DVI line about 2 hours into the transient. This is the time when the level in the PXS-B room becomes high enough to provide sufficient driving head for this to happen. This time, unlike the Case F DVI break scenario, the transition into reversed injection of water through the break into the downcomer occurs a little earlier, and is somewhat softer. As a result, the increased depressurization rate observed in Case F does not occur. Still, the levels in the downcomer (Figures RAI 720.013-15), the reactor core (Figures RAI 720.013-16) and the upper plenum (Figures RAI 720.013-22) are maintained high enough by the available DVI injection.

The availability of all ADS Stage 4 valves provides enough venting capacity to assure adequate depressurization and successful performance of the passive safety systems (Figures RAI 720.013-23 and RAI 720.013-24). The fuel remains covered throughout the transient and adequate core cooling is provided to remove the decay heat. The Hot Rod cladding temperature is about 20 deg F above saturation (Figure RAI 720.013-26) and steadily decreasing.

As the transient proceeds, the IRWST drains to a minimum of 106.9 feet at about 3.7 hours after the break. After that time, the level is kept constant at 106.9 feet, as predicted by MAAP4. The transient is terminated at about 4.4 hours after the break with the system being in a phase with stable DVI injection flows, adequate ADS 4 flows, and decreasing decay heat.

5. References

1. WCAP-14800, *AP600 PRA Thermal/Hydraulic Uncertainty Evaluation for Passive System Reliability*, June 1997.

2. WCAP-12945, *Code Qualification Document for Best Estimate Analysis*, Volumes 1 through 5, Revision 1 (Westinghouse Proprietary).

3. *AP1000 Design Control Document*, Chapter 15.6.5.4C, Revision 2.

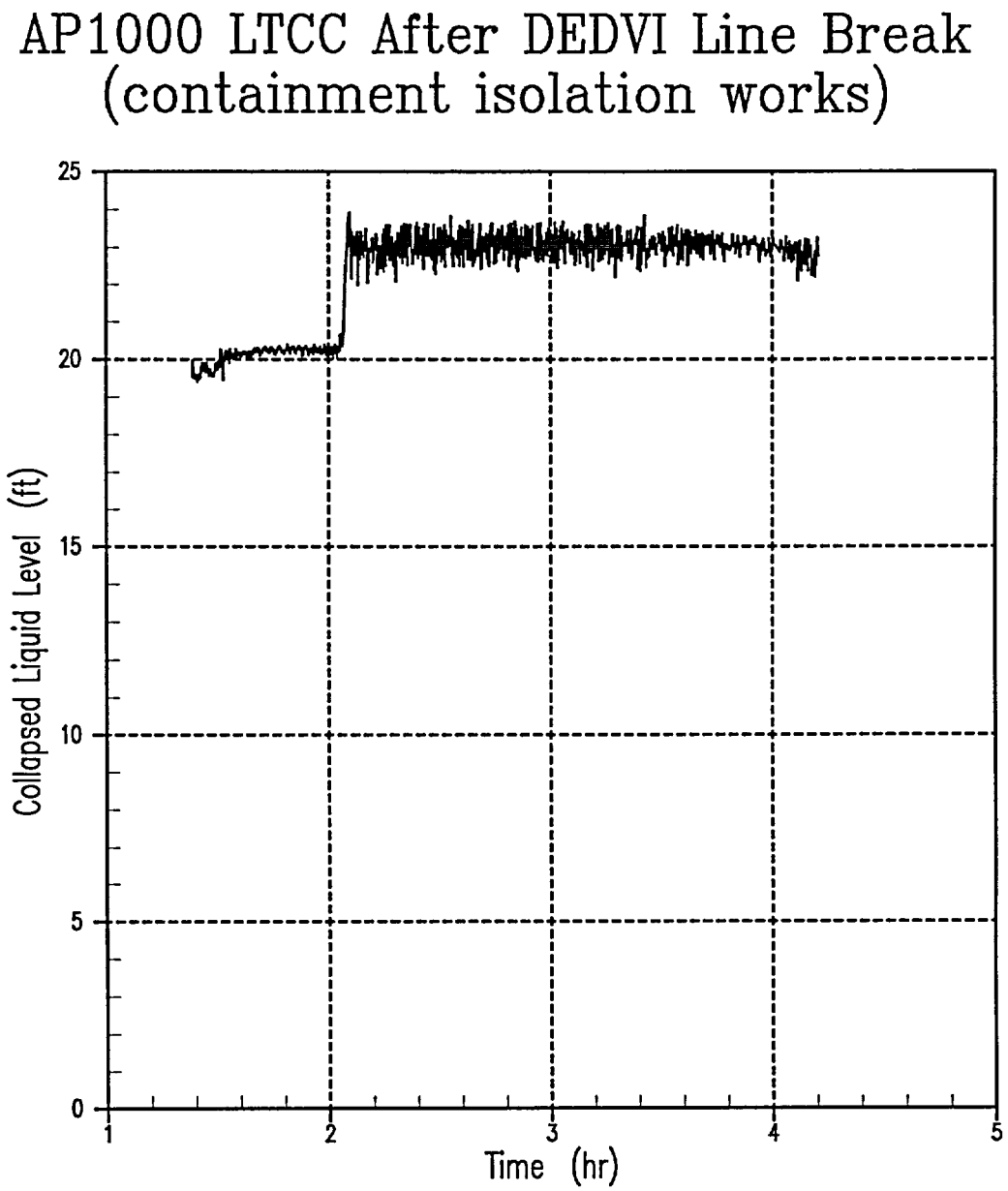


Figure RAI 720.013-1
Collapsed Level of Liquid in the Downcomer (Case F)

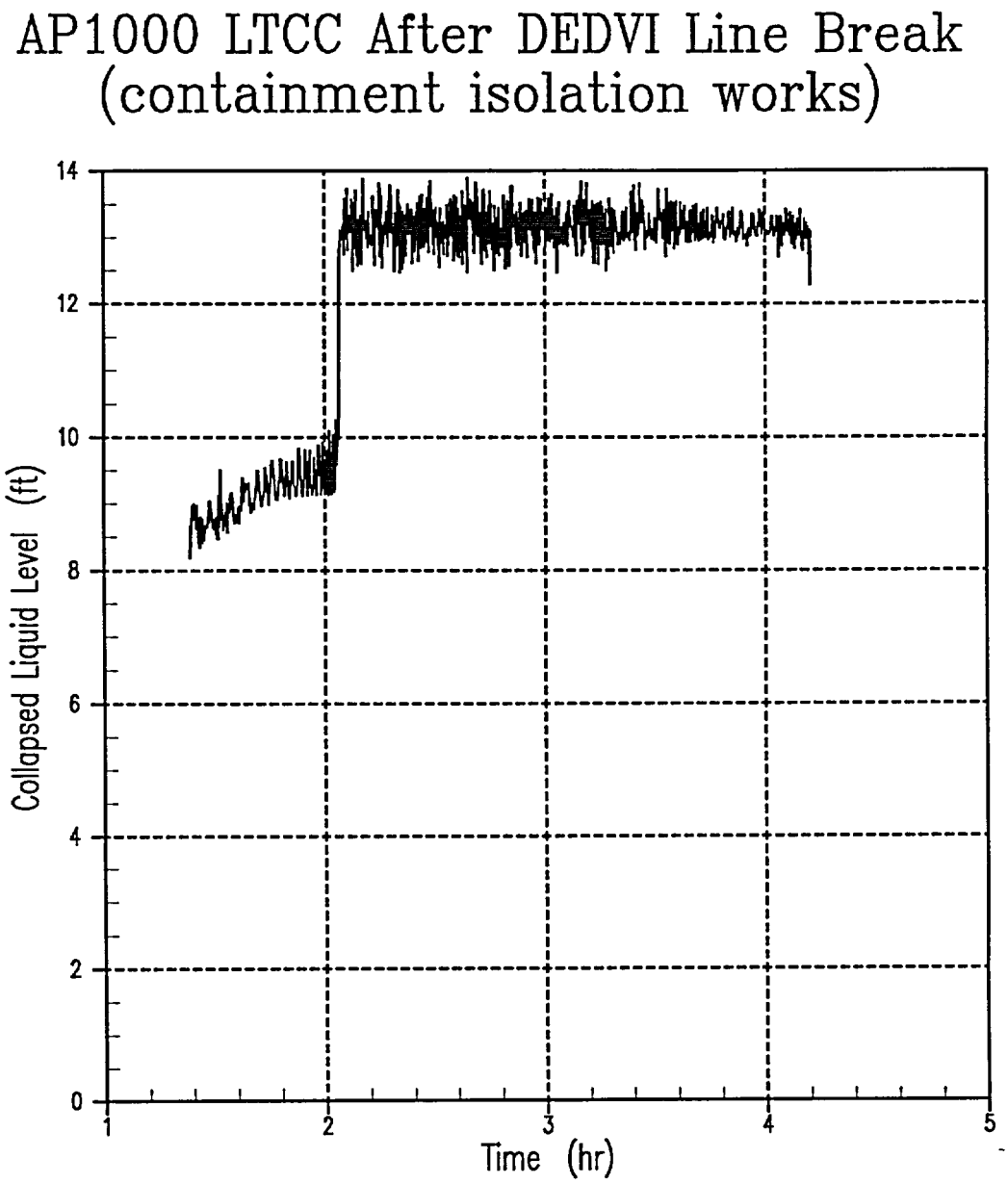


Figure RAI 720.013-2
Collapsed Level of Liquid Over the Heated Length of the Fuel (Case F)

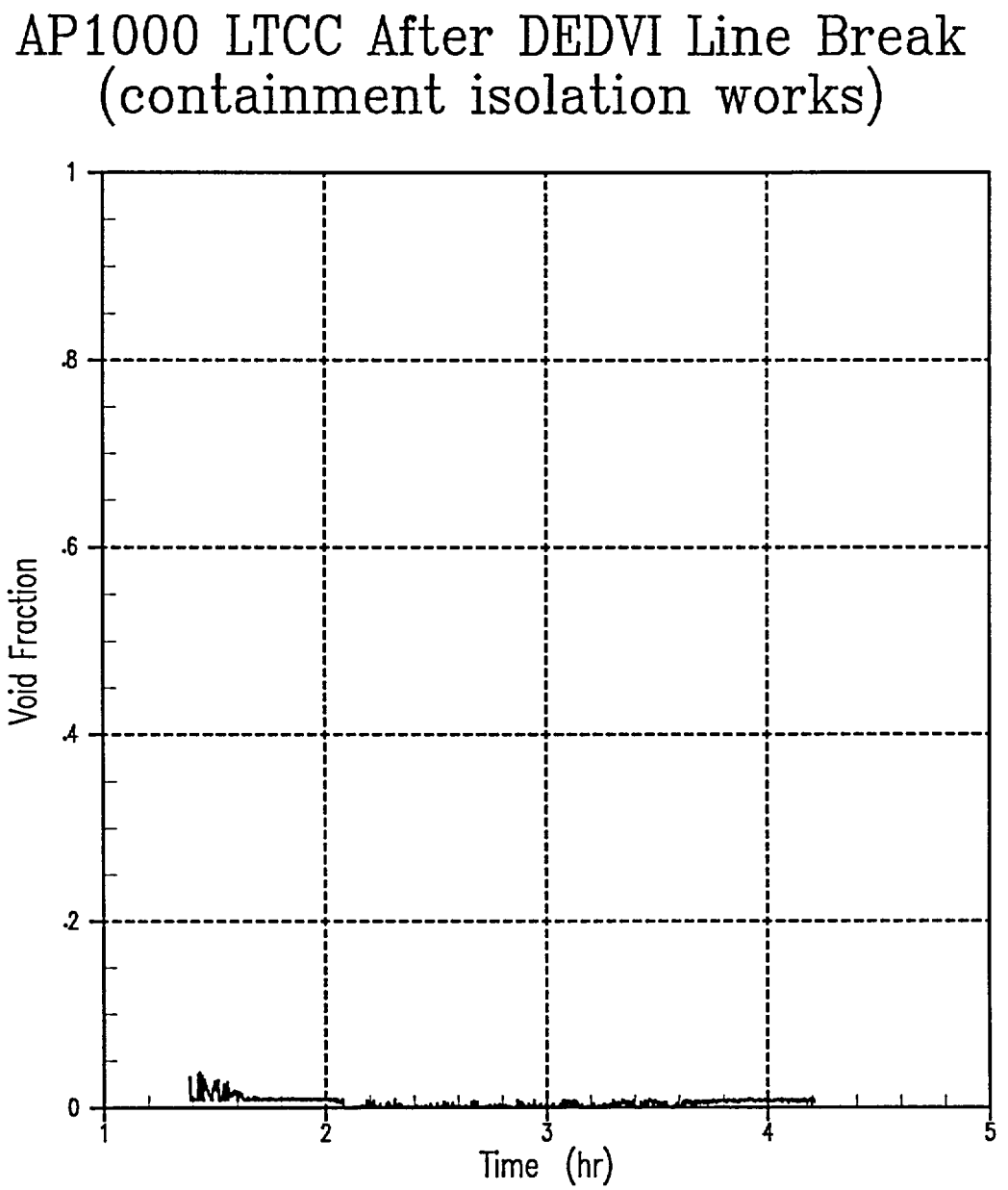


Figure RAI 720.013-3
Void Fraction in Core Cell Level 1 of 2 (Case F)

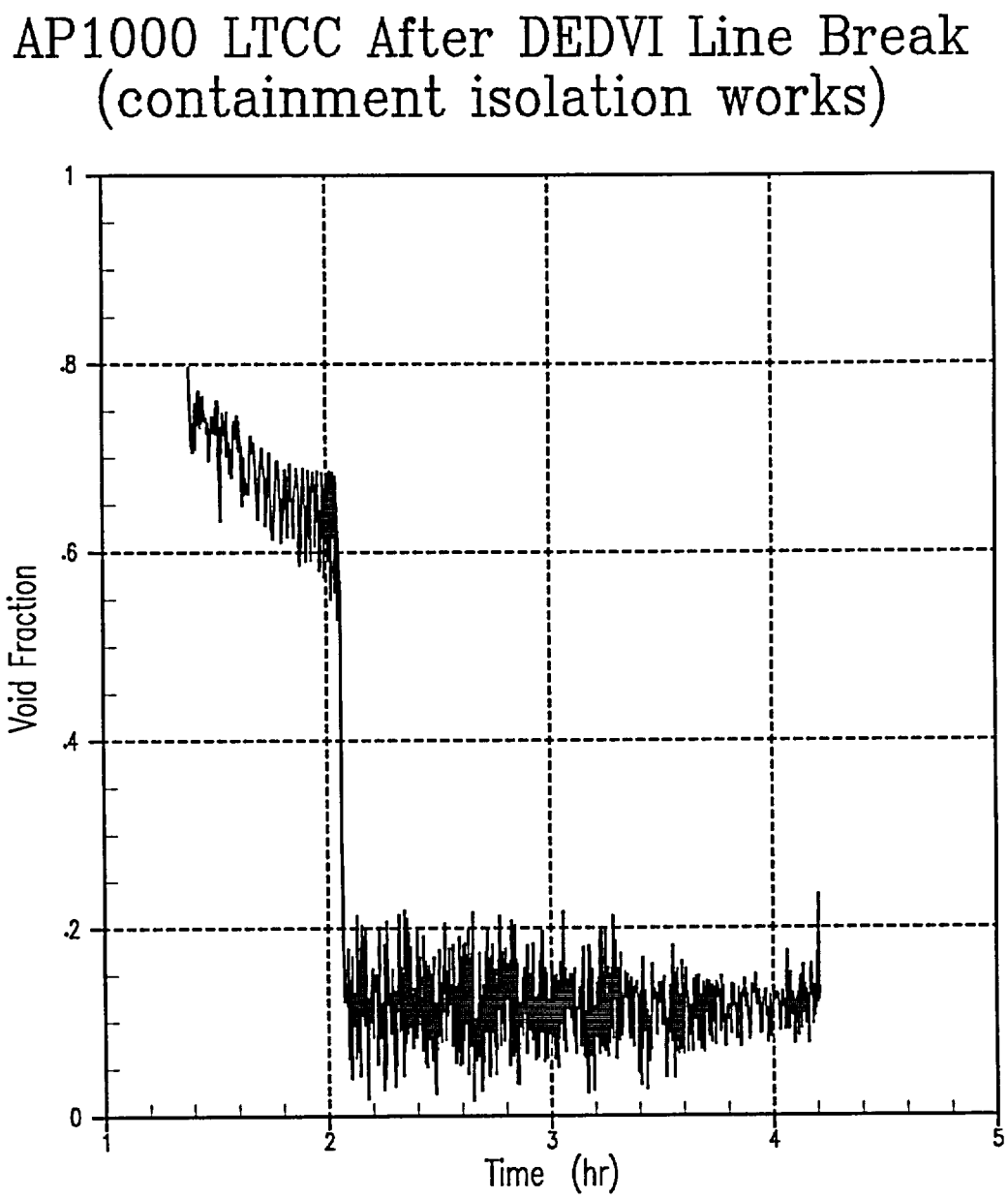


Figure RAI 720.013-4
Void Fraction in Core Cell Level 2 of 2 (Case F)

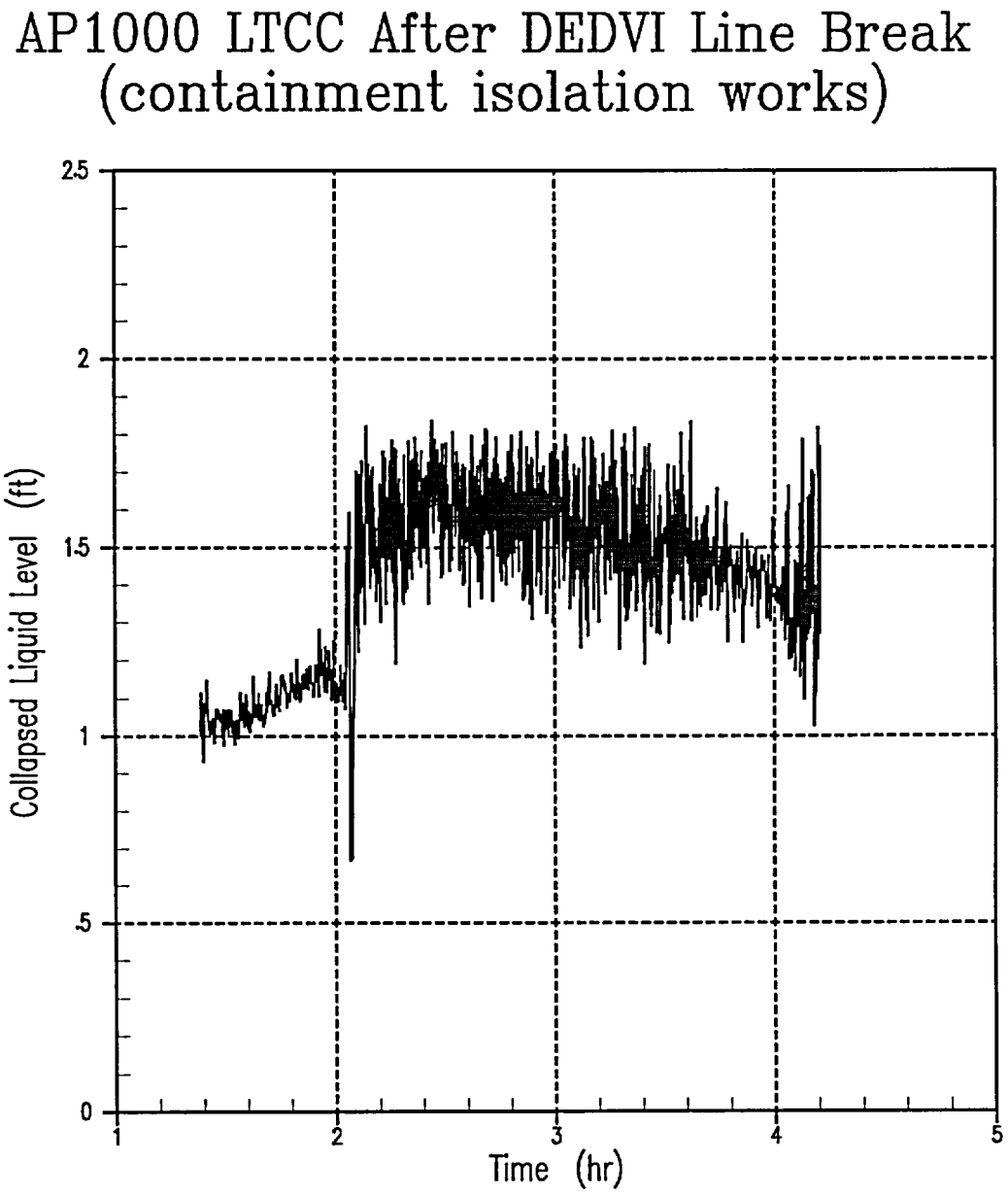


Figure RAI 720.013-5
Collapsed Liquid Level in the Hot Leg of Pressurizer Loop (Case F)

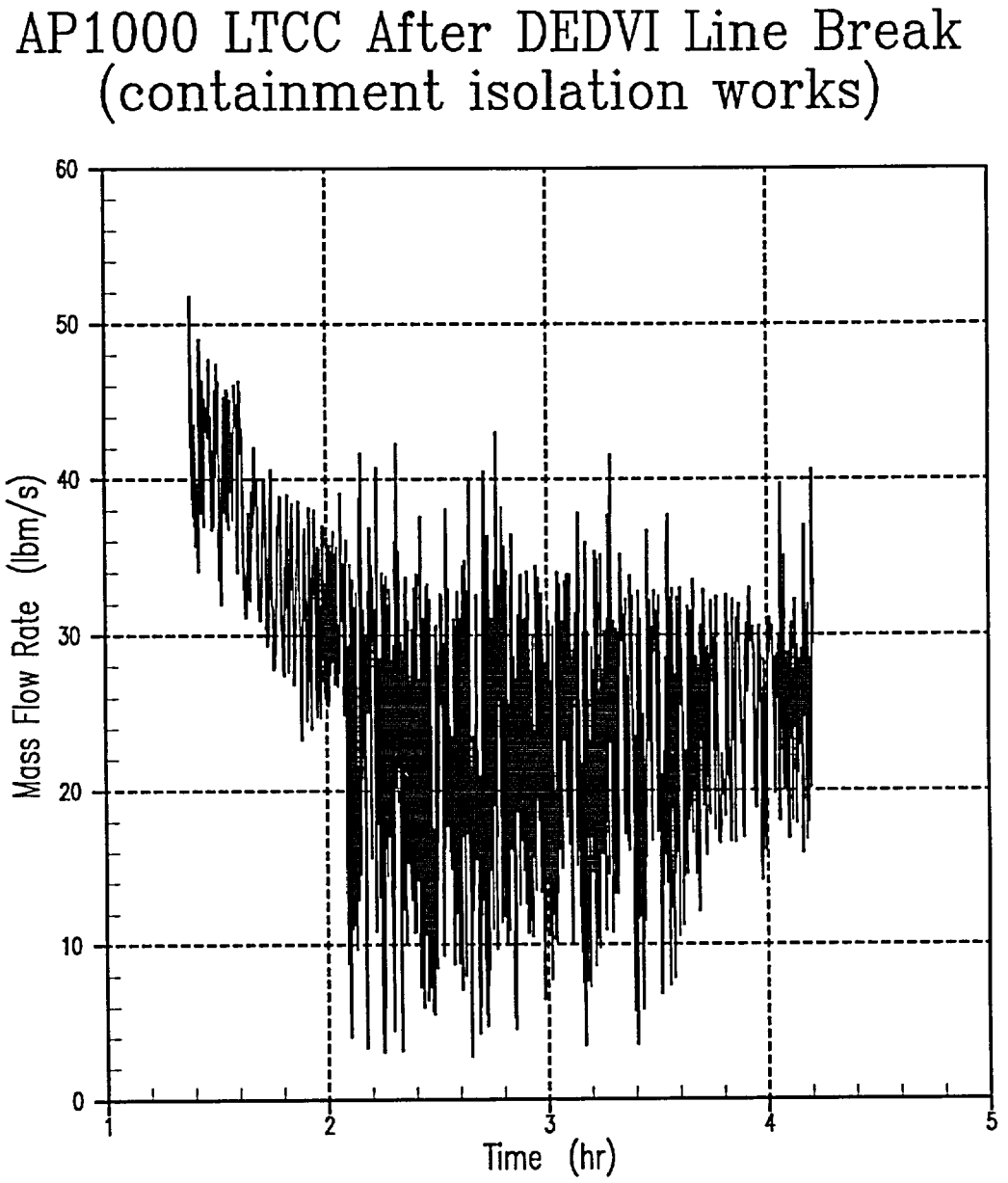


Figure RAI 720.013-6
Vapor Rate out of the Core (Case F)

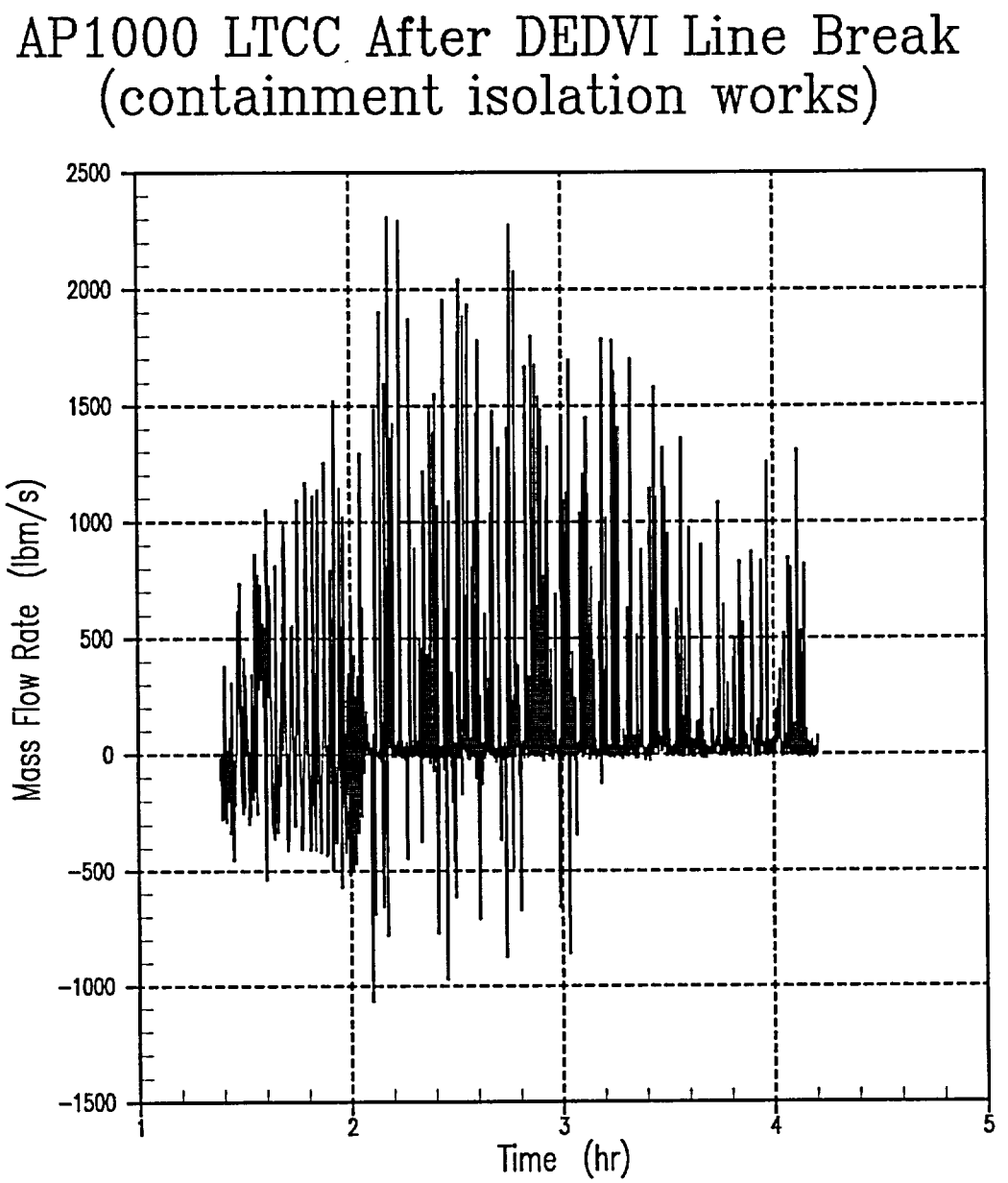


Figure RAI 720.013-7
Liquid Flow Rate Out of the Core (Case F)

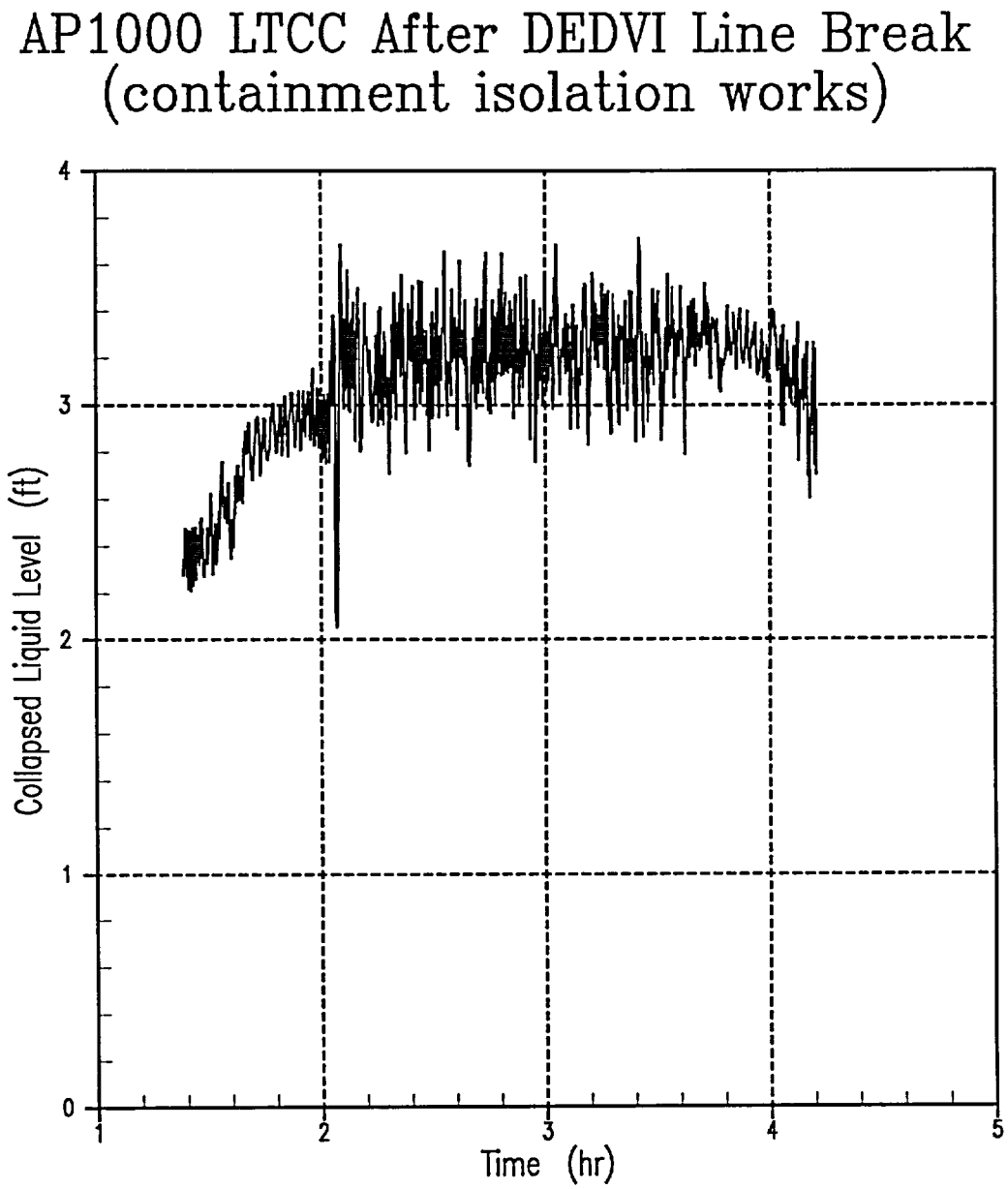


Figure RAI 720.013-8
Collapsed Liquid Level in the Upper Plenum (Case F)

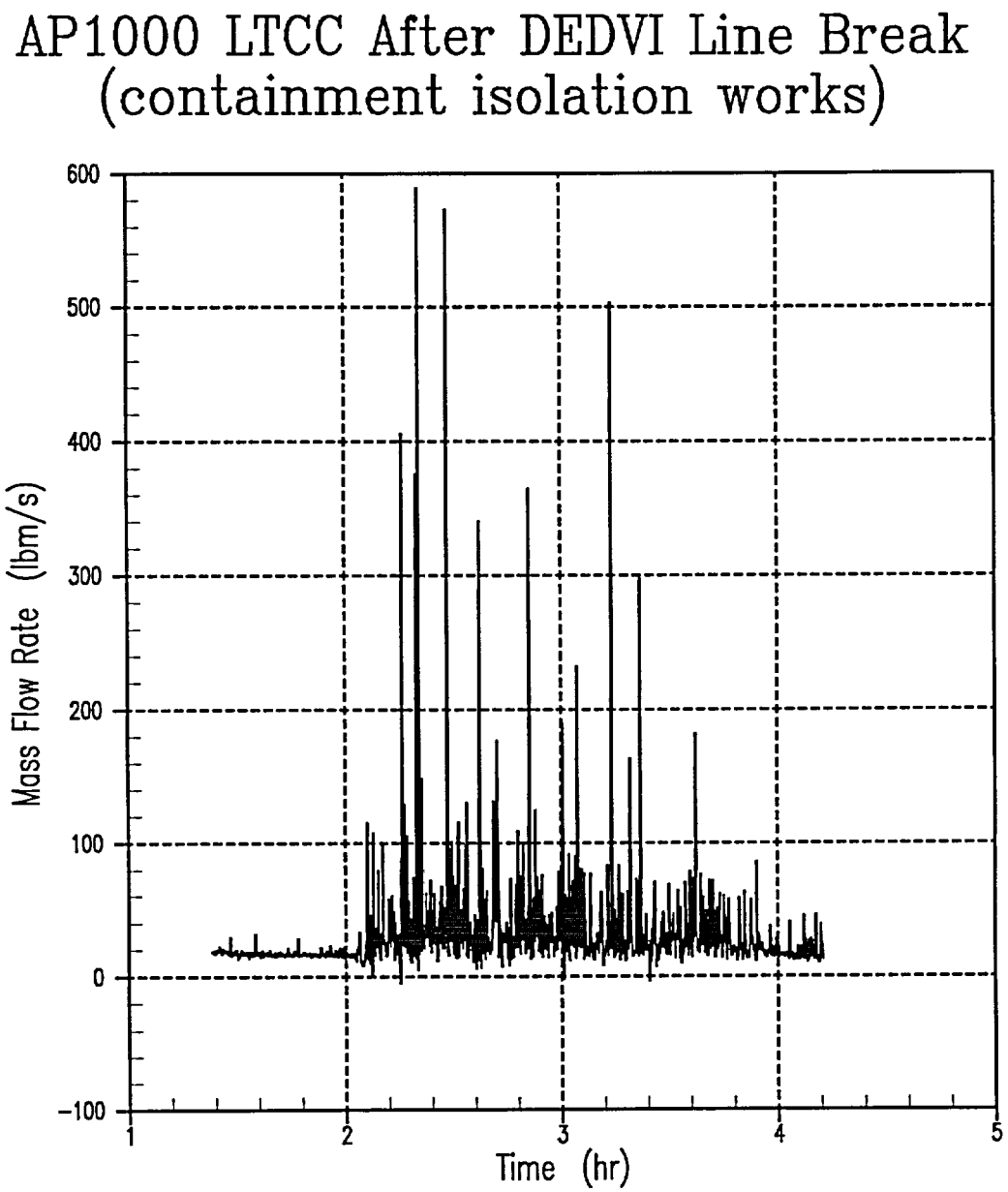


Figure RAI 720.013-9
Mixture Flowrate Through ADS Stage 4A Valves (Case F)

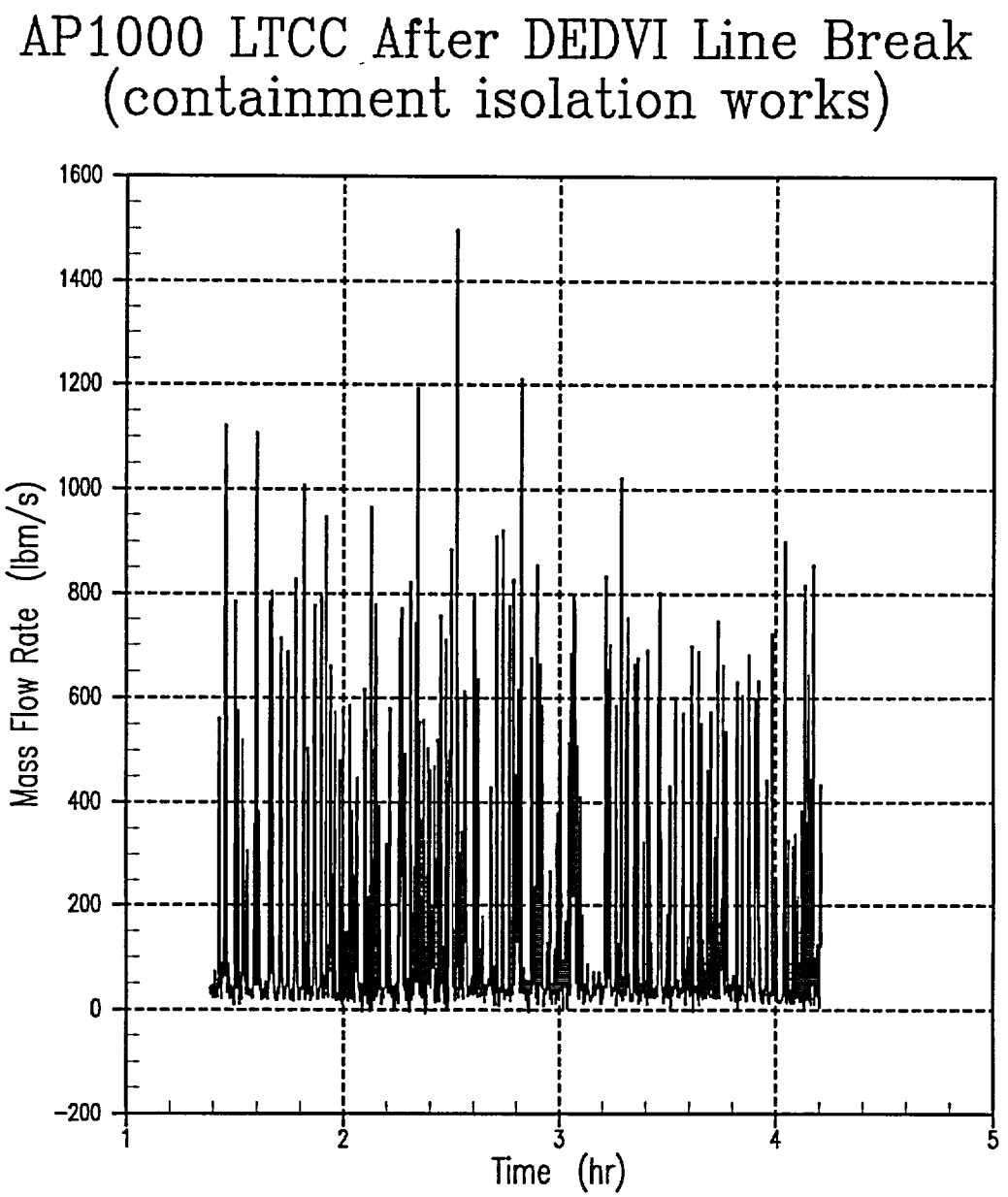


Figure RAI 720.013-10
Mixture Flowrate Through ADS Stage 4B Valves (Case F)

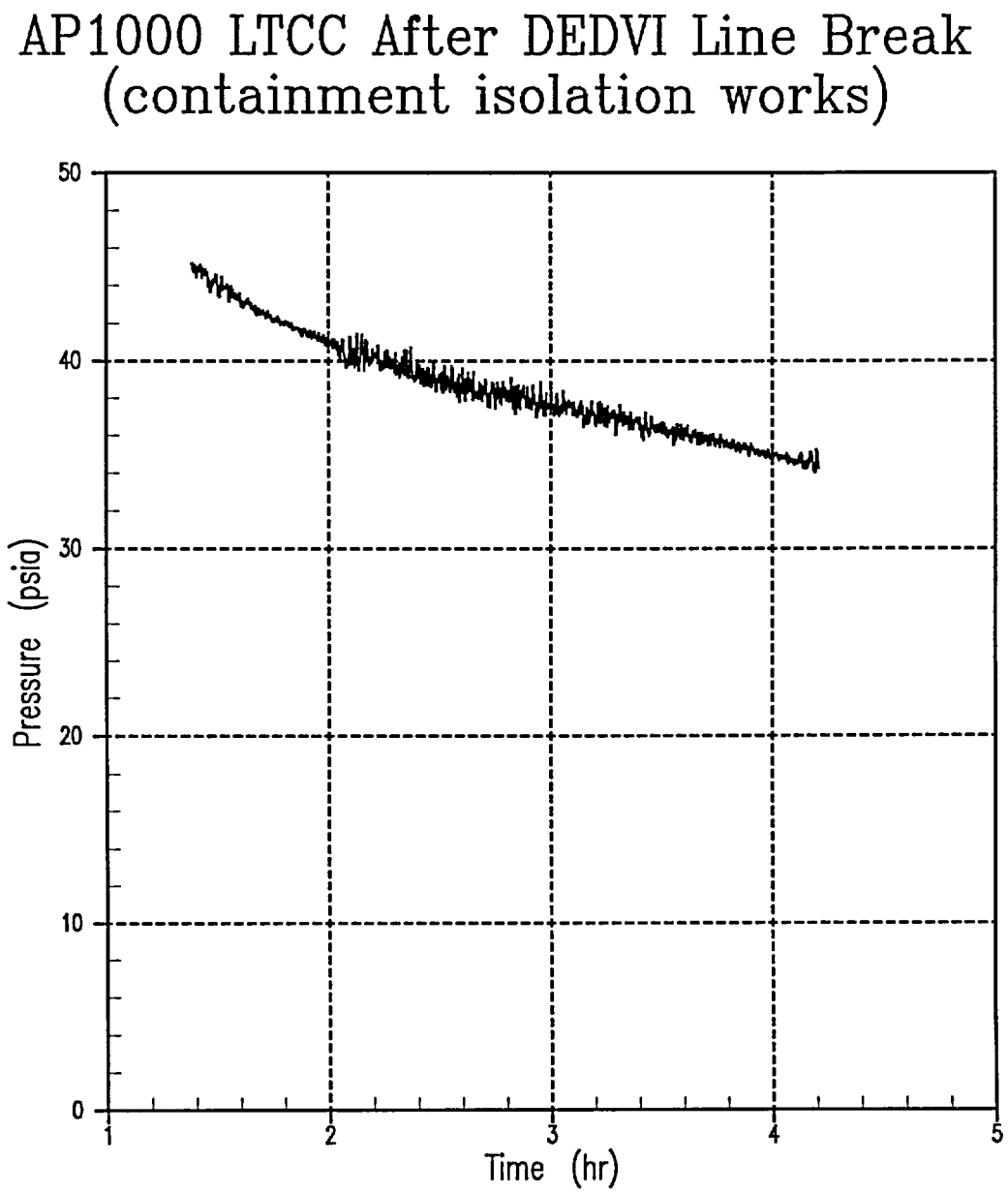


Figure RAI 720.013-11
Upper Plenum Pressure (Case F)

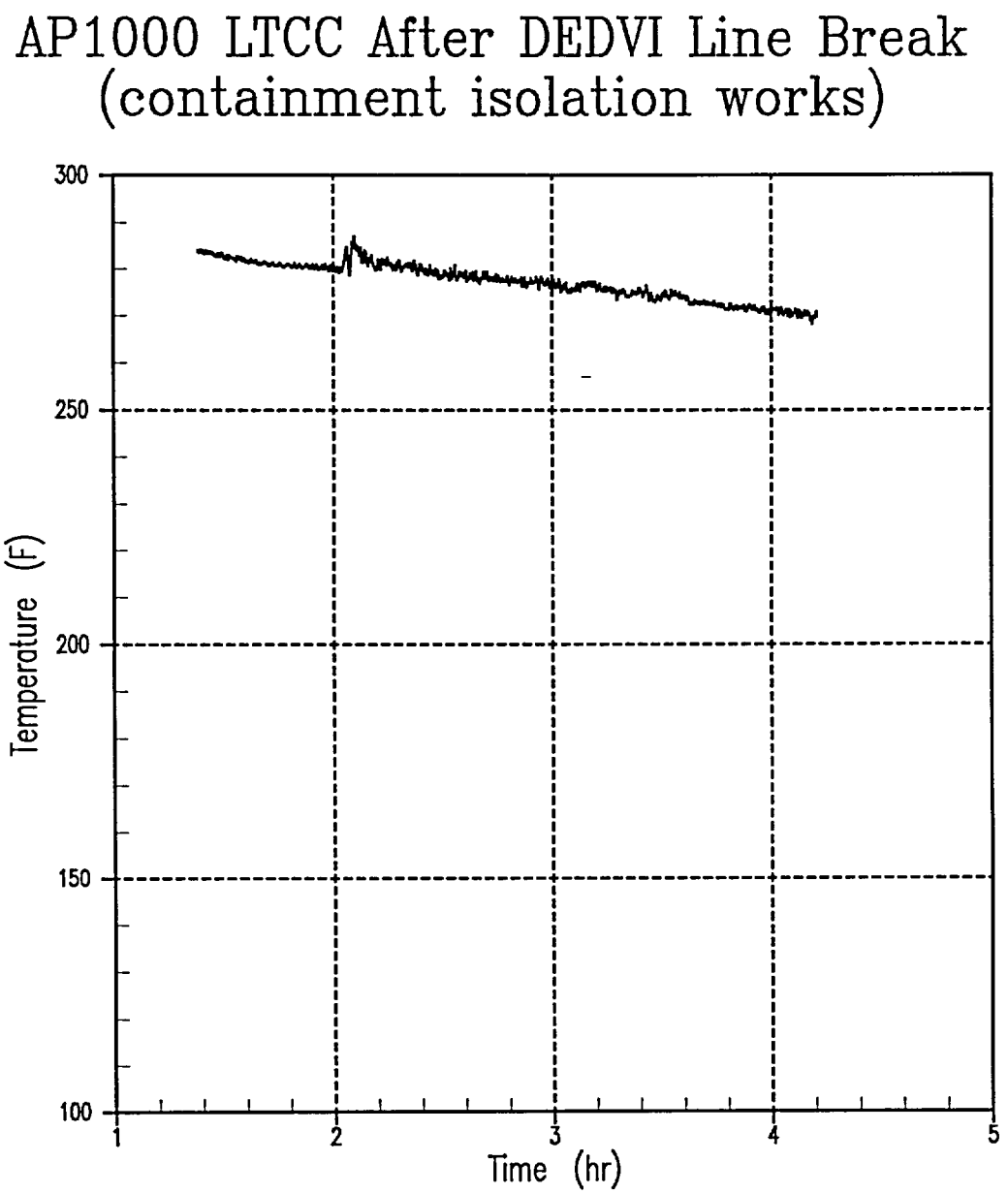


Figure RAI 720.013-12
PCT of the Hot Rod (Case F)

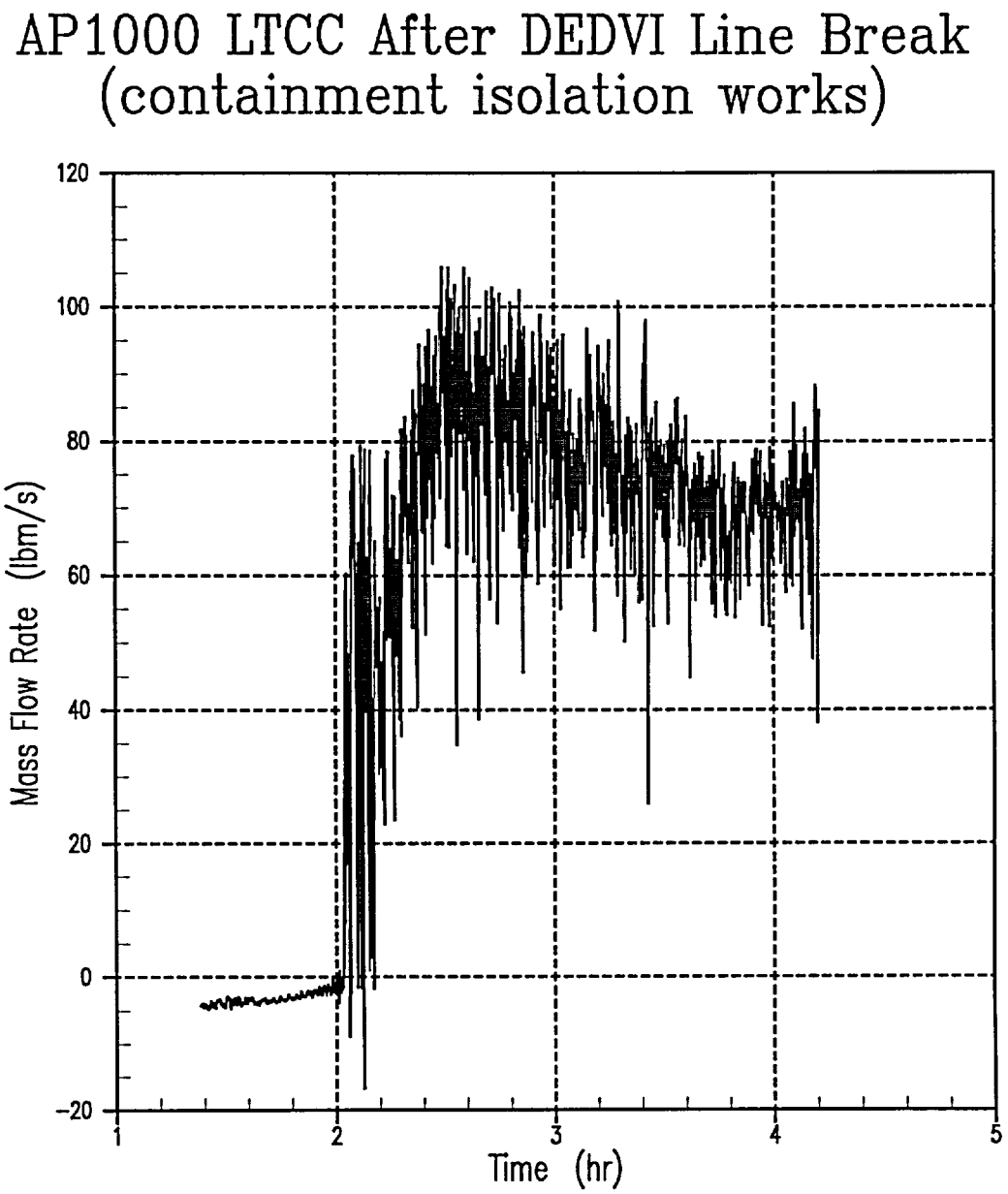


Figure RAI 720.013-13
DVI-A Mixture Flow Rate (Case F)

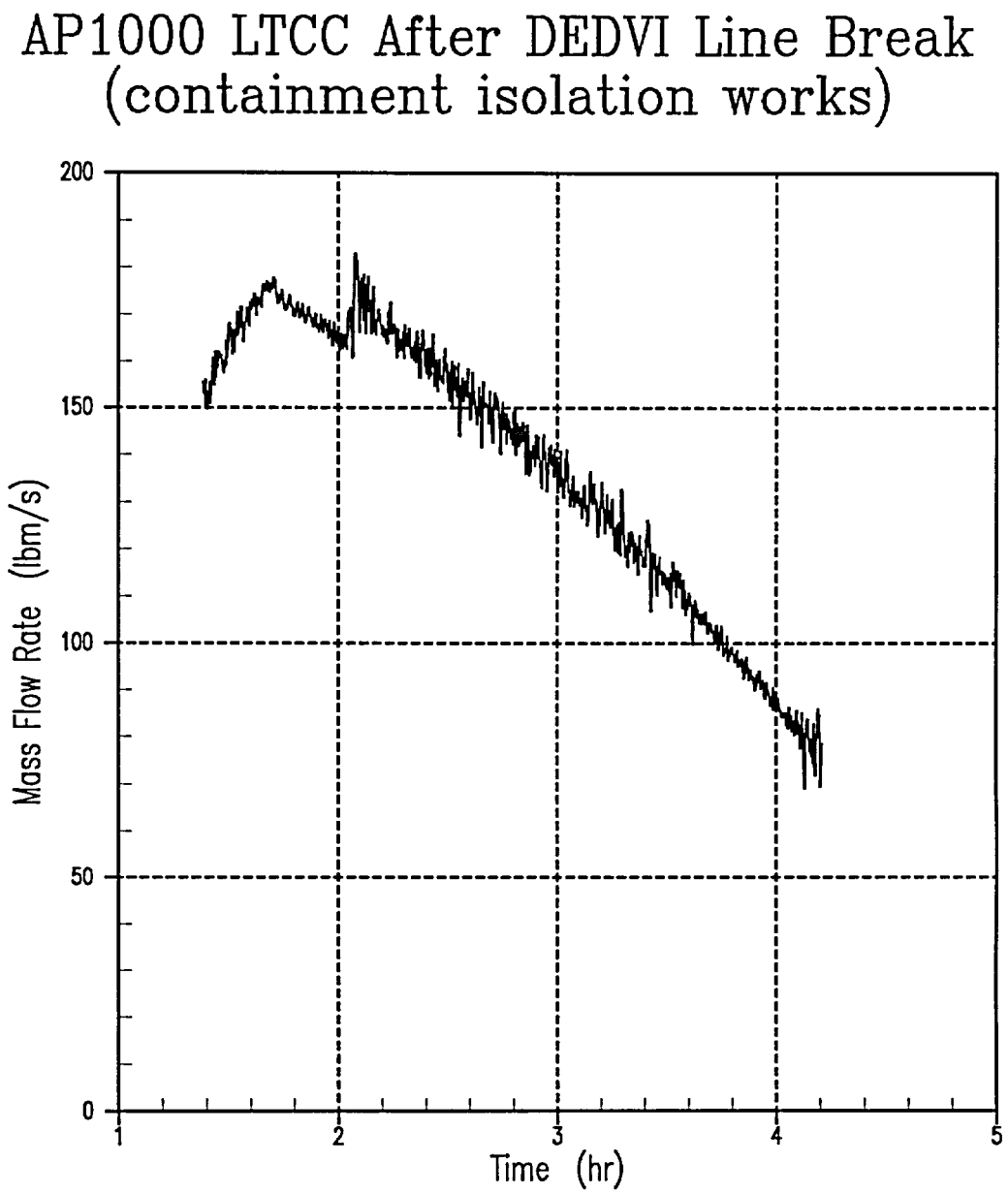


Figure RAI 720.013-14
DVI-B Mixture Flow Rate (Case F)

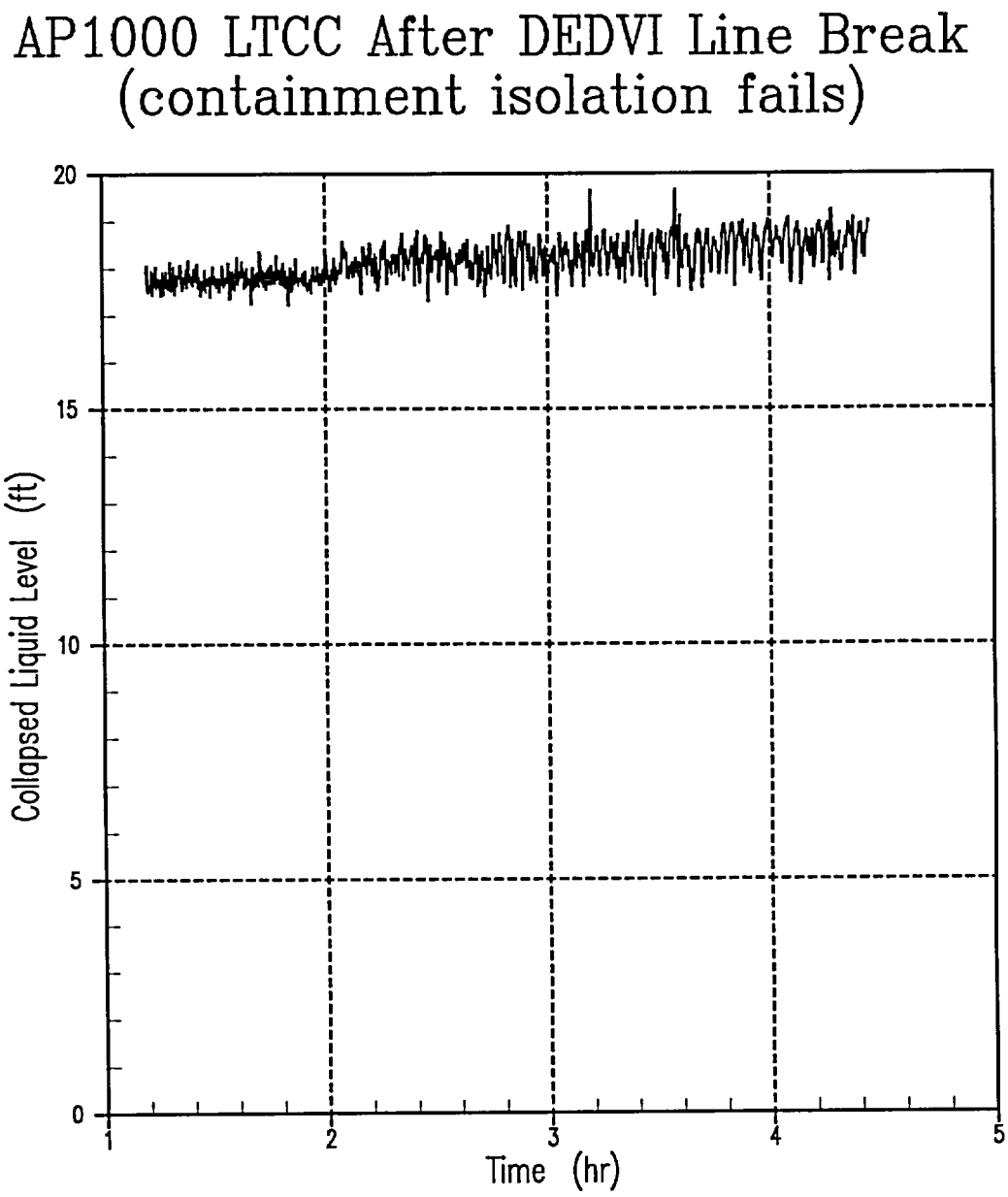


Figure RAI 720.013-15
Collapsed Level of Liquid in the Downcomer (Case G)

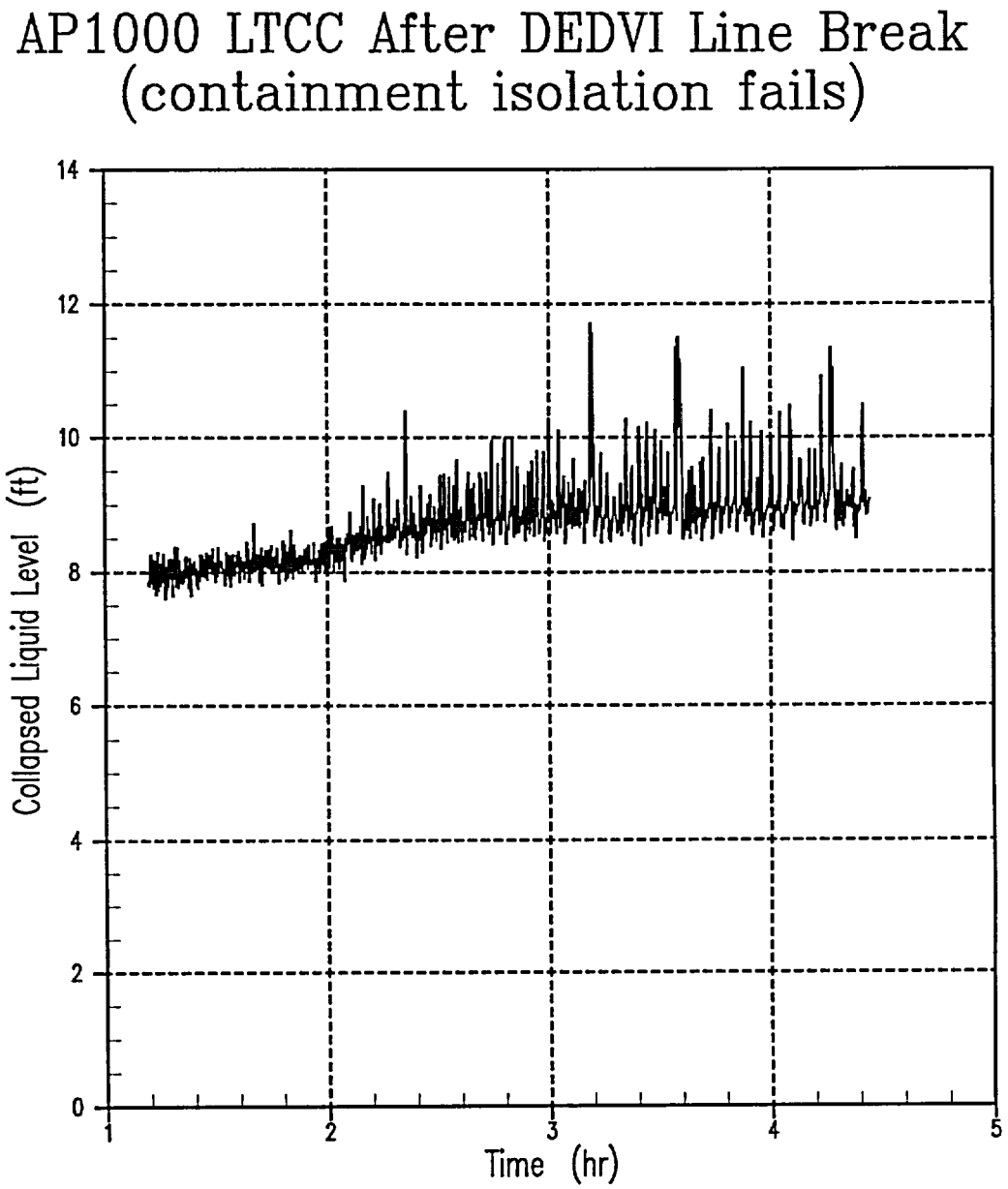


Figure RAI 720.013-16
Collapsed Level of Liquid Over the Heated Length of the Fuel (Case G)

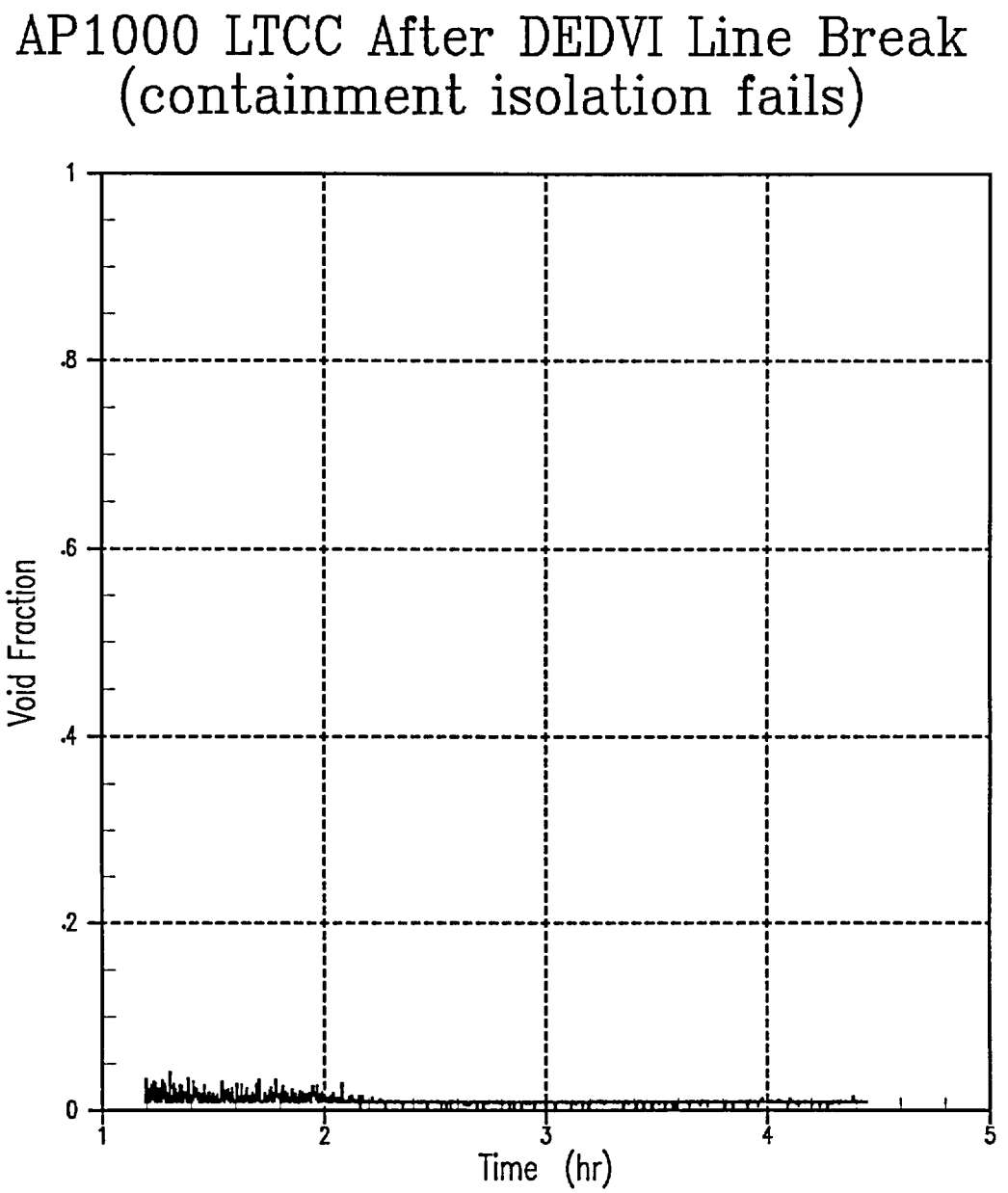


Figure RAI 720.013-17
Void Fraction in Core Cell Level 1 of 2 (Case G)

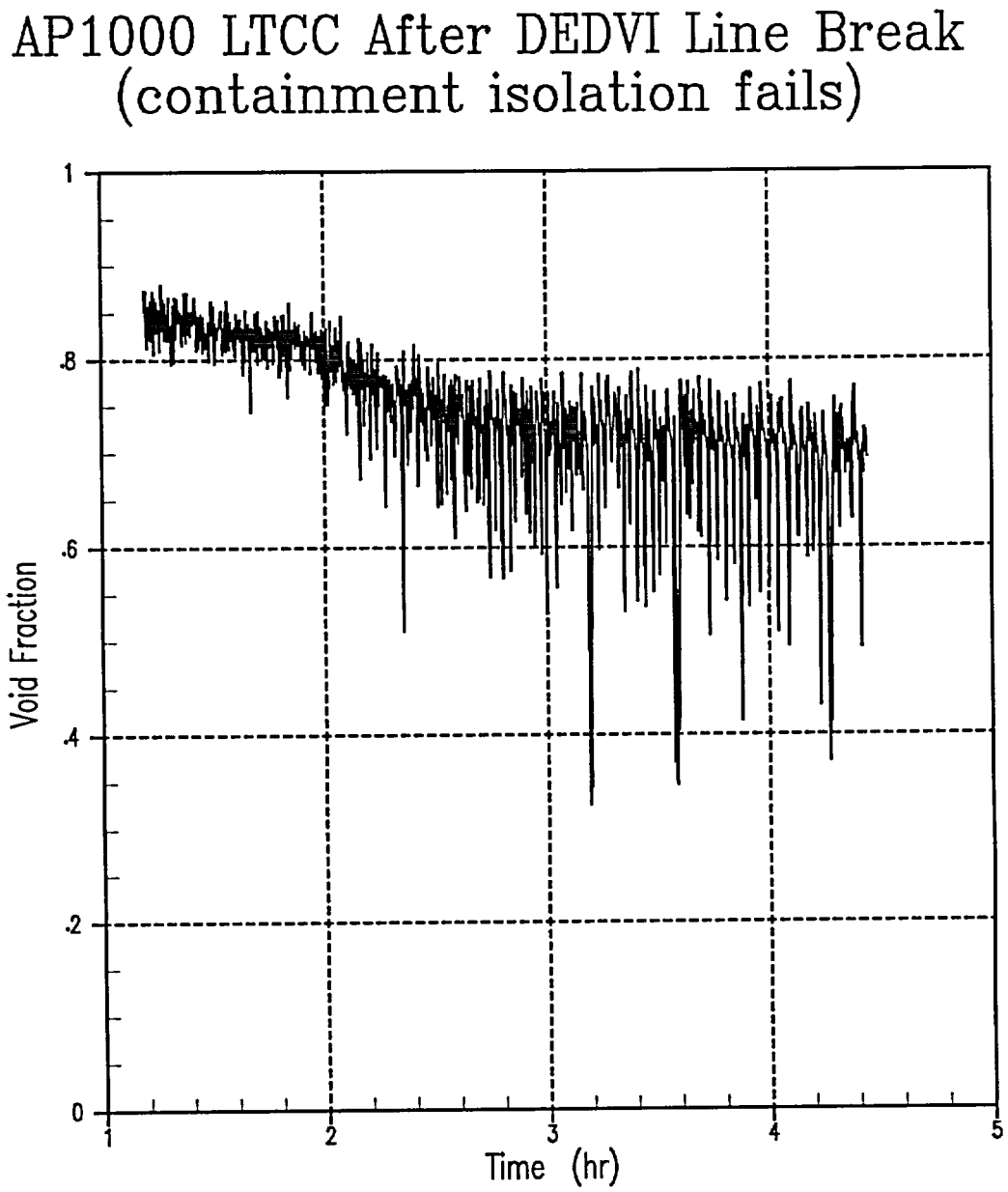


Figure RAI 720.013-18
Void Fraction in Core Cell Level 2 of 2 (Case G)

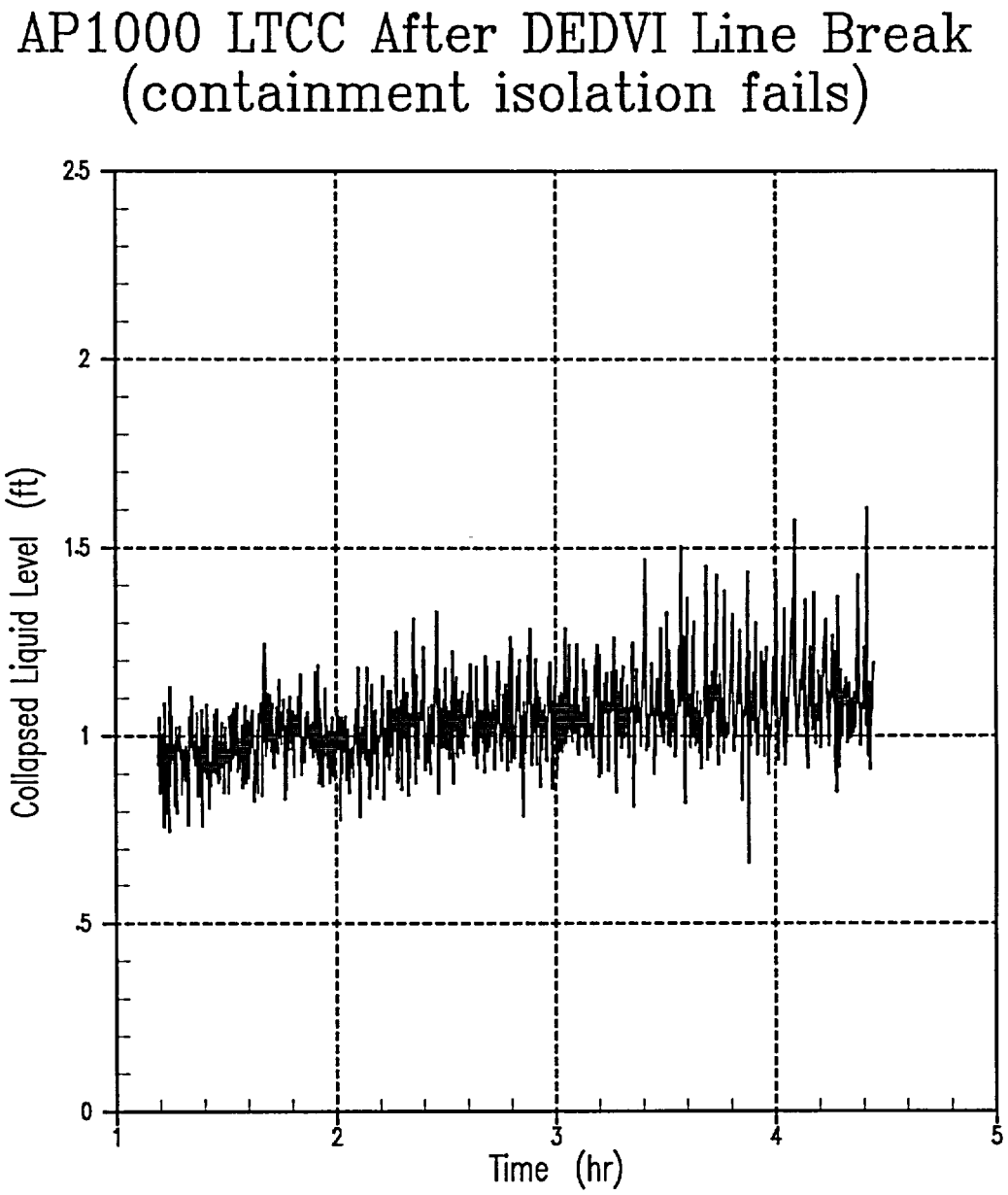


Figure RAI 720.013-19
Collapsed Liquid Level in the Hot Leg of Pressurizer Loop (Case G)

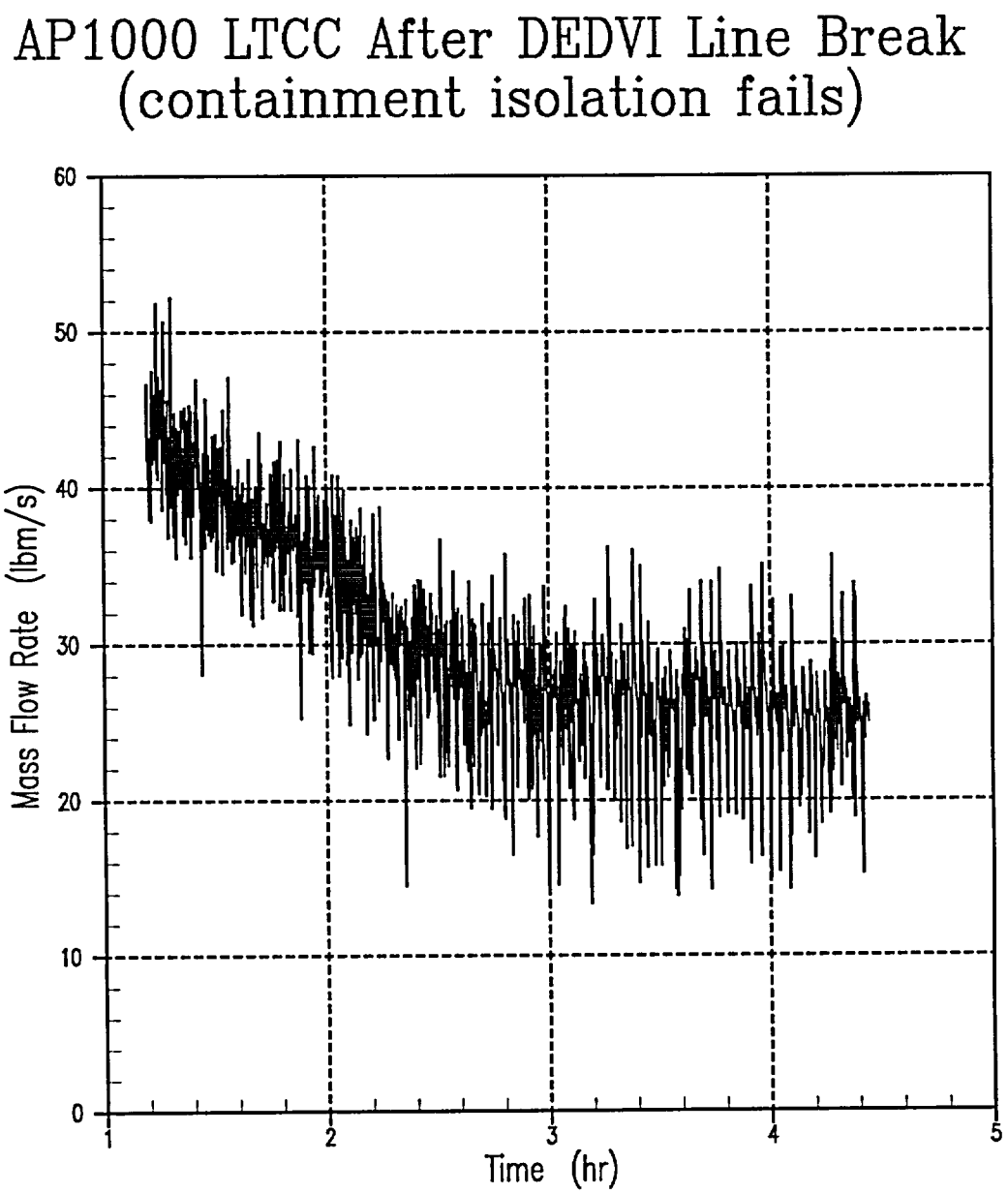


Figure RAI 720.013-20
Vapor Rate out of the Core (Case G)

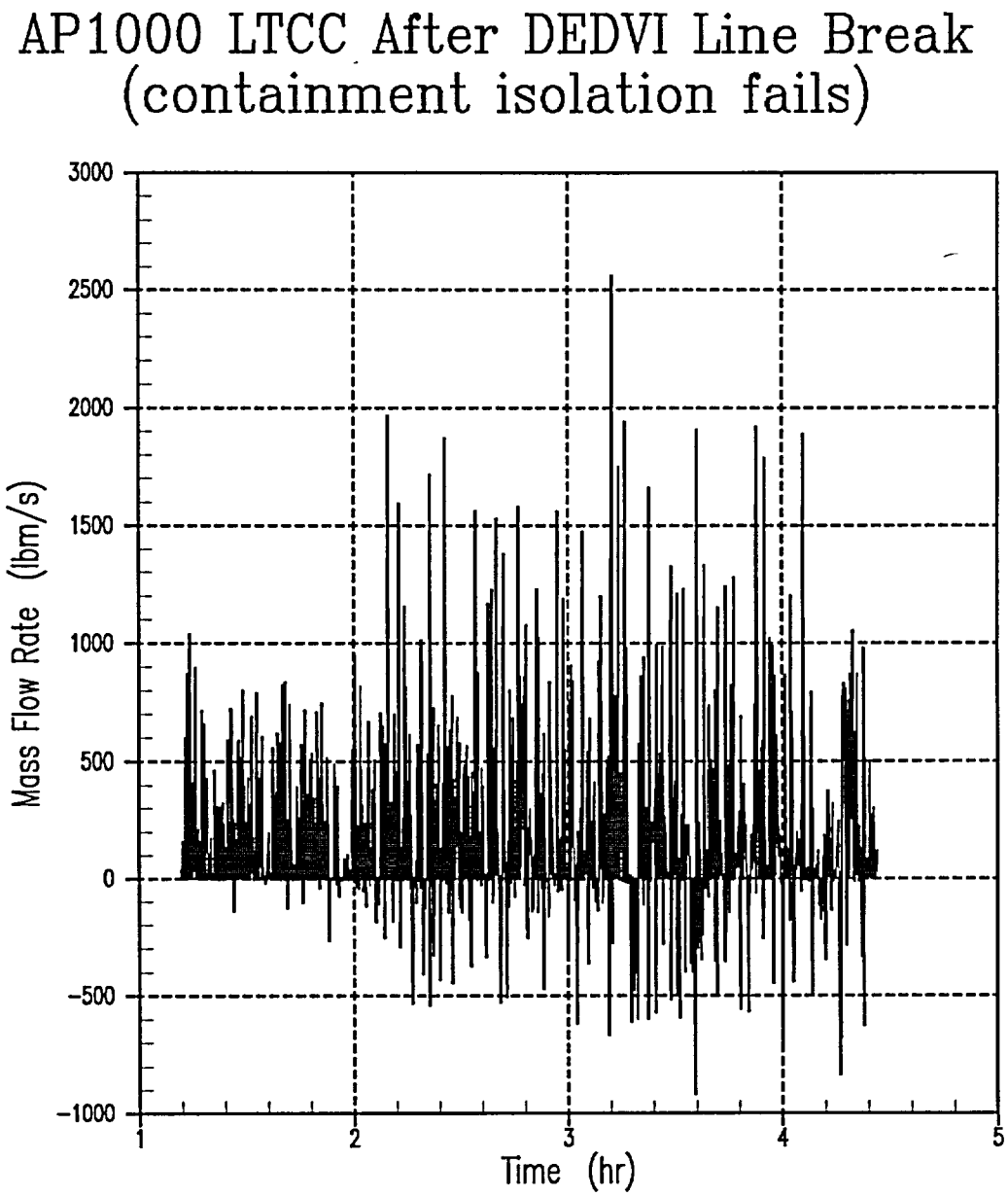


Figure RAI 720.013-21
Liquid Flow Rate Out of the Core (Case G)

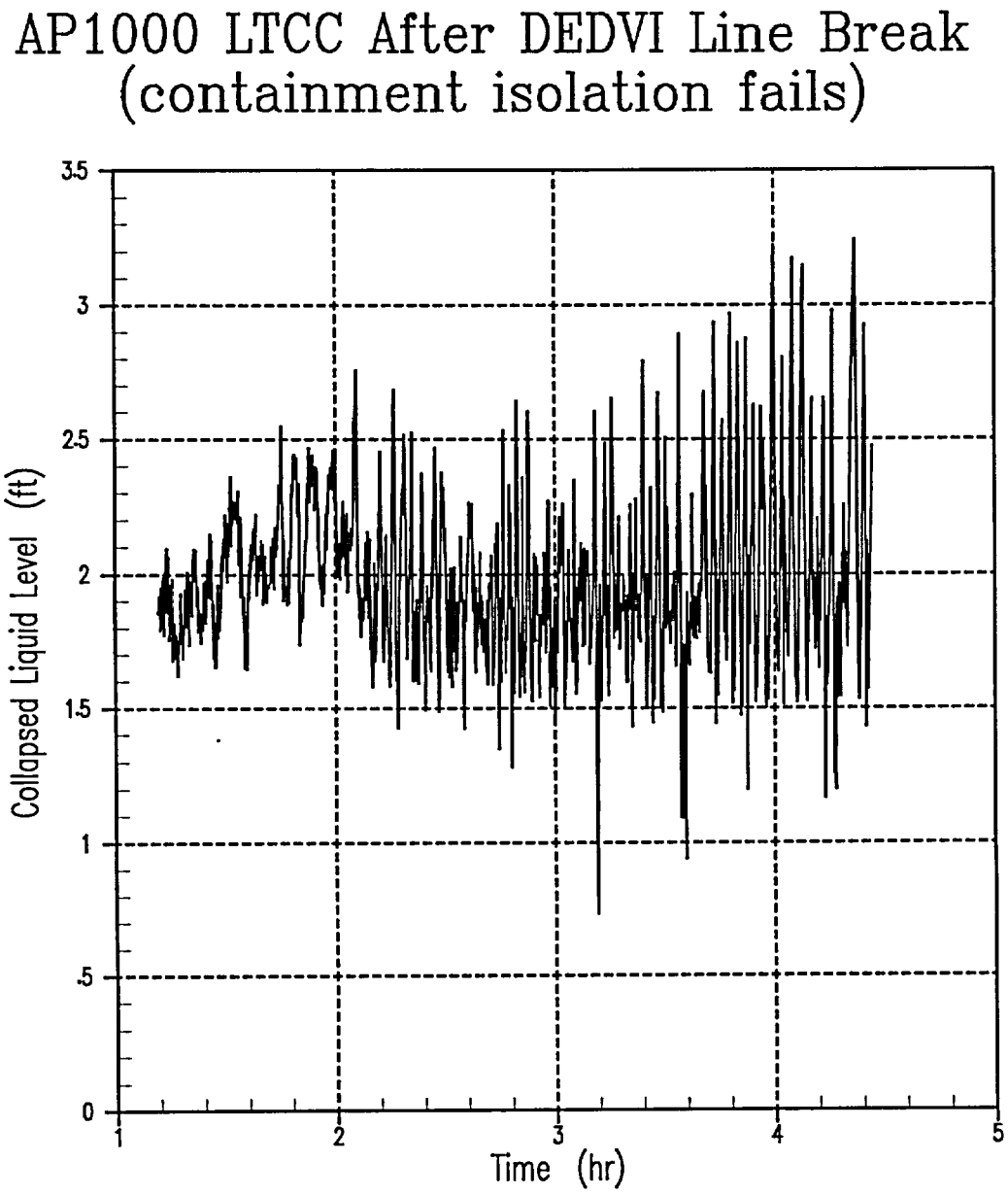


Figure RAI 720.013-22
Collapsed Liquid Level in the Upper Plenum (Case G)

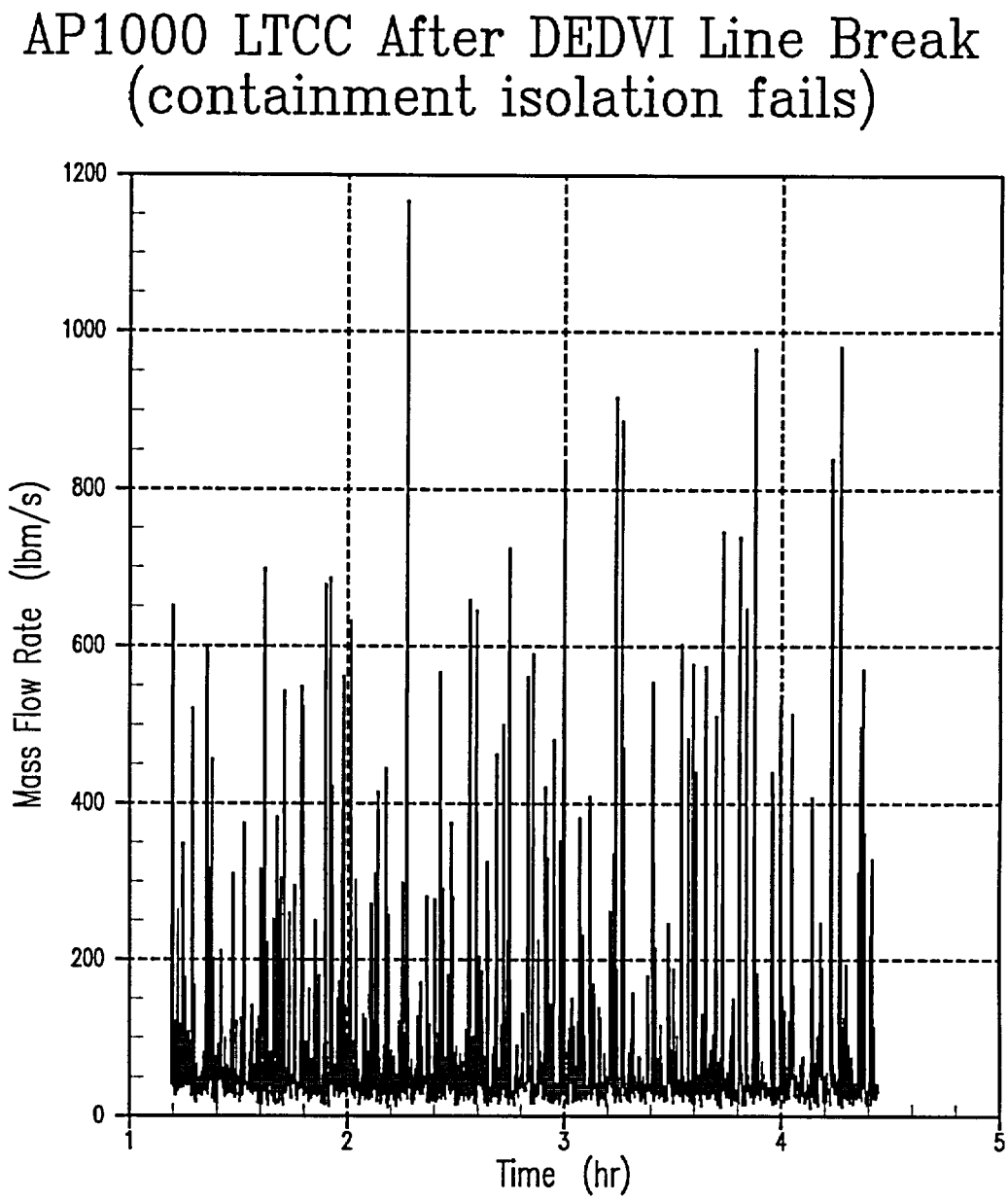


Figure RAI 720.013-23
Mixture Flowrate Through ADS Stage 4A Valves (Case G)

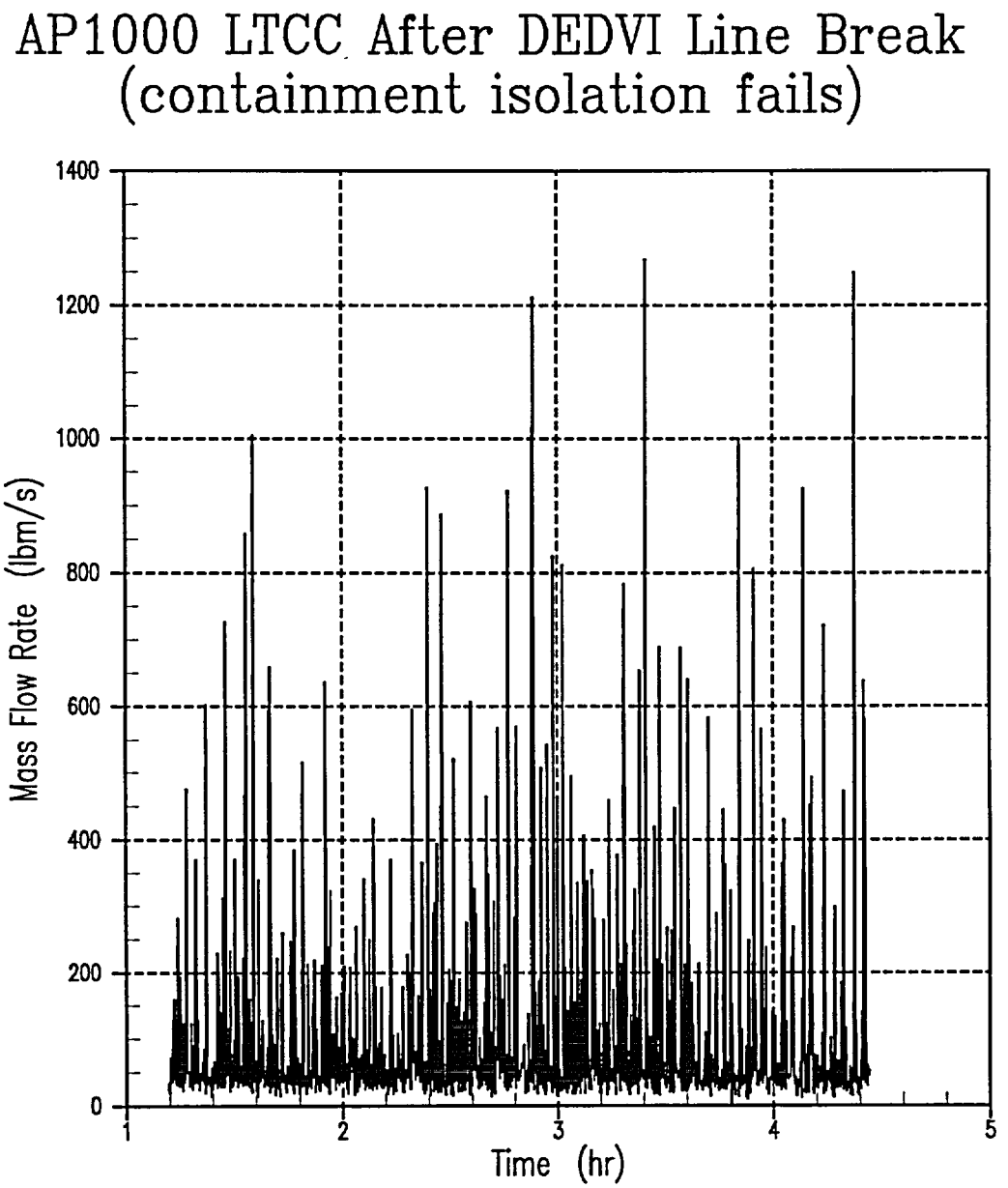


Figure RAI 720.013-24
Mixture Flowrate Through ADS Stage 4B Valves (Case G)

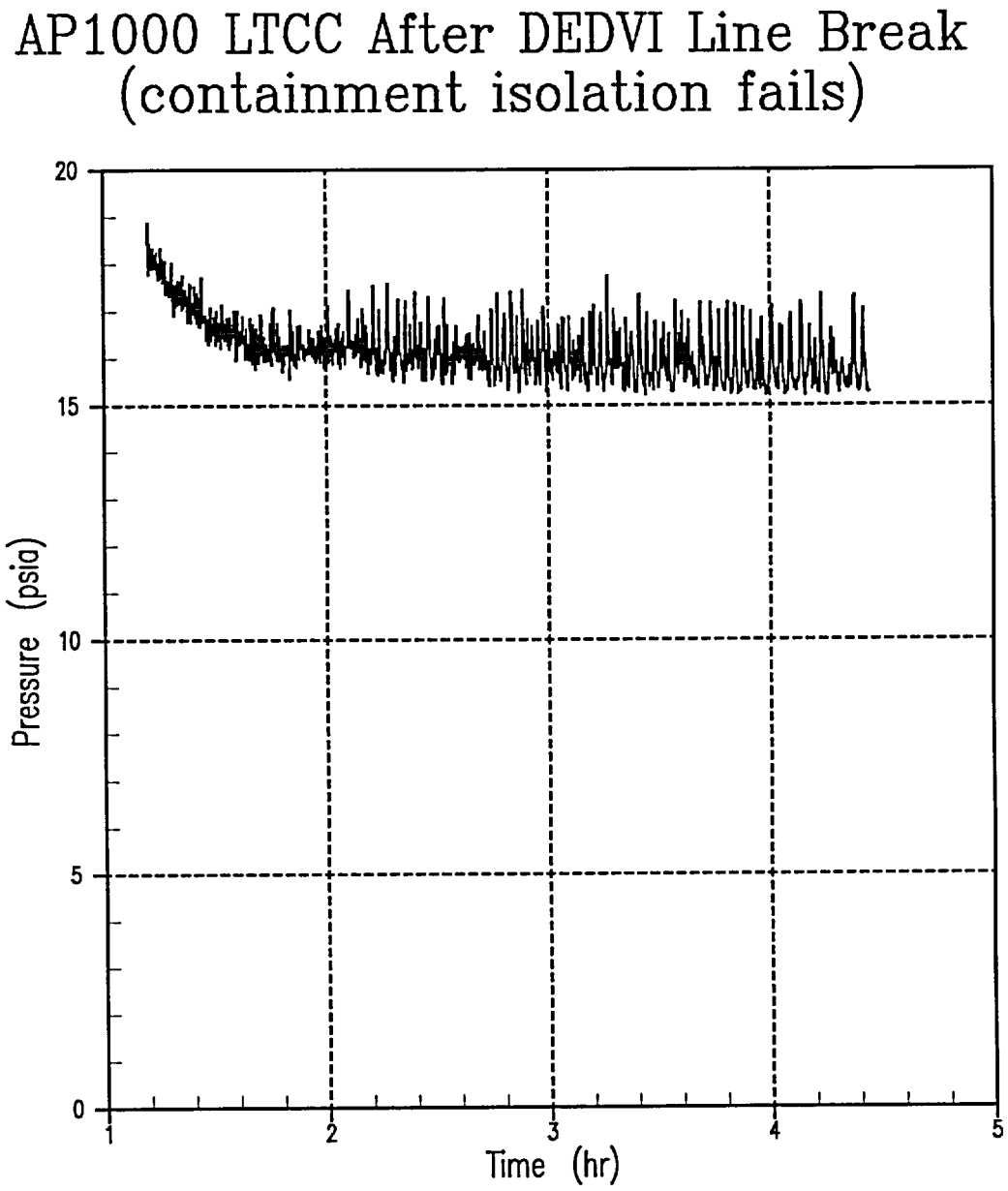


Figure RAI 720.013-25
Upper Plenum Pressure (Case G)

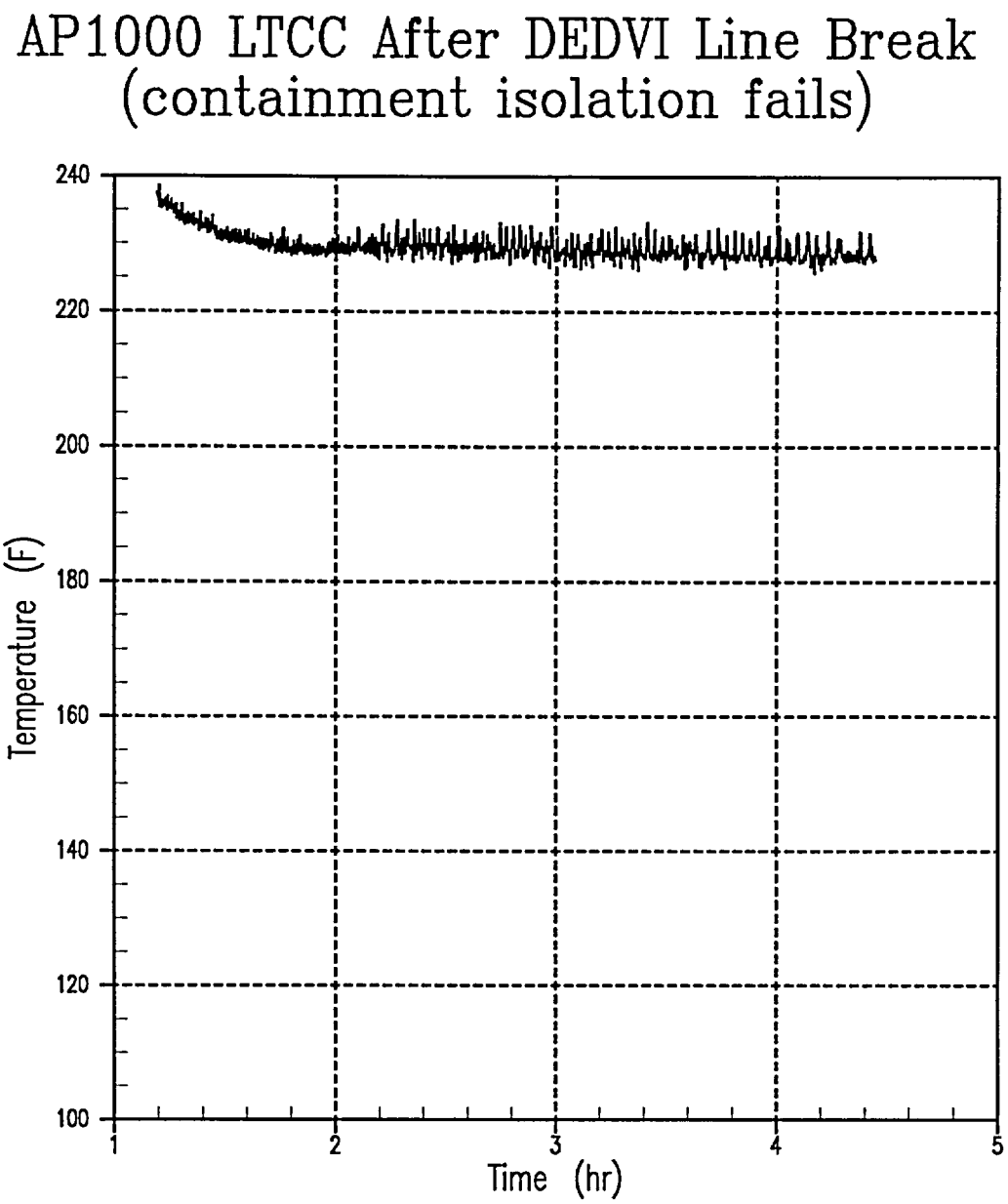


Figure RAI 720.013-26
PCT of the Hot Rod (Case G)

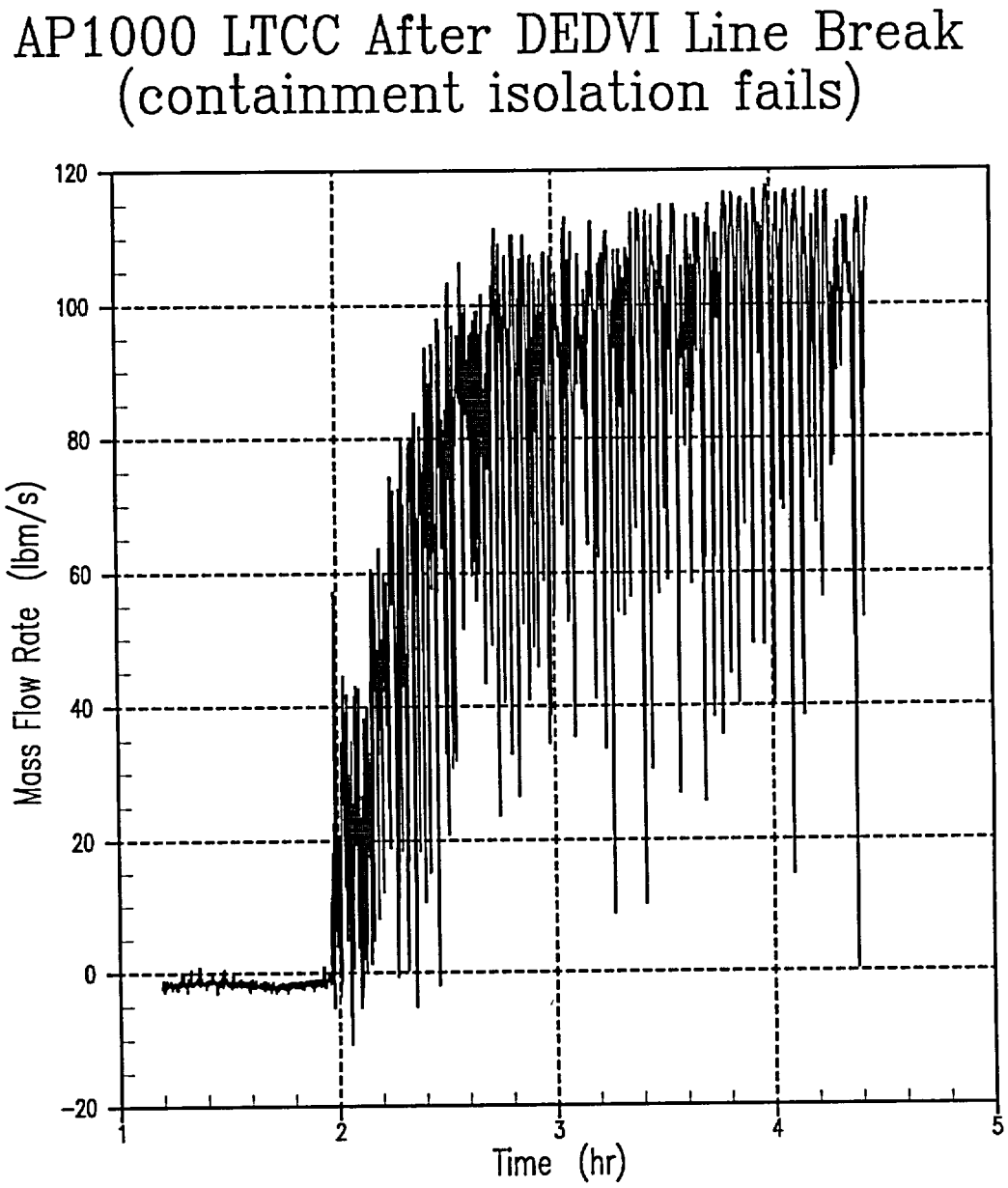


Figure RAI 720.013-27
DVI-A Mixture Flow Rate (Case G)

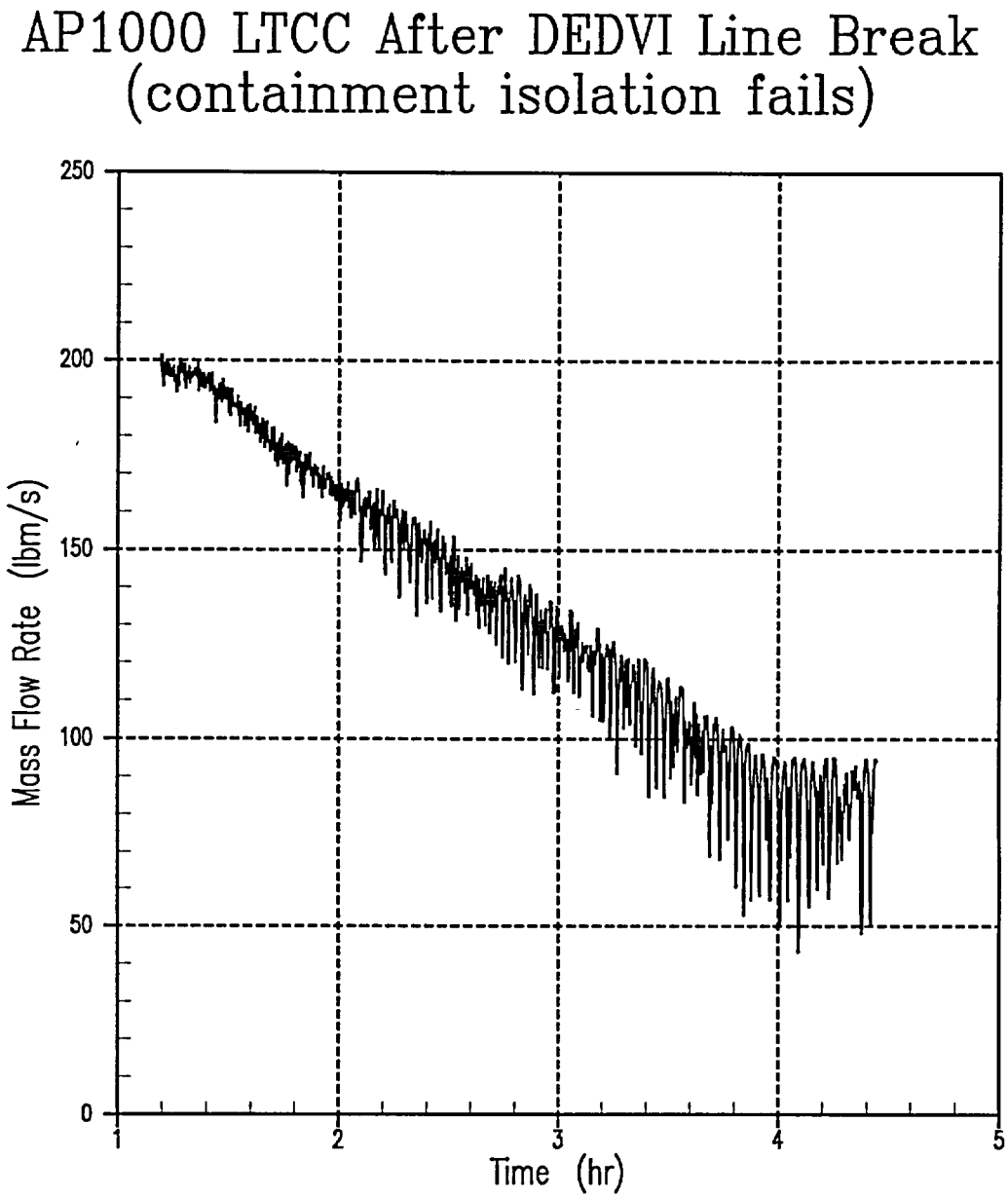


Figure RAI 720.013-28
DVI-B Mixture Flow Rate (Case G)

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.018

Question:

Comparisons with WCOBRA/TRAC in WCAP-15833, "WCOBRA/TRAC AP1000 ADS4/IRWST Phase Modeling," indicate that NOTRUMP is nonconservative for calculating reactor vessel inventory. See Figures 3-16 and 3-25 of the WCAP. The reduced vessel inventories are a result from more detailed models for reactor vessel, hot leg and ADS4 entrainment in WCOBRA/TRAC.

Provide sensitivity studies showing the effect on PCT if the more detailed models of WCOBRA/TRAC are used to calculate reactor vessel inventory for Uncertainty Case Nos. UC1, UC2 and UC3 in Section A5 of the PRA.

Westinghouse Response:

The Thermal-Hydraulic Uncertainty Analyses presented in Appendix A represent the limiting, low margin, high probability success sequences. These cases (UC1, UC2 and UC3) are analyzed using conservative bounding methods consistent with the design basis accident analyses presented in DCD Chapter 15. Similar to the approach for AP600, demonstration of success in these cases provides high confidence in the success criteria for the PRA, even considering uncertainties associated with thermal-hydraulic phenomena associated with the passive safety systems.

Case UC3 bounds the other cases cited (UC1, UC2) in severity. Therefore, the ADS-4 IRWST initiation phase of Case UC3 has been analyzed using the WCOBRA/TRAC modeling described in WCAP-15833, Revision 1. Case UC3 is a DEDVI break in which the intact loop CMT provides injection to the RCS. Only one of two IRWST lines injects into the reactor vessel, and one of its two parallel paths is assumed to fail to open.

The WCOBRA/TRAC analysis was initiated at the time of ADS-4 actuation at the system condition that is present in the NOTRUMP run at that time. The CMT flow rate from the NOTRUMP analysis is used directly as a boundary condition in WCOBRA/TRAC. Figures 720.018-1 through 6 provide plots of the plant response beyond the time at which the cladding temperature reaches its maximum value. The calculated peak clad temperature of 1084F is well below the value approaching 1600F that is calculated using LOCTA based on the NOTRUMP thermal-hydraulic transient described in the Westinghouse response to RAI 720.016.

WCOBRA/TRAC predicts a faster depressurization to the IRWST actuation setpoint and much more entrained liquid flow through the ADS-4 flow paths than the Case UC3 NOTRUMP analysis; this is consistent with the code comparisons for the design basis LOCA scenarios presented in WCAP-15833, Revision 1. Figure 720.018-1 shows the RCS pressure transient after the first ADS stage 4 valve opens. Because the ADS-4 IRWST initiation phase depressurization proceeds more rapidly than in NOTRUMP, the core and downcomer collapsed liquid levels (Figures 720.018-2 and 3, respectively) replenish enough that the cladding temperature transient turns around at approximately 1750 seconds transient time. Figure 720.018-4 shows that IRWST injection begins at 1630 seconds, compared to 1960 seconds in the corresponding NOTRUMP case; 230 seconds are needed after ADS-4 actuation in WCOBRA/TRAC, less than half the time needed in NOTRUMP. Figure 720.018-5 provides the calculated peak cladding temperature transient.

The upper part of the core is uncovered for an extended period in the WCOBRA/TRAC simulation, producing fuel rod heatup. As the combined core makeup tank and IRWST injection flow refills the downcomer and recovers the core, quenching the fuel rods, the fuel stored energy is removed. This additional release of energy leads to the slight repressurization of the reactor vessel observed after 1700 seconds, which eventually temporarily shuts down the gravity injection (Figure 720.018-4) from the IRWST. The core liquid inventory is adequately replenished at this juncture that the heat transfer in the upper part of the core leads to a peak clad temperature reduction of over 200F from the peak value. Since most of the fuel stored energy that accumulated during the clad heatup has been removed, IRWST injection will soon recommence, and core recovery will proceed; any increase in the peak clad temperature will be limited.

Figure 720.018-6 shows the total mass flow through the ADS-4 flow path in which both valves open. The mass flow beyond the approximately 50 lbm/s of vapor flow is liquid. Hundreds of {lbm/s} of liquid leave the RCS in WCOBRA/TRAC, until the core collapsed level falls below 4 feet. In contrast, the NOTRUMP case predicts virtually no liquid flow to proceed through the ADS-4 paths (Figure A5.2-30 in the AP1000 PRA Appendix A).

In summary, this supplemental WCOBRA-TRAC analysis of the limiting Thermal-Hydraulic Uncertanty Analysis (Case UC3) performed in support of the PRA further confirms the robustness of the AP1000 passive safety systems, and further confirms the appropriateness of the success criteria defined in the AP1000 PRA. As can be seen by the comparison of the calculated results of peak cladding temperature between WCOBRA-TRAC and NOTRUMP, the NOTRUMP computer code provides a conservative calculation of core cooling and fuel clad heat-up for a small break LOCA.

Design Control Document (DCD) Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

PRA Revision:

None

Figure 720.018-1 Reactor Vessel Pressure UC3 Scenario, Post ADS-4 Actuation

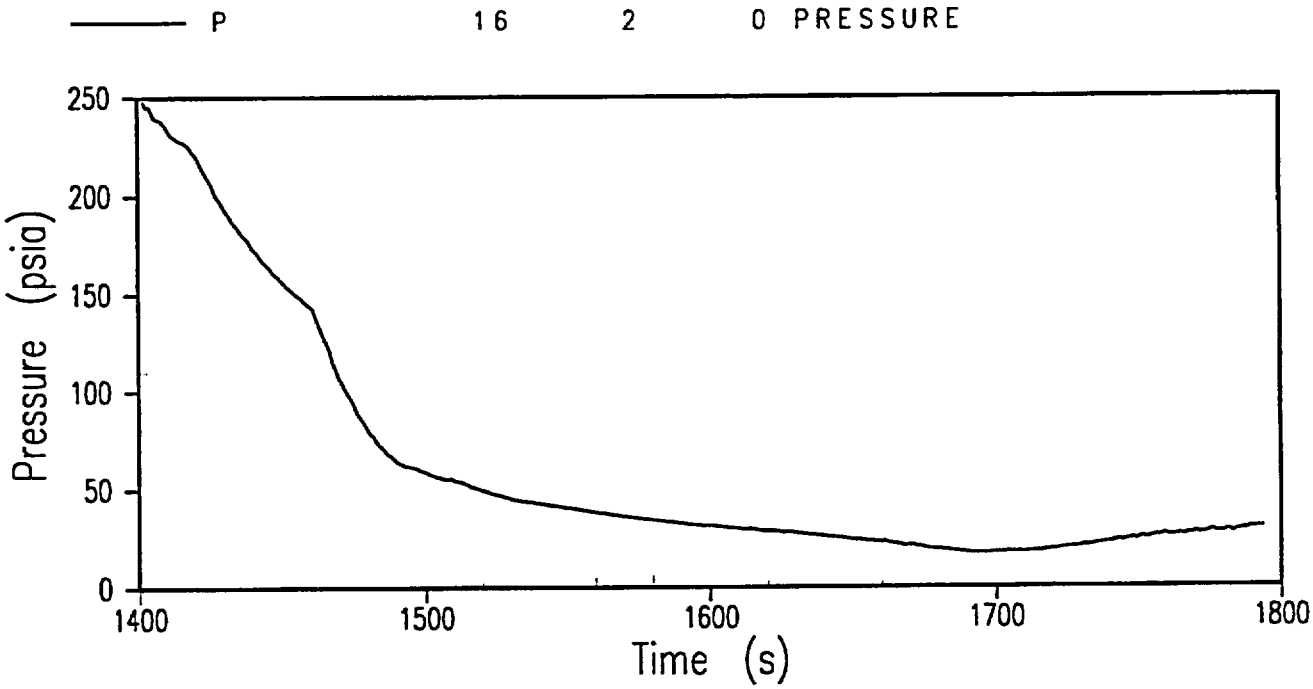


Figure 720.018-2 Core Collapsed Liquid Level UC3 Scenario, Post ADS-4 Actuation

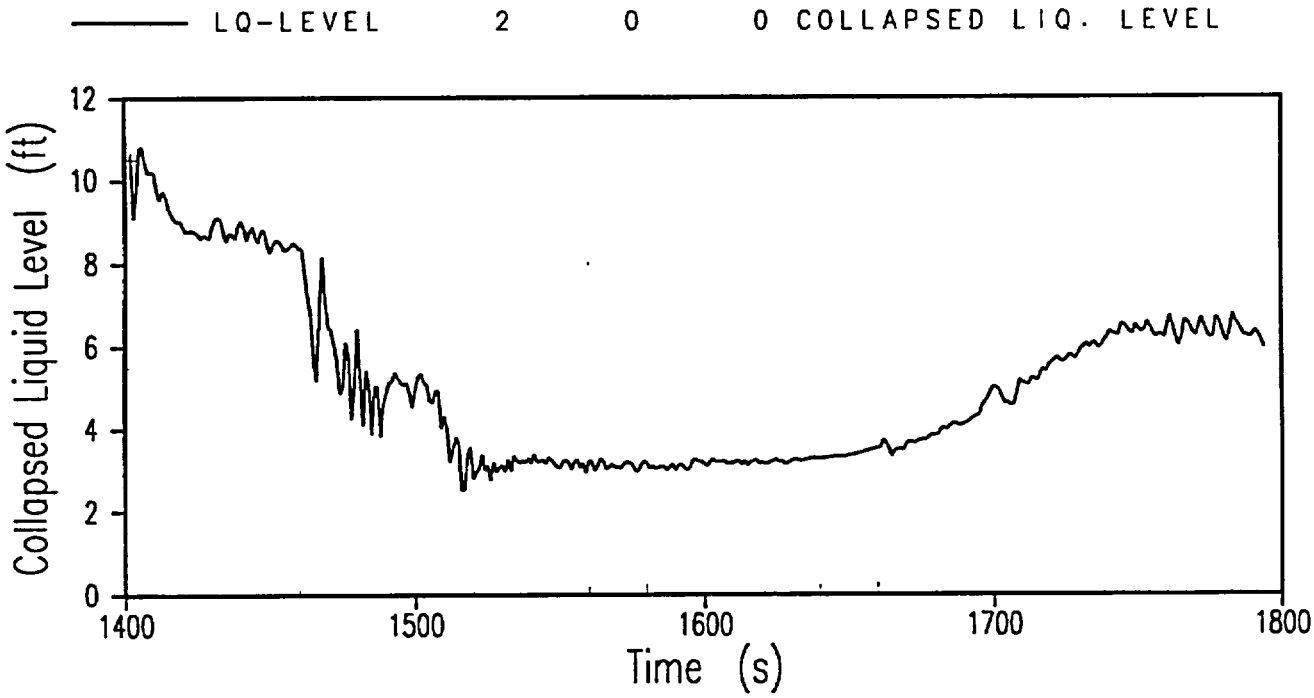


Figure 720.018-3 Downcomer Collapsed Liquid Level UC3 Scenario, Post ADS-4 Actuation

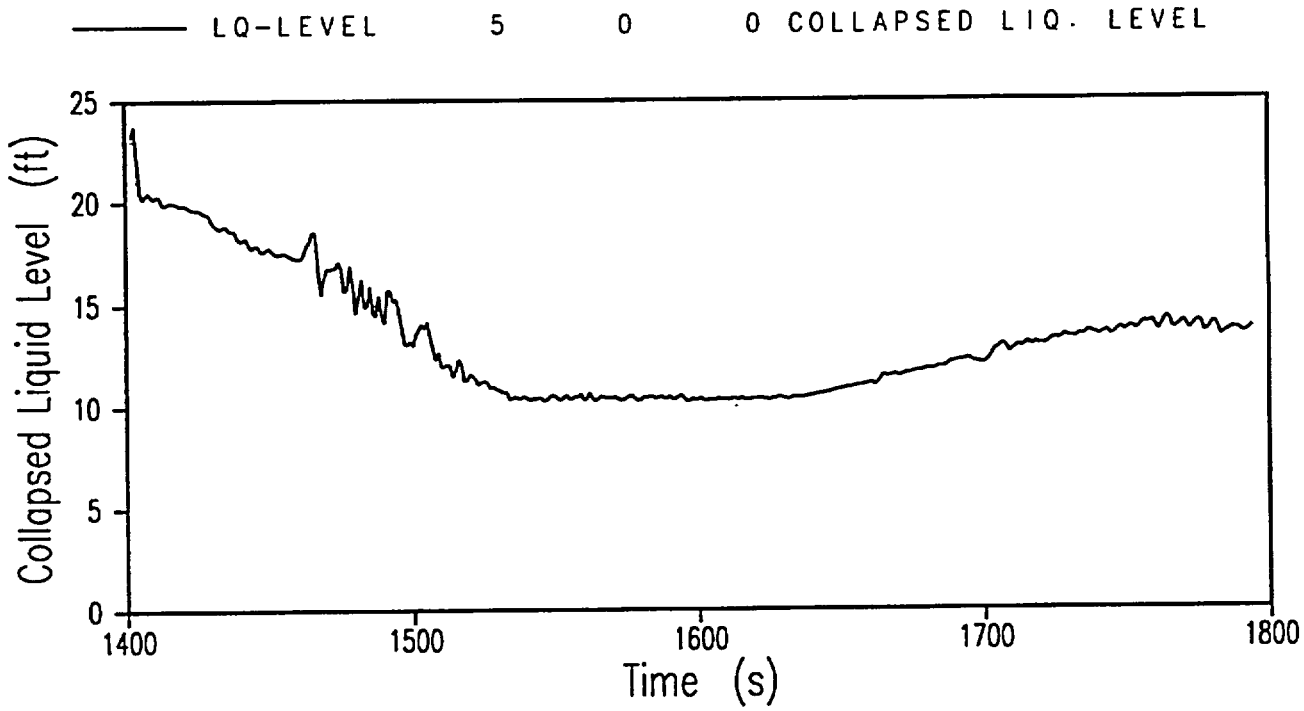


Figure 720.018-4 IRWST Injection Flow Rate UC3 Scenario, Post ADS-4 Actuation

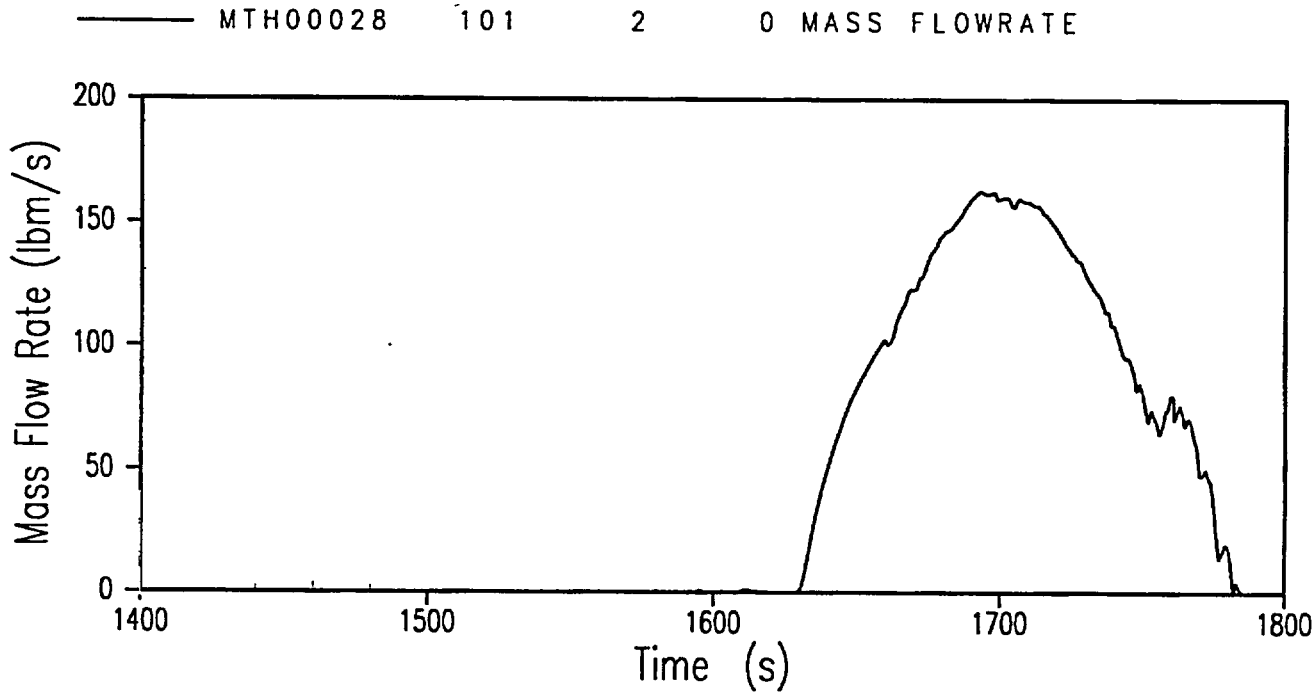
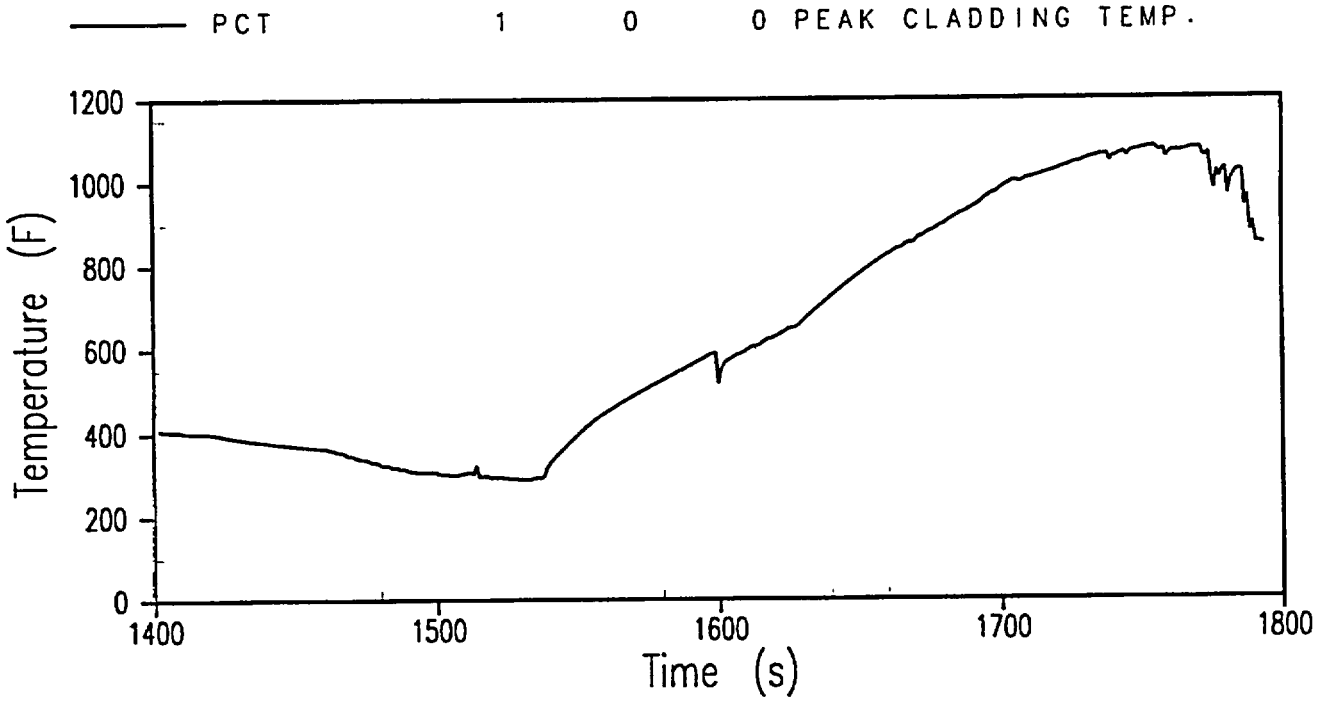


Figure 720.018-5 Peak Cladding Temperature UC3 Scenario, Post ADS-4 Actuation



AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.025

Question:

Section 6.3.2 discusses key operator actions within the various accident sequences, and the available time for these operator actions. For example, Section 6.3.2.1 states that, for a loss of feedwater event, the maximum time available for manual PRHR actuation is determined to be greater than 45 minutes. Section 6.3.2.5 states that, for medium-break LOCA, the time available for operator action to actuate CMT injection is determined to be 10 minutes from the time the actuation signal occurs, and 20 minutes if accumulator injection is successful. With successful accumulator injection and PRHR operation, the available time for operator action to depressurize the RCS is determined to be approximately 20 minutes from the time CMT actuation occurs, and the time available to start RNS injection is determined to be 20 minutes. For small-break LOCA, steam generator tube rupture and transients, the time available to manually actuate CMT, and RCS depressurization is determined to be at least 30 minutes. The maximum time available for manual actuation of RNS is determine to be approximately 10 minutes from ADS actuation if PRHR has not actuated. In all cases, Appendix A is referenced for the determination of the maximum available time for the operator actions.

A. Clarify where in Appendix A these available operator action times are described.

B. Provide bases and determination of these available times for the AP1000 design.

Westinghouse Response:

The following table indicates the basis for the time windows cited in PRA Section 6.3.2:

EVENT	Time Window	Available Time	Basis
Loss of Feedwater	PRHR signal to SG dryout	45 min	AP600 PRA Appendix A Figures A-2 through A-5 show that SG initial inventory supports core cooling for more than 5000 sec. AP1000 SG inventory per unit of core power is higher than for AP600.
MLOCA	CMT signal to manual CMT with Accumulator	20 min	AP1000 PRA Appendix A Figures A3.3-2 through A3.3-19 show that one accumulator supports core cooling for the first twenty minutes.
MLOCA	CMT signal to manual CMT without Accumulator	Not credited	Text in AP1000 PRA Chapter 6 will be revised as shown in PRA revision included in this response.
MLOCA	CMT signal to manual ADS with Accumulator	20 min	AP1000 PRA Appendix A Figures A3.3-2 through A3.3-19 show that one accumulator supports core cooling for the first twenty minutes.
MLOCA	CMT signal to manual ADS without Accumulator	Not credited	Table 6.3 in AP1000 PRA Chapter 6 will be revised as shown in PRA revision included in this response.
MLOCA	Event initiation to manual RNS with Accumulator	20 min	AP1000 PRA Appendix A Figures A3.3-2 through A3.3-19 show that one accumulator supports core cooling for the first twenty minutes.
MLOCA	Event initiation to manual RNS without Accumulator	Not credited	Table 6.3 in AP1000 PRA Chapter 6 will be revised as shown in PRA revision included in this response.
SLOCA, SGTR, TRAN	CMT signal to manual CMT/ADS	30 min	AP600 PRA Appendix A Figures A-17 through A-19 show adequate core cooling for more than 5000 sec with no CMT or Accumulator injection. AP1000 PRA Appendix A Figures A3.2-2 through A3.2-7 show that AP1000 response for SLOCA timing is very similar to AP600.
SLOCA, SGTR, TRAN	ADS actuation to manual RNS with CMT without PRHR	10 min	AP1000 PRA Appendix A Figures A3.2-2 through A3.2-7 show that RNS injection at 10 minutes after ADS would support core cooling in the same manner as IRWST injection.

Several inconsistencies in AP1000 PRA Chapter 6 are corrected in the PRA revision shown below.

Design Control Document (DCD) Revision:

None

PRA Revision:

6.3.2.1 Time to Respond to Loss of Decay Heat Removal

For events involving a loss of main and startup feedwater to the steam generators, the PRHR system would be expected to remove decay heat. If PRHR failed to automatically actuate, the operators would be expected to manually actuate this function. If this were unsuccessful, the operators could initiate RCS depressurization using the ADS, in order to actuate core makeup tank or accumulator injection (i.e., feed and bleed).

The limiting loss of decay heat removal events are those that result in a loss of secondary side heat sink. A loss of main feedwater without startup feedwater (resulting from either station blackout or failure of startup feedwater (SFW)) is analyzed, since this results in the minimum secondary side inventory at the time of reactor trip. Of interest is the time at which steam generator heat transfer is significantly degraded.

Given a loss of main feedwater, with a subsequent failure of startup feedwater, the maximum time available for manual PRHR actuation is the time between generation of the PRHR actuation signal (assumed to occur on low steam generator wide range level), and the time that steam generator dryout occurs. This is determined to be greater than 45 minutes ~~(see Appendix A)~~. The Human Reliability Analysis (HRA) as documented in Chapter 30, assumes only a 30-minute time window for this action.

Once dryout occurs, following a loss of main feedwater with a subsequent failure of startup feedwater and PRHR, the operators must initiate depressurization via the ADS in about 30 minutes.

6.3.2.5 Time to Actuate Core Makeup Tanks and Depressurize RCS

The core makeup tanks are expected to automatically actuate and inject following a LOCA, or following a transient with loss of decay heat removal and subsequent relief through the pressurizer safety valves. As the CMTs inject and their level drops, the ADS is automatically actuated to depressurize the RCS to allow for accumulator injection (self actuated) and RNS operation (manually actuated) or gravity injection from the IRWST (automatically actuated).

If the CMTs fail to actuate automatically, the operator can manually actuate them. If this fails, the operator is directed to manually initiate RCS depressurization with the ADS and initiate RNS injection. Actuation of the ADS also opens the IRWST injection squib valves. Thus, if RNS subsequently fails, depressurization will continue to the point at which gravity injection from the IRWST will occur; in this case the operator would be required to actuate the IRWST squib valves as well.

Large LOCA

For a large LOCA in the PRA, the RCS depressurizes rapidly and the accumulators empty in several minutes. As a result, the HRA assumes that there is insufficient time to credit operator action to actuate IRWST injection or ADS valves. Although not required to provide core cooling, CMT injection is required to provide automatic IRWST and ADS actuation signals.

Medium LOCA

The time available for operator action to actuate CMT injection is defined as the time from the occurrence of the first signal that would alert the operators to the fact that CMT actuation should have occurred until the latest time at which CMT injection can begin such that core damage will be prevented without accumulator injection (assuming one train of RNS operates in injection mode after CMT injection). For medium LOCA, ~~this is determined to be 10 minutes from the time the actuation signal occurs (Appendix A); the CMT actuation signal occurs shortly after event initiation~~ **no credit is taken for this action if no accumulator is available.** The available time is 20 minutes if accumulator injection is successful.

For medium LOCAs, the time available for operator action to depressurize the RCS is defined as the time from event initiation until the time at which the ADS valves must begin to open in order to ensure that gravity injection can occur in time to prevent core damage (given that RNS has failed and the IRWST valves have opened). For cases with successful accumulator injection and PRHR operation, this is determined to be approximately 20 minutes from the time CMT actuation occurs (Appendix A), conservatively evaluated for breaks at the large end of the medium LOCA spectrum. For cases without accumulator injection or PRHR operation, operator action is not modeled in the PRA because of the shorter time available.

If the IRWST injection line valves fail to open automatically, the operators must actuate them in order to allow gravity injection. This action is actually the same action as ADS actuation, since the same manual control initiates both ADS and IRWST valves. As a result, the HRA considers appropriate system dependencies with respect to this action.

The maximum time available for operator action to start RNS injection is defined as the time from event initiation until the time at which one train of RNS injection is required to prevent core damage and prevent initiation of gravity injection (assuming one accumulator has injected). For cases with successful accumulator injection and PRHR operation, this is determined to be 20 minutes (Appendix A), conservatively evaluated for breaks at the large end of the medium LOCA spectrum. For cases without accumulator injection or PRHR operation, operator action is not modeled in the PRA because of the shorter time available.

Small LOCA, Steam Generator Tube Rupture, and Transients

For small LOCAs, steam generator tube rupture with failure to isolate the ruptured steam generator and equalize primary and secondary side pressures, and transients with loss of decay heat removal, the operators manually actuate CMT injection, and RCS depressurization, if the automatic actuations fail. The time available to take these actions is defined as the time from generation of an automatic signal to actuate CMTs until the time at which the ADS valves must begin to open in order to ensure that either RNS operation or gravity injection can occur in time to prevent core damage. Under this assumption, manual RCS depressurization is assumed to occur at the same time as manual CMT actuation. This is determined to be at least 30 minutes (Appendix A) for these types of initiating events.

If the IRWST injection line valves fail to open automatically, the operators must actuate them in order to allow gravity injection. This action is actually the same action as ADS actuation, since the same manual control initiates both ADS and IRWST valves. As a result, the HRA considers appropriate system dependencies with respect to this action.

Actuation of RNS is assumed to be performed in conjunction with actuation of ADS. The maximum time available for operator action to start RNS injection is defined as the time from ADS actuation until the time at which one train of RNS injection is required to prevent core damage and prevent initiation of gravity injection (assuming at least one CMT has injected). This is determined to be approximately 10 minutes (Appendix A, for small LOCA) if PRHR has not actuated. This corresponds to over an hour from event initiation. With PRHR operation, there would be several hours available.

Table 6-3 (Sheet 2 of 4)

SUMMARY OF SUCCESS CRITERIA FOR OPERATOR ACTIONS AND MISSION TIMES

Operator Action Identifier	Operator Action Description/Where Performed	Used in Event Tree Cases	Available Operator Response Time (Minutes)	Reference/ Basis
CIC-MAN01	Isolate containment following core damage/In control room	CIC	120 (except LLOCA pre-core damage)	O, E
CMN-MAN01	Actuate CMTs if automatic actuation fails/In control room	CM1A, CM2AB, CM2L, CM2P, CM2NL, CM2SL	(Timing consistent with associated recognition action - LPM-MAN01/2)	A O, E
CMN-REC01	Actuate CMT injection after core damage/In control room	CMBOTH, CM2LLT	>60 (Based on CMN-MAN01)	O, E
CVN-MAN00	Align CVS auxiliary spray for SGTR/In control room	CSAX	30	O, E
CVN-MAN02	Align CVS boration following ATWS/In control room	CSBOR1	60	O, E
CVN-MAN03	Start CVS standby pump if aligned pump fails/In control room	CVS1, CSP	30	A, E
DUMP-MAN01	Control steam dump during SGTR/ In control room	COND1	>30	A, E
FWN-MAN02	Start startup feedwater pumps if automatic actuation fails/In control room	SFW, SFW1, SFWM, SFWT	>30	O, E
FWN-MAN03	Start startup feedwater pumps if automatic actuation fails (LOOP)/ In control room	SFWP	30	O, E
HPM-MAN01	Recognize need for high pressure decay heat removal/In control room	PRL, PRP, PRS, PRT, SFW, SFW1, SFWM, SFWP, SFWT	30	A, O, E
LPM-MAN01	Recognize need for RCS depressurization/In control room	AD1, AD1A, ADA, ADR, ADS, ADT, ADZ, ADV, ADW, CM2AB, CM2SL	30 (Transients, SLOCA)	A, E
LPM-MAN02	Recognize need for RCS depressurization/In control room	ADQ, ADUM, ADAB, ADAL, ADB, ADM, ADR, ADRA, ADU, CM1A, CM2L, CM2NL, CM2P	20 ~~(MLOCA with accumulator injection)~~ ~~10~~ ~~(MLOCA, without accumulator injection)~~	A, E

Table 6-3 (Sheet 4 of 4)

SUMMARY OF SUCCESS CRITERIA FOR OPERATOR ACTIONS AND MISSION TIMES

Operator Action Identifier	Operator Action Description/Where Performed	Used in Event Tree Cases	Available Operator Response Time (Minutes)	Reference/ Basis
RHN-MAN01	Align RNS after depressurization/ In control room	RNP, RNR	10 ~~(MLOCA without accumulator injection > 20 (Other)~~	A, E
RMN-MAN06	Recognize need and throttle RNS pump discharge valve if two RNS pumps are running with only one recirculation path available/ In control room	RECIRB, RECIRC, RECIRP, RNP, RNR	>30	E
RTN-MAN01	Perform controlled shutdown of the reactor	SDMAN	>60	E, O
SGHL-MAN01	Recognize need and actuate steam generator overfill protection/ In control room	SGHL	>30	O, E
VLN-MAN01	Recognize need and actuate containment hydrogen control system/In control room	VLH	>120	E
ZON-MAN01	Recognize need and start standby diesel generator/In control room	DGEN	>30	0

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.029

Question:

In Section 6.3.2.5, the time windows available for several operator actions associated with specific LOCA sequences are discussed. They include operator actions to actuate the core makeup tanks (CMTs), depressurize the reactor coolant system (RCS) , and actuate the normal residual heat removal (NRHR) pumps. In many cases, more than one time window (success criteria for operator action) is defined depending on the success or failure of other systems. For example, for medium LOCAs and CMT line breaks it was determined that the time available for operator action to manually actuate CMT injection (from the time the first signal would alert the operators to the time CMT actuation would occur) is 10 minutes without accumulator injection and 20 minutes with accumulator injection. Similarly, for medium LOCAs and CMT line breaks it was determined that the time available for operator action to manually depressurize the RCS is 20 minutes for cases with successful accumulator injection and PRHR injection. This time of availability was assessed to be too short to take credit for operator action for cases without accumulator injection or PRHR operation. The success criteria for operator actions used in the AP1000 human reliability analysis (HRA) are summarized in Table 6.3. For the specific example of medium LOCAs and CMT line breaks, two operator actions are listed: (1) manually actuate the CMTs if automatic actuation fails (event CMN-MAN01); and (2) recognize need for RCS depressurization (event LPM-MAN02). It appears from the HRA, documented in Chapter 30, that the human error probabilities for the events CMN-MAN01 and LPM-MAN02 were calculated assuming a 20 minutes time window, independent of the success or failure of the accumulators and PRHR (actually, the PRHR is not modeled at all in the MLOCA and CMT line break event trees). Please explain how the various success criteria for operator actions (time windows), determined by T-H analyses of the various accident sequences, were modeled in the PRA.

Westinghouse Response:

Some time window entries in Table 6.3 have been corrected as shown in the response to RAI720.025.

Design Control Document (DCD) Revision:

None

PRA Revision:

See the response to RAI 720.025.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.030

Question:

The AP1000 PRA event trees include a top event for containment cooling (event CHR). In Chapter 4 (Event Tree Models) it is stated that this top event models the need to successfully remove thermal energy from the containment atmosphere to the environment via the containment vessel following events that cause a significant increase in containment pressure and temperature, such as a LOCA and main steam line break accident inside containment. However, several statements made in Chapter 6 (Success Criteria Analysis) imply that the passive containment cooling system (PCS) water is not needed to prevent core damage. For example, on page 6-8 it is stated: "Therefore, sequences in which core damage has been avoided with successful IRWST injection and recirculation represent success (i.e., no core damage) regardless of the status of containment integrity or PCS water." Furthermore, on page 6-9 it is stated: "For success paths that result in steam release to the containment, the success of containment cooling (PCS or RNS) is modeled. If containment cooling is successful, then the path ends in an OK state. If PCS water cooling is not successful, then the path goes to a special OK end state to allow containment integrity sensitivity studies to be made." This "special OK" end state is labeled "late containment failure (LCF)" end state on page 4-141 and defined as an end state "...where the containment heat removal by either passive containment cooling system (PCS) or component cooling water (CCS) heat exchangers via normal residual heat removal (RHR) fails." These and other similar statements throughout the PRA create some confusion regarding what constitutes the PCS and what the criteria for its success are. In addition, no attempt is made to assess the impact of the "special OK end state" on PRA results and insights. Please address the following items:

A. Is containment cooling by heat transfer through the containment shell to the outside air included in the functions of the PCS system? Some parts of the PRA imply that containment cooling by air flow is part of the PCS while elsewhere in the PRA it is stated that the PCS is identical with the passive containment system water. If the answer is yes, please revise the system description and success criteria accordingly. Otherwise, include a statement at the beginning of each chapter discussing containment cooling, such as Chapter 4 (event tree models) and chapter 6 (success criteria), to clarify that air cooling is not part of PCS.

B. The success criteria for containment cooling are included in Section 6.3.1.5 under "Containment Isolation." Please include a separate section discussing the success criteria for "Containment Cooling."

C. In Chapter 13, on Passive Containment Cooling System (PCS), three fault tree models are listed. One, labeled PCT, is for all transients and LOCA events. Another, labeled PCP, is for loss of offsite power accidents. A third, labeled PCB, is for station blackout accidents. Table 13-2 summarizes the success criteria for fault tree PCT only. Since the PCS is a passive system, loss of offsite power or station blackout should not have an impact on the fault tree of this system. Please explain how fault trees PCP and PCB are different from PCT.

D. It is stated in Chapter 13 on PCS that blockage or plugging of the air flow paths is not modeled in the PRA because of the many design features that make such failure mechanisms highly unlikely (e.g., fifteen air flow path inlets covered by screens, each provided with a heating source to prevent blockage due to buildup of snow or ice). The staff believes that even "highly unlikely" failures should be included in the PRA models when such failures are associated with certain systems, structures and components (SSCs) whose unavailability or degradation would increase the plant's risk significantly (i.e., SSCs associated with high risk achievement worth). An SSC's risk achievement worth is used to identify operational requirements, such as requirements for surveillance and maintenance, to ensure that failure rates for these SSCs do not increase and remain "highly unlikely." In the case of containment air flow cooling paths, is air blockage highly unlikely when the availability of the heating sources to prevent buildup of snow or ice is not properly monitored and maintained? This issue maybe more significant for AP1000 than it is for AP600 because of the higher power and, thus, lower thermal margin associated with AP1000 as compared to AP600. Please provide a discussion on how this issue is being addressed in the AP1000 PRA.

E. On page 6-9 it is stated that "If PCS water cooling is not successful, then the path goes to a special OK end state to allow containment integrity sensitivity studies to be made." This "special OK" end state is labeled "late containment failure (LCF)" end state on page 4-141. Please explain the reasons for considering containment integrity sensitivity studies but not core damage sensitivity studies. Would the assumed late containment failure not cause core damage and large release? Also, is containment cooling by air flow alone adequate for long-term operation of the PRHR system even in the presence of an open or failed containment? Please explain and provide supporting analyses, as necessary.

F. The success criteria for containment heat removal, as listed in Table 6-2 and associated footnote #17, indicate that heat transfer through the containment shell to the outside air is sufficient to cool the containment atmosphere even with an open containment. According to Table 6-2, the success criteria for containment cooling by air flow for all accident sequences are based on T-H analyses documented in Appendix A of the PRA. However, the staff was not able to locate such analyses. Please explain.

G. If there is significant uncertainty whether the "late containment failure (LCF)" end state can lead to core damage, please perform a sensitivity study to assess the potential impact of this uncertainty on PRA results and insights, including insights related to the identification of non-safety-related equipment for availability control as well as the level of availability control for such equipment.

Westinghouse Response:

The responses to the questions are given below.

A. PCS containment heat removal functions are treated (and thus discussed) in two different contexts, which may be a source of confusion in the document).

Context 1:

Even if the PCS fails to deliver water cooling to the containment, for the plant CDF purposes, it is declared that PCS "air cooling mode" is sufficient to avoid core damage for 24 hours and beyond. For sequences with successful containment isolation, the effect of losing PCS water cooling is to increase the containment pressure but will not result in early containment failure as shown in the analyses provided in PRA Chapter 40. The higher containment pressure acts to increase the effectiveness of core cooling mechanisms, but this is ignored in the success criteria for core cooling. Sequences with loss of both PCS water cooling and containment isolation could require makeup to the containment water in the longer term depending on the size of the containment isolation failure, but the frequency of such sequences is too low to be significant. Also, other supplies of water are expected to be available from the fire protection system, demineralized water system, ancillary water system and temporary sources (fire trucks or water buffaloes) that can be brought on line by the operators to avoid dependence on air only cooling.

Context 2:

To address the potential uncertainty in avoiding core damage in the long term (after 24 hours) for a sequence which is "success" with air cooling being the only containment cooling mode, the success end states are further categorized as follows:

A event tree node is added to the end of the event trees (CHR – containment heat removal). This event tree node address only water cooling mode of PCS and decay heat removal by normal RHR (if applicable to a specific sequence).

When PCS is modeled by fault trees to be used in the at-power CDF event trees (under the event tree top event CHR), to identify and collect the late containment failure (LCF) end states for sequences, it includes only water cooling function. This function serves both as a short – term and a long term (24-72 hours). In this use of the PCS, the objective was to collect those

success sequences where only air cooling by PCS is deemed to be sufficient to avoid core damage, but both the water cooling by PCS and normal RHR are unavailable. This collection is stored under the LCF end state, which is not a CDF end state, but represents the uncertainty in the sufficiency of containment cooling solely by PCS air cooling.

As a sensitivity analysis, if the LCF end state frequency is added to the plant CDF, the plant CDF increases by 29%. This increase would consist of "late containment failure" end state.

Failure of air cooling is deemed to be less likely than the mechanical and actuation failure modes already accounted for in the PCS fault tree models. Thus, this failure mode is not assigned a failure probability. Other supplies of water are expected to be available from the fire protection system, demineralized water system, ancillary water system and temporary sources (fire trucks or water buffaloes) that can be brought on line by the operators to avoid dependence on air only cooling.

B. In the context of Section 6, the following success criteria is in effect for containment cooling:

Containment cooling either by

1. "Air cooling mode" of PCS

   or

2. "Water cooling mode" of PCS

   or

3. Decay heat removal mode of normal RHR

is sufficient to prevent core damage during the mission time specified for CDF event trees. Moreover, the probability of failure of all three of these functions for an other wise "success" sequence is deemed to be very small. Thus, this containment cooling function is not queried in the CDF event trees for CDF purposes."

However, it is queried for identifying LCF end state, as discussed in reply A above.

C. It is true that loss of offsite power or station blackout do not have a direct impact on the safety-related passive system PCS. However, the actuation signal models include non-safety related power sources in the base PCS fault tree. These sources are not credited when loss of offsite power and station blackout sequences are modeled. The valve mechanical failures are not impacted.

D. See the response to item G below.

E. Actually, a sensitivity study on postulating LCF end state as CDF has been performed and documented with the other CDF sensitivity analysis cases. However, this sensitivity analysis case was not provided in the AP1000 PRA report submitted to the NRC. According to this sensitivity analysis, if the LCF state becomes core damage (air cooling fails or is not sufficient, then the plant CDF increases by 29%.

F. The response to RAI 720.021 provides an analysis that shows PCS water cooling supports long term core cooling success even if an extreme containment isolation failure is assumed. Sequences with loss of both PCS water cooling and containment isolation could require makeup to the containment water in the longer term depending on the size of the containment isolation failure, but the frequency of such sequences is too low to be significant. Also, other supplies of water are expected to be available from the fire protection system, demineralized water system, ancillary water system and temporary sources (fire trucks or water buffaloes) that can be brought on line by the operators to avoid dependence on air only cooling.

G. As discussed in reply to question E, classifying LCF end state as core damage would increase CDF by 29%. This increase is deemed to be not large enough to warrant introduction of special attention from a CDF point of view.

From a LRF point of view, the Table below shows the relation between assuming different values for "air cooling" failure probability, and the resulting increase in LRF:

Air cooling Failure Probability	Current LCF with air cooling success	Increase in LRF if LCF and failure of air cooling occurs	increase in base LRF	Risk Significance
0.0001	6.92E-08	6.92E-12	very small	Insignificant
0.001	6.92E-08	6.92E-11	very small	Insignificant
0.01	6.92E-08	6.92E-10	3.5%	Insignificant

From this table, one sees that with any reasonable value for the air-cooling failure probability, the increase in LRF is not risk significant.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.053

Question:

The sequence used to quantify the intermediate containment failure probability given failures of PCS and containment venting (Figures 40-5 and 40-6) appears to be a loss-of-coolant accident (LOCA) with full RCS depressurization and successful core cooling. The resulting failure probability (0.02) is applied to all accident classes. Please justify the applicability of this probability value for each accident class since events with core damage could result in higher containment pressures than the sequence on which the probability value is based.

Westinghouse Response:

The cases being evaluated for intermediate containment failure at node IF on the containment event tree are pressurized over 24 hours by decay heat without PCS water cooling. There is no other challenge to the containment integrity other than the long-term pressurization. There is no significant hydrogen burning because the steam concentration in the containment is quite high, thus inerting combustion. There is no ex-vessel core debris, since the debris is retained in the vessel, and vessel failure is assume to result in early containment failure. It is conservative to use a successful core cooling sequence to evaluate containment pressure response for these cases because this results in higher mass and energy input to containment than would actually occur for these cases, for the following reasons:

- With unsuccessful core cooling, the core is uncovered and melts. A large fraction of the decay heat goes into melting fuel, not into pressurizing the containment atmosphere. The heat from the debris would be released at a rate dictated by boiling at the surfaces of the pool and reactor vessel lower head, which is approximately the same as decay heat. Much of the heat is absorbed by the large pool of subcooled water in the containment. Thus, the severe accident actually delays the start of the containment heatup, which then proceeds at approximately the same rate as the decay heat steaming, but lagging behind in the total pressure.

- With the core assumed to be intact and cooled, the decay heat goes into boiling water and producing steam in the reactor vessel that is released through the stage 4 ADS valves. The containment heatup begins earlier. Therefore, the containment pressure at 24 hours is higher for the success sequence than it is for the unsuccessful sequence. The successful core cooling scenario provides the bounding pressure for long-term heat up of the containment via decay heat steaming.

The 0.02 failure probability for at node IF is appropriate for the core damage accident classes.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.056

Question:

The offsite consequences for release categories CFI and CI for the AP1000 are similar to or less than the corresponding values for AP600. This is unexpected since the fission product inventories for AP1000 are larger than for AP600, and the same release fractions were assumed. Please explain the reasons for this inconsistency.

Westinghouse Response:

The AP1000 offsite consequences have now been calculated using release fractions from the AP1000-specific MAAP4.04 analyses incorporated into AP1000 Chapter 34 by the response to RAI 720.042. The CFI and CI Risk (REM/reactor year) are somewhat larger for AP1000 than for AP600, as expected.

Design Control Document (DCD) Revision:

None

PRA Revision:

AP1000 PRA Chapter 49 will be revised as shown in the attached markup.

CHAPTER 49

OFFSITE DOSE RISK QUANTIFICATION

49.1 Introduction

The potential ground-level exposure, expressed as both effective dose equivalent (EDE) whole-body dose and acute red bone marrow dose, resulting from the possible accidental release of radioactive fission products is discussed in this chapter. Chapter 45 provides the estimated source term information; that is, the accidental release conditions. These conditions are:

- The amount of released material
- Release energy
- Duration
- Location

for the ~~seven~~ six identified release categories~~ (note, CFV is not analyzed in this calculation note)~~.

This information is used here, along with data provided in the Electric Power Research Institute (EPRI) Advanced Light Water Reactor Utility Requirements Document (Reference 49-1), to perform the atmospheric dispersion analyses. These analyses are conducted to estimate the EDE whole-body dose and acute red bone marrow dose, both at the site boundary (0.5 miles). The population whole-body dose out to 80.5 kilometers and the downwind, centerline, ground-level thyroid dose at the site boundary (0.5 miles) are also calculated for information. The estimated site boundary whole-body dose and the acute red bone marrow dose are compared to the Westinghouse goal of <25 rems (0.25 sieverts), at a frequency not to exceed $1x10^{-6}$ per year. This is consistent with the goal provided in Reference 49-1.

It should be noted that Reference 49-1 recommends evaluation of the whole-body dose. However, it does not clearly identify whether this dose analysis should be based on an acute or committed dose (EDE) basis. Consequence codes such as the MELCOR Accident Consequence Code System (MACCS and MACCS2) and their predecessor CRAC2 (codes are recommended by Reference 49-1) can only calculate the EDE whole-body dose, therefore the committed dose has been used in previous ALWR analyses. It is felt however, that the whole-body acute centerline dose is more appropriate for this 25 rem dose calculation since the purpose of this calculation is to establish a margin to the Nuclear Regulatory Commission (NRC) safety goals and the NRC staff safety goal implementation requirement. In this context, it is acute health effects versus long term effects from a committed dose that are of significance. For consequence codes such as MACCS2, the acute red bone marrow dose may be used to represent the acute whole-body dose. These doses are determined at the site boundary (0.5-mile radius).

The thyroid (site boundary) and whole-body (population) doses are also calculated during the first 24 and 72 hours following the onset of core damage, based on the probabilistic atmospheric dispersion analysis of the dose associated with each release category, coupled

with multiple meteorological conditions. The thyroid and population doses are provided for information.

49.2 Conformance with Regulatory Requirements

MACCS2 version 1.12 (Reference 49-2) is used for the analysis. The NRC sponsored the development of this code. The code performs probabilistic estimates of offsite consequences from potential accidental releases in conformance with Chapter 9 of the probabilistic risk assessment (PRA) guidelines described in NUREG/CR-2300 (Reference 49-3).

The analysis was based on the Westinghouse design goals, which are consistent with the guidelines provided in Reference 49-1, as discussed above. This reference document also identifies use of the MACCS2 code for offsite consequence analysis.

49.3 Assumptions

This section discusses the information, including assumptions, required to perform the dose evaluation. The primary information required for the dose evaluation includes the release source terms; the site meteorological data; population distribution data; site economic data; agricultural and land use data; and food uptake, ingestion, and retention factors. Additionally, available data on site emergency plans, such as sheltering and evacuation, and site decontamination and interdiction plans, may be included in the dose evaluation. Since the Westinghouse design goal specifies the site boundary dose as the only consequence of concern, the population, land use, sheltering, evacuation, decontamination, and interdiction data are not required for this calculation.

The advanced light water reactor reference site information described in Reference 49-1 provides the meteorological and population data for the analysis. Since the advanced light water reactor site data does not provide sufficient topographical data to define the MACCS2 site input file, the site land use and crop data are based on representative data from the Surry Plant Site. These data are provided in Reference 49-2. Due to the proximity of the Surry Site to the ocean, those site sectors that are ocean were arbitrarily changed to land. This was done to allow use of the advanced light water reactor reference site population data (without having people assigned to ocean sectors). These changes to the land and crop characteristics are made to provide an acceptable MACCS2 input file. They have no effect on the calculated dose at the plant site boundary.

49.4 Methodology

The dose evaluation uses the MACCS2 accident consequence code to estimate the potential offsite effects of the postulated accidental releases, developed by the Level 2 analysis. The MACCS2 code performs multiple air dispersion analyses, based on the yearly meteorological data, to estimate the air and ground-level concentrations of the released nuclides of concern. Multiple dispersion analyses allow the application of statistical analysis to the full range of results, based on the probability of the meteorological sequences that caused those results. This accounts for the possibility of an accident occurring at any time during the year. The air and ground-level concentrations are then converted to exposure dose, per nuclide, for the following pathways: cloudshine, groundshine, inhalation (direct and resuspended material),

and ingestion. For the potential exposure during the initial 24 and 72 hours, the calculated dose does not consider the ingestion pathway.

The MACCS2 code permits evaluation of the effects associated with direct exposure to the radioactive cloud (that is, cloudshine, groundshine, and inhalation), during the period initially following the accident (up to 1 week), and the long-term (over many years) effects due to exposure to contaminated land (ingestion of local farm products, ground shine, resuspension inhalation). It also examines accident costs, which might include permanent relocation and/or decontamination. The code also permits the modeling of the protective effects of sheltering and/or evacuation of the population during the acute exposure phase.

The Westinghouse goal only requires dose determination for exposure resulting from the first 24 hours following the initiation of core damage. Additionally, the Westinghouse goal requires only the total dose to a hypothetical individual located at the site boundary, which is assumed to be one-half mile, directly downwind, during the entire exposure period. Therefore, dose calculations related to the actual site population distribution are not required, nor are calculation of potential health effects, such as deaths and cancers. Finally, the calculation of the site boundary dose ignores any potential mitigating effects, including sheltering and evacuation.

Therefore, the consequence level evaluated in this analysis includes the whole-body effective dose equivalent dose and the acute red bone marrow dose resulting from the first 24-hour exposure versus distance from the reactor.

Statistical evaluation is applied to the multiple dispersion analysis results so that the consequences are presented in terms of a mean value, a peak value, and as complementary cumulative distribution functions. These functions present the value of the consequence level (whole-body effective dose equivalent dose) versus the probability of exceeding this level. The Westinghouse goal and the Reference 49-1 guidelines provide a value for the site boundary dose, not to exceed 25 rems whole-body dose at a frequency not to exceed 1×10^{-6} events per year.

A brief description of the code follows.

The MACCS2 code performs its processing in three steps, or modules: ATMOS, EARLY, and CHRONC. The description of the source term and the dispersion calculations occur in the first module, i.e., the ATMOS module. The EARLY module performs the calculations relating to the initial exposure dose and can also account for sheltering or evacuation schemes. The CHRONC module performs the calculations relating to the long-term exposure dose (for many years) and can account for decontamination or food uptake parameters. Only the ATMOS and EARLY modules are used for this analysis.

The MACCS2 code models the atmospheric transport of fission products that are released from containment, as defined by the source term characteristics, using a Gaussian plume model, and calculates the air and ground-level concentrations for the radionuclides of concern. Vertical plume rise depends on the release energy. Plume motion depends on the available meteorological conditions; that is, wind speed, wind direction, and atmospheric stability. The code includes models for radioactive decay and daughter product buildup, wet

and dry deposition of the nuclides due to gravity settling, and washout due to precipitation. Noble gases are not removed by deposition.

The MACCS2 code first reviews the hourly meteorological data for one year and sorts the data into predefined and user-defined meteorological categories. This allows MACCS2 to assess the frequency of occurrence of the different meteorological types, and to provide a realistic representation of a full year of site weather. It does this without overlooking those meteorological conditions that, although infrequent, may be instrumental in producing peak impacts. The probability of each meteorological category is also determined by this analysis.

In performing the dispersion analysis for a specific source term, the code samples each of the meteorological categories several times. The number of sampling per category is specified by the user. Each sample consists of starting the postulated release during one of the 8,760 hours during the year, which is identified with the meteorological type being sampled. This is done for each of the meteorological types. For example, if the user specifies four samples per meteorological category, and there are 30 defined meteorological categories, and if the database has at least four hours of meteorological samples per category, then 4 times 30, or 120, dispersion analyses are performed by MACCS2.

Once the release start time is selected, then the actual meteorological data is used to model the subsequent dispersion. That is, the meteorological data are allowed to change as the material moves downwind. The calculation continues until the material reaches the boundary of the spatial grid (receptor grid) defined by the user. Each dispersion simulation, therefore, results in calculated, integrated air and ground concentrations, (plume centerline, ground level) as a function of downwind distance. Each analysis is then weighted by the probability of occurrence of the meteorological condition. As each calculation is performed, the results are accumulated to provide an average estimate of the downwind integrated air and ground concentrations, including effects from all possible meteorological types. The MACCS2 code also notes the peak downwind concentration at each receptor distance and the associated meteorological condition that produced the peak.

The MACCS2 code then performs conversion calculations to estimate the radiation doses based on the air and ground concentrations. The radiation doses received by individuals are due to the passing radioactive cloud and the material deposited on the ground. Radiation doses received from the cloud result from direct radiation (cloudshine) and inhalation of material suspended in the air (inhalation). These processes occur only during the time that the cloud passes over the affected population. Radiation doses associated with the material deposited to the ground include direct radiation of the nuclides on the ground (groundshine) and inhalation of materials that are resuspended into the air (resuspension). The MACCS2 code simulates these dose paths. Therefore, the code estimates the dose levels for each nuclide and for each dispersion analysis performed.

~~Seven~~ Six release categories are identified for evaluation of potential offsite doses. These categories are discussed in detail in Chapter 45, and are summarized as the following:

- IC – Containment integrity is maintained throughout the accident, and the release of radiation to the environment is due to nominal leakage.

- BP – Fission products are released from the reactor coolant system to the environment via the secondary system or other interfacing system bypass. Containment failure occurs prior to onset of core damage.

- CI – Fission product release occurs through a failure of the system or valves that close the penetrations between containment and the environment. Containment failure occurs prior to onset of core damage.

- CFE – Fission product release occurs through a containment failure caused by some dynamic severe accident phenomena occurring after the onset of core damage but prior to core relocation. Such phenomena include: hydrogen detonation, hydrogen diffusion flame, steam explosions, and vessel failures.

- CFI – Fission product release occurs through a containment failure caused by some dynamic severe accident phenomena occurring after core relocation but before 24 hours. Such phenomena include: hydrogen detonation and hydrogen deflagration.

- CFL – Fission product release occurs through a containment failure caused by some dynamic severe accident phenomena occurring after 24 hours. Such phenomena include the failure of containment heat removal (failure of passive containment cooling).

- ~~CFV – Fission product release occurs through containment venting to prevent containment failure (note, CFV is not analyzed in this calculation because no credit for venting is considered).~~

Additionally, one sensitivity evaluation (called DIRECT) is performed. The DIRECT release case is a modification of the IC release category in which no credit is assumed for aerosol nuclide deposition in the middle annulus. This case is conservative.

Based on the analysis described in Chapter 45, source terms are generated for each of the release categories. The source terms provide the necessary parameters to describe the conditions of the release. Tables 49-1 and 49-2 contain listings of the source terms and their parameters considered in this analysis. Table 49-1 summarizes the output provided by the Modular Accident Analysis Program (MAAP) code, while Table 49-2 presents the conversion of this data into MACCS2 input categories.

There are nine source terms, ~~seven~~ **six** release categories, and ~~two~~ **one** sensitivity ~~studies~~ **study** defined for this analysis~~ (CFV and BP sensitivity are not analyzed in this calculation)~~. To conservatively estimate the ground-level dose exposure at the site boundary, it is assumed that all the release categories occur at ground level. Finally, it is conservatively assumed that 5 percent of the iodine released from containment is volatile and would not deposit. Reference 49-1 provides a guideline of 3 percent volatile iodine.

Reference 49-1 provides some of the MACCS2 input data, including the site, and meteorological data. The dose data conversion file provided with the MACCS2 PC Code, version 1.12 (Reference 49-2) is used for this analysis. This file is required to convert the predicted nuclide concentrations to dose values.

49.5 Dose Evaluation Results and Discussion

Doses are determined for the early exposure effects resulting from the initial 24 and 72 hours following the core damage initiation. The dose evaluation provides the conditional probability distributions for the consequence measures, which includes the whole-body dose and the acute red bone marrow dose for this analysis. These consequence probability distributions are based on the assumption that the accident that produced the source term has occurred. Therefore, the consequence probability distributions presented result from the variation in dose levels due to the various meteorological conditions. Hence, the actual probability of the identified dose levels would be the probability of the release category that produced the source term occurring multiplied by the probability of the dose level. The actual probability of the identified dose levels is presented in Section 49.7.

Tables 49-3 through 49-6 present the summary of the dose evaluations (the MACCS2 output) for the six source terms and the DIRECT sensitivity. The information is provided in the following columns: the mean dose; the 50, 90, 95, 99, and 99.5 percent confidence values that the dose will not exceed; and the peak dose produced by any dispersion analysis. The dose (one sievert equals 100 rem) is presented for the following source terms: IC, BP, CFI, CFE, CFL, CI and DIRECT release sensitivity. Table 49-7 summarizes the calculated mean and peak dose values for the source terms evaluated.

Figures 49-1 through 49-56 present plots of the complementary cumulative distribution functions for the population whole-body, the site boundary whole-body effective dose equivalent, the acute red bone marrow and thyroid doses resulting from the following source terms: CFI, CFE, CFL, IC, BP, CI, and the DIRECT release sensitivity study.

Results in Table 49-7 show that for release categories IC, ~~CFL,~~ and the DIRECT sensitivity study, the mean whole-body EDE dose at the site boundary in 24 hours is less than ~~4~~ **6** rem. For all other release categories: BP, CI, CFE, ~~and~~ CFI, **and CFL** the mean dose at the site boundary in 24 hours is greater than 25 rem. The sum of the probabilities of the release categories including an intact containment excess leakage category is approximately 2.4×10^{-7} events per year for at power conditions. Therefore, for the IC **and the DIRECT**~~and the CFL~~ release categories, there is a large margin in both the dose as well as the probability for meeting the Westinghouse design goal of limiting the frequency of exceeding the 25 rem whole-body effective dose equivalent for an individual at the site boundary 24 hours after core damage to 1×10^{-6} events per year, without any emergency protective action. For the other release categories: BP, CI, CFE ~~and~~ CFI, **and CFL** there is a large margin in the probability for meeting the Westinghouse design goal.

Results in Table 49-7 also show that for release categories IC, ~~CFL,~~ and the DIRECT sensitivity study, the acute red bone marrow dose at the site boundary in 24 hours is less than 1 rem. For all other release categories: BP, CI, CFE, ~~and~~ CFI, **and CFL** the mean dose at the site boundary in 24 hours is greater than 25 rem. Again, the sum of the probabilities of the release categories including an intact containment excess leakage category is approximately 2.4×10^{-7} events per year for at power conditions. Therefore, for the IC, ~~CFL,~~ and the DIRECT release categories, there is a large margin in both the dose as well as the probability for meeting the Westinghouse design goal of limiting the frequency of exceeding the 25 rem whole-body effective dose equivalent for an individual at the site boundary 24 hours after

core damage to $1x10^{-6}$ events per year, without any emergency protective action. For the other release categories: BP, CI, CFE ~~and~~ CFI, **and CFL** there is a large margin in the probability for meeting the Westinghouse design goal.

49.6 Quantification of Site Risk

This section documents the calculation of total radiation dose risk at the site boundary and to the surrounding population for internal, at power, initiating events. Results are quantified based on both a 24-hour and a 72-hour exposure.

The dose risks are quantified by multiplying the calculated fission product release category frequency vector by the release category mean dose vectors. The frequencies for each of the six release categories are quantified in Chapter 45, while the mean doses for each release category are identified in this section. The total dose risk for each case is calculated as:

$$D_n = \Sigma_i (R_i \times d_{i,n})$$

where:

D_n = Total dose risk for case n=1,2,3,4 (site-24-hr, site-72-hr, population-24-hr, population-72-hr),
R_i = Release frequency for category i,
$d_{i,n}$ = mean dose for release category i for case n.

As previously described, the six release categories analyzed in this calculation are designated: IC, BP, CI, CFI, CFE, and CFL.

Tables 49-8 through 49-11 present the results of the dose risk calculations. Each table presents the release category identifier, the release frequency (per reactor-year), the mean dose (in rem), and the resulting risk (in rem per reactor-year). In addition, each table presents the total dose risk and the percent that each release category contributes to the total risk.

It is shown that release category CFE presents the largest risk to the site safety in each of the four presented cases.

49.7 Risk Quantification Results

The complementary cumulative distribution function (CCDF) probabilities presented in Figures 49-1 through 49-56 are based on the assumption that the respective release category has occurred. The actual dose probability is equal to the probability of the release multiplied by the CCDF probability. Figure 49-57 summarizes this calculation for the 24 hour, whole-body, site boundary dose for all release categories, excluding the sensitivity study. Figure 49-58 summarizes this calculation for the 24 hour, acute red bone marrow, site boundary dose for all release categories, excluding the sensitivity study. In addition, a total probability-dose curve, which sums all the release categories, is provided. This figure demonstrates compliance with the large release goal (24 hour, whole-body, site boundary dose greater than 25 REM has a frequency of less than $1x10^{-6}$).

49.8 References

49-1 "Advanced Light Water Reactor Utility Requirements Document," Volume III, Appendix A to Chapter 1, PRA Key Assumptions and Groundrules," EPRI, Rev. 5 & 6, December 1993.

49-2 Chanin, D., Young, M. L., "Code Manual for MACCS2, User's Guide," NUREG/CR-6613, SAND97-0594, Vol. 1, Sandia National Laboratories, U.S. Nuclear Regulatory Commission.

49-3 "PRA Procedures Guide," NUREG/CR-2300, U.S. Nuclear Regulatory Commission, Vol. 2, Chapter 9, Washington, D.C.

Table 49-1

AP1000 SOURCE TERMS FROM LEVEL 2 ANALYSIS (MAAP)

Case No.	Plume No.	Start Time (Seconds)	End Time (Seconds)	Duration (Seconds)	Release Fraction (MAAP Group) 1 Inert	2 CsI	3 TeO2	4 SrO	5 MoO2	6 CsOH	7 BaO	8 La2O3	9 CeO2	10 Sb	11 Te2	12 UO2	Plume Energy (Joules/sec) (Watts)	Plume Position
CFI	1	2930	10100	7170	2.88E-01	5.09E-03	7.92E-03	4.77E-03	1.03E-02	5.07E-03	8.12E-03	1.56E-04	1.82E-04	8.89E-03	0.00E+00	0.00E+00	0.00E+00	Leading
	2	10100	89850	79750	4.69E-01	3.50E-03	4.99E-03	9.23E-03	6.62E-03	3.49E-03	9.15E-03	1.08E-03	1.09E-03	5.96E-03	0.00E+00	0.00E+00	0.00E+00	Midpoint
	3	89850	176200	86350	6.98E-02	6.00E-06	1.00E-05	4.20E-04	1.00E-05	6.00E-06	1.70E-04	5.84E-04	5.84E-04	4.00E-05	0.00E+00	0.00E+00	0.00E+00	Midpoint
	4	176200	262600	86400	2.35E-02	0.00E+00	1.00E-05	1.50E-04	4.00E-05	0.00E+00	8.00E-05	3.23E-04	3.21E-04	4.00E-05	0.00E+00	0.00E+00	0.00E+00	Midpoint
CFE	1	3004	19810	16806	4.16E-01	5.53E-02	3.01E-02	3.14E-03	1.16E-02	5.35E-02	4.63E-03	5.57E-05	2.39E-04	2.03E-02	0.00E+00	0.00E+00	0.00E+00	Leading
	2	19810	89970	70160	4.05E-01	1.26E-03	1.48E-03	3.43E-04	2.58E-03	1.20E-03	6.45E-04	9.66E-06	1.14E-05	2.66E-03	0.00E+00	0.00E+00	0.00E+00	Leading
	3	89970	176300	86330	1.08E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
	4	176300	262700	86400	3.43E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E-05	0.00E+00	0.00E+00	0.00E+00	Midpoint
DIRECT	1	4378	84810	80432	2.95E-03	3.61E-05	2.86E-05	3.22E-05	3.94E-05	3.44E-05	3.61E-05	4.04E-06	4.39E-06	3.99E-05	0.00E+00	0.00E+00	0.00E+00	Midpoint
	2	84810	134400	49590	1.48E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.80E-08	2.40E-08	2.40E-07	0.00E+00	0.00E+00	0.00E+00	Leading
	3	134400	177600	43200	1.18E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.00E-08	0.00E+00	0.00E+00	0.00E+00	Midpoint
	4	177600	264000	86400	2.32E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.00E-08	0.00E+00	0.00E+00	0.00E+00	Midpoint
IC	1	4378	84810	80432	9.83E-04	1.20E-05	9.53E-06	1.07E-05	1.31E-05	1.15E-05	1.20E-05	1.35E-06	1.46E-06	1.33E-05	0.00E+00	0.00E+00	0.00E+00	Midpoint
	2	84810	134400	49590	4.93E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.00E-09	8.00E-09	8.00E-08	0.00E+00	0.00E+00	0.00E+00	Leading
	3	134400	177600	43200	3.94E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.00E-08	0.00E+00	0.00E+00	0.00E+00	Midpoint
	4	177600	264000	86400	7.72E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E-08	0.00E+00	0.00E+00	0.00E+00	Midpoint
BP	1	5984	22130	16146	6.32E-01	1.20E-02	1.85E-02	3.02E-04	4.97E-03	1.10E-02	7.91E-04	1.25E-05	3.52E-05	5.48E-02	0.00E+00	0.00E+00	0.00E+00	Midpoint
	2	22130	108500	86370	3.00E-04	5.00E-05	2.00E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
	3	108500	194900	86400	4.00E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
	4	194900	267800	72900	3.00E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
CI	1	101	50020	49919	5.73E-01	4.56E-02	2.12E-02	2.03E-02	4.04E-02	1.78E-02	3.16E-02	2.39E-04	7.42E-04	2.71E-02	0.00E+00	0.00E+00	0.00E+00	Midpoint
	2	50020	136400	86380	1.13E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E-07	0.00E+00	1.90E-04	0.00E+00	0.00E+00	0.00E+00	Midpoint
	3	136400	211700	75300	5.66E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.34E-03	0.00E+00	0.00E+00	0.00E+00	Midpoint
	4	211700	259600	47900	2.74E-02	0 00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.10E-04	0.00E+00	0.00E+00	0.00E+00	Midpoint
CFL	1	4353	23610	19257	7.51E-01	3.57E-02	4.36E-02	2.01E-03	9.90E-03	2.74E-02	3.02E-03	3.31E-05	1.60E-04	2.87E-02	1.14E-07	7.76E-10	0.00E+00	Midpoint
	2	23610	91280	67670	1.22E-01	9.30E-04	3.00E-05	0.00E+00	0.00E+00	2.70E-04	0.00E+00	0.00E+00	0.00E+00	2.90E-04	5.70E-09	2.00E-13	0.00E+00	Midpoint
	3	91280	177600	86320	6.93E-02	2.40E-04	3.00E-05	0.00E+00	0.00E+00	1.00E-04	0.00E+00	0.00E+00	0.00E+00	2.00E-04	1.00E-09	0.00E+00	0.00E+00	Midpoint
	4	177600	264000	86400	2.84E-02	1.00E-05	0.00E+00	0.00E+00	0.00E+00	1.00E-05	0.00E+00	0.00E+00	0.00E+00	1.20E-04	3.00E-10	0.00E+00	0.00E+00	Midpoint

Table 49-2

AP1000 SOURCE TERMS FOR DOSE EVALUATION (MACCS2)

Case No.	Plume No.	Start Time (Seconds)	End Time (Seconds)	Duration (Seconds)	Release Fraction (MACCS2 Group) 1 Inert	2 I	3 Cs	4 Te/Sb	5 Sr	6 Ru	7 La	8 Ce	9 Ba	Plume Energy Joules/sec (Watts)	Plume Position
CFI	1	2930	10100	7170	2.88E-01	5.09E-03	5.08E-03	5.14E-04	4.77E-03	1.03E-02	1.56E-04	6.04E-07	8.12E-03	0.00E+00	Leading
	2	10100	89850	79750	4.69E-01	3.50E-03	3.49E-03	3.45E-04	9.23E-03	6.62E-03	1.08E-03	3.61E-06	9.15E-03	0.00E+00	Midpoint
	3	89850	176200	86350	6.98E-02	6.00E-06	6.00E-06	2.31E-06	4.20E-04	1.00E-05	5.84E-04	1.94E-06	1.70E-04	0.00E+00	Midpoint
	4	176200	262600	86400	2.35E-02	0.00E+00	0.00E+00	2.31E-06	1.50E-04	4.00E-05	3.23E-04	1.06E-06	8.00E-05	0.00E+00	Midpoint
CFE	1	3004	19810	16806	4.16E-01	5.53E-02	5.37E-02	1.17E-03	3.14E-03	1.16E-02	5.57E-05	7.91E-07	4.63E-03	0.00E+00	Leading
	2	19810	89970	70160	4.05E-01	1.26E-03	1.21E-03	1.54E-04	3.43E-04	2.58E-03	9.66E-06	3.78E-08	6.45E-04	0.00E+00	Leading
	3	89970	176300	86330	1.08E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
	4	176300	262700	86400	3.43E-02	0.00E+00	0.00E+00	5.78E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
DIRECT	1	4378	84810	80432	2.95E-03	3.61E-05	3.46E-05	2.31E-06	3.22E-05	3.94E-05	4.04E-06	1.45E-08	3.61E-05	0.00E+00	Midpoint
	2	84810	134400	49590	1.48E-03	0.00E+00	0.00E+00	1.39E-08	0.00E+00	0.00E+00	1.80E-08	7.95E-11	0.00E+00	0.00E+00	Leading
	3	134400	177600	43200	1.18E-03	0.00E+00	0.00E+00	3.47E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
	4	177600	264000	86400	2.32E-03	0.00E+00	0.00E+00	1.73E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
IC	1	4378	84810	80432	9.83E-04	1.20E-05	1.15E-05	7.69E-07	1.07E-05	1.31E-05	1.35E-06	4.85E-09	1.20E-05	0.00E+00	Midpoint
	2	84810	134400	49590	4.93E-04	0.00E+00	0.00E+00	4.62E-09	0.00E+00	0.00E+00	6.00E-09	2.65E-11	0.00E+00	0.00E+00	Leading
	3	134400	177600	43200	3.94E-04	0.00E+00	0.00E+00	1.16E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
	4	177600	264000	86400	7.72E-04	0.00E+00	0.00E+00	5.78E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
BP	1	5984	22130	16146	6.32E-01	1.20E-02	1.11E-02	3.17E-03	3.02E-04	4.97E-03	1.25E-05	1.17E-07	7.91E-04	0.00E+00	Midpoint
	2	22130	108500	86370	3.00E-04	5.00E-05	5.73E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
	3	108500	194900	86400	4.00E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
	4	194900	267800	72900	3.00E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
CI	1	101	50020	49919	5.73E-01	4.56E-02	2.10E-02	1.57E-03	2.03E-02	4.04E-02	2.39E-04	2.46E-06	3.16E-02	0.00E+00	Midpoint
	2	50020	136400	86380	1.13E-01	0.00E+00	0.00E+00	1.10E-05	0.00E+00	0.00E+00	1.00E-07	0.00E+00	0.00E+00	0.00E+00	Midpoint
	3	136400	211700	75300	5.66E-02	0.00E+00	0.00E+00	7.74E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
	4	211700	259600	47900	2.74E-02	0.00E+00	0.00E+00	1.21E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0 00E+00	Midpoint
CFL	1	4353	23610	19257	7.51E-01	3.57E-02	2.84E-02	1.66E-03	2.01E-03	9.90E-03	3.31E-05	5.30E-07	3.02E-03	0.00E+00	Midpoint
	2	23610	91280	67670	1.22E-01	9.30E-04	3.46E-04	1.68E-05	0.00E+00	0.00E+00	0.00E+00	1.99E-13	0.00E+00	0.00E+00	Midpoint
	3	91280	177600	86320	6.93E-02	2.40E-04	1.16E-04	1.16E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint
	4	177600	264000	86400	2.84E-02	1.00E-05	1.00E-05	6.93E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	Midpoint

Table 49-3

SITE BOUNDARY WHOLE BODY DOSE (EFFECTIVE DOSE EQUIVALENT, EDE), SIEVERTS

24-hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	
CFI	3.25E+01	2.61E+01	7.11E+01	8.48E+01	NOT-FOUND	NOT-FOUND	9.29E+01
CFE	4.23E+01	2.22E+01	1.03E+02	1.35E+02	NOT-FOUND	NOT-FOUND	1.55E+02
DIRECT	5.48E-02	3.37E-02	1.36E-01	1.77E-01	NOT-FOUND	NOT-FOUND	2.37E-01
IC	1.82E-02	1.08E-02	4.84E-02	6.24E-02	NOT-FOUND	NOT-FOUND	7.88E-02
BP	1.15E+01	7.72E+00	3.15E+01	3.92E+01	NOT-FOUND	NOT-FOUND	4.39E+01
CI	5.10E+01	3.53E+01	1.24E+02	1.55E+02	NOT-FOUND	NOT-FOUND	2.27E+02
CFL	2.58E+01	2.10E+01	5.39E+01	6.47E+01	7.52E+01	7.81E+01	9.23E+01
72-hour Case Source Term	**Quantiles**						**Peak Consequence**
	Mean	**50th**	**90th**	**95th**	**99th**	**99.5th**	
CFI	3.49E+01	3.01E+01	7.12E+01	8.74E+01	NOT-FOUND	NOT-FOUND	9.71E+01
CFE	4.60E+01	2.31E+01	1.16E+02	1.47E+02	NOT-FOUND	NOT-FOUND	1.67E+02
DIRECT	6.09E-02	3.67E-02	1.66E-01	2.27E-01	NOT-FOUND	NOT-FOUND	2.56E-01
IC	2.21E-02	1.31E-02	5.73E-02	7.96E-02	NOT-FOUND	NOT-FOUND	8.97E-02
BP	1.23E+01	8.02E+00	3.18E+01	4.09E+01	NOT-FOUND	NOT-FOUND	4.66E+01
CI	5.40E+01	3.75E+01	1.39E+02	1.89E+02	NOT-FOUND	NOT-FOUND	2.38E+02
CFL	2.80E+01	2.13E+01	6.32E+01	7.12E+01	7.86E+01	8.20E+01	9.92E+01

Table 49-4

SITE BOUNDARY THYROID DOSE, SIEVERTS

24-hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	
CFI	5.67E+01	5 09E+01	1.14E+02	1.44E+02	NOT-FOUND	NOT-FOUND	1.62E+02
CFE	3.59E+02	2.02E+02	1.01E+03	1.18E+03	NOT-FOUND	NOT-FOUND	1.28E+03
DIRECT	1.34E-01	8.58E-02	3.55E-01	4.48E-01	NOT-FOUND	NOT-FOUND	5.23E-01
IC	4 47E-02	3.08E-02	1.20E-01	1.54E-01	NOT-FOUND	NOT-FOUND	1.74E-01
BP	7.71E+01	5 40E+01	2.04E+02	2.23E+02	2.73E+02	NOT-FOUND	2.73E+02
CI	1.82E+02	1.23E+02	4.58E+02	5.74E+02	NOT-FOUND	NOT-FOUND	7.30E+02
CFL	2.20E+02	1.68E+02	5.34E+02	6.39E+02	7.07E+02	7.11E+02	7.30E+02
72-hour Case Source Term	**Quantiles**						**Peak Consequence**
	Mean	**50th**	**90th**	**95th**	**99th**	**99.5th**	
CFI	6.14E+01	5.26E+01	1.23E+02	1.54E+02	NOT-FOUND	NOT-FOUND	1.73E+02
CFE	3.81E+02	2.11E+02	1.02E+03	1.23E+03	NOT-FOUND	NOT-FOUND	1.36E+03
DIRECT	1.56E-01	1.05E-01	4.20E-01	5.47E-01	NOT-FOUND	NOT-FOUND	5.94E-01
IC	5.65E-02	3.86E-02	1.33E-01	1.76E-01	NOT-FOUND	NOT-FOUND	2 09E-01
BP	8.19E+01	5.64E+01	2.10E+02	2.60E+02	NOT-FOUND	NOT-FOUND	2.89E+02
CI	1.98E+02	1.40E+02	4.70E+02	5.79E+02	NOT-FOUND	NOT-FOUND	7.90E+02
CFL	2.35E+02	2 08E+02	5.34E+02	6.39E+02	7.18E+02	7.29E+02	7.76E+02

Table 49-5

POPULATION WHOLE BODY DOSE (EFFECTIVE DOSE EQUIVALENT, EDE), 0-80.5 KM PERSON-SIEVERTS

24-hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	
CFI	7.88E+03	6.11E+03	1.47E+04	2.01E+04	3.21E+04	3.51E+04	5.34E+04
CFE	8.51E+03	6.25E+03	1.62E+04	2.31E+04	4.13E+04	5.06E+04	6.40E+04
DIRECT	2.16E+01	1.20E+01	4.78E+01	8.13E+01	1.14E+02	1.23E+02	1.68E+02
IC	7.19E+00	4.21E+00	1.71E+01	2.95E+01	3.56E+01	3.84E+01	5.60E+01
BP	2.91E+03	1.74E+03	5.90E+03	1.00E+04	1.52E+04	1.81E+04	2.58E+04
CI	2.01E+04	1.13E+04	4.71E+04	6.60E+04	1.23E+05	1.48E+05	1.61E+05
CFL	5.32E+03	3.87E+03	1.04E+04	1.35E+04	2.32E+04	2.77E+04	4.35E+04
72-hour Case Source Term	**Quantiles**						**Peak Consequence**
	Mean	**50th**	**90th**	**95th**	**99th**	**99.5th**	
CFI	8.89E+03	6.89E+03	1.63E+04	2.21E+04	3.42E+04	3.84E+04	5.73E+04
CFE	9.36E+03	6.89E+03	1.88E+04	2.54E+04	4.25E+04	5.12E+04	6.77E+04
DIRECT	2.45E+01	1.43E+01	5.50E+01	8.33E+01	1.16E+02	1.26E+02	1.78E+02
IC	8.80E+00	5.57E+00	1.98E+01	3.14E+01	4.41E+01	5.03E+01	6.33E+01
BP	3.11E+03	1.85E+03	6.31E+03	1.03E+04	1.54E+04	1.82E+04	2.69E+04
CI	2.14E+04	1.25E+04	4.90E+04	7.40E+04	1.27E+05	1.53E+05	1.67E+05
CFL	5.84E+03	4.32E+03	1.12E+04	1.48E+04	2.53E+04	3.04E+04	4.62E+04

Table 49-6

SITE BOUNDARY RED MARROW DOSE (TOTAL ACUTE), SIEVERTS

24-hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	
CFI	2.80E+00	1.89E+00	7.07E+00	7.93E+00	NOT-FOUND	NOT-FOUND	8.41E+00
CFE	7.81E+00	4.83E+00	2.01E+01	2.07E+01	2.22E+01	2.29E+01	2.70E+01
DIRECT	2.47E-03	1.49E-03	6.40E-03	8.69E-03	1.01E-02	1.01E-02	1.03E-02
IC	8.22E-04	5.34E-04	2.09E-03	2.67E-03	3.10E-03	3.15E-03	3.42E-03
BP	1.53E+00	1.09E+00	3.48E+00	4.77E+00	NOT-FOUND	NOT-FOUND	5.32E+00
CI	2.46E+00	1.67E+00	6.13E+00	7.72E+00	1.00E+01	NOT-FOUND	1.00E+01
CFL	3.01E+00	2.28E+00	6.38E+00	7.12E+00	7.86E+00	8.20E+00	9.90E+00
72-hour Case Source Term	**Quantiles**						**Peak Consequence**
	Mean	**50th**	**90th**	**95th**	**99th**	**99.5th**	
CFI	3.70E+00	3.14E+00	7.52E+00	9.09E+00	1.04E+01	1.07E+01	1.19E+01
CFE	1.02E+01	5.68E+00	3.01E+01	3.11E+01	3.36E+01	3.48E+01	4.17E+01
DIRECT	5.70E-03	4.48E-03	1.06E-02	1.13E-02	1.33E-02	1.43E-02	2.09E-02
IC	2.78E-03	2.44E-03	5.59E-03	7.00E-03	7.52E-03	7.76E-03	9.02E-03
BP	2.04E+00	1.28E+00	5.30E+00	6.75E+00	NOT-FOUND	NOT-FOUND	7.00E+00
CI	4.23E+00	3.13E+00	1.01E+01	1.08E+01	1.26E+01	1.34E+01	1.70E+01
CFL	4.34E+00	3.31E+00	9.35E+00	1.03E+01	1.13E+01	1.18E+01	1.42E+01

Table 49-7

DOSE SUMMARY

	Population Dose (Sieverts)				Site Boundary Whole Body Dose (Sieverts)				Site Boundary Thyroid Dose (Sieverts)				Site Boundary Red Marrow Dose Sieverts)			
	24-Hour		72-Hour		24-Hour		72-Hour		24-Hour		72-Hour		24-Hour		72-Hour	
	Mean	Peak	Mean	Peak	Mean	Peak	Mean	Peak	Mean	Peak	Mean	Peak	Mean	Peak	Mean	Peak
CFI	7 88E+03	5.34E+04	8.89E+03	5.73E+04	3 25E+01	9 29E+01	3 49E+01	9.71E+01	5 67E+01	1.62E+02	6.14E+01	1.73E+02	2 80E+00	8 41E+00	3.70E+00	1.19E+01
CFE	8 51E+03	6 40E+04	9 36E+03	6 77E+04	4.23E+01	1.55E+02	4 60E+01	1 67E+02	3.59E+02	1.28E+03	3 81E+02	1.36E+03	7 81E+00	2.70E+01	1 02E+01	4.17E+01
DIRECT	2 16E+01	1 68E+02	2.45E+01	1.78E+02	5 48E-02	2 37E-01	6 09E-02	2.56E-01	1.34E-01	5.23E-01	1.56E-01	5.94E-01	2.47E-03	1.03E-02	5.70E-03	2 09E-02
IC	7.19E+00	5 60E+01	8 80E+00	6 33E+01	1.82E-02	7.88E-02	2 21E-02	8 97E-02	4 47E-02	1.74E-01	5 65E-02	2 09E-01	8 22E-04	3 42E-03	2.78E-03	9 02E-03
BP	2 91E+03	2.58E+04	3.11E+03	2 69E+04	1.15E+01	4 39E+01	1.23E+01	4 66E+01	7.71E+01	2.73E+02	8.19E+01	2.89E+02	1.53E+00	5.32E+00	2 04E+00	7.00E+00
CI	2 01E+04	1 61E+05	2.14E+04	1 67E+05	5.10E+01	2.27E+02	5 40E+01	2.38E+02	1.82E+02	7.30E+02	1 98E+02	7.90E+02	2 46E+00	1 00E+01	4 23E+00	1.70E+01
CFL	5 32E+03	4.35E+04	5 84E+03	4 62E+04	2.58E+01	9 23E+01	2 80E+01	9.92E+01	2.20E+02	7.30E+02	2.35E+02	7.76E+02	3 01E+00	9.90E+00	4.34E+00	1.42E+01

Table 49-8

SITE BOUNDARY WHOLE BODY EDE DOSE RISK – 24 HOURS

Release Category	Release Frequency (/reactor year)	Mean Dose (sieverts)	Dose (REM)	Risk (REM/reactor year)	Percent Contribution to Total Risk
CFI	1.89E-10	3.25E+01	3.25E+03	6.14E-07	1.2%
CFE	7.47E-09	4.23E+01	4.23E+03	3.16E-05	61.4%
IC	2.21E-07	1.82E-02	1.82E+00	4.02E-07	0.8%
BP	1.05E-08	1.15E+01	1.15E+03	1.21E-05	23.5%
CI	1.33E-09	5.10E+01	5.10E+03	6.78E-06	13.2%
CFL	3.45E-13	2.58E+01	2.58E+03	8.90E-10	0.0%
			Total Risk =	5.150E-05	100.0%

Table 49-9

SITE BOUNDARY WHOLE BODY EDE DOSE RISK – 72 HOURS

Release Category	Release Frequency (/reactor year)	Mean Dose (sieverts)	Dose (REM)	Risk (REM/reactor year)	Percent Contribution to Total Risk
CFI	1.89E-10	3.49E+01	3.49E+03	6.60E-07	1.2%
CFE	7.47E-09	4.60E+01	4.60E+03	3.44E-05	61.8%
IC	2.21E-07	2.21E-02	2.21E+00	4.88E-07	0.9%
BP	1.05E-08	1.23E+01	1.23E+03	1.29E-05	23.2%
CI	1.33E-09	5.40E+01	5.40E+03	7.18E-06	12.9%
CFL	3.45E-13	2.80E+01	2.80E+03	9.66E-10	0.0%
			Total Risk =	5.56E-05	100.0%

Table 49-10

POPULATION WHOLE BODY EDE DOSE RISK – 24 HOURS

Release Category	Release Frequency (/reactor year)	Mean Dose (person-sieverts)	Dose (person-REM)	Risk (person-REM/ reactor year)	Percent Contribution to Total Risk
CFI	1.89E-10	7.88E+03	7.88E+05	1.49E-04	1.2%
CFE	7.47E-09	8.51E+03	8.51E+05	6.36E-03	51.3%
IC	2.21E-07	7.19E+00	7.19E+02	1.59E-04	1.3%
BP	1.05E-08	2.91E+03	2.91E+05	3.06E-03	24.7%
CI	1.33E-09	2.01E+04	2.01E+06	2.67E-03	21.6%
CFL	3.45E-13	5.32E+03	5.32E+05	1.84E-07	0.0%
			Total Risk =	1.24E-02	100.0%

Table 49-11

POPULATION WHOLE BODY EDE DOSE RISK – 72 HOURS

Release Category	Release Frequency (/reactor year)	Mean Dose (person-sieverts)	Dose (person-REM)	Risk (person-REM/ reactor year)	Percent Contribution to Total Risk
CFI	1.89E-10	8.89E+03	8.89E+05	1.68E-04	1.2%
CFE	7.47E-09	9.36E+03	9.36E+05	6.99E-03	51.9%
IC	2.21E-07	8.80E+00	8.80E+02	1.94E-04	1.4%
BP	1.05E-08	3.11E+03	3.11E+05	3.27E-03	24.2%
CI	1.33E-09	2.14E+04	2.14E+06	2.85E-03	21.1%
CFL	3.45E-13	5.84E+03	5.84E+05	2.01E-07	0.0%
			Total Risk =	1.35E-02	100.0%

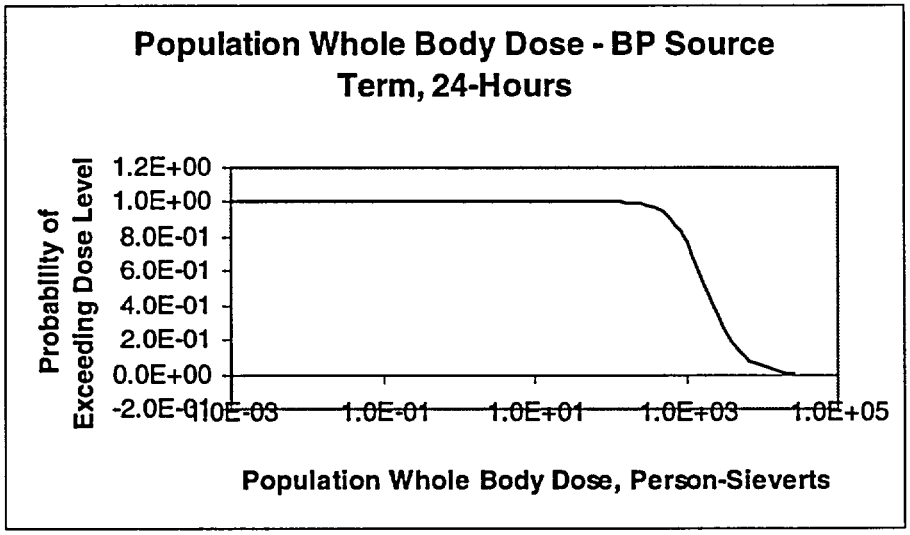


Figure 49-1

Population Whole Body Dose – BP Source Term, 24 Hours

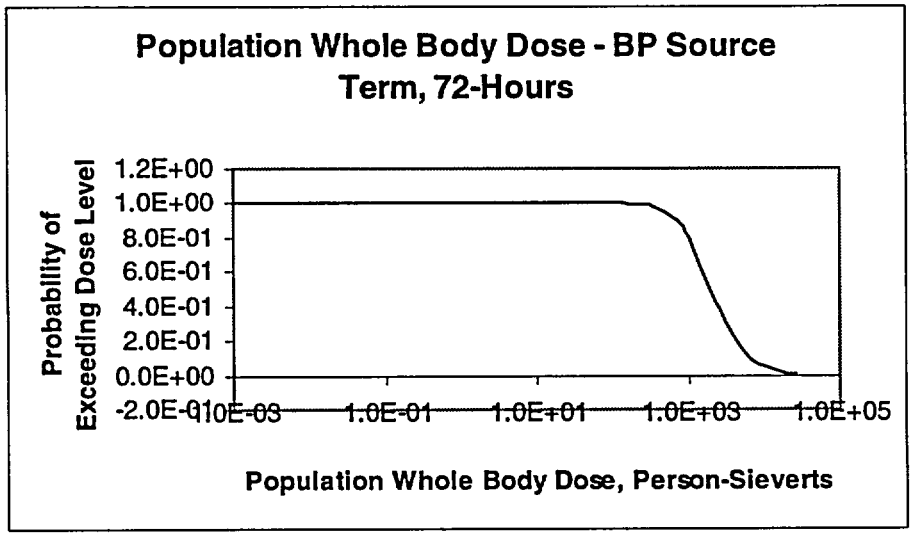


Figure 49-2

Population Whole Body Dose – BP Source Term 72 Hours

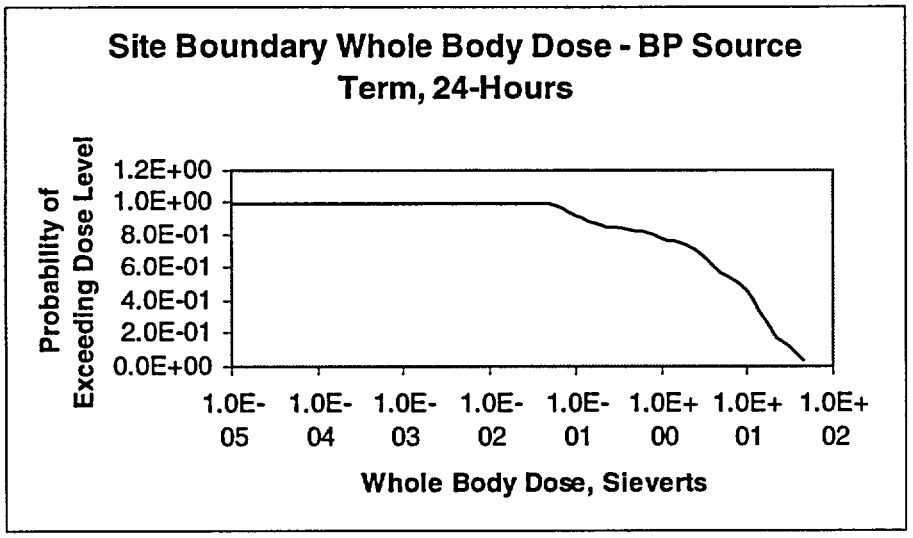


Figure 49-3

Site Boundary Whole Body Dose – BP Source Term, 24 Hours

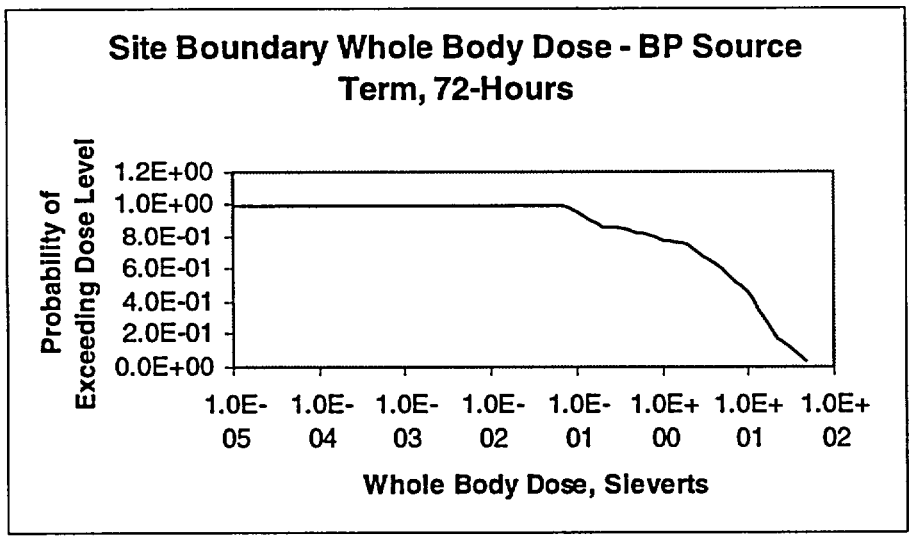


Figure 49-4

Site Boundary Whole Body Dose – BP Source Term, 72 Hours

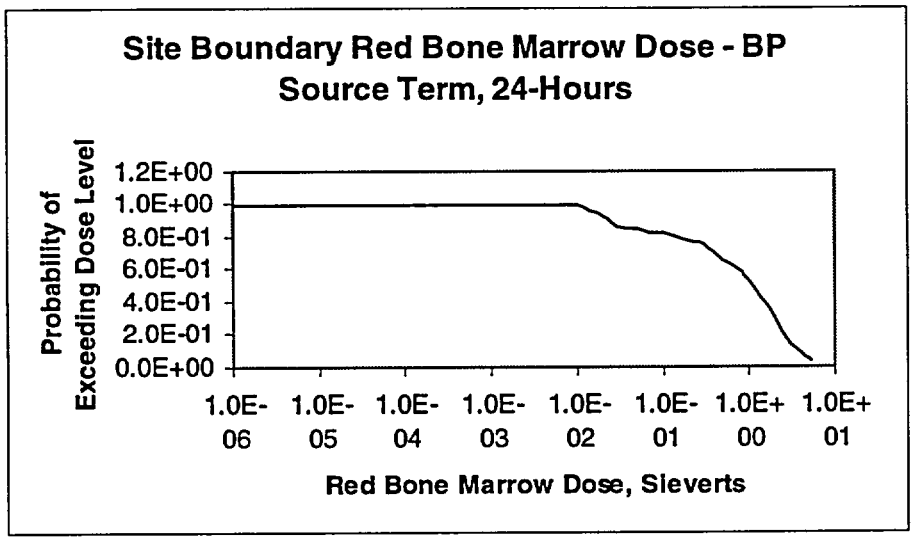


Figure 49-5

Site Boundary Red Bone Marrow Dose – BP Source Term, 24 Hours

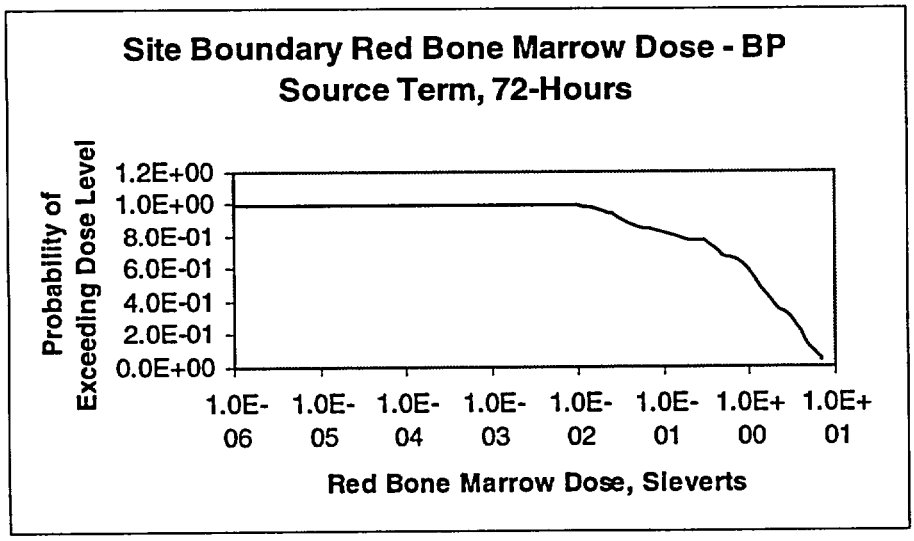


Figure 49-6

Site Boundary Red Bone Marrow Dose – BP Source Term, 72 Hours

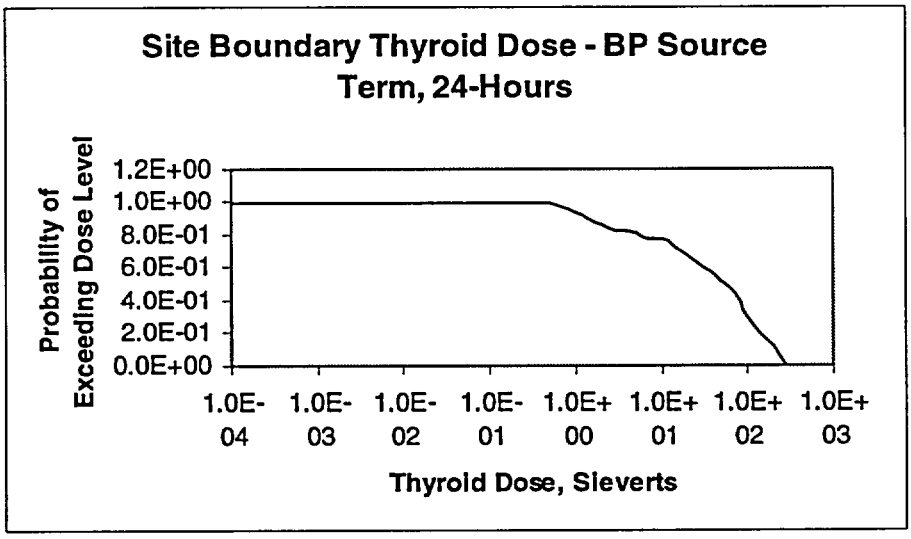


Figure 49-7

Site Boundary Thyroid Dose – BP Source Term, 24 Hours

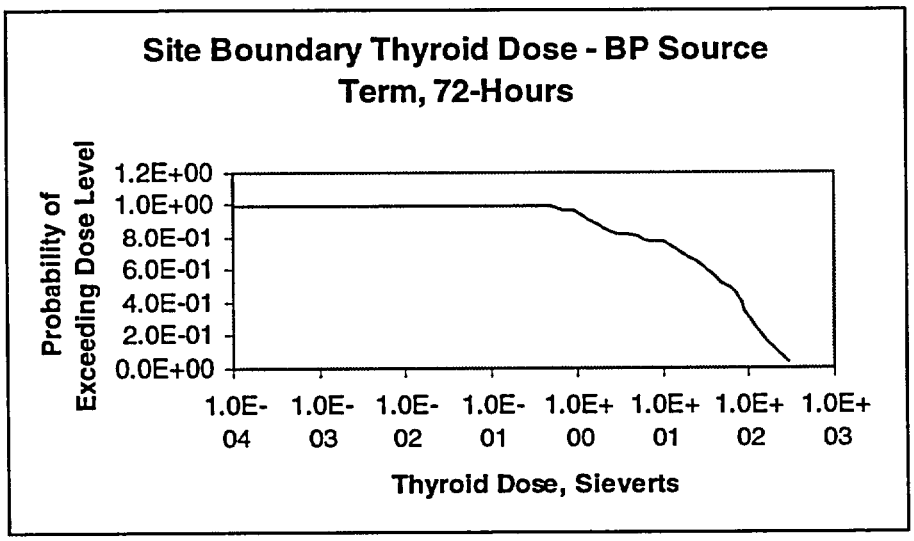


Figure 49-8

Site Boundary Thyroid Dose – BP Source Term, 72 Hours

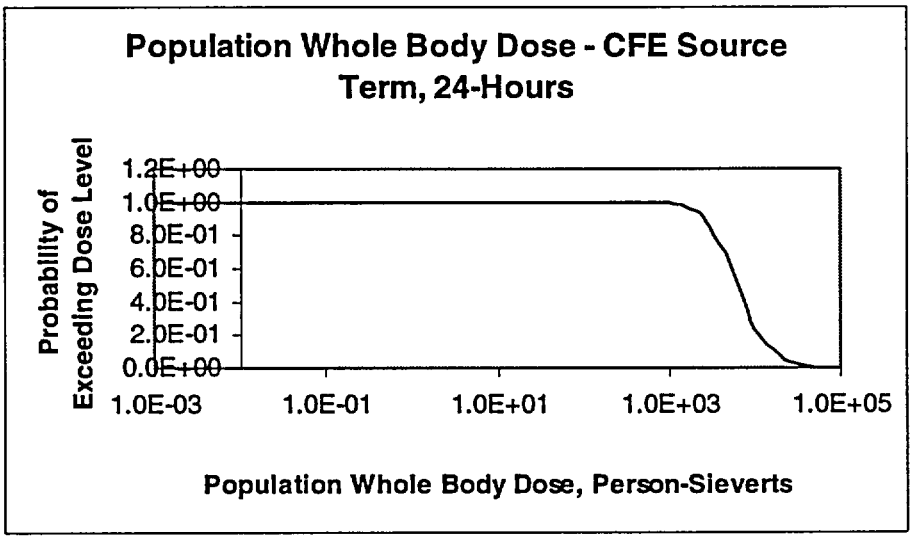


Figure 49-9

Population Whole Body Dose – CFE Source Term, 24 Hours

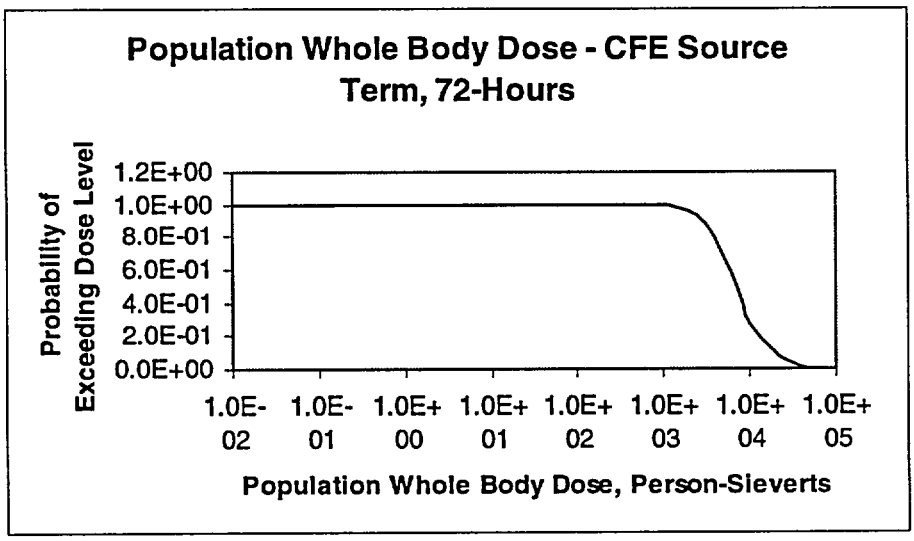


Figure 49-10

Population Whole Body Dose – CFE Source Term, 72 Hours

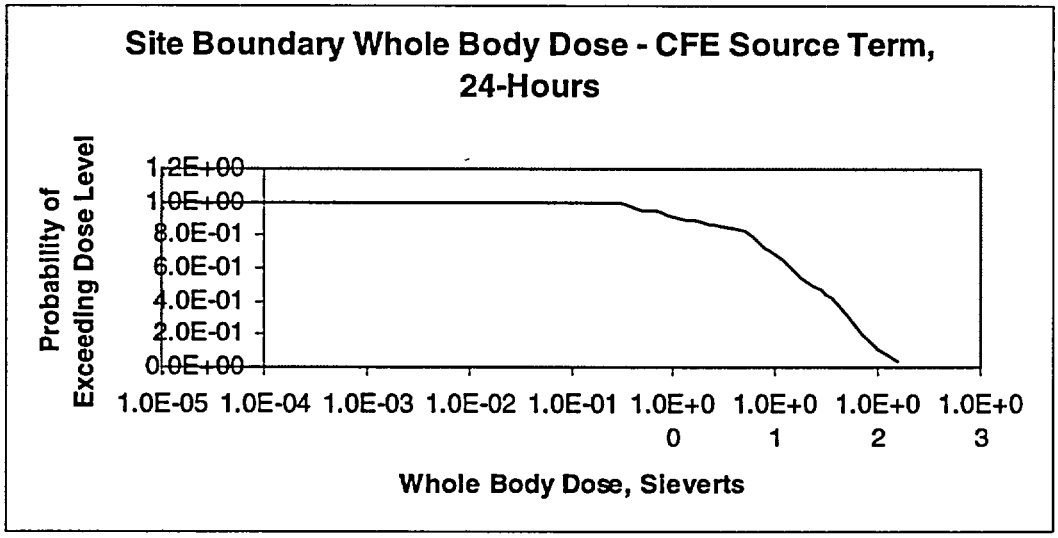


Figure 49-11

Site Boundary Whole Body Dose – CFE Source Term, 24 Hours

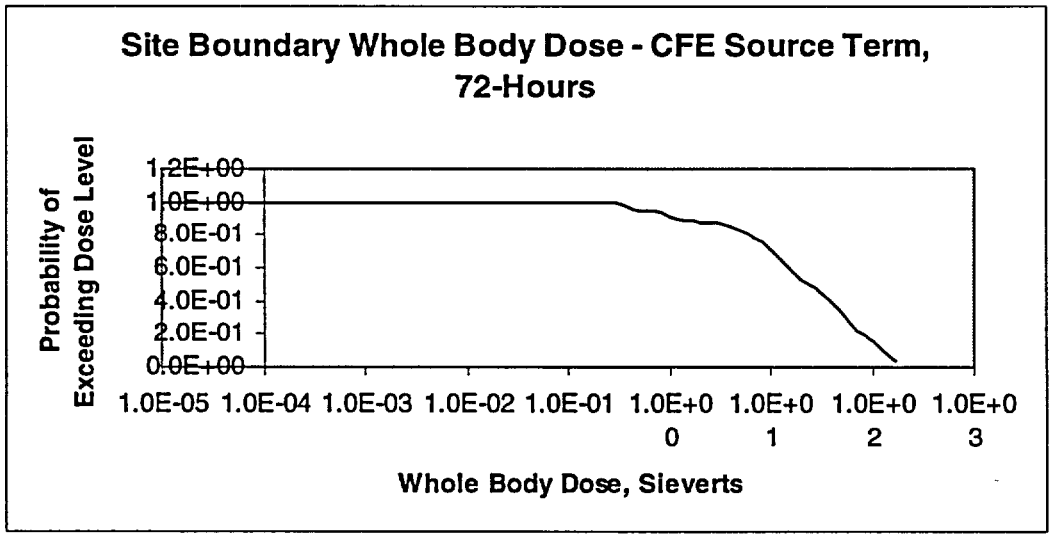


Figure 49-12

Site Boundary Whole Body Dose – CFE Source Term, 72 Hours

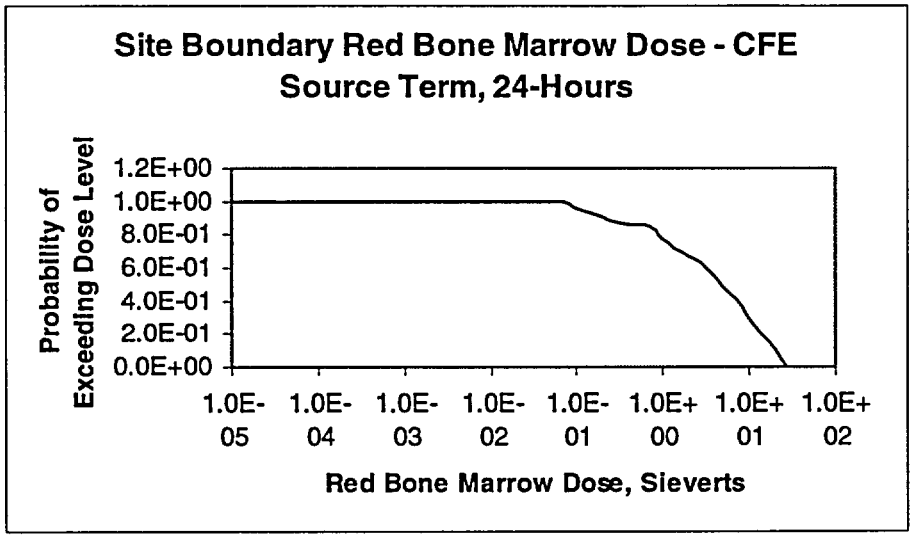


Figure 49-13

Site Boundary Red Bone Marrow Dose – CFE Source Term, 24 Hours

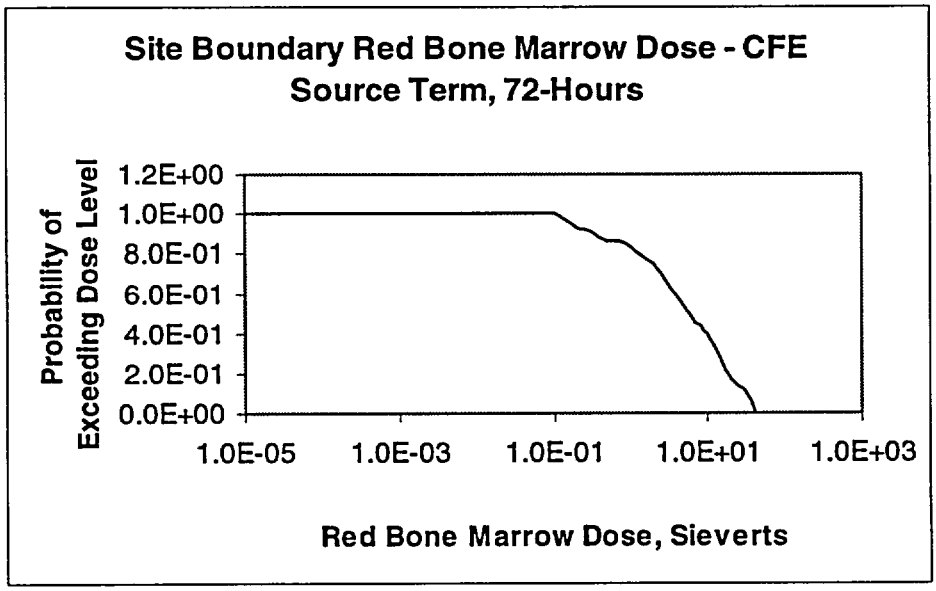


Figure 49-14

Site Boundary Red Bone Marrow Dose – CFE Source Term, 72 Hours

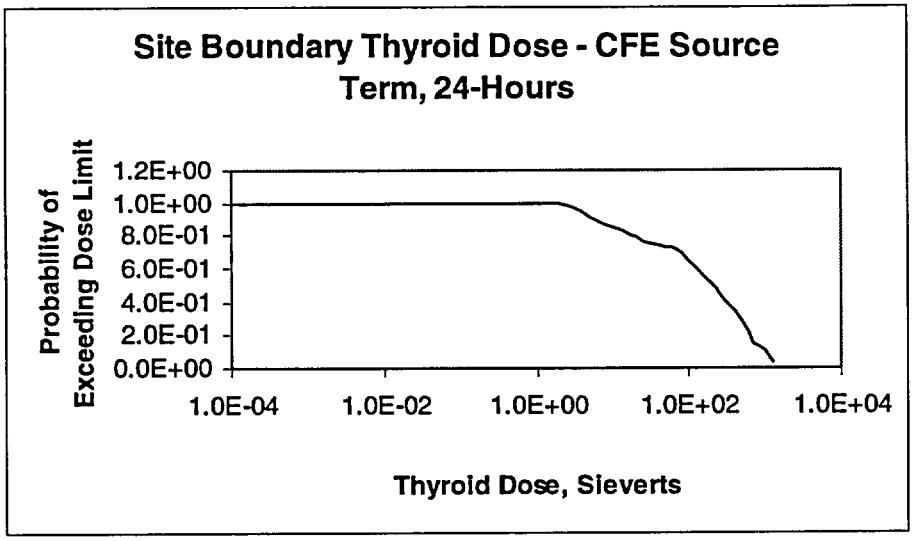


Figure 49-15

Site Boundary Thyroid Dose – CFE Source Term, 24 Hours

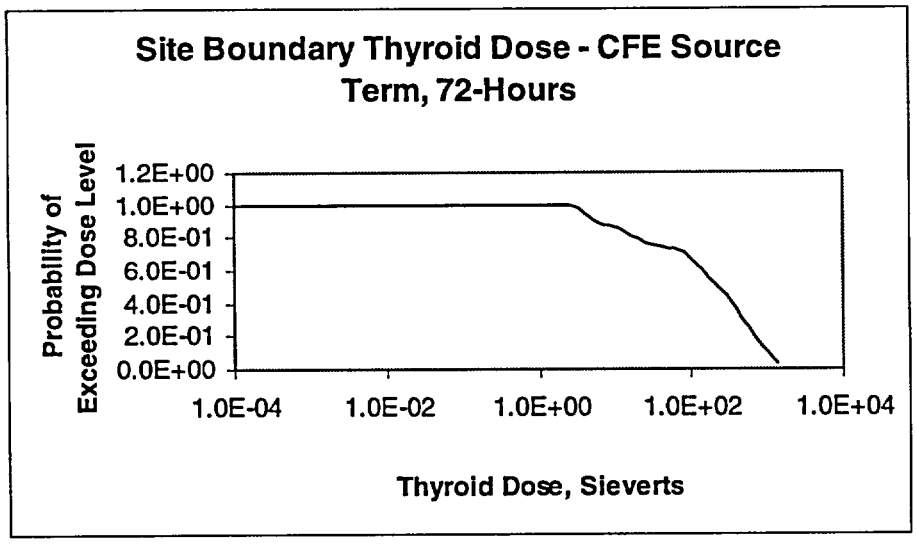


Figure 49-16

Site Boundary Thyroid Dose – CFE Source Term, 72 Hours

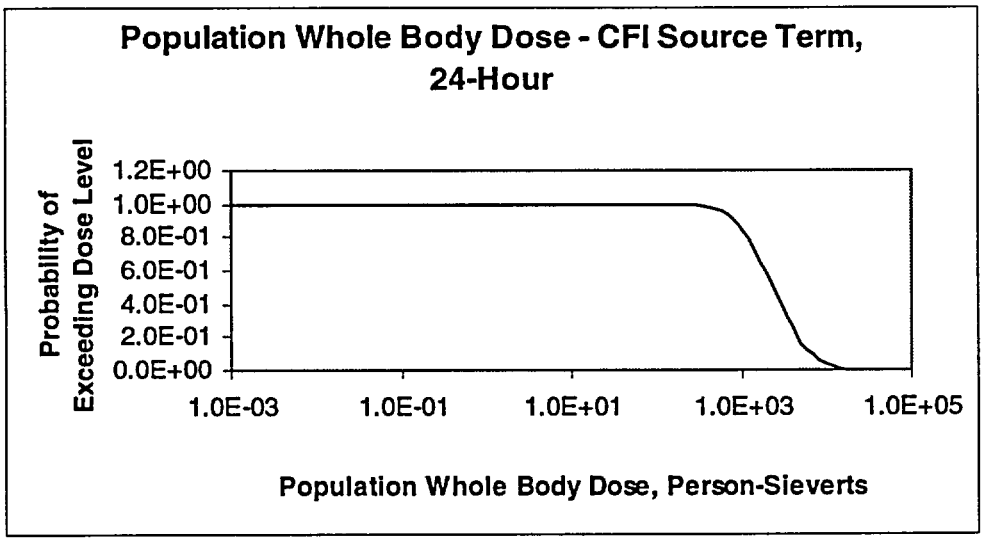


Figure 49-17

Population Whole Body Dose – CFI Source Term, 24 Hours

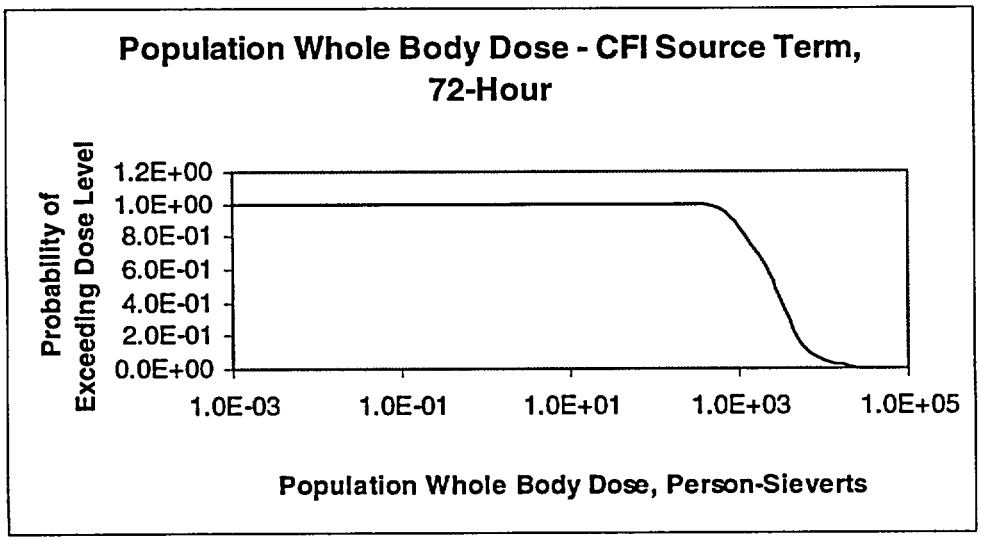


Figure 49-18

Population Whole Body Dose – CFI Source Term, 72 Hours

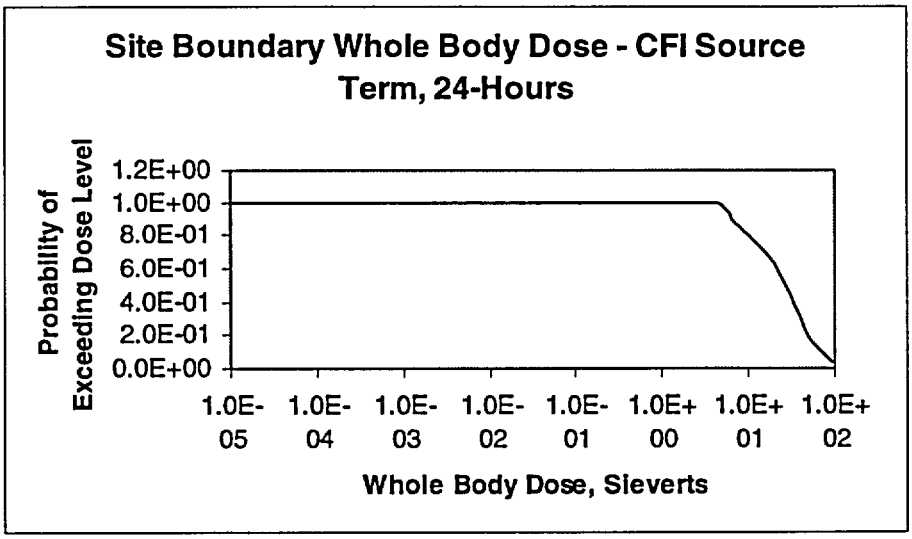


Figure 49-19

Site Boundary Whole Body Dose – CFI Source Term, 24 Hours

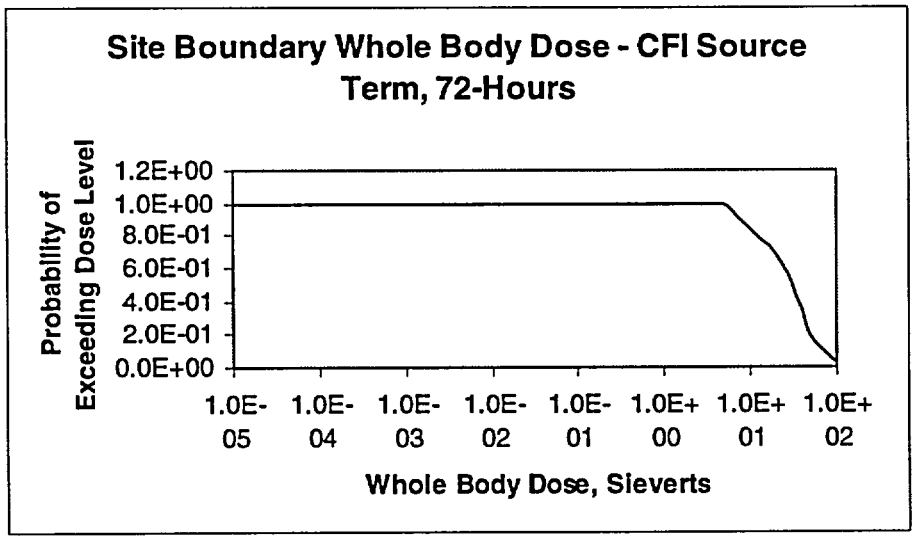


Figure 49-20

Site Boundary Whole Body Dose – CFI Source Term, 72 Hours

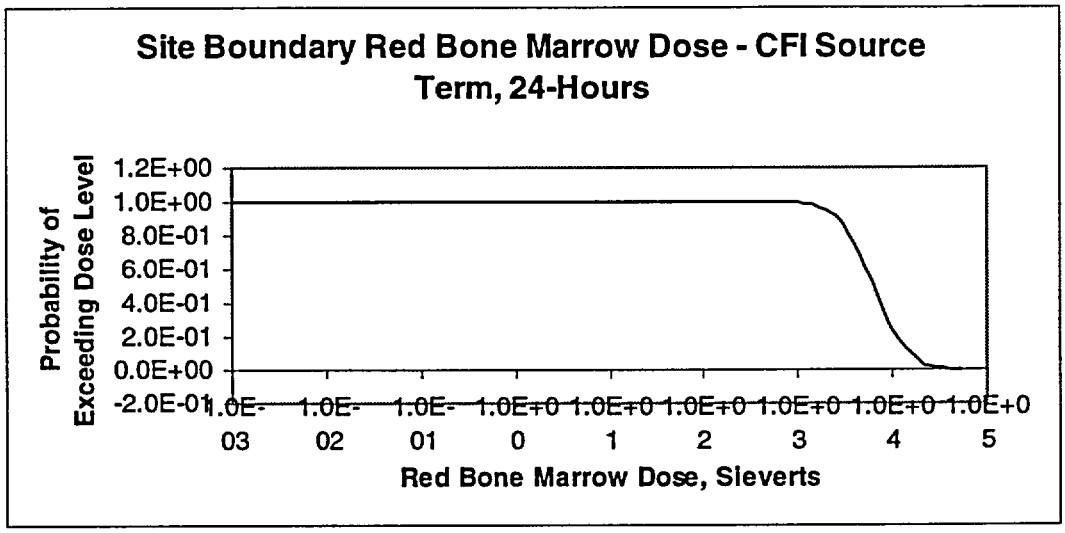


Figure 49-21

Site Boundary Red Bone Marrow Dose – CFI Source Term, 24 Hours

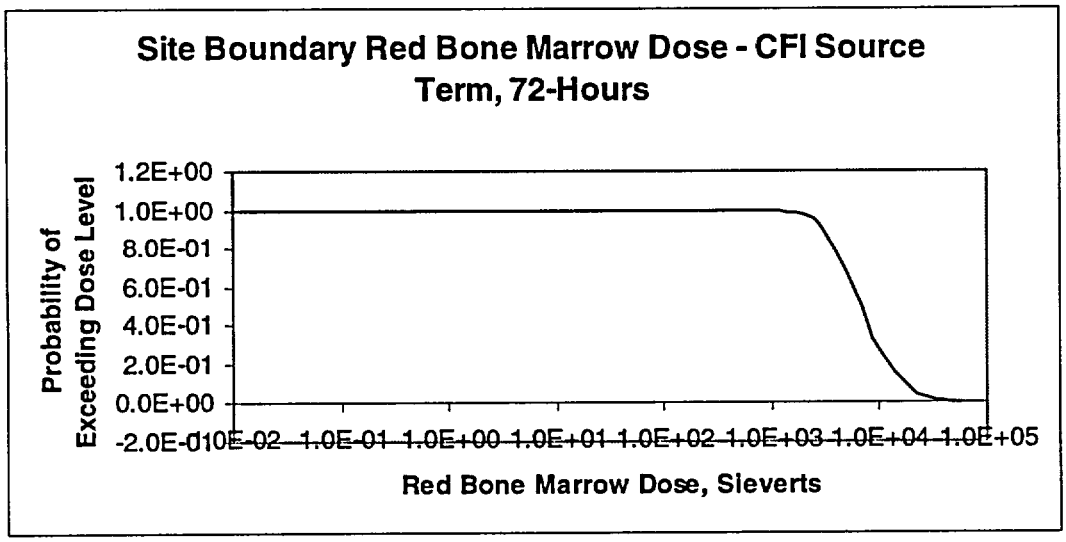


Figure 49-22

Site Boundary Red Bone Marrow Dose – CFI Source Term, 72 Hours

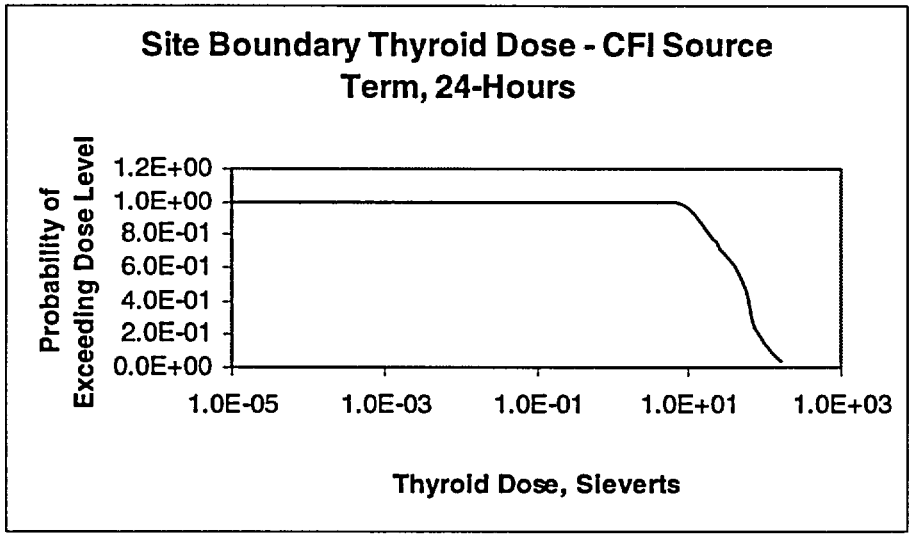


Figure 49-23

Site Boundary Thyroid Dose – CFI Source Term, 24 Hours

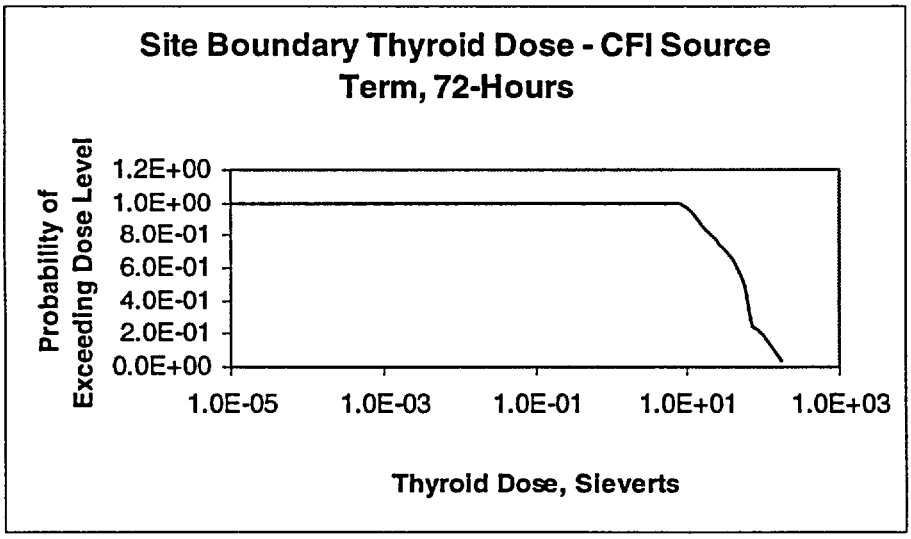


Figure 49-24

Site Boundary Thyroid Dose – CFI Source Term, 72 Hours

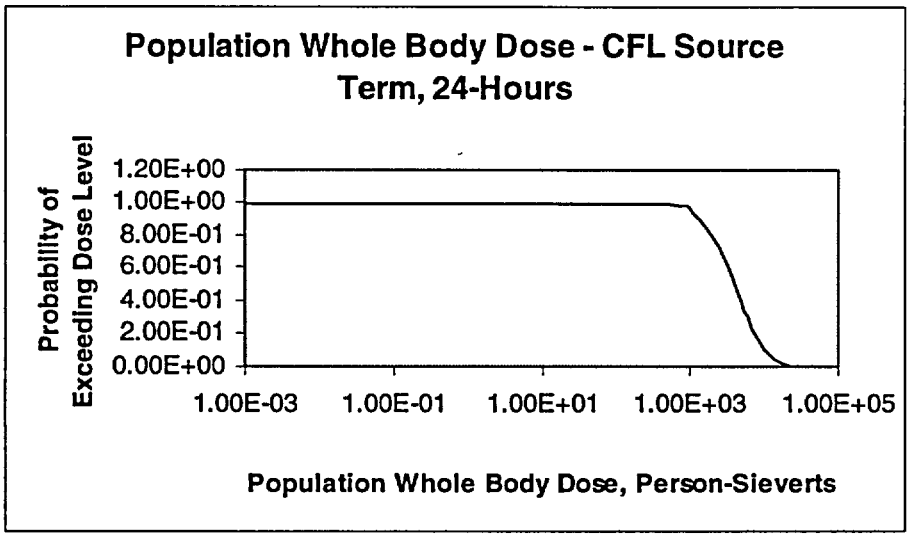


Figure 49-25

Population Whole Body Dose – CFL Source Term, 24 Hours

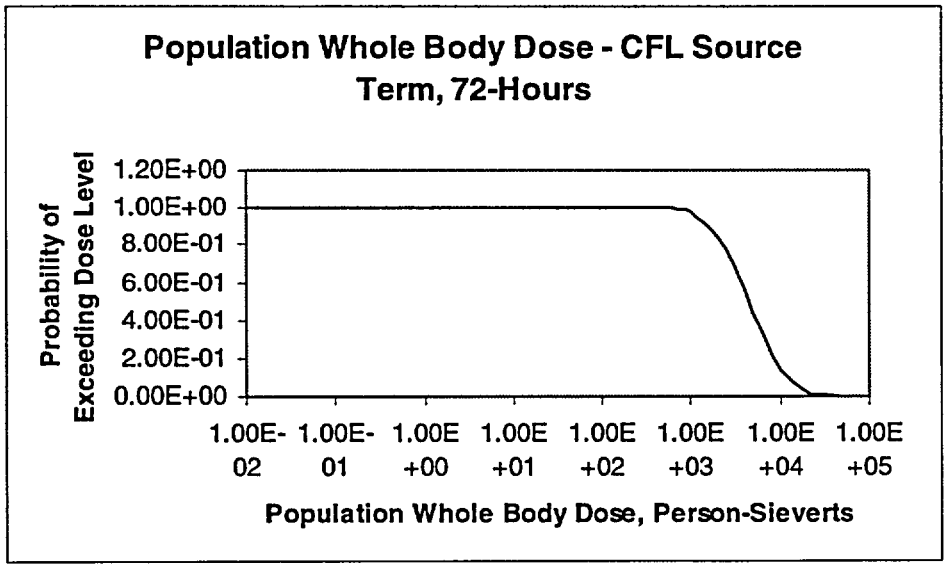


Figure 49-26

Population Whole Body Dose – CFL Source Term, 72 Hours

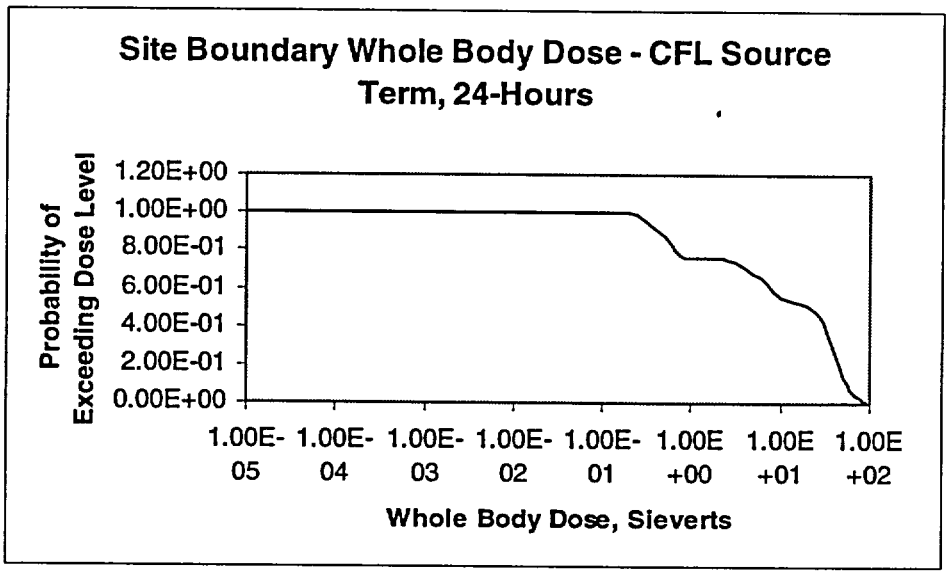


Figure 49-27

Site Boundary Whole Body Dose – CFL Source Term, 24 Hours

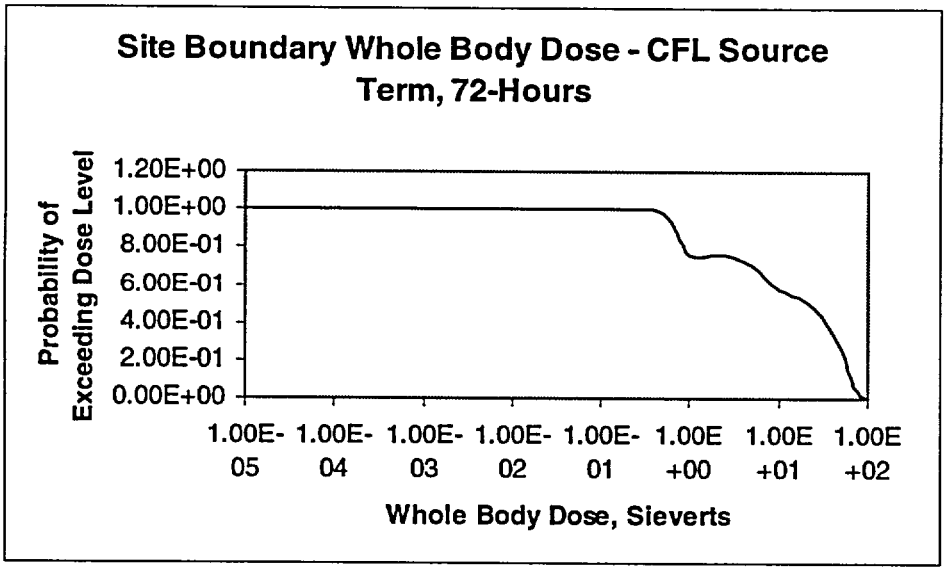


Figure 49-28

Site Boundary Whole Body Dose – CFL Source Term, 72 Hours

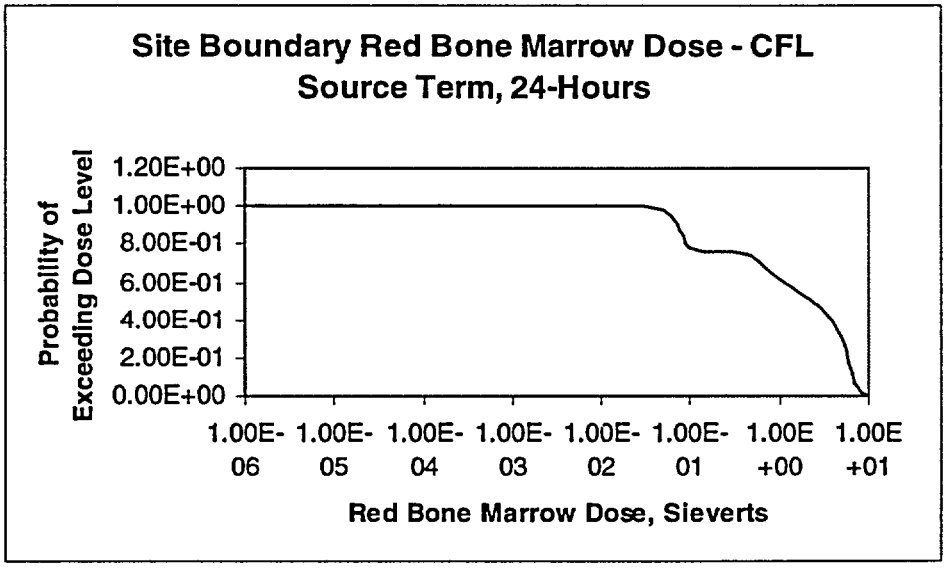


Figure 49-29

Site Boundary Red Bone Marrow Dose – CFL Source Term, 24 Hours

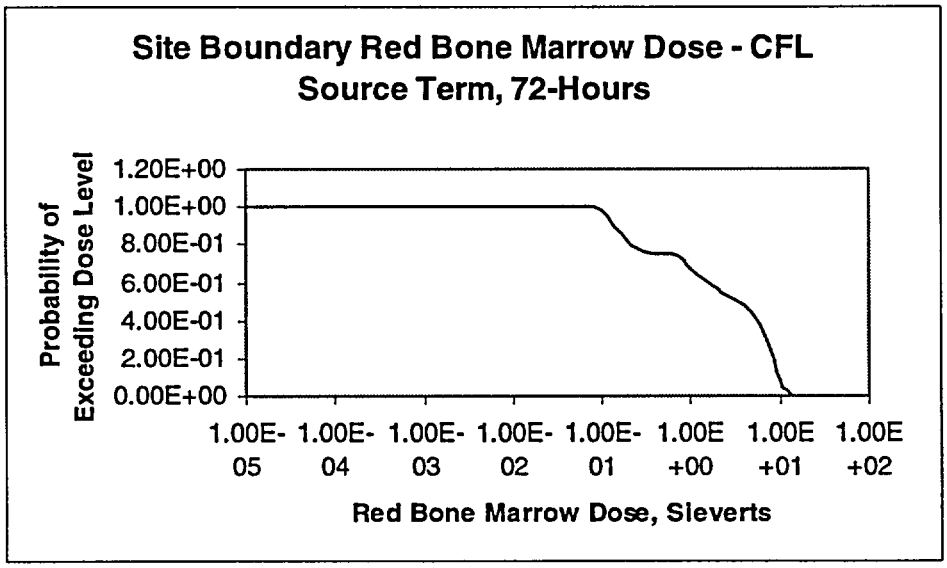


Figure 49-30

Site Boundary Red Bone Marrow Dose – CFL Source Term, 72 Hours

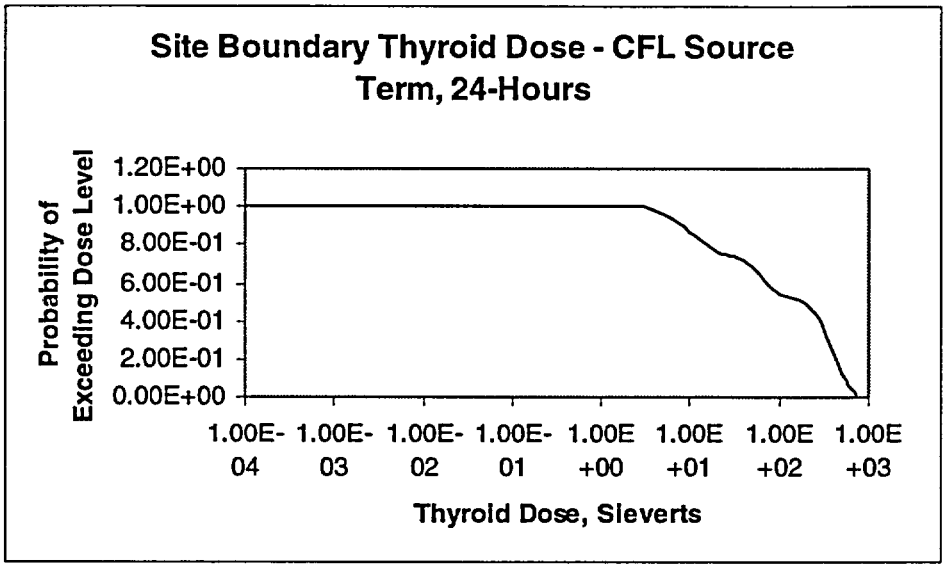


Figure 49-31

Site Boundary Thyroid Dose – CFL Source Term, 24 Hours

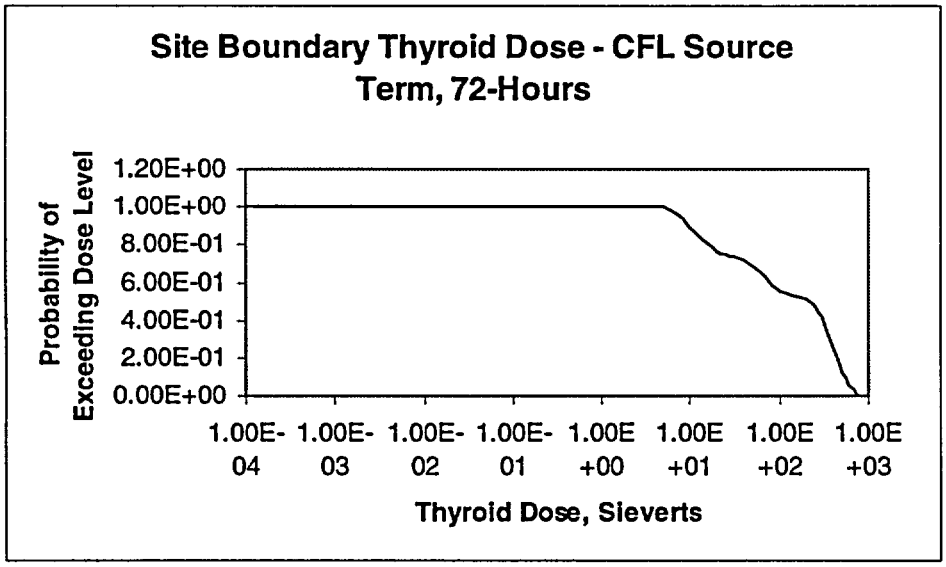


Figure 49-32

Site Boundary Thyroid Dose – CFL Source Term, 72 Hours

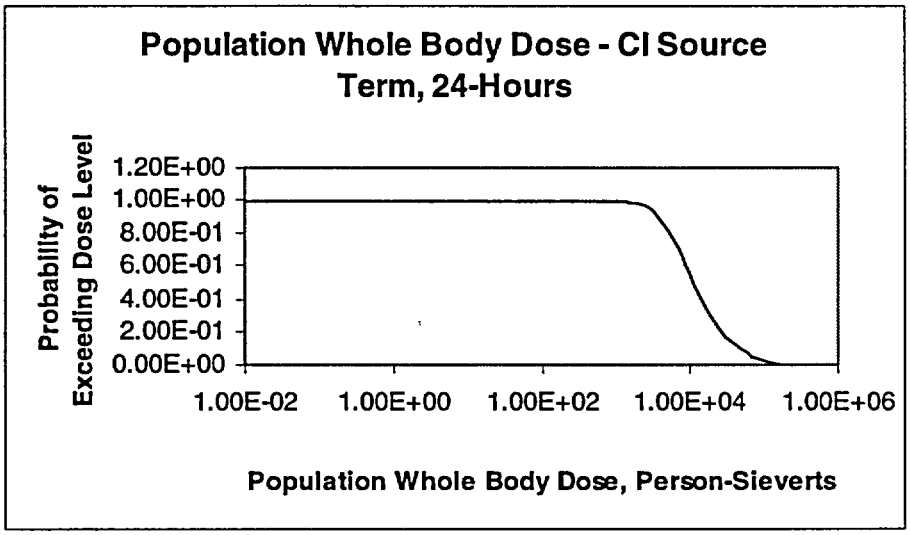


Figure 49-33

Population Whole Body Dose – CI Source Term, 24 Hours

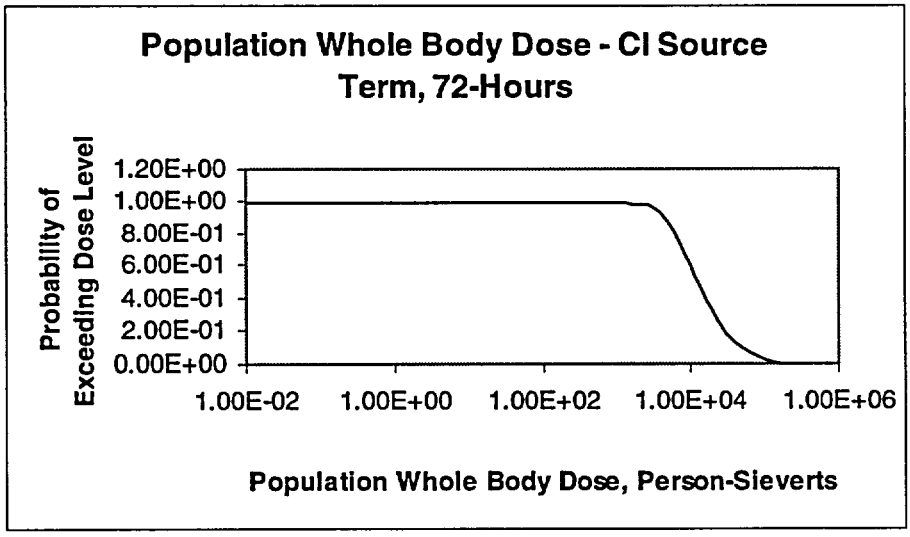


Figure 49-34

Population Whole Body Dose – CI Source Term, 72 Hours

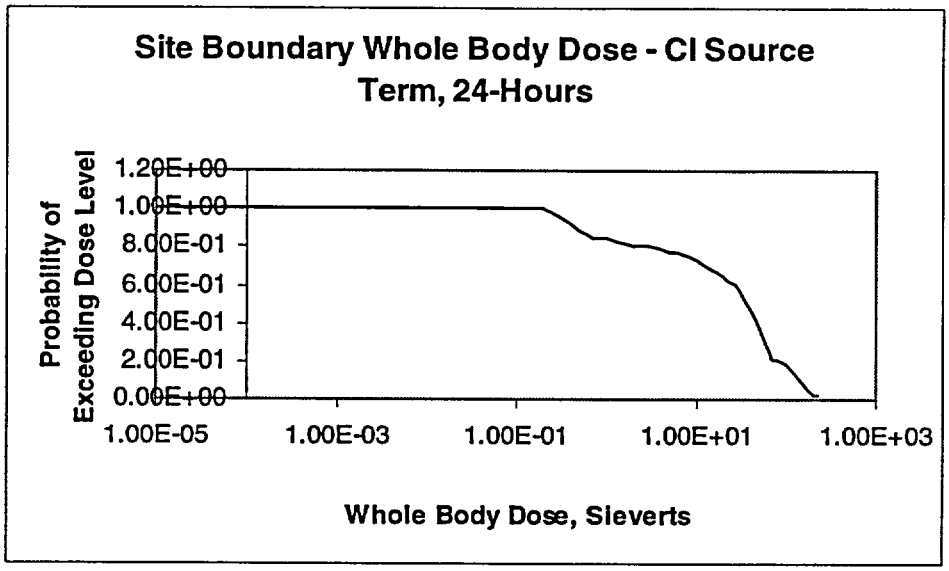


Figure 49-35

Site Boundary Whole Body Dose – CI Source Term, 24 Hours

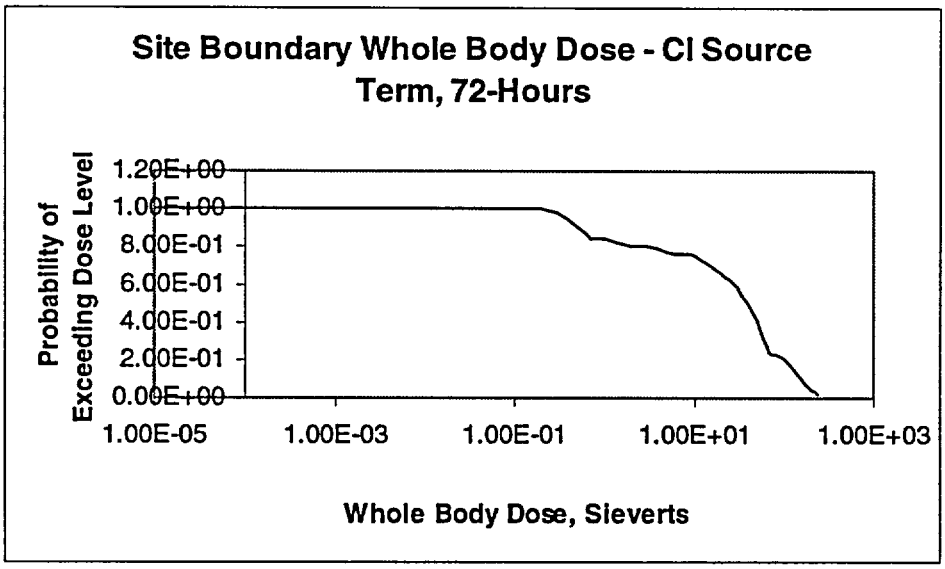


Figure 49-36

Site Boundary Whole Body Dose – CI Source Term, 72 Hours

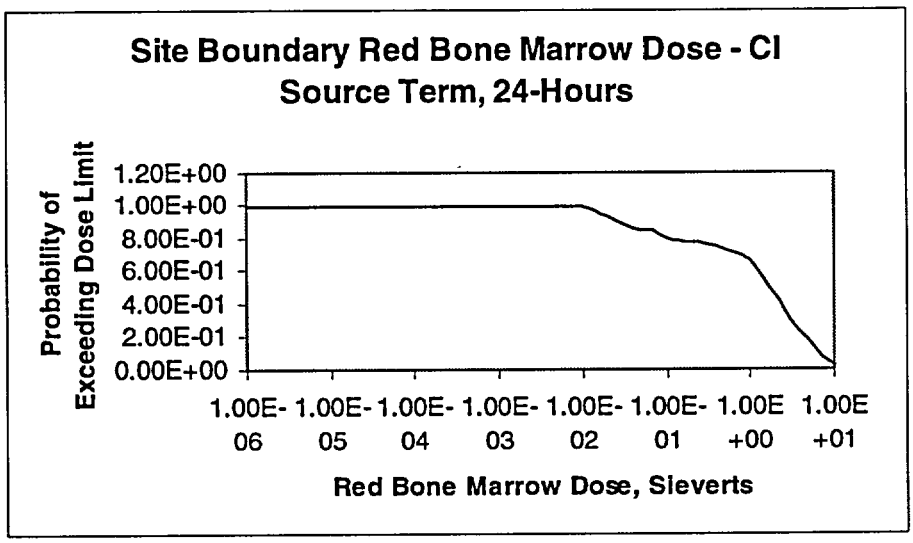


Figure 49-37

Site Boundary Red Bone Marrow Dose – CI Source Term, 24 Hours

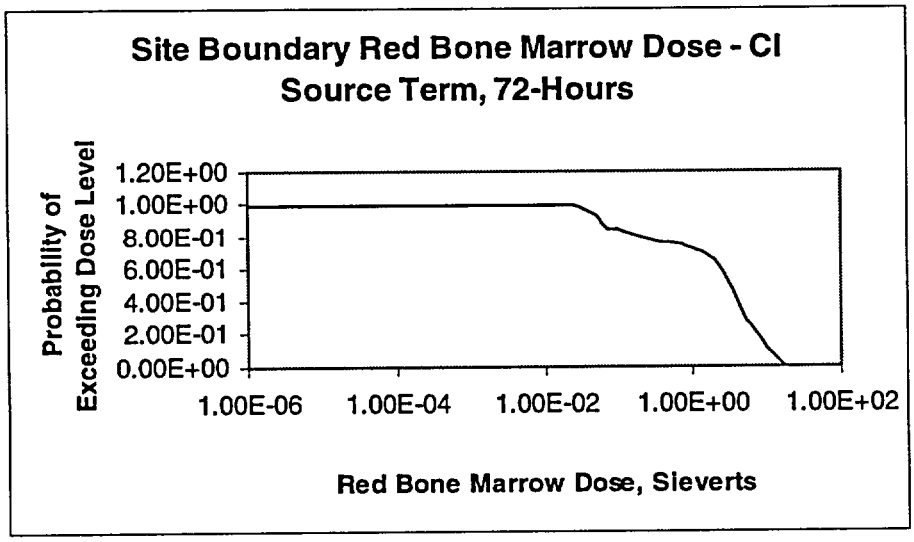


Figure 49-38

Site Boundary Red Bone Marrow Dose – CI Source Term, 72 Hours

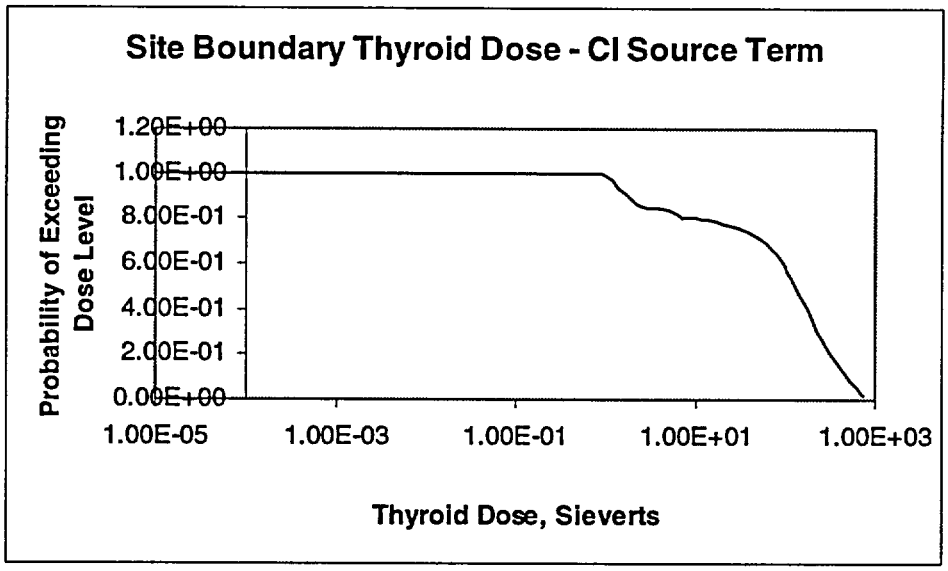


Figure 49-39

Site Boundary Thyroid Dose – CI Source Term, 24 Hours

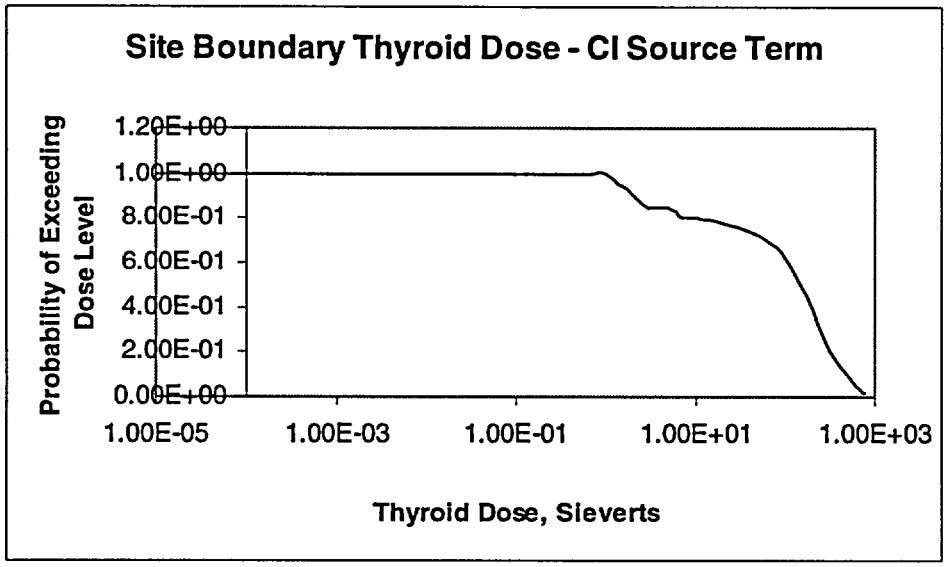


Figure 49-40

Site Boundary Thyroid Dose – CI Source Term, 72 Hours

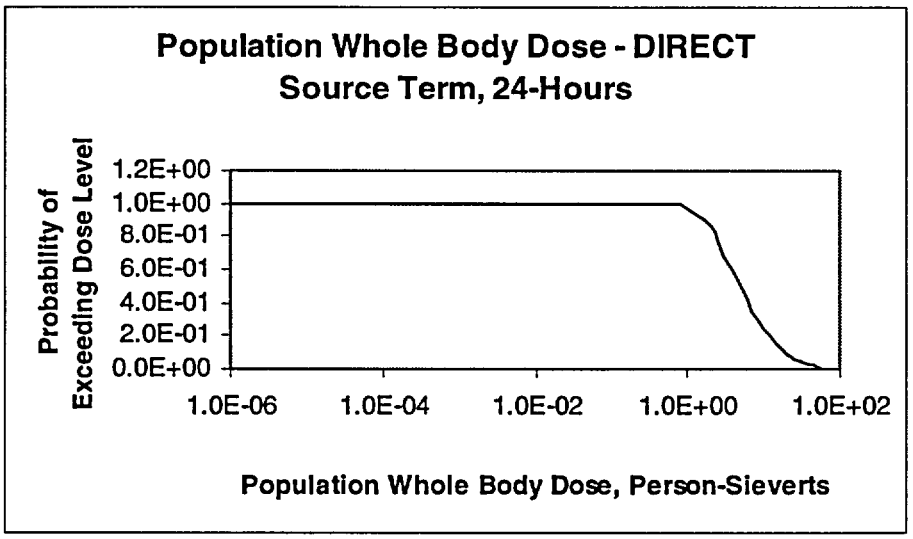


Figure 49-41

Population Whole Body Dose – DIRECT Source Term, 24 Hours

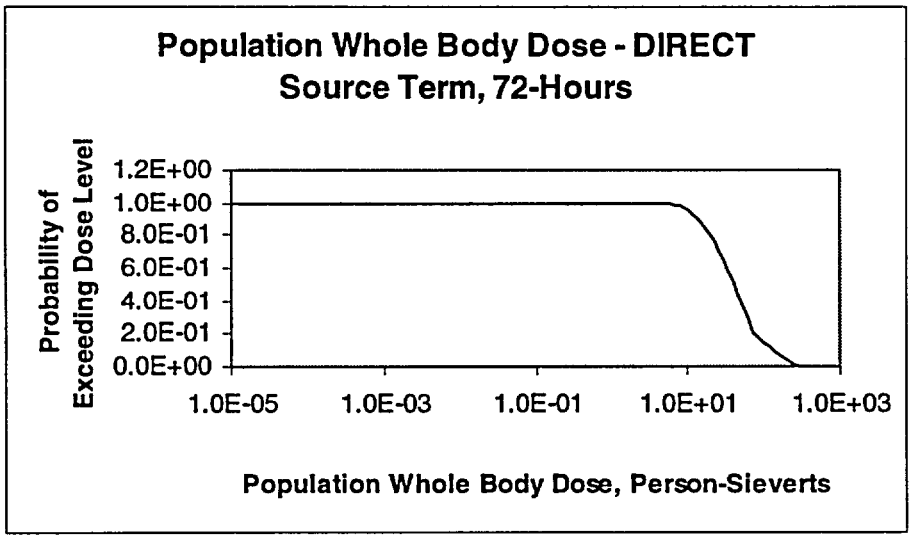


Figure 49-42

Population Whole Body Dose – DIRECT Source Term, 72 Hours

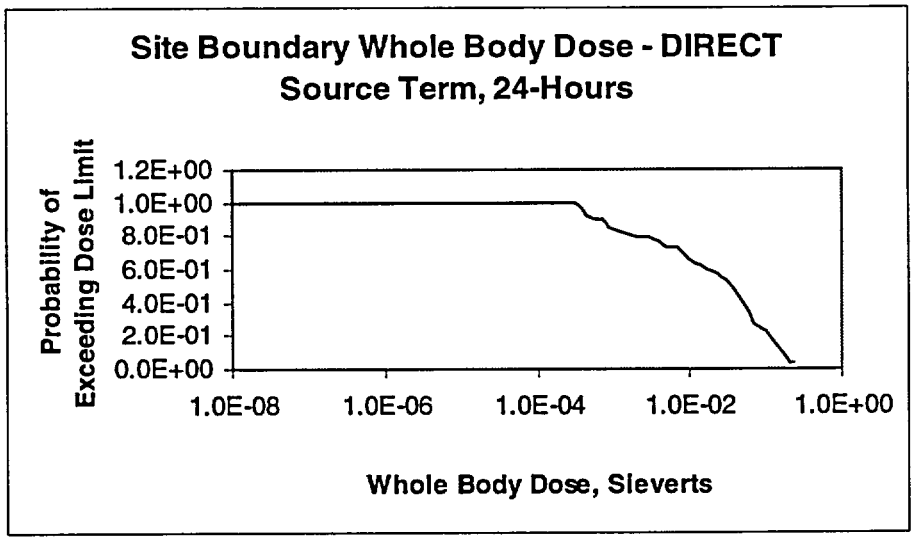


Figure 49-43

Site Boundary Whole Body Dose – DIRECT Source Term, 24 Hours

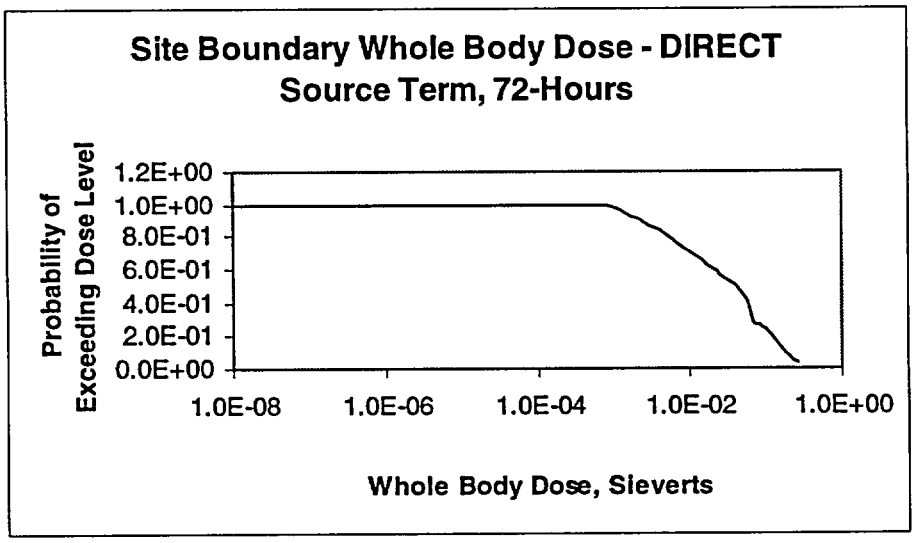


Figure 49-44

Site Boundary Whole Body Dose – DIRECT Source Term, 72 Hours

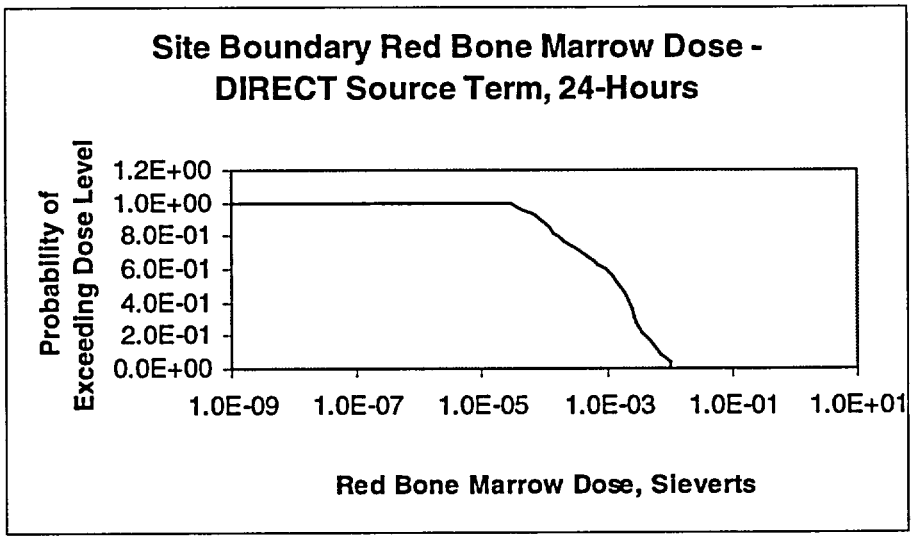


Figure 49-45

Site Boundary Red Bone Marrow Dose – DIRECT Source Term, 24 Hours

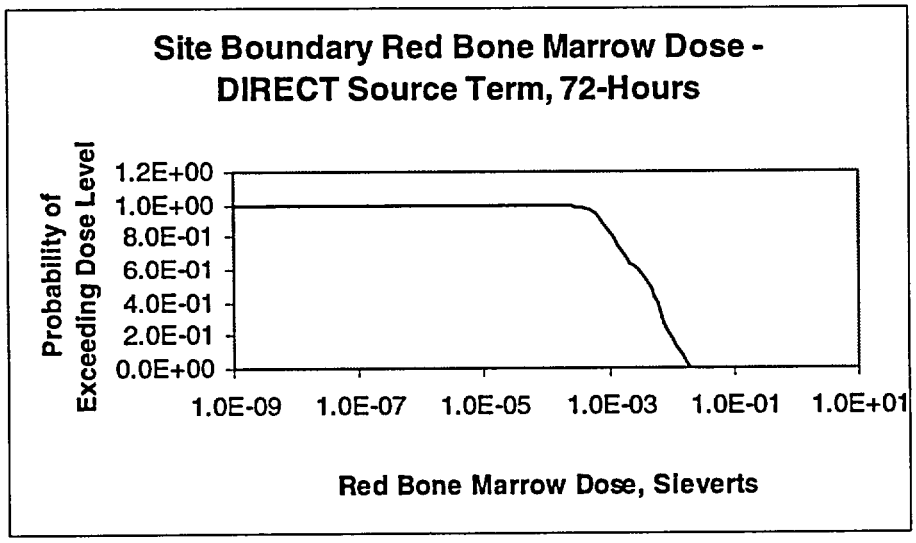


Figure 49-46

Site Boundary Red Bone Marrow Dose – DIRECT Source Term, 72 Hours

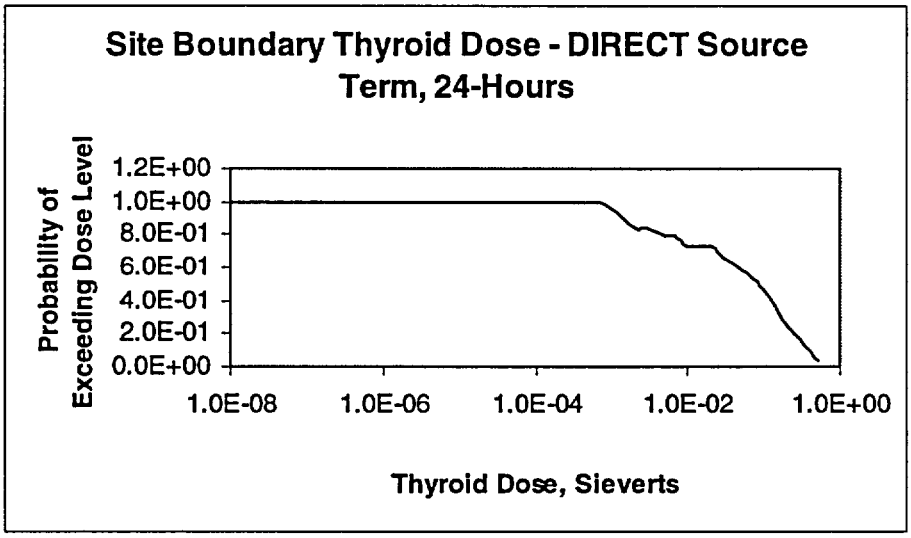


Figure 49-47

Site Boundary Thyroid Dose – DIRECT Source Term, 24 Hours

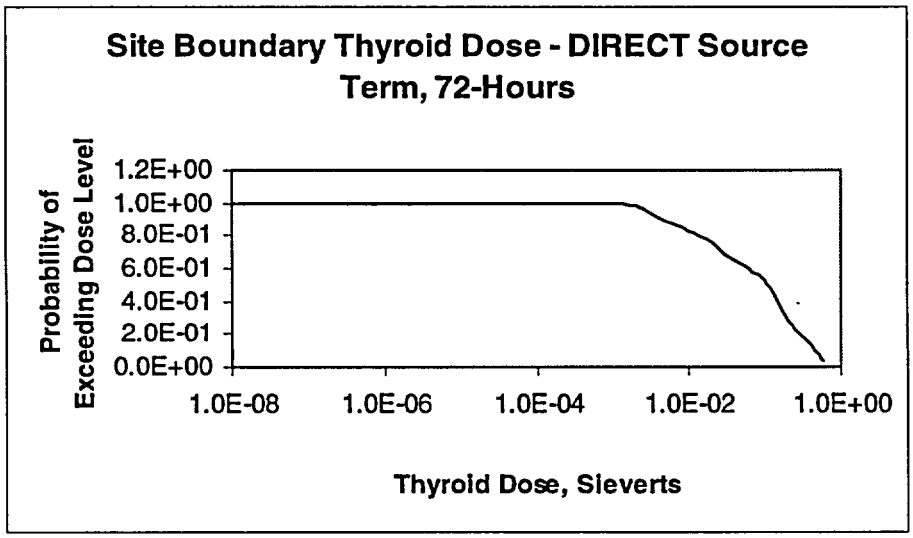


Figure 49-48

Site Boundary Thyroid Dose – DIRECT Source Term, 72 Hours

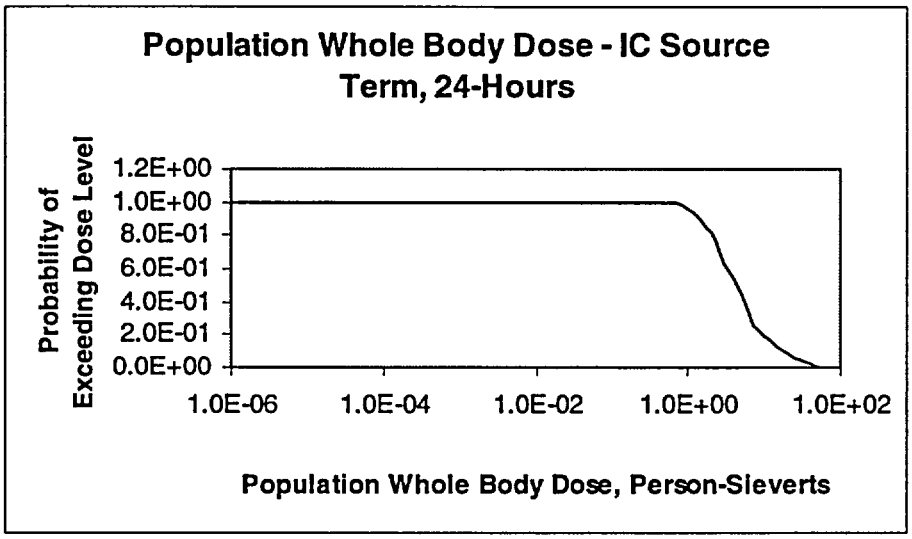


Figure 49-49

Population Whole Body Dose – IC Source Term, 24 Hours

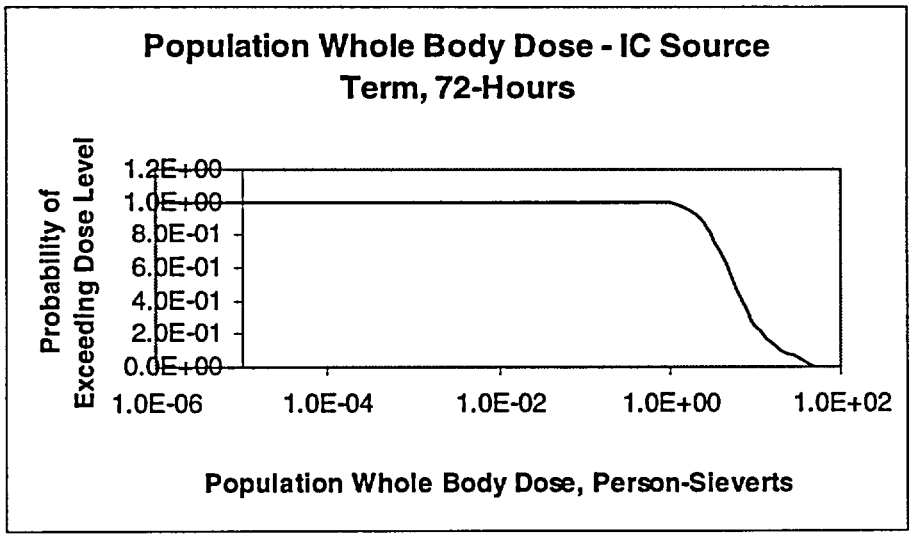


Figure 49-50

Population Whole Body Dose – IC Source Term, 72 Hours

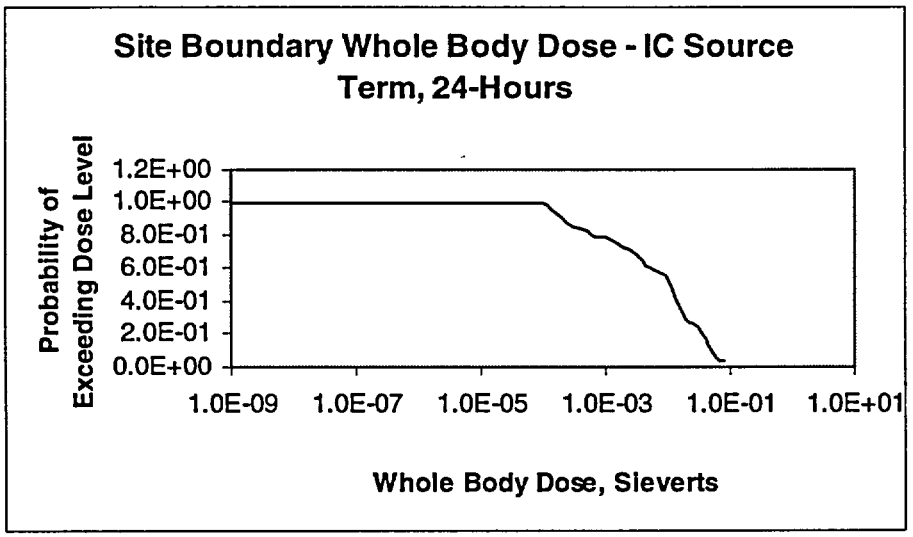


Figure 49-51

Site Boundary Whole Body Dose – IC Source Term, 24 Hours

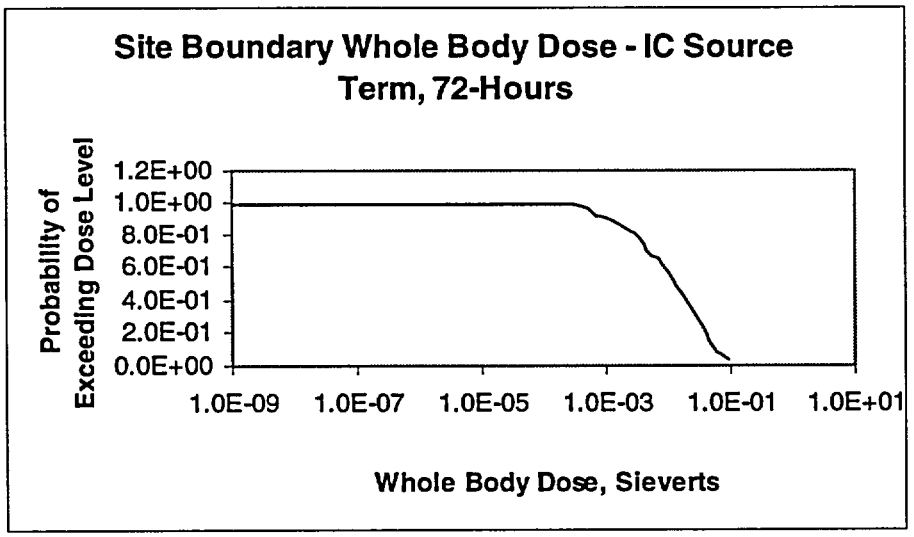


Figure 49-52

Site Boundary Whole Body Dose – IC Source Term, 72 Hours

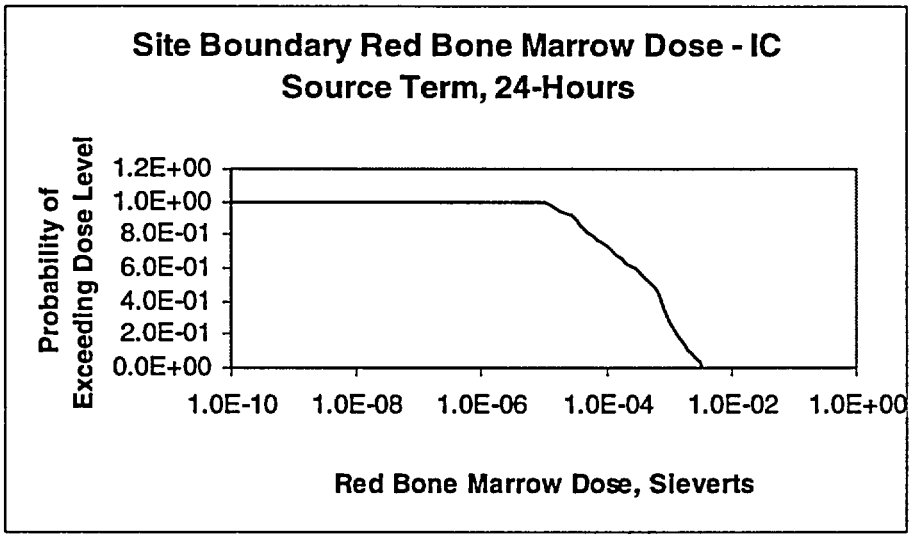


Figure 49-53

Site Boundary Red Bone Marrow Dose – IC Source Term, 24 Hours

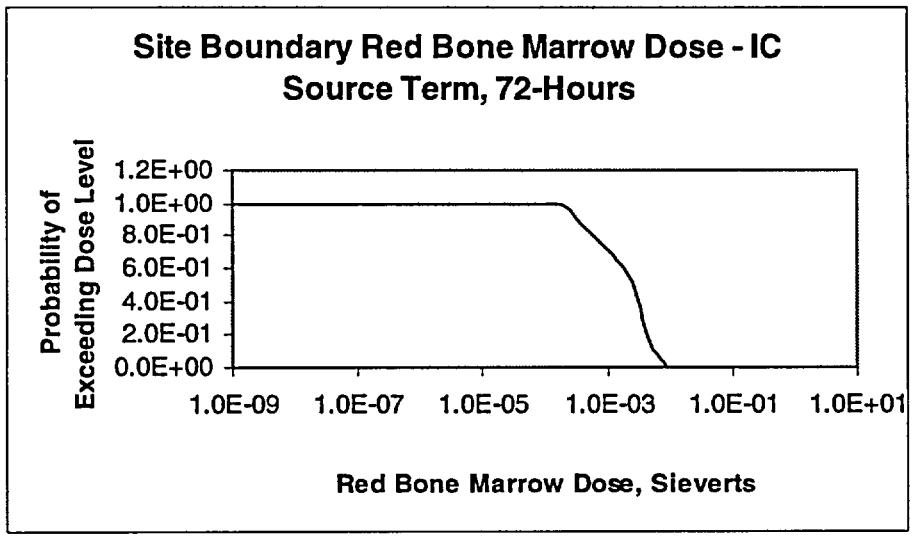


Figure 49-54

Site Boundary Red Bone Marrow Dose – IC Source Term, 72 Hours

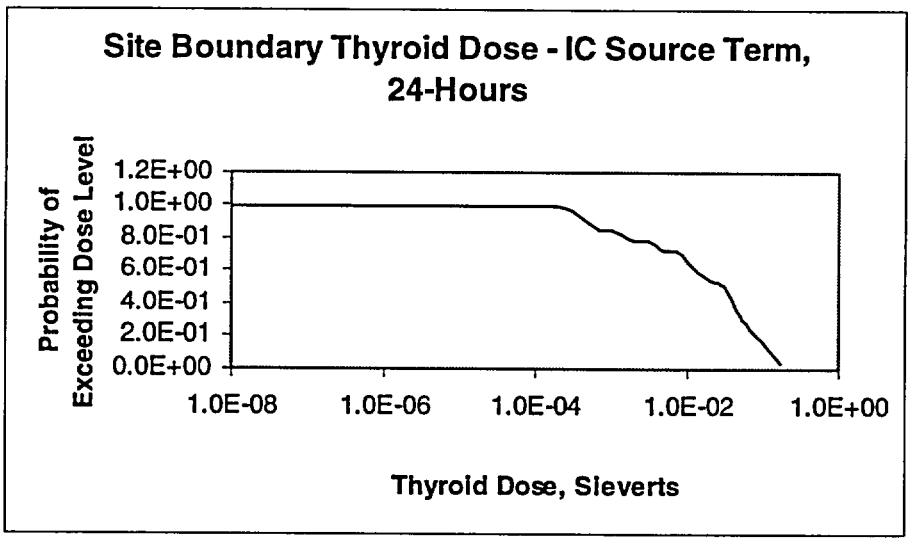


Figure 49-55

Site Boundary Thyroid Dose – IC Source Term, 24 Hours

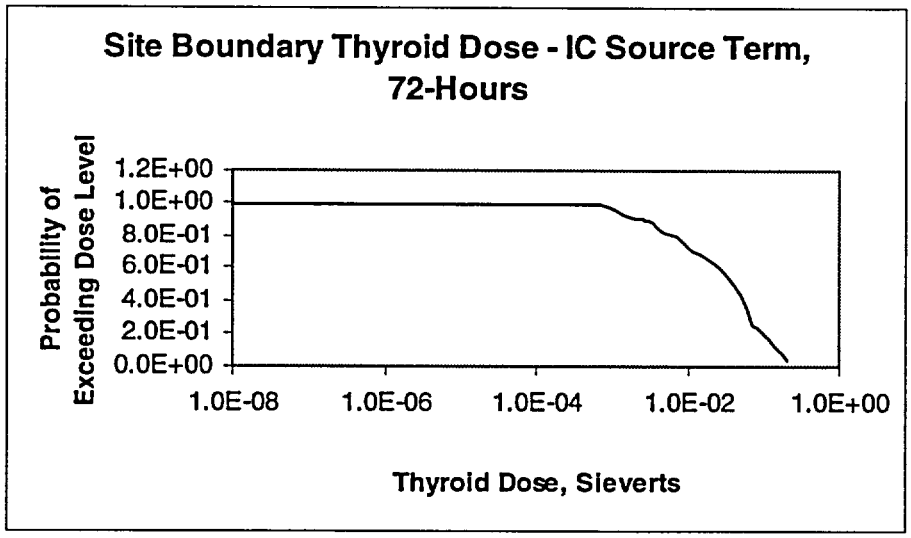


Figure 49-56

Site Boundary Thyroid Dose – IC Source Term, 72 Hours

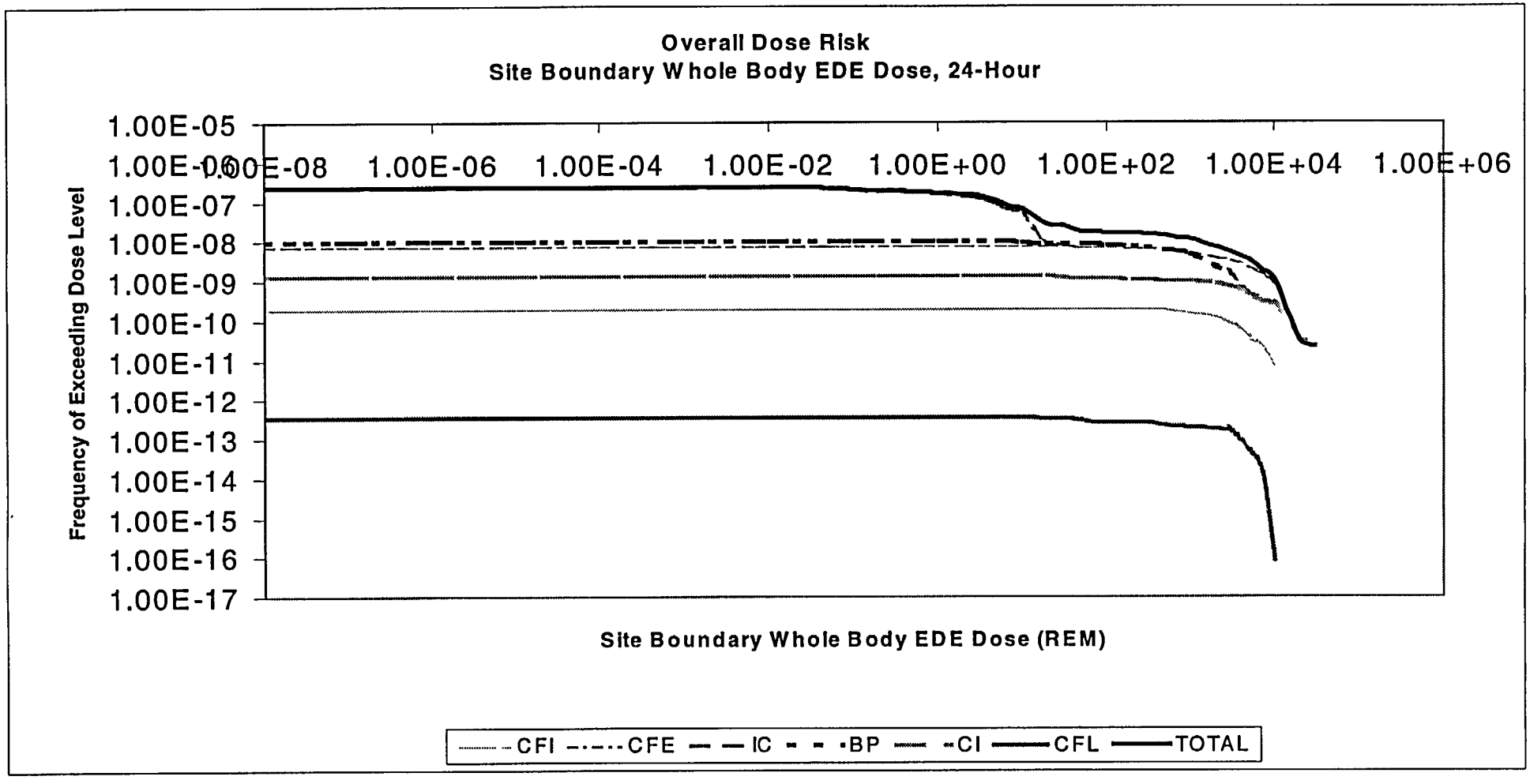


Figure 49-57

Overall Dose Risk Site Boundary Whole Body EDE Dose, 24 Hours

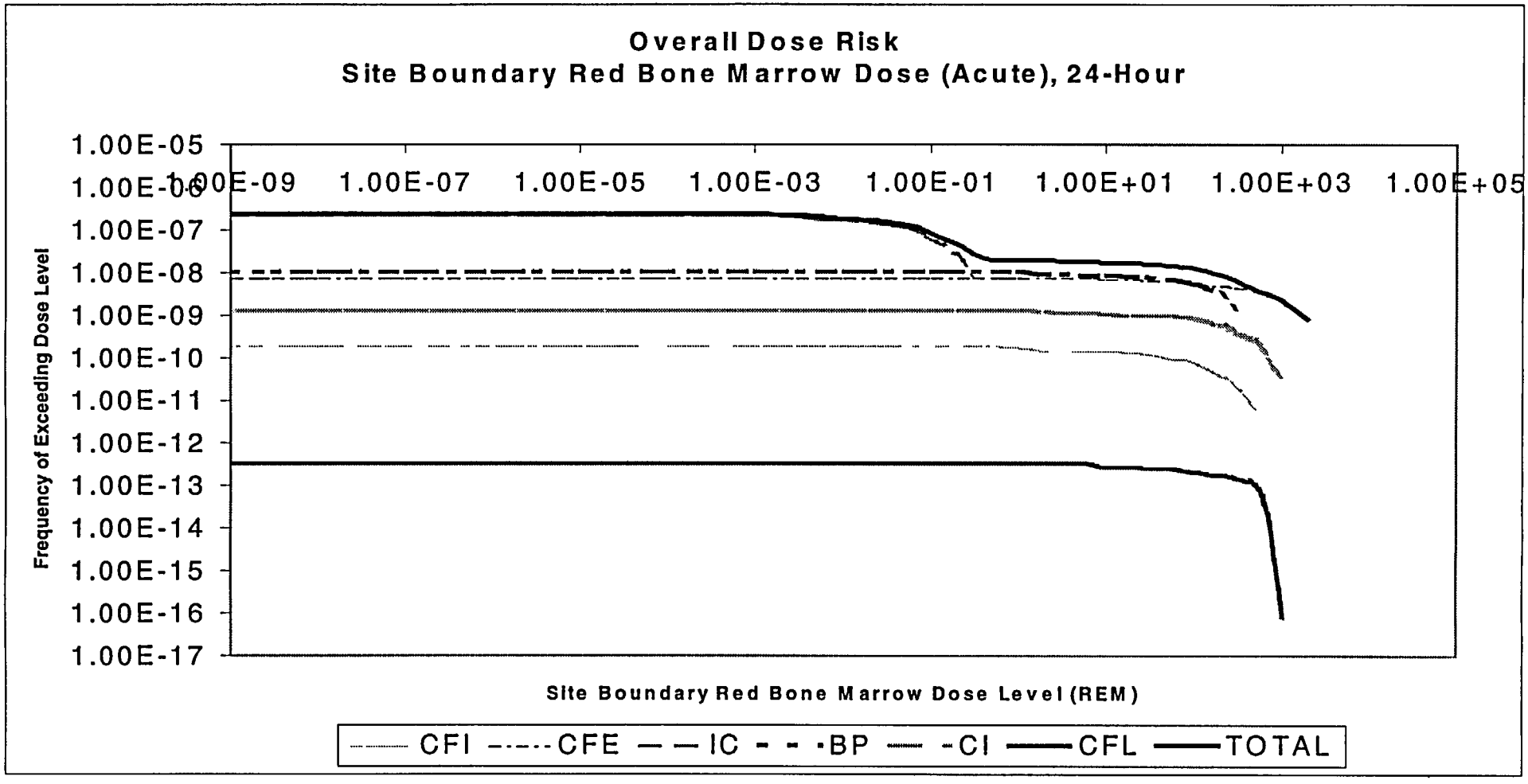


Figure 49-58

Overall Dose Risk Site Boundary Red Bone Marrow Dose (Acute), 24 Hours

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.058

Question:

Westinghouse claims that the concrete penetration on the reactor coolant drain tank (RCDT) (sump) side of the cavity is minimal following a hinged failure mode of the reactor vessel (RV), compared to the penetration on the RV side of the cavity. However, this is predicated on the core debris separating, with the oxide component (about 85 - 90 percent oxide) remaining on the reactor vessel side of the cavity, and a metallic component (about 75 to 85 percent metal) reaching the RCDT side of the cavity. This debris separation behavior is used by Westinghouse as the basis for concluding that core debris accumulation in the cavity sump would not be controlling for basemat melt-through. It is unclear whether this separation will actually occur given the large uncertainties in the configuration of molten core debris prior to vessel breach (i.e., mixed versus stratified), and the turbulence and mixing that would occur as the debris enters and spreads within the reactor cavity. Please confirm the robustness of your conclusion and the adequacy of the sump curb design by providing an assessment of the impact on basemat melt-through times and containment pressure (for both limestone and basaltic concretes) assuming that this oxide/metallic separation does not occur following a hinged failure of the reactor vessel, i.e., either a homogeneous melt or an oxide melt reaches the RCDT side of the reactor cavity and enters the sump.

Westinghouse Response:

The MELTSPREAD analyses that were performed for the two vessel failure scenarios determined the partitioning of the core debris between the reactor cavity and the reactor coolant drain tank room. The bounds of debris quenching and spreading during relocation are included in the investigation.

The masses of the debris released to the cavity from the hinged vessel failure are presented in Table B-1 of the AP1000 PRA report. The masses are summarized here in Table 1 and presented as the components uranium dioxide, zirconium, zirconium dioxide, and stainless steel.

Table 1 – Debris Volume in the Cavity

Debris Component	Mass (kg)	Volume (m^3)
UO_2	96,500	11.0
Zr	14,755	2.4
ZrO_2	10,726	1.8
SS	51,000	7.3
Total Volume		22.5

The cavity floor surface area is 48.3 m^2. If the debris is uniformly spread over the surface of the cavity floor, the debris depth is 47 cm or 18.3 inches. The height of the curb currently is 18 inches. The height of the curb will be increased to 24 inches (see PRA revision below) to prevent debris from entering the sump even for this non-mechanistic case.

Design Control Document (DCD) Revision:

None

PRA Revision:

B.4 Core Concrete Interactions

If the reactor vessel fails when the RCS is at a low pressure, the molten core debris will pour from the reactor vessel onto the reactor cavity floor. If a steam explosion does not occur, the pour will spread over the cavity floor and begin to transfer heat to the concrete floor of the reactor cavity. Due to the predicted mode of reactor vessel failure and the shape of the AP600 reactor cavity, analyses of the possible spreading of the core debris over the cavity floor were conducted using the MELTSPREAD code (Reference B-4). The AP1000 cavity geometry is the same as AP600. In addition, the AP1000 initial debris location from the vessel to the cavity is similar to AP600 in terms of mass flowrate and superheat, and therefore, the MELTSPREAD analyses performed for AP600 can be extended to AP1000. The results of the MELTSPREAD analyses were used as input to the MAAP4 code for analysis of core concrete interactions for AP1000.

The AP1000 reactor cavity is at containment elevation 71′ 6″ and consists of two interconnected volumes. The volume, which includes the reactor vessel, is octagonal in shape. The other volume is rectangular in shape and houses the reactor coolant drain tank (RCDT) and also contains the reactor cavity sump. The two volumes are connected by a 5-foot wide tunnel whose floor is also at elevation 71′ 6″ and a 3-foot wide ventilation duct whose bottom is 4 inches above the cavity floor. The cavity sump is situated between the tunnel and the ventilation duct at the side of the reactor coolant drain tank room closest to the reactor vessel. There is a 3-foot thick wall that separates the reactor cavity drain tank region from the reactor vessel region of the cavity. The floor of the cavity sump is at elevation 69′ 6″ and is completely encompassed by a curb whose top is at elevation **73′ 6″ (24-inch high curb)** ~~73′ 0″ (18-inch high curb)~~. The tunnel between the reactor vessel and reactor coolant drain tank portions of the cavity is protected by a door and shielding material to minimize radiation exposure to persons working in the reactor coolant drain tank area of the cavity. The door and shielding are not important to the analyses of core debris spreading in the reactor cavity due to the dynamic forces of the fuel coolant interactions that will occur at reactor vessel failure. Since the door and shielding are not designed to withstand "blast loading," they are expected to be destroyed prior to the arrival of core debris at their pre-vessel failure location. As added assurance that the door and shielding will not remain in their pre-vessel failure location, the high temperature of the core debris will quickly ablate and/or physically move any door and/or shielding components that might remain in place after the fuel coolant interaction loading. A schematic layout of the cavity region is provided in Figure B-3.

RAI Number: 720.060

Question:

In response to 10 CFR 50.34(f)(1)(i), Westinghouse provided an evaluation of potential AP600 design improvements (Severe Accident Mitigation Design Alternatives) in Appendix 1B of the AP600 standard safety analysis report (SSAR)/DCD Tier 2. The details of this evaluation, which included a design description and estimated risk reduction and costs for each alternative, and estimated offsite exposure for each of the major release categories, formed the basis for the staff's review. A similar evaluation has not been provided in Appendix 1B of the DCD or in the PRA for AP1000. In order to support the staff's review of potential design improvements, please provide an AP1000-specific evaluation of SAMDAs similar in scope and content to that provided for AP600. Please include the following within the response:

A. a summary of the risk-significant enhancements (i.e., impacting core damage frequency [CDF] and person-rem doses) incorporated subsequent to the AP600 design, such as the third PCS water injection line and the IRWST vent dampers,

B. the risk (CDF and population dose per year) associated with operation of an AP1000 at the reference site. Include the risk associated with internally and externally-initiated events, and events at shutdown to the degree that this can be inferred from the associated analyses,

C. the specific site characteristics, including population, meteorology, economic data, and evacuation assumptions on which the population dose estimates are based (provide these characteristics/interface assumptions in such a way that one can readily determine whether a potential new reactor site is enveloped by the same analysis),

D. the estimated dollar value of completely eliminating all severe accident risk for an AP1000 plant at the reference site, broken down by major cost category (i.e., public exposure, offsite property damage, occupational exposure, onsite cleanup and decontamination, and replacement power)

E. an explanation of how insights from the AP1000-specific PRA and supporting risk analyses for external and shutdown events, including importance analyses and cutset screening, were used to identify potential plant improvements, and

F. justification that the potential improvements identified through a systematic process as suggested in (e) are included within the set of 15 SAMDAs identified in Appendix 1B of the AP1000 DCD. Provide a supplemental analysis for those risk-significant improvements not included within the list of 15.

Westinghouse Response:

The responses to question A through F are given below in the AP1000 SAMDA evaluation:

- For response to question A see Section 5.2
- For response to question B see Section 4.2
- For response to question C see Section 6
- For response to question D see Section 7
- For response to question E see Section 5.1
- For response to question F see Section 8.

AP1000 SAMDA EVALUATION

1 Introduction

This response provides an evaluation of Severe Accident Mitigation Design Alternatives (SAMDA) for the Westinghouse AP1000 design. This evaluation is performed to evaluate whether or not the safety benefit of the SAMDA outweighs the costs of incorporating the SAMDA in the plant, and is conducted in accordance with applicable regulatory requirements as identified below.

The National Environmental Policy Act (NEPA), Section 102.(C)(iii) requires, in part, that

...all agencies of the Federal Government shall ... (C) include in every recommendation or report on proposals for legislation and other major Federal actions significantly affecting the quality of the human environment, a detailed statement by the responsible official on ... (iii) alternatives to the proposed action.

10 CFR 52.47(a)(ii) requires an applicant for design certification to demonstrate

> ... compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f) ...

A relevant requirement of 10 CFR 50.34(f) contained in subparagraph (1)(i) requires the performance of

> ... a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant ...

In SECY-91-229, the NRC staff recommends that severe accident mitigation design alternatives be addressed for certified designs in a single rulemaking process that would address both the 10 CFR 50.34 (f) and NEPA considerations in the 10 CFR Part 52 design certification rulemaking. SECY-91-229 further recommends that

applicants for design certification assess SAMDAs and the applicable decision rationale as to why they will or will not benefit the safety of their designs. The Commission approved the staff recommendations in a memorandum dated October 25, 1991 (Reference 1).

2 Summary

Note that AP1000 is very similar to AP600, which has received Design Certification. The evaluation for AP1000 uses the conclusions of the AP600 SAMDA investigation as described below. An evaluation of candidate modifications to the AP600 design was conducted to evaluate the potential for such modifications to provide significant and practical improvements in the radiological risk profile of the AP600 design. Since AP1000 is so similar to AP600, the list of candidate modifications is the same.

The process used for identifying and selecting candidate design alternatives included a review of SAMDAs evaluated for other plant designs. Several SAMDA designs evaluated previously for other plants were excluded from the present evaluation because they have already been incorporated or otherwise addressed in the AP600 and AP1000 designs. These include:

Hydrogen ignition system
Reactor cavity flooding system
Reactor coolant pump seal cooling
Reactor coolant system depressurization
Reactor vessel exterior cooling.

Additional design alternatives were identified based upon the results of the AP600 probabilistic risk assessment (Reference 3). The AP1000 probabilistic risk results are similar to those developed for AP600. Fifteen candidate design alternatives were selected for further evaluation.

An evaluation of these alternatives was performed using a bounding methodology such that the potential benefit of each alternative is conservatively maximized. As part of this process, it was assumed that each SAMDA performs beyond expectations and completely eliminates the severe accident sequences that the design alternative addresses. In addition, the capital cost estimates for each alternative were intentionally biased on the low side to maximize the risk reduction benefit. This approach maximizes the potential benefits associated with each alternative.

The results show, for AP600 and AP1000, that despite the significant conservatism employed in the evaluation, none of the SAMDAs evaluated provide risk reductions which are cost beneficial. The results also show that even a conceptual "ideal SAMDA", one which reduces the total plant radiological risk to zero, would not be cost effective. This is due primarily to the already low risk profile of the AP600 and AP1000 designs.

3 Selection of SAMDAS

Candidate design alternatives were selected based upon design alternatives evaluated for other plant designs (References 4, 5, and 6) as well as suggestions from AP600 and AP1000 design personnel. Additional candidate design alternatives were selected based upon an assessment of the AP600 and AP1000 probabilistic risk assessment results. Fifteen design alternatives were finally selected for further evaluation. These fifteen SAMDAs are:

- Chemical volume and control system (CVS) upgraded to mitigate small LOCAs
- Filtered containment vent
- Normal residual heat removal system (RNS) located inside containment
- Self-actuating containment isolation valves
- Passive containment spray
- Active high pressure safety injection system
- Steam generator shell side passive heat removal system
- Steam generator safety valve flow directed to in-containment refueling water storage tank (IRWST)
- Increase steam generator secondary side pressure capacity
- Secondary containment filtered ventilation
- Diverse IRWST injection valves
- Diverse containment recirculation valves
- Ex-vessel core catcher
- High pressure containment design
- Diverse actuation system (DAS) improved reliability.

A description of each design alternative evaluated for AP600 is presented in Appendix A to this document.

4 Methodology

The severe accident mitigation design alternatives analysis employs a bounding methodology such that the benefit is conservatively maximized and the capital cost is conservatively minimized for each SAMDA.

4.1 Total Population Dose

To assess the potential benefits associated with a design alternative, estimates are made of the total offsite population dose resulting from each of the release categories (i.e., source terms). MACCS2 version 1.12 (Reference 9) is used for the analysis. The NRC sponsored the development of this code. The code performs probabilistic estimates of offsite consequences from potential accidental releases in conformance with Chapter 9 of the probabilistic risk assessment (PRA) guidelines described in NUREG/CR-2300 (Reference 10).

Doses are determined for the early exposure effects resulting from the initial 24 hours following the core damage initiation. The dose evaluation provides the conditional probability distributions for the consequence measures, which includes the whole-body dose for this analysis. These consequence probability distributions are based on the assumption that the accident that produced the source term has occurred. Therefore, the consequence probability distributions presented result from the variation in dose levels due to the various meteorological conditions. Hence, the actual probability of the identified dose levels would be the probability of the release category that produced the source term occurring multiplied by the probability of the dose level.

The dose risks are quantified by multiplying the calculated fission product release category frequency vector by the release category mean dose vectors. The frequencies for each of the six release categories are quantified in Chapter 45 of the AP1000 PRA (Reference 2), while the mean doses for each release category are identified in revised Chapter 49 (see response to RAI 720.056). Table B-2 presents the results of the dose risk calculations at the site boundary for 2 hours of exposure. The table presents the release category identifier, the release frequency (per reactor-year), the mean dose (in rem), and the resulting risk (in rem per reactor-year). In addition, each table presents the total dose risk and the percent that each release category contributes to the total risk.

It is shown that release category CFE presents the largest risk to the site safety.

4.2 AP1000 RISK (CDF, LRF, and POPULATION Dose)

Summary of AP1000 Risk (CDF and LRF)

	CDF	LRF
Internal Events at Power	2.41E-07/yr	1.95E-08/yr
Events at Shutdown	1.23E-07/yr	2.05E-08/yr (2)
Internal Fire	5.61E-08/yr	4.54E-09/yr (2)
Internal Flooding	8.82E-10/yr	negligible
Seismic Events	not quantified (1)	not quantified (1)

Notes:
(1) = Seismic margins method is used. CDF and LRF not quantified.
(2) = LRF is not quantified, but is estimated by a ratio of CDF to LRF for corresponding cases: namely AP600 for shutdown, internal events for fire.

Level 3 analysis is performed only for internal events at power. The ensuing population dose was very low and it was not pursued for other events. The population dose for internal events is given below, in Table 4-1.

Table 4-1

POPULATION WHOLE BODY DOSE (EFFECTIVE DOSE EQUIVALENT, EDE), 0-80.5 KM PERSON-SIEVERTS

24-hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	
CFI	7.88E+03	6.11E+03	1.47E+04	2.01E+04	3.21E+04	3.51E+04	5.34E+04
CFE	8.51E+03	6.25E+03	1.62E+04	2.31E+04	4.13E+04	5.06E+04	6.40E+04
DIRECT	2.16E+01	1.20E+01	4.78E+01	8.13E+01	1.14E+02	1.23E+02	1.68E+02
IC	7.19E+00	4.21E+00	1.71E+01	2.95E+01	3.56E+01	3.84E+01	5.60E+01
BP	2.91E+03	1.74E+03	5.90E+03	1.00E+04	1.52E+04	1.81E+04	2.58E+04
CI	2.01E+04	1.13E+04	4.71E+04	6.60E+04	1.23E+05	1.48E+05	1.61E+05
CFL	5.32E+03	3.87E+03	1.04E+04	1.35E+04	2.32E+04	2.77E+04	4.35E+04
72-hour Case Source Term	**Quantiles**						**Peak Consequence**
	Mean	**50th**	**90th**	**95th**	**99th**	**99.5th**	
CFI	8.89E+03	6.89E+03	1.63E+04	2.21E+04	3.42E+04	3.84E+04	5.73E+04
CFE	9.36E+03	6.89E+03	1.88E+04	2.54E+04	4.25E+04	5.12E+04	6.77E+04
DIRECT	2.45E+01	1.43E+01	5.50E+01	8.33E+01	1.16E+02	1.26E+02	1.78E+02
IC	8.80E+00	5.57E+00	1.98E+01	3.14E+01	4.41E+01	5.03E+01	6.33E+01
BP	3.11E+03	1.85E+03	6.31E+03	1.03E+04	1.54E+04	1.82E+04	2.69E+04
CI	2.14E+04	1.25E+04	4.90E+04	7.40E+04	1.27E+05	1.53E+05	1.67E+05
CFL	5.84E+03	4.32E+03	1.12E+04	1.48E+04	2.53E+04	3.04E+04	4.62E+04

5. A Summary of Risk Significant Enhancements

5.1 Use of PRA Insights and Design Improvements

Three specific discussions of how Insights from the AP1000 PRA and supporting analyses are given in existing documents as follows:

1. Response to RAI 720.040 lists and discusses the PRA insights that impacted the design;

2. AP1000 RTNSS WCAP-15985 "AP1000 Implementation of the regulatory Treatment of Nonsafety-Related System Process" uses additional insights to

classify some non-safety systems to be considered for Technical Specifications and Administrative Controls (see response to RAI 720.039);

3. Reliability assurance program (RAP) (DCD Section 17.4) uses component importances and expert input to identify those components that are risk significant.

Documents for items 2 and 3 can be consulted for further details. Item 1 is summarized below.

5.2 AP1000 Design Enhancements Already Implemented

This section summarizes the design enhancements that are already incorporated into the AP1000 plant due to PRA insights and results.

1. Changed the normal position of the two Containment motor operated recirculation valves (in series with squib valves) from closed to open

The normal position of the two MOV lines in the two sump recirculation lines have been changed from NORMALLY CLOSED to NORMALLY OPEN to improve the reliability of opening these paths. These 2 paths support containment recirculation for core cooling and IRWST draining for IVR. This change reduced the CDF and LRF contribution from the failure modes to open the MOVs.

2. Changed IRWST drain procedure so it occurs earlier for IVR support

Credit is taken for operator action to drain the IRWST into the sump to preserve reactor vessel integrity following core melt. The procedure for this severe accident response has been modified so that the operator action associated with IRWST draining is moved to the beginning of the procedure to allow more time for operator success and also to fill the cavity as soon as possible. This improves the probability of success of the operator action.

3. Improved IVR heat transfer

In going from AP600 to AP1000, the heat loads during IVR are increased due to the larger core power level which reduced the margins in the heat removal capability through the reactor vessel head during IVR . To compensate for the increase in core power, the critical heat flux limit on the outside of the reactor vessel has been increased by changes made to the flow path between the outside of the reactor vessel and the reactor vessel insulation. Testing has confirmed the robustness of the IVR heat transfer.

4. Improved IRWST vents

The larger core in the AP1000 can generate more hydrogen in a severe accident. In the AP1000 hydrogen analysis for Level II, it was observed that the standing hydrogen diffusion flames at the IRWST vents resulted in a larger thermal loads to the containment steel shell, potentially leading to containment wall failure. The design of the vents were changed so that the IRWST vents located well away from the containment would open and the IRWST vents located next to the containment would not open during a severe to eliminate or minimize this potential concern.

5. Incorporated low boron core (ATWT)

In AP600, ATWS contribution to LRF was noticed to be high relative to other initiating events. A low boron core was incorporated into the design to reduce the potential contribution of ATWS to plant risk.

6. Added 3rd Passive Containment Cooling drain valve (MOV diverse to AOV)

Due to reduced containment surface area per MW of core power, natural air circulation without PCS water drain may not always be sufficient for long term (> 1 day) containment heat removal in AP1000. For AP600 it was always sufficient for an indefinite time. To reduce the uncertainty in whether air cooling is sufficient to provide adequate long-term containment heat removal, a third path was added to the PCS drain lines to increase PCS reliability. The isolation valve used in the third path is an MOV, which is diverse from the AOVs used in the other two lines. This provides considerable improvement in the PCS water drain reliability.

7. Reduced Potential Recirculation-Line Squib Valve Failures

An examination of AP1000 plant CDF cutsets revealed that the CCF of 4/4 recirculation line squib valves is a dominant contributor to CDF and LRF. This failure mode can be reduced by re-aligning the diverse squib valves already used in the AP1000 (and AP600) IRWST injection paths (high pressure valves) and the containment recirc paths (low pressure valves). By making the recirculation squib valves two sets of two LP and HP squib valves, which are different and belong to different CCF groups. This design change reduces the CCF failure contribution of the recirculation squib valves. The increase in the group size of the HP squib valves from 4 to 6 (including the four from the IRWST injection lines) does not add an appreciable contribution to the plant CDF.

6. The Specific Site Characteristics

AP1000 Chapter 49, "Offsite Dose Risk Quantification," is based on an EPRI report (Reference 11) to establish the specific site characteristics for AP1000. Reference 11 Annex B, "ALWR Reference Site," establishes a conservative reference site to represent the consequences of most potential sites with respect to exposure at the site boundary. This reference site was based on the characteristics of 91 U.S.

reactor sites that are tabulated in the NRC document, "Technical Guidance for Siting Criteria Development," (NUREG CR-2239) (Reference 12). Annex B provides a summary of the meteorological data to be used in calculating off-site dose. The following information has been obtained from Reference 12.

Base Case Used for Siting Criteria Development

- A standard 1120 MWe PWR

Chosen because many reactors operating in December 1982 and most under construction at that time were about this size.

- AN SST1 release

For the purpose of decision-making in such areas as siting and emergency response, NRC defined a set of five siting source terms (SST1-5) to represent the five accident groups. SST1 is as follows:

  - Severe core damage
  - Loss of all installed safety features
  - Severe direct breach of containment

Based on, at that time (December 1982) the number of PRAs available, NRC suggested that a representative probability for SST1 would be 1E-05. SST1 was then normalized to 100 and comparisons and sensitivities were made to the other four SSTs for conditional mean consequences of:

  - Mean early fatalities – Indian Point 710-1300*
    The maximum distance, to which early fatalities occurred for an SST1 release ranged from 13 to 25 miles, depending on meteorology, and 18 miles for New York meteorology. Improbable events with conditional probabilities of ≤ E-03 caused by adverse weather, e.g., rainout of the radioactive plume onto a population center.
  - Mean early injuries – Indian Point 2400-14000*
  - Mean latent cancer fatalities – Indian Point 7600-9300*
  - Mean thyroid modules
  - Mean interdicted land area – Indian Point 72-140 sq. miles*

  * All #s based on Complementary Cumulative Distribution Functions (CCDFs) generated with meteorological data from 29 National Weather Service Stations

- New York City meteorology

  Observed rainfall for New York is 697 hours and annual rain of 49 inches

- Indian Point wind rose and population
- Summary evacuation

Summary of Consequence Distances (Miles)

Source	Consequence	Conditional Probability Level		
		Mean	99%	Calc Max
SST1	Early Fatalities	<5	≤15	<25
	Early Injuries	~10	~30	≥50
	Land Interdiction	~20	>50	>50
	PAGs	≥50	>50	>50

Mean distances are the average of the probability distributions of distance.
99% distances refer to those beyond which a consequence or dose is calculated to occur in 1 in 100 accidents.
Calculated maxima represent the largest distances calculated
PAG is defined as the "projected" dose to an individual in the general public that warrants the initiation of emergency protective actions. PAGs range from 1 to 5 rems for whole body exposure and from 5 to 25 rems for projected dose to the thyroid.

Sensitivity of Fatal, Injury, and Interdiction Distances to Release Magnitude

Source (mi)	Fatal Distance (mi)			Injury Distance (mi)			Interdiction Distance		
SST1	Mean	99%	Peak	Mean	99%	Peak	Mean	99%	Peak
1	3.9	1.2	18	11	35	50	19	55	85
½	2.5	10	18	7.0	20	25	14	45	50
1/10	0.9	2.2	5.0	2.8	10	18	5.5	18	25
1/20	0.5	2.0	2.0	1.9	7.0	10	3.6	12	18
1/100	0	1.0	1.0	0.9	4.0	5.0	1.1	10	10

Peak result is that obtained for the most favorable weather conditions sampled.

7. Value of Eliminating Risk

The dollar value of completely eliminating all severe accident risk for an AP1000 plant at the reference site is calculated below for a base case, and various sensitivity analyses.

The following cost categories are considered:

1. Public Exposure	$2000 per man-rem
2. Loss of Plant	$2.0E+09
3. Offsite Property Damage/Cleanup	$2.0E+09
4. Onsite Cleanup and Decontamination	$1.0E+09
5. Replacement Power	not considered since the plant is written off

NUREG/CR-3568 (A Handbook for Value-Impact Assessment, 1983) is consulted for setting up the base case.

Following additional input are used for the estimate:

Delta CDF	2.41E-07/yr
Delta LRF	1.95E+08/yr
Average Population Whole Body Dose	6.4E+05 man-rem
Plant Life	40 years

Inflation rate is taken as equal to opportunity cost of money. Note that this is conservative in most cases since generally, the opportunity cost of money is larger than inflation, which makes the current value of a series of future expenditures less than a simple sum of all these expeditures.

The calculations for the base case and three more conservative cases are shown in Table 7-1.

The following equations are used for calculating dollar value of eliminating risk (with the assumption that inflation rate is equal to opportunity cost of money):

$Q = Q1 + Q2 + Q3 + Q4$

$Q1 = q1 * f2 * r * t$

$Q2 = q2 * f1 * t$

$Q3 = q3 * f2 * t$

$Q4 = q4 * f1 * t$

The symbols are defined in Table 7-1.

From Table 7-1, it is seen that, even with generously conservative assumptions, the value of eliminating AP1000 risk totally is very small. Our best estimate ranges from a best estimate value of $30,000 to a more conservative estimate of $405,000 of initial investment money.

Table 7-1. Value of Eliminating Risk					
					Not used
		Base Case	Case 2	Case 3	Case 4
f1	Delta CDF	2.41E-07	2.41E-06	2.41E-07	2.41E-06
f2	Delta LRF	1.95E-08	1.95E-07	1.95E-08	1.95E-07
r	Man-REM Exposure	6.40E+05	6.40E+06	6.40E+05	6.40E+06
t	Plant Life	4.00E+01	4.00E+01	4.00E+01	6.00E+01
q1	Cost of Exposure	$2,000	$2,000	$4,000	$4,000
q2	Cost of Plant	$2,000,000,000	$2,000,000,000	$4,000,000,000	$4,000,000,000
q3	Offsite Damage	$2,000,000,000	$2,000,000,000	$4,000,000,000	$4,000,000,000
q4	Onsite Cleanup	$1,000,000,000	$1,000,000,000	$2,000,000,000	$2,000,000,000
Q1	Value of Exposure =	$1,154	$115,440	$2,309	$346,320
Q2	Value of Plant =	$19,280	$192,800	$38,560	$578,400
Q3	Value of Offsite damage =	$1,560	$15,600	$3,120	$46,800
Q4	Value of Onsite cleanup =	$9,640	$96,400	$19,280	$289,200
Q	Total Value of Eliminating Risk =	**$31,478**	$404,640	$62,957	$1,213,920

8. Evaluation of Potential Improvements

We have already estimated value of eliminating AP1000 total risk as $30,000, as discussed in Section 7 above. This value is an upper bound for any single engineered design alternative, which would actually reduce CDF and/or LRF a fraction of the values assumed in the base case for calculating the $30,000 value. Moreover, only two percent of the $30,000 comes from reduction of man-rem exposure. Thus, any design alternative that does not reduce CDF considerably, even if it does reduce the man-rem exposure, would not be cost beneficial.

For AP600, 14 design alternatives were discussed in the SAMDA section (Appendix 1B) and were found to be not cost effective. One of these alternatives is actually implemented in the AP1000 design (diverse containment recirculation squib valves). The costs associated with the remaining 13 design alternatives are provided in Table 8.1. Note that only one design alternative, # 3, namely self-actuating containment isolation valves has a cost near $30,000; the remaining alternatives are at least an order of magnitude more costly than $30,000. Thus, only design alternative #3 needs to be further discussed.

Table 8-1 Design Alternatives for SAMDA

#	Design Alternative	Cost
1	Upgrade CVS for Small LOCA	1,500,000
2	Containment Filtered Vent	5,000,000
3	Self-Actuating Containment Isolation Valves	33,000
4	Safety Grade Passive Containment Spray	3,900,000
6	SG Shell Side Heat Removal	1,300,000
7	SG Relief Flow to IRWST	620,000
8	Increased SG Pressure Capability	8,200,000
9	Secondary Containment Ventilation with Filtration	2,200,000
10	Diverse IRWST Injection Valves	570,000
11	Diverse Containment Recirculation Valves	Already Implemented
12	Ex-Vessel Core Catcher	1,660,000
13	High Pressure Containment Design	50,000,000
14	More Reliable DAS	470,000

Self-Actuating Containment Isolation Valves

This SAMDA consists of improved containment isolation provisions on all normally open containment penetrations. The category of "normally open" is limited to normally open pathways to the environment during power and shutdown conditions, excluding closed systems inside and outside the containment such as RNS and component cooling. The design alternative would be to add a self-actuating valve or

enhance the existing inside containment isolation valve to provide for self actuation in the event that containment conditions are indicative of a severe accident. Conceptually, the design would either be an independent valve or an appendage to an existing fail-closed valve that would respond to post-accident containment conditions within containment. For example, a fusible link would melt in response to elevated ambient temperatures resulting in venting the air operator of a fail-closed valve, thus providing the self-actuating function. To evaluate the benefit of this SAMDA, this design change is assumed to eliminate the CI release category. This does not include induced containment failures which occur at the time of the accident such as in cases of vessel rupture or anticipated transients without scram (ATWS). This design alternative provides almost no benefit in reducing plant CDF.

Generously assuming that this design alternative will eliminate CI release totally, the delta LRF is 1.33E-09/yr (see Table 8-2). Delta CDF is zero. The benefit of this design alternative is calculated as $320, with a conservative estimate range of up to $22,500. Based on these calculations, even the cheapest design alternative does not meet the benefit/cost ratio of 1.

Other New design Changes:

Other design changes, as discussed in Section 5, are already incorporated into AP1000. There is no cost/benefit analysis available for those changes already incorporated.

Table 8-2

POPULATION WHOLE BODY EDE DOSE RISK – 24 HOURS

Release Category	Release Frequency (/reactor year)	Mean Dose (person-sieverts)	Dose (person-REM)	Risk (person-REM/ reactor year)	Percent Contribution to Total Risk
CFI	1.89E-10	7.88E+03	7.88E+05	1.49E-04	1.2%
CFE	7.47E-09	8.51E+03	8.51E+05	6.36E-03	51.3%
IC	2.21E-07	7.19E+00	7.19E+02	1.59E-04	1.3%
BP	1.05E-08	2.91E+03	2.91E+05	3.06E-03	24.7%
CI	1.33E-09	2.01E+04	2.01E+06	2.67E-03	21.6%
CFL	3.45E-13	5.32E+03	5.32E+05	1.84E-07	0.0%
			Total Risk =	1.24E-02	100.0%

Table 8-3. Value of Eliminating Risk for Alternative 3				
				Not used
	Base Case	Case 2	Case 3	Case 4
Delta CDF	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Delta LRF	1.33E-09	1.33E-08	1.33E-09	1.33E-08
Man-REM Exposure	2.01E+06	2.01E+07	2.01E+06	2.01E+07
Plant Life	4.00E+01	4.00E+01	4.00E+01	6.00E+01
Cost of Exposure	$2,000	$2,000	$4,000	$4,000
Cost of Plant	$2,000,000,000	$2,000,000,000	$4,000,000,000	$4,000,000,000
Offsite Damage	$2,000,000,000	$2,000,000,000	$4,000,000,000	$4,000,000,000
Onsite Cleanup	$1,000,000,000	$1,000,000,000	$2,000,000,000	$2,000,000,000
Value of Exposure =	$225	$22,450	$449	$67,351
Value of Plant =	$0	$0	$0	$0
Value of Offsite damage =	$106	$1,064	$213	$3,192
Value of Onsite cleanup =	$0	$0	$0	$0
Total Value of Eliminating Risk =	$320	$22,450	$641	$67,351

9. Results

Due to existing low risk of the AP1000 plant, none of the design alternatives described in Appendix A meet an acceptable benefit to cost ratio of 1 or greater.

Several of the design alternatives evaluated in other SAMDA analyses are included in the current AP1000 design. These design features include:

RCS depressurization system
Passive residual heat removal system located inside containment
Cavity flooding system
Passive containment cooling system
Hydrogen igniters in a large-dry containment
Diverse actuation system

Canned motor RCPs
Interfacing system with high design pressure

As the AP1000 plant core damage frequency is lower than for existing plants, the benefits of additional design alternatives are very small. The fifteen SAMDAs analyzed provided little or no benefit to the AP1000 design.

Assuming a hypothetical design alternative was developed which provides a 100 percent reduction in overall plant risk, representing an averted risk of 1.24×10^{-2} man-rem per year, the capital benefit only amounts to $31,500.

10. References

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3. "AP600 Probabilistic Risk Assessment," Westinghouse Electric Corporation and ENEL, Revision 8, September 1996.

4. "Supplement to the Final Environmental Statement - Limerick Generating Station, Units 1 and 2," Docket Nos. 50-352/353, August 1989.

5. "Supplement to the Final Environmental Statement - Comanche Peak Steam Electric Station, Units 1 and 2," Docket Nos. 50-445/446, October 1989.

6. "System 80+ Design Alternatives Report," Docket No. 52-002, April 1992.

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8. Nuclear Energy Cost Data Base, DOE/NE-0095, U.S. Department of Energy, September 1988.

9. Chanin, D., Young, M. L., "Code Manual for MACCS2, User's Guide," NUREG/CR-6613, SAND97-0594, Vol. 1, Sandia National Laboratories, U.S. Nuclear Regulatory Commission.

10. "PRA Procedures Guide," NUREG/CR-2300, U.S. Nuclear Regulatory Commission, Vol. 2, Chapter 9, Washington, D.C.

11. EPRI Advanced Light Water Reactor Utility Requirements Document Volume III Annex B "ALWR Reference Site," Revisions 5 & 6, December 1993.

12. NRC NUREG/CR-2239 "Technical Guidance for Siting Criteria Development," prepared by Sandia National Laboratories, D.C. Aldrich, et al, December 1982.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

APPENDIX A

Design Alternatives

This subsection describes each SAMDA and the benefit expected due to the modification. In the evaluation of the risk reduction benefit, each SAMDA is assumed to operate perfectly with 100 percent efficiency, without failure of supporting systems. A perfect SAMDA reduces the frequency of accident sequences which it addresses to zero. This is conservative as it maximizes the benefit of each design alternative. The SAMDA will reduce the risk by lowering the frequency, attenuating the release, or both. The benefit will be described in terms of the accident sequences and dose which are affected by the SAMDAs, as well as the overall risk reduction. Note that for the purposes of these evaluations, increases to release category IC are not factored into the risk benefit calculations. The IC dose is sufficiently small that changes to the IC total frequency do not result in an appreciable change to overall results. This is also a conservative representation since this maximizes the risk reduction.

Since AP1000 alternatives are the same for AP1000 as for AP600, specific AP1000 risk reduction factor calculations were not performed for AP1000. To recognize the effect of the differences in release frequencies between AP600 and AP1000, they were compared. The largest difference in release category frequency between AP600 and AP1000 is for CFI, which is 14.5 times larger in AP1000 than for AP600. For conservatism, each of the AP600 SAMDA risk reduction factors was multiplied by 15 and applied to AP1000.

1 Upgrade the CVS for Small LOCAs

The chemical, volume, and control system (CVS) is currently capable of maintaining the reactor coolant system (RCS) inventory to a level in which the core remains covered in the event of a very small (≤ 3/8" diameter break) loss of coolant accident (LOCAs). This SAMDA involves providing in-containment refueling water storage tank (IRWST) / containment recirculation connections to the CVS and adding a second line from the CVS makeup pumps to the RCS in order to be able to use the system to keep the core covered during small and intermediate LOCAs.

A perfect, upgraded CVS system is assumed to prevent core damage in the RCS leak, passive RHR heat exchanger tube ruptures, small LOCA, and intermediate LOCA release categories. The CVS is assumed to have perfect support systems (power supply, component cooling) and to work in all situations regardless of the common cause failures of other systems.

2 **Filtered Vent**

This SAMDA consists of placing a filtered containment vent and all associated piping and penetrations into the AP1000 containment design. The filtered vent could be used to vent the containment to prevent catastrophic overpressure failure, and also provides filtering capability for source term release. With respect to the AP1000 PRA, the possible scenario in which the filtered vent could result in risk reduction would be late containment overpressure failures (release category CFL). Other containment overpressure failures occur due to dynamic severe accident phenomena, such as hydrogen burn, steam explosion, etc. The late containment failures for AP1000 are failures of the passive containment cooling system (PCS). Analyses have indicated that for scenarios with PCS failure, air cooling may limit the containment pressure to less than the ultimate pressure. However, for the purposes of the Level 2 PRA, failure of PCS is assumed to result in containment failure based on an adiabatic heatup. To conservatively consider the risk reduction of a filtered vent, the use of a filtered vent to preclude a late containment failure will be evaluated. A decontamination factor (DF) of 1000 will conservatively be assumed for each PRA Level 1 accident classification, even though it is realized that the dose due to noble gases will not be impacted by the filtered vent since 100% of the noble gas fission products will still be released. Therefore, the risk reduction is equal to the decontamination factor assumed, since the PRA Level 1 accident classification frequencies do not change.

3 **Self-Actuating Containment Isolation Valves**

This SAMDA consists of improved containment isolation provisions on all normally open containment penetrations. The category of "normally open" is limited to normally open pathways to the environment during power and shutdown conditions, excluding closed systems inside and outside the containment such as RNS and component cooling. The design alternative would be to add a self-actuating valve or enhance the existing inside containment isolation valve to provide for self actuation in the event that containment conditions are indicative of a severe accident. Conceptually, the design would either be an independent valve or an appendage to an existing fail-closed valve that would respond to post-accident containment conditions within containment. For example, a fusible link would melt in response to elevated ambient temperatures resulting in venting the air operator of a fail-closed valve, thus providing the self-actuating function. To evaluate the benefit of this SAMDA, this design change is assumed to eliminate the CI release category. This does not include induced containment failures which occur at the time of the accident such as in cases of vessel rupture or anticipated transients without scram (ATWS).

4 **Passive Containment Sprays**

This SAMDA involves adding a passive safety-related spray system and all associated piping and support systems to the AP1000 containment. A passive containment spray system could result in risk benefits in the following ways:

¥ scrubbing of fission products, primarily for CI failures,

¥ assuming appropriate timing, containment spray could be used as an alternate means for flooding the reactor vessel (in-vessel retention) and for debris quenching should vessel failure occur,

¥ containment spray could also be used to control containment pressure for cases in which PCS has failed.

In order to envelop these potential risk benefits, the risk reduction evaluation will assume that containment sprays are perfectly effective for each of these benefits, with the exception of fission product scrubbing for containment bypass. Thus the risk reduction can be conservatively estimated by assuming all release categories except BP are eliminated.

5 **Active High Pressure Safety Injection System**

This SAMDA consists of adding a safety-related active high pressure safety injection (HPSI) pump and all associated piping and support systems to the AP1000 design. A perfect high pressure safety injection system is assumed to prevent core melt for all events but excessive LOCA and ATWS. Therefore, to estimate the risk reduction, only the contributions to each release category of Level 1 accident classes 3C (vessel rupture) and 3A (ATWS) need be considered. The averted risk is shown in Table 8-1. This SAMDA would completely change the design approach from a plant with passive safety systems to a plant with passive plus active safety-related systems and is not consistent with design objectives.

6 **Steam Generator Shell-Side Heat Removal System**

This SAMDA consists of providing a passive safety-related heat removal system to the secondary side of the steam generators. The system would provide closed loop cooling of the secondary using natural circulation and stored water cooling, thus preventing a loss of primary heat sink in the event of a loss of startup feedwater and passive RHR heat exchanger. A perfect secondary heat removal system would eliminate transients from each of the release categories. In order to evaluate the benefit of this SAMDA, the frequencies of all the transient sequences is subtracted from the overall frequency of each of the release categories and the risk is recalculated. The total risk averted is shown in Table 8-1.

7 **Direct Steam Generator Relief Flow to the IRWST**

This SAMDA consists of providing all the piping and valves required for redirecting the flow from the steam generator safety and relief valves to the in-containment refueling water storage tank (IRWST). An alternate, lower cost option of this SAMDA consists of redirecting only the first stage safety valve to the IRWST. This system would prevent or reduce fission product release from bypassing the containment in the event of a steam generator tube rupture (SGTR) event. In order to evaluate the benefit from this SAMDA (both options), this design change is assumed to eliminate the BP release category.

8 **Increased Steam Generator Pressure Capability**

This SAMDA consists of increasing the design pressure of the steam generator secondary side and safety valve set point to the degree that a steam generator tube rupture will not cause the secondary system safety valve to open. The design pressure would have to be increased sufficiently such that the combined heat capacity of the secondary system inventory and the PRHR system could reduce the RCS temperature below T_{sat} for the secondary design pressure. Although specific analysis would have to be performed, it is estimated that the design pressure would have to be increased several hundred psi. This design would also prevent the release of fission products which bypasses the containment via the SGTR.

9 **Secondary Containment Filtered Ventilation**

This SAMDA consists of providing the middle and lower annulus (below the 135′ 3″ elevation) of the secondary concrete containment with a passive annulus filter system to for filtration of elevated releases. The passive filter system is operated by drawing a partial vacuum on the middle annulus through charcoal and HEPA filters. The partial vacuum is drawn by means of an eductor with motive flow from compressed gas tanks. The secondary containment would then reduce particulate fission product release from any failed containment penetrations (containment isolation failure). In order to evaluate the benefit from such a system, this design change is assumed to eliminate the CI release category.

10 **Diverse IRWST Injection Valves**

This SAMDA consists of changing the in-containment refueling water storage tank (IRWST) injection valve designs so that two of the four lines use diverse valves. Each of the four lines is currently isolated by a squib valve in series with a check valve. In order to provide diversity, the valves in two of the lines will be provided by a different vendor. For the check valves, alternate vendors are available. However, it is questionable if check valves of different vendors would be sufficiently different to be considered diverse unless the type of check valve was changed from the current swing disk check to another type. The swing disk type is the preferred type for this application and other types are considered to be less reliable. Squib valves are specialized valve designs for which there are few vendors. A vendor may not be willing to design, qualify, and build a reasonable squib valve design for this AP1000 application considering that they would only supply two valves per plant. As a result, this SAMDA is not really practicable because of the uncertainty in availability of a second squib valve design/vendor and because of the uncertainty in the reliability of another check valve type. However, the cost estimate for this SAMDA assumes that a second squib valve vendor exists and that vendor only provides the two diverse IRWST squib valves. The cost impact does not include the additional first time engineering and qualification testing that will be incurred by the second vendor. Those costs are expected to be more than a million dollars.

This change will reduce the frequency of core melt by eliminating the common cause failure of the IRWST injection. To estimate the benefit from this SAMDA, all core damage sequences resulting from a failure of IRWST injection are assumed to be averted. Core damage sequences resulting from a failure of IRWST injection correspond to PRA Level 1 accident classification 3BE; thus, release category 3BE is eliminated.

11 **Diverse Containment Recirculation Valves**

This SAMDA consists of changing the containment recirculation valve designs so that two out of the four lines use diverse valves. Each of the four lines currently contains a squib valve; two of the lines contain check valves and the other two contain motor-operated valves. In order to provide diversity, the squib valves in two lines will be made diverse by supplying them from a different vendor. This change will reduce the frequency of core melt by eliminating the common cause failure of the containment recirculation. To estimate the benefit from this SAMDA, all core damage sequences resulting from a failure of containment recirculation are assumed to be averted. Core damage sequences resulting from failure of containment recirculation correspond to PRA Level 1 accident classification 3BL; thus, release category 3BL is eliminated.

12 Ex-Vessel Core Catcher

This SAMDA consists of designing a structure in the containment cavity or using a special concrete or coating which will inhibit core-concrete interaction (CCI), even if the debris bed dries out. A perfect core catcher would prevent CCI for all cases. However, the AP1000 incorporates a wet cavity design in which ex-vessel cooling is used to maintain the core debris in the vessel thus preventing ex-vessel phenomena, such as CCI. Consequently, containment failure due to CCI is not considered in detail for the AP1000 Level 2 PRA. For cases in which reactor vessel flooding is failed, it is assumed that containment failure occurs due to ex-vessel steam explosion or CCI. This containment failure is assumed to be an early containment failure, CFE, (due to ex-vessel steam explosion) even though CCI and basemat meltthrough would be a late containment failure. To conservatively estimate the risk reduction of an ex-vessel core catcher, this design change is assumed to eliminate the CFE release category.

13 High Pressure Containment Design

This SAMDA design consists of using the massive high pressure containment design in which the design pressure of the containment is approximately 300 psi (20 bar) for the AP1000 containment. The massive containment design has a passive containment cooling feature much like the AP1000 containment. The high design pressure is considered only for prevention of containment failures due to severe accident phenomena such as steam explosions and hydrogen detonation. A perfect high pressure containment design would reduce the probability of containment failures, but would have no reduction of the frequency or magnitude of the release from an unisolated containment (containment isolation failure or containment bypass). To estimate the risk reduction of a high pressure containment design, this design is assumed to eliminate the CFE, CFI and CFL release categories.

14 Increase Reliability of Diverse Actuation System

This SAMDA design consists of improving the reliability of the diverse actuation system (DAS) which actuates engineered safety features and allows the operator to monitor the plant status. The design change would add a third instrumentation and control cabinet and a third set of diverse actuation system instruments to allow the use of 2-out-of-3 logic instead of 2-out-of-2 logic. Other changes, such as adding another set of batteries, have not been included in the cost estimates. A perfectly reliable DAS system would reduce the frequency of the release categories by the cumulative frequencies of all sequences in which DAS failure leads to core damage. In order to evaluate the benefit from the DAS system upgrade, a Level 1 sensitivity analysis assuming perfect reliability of DAS was completed.

15 Locate Normal Residual Heat Removal Inside Containment

This SAMDA consists of placing the entire normal residual heat removal system (RNS) and piping inside the containment pressure boundary. Locating the RNS inside the containment would prevent containment bypass due to interfacing system LOCAs (ISLOCA) of the residual heat removal (RHR) system. In past probabilistic risk assessments of current generation nuclear power plants, the ISLOCA is the leading contributor of plant risk because of large offsite consequences. A failure of the valves which isolate the low pressure RHR system from the high pressure RCS causes the RHR system to overpressurize and fail, releasing RCS coolant outside the containment where it cannot be recovered for recirculation cooling of the core. The result is core damage and the direct release of fission products outside the containment.

In the AP1000, the RNS is designed with a higher design pressure than the systems in current pressurized water reactors, and an additional isolation valve is provided in the design. In the probabilistic risk assessment, no ISLOCAs contribute significantly to the core damage frequency of the AP1000 (Reference 2, Chapter 33). Therefore, relocating the RNS of the AP1000 inside containment will provide virtually no risk reduction benefit and will not be investigated further in terms of cost.

APPENDIX B

The release categories for the AP1000 are defined as follows:

IC - intact containment. Containment integrity is maintained throughout the accident, and the release of radiation to the environment is due to nominal leakage.

CFE - containment failure early. Fission-product release through a containment failure caused by severe accident phenomenon occurring after the onset of core damage but prior to core relocation.

CFI - containment failure intermediate. Fission-product release through a containment failure caused by severe accident phenomenon occurring after core relocation but before 24 hours.

CFL - containment failure late. Fission-product release through a containment failure caused by severe accident phenomenon occurring after 24 hours.

CI - containment isolation failure. Fission-product release through a failure of the system or valves that close the penetrations between the containment and the environment. Containment failure occurs prior to onset of core damage.

BP - containment bypass. Fission products are released directly from the Reactor Coolant System to the environment via the secondary system or other interfacing system bypass. Containment failure occurs prior to onset of core damage.

The following subsections present a brief description of the AP1000 release categories.

The release category frequencies are shown in Table B-1.

1 Release Category IC - Intact Containment

If the containment integrity is maintained throughout the accident, then the release of radiation from the containment is due to nominal leakage and is expected to be within the design basis of the containment. This is the "no failure" containment failure mode and is termed intact containment. The main location for fission-product leakage from the containment is penetration leakage into the auxiliary building where significant deposition of aerosol fission products may occur.

2 **Release Category CFE - Early Containment Failure**

Early containment failure is defined as failure that occurs in the time frame between the onset of core damage and the end of core relocation. During the core melt and relocation process, several dynamic phenomena can be postulated to result in rapid pressurization of the containment to the point of failure. The combustion of hydrogen generated in-vessel, steam explosions, and reactor vessel failure from high pressure are major phenomena postulated to have the potential to fail the containment. If the containment fails during or soon after the time when the fuel is overheating and starting to melt, the potential for attenuation of the fission-product release diminishes because of short fission-product residence time in the containment. The fission products released to the containment prior to the containment failure are discharged at high pressure to the environment as the containment blows down. Subsequent release of fission products can then pass directly to the environment. Containment failures postulated within the time of core relocation are binned into release category CFE.

3 **Release Category CFI - Intermediate Containment Failure**

Intermediate containment failure is defined as failure that occurs in the time frame between the end of core relocation and 24 hours after core damage. After the end of the in-vessel fission-product release, the airborne aerosol fission products in the containment have several hours for deposition to attenuate the source term. The global combustion of hydrogen generated in-vessel from a random ignition prior to 24 hours can be postulated to fail the containment. The fission products in the containment atmosphere are discharged at high pressure to the environment as the containment blows down. Containment failures postulated within 24 hours of the onset of core damage are binned into release category CFI.

4 **Release Category CFL - Late Containment Failure**

Late containment failure is defined as containment failure postulated to occur later than 24 hours after the onset of core damage. Since the PRA assumes the dynamic phenomena, such as hydrogen combustion, to occur before 24 hours, this failure mode occurs only from the loss of containment heat removal via failure of the passive containment cooling system. The fission products that are airborne at the time of containment failure will be discharged at high pressure to the environment, as the containment blows down. Subsequent release of fission products can then pass directly to the environment. Accident sequences with failure of containment heat removal are binned in release category CFL.

5 **Release Category CI - Containment Isolation Failure**

A containment isolation failure occurs because of the postulated failure of the system or valves that close the penetrations between the containment and the environment. Containment isolation failure occurs before the onset of core damage. For such a failure, fission-product releases from the reactor coolant system can leak directly from the containment to the environment with diminished potential for attenuation. Most isolation failures occur at a penetration that connects the containment with the auxiliary building. The auxiliary building may provide additional attenuation of aerosol fission-product releases. However, this decontamination is not credited in the containment isolation failure cases. Accident sequences in which the containment does not isolate prior to core damage are binned into release category CI.

6. **Release Category BP - Containment Bypass**

Accident sequences in which fission products are released directly from the reactor coolant system to the environment via the secondary system or other interfacing system bypass the containment. The containment failure occurs before the onset of core damage and is a result of the initiating event or adverse conditions occurring at core uncovery. The fission-product release to the environment begins approximately at the onset of fuel damage, and there is no attenuation of the magnitude of the source term from natural deposition processes beyond that which occurs in the reactor coolant system, in the secondary system, or in the interfacing system. Accident sequences that bypass the containment are binned into release category BP.

TABLE B 1 POPULATION WHOLE BODY EDE DOSE RISK – 24 HOURS					
Release Category	Release Frequency (/reactor year)	Mean Dose (person-sieverts)	Dose (person-REM)	Risk (person-REM/ reactor year)	Percent Contribution to Total Risk
CFI	1.89E-10	7.88E+03	7.88E+05	1.49E-04	1.2%
CFE	7.47E-09	8.51E+03	8.51E+05	6.36E-03	51.3%
IC	2.21E-07	7.19E+00	7.19E+02	1.59E-04	1.3%
BP	1.05E-08	2.91E+03	2.91E+05	3.06E-03	24.7%
CI	1.33E-09	2.01E+04	2.01E+06	2.67E-03	21.6%
CFL	3.45E-13	5.32E+03	5.32E+05	1.84E-07	0.0%
			Total Risk =	1.24E-02	100.0%

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.063

Question:

Westinghouse added a check valve to the AP1000 refueling canal drain to permit the reactor cavity to flood more rapidly (AP1000 PRA page 39-2). This valve should be added to the system description in the DCD (e.g., Chapter 5.3.5.4) and included within the ITAAC related to the PXS (Chapter 2.2, Table 2.2.3-4).

Westinghouse Response:

The PRA is referring to check valves that have been incorporated in the drain line from the refueling cavity to the steam generator compartment. This drain line and associated valves are part of the spent fuel pool cooling system and are included in the Tier 1 material for this system (Tier 1 – Section 2.3.7). DCD Section 9.1.3 will be revised to incorporate this design feature. DCD Section 5.3.5.3 Description of External Vessel Cooling Flooded Compartments is revised to reference the appropriate sections of 6.3.2.1.3 and 9.1.3.3.8 as they relate to their role in post-accident flooding of the reactor cavity. DCD Section 5.3.5.4 Determination of Forces on Insulation and Support System is not changed. DCD 9.1.3.3.8 is added as shown.

Design Control Document (DCD) Revision:

5.3.5.3 Description of External Vessel Cooling Flooded Compartments

Ex-vessel cooling during a severe accident is provided by flooding the reactor coolant system loop compartment including the vertical access tunnel, the reactor coolant drain tank room, and the reactor cavity. Water from these compartments replenishes the water that comes in contact with the reactor vessel and is boiled and vented to containment. The opening between the vertical access tunnel and the reactor coolant drain tank room is approximately 100 ft^2. Removable steel grating is provided over the inlet to the vertical access tunnel to restrict access to the lower compartments. This grating precludes large debris from being transported into the reactor cavity during ex-vessel cooling scenarios. Figure 5.3-8 depicts the flooded compartments that provide the water for ex-vessel cooling. The doorway between the reactor cavity compartment and the reactor coolant drain tank room consists of a normally closed door and a damper above the door. The door and damper arrangement, shown in Figure 5.3-9, maintains the proper air flow through the reactor cavity during normal operation. The damper prevents air from flowing into the reactor coolant drain tank compartment, but opens to permit flooding of the reactor cavity from the reactor coolant drain tank compartment. The damper opening has a minimum flow area of 8 ft^2 and is not susceptible to clogging from debris that can pass through the grating over the inlet to the vertical access tunnel. It is constructed of light-weight material to minimize the force necessary to open the damper and permit flooding and continued water flow through the opening during ex-vessel cooling. The damper provides an acceptable pressure drop through the opening during ex-vessel cooling.

DCD Section 6.3.2.1.3 discusses post-accident operation of the passive core cooling system that operates to flood the reactor cavity following an accident. DCD Section 9.1.3 provides a discussion of the connections to the refueling cavity to the steam generator compartment that are provided to facilitate flooding of the reactor cavity following an accident.

9.1.3.3.8 Reactor Cavity Connections

The spent fuel pool cooling system contains connections to the refueling cavity to prevent excessive holdup of water in the reactor cavity following an accident. The piping connection facilitates draining of the reactor cavity to the steam generator compartment following a postulated accident. The line connects at the bottom of the reactor cavity and discharges to a steam generator compartment and contains a manual locked-open isolation valve and two check valves in series. The isolation valve is closed during refueling operations to facilitate flooding of the reactor cavity for refueling operations. Other connections are provided to the refueling cavity to facilitate proper draining, filling and purification of the reactor cavity to support refueling operations.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.078

Question:

In SECY-93-087, dated April 2, 1993, and the associated Commission's staff requirements memorandum (SRM), dated July 21, 1993, sets forth the policy regarding the criteria that the staff will use to evaluate an advanced light-water reactor (ALWR) vendor's review of the various severe accident scenarios that were analyzed and that identified the equipment needed to perform during a severe accident and the environmental conditions under which the equipment must function. The staff concluded in NUREG-1512, "Final Safety Evaluation Report [FSER] Related to Certification of the AP600 Standard Design," that applicable environments described in Section 19.2.3.3.7 of the AP600 FSER met the above guidance. Table D.7-2 and Figures D.7-3 thru D.7-69 of the AP600 PRA were an integral part of defining that pressure, temperature, and event timing and were the basis for defining the applicable environment described in Section 19.2.3.3.7 of the AP600 FSER. These figures do not appear to have been provided for the AP1000. Please provide similar figures for the AP1000 and/or additional information to support the position that the applicable environments meet the above-mentioned guidance for AP1000.

Westinghouse Response:

AP1000-specific MAAP4 cases defining Thermal Hydraulic Environments for equipment survivability will be added to PRA Appendix D as shown below.

Design Control Document (DCD) Revision:

None

PRA Revision:

AP1000 PRA Appendix D will be revised as shown in the following markup.

D.7.2 Thermal-Hydraulic Environments

~~D.7.2.1 Controlled Hydrogen Combustion~~

~~The bounding severe accident environments for each of the equipment survivability time frames defined in Section D.4 are provided in this section. These bounding environments for the reactor coolant system and containment are used in the assessments described in Section D.8.~~

~~The MAAP4 computer code (version 4.0.2) was used to support the quantification of the equipment survivability time frames and the bounding environment within each time frame for AP600, and this information is considered valid for AP1000. Two basic sequences and five sensitivity cases were quantified to establish the bounding environments including hydrogen combustion in the containment. Each sequence input data were adjusted to assure that a 100% fuel clad metal-water reaction occurred so that the required bounding hydrogen source was considered.~~
~~The two base sequences were a large (2.2 ft^2) hot leg break into a steam generator compartment and a 4-inch direct vessel injection (DVI) line break in a valve vault room. For each of these LOCA sequences, four sensitivity cases were run to determine the effects of cavity flooding, core-concrete interaction, igniters (local burn versus global burn) and jet burning of the heated hydrogen-rich RCS gas discharge. A total of ten sequences were quantified. The designator and description for each of the ten sequences are summarized in Table D-6.~~

~~These key events in the severe accident progression directly relate to the equipment survivability time frames. Time Frame 1 is the interval between core uncovery and a core exit gas temperature exceeding 2000°F (1367 K). Time Frame 2 is the interval between the core exit gas temperature exceeding 1367 K and either the end of core material relocation into the lower head or vessel failure. Time Frame 3 is the interval between vessel failure and the end of the sequence.~~

~~The MAAP4 results provide the bounding containment environment associated with the combustion of hydrogen resulting from the equivalent of 100% oxidation of the active fuel cladding where: 1) igniters are functioning (local burning scenario), 2) igniters were artificially defeated (global burning scenario), and 3) jet burning and igniters were defeated (global burning scenario). To calculate more severe bounding containment environments, the cavity flooding was defeated in some sequences resulting in ex-vessel hydrogen generation due to core-concrete interaction.~~

~~The results of 4-inch DVI line break sequences are very similar to the hot leg large LOCA results because the ADS 4th stage valves are opened in both sequences. The RCS response for these low pressure sequences is very similar.~~
~~In general, results with and without igniters were very similar because of the jet burn of the gas flow coming out from the primary system. For cases without cavity flooding, more hydrogen was generated due to the core-concrete interaction such that late hydrogen burns were observed.~~

~~For the hot leg LOCA sequences with the cavity flooding available, the water level in the containment eventually reached the hot leg break elevation and the whole core became submerged by water in the later~~

~~transient. The reverse water flow through the break did not occur in the DVI line break sequences because of the high floor elevation of the valve vault.~~

~~**D.7.2.2 Global Hydrogen Combustion**~~

~~This section presents the containment environmental response to a scenario in which the hydrogen generated from 100 percent cladding reaction is allowed to mix in the containment and then is globally burned. The case presented here is the same as case FRF1 from the AP1000 hydrogen combustion analysis (Chapter 41), except that, after the cladding reaction is completed and the hydrogen is mixed in the containment, a burn is initiated that produces a pressure and temperature spike in the containment.~~

Bounding severe accident environments are provided in this section. Five severe accident cases are analyzed with the MAAP4.04 code to generate the environment. The MAAP4 code input parameters are set to produce bounding cladding oxidation in each of the analyses.

The five cases are:

- IGN – DVI line break with vessel reflood, cavity flooding and igniter
- IVR – DVI line break with cavity flooding and igniters, no vessel reflood
- NOIGN – 4-inch DVI line break with vessel reflood, cavity flooding and no igniters
- CCI – Large LOCA with igniters, no vessel reflood and no cavity flooding
- GLOB – Global burning of hydrogen from 100% cladding reaction

The event timing for each case is presented in Table D-6. These key events relate directly to the equipment survivability time frames.

D.7.2.1 Case IGN – Large In-Vessel Hydrogen Release Burned at Igniters

Case IGN provides a containment environment with high rate of hydrogen generation from vessel reflooding, sustained steaming from stage 4 ADS, and hydrogen burning at the igniters. The MAAP4 results are presented in Figures D-3 through D-11.

The accident sequence is initiated by a DVI line break into a PXS compartment. The compartment floods with water from the IRWST and fills above the break elevation, allowing the vessel to reflood. Reflooding the overheated core causes a large fraction of the zirconium cladding to oxidize, however, relocation of the core to the lower plenum of the reactor vessel is prevented.

The hydrogen produced in vessel is released to the containment through the ADS system and through the break. It burns at igniters placed throughout the containment.

D.7.2.2 Case IVR – In-Vessel Retention of Core Debris with Cavity Flooding and Igniters, No Vessel Reflood

Case IVR provides a containment environment from the bounding in-vessel retention case. The MAAP4 results are presented in Figures D-12 through 20.

The case is initiated by a DVI line break. The cavity is flooded, but the PXS compartment is not. The break is above the water level in the compartment. Water is unable to get back into the vessel and reflood the core. The core melts and collects in the lower plenum of the vessel, but is not quenched. The external surface of the vessel lower head is cooled with water from the IRWST, and the vessel remains intact. The amount of hydrogen generation in-vessel is low since the cladding oxidation reaction is water limited.

Hydrogen that is generated in-vessel is released to the containment through the ADS system, and burns at the igniters placed throughout the containment.

D.7.2.3 Case NOIGN – Igniter Failure

Case IGN provides a containment environment with high rate of hydrogen generation from vessel reflooding, sustained steaming from stage 4 ADS, and a large global hydrogen burn in the long term. The MAAP4 results are presented in Figures D-21 through 29.

The accident sequence is initiated by a DVI line break into a PXS compartment. The compartment floods with water from the IRWST and fills above the break elevation, allowing the vessel to reflood. Reflooding the overheat core causes a large fraction of the zirconium cladding to oxidize, however, relocation of the core to the lower plenum of the reactor vessel is prevented.

The hydrogen produced in vessel is released to the containment through the ADS system and mixes in the containment. It is assumed to be ignited randomly at 8 hours and produces a large global burn.

D.7.2.4 Case CCI – Vessel Failure and Core Concrete Interaction

Case CCI provide a post-vessel failure containment environment with an unquenched and non-coolable debris bed in the reactor cavity. The MAAP4 results are presented in Figures D-30 through D-38.

The accident sequence is initiated by a spurious opening of an ADS stage 4 valve. The cavity is not flooded. The core melt progresses to vessel failure and debris is released to the containment.

There is little hydrogen produced in-vessel since the oxidation reaction is water limited. However, hydrogen is released from the debris during the core concrete interaction. The hydrogen burns at the igniters until the containment becomes oxygen starved.

D.7.2.5 Case GLOB - Global Combustion of Hydrogen Produced from 100% Cladding Reaction

Case GLOB presents a bounding hydrogen combustion case, burning the mass of hydrogen produced from 100% oxidation of the active cladding in the core. The oxidation reaction produces 788 kg of hydrogen. The MAAP4 results are presented in Figures D-39 through D-46.

D.7.2.36 Sustained Burning Environments

Sustained burning of combustible gases can occur in the containment as a diffusion flame at the location where the gas plume encounters a continuous oxygen source. Equipment that is needed after core damage (Time Frames 2 and 3) should either be located well away or shielded from these locations or there should be other redundant equipment located outside the zone of influence to demonstrate reasonable assurance of the function survivability.

Burning at igniters away from combustible gas sources will be limited. The igniters will light off the plume, and the flame will flash back to the source of the combustible gas within a compartment supplied with air. The locations of the diffusion flames can be identified by identifying the combustible gas release points in the containment. Combustible gas generation begins as hydrogen is released from the RCS to the containment during the in-vessel phase of the accident (Time Frame 2). After vessel failure (Time Frame 3), hydrogen and carbon monoxide can be released by core-concrete interaction. A continuous oxygen source can be provided in the compartments, which form the natural circulation flow path in the containment, the steam generator and loop compartment rooms, the CMT room, and the upper compartment. Therefore, diffusion flame environments can be postulated in these compartments near the combustible gas source.

Except for the break, the source locations are pre-determined by ADS vent points and the geometry of the containment. During the in-vessel phase of the accident, hydrogen will be released from the break and from the ADS system. If the system is fully depressurized, the sustained hydrogen release will be from the stage 4 ADS valves and, depending on the size and location, the break. ADS stages 1 through 3 relieve to the IRWST. The hydrogen is preferentially released from the IRWST through the pipe vents along the steam generator doghouse wall. The stage 4 valves relieve to the steam generator loop compartments. Generally, the break location can be postulated to be in one of the steam generator loop compartments also, which is essentially lumped together with the hydrogen release from the ADS stage 4 valves, but piping connected to the RCS is also located in the accumulator rooms or valve vaults and the CVS compartment. For releases to these dead-ended compartments, the plume cannot encounter an oxygen supply until it reaches the CMT room.

Ex-vessel combustible generation occurs in the reactor cavity in Time Frame 3. The reactor cavity does not have a continuous air supply, so the first locations where oxygen is available along the flow pathways is where the sustained burning can be postulated. The flow paths from the reactor cavity to the containment air supply are through the RCDT room access into the vertical access tunnel, through the loop nozzle holes into the steam generator rooms, and past the reactor vessel flange through the seal ring (should the seal ring fail) into the refueling pool.

Therefore, sustained burning can be postulated in the following locations:

1. IRWST pipe vent exits along the SG doghouse wall (Time Frame 2)
2. Stage 4 valves outlet in the steam generator loop compartments at 112-ft elevation (Time Frame 2)

3. Vents from the accumulator rooms into the CMT room (Time Frame 2). Sustained burning will, however, exist from only one of the two accumulator rooms for a given break.
4. RCDT room access into the vertical access tunnel (Time Frame 3)
5. Loop nozzle holes into the steam generator loop compartment (Time Frame 3)
6. Seal ring into the refueling pool (Time Frame 3)

~~Table D-6~~

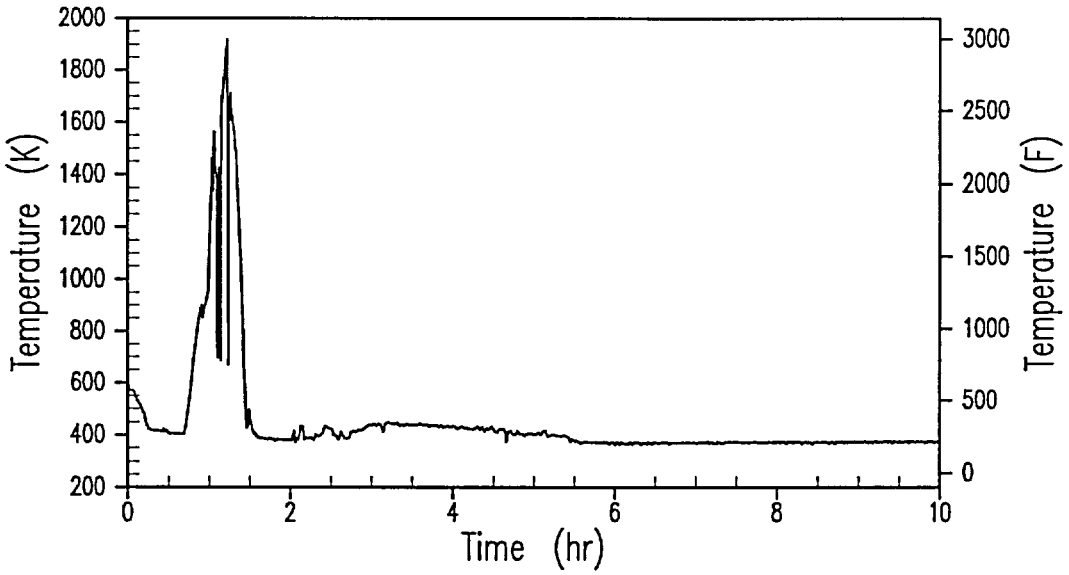
~~SEQUENCE DESIGNATOR~~	
~~Hot leg LOCA with a break area of 2.2 ft^2 with no reflood: 2 ADS Stage 1-3 and 4 ADS Stage 4, 1 accumulator, 1 CMT, and PCS available.~~	
~~APLHL~~	~~Large hot leg with cavity flooding, similar to the A3BE, except the break location and the area. The break location is in S/G compartment 1, rather than valve vault.~~
~~APLHL-N~~	~~APLHL + no cavity flooding + no ex-vessel cooling~~
~~APLHL-G~~	~~APLHL + no igniters~~
~~APLHL-GN~~	~~APLHL + no igniters + no cavity flooding + no ex-vessel cooling~~
~~APLHL-GJ~~	~~APLHL + no igniters + no jet burn~~
~~4-inch DVI line break with no reflood and no PRHR: 2 ADS Stage 1-3 and 4 ADS Stage 4, 1 accumulator, 1 CMT, and PCS available.~~	
~~A3BE~~	~~AP600 3BE sequence with cavity flooding, ex-vessel cooling, igniters, and jet burning~~
~~A3BE-N~~	~~A3BE + no cavity flooding + no ex-vessel cooling~~
~~A3BE-G~~	~~A3BE + no igniters~~
~~A3BE-GN~~	~~A3BE + no igniters + no cavity flooding + no ex-vessel cooling~~
~~A3BE-GJ~~	~~A3BE + no igniters + no jet burn~~

Table D-6

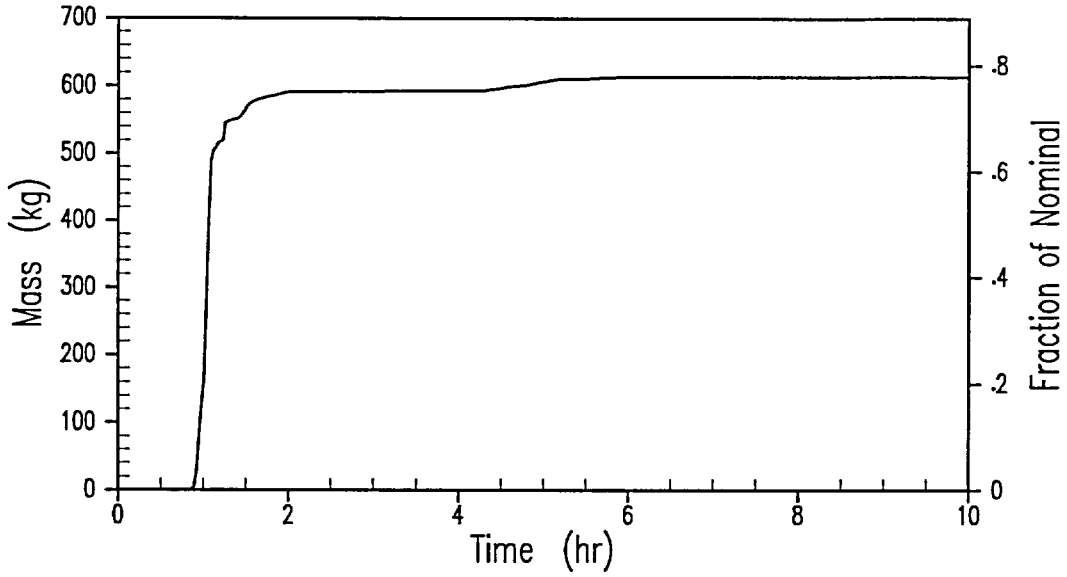
SUMMARY OF MAAP4 ANALYSES: EQUIPMENT SURVIVABILITY TIME FRAMES

Key Quantity or Timing	SEQUENCES				
	IGN	IVR	NOIGN	CCI	GLOB
Cladding Oxidation in-vessel (%)	78	48	86		100
Time of Core Uncovery (s)	2481	2483	2481	2285	14
Time (s) Core Exit Gas Temp >1367°K	3318	3320	3318	3672	160
Time of Initial Core Material Relocation to Lower Plenum (s)	-	11100	-	5940	-
Time Core Material Relocation to Lower Plenum Ends (s)	-	11600	-	9000	-
Time of Vessel Failure (s)	-	-	-	9288	-

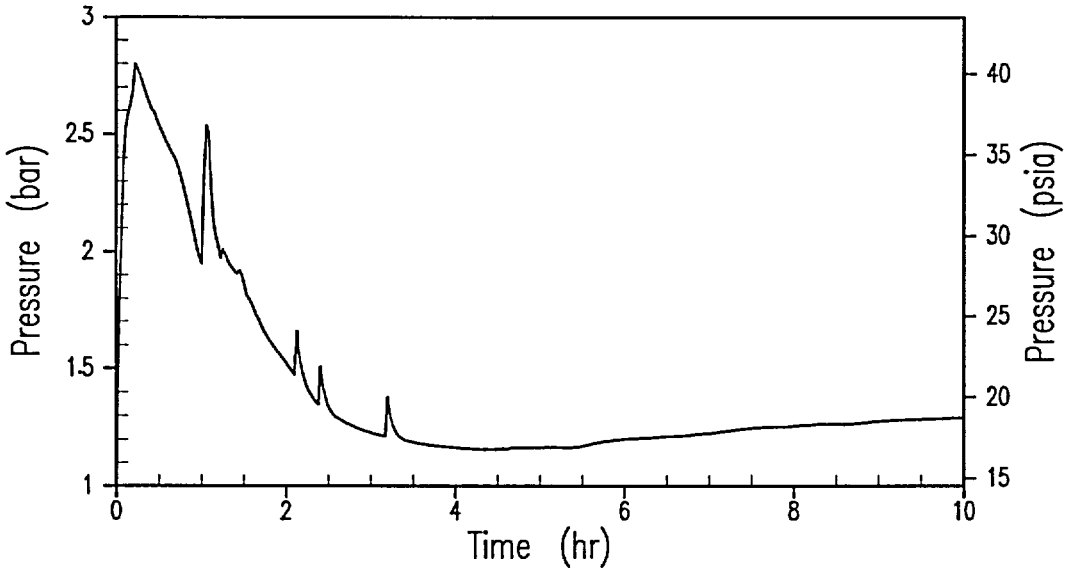
AP1000 Eq Surv Case IGN – Hydrogen Burning at Igniters
Upper Plenum Gas Temperature



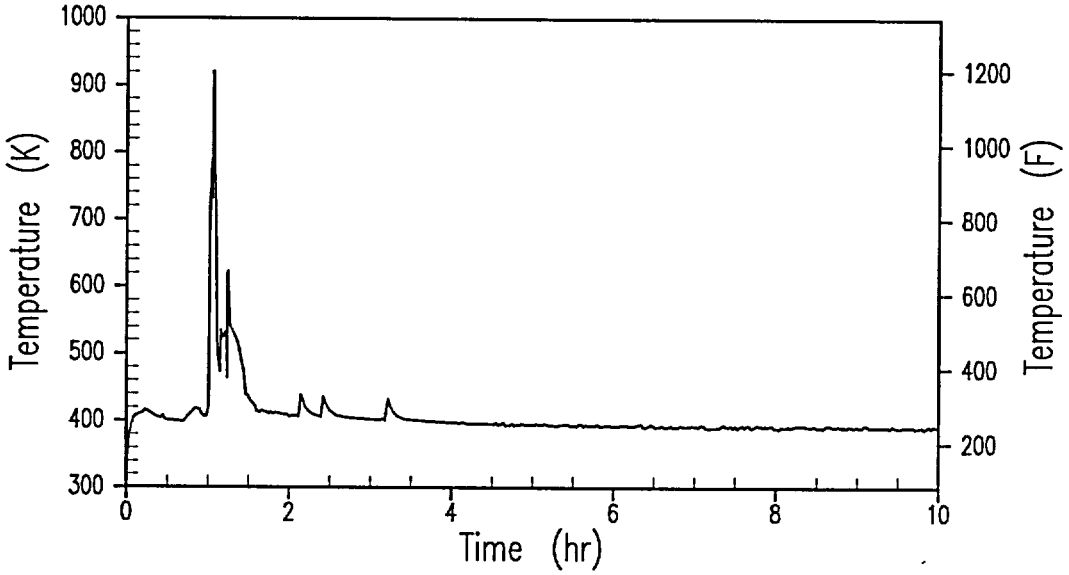
AP1000 Eq Surv Case IGN – Hydrogen Burning at Igniters
In–Vessel Hydrogen Generation



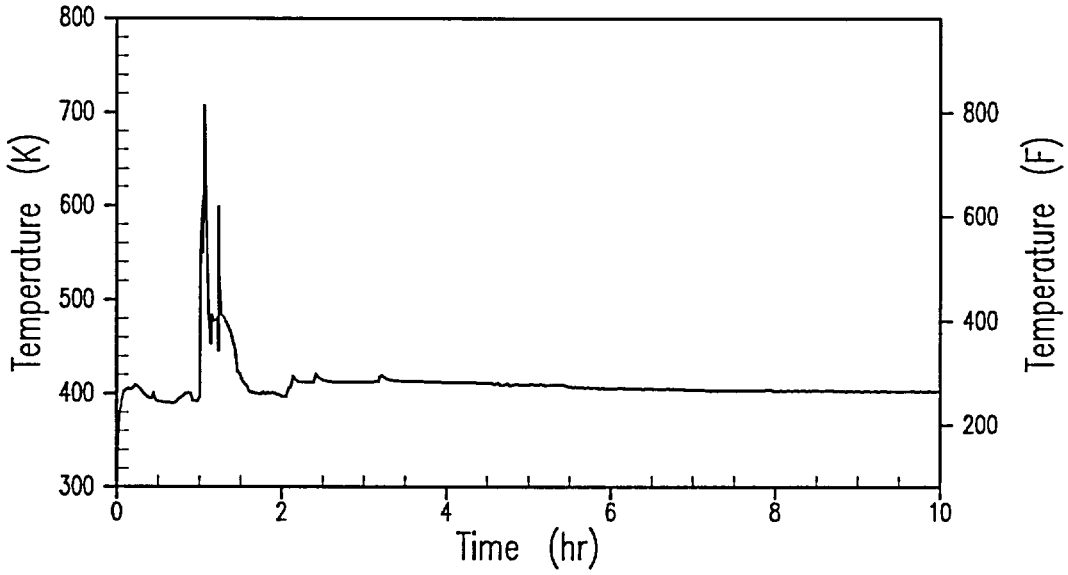
AP1000 Eq Surv Case IGN – Hydrogen Burning at Igniters
Containment Upper Compt Pressure



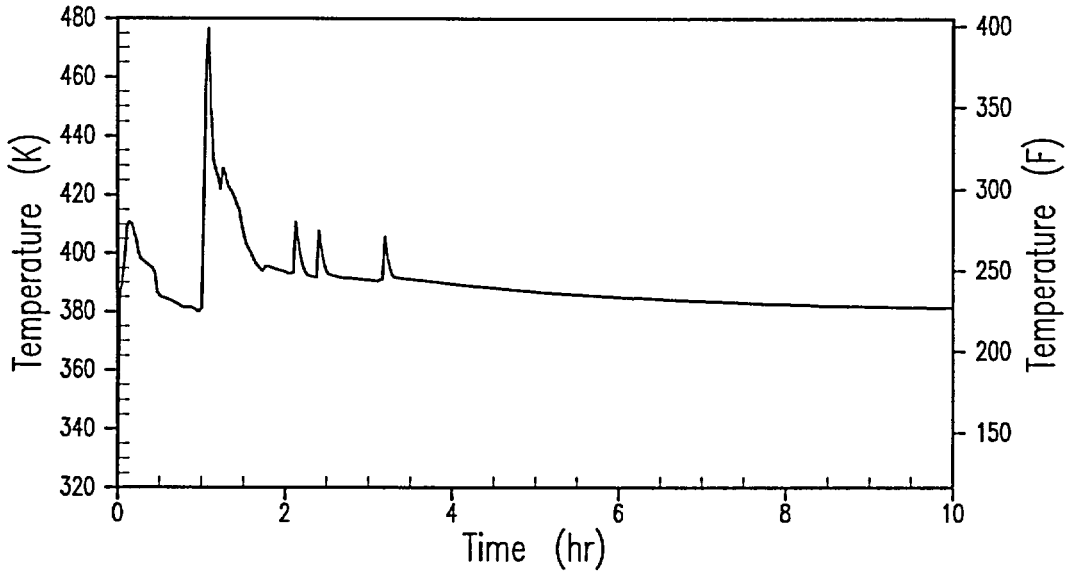
AP1000 Eq Surv Case IGN – Hydrogen Burning at Igniters
SG Compt 1 Gas Temperature



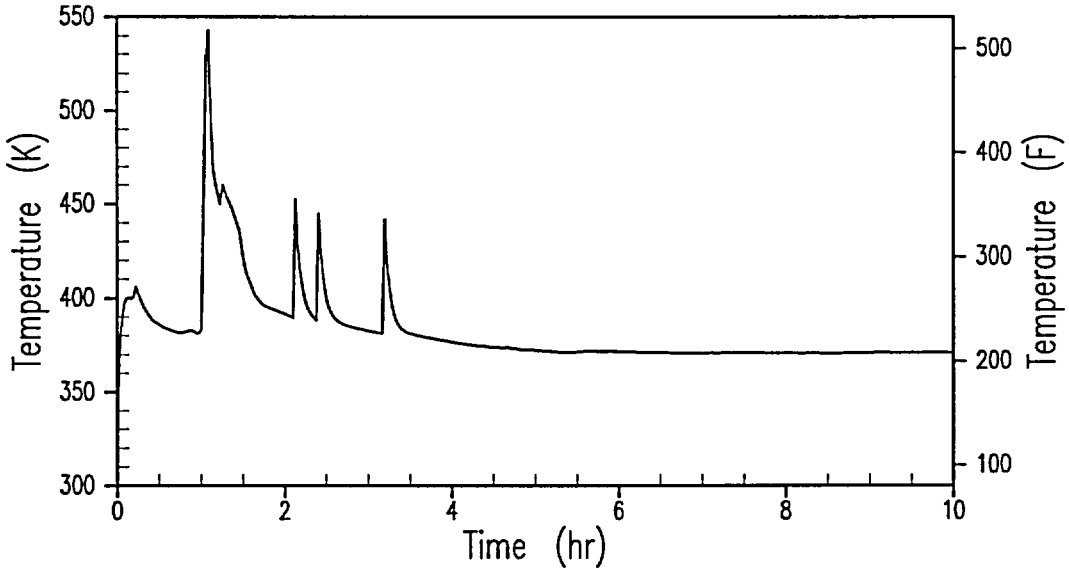
AP1000 Eq Surv Case IGN – Hydrogen Burning at Igniters
SG Compt 2 Gas Temperature



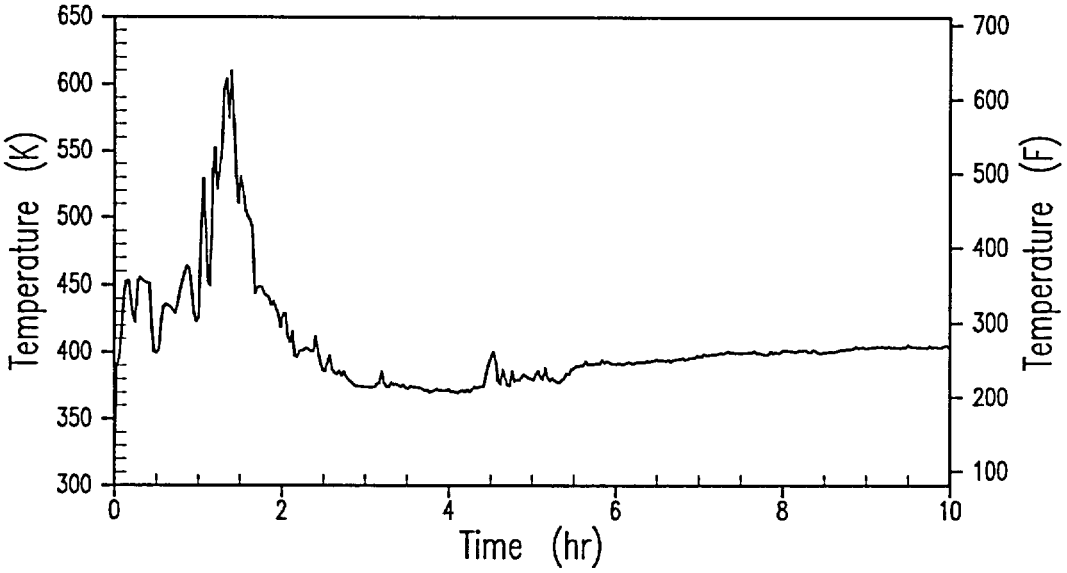
AP1000 Eq Surv Case IGN – Hydrogen Burning at Igniters
CMT Room Gas Temperature



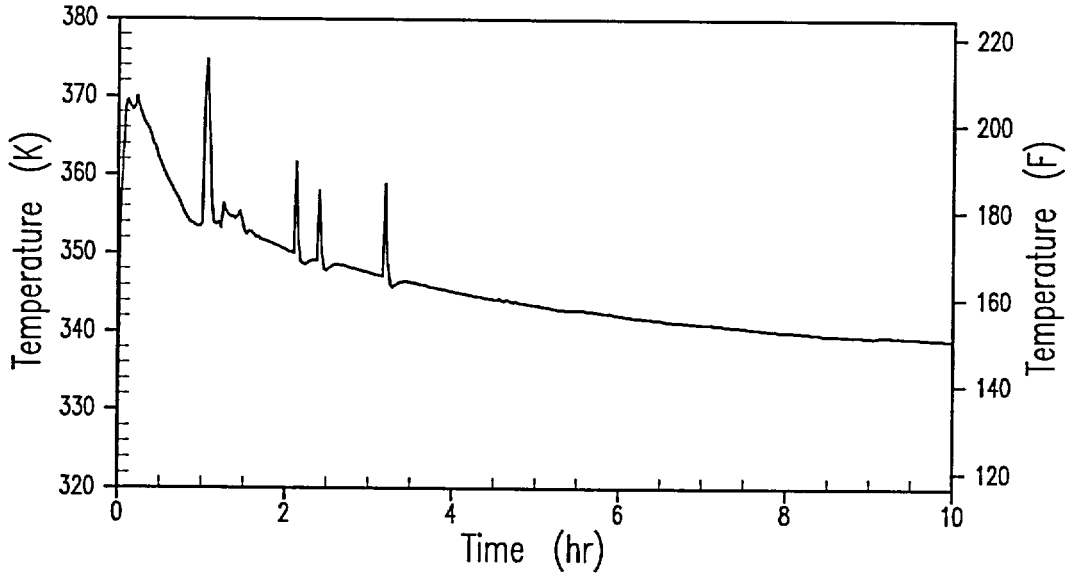
AP1000 Eq Surv Case IGN – Hydrogen Burning at Igniters
Upper Compt Gas Temperature



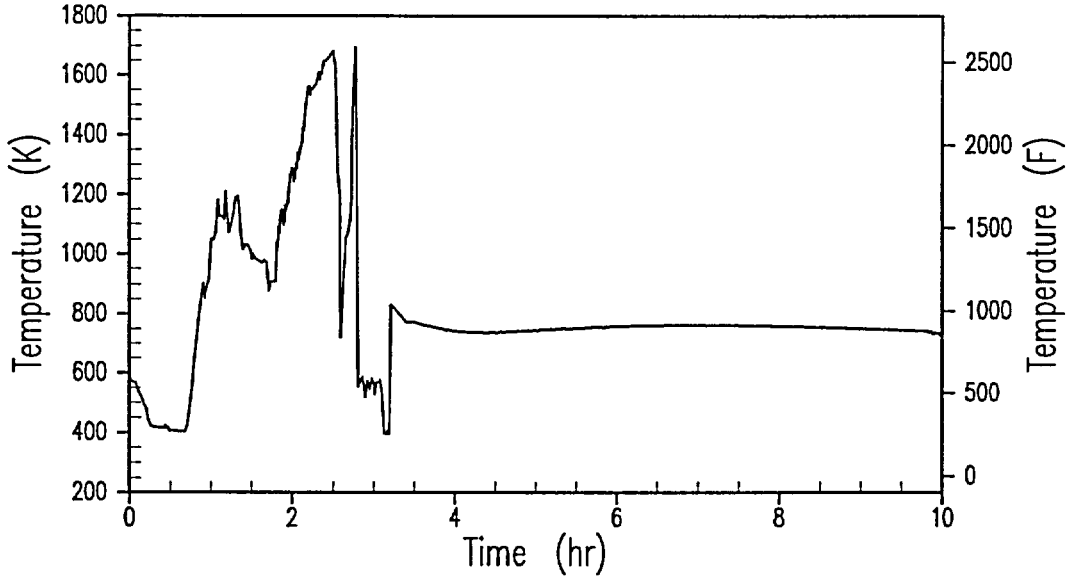
AP1000 Eq Surv Case IGN – Hydrogen Burning at Igniters
Faulted PXS Compt Gas Temperature



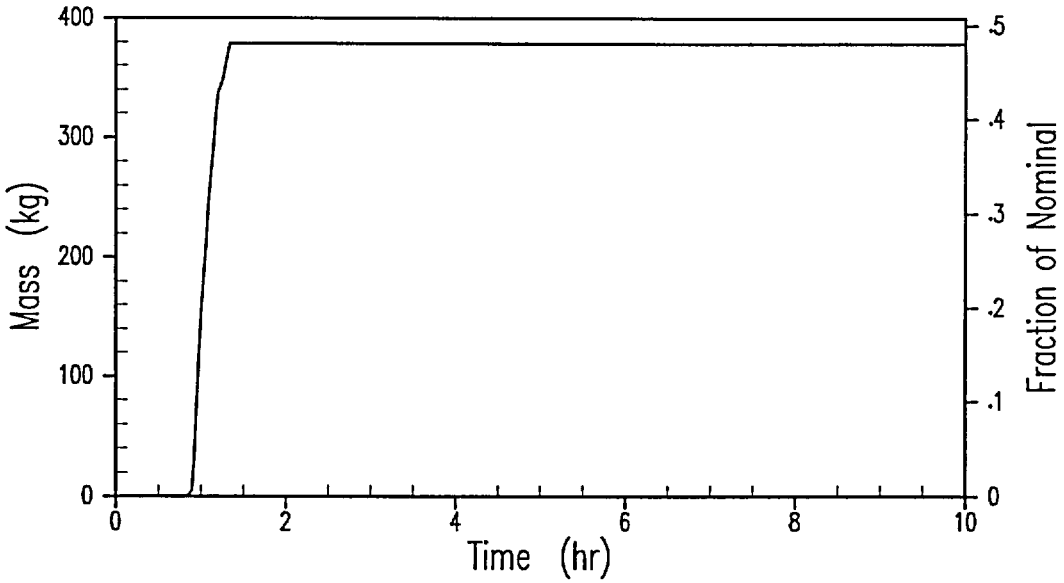
AP1000 Eq Surv Case IGN – Hydrogen Burning at Igniters
Intact PXS Compt Gas Temperature



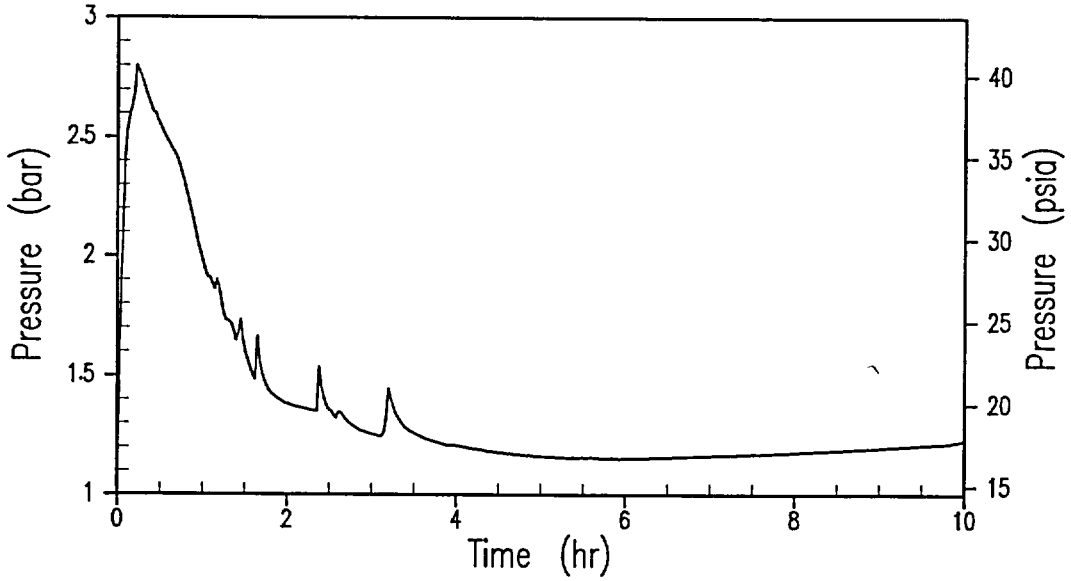
AP1000 Eq Surv Case IVR – In–Vessel Retention of Core Debris
Upper Plenum Gas Temperature



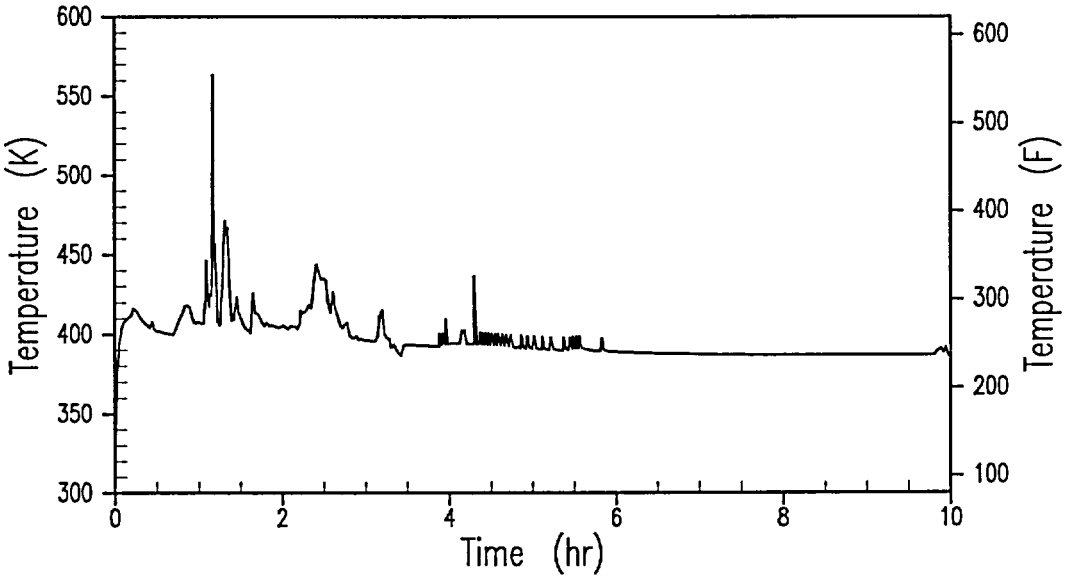
AP1000 Eq Surv Case IVR – In–Vessel Retention of Core Debris
In–Vessel Hydrogen Generation



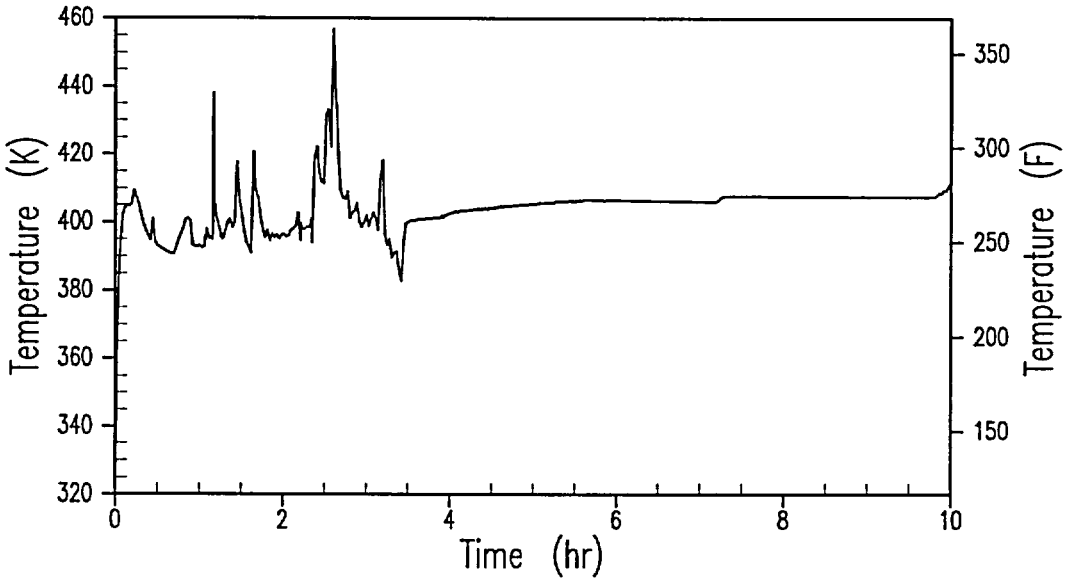
AP1000 Eq Surv Case IVR – In–Vessel Retention of Core Debris
Containment Upper Compt Pressure



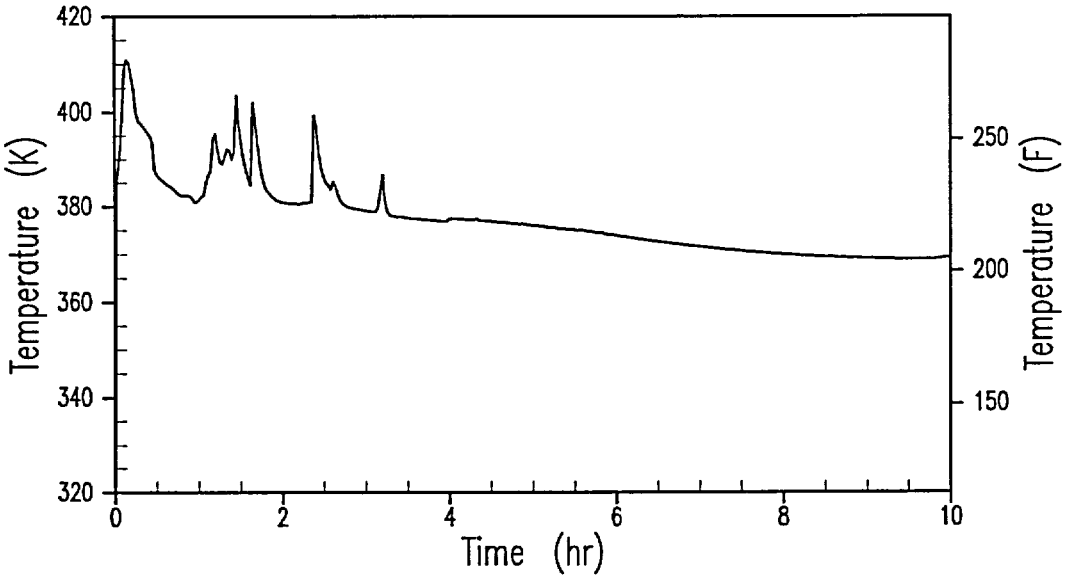
AP1000 Eq Surv Case IVR – In-Vessel Retention of Core Debris
SG Compt 1 Gas Temperature



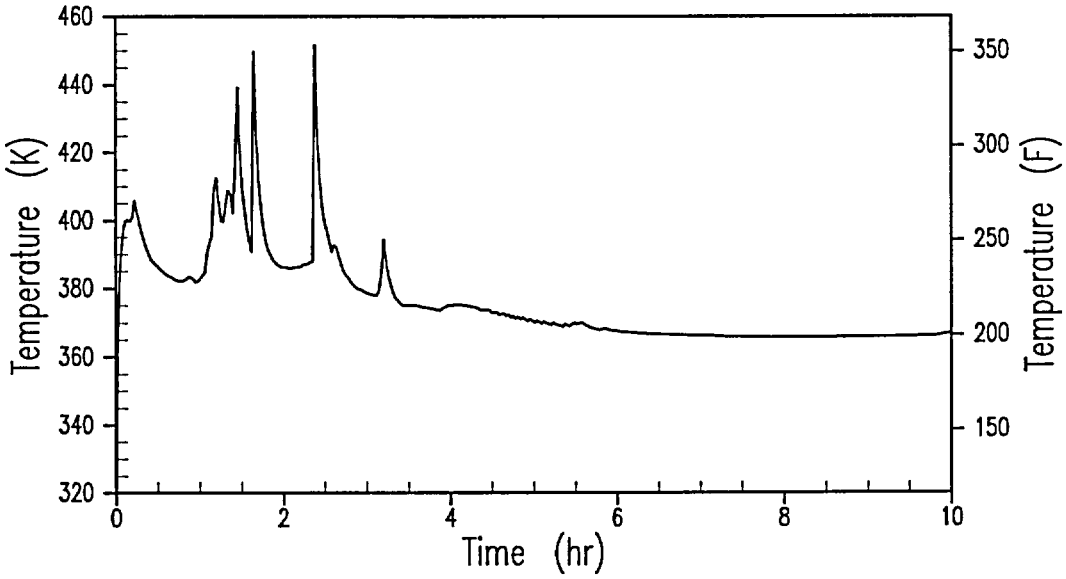
AP1000 Eq Surv Case IVR – In–Vessel Retention of Core Debris
SG Compt 2 Gas Temperature



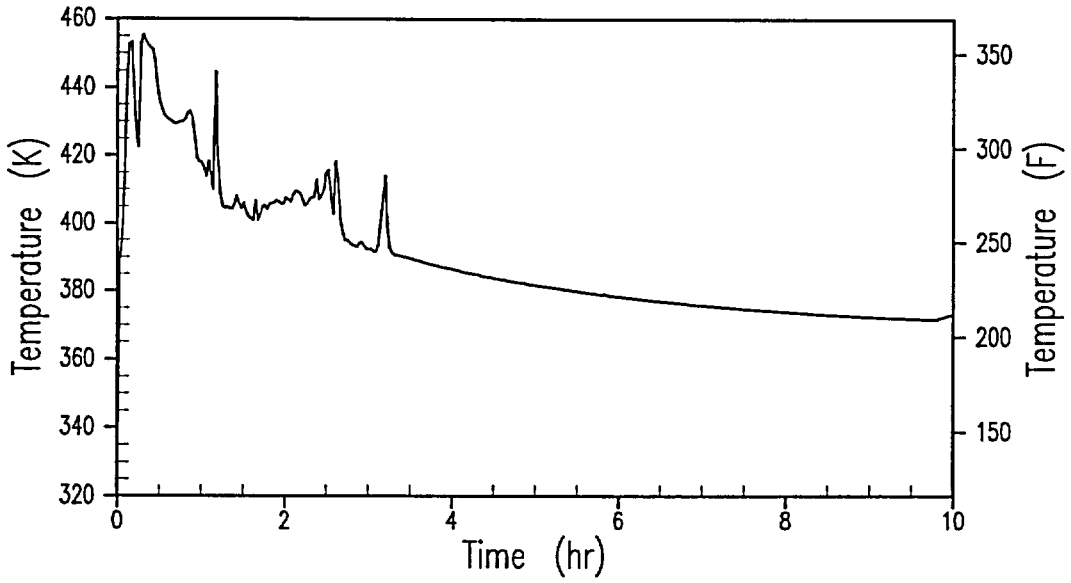
AP1000 Eq Surv Case IVR – In–Vessel Retention of Core Debris
CMT Room Gas Temperature



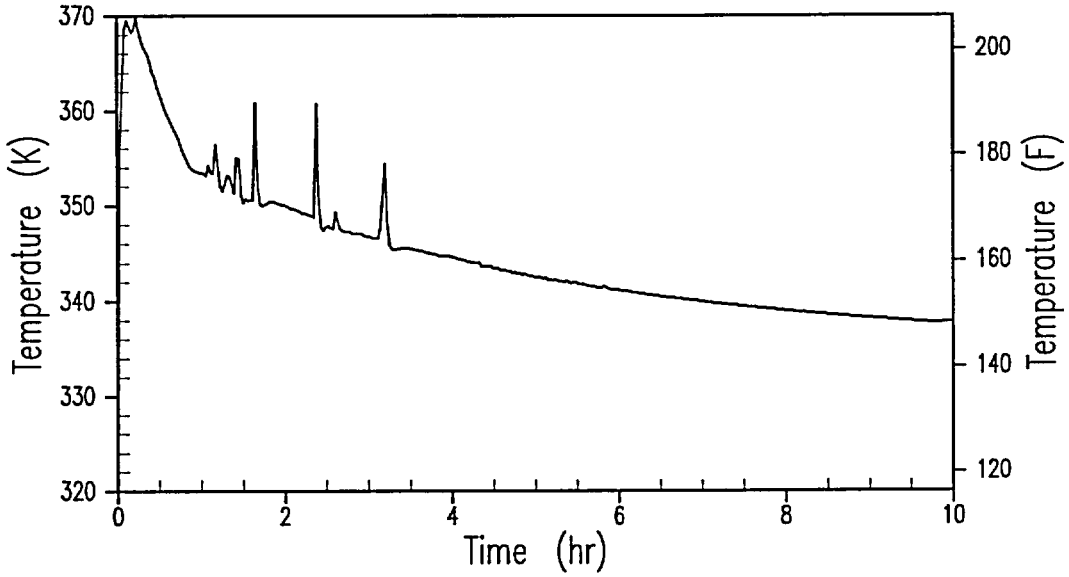
AP1000 Eq Surv Case IVR – In–Vessel Retention of Core Debris
Upper Compt Gas Temperature



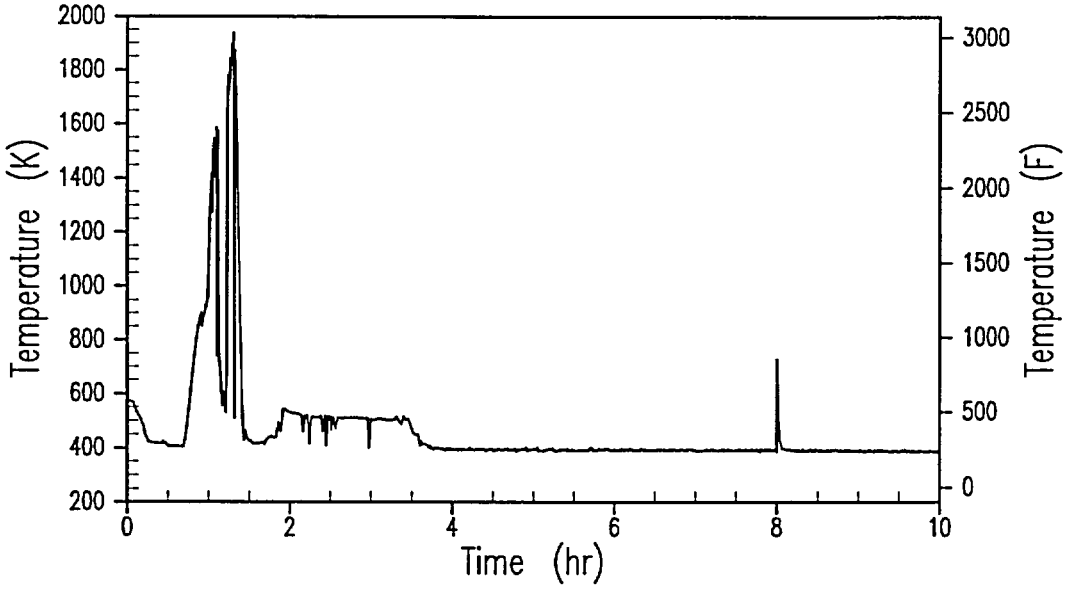
AP1000 Eq Surv Case IVR – In-Vessel Retention of Core Debris
Faulted PXS Compt Gas Temperature



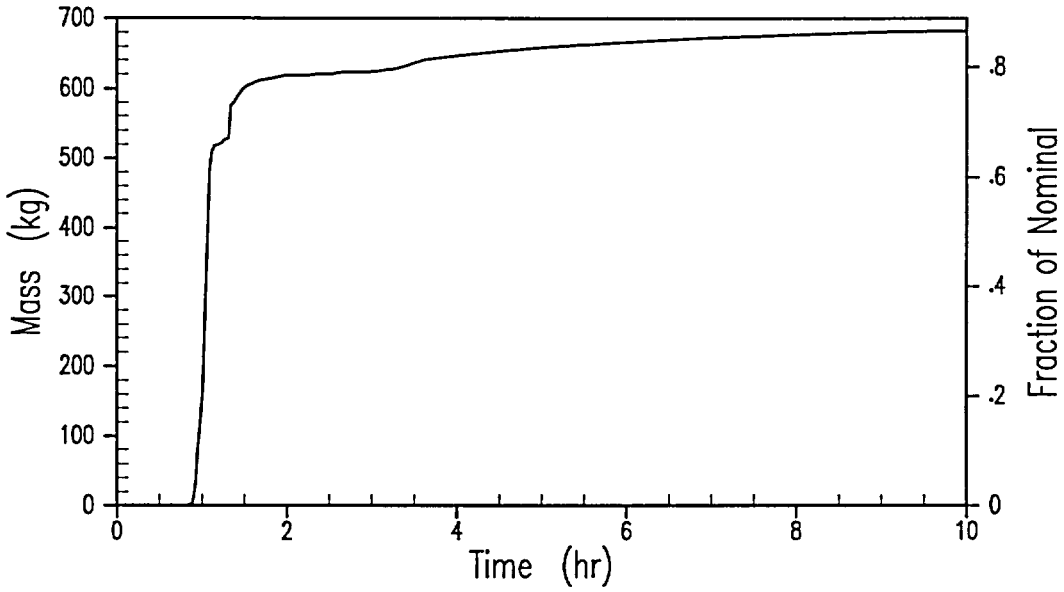
AP1000 Eq Surv Case IVR – In–Vessel Retention of Core Debris
Intact PXS Compt Gas Temperature



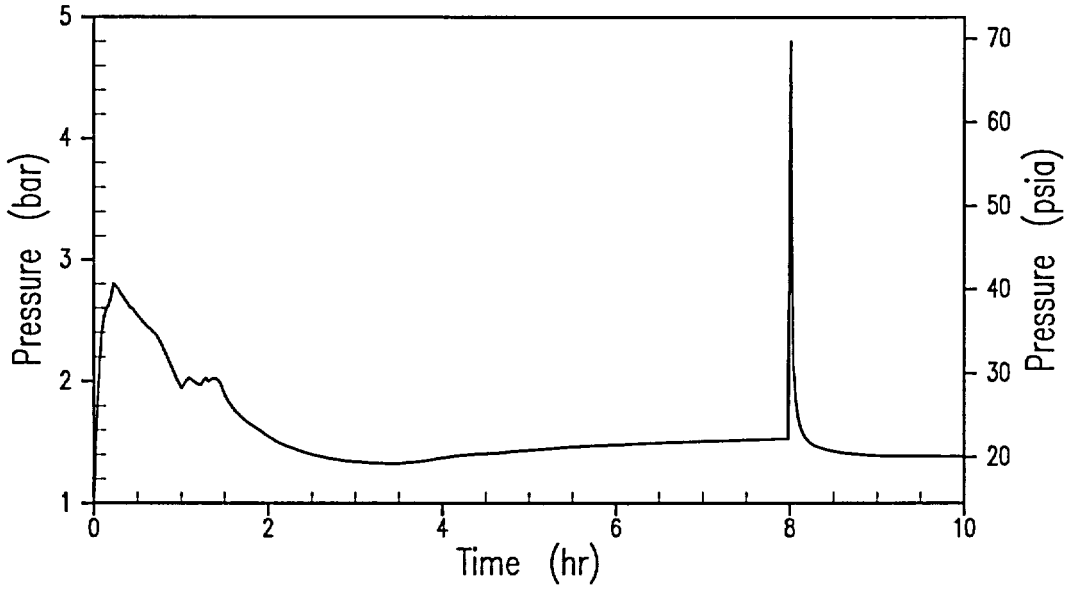
AP1000 Eq Surv Case NOIGN – DVI Line Break without Igniter Operation
Upper Plenum Gas Temperature



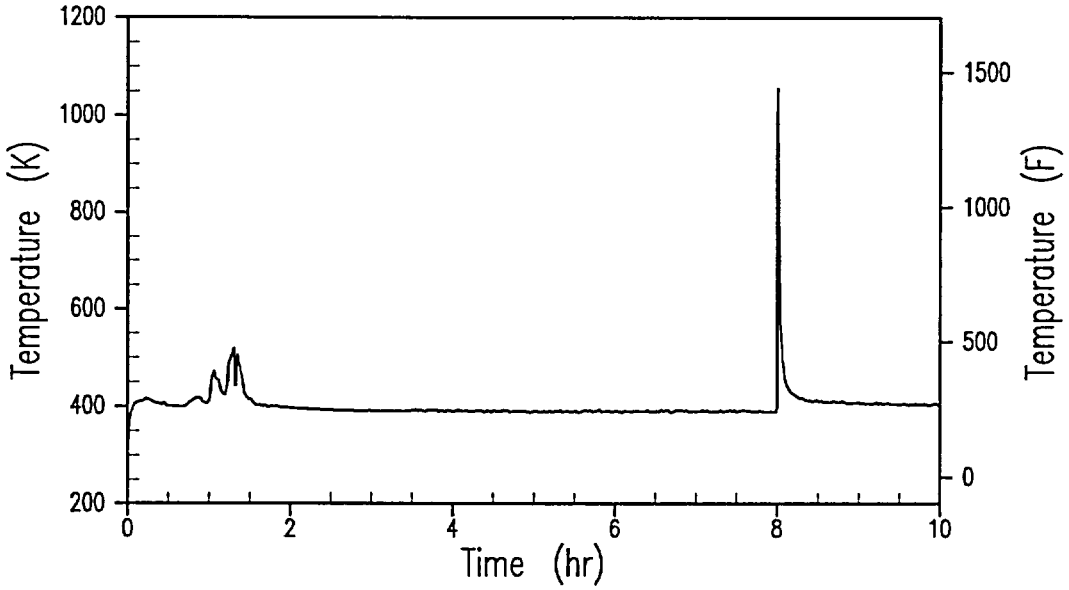
AP1000 Eq Surv Case NOIGN – DVI Line Break without Igniter Operation
In–Vessel Hydrogen Generation



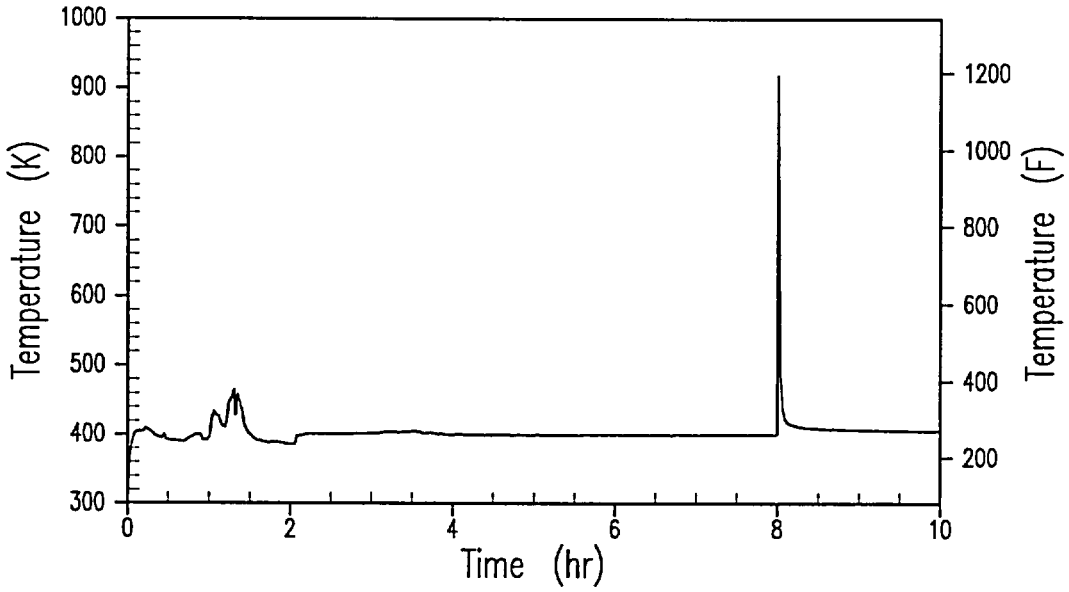
AP1000 Eq Surv Case NOIGN – DVI Line Break without Igniter Operation
Containment Upper Compt Pressure



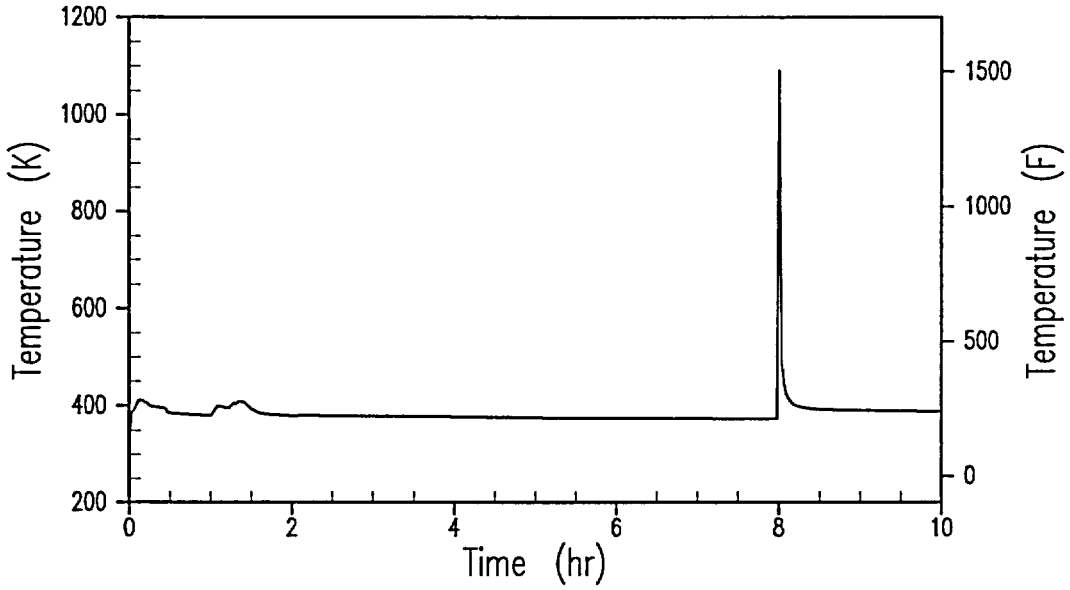
AP1000 Eq Surv Case NOIGN – DVI Line Break without Igniter Operation
SG Compt 1 Gas Temperature



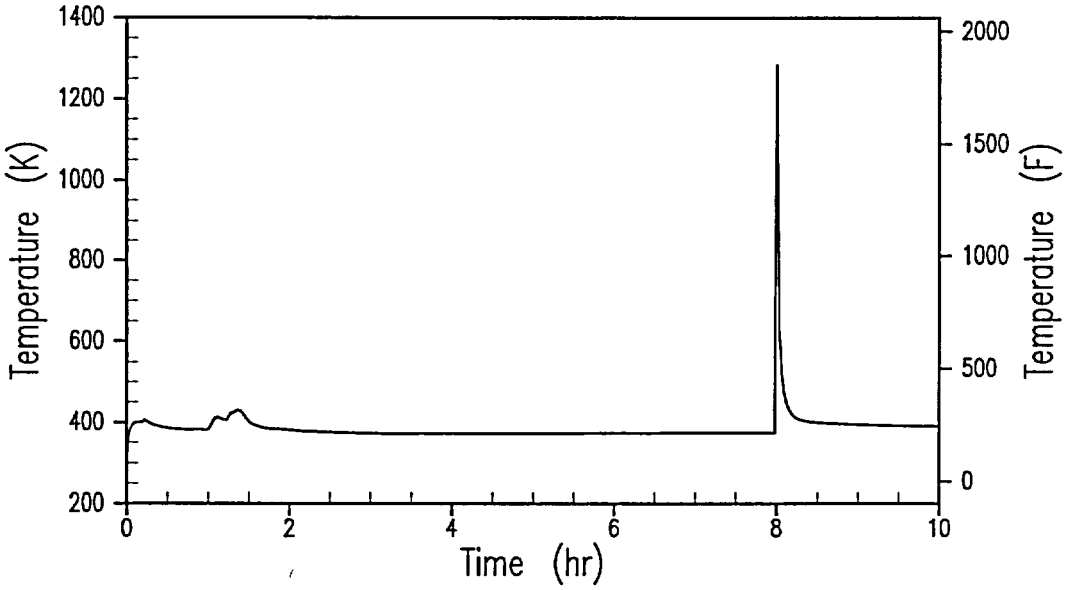
AP1000 Eq Surv Case NOIGN – DVI Line Break without Igniter Operation
SG Compt 2 Gas Temperature



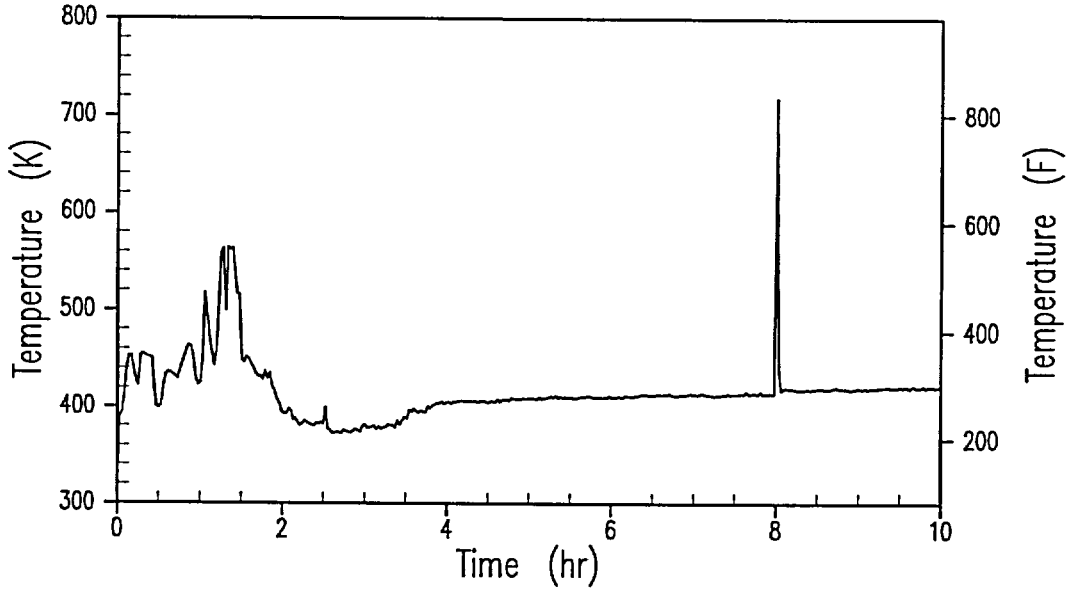
AP1000 Eq Surv Case NOIGN – DVI Line Break without Igniter Operation
CMT Room Gas Temperature



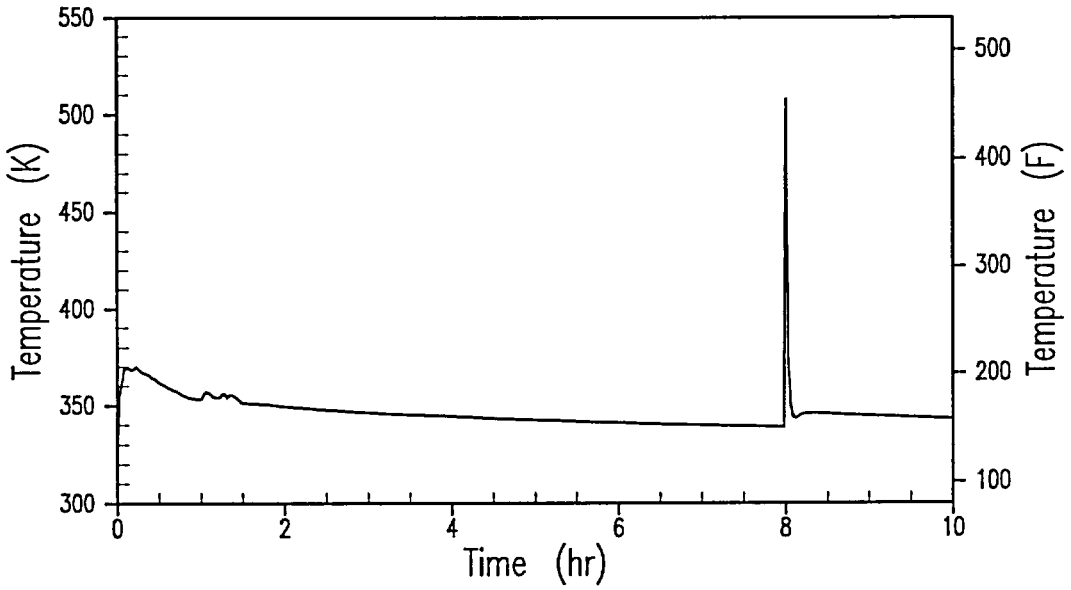
AP1000 Eq Surv Case NOIGN – DVI Line Break without Igniter Operation
Upper Compt Gas Temperature



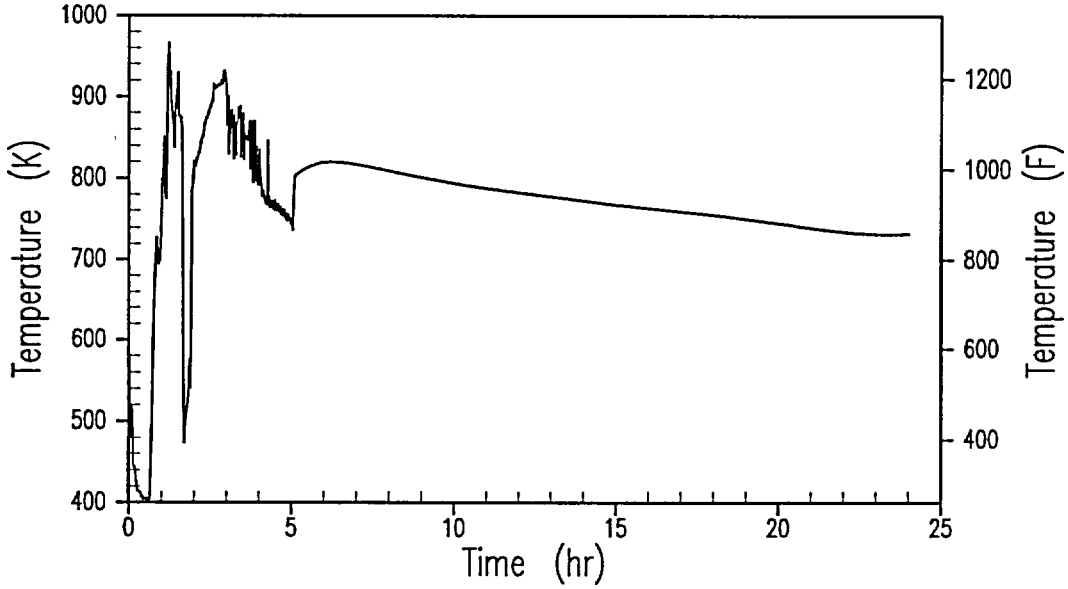
AP1000 Eq Surv Case NOIGN – DVI Line Break without Igniter Operation
Faulted PXS Compt Gas Temperature



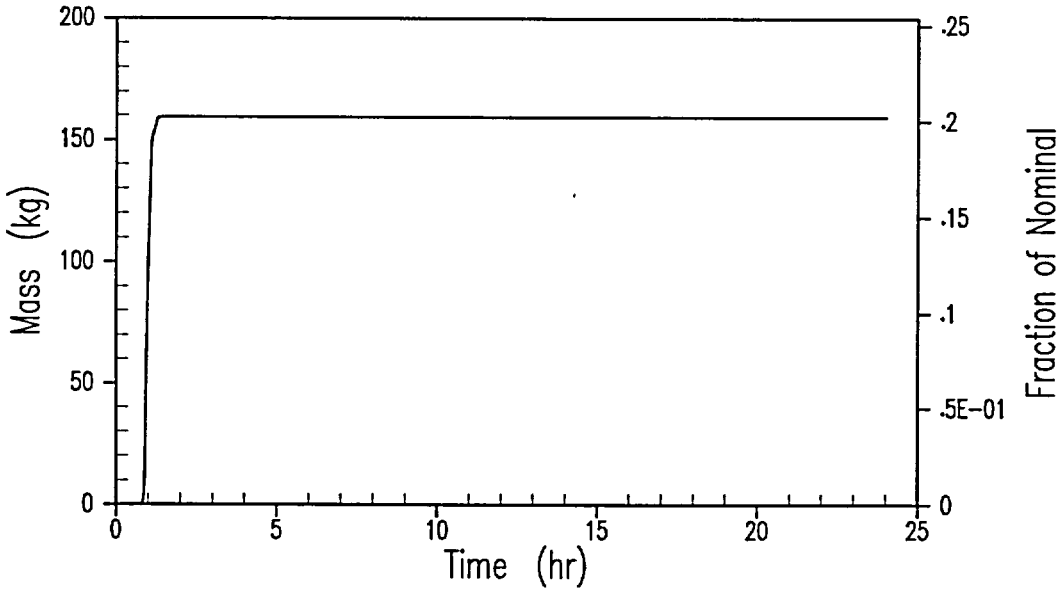
AP1000 Eq Surv Case NOIGN – DVI Line Break without Igniter Operation
Intact PXS Compt Gas Temperature



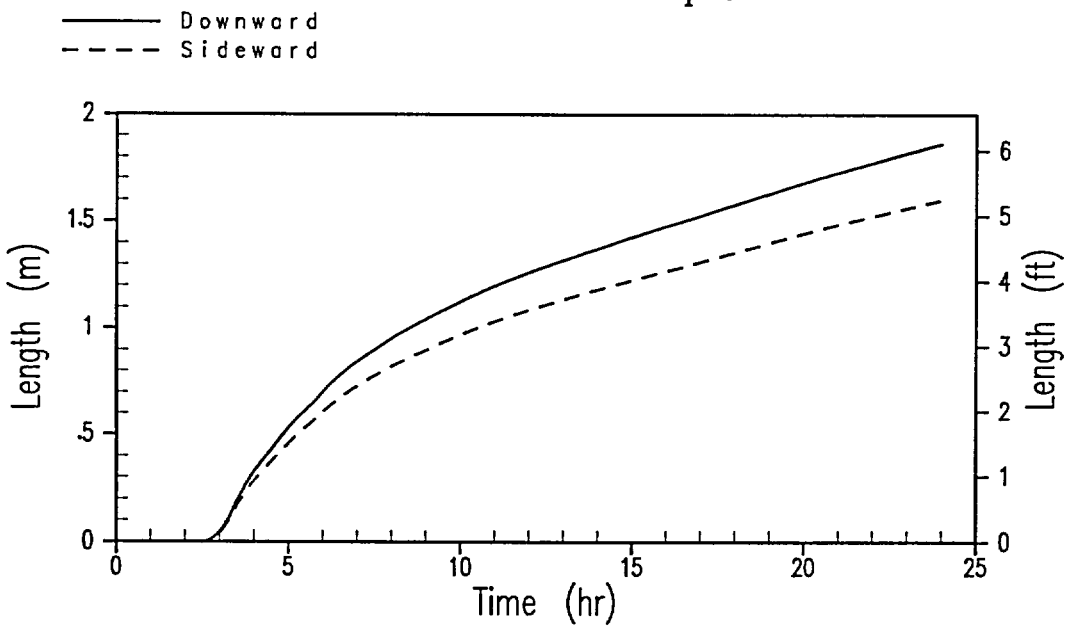
AP1000 Eq Surv Case CCI – Uncoolable Debris Bed in the Reactor Cavity
Upper Plenum Gas Temperature



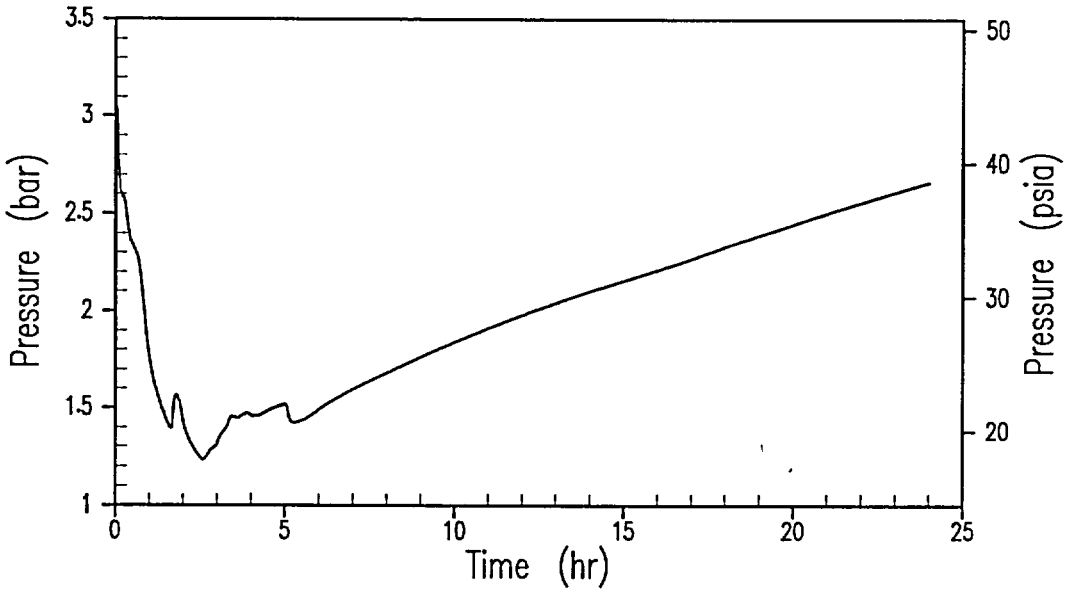
AP1000 Eq Surv Case CCI – Uncoolable Debris Bed in the Reactor Cavity
In–Vessel Hydrogen Generation



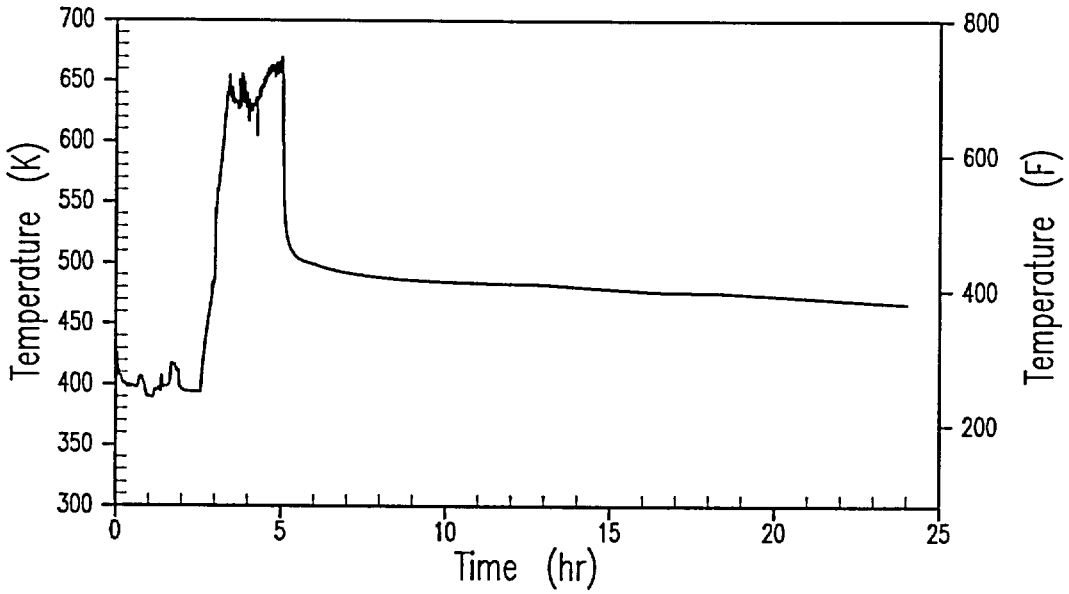
AP1000 Eq Surv Case CCI – Uncoolable Debris Bed in the Reactor Cavity
Concrete Penetration Depth



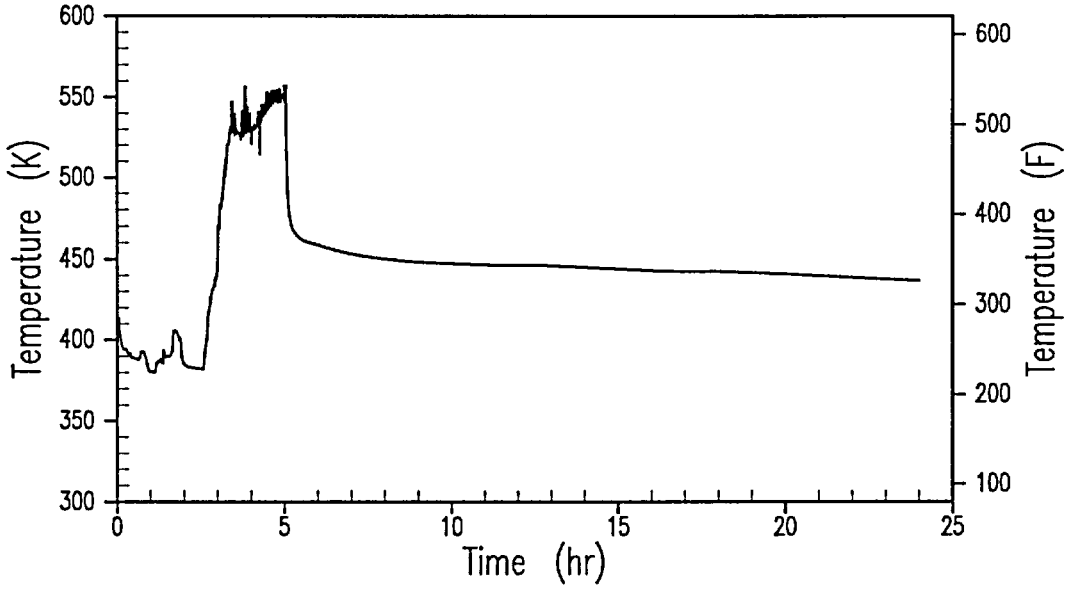
AP1000 Eq Surv Case CCI – Uncoolable Debris Bed in the Reactor Cavity
Containment Upper Compt Pressure



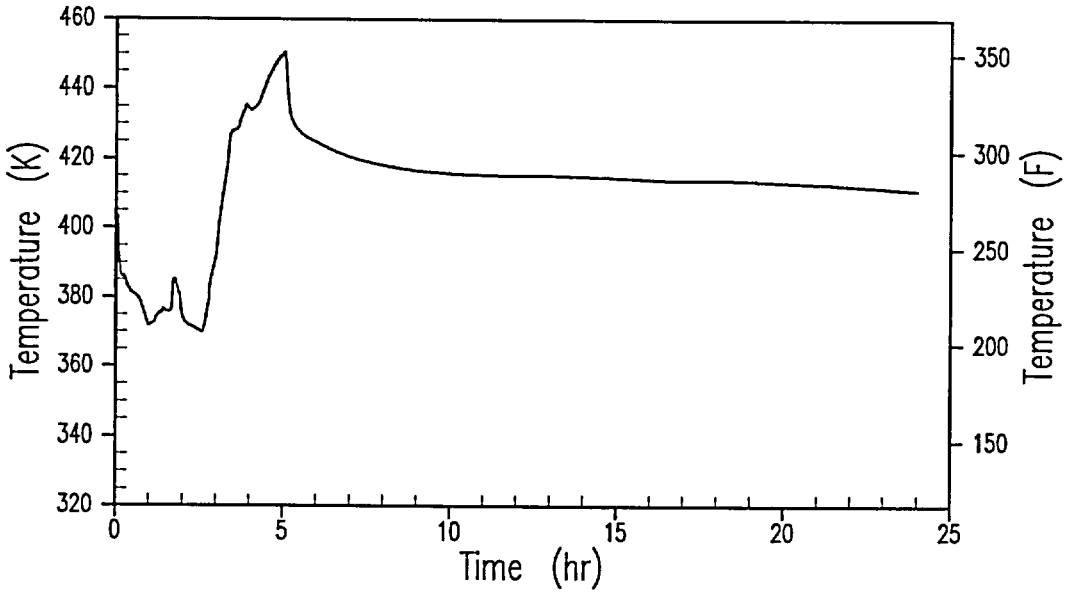
AP1000 Eq Surv Case CCI – Uncoolable Debris Bed in the Reactor Cavity
SG Compt 1 Gas Temperature



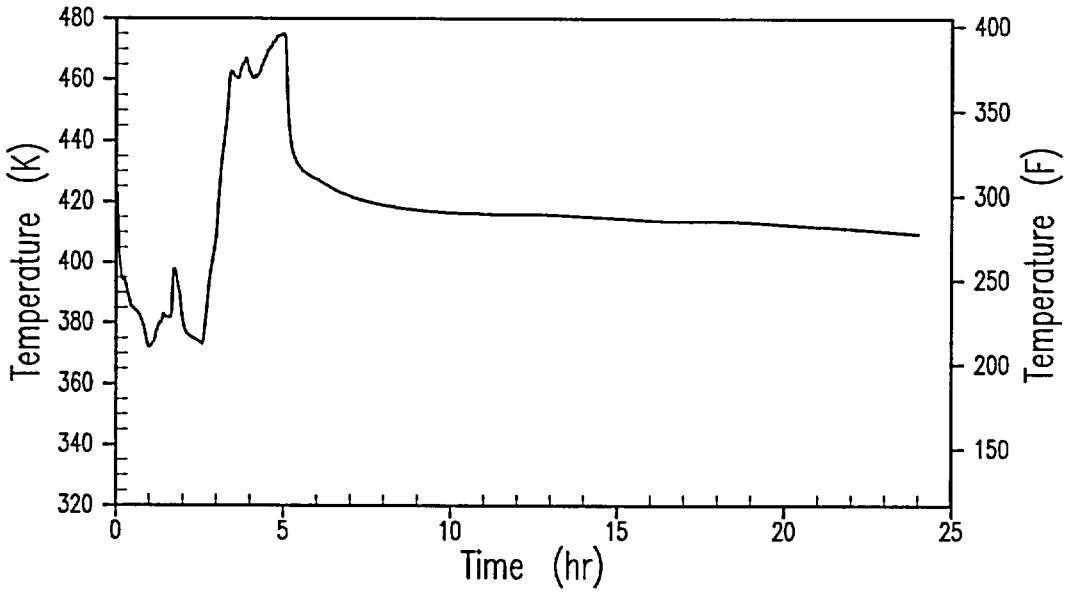
AP1000 Eq Surv Case CCI – Uncoolable Debris Bed in the Reactor Cavity
SG Compt 2 Gas Temperature



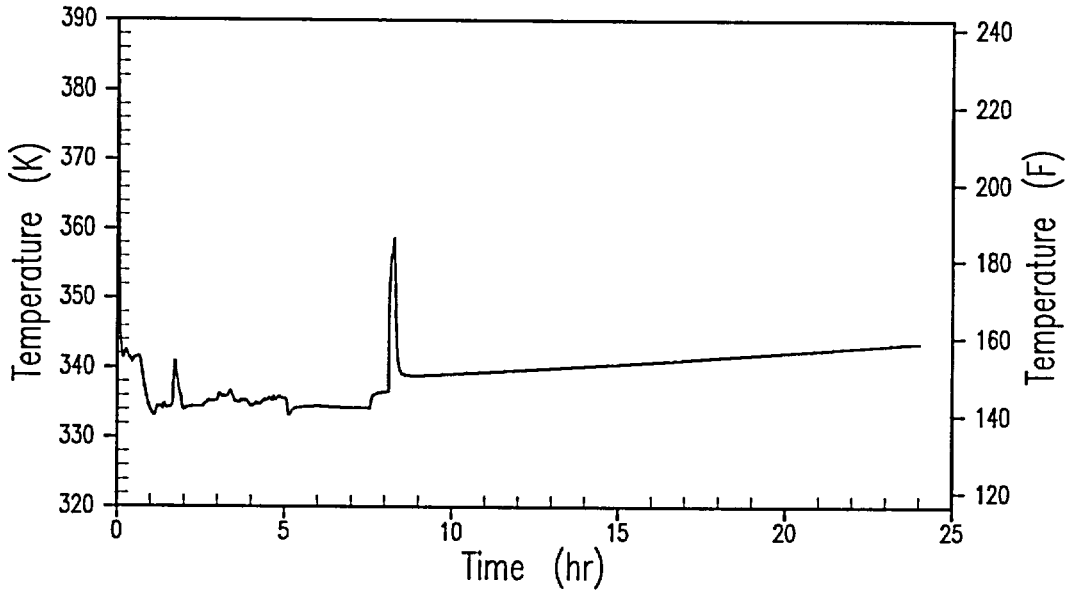
AP1000 Eq Surv Case CCI – Uncoolable Debris Bed in the Reactor Cavity
CMT Room Gas Temperature



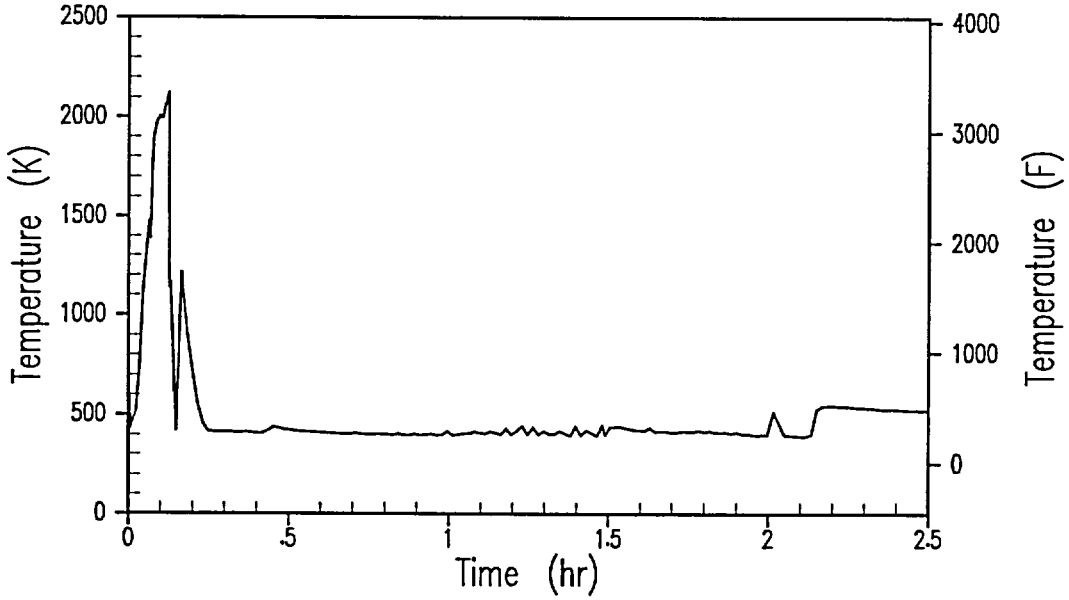
AP1000 Eq Surv Case CCI – Uncoolable Debris Bed in the Reactor Cavity
Upper Compt Gas Temperature



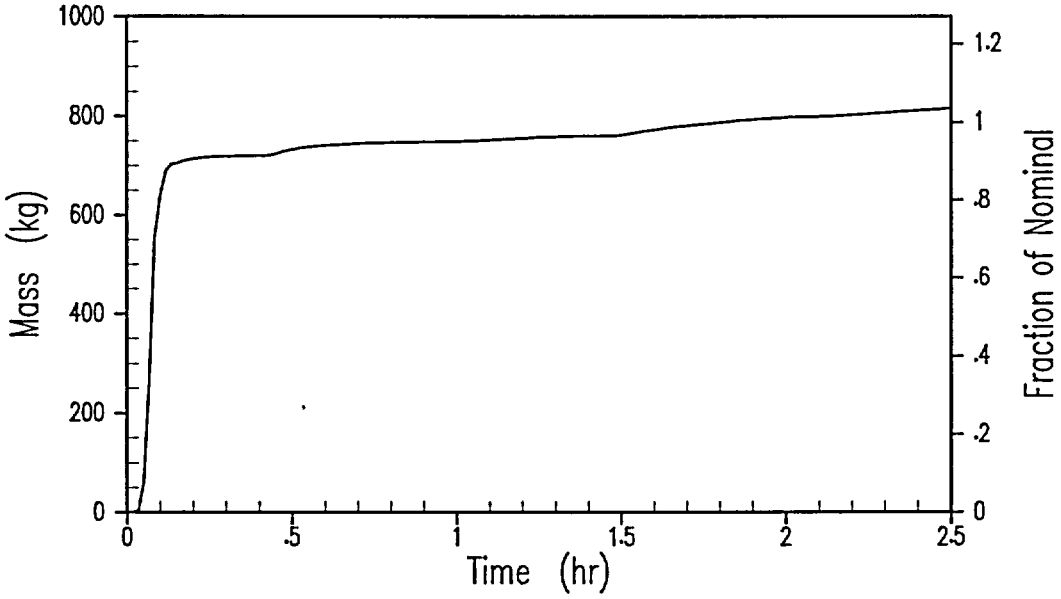
AP1000 Eq Surv Case CCI – Uncoolable Debris Bed in the Reactor Cavity
PXS Compt Gas Temperature



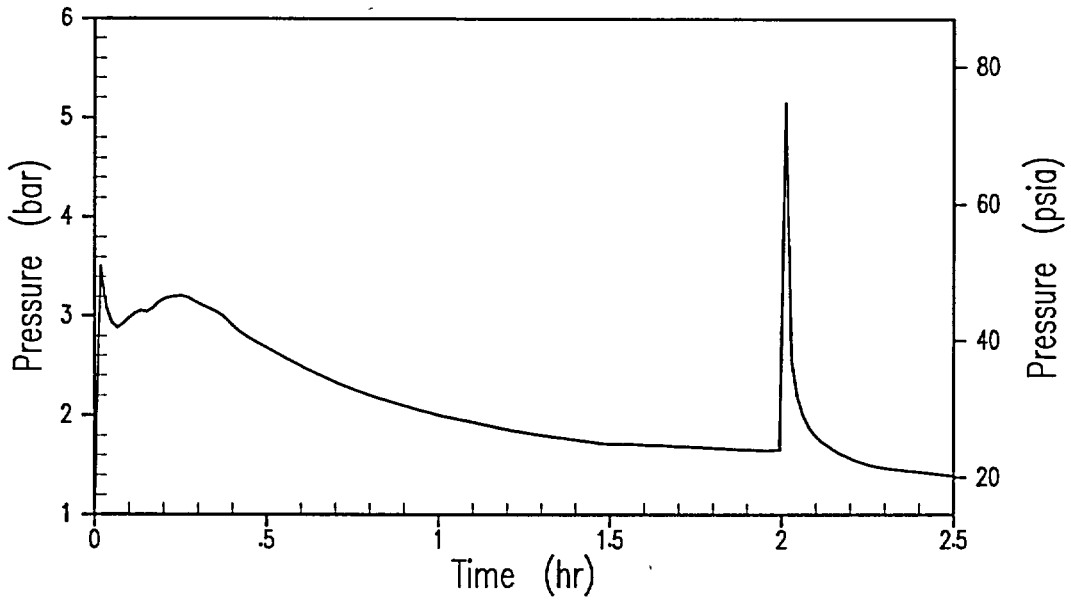
AP1000 Eq Surv Case GLOB – Global Burning of Hydrogen From 100% Clad Oxidation
Upper Plenum Gas Temperature



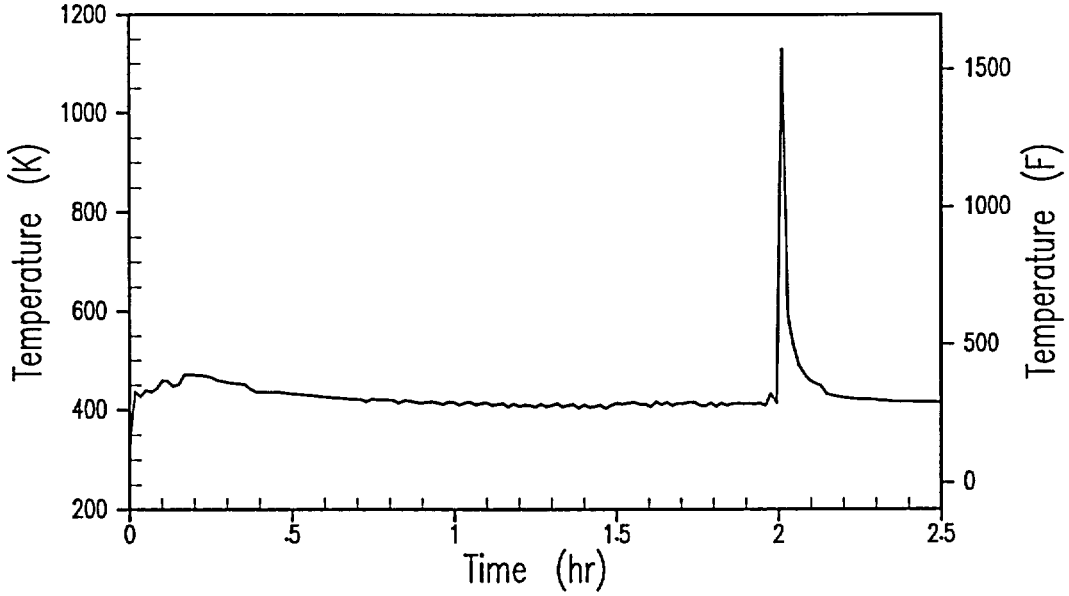
AP1000 Eq Surv Case GLOB – Global Burning of Hydrogen From 100% Clad Oxidation
In-Vessel Hydrogen Generation



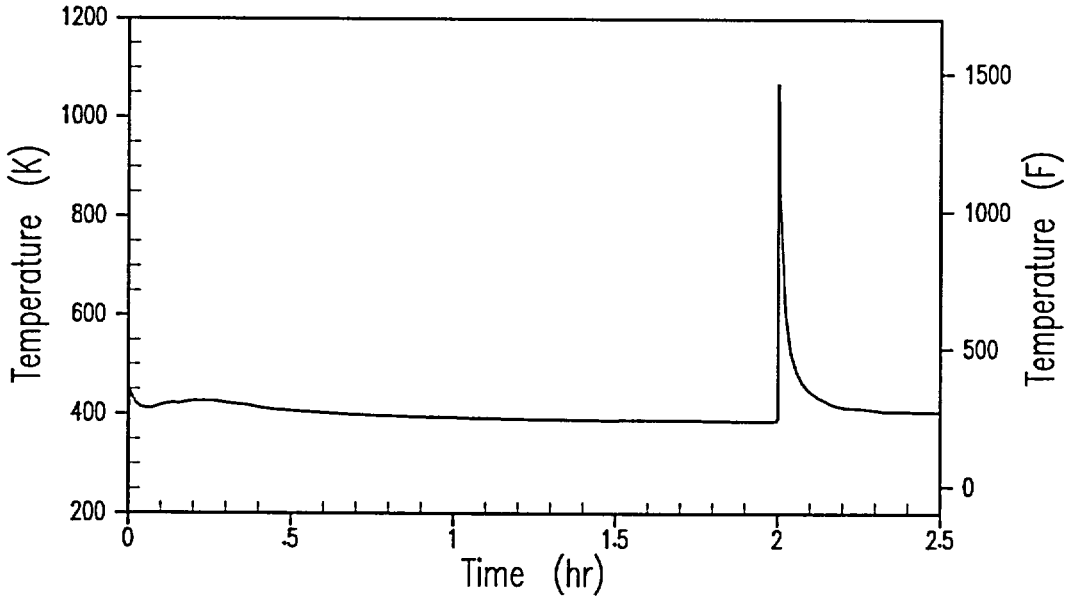
AP1000 Eq Surv Case GLOB – Global Burning of Hydrogen From 100% Clad Oxidation
Containment Upper Compt Pressure



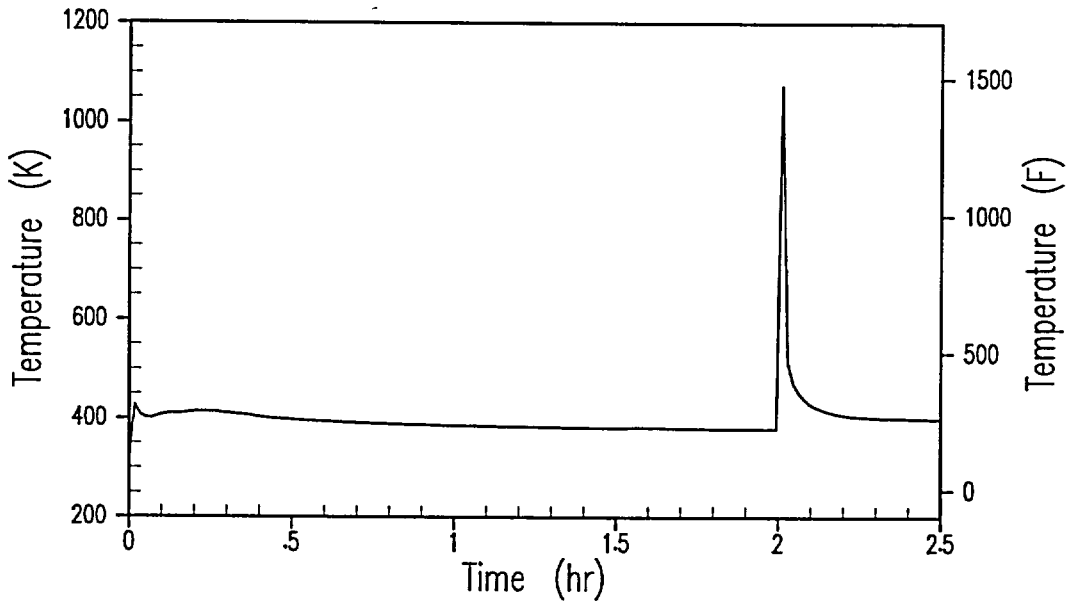
AP1000 Eq Surv Case GLOB – Global Burning of Hydrogen From 100% Clad Oxidation
SG Compt 1 Gas Temperature



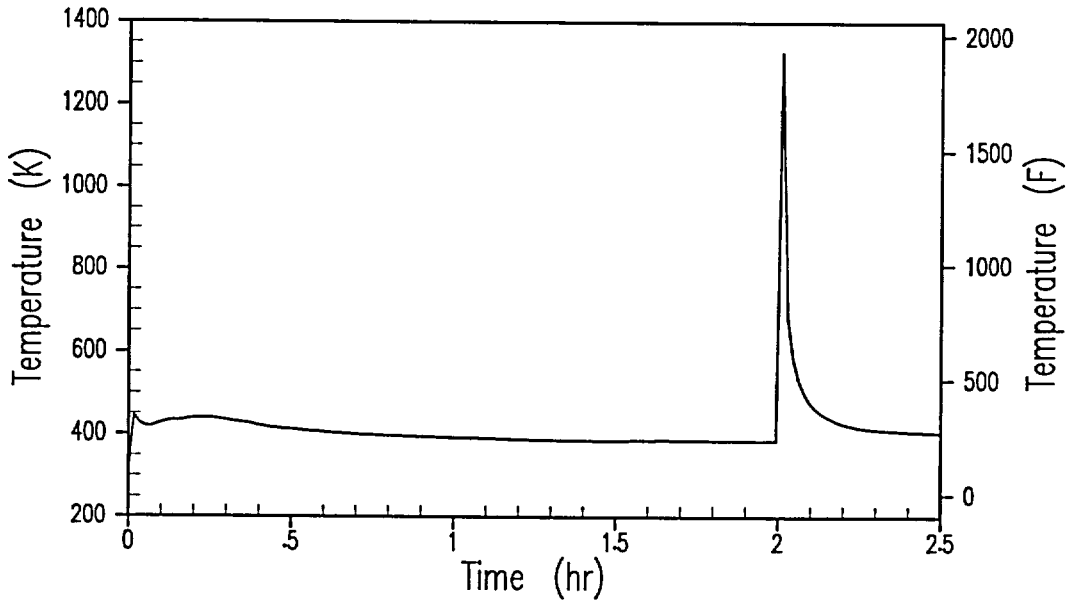
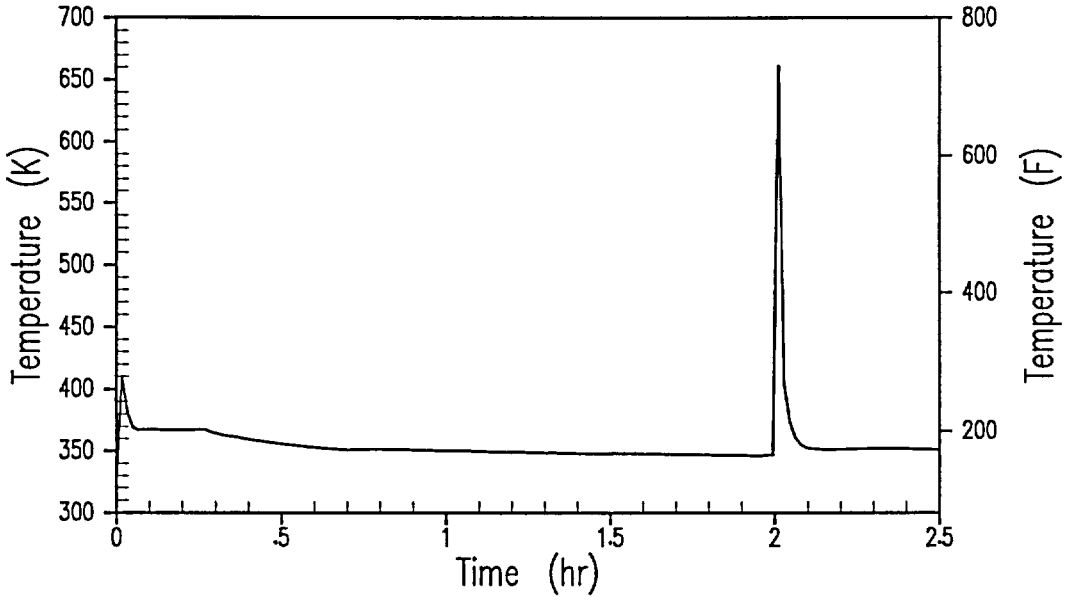
AP1000 Eq Surv Case GLOB – Global Burning of Hydrogen From 100% Clad Oxidation
SG Compt 2 Gas Temperature



AP1000 Eq Surv Case GLOB – Global Burning of Hydrogen From 100% Clad Oxidation
CMT Room Gas Temperature



AP1000 Eq Surv Case GLOB – Global Burning of Hydrogen From 100% Clad Oxidation
Upper Compt Gas Temperature



AP1000 Eq Surv Case GLOB – Global Burning of Hydrogen From 100% Clad Oxidation
PXS Compt Gas Temperature