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Docket No. 52-002

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Response to NRC Requests for Additional Information

Reference: Letter, Risk Assessment Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated October 30, 1991

Dear Sirs:

The Reference requested additional information for the NRC staff review of the Combustion Engineering Standard Safety Analysis Report - Design Certification (CESSAR-DC). Enclosure I to this letter provides our responses to a number of these questions including corresponding revisions to CESSAR-DC.

Should you have any questions on the enclosed material, please contact me or Mr. Stan Ritterbusch of my staff at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

C. B. Brinkman

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Acting Director
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gdh:
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cc: J. Trotter (EPRI)
T. Wambach (NRC)

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Enclosure I to
LD-92-023

RESPONSE TO NRC REQUESTS FOR ADDITIONAL INFORMATION
RISK ASSESSMENT BRANCH

Question 720.3

On page 8-30 of the CESSAR PRA, it is stated that, for a small LOCA with failure of SIS, the shutdown cooling system (SCS) can be used to provide injection for RCS inventory control, if the primary system can be depressurized below the SCS pump shut-off head by aggressive secondary cooldown. One of the success criteria for aggressive secondary cooldown is that all four safety injection tanks (SITs) inject borated water into the RCS during the primary-side depressurization. Examination of the fault trees developed for aggressive secondary cooldown during small LOCAs (page 6-862 of the PRA) revealed that the success criterion requiring injection of SIT water is not modeled. Please explain why or modify the fault trees.

Response 720.3

The aggressive cooldown models will be modified to reflect the appropriate SIT success criterion. This will not significantly affect the PRA results as failure of aggressive cooldown is dominated by the operator failing to initiate the cooldown in time.

Question 720.4

To succeed in long-term decay heat removal using the shutdown cooling system (SCS), the operator must properly align the flow path (taking suction from the hot legs) and start the relevant pumps. This operator action is modeled in the fault trees by the basic event, JSXOSDC (operator fails to establish shutdown cooling) and assigned a failure probability of 1.7E-06. Note, however, that the required operator actions differ somewhat if the SCS has been used for injection purposes (taking suction from the IRWST) prior to being used in the long term decay heat removal (compare, for example, sequence 3 and sequence 9 of the small LOCA event tree).

(a) Are these differences in the required operator's actions taken into consideration when quantifying the fault trees?

(b) Please explain how the human error probability of 1.7E-06 was obtained.

Response 720.4

(a) The differences in the operator actions were not taken into consideration when quantifying the fault trees. It was assumed that the operator would be working from procedures, would have good displays as to the system alignment and would have ample time in which to make the appropriate system alignments. Based on these assumptions, it was deemed that there was no significant difference between the required actions for initiating shutdown cooling from various configurations.

(b) The human error rate was obtained based on several HCR calculations. The basic assumptions in these calculations were that initiating shutdown cooling did not require a lot of time to accomplish and that there was a lot of time available.

C-E is currently updating the System 80+ PRA. As stated in the response to Questions 721.1 through 721.17, C-E is revising the calculation of human error rates as part of this update.

Question 720.10

Sequence #11 of the ATWS event tree involves a SGTR, failure of the safety injection system, with success of aggressive secondary cooldown and success of SCS (injection mode). This sequence is considered to lead directly to a "success" state without core damage. The success criteria for aggressive secondary cooldown, however, require that both steam generators and their ADVs be used in the RCS heat removal.

(a) What is your basis for assuming that the broken generator can be successfully used in aggressive cooldown? What are your estimates of the doses compared to Part 100 due to the radioactive nuclides transported to the secondary-side of the broken generator and released through the ADVs during this operation?

(b) How long does it take to reach the SCS entry conditions by employing aggressive secondary cooldown? For all other transients, it is said to take 6 to 8 hours. With no RCS inventory makeup and continuous leakage to the SG secondary-side , can core uncovery occur prior to reaching the SCS entry conditions?

(c) Why does the success of subsequent SCS injection by itself lead directly to a "success" state without taking credit for isolating the broken generator? Based on the capacity of a single SI pump (800)gpm), the IRWST (116,000 cubic feet) will be exhausted in less than 20 hours.

Response 720.10

(a) A SGTR does not physically preclude the use of the affected generator for heat removal. Although the affected generator is normally isolated following a SGTR, the affected generator can be steamed as a way of controlling excess inventory in the affected generator, and the isolation valves can be manually opened. It is felt that if the operator has the choice of steaming the affected generator or melting the core, the operator will steam the affected generator. No specific calculations were made as to the doses resulting from using the affected generator since no core damage is predicted.

(b) For a small LOCA or a SGTR, shutdown cooling entry conditions can be reached in 4 to 5 hours without violating the Technical Specification 100^{0} F/hour cooldown rate. If the Technical Specification cooldown rate is exceeded, shutdown cooling entry conditions can be reached much sooner. In analysis performed for CEN-239, "Depressurization and Decay Heat Removal - Response to NRC Questions, it was shown that for a System 80 plant, given failure of the HPSI system, SCS entry conditions could be reached and RHR injection initiated prior to core uncovery using aggressive secondary cooldown.

(c) With success of SCS injection, the RCS pressure has been reduced and heat removal in conjunction with pressure control is established.

The SGTR leak rate is significantly reduced from its initial value of about 600 gpm because of the reduced pressure differential between the SG and the RCS. Calculations performed for a System 80 plant under equivalent conditions indicated that it would take well in excess of 24 hours to deplete the equivalent of the inventory in the IRWST. This provides sufficient time to cool the plant down and terminate the leak.

C-E is currently updating the System 80+ PRA. Additional documentation on aggressive cooldown, particularly as it applies to SGTR, will be provided in the updated report.

Question 720.26

Although the event stuck-open primary safety valves (PSV) is modeled in the ATWS event tree, failure of the PSVs to open to relieve pressure is not modeled. Please explain why. Note that for an ATWS, several PSVs may have to open to prevent RCS for overpressurization.

Response 720.26

CE is currently updating the System 80+ PRA. The potential impact of failure of a PSV to open following an ATWS will be evaluated as part of this update.

Question 720.27

For an ATWS with consequential SGTR, a large leakage rate can be expected because of the high reactor power and high RCS pressure. It is thus more difficult to manually control the RCS pressure to stop the leakage and to bring the reactor to stable cold shutdown conditions. Examination of the human error probability, PPAX01BX (operator controls RCS pressure), however, revealed that the same failure probability of 1.0E-03 is used regardless of whether ATWS is involved. Please explain why.

Response 720.27

While the initial pressures and leakage rates for an SGTR following an ATWS are higher than those for standard SGTR, the process of controlling the RCS pressure to stop the leakage is assumed to be similar whether or not an ATWS was involved. In addition, the RCS boration and pressure control process for responding to the ATWS is complementary to the process for responding to the SGTR.

Question 720.31

ATWS sequence #19 involves an ATWS with consequential SGTR, failure of secondary-side cooling through an intact SG, but success of safety injection and late safety depressurization (bleed). The IRWST, in this case, serves as the heat sink for the "bleed" function and the water source for the safety injection. In view of the high reactor power and the continuous loss of RCS coolant through the ruptured tube, what will be the success criteria for cooling the IRWST? Will the IRWST be depleted, thereby leading to core damage? If not, why not?

Response 720.31

The success criteria used for IRWST cooling was that one of the four RHR or Containment Spray pumps must recirculate the IRWST inventory through its respective heat exchanger. This success criteria was based on normal feed and bleed heat loads. Evaluations of IRWST depletion for an SGTR indicated that the IRWST inventory will last well in excess of the 24 hour mission. The combination of an ATWS with a consequential SGTR and feed and bleed cooling was not, however, explicitly considered when setting the success criteria. C-E is currently updating the System 80+ PRA. The success criteria, timing, and logic for this sequence will be included as part of this update.

Question 720.50

Based on the seismic event tree shown in Figure 7.3-3, SIT injection (2/4) is required if a large LOCA is brought about by a seismic event. However, no seismic fault tree for the SIT injection can be found in the System 80+ PRA. Where, for example, is the seismic basic event, LTXZ (seismic failure of the accumulators shown in Table 7.3-2) used?

Response 720.50

The seismic fault tree for the SITs was deleted during an early phase of the analysis because the system is a passive system with no dependencies on power or cooling water, and the fragilities for the components in the model are all 2.5g or greater. C-E is currently updating the System 80+ PRA. The seismic model for the SITs will be reconstructed and included as part of the update. This should not impact the results.

Question 720.55

Examination of the seismic event tree for ATWS sequences revealed that no top event was allocated for "turbine trip" or "PSV Open" following the occurrence of an ATWS and loss of main feedwater. Justify omission of these events from your seismic core damage frequency estimate.

Response 720.55

Failure of the turbine to trip was assumed in all ATWS analyses. With respect to "Failure of the PSVs to open", see the response to Question 720.26.

Question 720.61

Please provide a list of all the human error probabilities used in quantifying the core damage frequencies attributable to tornado strike events and seismic events. How do these human error probabilities reflect the stressful circumstances encountered by the operators?

Response 720.61

All human error probabilities used in the System 80+ PRA are listed in Tables 5-6 and 5-7 of the System 80+ PRA Report, DCTR-RS-02 Rev. 0, January, 1991. No special differentiation was used for human errors for seismic and tornado strike events. As stated in the response to Questions 721.1 through 721.17, C-E is re-evaluating the human reliability analysis as part of the update of the System 80+ PRA.

Question 720.63

Please provide discussions regarding the modeling of electrical equipment (such as an electrical breaker) to account for relay chattering effect in fragility quantification. Provide discussion regarding accident sequences (such as loss of containment isolation function) that could result from relay chattering failure mode, and method of quantifying such failure modes, including human recovery actions, if any.

Response 720.63

System 80+ employs a primarily solid state protection system and component control system as opposed to the relay logic employed in previous generation C-E plants. However, it was conservatively assumed for this analysis that relays would be used in areas such as motor controllers. During a seismic event, the relay contact chatter failure mode was assumed to result in a momentary opening of the relay contact. The assumed result of this was a loss of control signal to the affected component. This analysis also used the "one fail- all fail" assumption. This meant that all relays were assumed to be affected by relay chatter. As a consequence of these assumptions, all safety equipment which needed actuation were assumed not to be actuated. The median fragility used for relay chatter failure was 1.0 g. At this g level, there would be little other significant damage and the transient would be essentially a loss of offsite power. The required safety system responses for this transient would be starting of the diesel generators and the actuation of the emergency feedwater system. The available response time would be on the order of one to two hours. Following the loss of control/actuation attributable to relay chatter, it was assumed that the operators would have to reset the relays and re-issue the actuation signals. The recovery action, ECHATTER, "Failure to recover bus chatter failure", was applied to all chatter related seismic cutsets. The value used for this element was 0.05 with an assigned error factor of 1.0

Question 720.85

Section 4.1.1 - The PRA states that " .. general aviation aircraft cannot damage equipment protected by structures of reinforced concrete with a minimum wall and roof thickness of structures greater than 18 inches." Is 18 inches the minimum wall and roof thickness for all System 80+ Category I structures? Is the statement true even if such walls and roofs are not specifically reinforced to withstand the impact due to general aviation aircraft?

Response 720.85

Eighteen inches is not the minimum wall and roof thickness for all System 80+™ Category I structures. The minimum wall and roof thickness currently being used for System 80+ Category I structures in areas subject to aircraft impact is 36 inches.

All System 80+™ Category I structures are designed to protect against design basis aircraft impacts and fire. System 80+™ Category I structures are designed to be fully capable of withstanding the effects of postulated aircraft impacts and fires without loss of safe shutdown capability and without causing a release of radioactivity which would exceed 10 CFR Part 100 dose guidelines.

Barrier thicknesses less than 18 inches could be used to protect against aircraft impact and fire provided sufficient justification is presented to support their design.

Question 720.86

Section 4.2.3 - Provide wind/tornado fragility descriptions for structures and outdoor equipment used to calculate the core damage frequency values given in Table B4.2.3-1.

Response 720.86

The structures for the nuclear annex and the structures housing the service water pumps are designed to withstand winds with speeds of up to 330 mph. Thus, these structures and the safety-related equipment within them were considered not to be susceptible to wind/tornado-induced failures. Equipment in other structures, such as the standby combustion turbine, were assumed to be unavailable due to wind/tornado-induced failures. Also, the potential for blockage of the service water intake structure due to tornado generated debris was modeled.

Question 720.87

Section 4.2.2.2 - The wind speed of 360 mph (Amendment H) is not consistent with number (330 mph) presented in Section 3.3.2.1 (Amendment 1). Clarify the discrepancy.

Response 720.87

The wind speed of 330 mph is the correct value. Section 4.2.2.2 will be revised to reflect this.

Question 720.90

Tables B4.3.1-1 and B4.3.1-2 - Why is the median capacity of seismically isolated 125 Vdc-480 Vac inverters presented in Table 4.3.1-2 greater than that of the battery chargers and inverters presented in Table 4.3.1-1? Note that the EPRI document recommends a value of 2.5.

Response 720.90

The median capacity for battery chargers and inverters presented in Table B4.3.1-1 is the value recommended by EPRI in the EPRI ALWR Key PRA Assumptions and Groundrules Document, Revision 0, July, 1989. Based on the preliminary Seismic PRA, the DC-AC inverters for the cavity flood valves were found to be key components for seismic severe accidents. Based on this, an interface requirement was established that these inverters be seismically isolated such that their effective median capacity would be reasonably close to that of the batteries, the other key component. It was felt that with reasonable anchorage and isolation, the median capacity of the inverters would approach that of motor control centers. Since 1989, EPRI has revised their estimate of the median capacity of battery chargers and inverters from 1.6g to 2.5g. C-E believes that a median capacity of 3.0g accurate if the inverters are seismically isolated.

Question 720.91

Table B4.3.1-2 - Describe how the median seismic capacity of the diesel generator building (Event EDGZBLDG) was derived. Describe how the median capacity of the seismically-induced large break LOCA (Event LBLOCA) was derived. Why is it 2.0g, whereas the piping capacity is 3.8g (Table B4.3.1-1)?

Response 720.91

The preliminary System 80+ Seismic PRA was performed by an Advanced Reactor Severe Accident Program (ARSAP) contractor and transmitted to C-E via the ARSAP Task Report RDC-69-89, "Interim Estimation of Core-Damage Frequency For CE System 80+ Due to Seismic Initiating Events". (Note: This report has been supplied to the NRC.) The median seismic capacities for the diesel generator building (Event EDGZBLDG) and seismically-induced Large break LOCAs (event LBLOCA) were provided in this report, but there was no description of how they were derived. The median seismic capacity for LBLOCA may be based on the EPRI estimate of the median seismic capacity of the reactor pressure vessel. C-E is currently updating the System 80+ PRA and will reassess the median capacities of these two events as part of the update.

Question 720.95

Describe to what extent the documentation and documentation process for the System 80+ PRA meets Appendix B requirements.

Response 720.95

The System 80+ PRA documentation and documentation process meets the intent of Appendix B. The System 80+ PRA Report, (DCTR-RS-02, Rev. 0, January, 1991) is the primary document for the System 80+ PRA. This document describes the methodology and computer codes used, states the analysis assumptions, presents the data used, the systems analysis models and results, and presents the final results. Sufficient information has been presented so that a skilled PRA practioneer should be able to reproduce the results. The PRA report was reviewed by the engineering and analysis groups in C-E.

Question 722.1

Please provide detailed drawings showing the geometry of the reactor cavity region including: floor area, volume, openings to other parts of the containment, thickness of the walls, details of the basemat (species of concrete to be specified, location and depths of sump(s), and existence of any encapsulated components within the basemat), and equipment/structures located in the cavity. Discuss whether the cavity wall supports the vessel load in such a way that damage to or erosion of the wall could result in relocation of the vessel, other RCS components, or any of the containment penetrations.

Response 722.1

A drawing of the reactor cavity is presented on the attached figure.

As shown on the figure, the reactor vessel is supported by columns which bear on concrete support corbels. These corbels project from the primary shield wall which has a minimum thickness of 6'-0". No scenarios exist which would result in erosion or damage of the concrete in this area. Therefore, structural integrity is maintained.

Question 722.2

Please provide the following additional information regarding the reactor cavity flood system: P&ID drawings showing number, type and location of valves; estimated flow rate for the system; and philosophy (and any operating procedures) regarding system actuation. Specifically address the time at which the system would be actuated relative to the time of core slump and vessel failure, and the plant parameters, when the system should be actuated (e.g., core exit thermocouples exceeding some prescribed value).

Response

A P&ID of the Cavity Flooding System (CFS) is shown in Figure 6.8-4 in chapter 6 of CESSAR-DC. The CFS has six motor operated, gate valves consisting of four holdup volume tank (HVT) spillway valves (SI-390, SI-391, SI-392, SI-393) and two reactor cavity (RC) spillway valves (SI-394, SI-395). Figure 6.8-2 illustrates the typical location of these valves. All CFS valves are located in the HVT so that these valves can be accessed for maintenance and inspection. The function of the CFS is to flood the reactor cavity with water during beyond design basis events.

The water flow rate through the CFS varies with time and is a function of the water height in the IRWST. Assuming a normal water level in the IRWST, the average cavity flooding flow rate will be approximately 6,000 gpm. The instantaneous flow rate rises to a maximum in the first several minutes following CFS actuation and then decreases to zero with time as the water levels in the IRWST, HVT and RC tend to equalize. Flooding time, initiated when there is a normal initial water level in the IRWST, is less than one hour. Over 100,000 gallons of water will have been delivered to the reactor cavity during this time frame.

The CFS is actuated manually by the operator. This allows flexibility in responding to changing plant conditions during a severe accident. Manual control of the CFS also gives the operator the option of terminating RC flooding, if necessary.

The general philosophy regarding the time at whic. the operatcr would actuate the CFS is that such actuation would occur as soon as possible so that the reactor cavity is flooded prior to the time of vessel failure. Typical plant conditions that would indicate the need for cavity flooding include (1) core exit temperature exceeding a threshold value and increasing, and/or (2) water level below the top of the core and decreasing with no inventory makeup expected for a sustained period of time. Specific details regarding actuation time and the parameters that will be used to determine the need for cavity flood would be developed as part of the Accident Management Guidance for severe accidents. This effort will employ NUMARC and EPRI recommendations for Accident Management Strategies.

Question 722.3

Pressurization of containment as a result of steam generation or ex-vessel steam explosion would appear to be a potential disadvantage of flooding the cavity prior to vessel failure in sequences in which sprays are unavailable. As evidence, the containment release characteristics presented in Table 9.3-1 indicate that dry cavity sequences (RCs 6.2 and 6.4) result in basemat melt through in about 180 or more hours, in contrast to wet cavity sequences resulting in containment overpressure at much earlier times. This would imply that more time is available in the case of dry cavity than in the case of a wet cavity before containment failure. If this is true, use of the cavity flood system only when the containment heat removal system is available would appear to offer risk reduction potential. Discuss the significance of this challenge, and whether any constraints on the use of the cavity flood system (such as manually flooding the cavity only when containment heat removal is available). Discuss the value of adding success a strategy/constraint recognizing the competing effects of delaying containment failure at the expense of reducing the fission product scrubbing and containment cooling afforded by a flooded cavity. (Also see Question 80.)

Response 722.3

The cavity flood system upon which this analysis was based was a design arrived at in the earlier stages of the System 80+ design. For this design, when the system was actuated, an initial volume of water was released to the cavity, and this volume was not replenished except by the containment sprays. It was also assumed that if the cavity was not fully flooded within a short time after vessel failure a coolable geometry for the corium could not be established. Thus, all dry cavity cases would lead to a late containment failure due to basemat meltthrough. On the other hand, if the cavity was flooded and a coolable geometry established for the corium, even with the containment heat removal unavailable, there was the potential for recovering containment heat removal in time to prevent containment failure.

The System 80+ Cavity Flood System design has significantly changed. The current Cavity Flood System design provides more operational flexibility. Thus, when containment heat removal is avvailable, the operator can immediately flood the cavity. However, if containment heat removal is not available, the operator can delay flooding the cavity until containment heat removal is restored and still have reasonable assurance that the cavity can be flooded and the corium will be coolable. C-E is currently updating the System 80+ PRA and will assess the impact of the new cavity flood system design on the issues discussed above.

Question 722.4

Provide a discussion of the power supply to the cavity flood system valves, the impact of station blackout or battery depletion on operation of the valves, and the modelling of failure of the operator to open these valves before the depletion of batteries (or other motive sources). Discuss whether (and how far) the IRWST would be drained if these valves or connected piping are damaged, how this would affect the core injection or containment spray capability, and whether the possibility is included in the Levels I and II analyses, especially for seismically initiated events. Discuss whether water in the cavity can be recycled to the IRWST or vessel, and whether spray can take suction from the cavity.

Response

The Cavity Flooding System (CFS) consists of six, motor operated valves. Four valves at the discharge of the Hold-up Volume Tank (HVT) spillway connect the Incontainment Refueling Water Storage Tank (IRWST) to the HVT. Two valves at the inlet to the reactor cavity (RC) spillways connect the HVT to the RC. All CFS valves are in the HVT and are connected to emergency battery power. Actuation of the CFS causes all of these valves to open simultaneously. The location of these valves in the HVT is illustrated in figure 6.8-2 of CESSAR-DC.

The plant has two redundant and independent Class 1E Auxiliary Power Systems, identified as Safety Divisions I and II. Two of the HVT spillway valves are powered from Division I buses and the remaining two from Division II buses. One of the RC spillway valves is powered from a Division I bus and the other from a Division II bus. Given a worst-case design bases accident (DBA) and a single failure, actuation of the CFS will result in at least two HVT spillway valves and one RC spillway valve opening and flooding the RC.

During a station blackout scenario, the CFS valves will be powered by emergency battery power. The need for dedicated battery power will be considered in the design in the context of minimum battery life and depletion due to the use of other equipment of this power source.

In the event that all CFS valves are inadvertently actuated or damaged so as to cause flooding of the RC during a worst-case DBA, sufficient water exists in the IRWST to flood the HVT and RC and still maintain sufficient static suction head for all Engineered Safety Features (ESF) system pumps such that the available net positive suction head ($NPSH_a$) exceeds that required ($NPSH_r$) by the pumps. All ESF system pumps take suction from the IRWST only.

In the event that only the RC spillway valves are inadvertently actuated or damaged so as to cause flooding of the RC from the HVT during a worst-case DBA, sufficient water exists in the IRWST to allow the HVT and RC to fill to an elevation high enough to allow water to replenish the IRWST through the IRWST spillways, accounting for water held up in containment from sprays. The minimum water level in the IRWST during this scenario is still sufficient to satisfy the ESF system pump NPSH requirements. Therefore, the CS pumps continue to take suction from the IRWST. The design only permits the CS pumps from taking suction from the IRWST, and there is no need to take suction from the RC.

Water flooded into the RC cannot be directly recycled to the IRWST. However, as explained above, there is sufficient water in the IRWST to allow IRWST replenishment during worst-case DBA's and satisfy ESF system pump suction requirements.

Question 722.5

Provide information regarding the water level that would be reached in the reactor cavity after actuation of the cavity flood system, and the extent to which the reactor vessel would be submerged. Discuss any strategies planned or under study for flooding the reactor vessel externally (prior to vessel failure) to arrest core damage in-vessel. Provide any supporting analyses of heat transfer from the vessel to the surrounding water, including the effect of thermal insulation on this heat transfer.

Response

The Cavity Flooding System (CFS) is designed to operate only during beyond design bases events. For those events, actuation of the CFS will flood the reactor cavity (RC) to a level below the reactor vessel (RV).

There are no strategies currently planned or under study for flooding the RV externally prior to vessel failure to arrest core damage in-vessel.

Question 722.6 :

Discuss whether the CE System 80+ design includes any drain connections from the reactor cavity or sump whose failure could result in drainage of the cavity water. Describe how the potential for failure of these connections (e.g., from seismic events or contact with core debris) and subsequent drainage of the cavity was assessed.

Response 722.6 :

The reactor cavity sump is located in the reactor cavity area and has no drain connections. Discharge piping from the sump pumps is routed up and out of the reactor cavity and will exit the containment in the containment penetration area above elevation area 91+0. The difference in elevation of the sump discharge piping containment penetration and the maximum expected wate level in the cavity in a severe accident prevents gravity draining of the cavity water.

Question 722.7

Please provide details on the geometry and orientation of the IRWST. Discuss whether failure of RHR piping (such as heat exchanger piping) during a seismic event can drain the IRWST, how an accident such as this would affect the core injection and containment spray capability, and how this failure potential was treated in the PRA.

Response 722.7

Details on the layout of the IRWST are provided in the System 80+^{TM} General Arrangement figures as listed below:

CESSAR-DC Figure Number	Drawing Description
Figure 1.2-4	Plan at Elevation 50+0
Figure 1.2-5A	Plan at Elevation 70+0
Figure 1.2-3	Section A-A
Figure 1.2-2	Section B-B

These figures were recently submitted via C-E letter LD-92-005, dated January 28,1992. For design basis accidents no scenarios exist whereby a failure of the RHR piping could affect the IRWST such that draining of the tank could occur.

For accidents beyond the design bases, it is possible that failure of piping upstream of the isolation valve could cause drainage of the IRWST.

C-E is currently updating the System 80+^{TM} PRA. The impact of this potential failure will be addressed in this revision.

Question 722.8

In Section 9.1.4.1, containment heat removal is defined to be available if containment spray is working or if the IRWST inventory is being cycled through the containment spray or RHR heat exchangers. If the containment spray headers are not available and containment heat removal is being accomplished by cycling IRWST inventory, please describe how energy released into the containment atmosphere from the reactor coolant system and from ex-vessel severe accident phenomena is transported to the IRWST and then removed from the containment.

Response 722.8

Prior to vessel failure, the reactor core transfers energy to the water in the reactor coolant system, generating steam which is discharged to containment, either directly or via the IRWST. After vessel failure, the corium transfers energy to water in the cavity, generating steam which is directly discharged to the upper containment. As the containment reaches saturation conditions, some of the steam condenses on the containment shell or other heat sinks inside containment. All condensate flow is directed to the holdup volume and then back into the IRWST. The containment spray pumps and RHR pumps can take suction from the IRWST and discharge flow back to the IRWST via either the containment spray heat exchangers or the RHR heat exchangers. The IRWST inventory being pumped through these heat exchangers transfers energy to the component cooling water. From the component cooling water, the energy is transferred to the service water and thence to the ultimate heat sink.

Question 722.9

Please describe how containment spray water, breakflow from the RCS, and condensed water on the containment wall flow will return to the reactor cavity or to the IRWST. If this return flow is split between the cavity and IRWST, please provide information regarding water levels in the reactor cavity and containment as a function of volume of water in containment, and any related flow split fractions. Discuss how long the spray system will continue to operate after the containment fails.

Response 722.9

Containment spray water, RCS breakflow, and condensed water on the containment wall will drain first into the holdup volume tank which is located adjacent to the IRWST. The water draining into the holdup volume tank is ultimately returned to the IRWST through spillways connecting the IRWST to the holdup volume tank once the water in the holdup volume tank reaches the spillway elevation. These spillways are always open and contain no valves. When the water level in the holdup volume tank reaches the inlet of the spillway, water flows by gravity from the holdup volume tank into the IRWST. The spillways are located at an elevation above the IRWST normal water level.

The return flow is not split between the reactor cavity and the IRWST. In order for cavity flooding to occur, manual actuation of spillway valves allows flow from the IRWST to the holdup volume tank, and from the holdup volume tank to the reactor cavity. These spillways rely on static head and gravity flow following valve actuation for communication of water between the IRWST, holdup volume tank and the reactor cavity.

This process, of providing water to the safety systems, continues until the system is removed from actuation. In the PRA analysis, once the containment fails, no credit is taken for continued operation of the system.

Question 722.10

Please describe the operating philosophy (and any operating procedures) regarding use of the containment spray system during severe accidents. Specifically address any provisions (and associated procedures) for: connecting the spray system to sources of water other than the IRWST (such as external sources of water), throttling containment spray flow rate in order to conserve IRWST inventory, and replenishing the water in the IRWST.

Response 722.10

Containment spray will be actuated on high containment pressure. Its function is to provide containment heat removal and pressure control and radioisotope scrubbing from the containment atmosphere. The general operating philosophy would be that once the spray system is actuated, it would remain operating as long as it is needed. The operator guidance is the same as CEN-152. Additional guidance may have to be developed for conditions where containment spray can be recovered late in a severe accident sequence. The containment spray system draws suction from the IRWST and discharges back to the IRWST via the holdup volume. Thus, containment spray flow does not have to be throttled to conserve IRWST inventory. No specific provisions have been made to connect the containment spray to external water sources. The IRWST can be connected to the Boric Acid Storage Tank (BAST) which can be used to provide additional inventory to the IRWST in the unlikely event that it is needed.

Question 722.11

Please specify the igniter system that will be utilized in the CE System 80+ design, and provide references for the evaluation of their effectiveness and reliability under accident conditions (such as moderate steam concentrations).

Response 722.11

The System 80+ Hydrogen Mitigation System (HMS) igniters are described in CESSAR-DC, Section 6.2.5. These igniters are of the glow plug design, with at least two igniters at each location. Igniters will be located globally with special consideration of areas where hydrogen may be produced (e.g., near the In-containment Refueling Water Storage Tank), or where hydrogen could accumulate. The two igniters at each location are powered from separate electrical divisions to provide redundancy and protecti· · against loss of electrical power in a division.

The reliability of the HMS is determined by the reliability of the igniters themselves and igniter electrical power sources.

- The igniter electrical power sources provide the maximum reliability and availability during all accident scenarios through redundancy and diversity. The igniters can receive power from offsite power, the Class 1E diesel generators, the Class 1E divisional batteries, or the diverse Alternate AC source (combustion turbine generator).

- The glow plug igniters are a proven design, used in the ice condenser PWRs. These igniters are of a simple design and are capable of remote testing during power operation. Evaluation of glow plug igniter reliability, and testing under accident conditions was performed as part of their qualification for the ice condenser PWRs. This qualification testing determined that the glow plug igniters effectively burn hydrogen at concentrations greater than 5%, and that 100% humidity or steam concentrations up to 40% do not hinder igniter performance. The HMS igniters will be comparable to those tested.

Question 722.12

It is claimed (page 9-73) that a small igniter failure probability (0.01) is used in the analysis since hydrogen igniters are available which are independent of station power. This value is very low and appears to have ignored several mechanisms which would tend to compromise igniter effectiveness. These include: early battery failure/ unavailability due to improper/missed surveillance; battery depletion (especially for late containment failure, in which the mission time may exceed the stated 40 hour life of the internal battery packs); failure of the operator to actuate the igniters; preexisting contamination of the passive igniter catalytic surface due to long-term exposure to containment environment; and failure of catalytic igniter assemblies to properly open (assuming they are sealed during normal operation). Also, moderate steam concentrations (20-30 percent), while not high enough to inert a hydrogen/steam/air mixture, can significantly shift the lower flammability limit. Please provide justification for the assumed igniter failure probability effectiveness in view of these factors. Provide an assessment of the effect of each factor in your response.

Response 722.12

The System 80+ hydrogen igniter design has been revised since the original PRA analysis which is summarized in CESSAR-DC, Appendix B. The igniters are of the glow plug design as described below. The PRA analysis in CESSAR-DC, Appendix B will be updated to address the revised glow plug igniter design.

The hydrogen igniter availability is maximized by the igniter design and the reliability of the electrical power sources. The glow plug igniters are of a proven design, which have undergone qualification testing and are currently used in the ice condenser PWRs. The availability of the igniter electrical power sources provides maximum reliability and availability during all accident scenarios through redundancy and diversity. The igniters can receive power from offsite power, the Class 1E diesel generators, the Class 1E divisional batteries, or the diverse Alternate AC (AAC) source (combustion turbine generator).

Two igniters are placed at each location where hydrogen is produced or could accumulate. The two igniters at each location are powered from separate electrical divisions to provide redundancy and protection against loss of electrical power in a division.

Response 722.12 (Cont'd)

The specific factors listed above are addressed as follows:

1. Early battery failure/unavailability due to improper/missed surveillance:

   The Class 1E divisional batteries provide power to the igniters in the highly unlikely event of a loss of all AC power, including the diverse AAC source. These batteries provide the most reliable power source available with at least eight hours of capacity following the loss of all AC power.

2. Battery depletion (especially for late containment failure, in which the mission time may exceed the stated 40 hour life of the internal battery packs):

   For the station blackout scenario (loss of offsite power and failure of both diesel generators to start), the AAC source provides a diverse source of electrical power for the igniters. Upon loss of all AC power, the Class 1E batteries are the most reliable power sources available, and provide at least eight hours of capacity following the loss of all AC power. By the time the batteries are depleted, the containment is anticipated to be inerted with steam which will preclude accumulation of detonable hydrogen concentrations.

3. Failure of the operator to actuate the igniters:

   Igniter actuation and operation will be addressed in the detailed Severe Accident Management Procedures prepared by the owner/operator using the most current knowledge available at that time. Failure to actuate the igniters would mean either that the operators were not following the Severe Accident Management Procedures, or that they did not realize that the reactor core was being uncovered, both of which are highly unlikely.

4. Preexisting contamination of the passive igniter catalytic surface due to long-term exposure to containment environment:

   This factor does not apply to the glow plug igniter design.

5. Failure of catalytic igniter assemblies to properly open (assuming they are sealed during normal operation):

   This factor does not apply to the glow plug igniter design.

Response 722.12 (Cont'd)

6. Moderate steam concentrations (20-30 percent), while not high enough to inert a hydrogen/steam/air mixture, can significantly shift the lower flammability limit:

   Evaluation of glow plug igniter performance under accident conditions was performed as part of the qualification testing for the ice condenser PWRs. This qualification testing determined that the glow plug igniters effectively burn hydrogen at concentrations greater than 5%, and that 100% humidity or steam concentrations up to 40% do not hinder igniter performance. The System 80+ igniters will be comparable to those tested.

Question 722.13

Describe the locations of the hydrogen igniters and the philosophy and procedures for activating the system. Specifically address any special procedures for securing the igniter system upon loss of station power, and for reactivating the igniter system following recovery of power (provided, of course, that they are remote-manually controlled).

Response 722.13

The evaluation to determine hydrogen igniter locations, based on engineering judgement, is currently in progress. Conditions to be considered are hydrogen sources, equipment locations, and areas where hydrogen could accumulate (i.e., dead-ended compartments, upper portions of containment, etc.). The results of this evaluation are anticipated to be included in a future CESSAR-DC amendment. See response to NRC RAIs 730.7 and 722.11.

The hydrogen igniters are remote-manually controlled (See CESSAR-DC, Section 6.2.5). Igniter actuation and operation will be addressed in the detailed Severe Accident Management Procedures as indicated in the response to RAI 722.12(3).

Upon loss of offsite power and failure of both diesel generators to start, the igniters can receive power from the diverse Alternate AC (AAC) source (combustion turbine generator). Upon additional loss of the AAC source, the Class 1E divisional batteries can provide power to the igniters until depleted.

In the highly unlikely event of loss of offsite power, both diesel generators, and the AAC source, along with battery depletion, the containment is anticipated to be inerted with steam. When electrical power is restored, the Severe Accident Management Procedures will address activating the igniters and controlling the rate of containment de-inerting with containment spray to preclude achieving detonable hydrogen concentrations.

Question 722.14

Please provide justification for not addressing in the PRA the potential for either global or local detonations. Include in your response a description and assessment of: the transport and mixing of gases from the reactor cavity and IRWST to other regions within containment, for sequences with and without containment sprays available; areas in containment where detonable concentrations of combustible gases can form (during sequences in which igniters are unavailable); the potential for detonations in these regions; and the impact of such detonations on structures and equipment. Also discuss any credit taken in the MAAP analyses for recombination of hydrogen in the reactor cavity or other areas in containment.

Response 722.14

C-E is currently updating the System 80+ PRA. As part of this update, the potential for hydrogen detonations will be re-evaluated and incorporated in the PRA as appropriate. The System 80+ containment has a large, open, unobstructed upper containment area. Based on a review of other PRAs, it is felt that hydrogen detonation will not be a problem for System 80+.

QUESTION 722.15

Please provide the following additional information regarding the reactor depressurization system: operating philosophy for use of the system under accident conditions; details regarding the power supply for the depressurization valves, including a description of other loads carried by the batteries; a description of any procedures/provisions to assure the availability of the batteries when it is decided by operators to open these valves at the later stages of a severe accident (such as after a prolonged operation of turbine driven AFW pumps); and a description of any procedures/provisions to connect the valves to external sources of motive power. Discuss how the failure of the operator to open these valves before battery depletion was treated in the evaluation of unavailability of this system during high pressure sequences.

RESPONSE TO 722.15

Operating Philosophy

The Rapid Depressurization System (RDS) has been designed to permit a rapid depressurization of the the Reactor Coolant System (RCS). The system operates in conjunction with the Safety Injection System (SIS) for feed and bleed cooling of the RCS as a last resort for the beyond design basis event of total loss of feedwater (TLOFW). The RDS is a manual safety-grade means of quickly depressurizing the RCS when normal and emergency feedwater are unavailable for RCS heat removal from the steam generators.

A TLOFW, if not corrected, prevents the steam generators from performing the RCS heat removal function. The operator actions for a TLOFW are directed at determining the cause of the TLOFW and regaining and establishing a feedwater source to one of the two steam generators. If it is not possible to restore feedwater flow to one of the two steam generators, operator actions are then directed at establishing feed and bleed (feed and bleed is used as a last-resort method of core cooling if steam generator heat removal is no longer adequate). All safety functions will be monitored to assure public safety, or to detect changes in plant conditions.

Once the operator has successfully established feed and bleed, efforts to restore steam generator heat removal capability will continue. When normal RCS and core heat removal are re-established via one of the two steam generators, then feed and bleed will be terminated. This will re-establish the normal mode of RCS heat removal.

If normal RCS and core heat removal cannot be established, the operator will have to perform a cooldown to the shutdown cooling system entry conditions using the RDS and the SIS.

Power to SDS Valves

As stated in CESSAR-DC Section 6.7.1.2.C.12 the power supply for each RDS valve is from a DC bus. The power is connected such that in the case of a loss of both sources of offsite power, both EDGs, the combustion turbine, and the loss of one battery bank, a RDS bleed path can be established. Each DC load group, as stated in CESSAR-DC Section 8.3.2.1.2.1.1, is provided with a separate and independent 125 volt battery charger. The battery chargers are powered from Division I and II of the Class 1E Auxiliary Power Systems. Normally, each battery charger supplies the loads to its associated distribution center while maintaining a float charge on its associated battery. The Class 1E DC loads have an operating voltage range of 105 to 140 volts. The minimum battery discharge voltage is 105 volts. As stated in CESSAR-DC Section 8.3.2.1.2.1.2 each battery is sized to supply the continuous emergency load of its own load group for a period of 4 hours.

In addition, the batteries provide a station blackout coping capability which, assuming manual load shedding or the use of load management programs, exceeds 4 hours and, as a minimum, permits operating the instrumentation and control loads associated with the turbine-driven emergency feedwater pumps for 8 hours.

As stated in CESSAR-DC Section 8.3.1.1.2.2 the 480 volt Class 1E Auxiliary Power System receives its power from the 4160 Volt Class 1E Auxiliary Power System. Further, as stated in CESSAR-DC Section 8.3.1.1.2.1, on a loss of normal power, emergency power is provided to each of the redundant 4160 volt Class 1E Auxiliary Power System Divisions by two (one per division) separate and completely independent emergency diesel generators (EDGs). In the event a diesel generator is out of service or fails, the Alternate AC Source can be aligned to provide emergency power to either Class 1E Auxiliary Power System Division.

Typical loads carried by the Class 1E DC Vital Power System are given in CESSAR-DC Table 8.3.2-4. Generally, the batteries are supplied with a float charge by their respective battery chargers and do not carry any loads. However, a description of loads that might be carried by the batteries during a station blackout are also listed in Table 8.3.2-4.

Battery Availability

Battery availability procedures/provisions will be such that during a station blackout manual load shedding will be used in conjunction with load management programs to extend battery life. Further, procedures/provisions will consider the effect of battery depletion on the operability of the RDS valves.

External Power Source

Normally, the batteries are always available to provide power to the RDS valves. In the unlikely event of a station blackout in conjunction with a TLOFW, the Alternate AC Source (combustion turbine) can be manually aligned to provide power to one permanent non-safety bus and one safety bus. This will then provide power to the 480 volt Class 1E Auxiliary Power System, which, will in turn, provide power to one set of batteries through their respective battery chargers. The batteries provide a standby power source to operate the RDS valves. If a station blackout and a TLOFW occurred simultaneously, the batteries would be required to provide power for approximately the first ten minutes in order to operate the RDS valves. This will allow the

Alternate AC Source to come up to power and be manually aligned to the appropriate safety bus to provide power to one train of RDS valves.

System 80+ does not need sources of motive power to operate the RDS valves other than those described in CESSAR-DC.

Unavailability

In the PRA evaluation of high pressure sequences, the failure of the operator to open the RDS valves before battery depletion was treated as a failure to depressurize as stated in sub-sections of CESSAR-DC Appendix B dealing with high pressure sequences.

QUESTION 722.16

Please provide an assessment of the influence of containment pressure on the operability of the reactor depressurization valves. Discuss whether the valves are subject to reclosing upon high containment pressure, or battery depletion.

RESPONSE TO 722.16

As described in CESSAR-DC Section 6.7.2.2.2 the Rapid Depressurization System (RDS) valves are motor-operated. The operators are sized for the maximum expected differential pressure across the valve. Therefore, the valves are designed to operate under the expected range of containment conditions.

For the "beyond design basis" event of a Total Loss of Feedwater, the RDS valves will not be subject to reclosing on high containment pressure. Containment pressure is not expected to be significantly affected during RDS operation since the RDS flow discharges directly to a sparger network in the In-Containment Refueling Water Storage Tank (IRWST) and not the containment atmosphere. The IRWST is designed with the capability of condensing the RDS flow. The IRWST is cooled by the SCS or the CSS heat exchangers during flow from the RDS.

The RDS valves will not be subject to automatic reclosing upon battery depletion because the valves fail as-is. The RDS valves are powered from an extremely reliable set of power sources. Loss of all motive power is not considered a credible event. As described in the response to RAI 722.15 the batteries provide a standby power source to the RDS valves. Normally, the RDS valves are powered from the DC bus. The DC bus is supplied power from the 480 volt Class 1E Auxiliary Power System which receives its power from the 4160 volt Class 1E Auxiliary Power System. Upon a loss of normal power, the 4160 volt Class 1E Auxiliary Power System receives power from the Emergency Diesel Generators which, in turn, provide power to the 480 volt Class 1E Auxiliary Power System which provides power to the DC bus through battery chargers.

During a station blackout and a total loss of feedwater event the batteries will be required to provide power to the RDS valves for approximately the first ten minutes of the event. This will allow time for the Alternate AC Source to come up to speed and be manually aligned to provide power to one 4160 volt Class 1E Auxiliary Power System, which through the above distribution system, provides power to the DC bus and charges the batteries.

Question 722.17

The System 80+ containment is substantially different from the containments considered in the NUREG-1150 study in terms of shape materials and design pressures. Yet, the uncertainty distribution for the containment strength of the System 80+ design was obtained by fitting to the distributions developed in the NUREG-1150 study. Experience in containment overpressure evaluations on several containments shows that pressure capacities and failure modes are very plant specific. In this regard, please justify the applicability of containment ultimate pressure capacity estimates and uncertainty distributions for the NUREG-1150 plants to the System 80+ design considering the radically different containment designs.

Response 722.17

A distribution of containment failure probability versus containment pressure, the uncertainty distribution for containment strength, is needed for the level 2 PRA analyses. At the time of the analyses, the System 80+ containment design pressure and ultimate capacity had been established, but there was not sufficient design detail to determine the plant specific uncertainty distribution for containment strength. An examination of the various uncertainty distributions for containment strength developed in the NUREG-1150 study indicated that the distributional shapes were reasonably similar despite the containment design differences. It was assumed that a curve fit to these distributions and normalized to the calculated ultimate capacity for the System 80+ containment would provide a reasonable estimate of the uncertainty distribution for the System 80+ containment strength. A least squares fit curve was developed to obtain the best possible fit to all of the NUREG-1150 curves. This fitted curve was constrained to yield a failure probability of 50% at the ultimate capacity pressure. This is consistent with the definition of the code calculated ultimate capacity. Because containments are pressure tested to the full design pressure, it was assumed that the probability of containment failure at or below the design pressure was 0. Therefore, the fitted curve was further constrained to yield a failure probability of 0 at the design pressure. It is agreed that containment pressure capacities and failure modes are plant specific. The uncertainty distribution for the System 80+ containment will be confirmed during the detailed design phase as part of the Reliability Assurance Program.

Question 722.18

Describe the structural analyses performed to assess the potential for failure of equipment hatches, personnel airlocks, electrical and piping penetration assemblies, and seals. Provide an estimate of the pressure capacity for each of these components.

Response 722.18

Personnel airlocks, equipment hatches, electrical and piping penetration assemblies and seals are items that will be provided by equipment vendors via procurement design specifications. The design for these items will be specified such that the structural performance meets or exceeds that of the steel containment vessel. The specifications will outline the details for structural design and manufacture using design information for the required pressure capacity when this information is more defined, i.e., detailed design stage.

QUESTION 722.19

Discuss how the effect of temperature in the containment during severe accidents was accounted for in the evaluation of ultimate pressure capacity for the containment boundary and key penetrations. Provide justification for not addressing temperature induced failure in the PRA.

RESPONSE 722.19

The effect of temperature in the ultimate capacity analysis was accounted for by determining the containment material properties, i.e., yield strength, modulus of elasticity, etc., for the SA 537 Class 2 steel vessel at the design temperature of 290 degree F. C-E is currently updating the System 80+ PRA. This revised PRA would employ containment ultimate pressure capacity that is calculated on the basis of containment temperatures realized during severe accident scenarios to quantify the containment failure times and fission product release fractions.

Question 722.20

The mathematical procedure used to fit the NUREG-1150 data is not a rigorous least squares fitting of data in that the parameters in equations 5.2-8 and 5.2-9 were adjusted to fit the presumed distribution (i.e., the median and zero probability point), separately from the least fitting procedure. Please comment on this.

Response 722.20

See the response to Question 722.17.

Question 722.21

Please describe the technical basis for assumptions in the PRA concerning the likely locations of containment failure, and associated containment leak areas. Discuss whether the probability of the various failure locations/areas would realistically be dependent on containment challenge (e.g. rapid versus gradual overpressurization). Discuss how a catastrophic failure of the SCV will be avoided when the internal pressure approaches the ultimate capacity.

Response 722.21

C-E concurs that the probability of the various failure locations/areas is to an extent dependent on the type of containment challenge. Failure locations and areas are also dependent on the specific as procured details of the containment penetrations. This information will not be available until the detailed design phase. For the System 80+ PRA at this stage, very conservative assumptios were made regarding the specific location of containment failures. For overpressure failures, the failure was considered to be catastrophic, and leak size of several square feet was assumed. With the exception of the interfacing systems LOCA, all failures were assumed to occur directly to the atmosphere with no deposition of the fission products. When making CRAC2 runs to determine doses at 0.5 miles, high pressure releases were assumed to occur at the top of containment and low pressure releases were assumed to occur at ground level. The primary reason for this is that these are the only two release locations permitted by CRAC2. A more extensive evaluation of containment failure modes and locations will be included in the PRA as it is updated during the detailed design.

QUESTION 722.22

Please provide MAAP input listings for a representative set of those accident scenarios used to develop the source term information for each release category. For several accident sequences which have relatively high frequencies of occurrence, provide MAAP output (in graphical form) showing: containment pressure (including partial pressures of steam and non-condensible gas); temperature of water and air space; distribution of core material and non-condensible gases in each compartment; mass of water in the IRWST and cavity; heat loss through the containment wall; fission product release fractions; and erosion depth of the basemat and the cavity wall. Include a basemat melt-through sequence (i.e., a dry cavity case) in these outputs.

RESPONSE 722.22

A listing of the MAAP Parameter File which contains the plant design and operations data for the System 80+ design is provided in Attachment 1. The parameter file along with specific case data file are employed to perform MAAP analyses for specific accident scenarios used to develop the source term information. Case data file listings for a representative set of accident scenarios are provided in Attachment 2. These accident scenarios consist of the following Release Classes (RCs):

RC 2.2 -- Small LOCA (0.02 sq ft) in hot leg with no safety injection and no emergency feedwater; containment isolation failure at the start of the transient.

RC 2.4 -- Loss of Offsite Power transient with diesel generators available.

RC 3.1 -- Total Loss of Feedwater transient with manual feed and bleed cooling of the RCS. Containment heat removal is assumed to be unavailable.

RC 4.1 -- Large hot leg LOCA coincident with failure of safety injection.

RC 5.1 -- Station Blackout with battery depletion at 8 hours.

RC 6.2 -- Large cold leg LOCA with coincident failure of safety injection; no cavity flooding (dry cavity).

RC 7.1 -- Station Blackout with battery depletion at 8 hours and late recovery of power and containment heat removal.

Readily available output parameters for the above accident scenarios are provided in Attachment 3. These include the containment pressure plot, tabular data for fission product release fractions, height of water level in the cavity and/or the IRWST, concrete erosion of the basemat, and mass of hydrogen generated.

The proprietary versions of Attachments 1 and 2 are being submitted under a separate cover letter.

ATTACHMENT 1

LISTING OF MAAP PARAMETER FILE FOR THE SYSTEM 80+ DESIGN

THE FOLLOWING 16 PAGES CONTAIN COMBUSTION ENGINEERING PROPRIETARY INFORMATION

ATTACHMENT 2

LISTING OF CASE DATA FILE FOR REPRESENTATIVE ACCIDENT SCENARIOS

THE FOLLOWING 11 PAGES CONTAIN COMBUSTION ENGINEERING PROPRIETARY INFORMATION

ATTACHMENT 3

MAAP OUTPUT FOR REPRESENTATIVE ACCIDENT SCENARIOS

LEAKED FROM CONTMT:	GAS MASS(LB)	AEROSOL MASS(LB)	RELEASE FRACTION
NOBL. IN	2.9592E+02	7.6472E-03	1.8849E-01
CSI	-7.2752E-23	1.4732E-02	1.0943E-04
TEO2	0.0000E+00	0.0000E+00	0.0000E+00
SRO	0.0000E+00	8.5466E-05	2.9258E-07
MOO2	0.0000E+00	1.0109E-02	8.2970E-06
CSOH	2.3102E-11	8.6979E-02	1.1136E-04
BAO	0.0000E+00	1.3530E-03	3.0846E-06
LA2O3	0.0000E+00	3.9238E-05	1.6830E-08
CEO2	0.0000E+00	3.2091E-04	3.3955E-07
SB	2.0973E-12	3.3796E-04	3.6815E-05
TE2	0.0000E+00	0.0000E+00	0.0000E+00
UO2. ACT	0.0000E+00	0.0000E+00	0.0000E+00

RELEASE CLASS 2.2

Case 9

t = 24 hours

NO. ITERATIONS IN PRIM SYS TRANSPORT MODEL: 7 IN CONTMT GAS TRANSPORT MODEL: 1

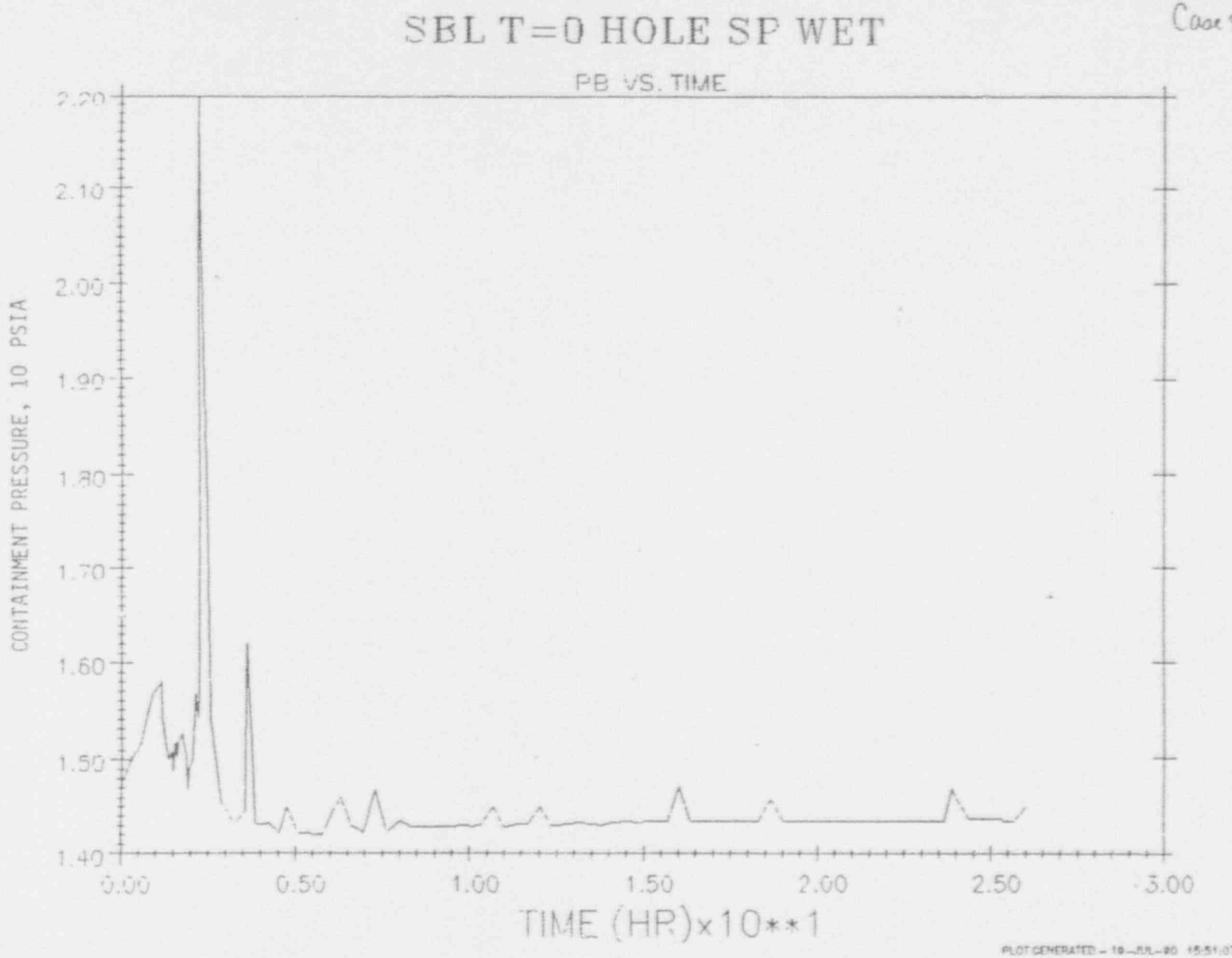
FISSION PRODUCT MASS BALANCES:

BALANCE 1 = TOTAL MASS AS USED BY MAAP
BALANCE 2 = IN FUEL + INTEGRATED RELEASE FROM FUEL

	NOBLES	CSI	TEO2	SRO	MOO2	CSOH
BAL. 1:	712.10120	61.06284	0.00000	133.30383	556.01569	354.29037
BAL. 2:	712.10010	61.06260	0.00000	133.28671	555.92700	354.29022
INITIAL:	712.09998	61.06259	-0.45515	132.50000	552.65625	354.29031
CORE:	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
CORIUM:	0.00000	0.00000	0.00000	132.69112	514.14886	0.00000
PS:	10.56961	41.33510	0.00000	0.50412	28.85357	240.39081
CONT:	701.53156	19.72774	0.00000	0.10859	13.01324	113.89957
INV REL:	712.10010	61.06260	0.00000	0.59560	41.77815	354.29022
EXV REL:	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000

	BAO	LA2O3	CEO2	SB	TE2	UO2
BAL. 1:	200.33818	1063.83435	431.24356	4.17682	61.53891	113399.40625
BAL. 2:	200.09187	1063.83057	431.24207	4.17681	61.53891	113399.40625
INITIAL:	198.95653	1057.52490	428.68570	4.16393	61.17422	113392.09375
CORE:	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
CORIUM:	191.07890	1063.63000	430.17029	2.38530	61.53891	113399.40625
PS:	7.54105	0.15566	0.69993	1.44551	0.00000	0.00000
CONT:	1.71822	0.04864	0.37335	0.34601	0.00000	0.00000
INV REL:	9.01296	0.20060	1.07179	1.79151	0.00000	0.00000
EXV REL:	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000

RELEASE CLASS 2.2
Case 9



UO2.ACT GAS	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
AERO	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	-0.0000E+00
DEP AERO/GAS	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
IN CORIUM	0.0000E+00	0.0000E+00	2.5001E+05				

LEAKED FROM CONTMT:	GAS MASS(LB)	AEROSOL MASS(LB)	RELEASE FRACTION
NOBL.IN	1.5420E+03	2.0707E-01	9.8221E-01
CSI	-6.0273E-24	4.7803E+00	3.5510E-02
TEO2	0.0000E+00	0.0000E+00	0.0000E+00
SRO	0.0000E+00	1.9777E-02	6.7704E-05
MOO2	0.0000E+00	4.1173E+00	3.3793E-03
CSOH	1.3366E-06	2.0614E+01	2.6391E-02
BAO	0.0000E+00	1.8091E-01	4.1246E-04
LA2O3	0.0000E+00	2.2504E-03	9.6523E-07
CEO2	0.0000E+00	2.5711E-02	2.7205E-05
SB	8.6976E-08	7.9807E-01	8.6936E-02
TE2	0.0000E+00	0.0000E+00	0.0000E+00
UO2.ACT	0.0000E+00	0.0000E+00	0.0000E+00

RELEASE CLASS 2.4

Case 10
t=36 hours

NO. ITERATIONS IN PRIM SYS TRANSPORT MODEL: 2 IN CONTMT GAS TRANSPORT MODEL: 1

FISSION PRODUCT MASS BALANCES:

BALANCE 1 = TOTAL MASS AS USED BY MAAP
BALANCE 2 = IN FUEL + INTEGRATED RELEASE FROM FUEL

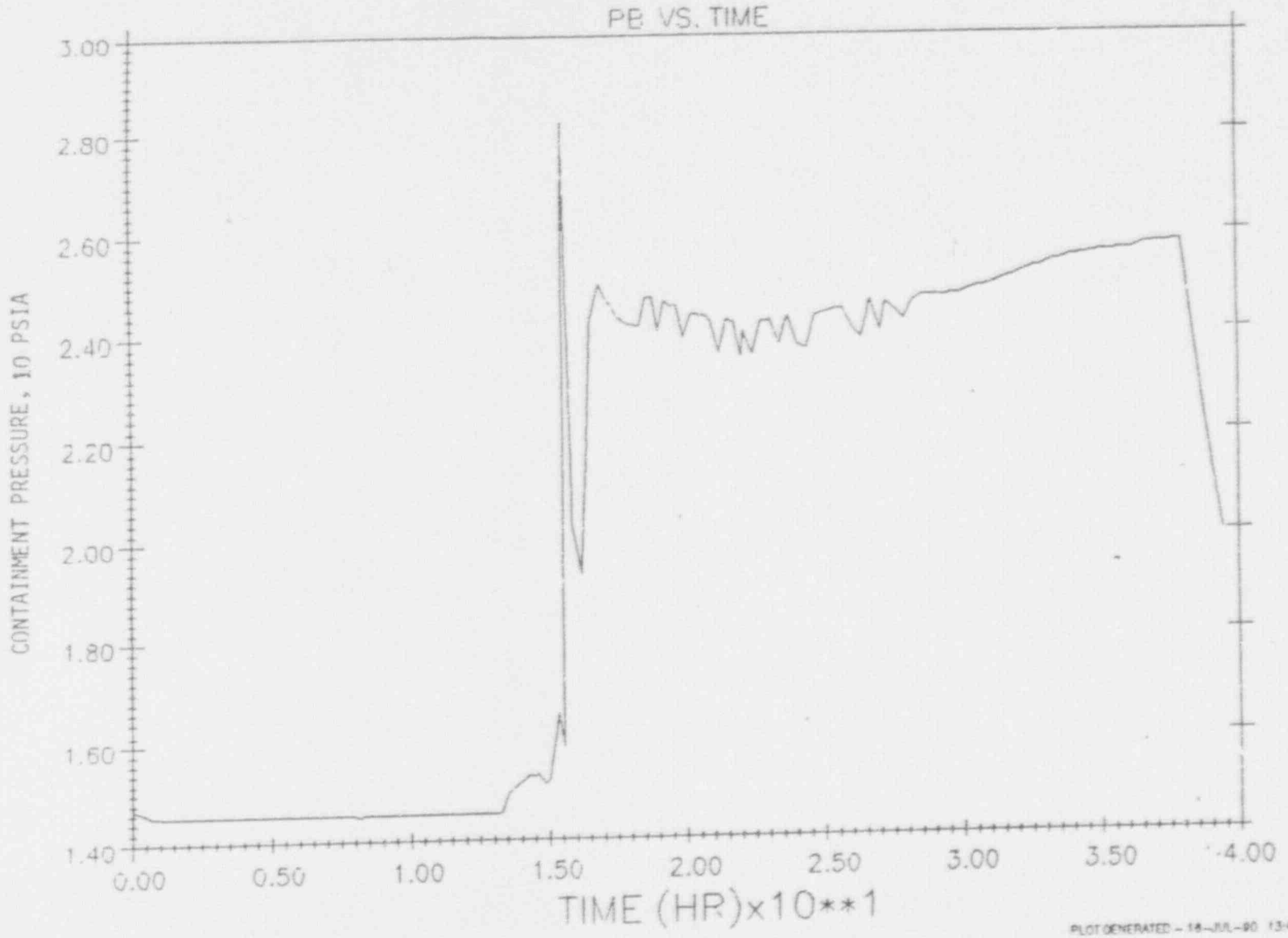
	NOBLES	CSI	TEO2	SRO	MOO2	CSOH
BAL. 1:	712.09875	61.06263	0.00000	133.53567	555.59937	354.29053
BAL. 2:	712.09991	61.06264	0.00000	133.53468	555.45697	354.29034
INITIAL:	712.09998	61.06259	-0.60084	132.50000	552.65625	354.29031
CORE:	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
CORIUM:	0.00000	0.00000	0.00000	132.70961	427.19827	0.00000
PS:	1.36874	53.55537	0.00000	0.78049	120.31207	320.09421
CONT:	710.73004	7.50725	0.00000	0.04578	8.08900	34.19633
INV REL:	712.09991	61.06264	0.00000	0.82507	128.25867	354.29034
EXV REL:	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000

	BAO	LA2O3	CEO2	SB	TE2	UO2
BAL. 1:	200.42085	1065.85315	432.05881	4.17633	61.65564	113400.68750
BAL. 2:	200.40009	1065.84985	432.05746	4.17441	61.65564	113400.68750
INITIAL:	198.95653	1057.52490	428.68570	4.16393	61.17422	113392.09375
CORE:	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
CORIUM:	188.85867	1065.58472	430.90033	2.33961	61.65564	113400.68750
PS:	11.08803	0.24125	1.10423	0.83416	0.00000	0.00000

RELEASE CLASS 2.4

Case 10

LOOP T=0 HOLE NOSP LATE WET



PLOT GENERATED - 18-JUL-90 13:48:47

	GAS MASS (LB)	AEROSOL MASS (LB)	RELEASE FRACTION
NOBL. IN	1.5369E+03	8.0695E+02	9.7896E-01
CSI	3.7522E-28	1.3609E+01	1.0109E-01
TEO2	4.0958E-34	1.4456E-01	8.5886E-04
SRO	0.0000E+00	5.4676E+00	1.0717E-02
MOO2	0.0000E+00	8.5031E-01	6.9789E-04
CSOH	4.8215E-05	8.2226E+01	1.0527E-01
BAO	0.0000E+00	3.3068E+00	7.5391E-03
LA2O3	0.0000E+00	1.0467E-01	4.48[illegible]E-05
CEO2	0.0000E+00	9.8831E-01	1.0457E-03
SB	6.3329E-06	1.5089E+00	1.6437E-01
TE2	3.5065E-04	9.5413E+00	7.0749E-02
UO2. ACT	0.0000E+00	9.7935E-01	3.9176E-06

RELEASE CLASS 3.1

Case 8
t=60 hours

NO. ITERATIONS IN PRIM SYS TRANSPORT MODEL: 1 IN CONTMT GAS TRANSPORT MODEL: 7

FISSION PRODUCT MASS BALANCES:

BALANCE 1 = TOTAL MASS AS USED BY MAAP
BALANCE 2 = IN FUEL + INTEGRATED RELEASE FROM FUEL

	NOBLES	CSI	TEO2	SRO	MOO2	CSOH
BAL. 1:	712.34180	61.06352	0.70777	133.38124	555.16205	354.29794
BAL. 2:	712.33557	61.06355	0.70777	133.38077	555.08331	354.29703
INITIAL:	712.09998	61.06259	0.19845	132.50000	552.65625	354.29031
CORE:	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
CORIUM:	0.00000	0.00000	0.00000	97.90195	281.66031	0.00000
PS:	3.92870	52.54654	0.00114	2.19301	272.29065	303.04590
CONT:	708.41309	8.51698	0.70663	33.28629	1.21108	51.25203
INV REL:	712.33557	61.06355	0.00000	2.17233	273.42291	354.29703
EXV REL:	4641.00977	0.00000	0.70777	33.30649	0.00004	0.00000

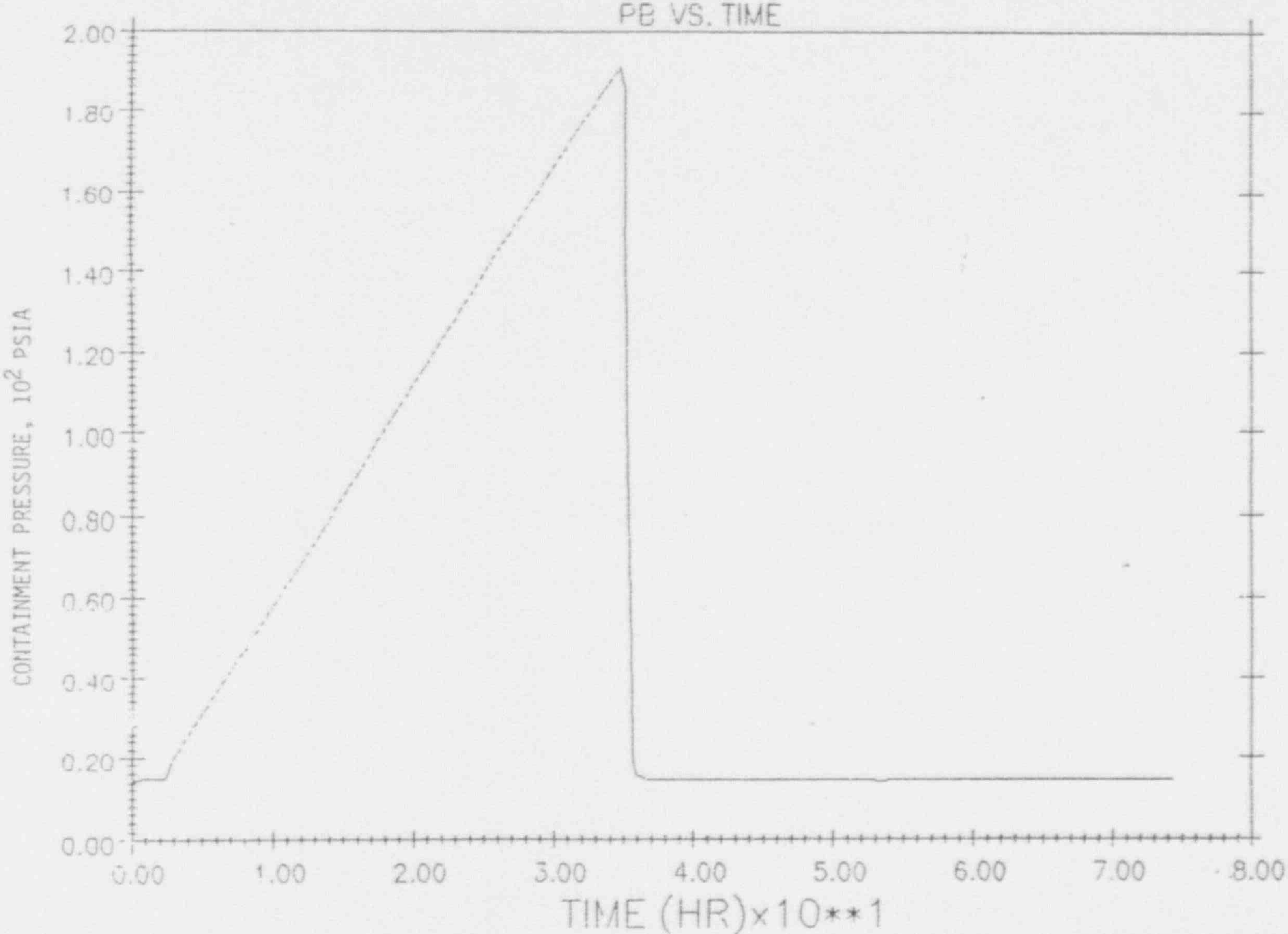
	BAO	LA2O3	CEO2	SB	TE2	UO2
BAL. 1:	200.17035	1064.59302	431.51526	4.17629	61.01521	113403.17969
BAL. 2:	200.16180	1064.59302	431.51517	4.17569	61.01535	113403.17969
INITIAL:	198.95653	1057.52490	428.68570	4.16393	60.60712	113392.09375
CORE:	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
CORIUM:	151.30380	1063.24011	410.66852	0.33854	0.00604	113398.53125
PS:	29.48168	0.77718	15.28096	2.60452	0.04928	0.00422
CONT:	19.38486	0.57568	5.56578	1.23322	60.95990	4.64107
INV REL:	29.57733	0.77729	15.27999	3.47814	0.00000	0.00000
EXV REL:	19.28067	0.57551	5.56667	0.35901	61.00932	4.64529

ADDITIONAL FIGURES-OF-MERIT OF THIS RUN

RELEASE CLASS 3.1

Case 1

TRCP TRIP LATE WET



PLOT GENERATED – 19-JUL-90 09:28:37

IN CORIUM	0.0000E+00	0.0000E+00	2.2830E+03				
CEO2 GAS	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
AERO	2.7345E-09	3.4056E-03	7.4173E-08	6.0540E-08	0.0000E+00	0.0000E+00	2.8224E-09
DEP AERO/GAS	9.3369E-09	5.1262E-07	9.1804E+00	4.2226E+00	0.0000E+00	0.0000E+00	1.C138E+02
IN CORIUM	0.0000E+00	0.0000E+00	7.7884E+02				
SB GAS	5.4684E-14	8.6769E-14	3.0132E-06	3.4309E-13	0.0000E+00	0.0000E+00	3.7246E-14
AERO	1.9291E-06	2.1838E-03	1.3311E-14	4.2928E-05	0.0000E+00	0.0000E+00	2.0863E-06
DEP AERO/GAS	6.8445E-06	3.2682E-04	7.8911E-12	1.4009E-01	0.0000E+00	0.0000E+00	7.7303E+00
IN CORIUM	0.0000E+00	0.0000E+00	2.4697E-01				
TE2 GAS	3.5837E-09	3.2916E-09	1.1515E-11	1.2497E-08	0.0000E+00	0.0000E+00	9.5566E-10
AERO	1.5863E-06	1.3476E-05	0.0000E+00	3.4767E-05	0.0000E+00	0.0000E+00	1.7583E-06
DEP AERO/GAS	5.8130E-06	2.6154E-04	0.0000E+00	8.2113E-01	0.0000E+00	0.0000E+00	3.6775E+01
IN CORIUM	0.0000E+00	0.0000E+00	1.1083E-04				
UO2.ACT GAS	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
AERO	1.9582E-06	2.4529E-05	5.3450E-05	4.3369E-05	0.0000E+00	0.0000E+00	1.9617E-06
DEP AERO/GAS	6.6796E-06	3.6739E-04	1.4573E+01	5.6184E+00	0.0000E+00	0.0000E+00	1.3100E+02
IN CORIUM	0.0000E+00	0.0000E+00	2.4978E+05				

LEAKED FROM CONTMT:	GAS MASS(LB)	AEROSOL MASS(LB)	RELEASE FRACTION
NOBL.IN	1.3762E+03	2.5627E+00	8.8935E-01
CSI	-1.3435E-20	9.9046E-02	7.3574E-04
TEO2	1.6006E-20	3.1328E-01	1.8612E-03
SRO	-2.2316E-27	5.8031E-03	1.9866E-05
MOO2	-3.1923E-26	7.6121E-03	6.2476E-06
CSOH	1.0256E-11	5.8558E-01	7.4971E-04
BAO	-9.9164E-30	6.6953E-03	1.5264E-05
LA2O3	0.0000E+00	1.0795E-03	4.6301E-07
CEO2	0.0000E+00	3.9668E-03	4.1973E-06
SB	1.6292E-11	2.4522E-02	2.6713E-03
TE2	2.5455E-07	1.2140E-01	9.0019E-04
UO2.ACT	0.0000E+00	1.1934E-02	4.7737E-08

RELEASE CLASS 4.1

Case 6
t=30 hours

NO. ITERATIONS IN PRIM SYS TRANSPORT MODEL: 4 IN CONTMT GAS TRANSPORT MODEL: 1

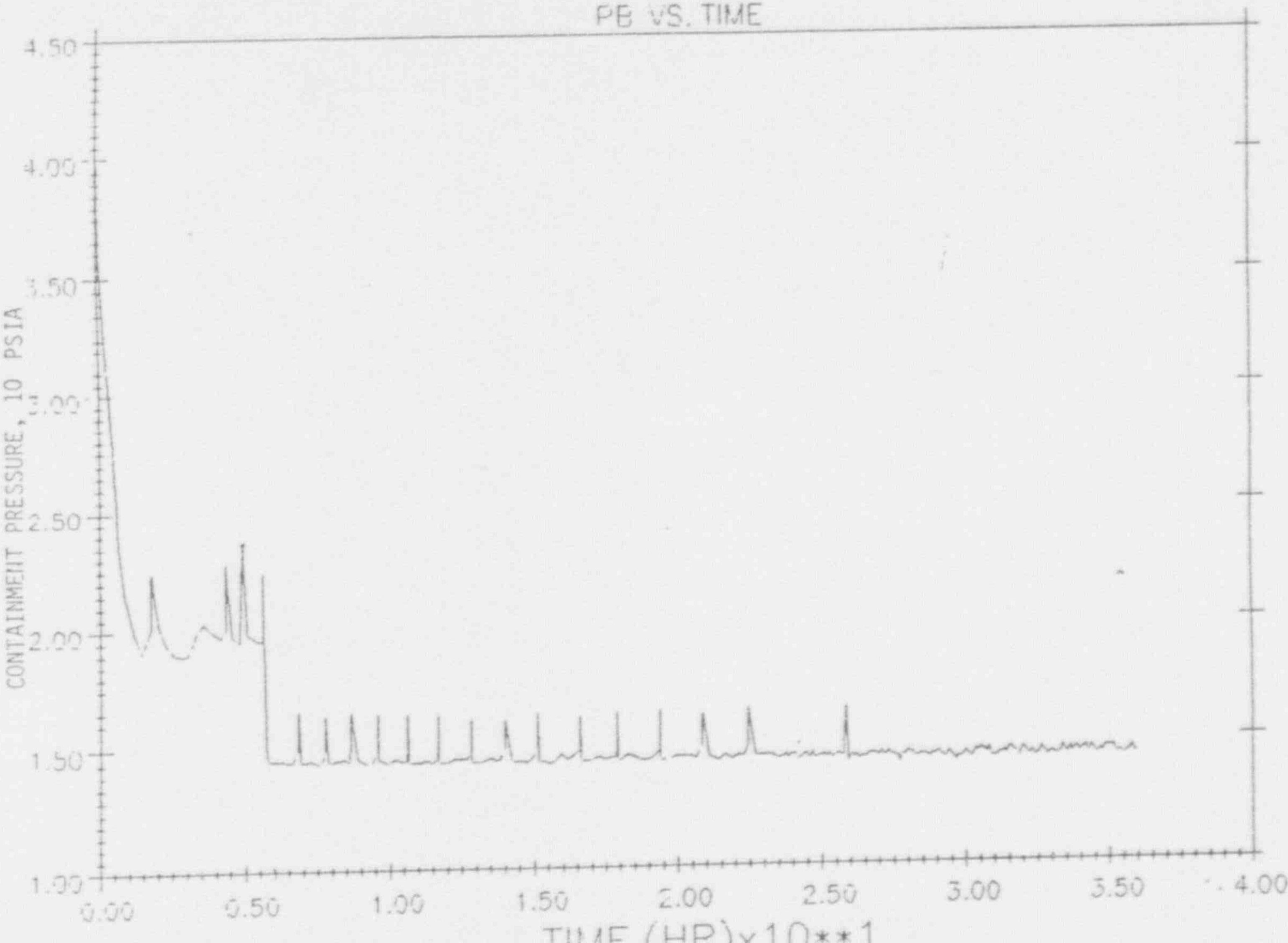
FISSION PRODUCT MASS BALANCES:

BALANCE 1 = TOTAL MASS AS USED BY MAAP
BALANCE 2 = IN FUEL + INTEGRATED RELEASE FROM FUEL

	NOBLES	CSI	TEO2	SRO	MOO2	CSOH
BAL. 1:	712.09821	61.06258	54.99501	132.30014	552.64917	354.28506
BAL. 2:	712.09869	61.06245	54.99578	132.30047	552.64978	354.29047

RELEASE CLASS 4.1
Case 6

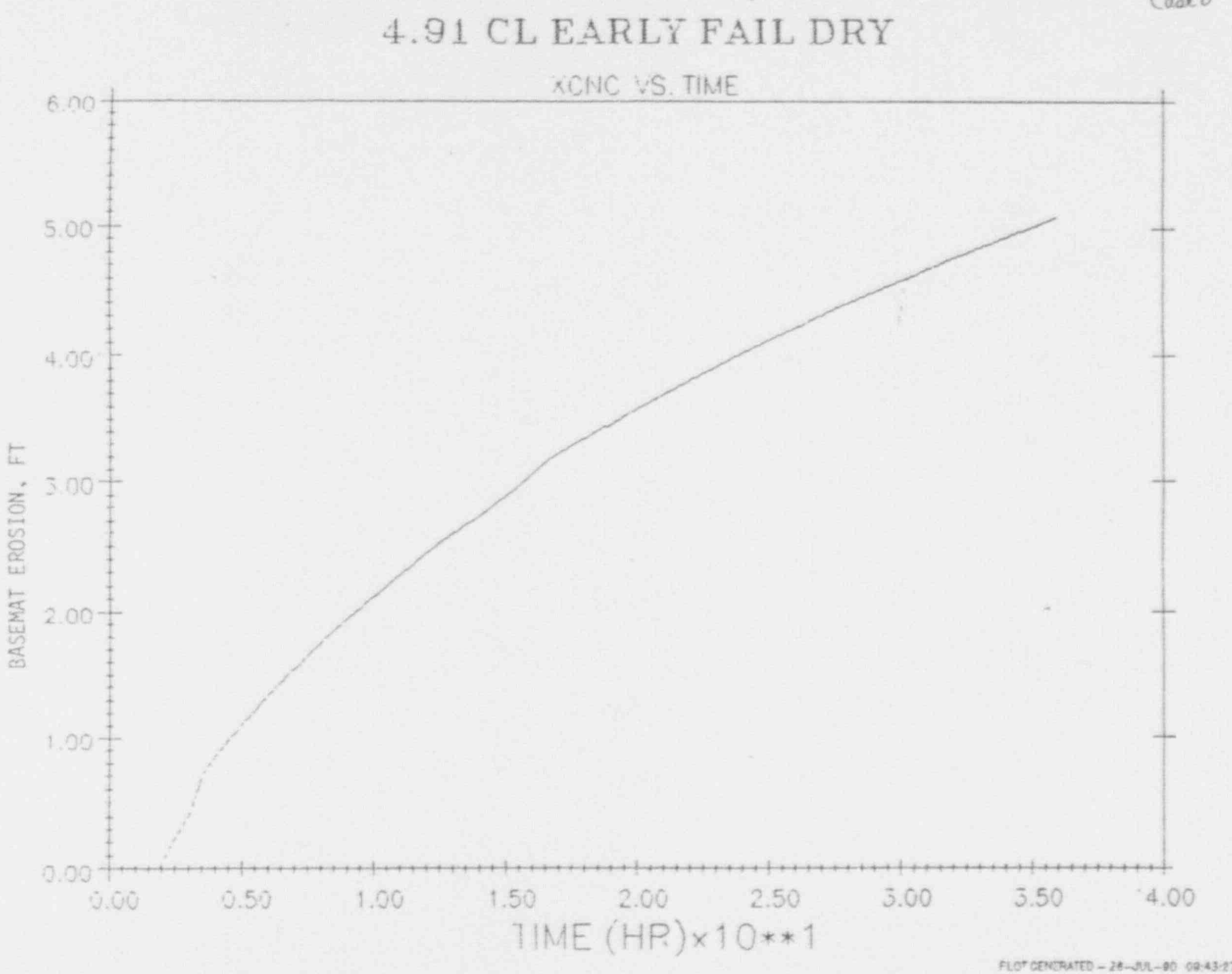
4.91 CL EARLY FAIL DRY



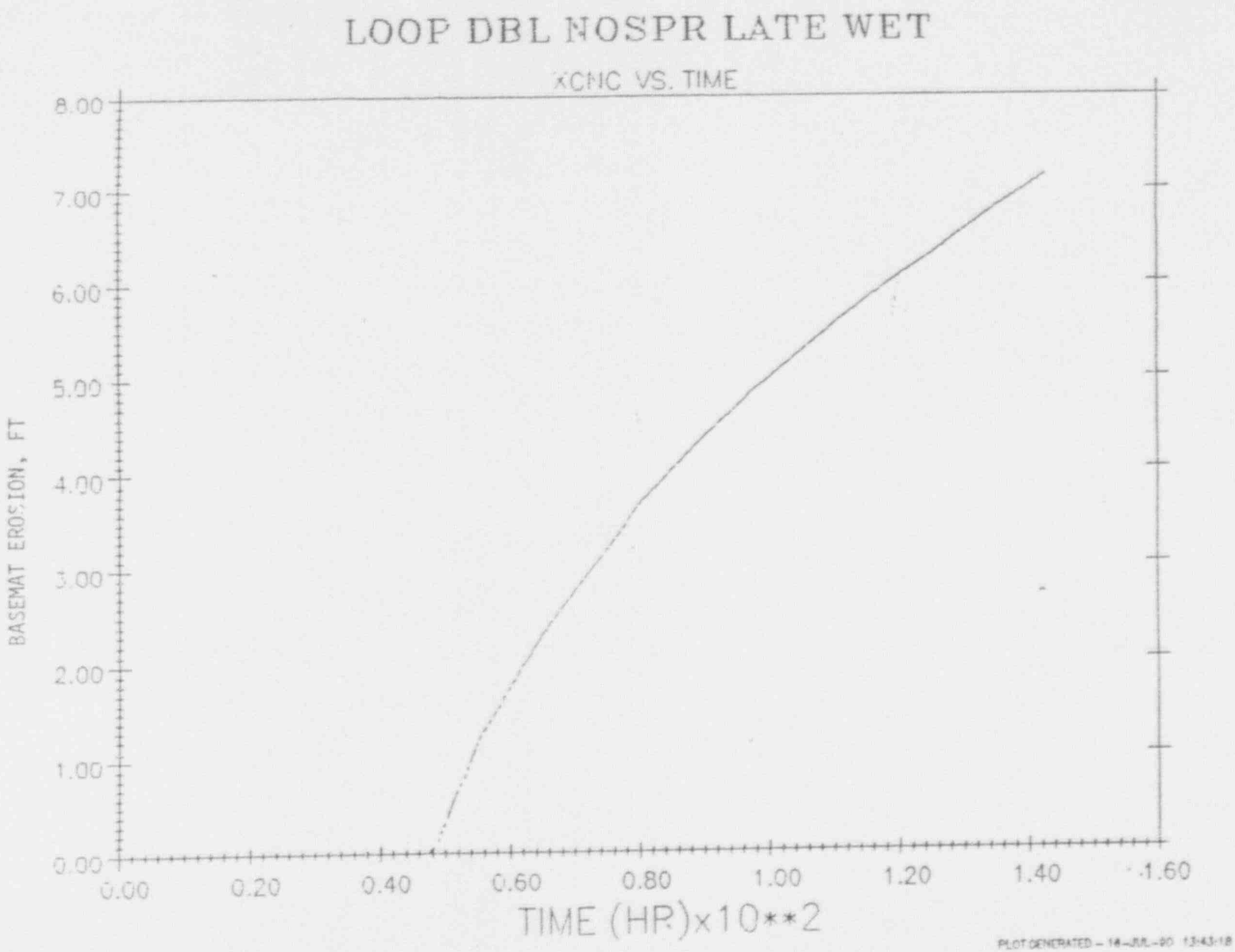
PLOT GENERATED – 28–JUL–90 09:39:38

RELEASE CLASS 4.1

Case 6



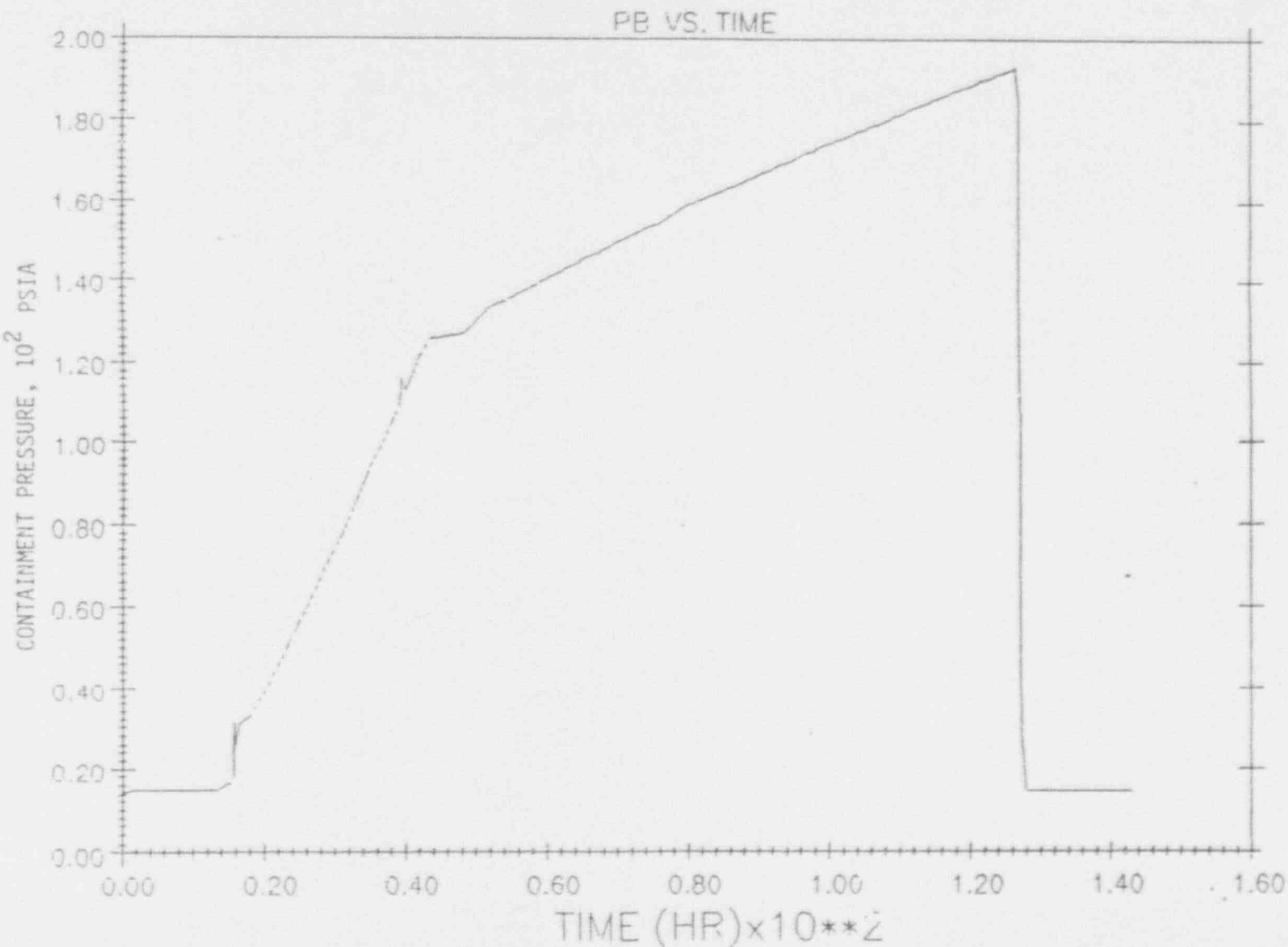
RELEASE CLASS 5.1
Cont



RELEASE CLASS 5.1

Case 4

LOOP DBL NOSPR LATE WET



PLOT GENERATED - 16-JUL-80 13:39:20

UO2.ACT GAS	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
AERO	2.6404E-03	1.1766E-03	3.5328E-05	2.7504E-03	0.0000E+00	0.0000E+00	4.2961E-09
DEP AERO/GAS	6.6114E-01	1.2763E+00	9.9038E-02	1.3413E+00	0.0000E+00	0.0000E+00	7.0742E-03
IN CORIUM	0.0000E+00	0.0000E+00	2.5000E+05				

LEAKED FROM CONTMT:	GAS MASS(LB)	AEROSOL MASS(LB)	RELEASE FRACTION
NOBL.IN	1.5592E+03	3.0295E+01	9.9318E-01
CSI	8.1989E-06	5.2376E+00	3.8906E-02
TEO2	8.7014E-29	2.2399E+00	[illegible]E-02
SRO	0.0000E+00	1.0928E-02	3.7411E-05
MOO2	0.0000E+00	1.7760E-02	1.4577E-05
CSOH	1.5077E-02	3.5562E+01	4.5549E-02
BAO	0.0000E+00	3.6747E-02	8.3778E-05
LA2O3	0.0000E+00	4.9348E-04	2.1166E-07
CEO2	0.0000E+00	2.3801E-03	2.5183E-04
SB	8.9238E-03	1.9074E+00	2.0875E-01
TE2	4.8811E-01	1.0952E+00	1.1740E-02
UO2.ACT	0.0000E+00	2.5730E-02	1.0292E-07

RELEASE CLASS 5.1

Case 4

t=140 hours

NO. ITERATIONS IN PRIM SYS TRANSPORT MODEL: 1 IN CONTMT GAS TRANSPORT MODEL: 3

FISSION PRODUCT MASS BALANCES:

BALANCE 1 = TOTAL MASS AS USED BY MAAP
BALANCE 2 = IN FUEL + INTEGRATED RELEASE FROM FUEL

	NOBLES	CSI	TEO2	SR	MOO2	CSOH
BAL. 1:	712.10083	61.06186	3.42127	133.30748	554.92566	354.29453
BAL. 2:	712.09961	61.06260	3.42127	133.30490	554.84570	354.29013
INITIAL:	712.09998	61.06259	2.94843	132.50000	552.65625	354.29031
CORE:	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
CORIUM:	0.00000	0.00000	0.00000	123.00731	427.78699	0.00000
PS:	0.14386	50.58181	0.00178	0.72725	122.35056	303.48398
CONT:	711.95697	10.48005	3.41949	9.57292	4.78812	50.81054
INV REL:	712.09961	61.06260	0.00000	0.74544	127.05857	354.29013
EXV REL:	6206.77490	0.00000	3.42127	9.55215	0.00014	0.00000

	BAO	LA2O3	CEO2	SB	TE2	UO2
BAL. 1:	2[illegible].14630	1063.86218	431.25928	4.17410	58.81182	113400.44531
BAL. 2:	200.13066	1063.86011	431.25839	4.17232	58.81169	113400.44531
INITIAL:	198.95653	1057.52490	428.68570	4.16393	58.43295	113392.09375
CORE:	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
CORIUM:	184.82861	1063.32910	428.95325	2.18885	8.90633	113398.89844
PS:	9.68557	0.27026	0.80507	0.26887	0.14393	0.00331
CONT:	5.63211	0.26283	1.50098	1.71637	49.76155	1.54676

CEO2	GAS	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
	AERO	3.2043E-07	1.5093E-07	4.1308E-09	3.2744E-07	0.0000E+00	0.0000E+00	1.3530E-07
	DEP AERO/GAS	2.8727E-02	5.1188E+01	1.1266E+01	2.2965E+01	0.0000E+00	0.0000E+00	1.2871E+00
	IN CORIUM	0.0000E+00	0.0000E+00	7.9490E+02				
SB	GAS	1.9654E-02	8.7622E-03	1.0139E-05	1.5263E-02	0.0000E+00	0.0000E+00	1.3082E-10
	AERO	-8.6591E-13	-7.4996E-13	-7.9819E-15	4.6744E-03	0.0000E+00	0.0000E+00	7.1525E-05
	DEP AERO/GAS	1.0470E-01	3.6150E+00	6.9802E-11	2.4769E+00	0.0000E+00	0.0000E+00	8.6451E-02
	IN CORIUM	0.0000E+00	0.0000E+00	1.0973E+00				
TE2	GAS	3.0820E-02	1.4264E-02	3.0867E-08	3.2370E-02	0.0000E+00	0.0000E+00	2.6794E-07
	AERO	-2.8986E-17	-3.4161E-17	-9.6602E-20	-3.1838E-16	0.0000E+00	0.0000E+00	8.0712E-04
	DEP AERO/GAS	0.0000E+00	5.8330E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	1.2637E-01
	IN CORIUM	0.0000E+00	0.0000E+00	9.6763E-03				
UO2,ACT	GAS	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
	AERO	5.0214E-04	2.3715E-04	7.4957E-06	5.1477E-04	0.0000E+00	0.0000E+00	1.1068E-06
	DEP AERO/GAS	9.0435E-02	3.8280E+01	1.7479E+01	1.1910E+01	0.0000E+00	0.0000E+00	6.1489E-01
	IN CORIUM	0.0000E+00	0.0000E+00	2.4988E+05				

LEAKED FROM CONTMT:	GAS MASS(LB)	AEROSOL MASS(LB)	RELEASE FRACTION
NOBL. IN	1.5297E+03	3.3049E+01	9.7308E-01
CSI	1.1260E-04	2.9340E-01	2.1803E-03
TEO2	8.9105E-09	3.0623E-01	1.8193E-03
SRO	0.0000E+00	1.4308E-02	4.9254E-05
MOO2	2.8669E-36	4.2447E-02	3.4838E-05
CSOH	7.3339E-02	1.6288E+00	2.1772E-03
BAO	0.0000E+00	1.9965E-02	4.5518E-05
LA2O3	0.0000E+00	2.8345E-03	1.2158E-06
CEO2	0.0000E+00	2.4222E-02	2.5630E-05
SB	6.0594E-02	1.0112E-01	1.7616E-02
TE2	1.0817E+00	1.5305E+00	1.9369E-02
UO2, ACT	0.0000E+00	1.3098E-02	5.2394E-08

RELEASE CLASS 6.2

Case 2

t=350 hour

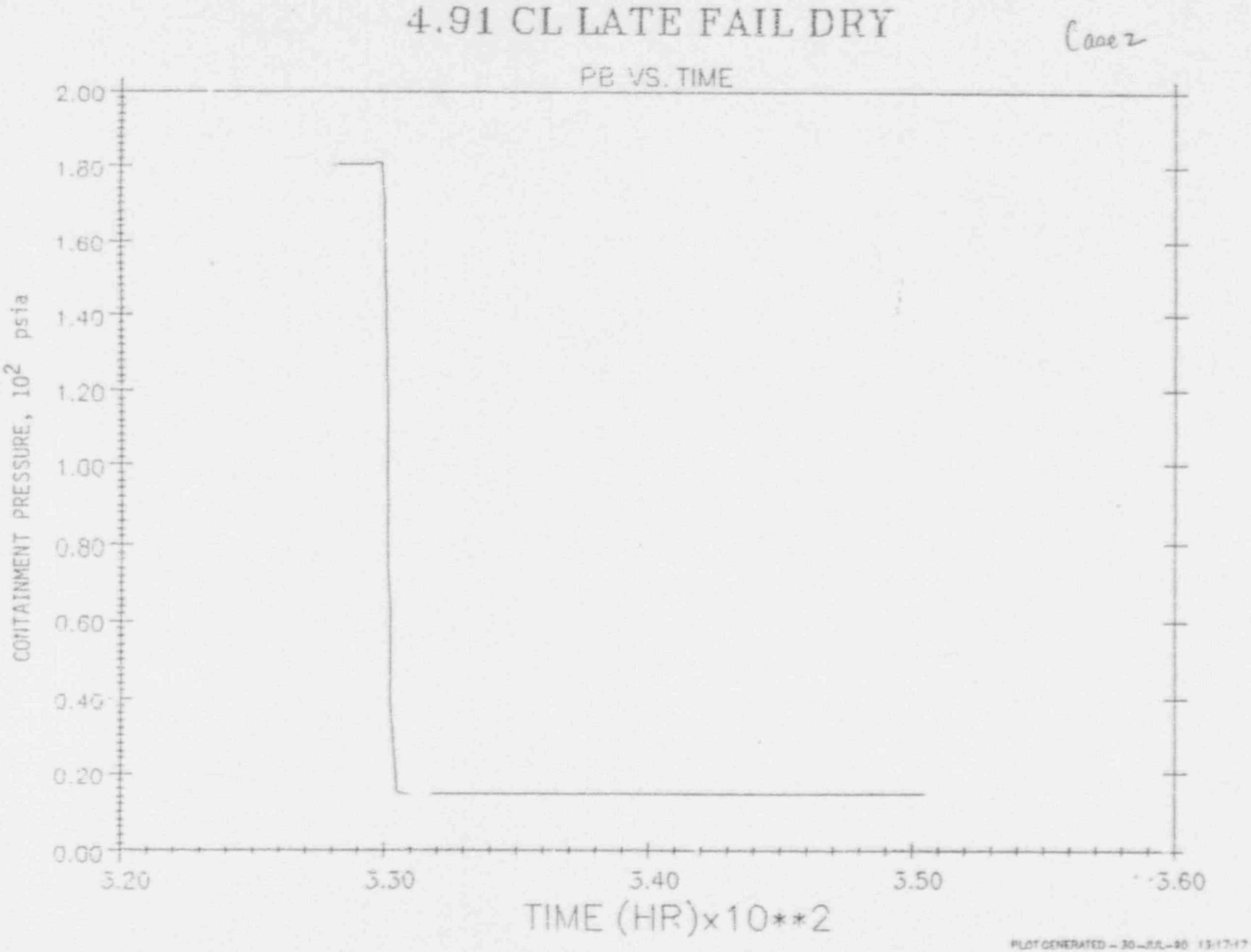
NO. ITERATIONS IN PRIM SYS TRANSPORT MODEL: 2 IN CONTMT GAS TRANSPORT MODEL: 1

FISSION PRODUCT MASS BALANCES:

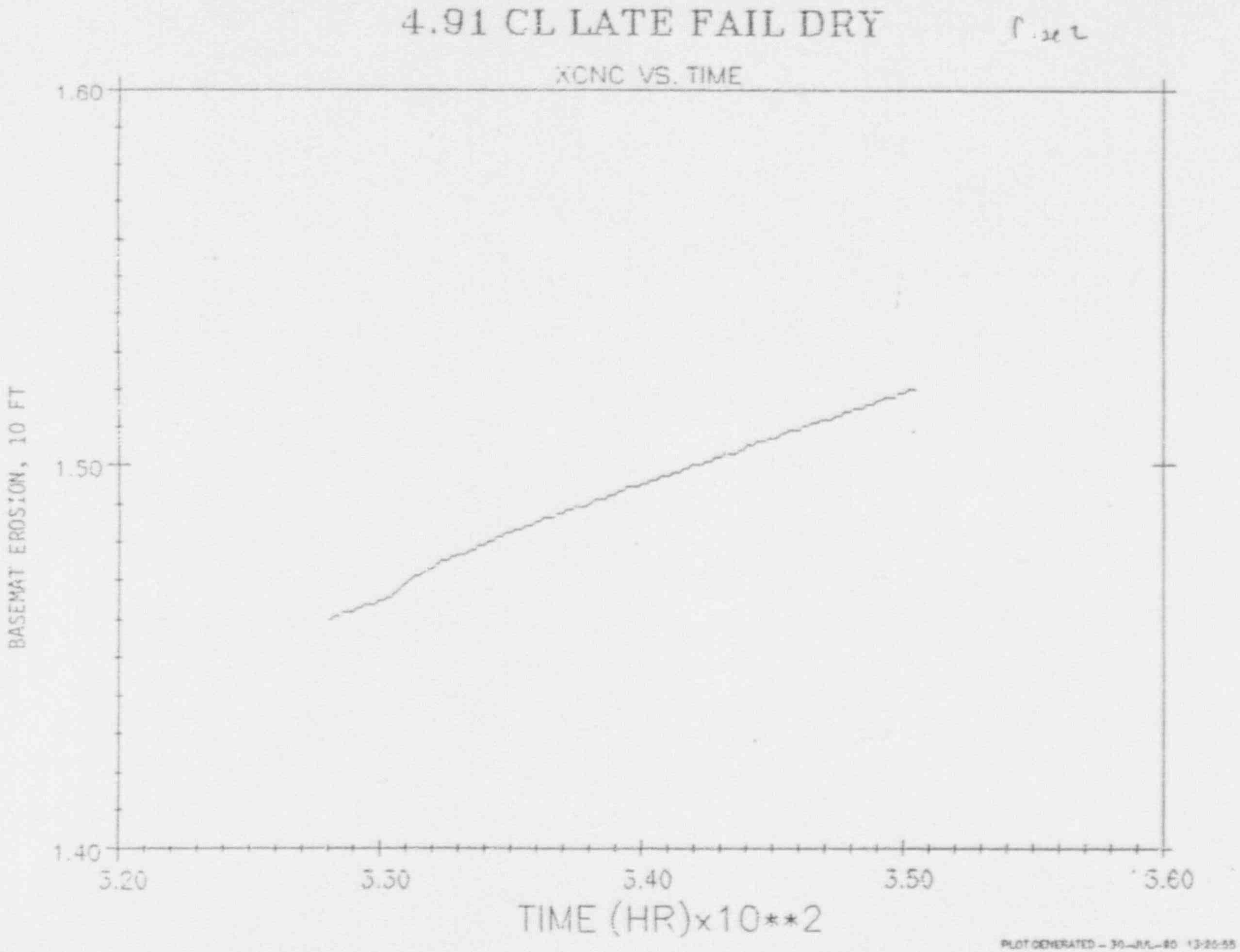
BALANCE 1 = TOTAL MASS AS USED BY MAAP
BALANCE 2 = IN FUEL + INTEGRATED RELEASE FROM FUEL

	NOBLES	CSI	TEO2	SRO	MOO2	CSOH
BAL. 1:	712.09729	61.06078	71.44212	132.51192	552.61133	354.25201
BAL. 2:	712.09955	61.06253	71.44342	132.51312	552.61243	354.29053
INITIAL	712.07998	61.06259	71.44608	132.50000	552.65625	354.29031

RELEASE CLASS 6.2



RELEASE CLASS 6.2



CSOH GAS	4.5797E-07	4.2441E-09	4.0815E-09	6.8670E-08	0.0000E+00	0.0000E+00	2.3324E-09
AERO	3.8066E-03	2.8380E-04	8.1272E-07	4.6026E-02	0.0000E+00	0.0000E+00	1.6080E-03
DEP AERO/GAS	1.3717E-02	4.3017E-01	5.0894E+01	3.4836E+01	0.0000E+00	0.0000E+00	8.3424E+01
IN CORIUM	0.0000E+00	0.0000E+00	0.0000E+00				
BAO GAS	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
AERO	1.7719E-08	5.3730E-08	6.8646E-11	2.2235E-07	0.0000E+00	0.0000E+00	7.6378E-06
DEP AERO/GAS	7.8097E-08	2.1087E-06	3.4547E+00	2.4597E+00	0.0000E+00	0.0000E+00	6.6355E+00
IN CORIUM	0.0000E+00	0.0000E+00	4.1943E+02				
LA2O3 GAS	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
AERO	3.6495E-10	2.3829E-09	3.0432E-12	1.081[illegible]-08	0.0000E+00	0.0000E+00	3.3662E-07
DEP AERO/GAS	3.8905E-09	1.0255E-07	1.2276E-01	8.[illegible]81E-02	0.0000E+00	0.0000E+00	2.5342E-01
IN CORIUM	0.0000E+00	0.0000E+00	2.3672E+03				
CEO2 GAS	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
AERO	5.4310E-09	1.4059E-08	1.7994E-11	6.7771E-08	0.0000E+00	0.0000E+00	1.9826E-06
DEP AERO/GAS	2.4570E-08	6.4250E-07	8.3229E-01	4.3718E-01	0.0000E+00	0.0000E+00	1.5959E+00
IN CORIUM	0.0000E+00	0.0000E+00	9.5520E+02				
SR GAS	1.0509E-10	1.1851E-10	3.2676E-10	3.4920E-09	0.0000E+00	0.0000E+00	1.0825E-10
AERO	3.7044E-05	2.7824E-06	7.6399E-09	4.4790E-04	0.0000E+00	0.0000E+00	1.8523E-05
DEP AERO/GAS	1.8216E-04	4.2156E-03	5.1712E-01	6.4058E-01	0.0000E+00	0.0000E+00	1.5298E+00
IN CORIUM	0.0000E+00	0.0000E+00	6.3828E+00				
TE2 GAS	6.2809E-07	5.4461E-07	1.0807E-08	5.6394E-06	0.0000E+00	0.0000E+00	2.5023E-07
AERO	5.4672E-07	3.2711E-07	-2.6146E-18	-5.9242E-16	0.0000E+00	0.0000E+00	4.5037E-05
DEP AERO/GAS	1.6926E-06	2.6864E-05	1.1839E+01	1.0462E+01	0.0000E+00	0.0000E+00	2.7872E+01
IN CORIUM	0.0000E+00	0.0000E+00	8.6004E+01				
UO2,ACT GAS	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
AERO	4.2268E-09	1.1589E-08	1.4203E-11	5.2844E-08	0.0000E+00	0.0000E+00	1.6369E-06
DEP AERO/GAS	1.9020E-08	5.0105E-07	4.7991E-01	3.6704E-01	0.0000E+00	0.0000E+00	1.1110E+00
IN CORIUM	0.0000E+00	0.0000E+00	2.5003E+05				

LEAKED FROM CONTMT:	GAS MASS(LB)	AEROSOL MASS(LB)	RELEASE FRACTION
NOBL.IN	4.7231E+01	2.0496E+00	3.0085E-02
CSI	1.6483E-11	3.4513E-02	2.5638E-04
TEO2	0.0000E+00	0.0000E+00	0.0000E+00
SRO	0.0000E+00	3.7072E-03	1.2698E-05
MOO2	0.0000E+00	2.4130E-02	1.9805E-05
CSOH	7.8389E-07	1.2468E-01	1.5962E-04
BAO	0.0000E+00	2.6989E-03	6.1530E-06
LA2O3	0.0000E+00	7.9140E-05	3.3945E-08
CEO2	0.0000E+00	4.1874E-04	4.4307E-07
SR	1.8143E-07	2.3289E-03	2.5372E-04
TE2	1.4053E-05	8.7507E-03	6.4988E-05
UO2.ACT	0.0000E+00	2.9960E-04	1.1985E-07

RELEASE CLASS 7.1

Case 1

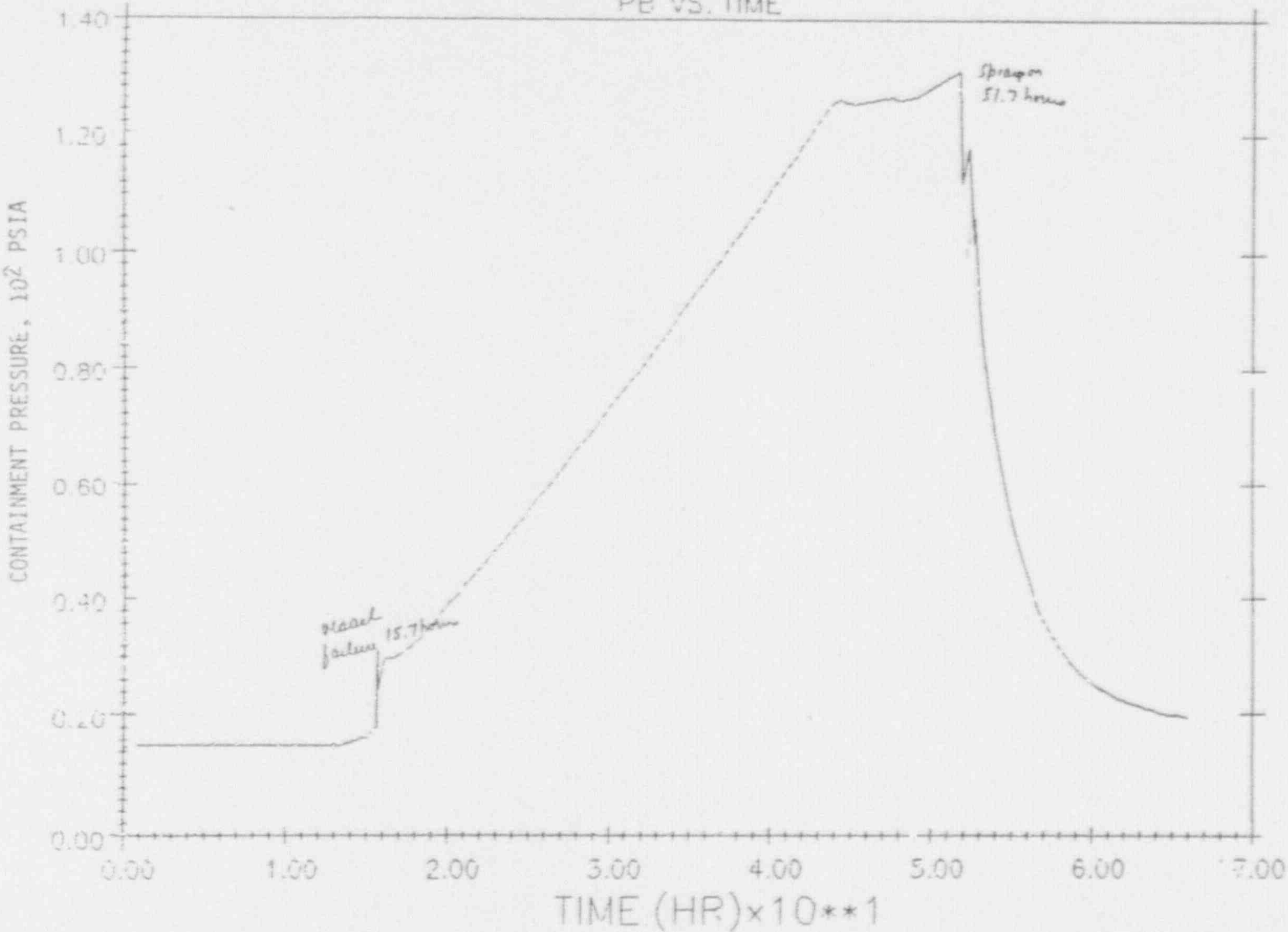
t=60 hours

RELEASE CLASS 7.1

LOOP DBL 1SPR LATE WET

36 hour delay
case 1

PB VS. TIME

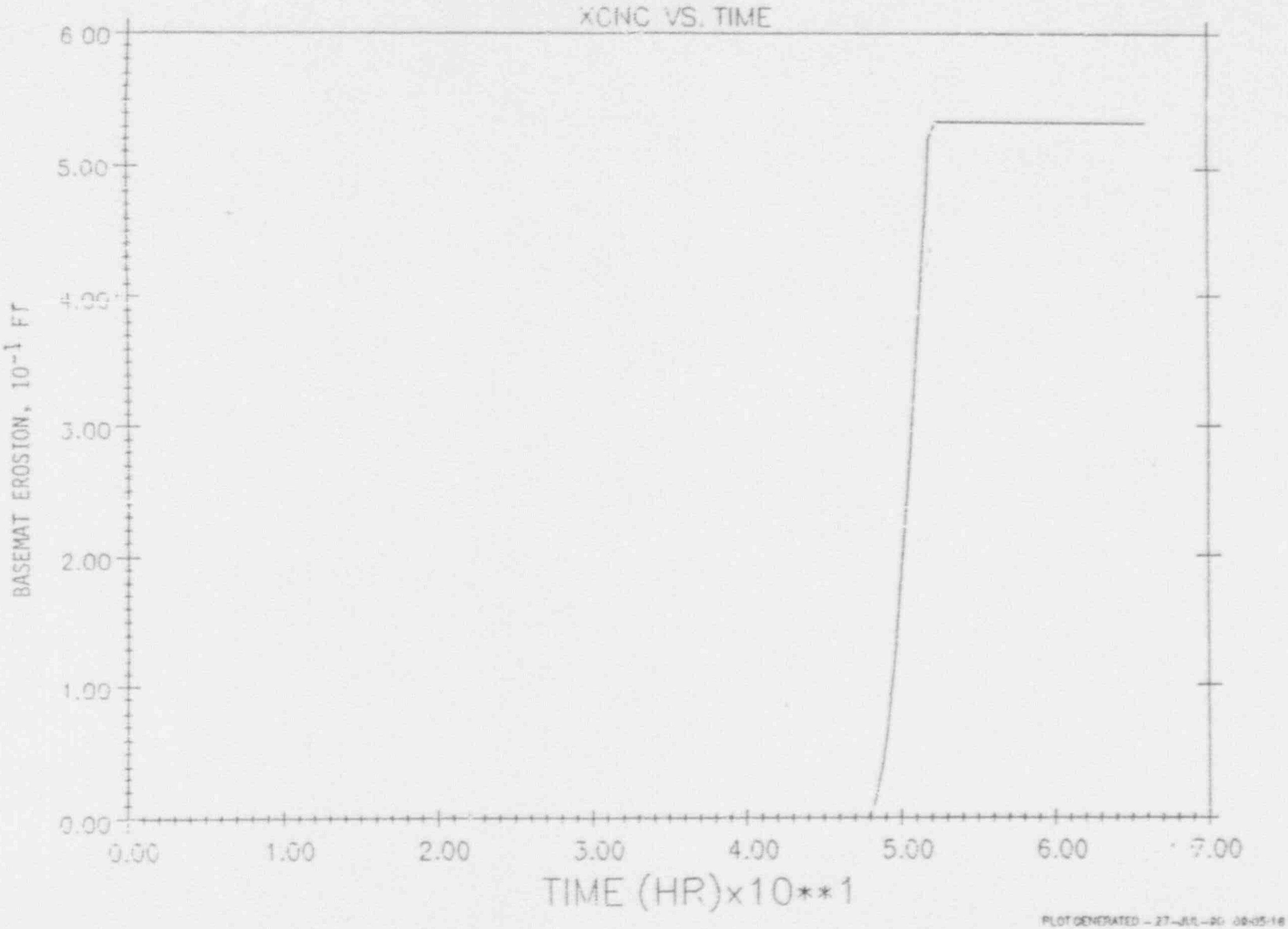


PLOT GENERATED – 27–JUL–90 09:01:18

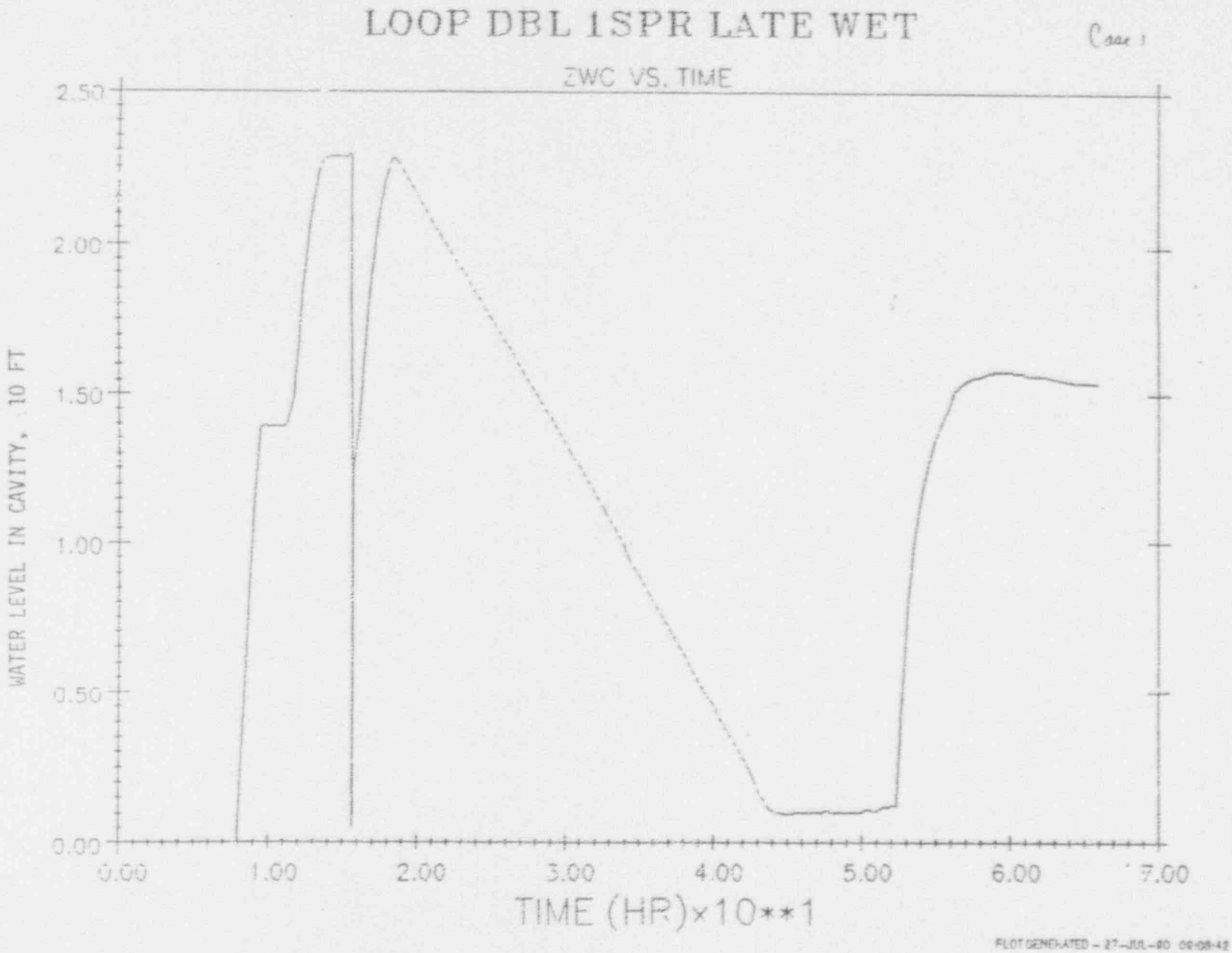
RELEASE CLASS 7.1

LOOP DBL 1SPR LATE WET

Case 1



RELEASE CLASS 7.1



QUESTION 722.23

Describe the version/revision of the MAAP code used for the containment and source term analyses, and how this version differs from MAAP3.0B, Rev. 7.0. Discuss any changes to the code/deck to accommodate the specific geometry of the CE system 80+ containment or to provide CE specific constitutive equations. Also identify major input assumptions regarding modelling of the NRC/IDCOR phenomenological issues, such as in-vessel flow blockage due to cladding relocation, hydrogen recombination in the reactor cavity, and debris coolability in the reactor cavity.

RESPONSE 722.23

Most of the severe accident deterministic analyses in support of the System 80+ PRA effort were performed using MAAP3.0B, Revision 16. In some of the early calculations Revision 11 was used, however, for each of these cases a benchmark Revision 16 calculation was also performed to validate the results. On January 31, 1991 Fauske and Associates issued Revision 17 of MAAP3.0B (CE assumes that "Rev. 7.0" in the above question was a typographical error, meant to read "Revision 17") to the MAAP Users Group members. There were many changes made in Revision 17 relative to Revision 16. These included enhanced core modeling, mid-loop operation modeling, better I/O capability, more efficient numerical calculations, and many engineered safety features model upgrades. Reference 1 (attached) provides brief descriptions of these changes/enhancements.

The computer data file that represents CE's System 80+ design contains fully integrated flexible models for simulating all essential features of the system containment design as well as the engineered safety features systems. This includes the capability to model the in-containment refueling water storage tank (IRWST), the cavity flooding system, and the safety depressurization system (SDS). The IRWST model is implemented using the quench tank model of MAAP3.0B, Revision 16 as a starting point. Additional modeling changes were made as necessary to model the cavity flooding system, the lower compartment, and the holdup volume. The SDS valves were modelled using the PORV model of MAAP3.0B, Revision 16 and by modifying the relevant input to reflect the design and operation of the SDS valves.

Reference 1. Letter to MAAP Users Group Members from M. Plys, Fauske & Associates, January 31, 1991.

Reference 1. of NRC
Question 722.23

Fauske & Associates, Inc.

DATE: January 31, 1991

TO: MAAP Users Group Members

FROM: Martin Plys, FAI MGP

SUBJECT: Release of MAAP 3.0B BWR Revision 7 and PWR Revision 17

FAI is pleased to announce the release of the new revisions of MAAP 3.0B to MAAP Users Group Members. We received approval from Ed Fuller of EPRI on January 17. His approval was based on code performance for sample problems run in both single and double precision on VAX, 386 PC, and special purpose CPUs. Since that time, a few last minute bugs were repaired, Quality Assurance was completed, and voluminous documentation was assembled.

Accompanying this memorandum are the new MAAP software, a description of the software transmittal format, and new code documentation as explained below.

We have created a documentation package for each code that explains the contents of the new revisions. You will find a summary of major updates, detailed descriptions of major new models plled from QA documentation that summarize individual code changes, a list of new parameter file inputs, examples of new input decks, and a list of all subroutines changed in these revisions, and station blackout sample problem plots. We anticipate that this level of documentation will be more than adequate for QA requirements of most MUG members. Note that we cannot send out our entire QA document on each code because each one is about 4 times larger than the abridged versions you received.

Sample problems for the new revisions differ from the old sample problems because the blockage model is turned off by default. Therefore, expect more in-vessel hydrogen with the new sample parameter file. Generally speaking, code performance is not significantly different for these new revisions compared to the old revisions. However, users should check for differences in results that may be caused by threshold criteria. For example, check sequences for which pressures or temperatures reached transient levels near failure levels to ensure that results are unchanged. In general, output should be scrutinized for these kinds of events and sensitivity studies should be made whenever failure thresholds are approached. We found that selection of the no blockage option for the BWR blackout sample problem can result in transient drywell pressure near the old assumed failure level immediately following vessel failure, and this provides a good example to all users concerning threshold sensitivity behavior.

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Telefax (708) 986-5481

Reference 1. of NRC
Question 722.23

Numerical performance for the PWR code is better than ever. However, numerical performance for the BWR code is now somewhat poorer. This is due to the new core model and is compounded by the use of the no blockage option. FAI is investigating BWR code numerical performance. Despite this shortcoming, the essential character of BWR code results is unchanged, and judgements concerning the risk significance of sequences or mitigative actions are unaffected. Since results for major event times are usually within the 3% acceptance criterion, EPRI and FAI have judged the codes acceptable for release.

In the new code revisions, there are new and revised parameter file entries and there is an upgrade to I/O for the parameter file and input deck. You can use the previous I/O features except that you must use the new plotting format (see *PLTMAP in the parameter file). You will only need to change or add a small number of parameter file entries to run these new code versions. However, you will need to change or add more parameter file entries to use new model features, such as the ESF upgrades and PWR half loop operation.

Major model additions include the engineered safety feature upgrades, an improved BWR HEATUP model, PWR changes for half loop operation, upgraded PWR numerical performance, and enhanced I/O for the parameter file and input decks, including upgrades to the user defined event codes. ESF upgrades are those approved at the June 1990 MUG meeting plus extra modifications agreed upon during the November steering committee meeting. The BWR HEATUP model now includes control blade motion separate from fuel motion. PWR half-loop operation modifications go beyond the proposed scope to include separate mass fractions for hydrogen and air in all primary system nodes.

Enhanced I/O allows definition of automatic operator actions that occur when event codes (essentially any desired intervention condition) change state. Since event codes are monitored continuously and the actions are automatic, you may in effect look for intervention conditions in parallel with this new change. As an example, suppose you wish to open a relief valve when the water level reaches a certain low threshold, and you want to turn on containment sprays when the temperature reaches a certain high threshold, but you do not know in advance which condition will be reached first. Simply create a user defined event code for each intervention condition, and create an automatic action corresponding to each. Each action will be taken when its condition is reached regardless of order. Order may be established by making one intervention condition contingent upon previous occurrance of another.

The mass and energy balance output developed for the MELCOR/MAAP comparison are not yet completely installed. They are not provided with the PWR code, and they are overridden by default in the BWR code. We will provided fully debugged mass and energy balances in future minor code revisions.

PC users may need a modified INPUT1 subroutine to open files. We have included our version of this file for PC's. We encourage PC users to inform us of PC difficulties and their resolution so that other MUG members may benefit from the experience and to identify simple remedies in future code releases.

Reference 1. of NRC
Question 722.23

FAI will transmit revised MAAP Users Manual and Users Guide sections through MAAP RAAP.

Please contact Barbara Schlenger if you have any difficulties installing the new revisions or require clarification of the new code input or models.

Question 722.24

In Section 2 of the CE System 80+ PRA it is stated that:

- a full recovery analysis was performed for each core damage sequence with point estimate frequency greater than 1.E-12,

- each core damage sequence with mean frequency greater than 1.E-11 was linked to containment safeguards states to generate a set of plant accident sequences,

- each plant damage state (PDS) contribution whose frequency was greater than 1.E-10 was reviewed ' identify the dominant PDS for each release class.

Please describe the rationale for se ting each of these screening thresholds. Specifically address what as done with the frequency associated with sequences below the noted threshold value at each successive step of the analysis, and what was done with those sequences whose release category frequency was less than 1.E-10.

Response 722.24

The basic criterion for determining which sequences would be subject to recovery analysis was that all major sequences would be subject to recovery analysis. The original numerical threshold was set to be approximately 3 orders of magnitude below the point estimate for total core damage frequency, or about 1.E-10. Several interesting sequences with frequencies less than 1.E-10 were found, so the threshold value was reduced so that these sequences would be included. All sequences, both recovered and unrecovered, were used to calculate the final total core damage frequency.

In selecting sequences to link with the containment safeguards states, the intent was to account for at least 99% of the core damage frequency while minimizing the number of sequences that had to be linked with containment safeguards states because of the multiplicative effect. Originally a threshold value of 1.0E-11 was arbitrarily established. After a preliminary screening, it was determined that if a threshold value of 1.0E-10 was used, more than 99% of the core damage frequency would be covered while reducing the number of sequences that had to be dealt with by 20%. (Note: the threshold value of 1.0E-11 on page 2-17 should be changed to 1.0E-10.) The sequences with frequencies below the 1.0E-10 threshold were not propagated further in the analysis. This affected much less than 1% of the total core damage frequency.

The third threshold value was established for release class filtering. On page 2-18 of the PRA report, the statement reads "To do this, the row vector of PDS contributions for each release class whose frequency of occurrence was greater than or equal to 1.0E-10/reactor year was reviewed ...". The objective of setting this threshold value was to reduce the number of release classes for which CRAC2 runs were to be made while covering at least 99% of the total releases. One additional criterion used to establish this value was that each major containment failure mode would include at least one release class for which a CRAC2 run was made. The threshold value was set at 1.0E-10 so that Early Containment Failure would be covered by at least one CRAC2 run. Release classes below the threshold value were excluded from further analysis. Less than .1% of the total releases were thus excluded.

Question 722.25

Clarify the apparent inconsistency between the statement on page 2-17 that "each core damage sequence with a mean probability greater than or equal to 1.E-11 was linked to the set of containment safeguards states to generate a set of plant accident sequences." and the statement on page 9-21 that "only core damage sequences with an occurrence frequency of 1.E-10 or greater were propagated through the containment safeguards event tree.".

Response 722.25

As discussed in the response to Question 722.24, 1.0E-11 was the preliminary value set for the screening threshold. This value was latter revised to the 1.0E-10 value. The value provided on page 2-17 will be changed to reflect the actual screening value of 1.0E-10.

Question 722.26

It appears that core damage sequences with an occurrence frequency of 1.E-10 or smaller were truncated when mapped to plant accident sequences. Please provide an assessment of the total core damage frequency that is thus ignored, and what fraction of the overall core damage frequency this represents. Discuss the effect on the CCFP when these sequences are included (particularly in view of the fact that the calculated CCFP is 0.099).

Response 722.26

The total core damage frequency ignored due to truncation is 5.39E-10. This is less than 0.03% of the total core damage frequency. Since the truncated core damage frequency is so low, it would have no impact on the calculated CCFP.

Question 722.27

The largest accident sequence frequencies reported in Table 9.1-5 are approximately 1.E-8. Thus, propagation of core damage sequences greater than 1.E-10 through the containment safeguards event tree would appear to provide for retention of approximately 99% of the core damage frequency. However, the sequence frequencies entering the CSETs already reflect credit for operator recovery actions. From inspection of Tables 5-6 and 5-7, these actions typically reduce the sequence frequency by 3 to 4 orders of magnitude (human error probabilities of 1.E-3 to 1.E-4). As a result, sequences with a frequency of 1.E-6 to 1.E-7 before recovery (which would otherwise be dominant) would not be captured in the analysis, and any supporting sensitivity analyses. In order to reflect all potentially important sequences for subsequent treatment in sensitivity and uncertainty analyses, please provide a supplementary analysis of PDS frequencies (including revised Tables 9.1-5 through 9.1-9) assuming either a lower screening threshold or no credit for operator recovery action.

Response 722.27

As stated in the response to Question 722.26, the truncated sequences accounted for a total core damage frequency of only 5.4E-10, less than 0.03% of the total core damage frequency. Therefore, C-E does not believe that updating Tables 9.1-5 through 9.1-9 to include these sequences would provide any additional meaningful information or insights.

Question 722.28

Please provide details on how the frequencies of "Plant Accident Sequences" in Tables 9.1-5, 9.1-6, and 9.1-7 were obtained from the "Core Damage Sequences". Specifically, provide a table showing the branch split fractions of the CSET of figure 9.1-1 for each core damage sequence.

Response 722.28

The CSET in Figure 9.1-1 has six subsequences, each defined by specific set of faulted and unfaulted systems. Likewise, each core damage sequence is uniquely defined by a set of faulted and unfaulted systems as determined from the appropriate event tree. The "Plant Accident Sequences" presented in tables 9.1-5, 9.1-6, and 9.1-7 were developed by appending the definitions of the CSET subsequences to the definition of each core damage sequence, thus producing a possible six "Plant Accident Sequences" for each core damage sequence. This process was performed using the IRRAS 2.0 Beta Draft sequence definition module. These Plant Accident Sequences were then solved using the IRRAS 2.0 Beta Draft event sequence solution module in exactly the same way that the core damage sequences were solved. Only those Plant Accident Sequences with a frequency of greater than or equal to 1.0E-10 are presented in tables 9.1-5, 9.1-6, and 9.1-7. A table of split fractions of the CSET for each core damage sequence is not provided because split fractions were not developed and were not used. All Plant Accident Sequences were solved using full fault tree linking as described above.

Question 722.29

Please describe the rationale for selecting the pressure ranges presented in Table 9.1-1. For example, are they an outgrowth of accumulator discharge pressure?

Response 722.29

The pressure ranges presented in Table 9.1-1 are essentially a quantitative representation of a qualitative statement regarding the susceptibility to a DCH event. Low pressure events, such as large LOCAs, are not susceptible to DCH. High pressure events such as those typified by discharges through a cycling relief valve have a high susceptibility to DCH, and medium pressure events, such as small LOCAs or SGTRs, have a moderate susceptibility to DCH. A review of the Chapter 6 and 15 analyses for small LOCAs and SGTRs indicated that this type of event might have pressures in approximately the 1200 psia range at the onset of core damage. This value was established as the upper bound for the medium pressure range. The lower bound for the medium pressure range was established based on the capability of the safety depressurization valves. The design intent of the safety depressurization system was that a high pressure sequence could be reduced to a low pressure sequence by opening both valves. Analyses demonstrated that with both valves open RCS pressures in the range of 400 psia or less could be reached with in an appropriate time frame. With only one valve open, the RCS pressure would be greater than 400 psia but less than 1200 psia. Thus, 400 psia was established as the upper bound of the low pressure sequences or the lower bound of the medium pressure sequences.

Question 722.30

Low pressure plant damage states are defined to have pressures less than 400 psia, and in the PRA do not contribute to direct containment heating (DCH). However, in Section 9.2.2.6.1, direct heating is said to be a problem for reactor system pressures as low as 250 psia. Thus, a sequence with a pressure of 350 psia would be assigned to a low pressure plant damage state but would still pose a threat for direct containment heating. Please address the apparent discrepancy between the pressure ranges used to define low pressure plant damage states and the threat of DCH, and the significance of this discrepancy on the results for early containment failure.

Response 722.30

In the early stages of the System 80+ level 2 analyses, a preliminary value of 250 psia was used as the break point between low and medium pressure sequences. This value was later changed to 400 psia. The 250 psia value in the text in section 9.2.2.6.1 should be 400 psia. This error will be corrected in the updated PRA Report. (See the response to Question 722.29 also.)

Question 722.31

Section 9.1.2.2 states that the second PDS parameter is "RCS pressure at vessel failure", and lists PDSs with medium or high RCS pressure at vessel failure in Table 9.1-8. These include PDSs 53 through 115. However, in Section 9.2.2.6.1, credit is taken for depressurizing these sequences is taken. Please clarify this apparent inconsistency. Is the second PDS parameter actually "RCS pressure at core damage"?

Response 722.31

The second PDS parameter is intended to represent RCS pressure at core damage as suggested. C-E is currently updating the System 80+ PRA and will correct the text in section 9.1.2.2.

Question 722.32

Core melt timing is not used for the definition of "plant Damage States" since it is argued that it primarily affects warning time. However, core melt timing is an important parameter in determining the time available to implement actions to prevent core damage and to mitigate core damage in-vessel. It also appears to be an important parameter for CET top event CFM3 (failure before core melt prevented). Please provide additional justification for not explicitly addressing core melt timing in the PRA, and explain further how time available for operator recovery was handled in view of the approach taken.

Response 722.32

A description of how operator recovery for core damage mitigation is provided on pages 2-13 and 2-14 in Section 2.5 of the System 80+ PRA Report, DCTR-RS-02, Rev. 0, January, 1991. The time from the initiating event until the onset of core damage was assessed in order to determine the amount of time available for an operator to perform a recovery action to prevent core damage. In-vessel mitigation of core damage was not credited at all in the System 80+ PRA. The CET top event, CFM3, was included to address a specific set of accident sequences. These sequences involve a primary system breach with successful inventory and core heat removal control, but with failure to remove decay heat by cooling the containment. For these sequences, it was assumed that if containment heat removal was not restored, the containment would eventually fail on overpressure. It was further assumed that on containment failure, the safety injection pumps would trip and core damage would occur shortly thereafter. Again, core melt timing was not a critical issue. The PDS parameters were selected based on their perceived importance to the progression of the severe accident following the onset of core damage. The time of core damage has a limited effect on the progression of the severe accident following the onset of core damage. Therefore, C-E believes that core damage timing was adequately treated in the level 1 PRA analyses and that it need not be treated as a primary Plant Damage State (PDS) parameter.

Question 722.33

Explain why the status of containment isolation is not treated as a plant damage state parameter. Since top event question CFM3 handles only those cases where containment failed before core damage due to loss of containment cooling, describe how other containment failures before vessel failure were handled (such as containment penetration failure in seismic events or pre-existing containment leakage).

Response 722.33

For the System 80+ PRA, it was decided to treat containment isolation failure as an independent event and address it directly in the containment event tree (see top event CFM2 on figure 9.2-1). The treatment of containment isolation failures is discussed in Sections 9.2.1.1.3 and 9.2.2.4 of the System 80+ PRA Report, DCTR-RS-02, Rev. 0, January, 1991.

Question 722.34

For the sequences whose release point is outside containment, such as a bypass sequence, in-vessel accident progression should not be significantly affected by whether the cavity is dry or wet since essentially the same amount of water is available for core cooling (if some of the IRWST water is used to flood the cavity before vessel failure, the vessel failure would actually occur somewhat earlier). Accident progression following vessel failure, however, would be influenced by the availability of water in the cavity. Please discuss the rationale for not including a PDS parameter combination deletion rule (in Table 9.1-2) which would further simplify the initial plant damage states by combining those PDSs with releases outside containment and wet/dry cavity conditions (e.g., PDs 13 and 15, and 14 and 16 in Table 9 1-3). In this regard please describe the difference in how PDS 14 and 16 were handled in the containment event tree analysis, and whether their containment failure modes were different (Table 9.2-18).

Response 722.34

The PDS parameter combination deletion rules were established to delete physically impossible combinations or combinations that would not influence the progression of the severe accident or the releases. As stated in the question, accident progression following vessel failure would be influenced by the availability of water in the cavity. In the case of the PDSs with releases outside containment, the primary effect is on the nature of the releases. With the cavity dry, isotopic releases associated with concrete ablation would occur whereas with the cavity wet, there would be a difference in the type of radioisotopes released. The treatment of PDSs 14 and 16 is described throughout section 9.2 of the System 80+ PRA Report, DCTR-RS-02, Rev. 0, January, 1991. For both PDS 14 and PDS 16, containment failure occurred prior to core damage. For PDS 14, containment failed on overpressure prior to core damage due to loss of containment heat removal while PDS 16 involved a containment bypass due to an interfacing system LOCA (see Table 9.1-8).

Question 722.35

Plant Damage State parameter deletion rule #3 would preclude interfacing LOCAs with a break area greater than 0.5 square feet. Provide the justification for excluding breaks in piping systems greater than this area, as well as multiple pipe breaks induced by seismic events.

Response 722.35

Plant Damage State parameter deletion rule #3 was based on the assumption that the largest line for an interfacing LOCA was a 10 inch line. This includes all lines except the RHR suction lines which use 16 inch pipe. C-E is currently updating the System 80+ PRA. Plant Damage State parameter deletion rule #3 will be revised and the PDSs will be re-evaluated. This change is not expected to affect the results.

Question 722.36

The reactor cavity is defined to be wet if containment spray is available (Section 9.1.2.7), yet in the list of initial PDSs (Table 9.1-3) several damage states exist in which sprays are available and the cavity is dry. Please explain this discrepancy. Does "spray available" imply that spray has always been actuated and is operating? Why is there no PDS deletion rule to address this situation?

Response 722.36

Just prior to completion of the System 80+ PRA, the Cavity Flood System design was changed such that when the Containment spray system was operating, a portion of the spray runoff into the holdup volume would be diverted into the cavity. This design change was factored into the level 2 analyses. However, several of the initial PDSs in Table 9.1-3 were not corrected, and no specific PDS deletion rule created for the situation. Since then, the Cavity Flood System design has changed again. CE is currently updating the System 80+ PRA and will correct the PDSs listed in Table 9.1-3 to reflect the final Cavity Flood System design. (See 722.9)

Question 722.37

Please identify the PDSs that involve containment failure due to a failure of containment heat removal capacity. Indicate what percentage of these PDSs are initiated by earthquakes. Identify and discuss any plant features considered by C-E to prevent some of these PDSs.

Response 722.37

PDSs 14, 24, 25, 67, 131, and 133 involve containment failure due to failure of containment heat removal. These PDSs have a total frequency of 2.24E-09. These PDSs do not include any seismic sequences. C-E has, in conjunction with EPRI ALWR Utility Requirements Document efforts, considered containment vents and external containment sprays as potential ways to prevent these PDSs. Containment vents were not included because of utility concerns for adverse public reaction. An external containment spray system was not included in the design because it has not been established that such a system would be able to remove sufficient containment heat to be of benefit. C-E is preparing a Severe Accident Mitigation Design Alternatives (SAMDA) evaluation Report for submittal in March. This report discusses the cost-benefit evaluation of these and other SAMDAs in more detail.

Question 722.38

Please explain how Branches C and D are possible or necessary in the CSET, i.e., how the containment heat removal is available when containment spray is not available. Please provide some examples (Table 9.1-5 does not show any sequences with these designations.).

Response 722.38

Containment cooling can be achieved by cooling the IRWST inventory using the RHR pumps and RHR heat exchangers. Given that the containment spray system has failed in certain ways, it is possible that containment cooling via this path would still be available. Table 9.1-5 does not contain any examples of C or D sequences because all C and D sequences had frequencies of less than 1.E-10.

Question 722.39

Discuss why there is not a corresponding E sequence for every F sequence in Table 9.1-5 (e.g., no LOFW-9E, no TOTH-9E, no ISL-9E), and a corresponding F sequence for every non-LOCA E sequence.

Response 722.39

There is not a corresponding E sequence for every F sequence nor a corresponding F sequence for every non-LOCA E sequence in Table 9.1-5 because these missing E or F sequences had frequencies less than 1.E-10.

Question 722.40

Discuss why there is not a corresponding B sequence for every A sequence in Table 9.1-5 (only two B sequences are shown in Table 9.1-5) Is this because frequencies of most B sequences are below 1.E-10? (e.g. ATWS-26B)

Response 722.40

There is not a corresponding B sequence for every A sequence in Table 9.1-5 because most of the B sequences had frequencies less than 1.E-10. A "B" sequence is defined as a sequence in which containment spray is working but containment heat removal is not available. By definition, the containment spray system is working. Thus, power and component cooling water (CCW) are available. The only way that containment heat removal could be lost, given that the Containment Spray System is operating, is if the valves controlling ccw flow through the heat exchanger are all failed. This is a low probability sequence.

Question 722.41

Each release class in the PRA was characterized by a representative sequence corresponding to the highest frequency CET end states. Please discuss the review performed to ascertain that the sequences selected represent the bounding or most conservative conditions for release.

Response 722.41

The PDSs chosen to represent each release class were chosen primarily based on frequency. The next higher PDSs were reviewed to determine, based on engineering judgement, if there was any outstanding indication or reason to believe that they might have larger or more energetic releases than the selected PDS.

Question 722.42

Please explain why further separation of Release Classes (RC) 1.1 through 1.8 are necessary. Specifically address why the top events STC7 and STC9 are relevant for CFM1 (containment bypass sequences), when the only relevant question here appears to be STC10 and thus RCs 1.7 and 1.8. (Your results show one entry each for classes 1.4 and 1.7, and none for the rest of these classes.)

Response 722.42

The CET was developed prior to propagating PDSs through it. Branch points were excluded only if they were deemed to be physically impossible or would not influence the containment failure mode or the type of release. Top event STC7 addresses the potential for vaporization releases which are primarily affected by whether or not the cavity is flooded. There is no physical reason the cavity would not be flooded for a containment bypass sequence. Thus, STC7 is a valid question for containment bypass sequences. STC9 addresses the potential for reduction of the fission product content of a release via the scrubbing effect of containment sprays. Further review of the bypass sequences indicates that the operability of the containment spray would have little impact on the releases for bypass sequences because of the primary release paths. C-E is currently updating the System 80+ PRA. The CET will be revised to reflect this change.

Question 722.43

Clarify whether top event STC10 is relevant for isolation failure. If so, explain why there are no entries in RCs 2.1, 2.3, and 2.7 in Table 9.2-22. We would expect that the PRA would have identified some containment isolation failure sequences which release to the auxiliary building. Please discuss and justify the lack of such sequences for the System 80+ design.

Response 722.43

As discussed in the response to Question 722.42, during the development of the CET, Branch points were deleted only if they were physically impossible or did not affect the containment failure mode or nature of the potential releases. Question STC10 is a valid question for release class 2. In the System 80+ PRA, a single probability for isolation failure was applied for all initiators. Containment isolation failures were not investigated individually, so specific isolation failure locations are not identified. As discussed in Section 9.2.2.12 of the System 80+ PRA report, DCTR-RS-02, Rev. 0, January, 1991, top event STC10 was primarily used to determined whether there was a potential for attenuation of the releases due to passage through extensive portions of the auxiliary building. In the analysis, it was assumed that this would occur only for the interfacing system LOCA. Because the specific locations of the isolation failures were not known, it was conservatively assumed that these failures would all occur in a location that would not result in significant attenuation of the releases. Thus, STC10 was answered "no" for all isolation failures. C-E is currently updating the System 80+ PRA. This update will include a more detailed evaluation of isolation failures. The CET and the propagation of PDSs through the CET will be revised as needed.

Question 722.44

Clarify whether top event CFM2 is the same for internal and seismic events, and whether CFM2 includes sequences with isolation failure due to seismic events.

Response 722.44

CFM2 is the same for both internal and seismic events. CFM2 does not include sequences with isolation failure due to seismic events. It was assumed that there would be no significant increase in isolation failure due to seismic events because the containment is well embedded in the concrete of the auxiliary building and they both have a common basemat.

Question 722.45

op events CFM2 and CFM3 both appear to represent sequences where containment fails before vessel failure. Please explain the difference between these two events , and whether CFM2 is actually a subset of CFM3. Discuss whether CFM3 includes containment failure due to ATWS, or only includes containment failures due to long term loss of containment cooling.

Response 722.45

Top event CFM3 represents a "catastrophic" containment failure on overpressure occurring prior to the onset of core damage specifically as a result of loss of long term containment heat removal. CFM2 represents a failure of containment isolation such as a failure of a valve or hatch or as a result of penetration leakage regardless of the core damage initiator. CFM3 pertains to all sequences in which core damage occurs as a direct result of loss of long-term containment heat removal regardless of the transient initiator. There is no ATWS sequence other than a loss of long-term containment heat removal that leads to containment failure prior to core damage.

Question 722.46

Describe how CDF is calculated for sequences which are caused by loss of containment cooling (i.e., those sequences where the top event CFM3 is relevant). Is loss of containment cooling always assumed to result in the core damage in these sequences?.

Response 722.46

Sequences involving containment failure prior to core damage due to the loss of long-term containment heat removal are always assumed to result in core damage. The CDFs for these sequences were calculated using fault tree linking for the fault trees for the system failures leading to the long-term loss of containment heat removal. See section 4.1.1.4 of the System 80+ PRA Report, DCTR-RS-02, Rev. 0, January, 1992 for an additional dis-ussion of this type of sequence.

Question 722.47

Scrubbing of fission products in the IRWST can be significant for the sequences where the fission is released through the IRWST during core damage perioc before vessel failure (compared to LOCAs). In view of this, explain why early scrubbing is not treated as a top event question in the CET.

Response 722.47

Early scrubbing of the fission products in the IRWST is treated via the "Release Point" parameter in the PDS definitions. It is felt that this provided adequate treatment of early scrubbing. Therefore a separate CET top event is not needed.

Question 722.48

Please justify how fission product scrubbing is possible when the cavity is dry in Figure 9.2-1, e.g., STC9 would not appear to be relevant for "STC7 NO" branches when the vessel and containment are failed. Also explain why there are no entries in Table 9.2-22 for many RCs representing these sequences while they are shown in your CET (e.g., RCs 1.5, 1.6, 2.5, 2.6, 4.3, 5.3, 5.5, 6.1 and 6.3)

Response 722.48

For the original System 80+ cavity flood system design, it was possible for the cavity to be dry with the containment spray system operating. (Note: top event STC9, "Fission Product scrubbing available" pertains to the time prior to containment failure.) Just prior to completion of the level 2 analyses, the cavity flood system design was changed such that the cavity would be flooded whenever the containment sprays were operating. Given the design change, top event STC9 was no longer appropriate for "STC7 NO" branches. The CET was not modified, but the affected PDSs were repropagated through the CET. That is why there are no entries for the listed RCs. There have been additional Cavity Flood System design changes since the submittal of the System 80+ PRA. C-E is currently updating the System 80+ PRA and will modify the CET to reflect effects of the final Cavity Flood System design. (see 722.9)

Question 722.49

Explain why RC 3.2 is necessary, and whether top event STC7 is relevant for the "CFM3 NO" branch. Doesn't this sequence presume a wet cavity?

Response 722.49

The "CFM3 NO" branch pertains to sequences for which the containment fails on overpressure due to loss of long-term containment heat removal prior to core damage. In general, containment spray is not available for these sequences, so the cavity is not automatically flooded by the runoff from the containment spray into the holdup volume. The cavity flood system requires manual actuation. Thus, it is possible for the cavity to be dry for these sequences and RC 3.2 is a physically achievable release class.

Question 722.50

Explain why Question STC9 isn't asked for RCs 5.1 and 5.2. For example, since it is possible to have spray but no containment heat removal, wouldn't PDS 24 contribute to this category?

Response 722.50

Further review of the definition of STC9 and the definitions of release classes 5.1 and 5.2 indicate that STC9 is potentially a valid question for release classes 5.1 and 5.2 because it is possible, however unlikely, that containment spray will be available with no containment heat removal. Sequences with this set of conditions are expected to be filtered out with frequencies less than 1.E-10. C-E is currently updating the System 80+ PRA and will include this set of conditions.

PDS 24 would not contribute to any "late containment failure" (RC5.x) release class because it involves containment failure prior to core damage.

Question 722.51

Explain why RCs 5.3 through 5.6 are not the same as RC 6, since both require dry cavities and would be expected to lead to containment melt-through. Similarly, explain why [sic] on the "STC7 NO" lead to late overpressurization rather than melt-through since they would also have a dry cavity.

Response 722.51

A late containment overpressure failure can occur as a result of steam generation or a late hydrogen burn. Sequences with dry cavities can produce significant amounts of hydrogen, thus creating the potential for a late containment overpressure failure due to a late hydrogen burn before the containment melt-through occurs. Release classes 5.3 through 5.6 pertain to these sequences. Per the discussion in the response to Question 722.48, top event STC9 is not applicable for these sequences. C-E is updating the System 80+ PRA and will correct the CET.

Question 722.52

Since it is possible for containment melt-through to occur simultaneously with other containment failures, the top event question CFM6 should be asked for all other failure modes which have dry cavities. If CFM6 is not important for early containment failure modes, shouldn't it be still important for late failures and asked after CFM5?

Response 722.52

For this analysis, the CET top event question CFM6 means "Is containment basemat melt-through the containment failure mode?". Containment basemat melt-through does not occur simultaneously with other containment failure modes. Dry cavity sequences can result in early containment overpressure failure due to DCH or early hydrogen burn. If the dry cavity sequence does not lead to an early containment failure, it can lead to a late containment overpressure failure due to a late hydrogen burn. If the dry cavity sequence does not result in either an early containment overpressure failure or a late containment overpressure failure, it will lead to containment failure due to basemat melt-through. Containment basemat melt-through requires several hundred hours. For the other containment failure modes, the impact of the concrete ablation on the release fractions is addressed via CET top event question STC7.

Question 722.53

Explain why CFM6 and STC7 were not combined in the CET since the supporting logic models for these questions (Figures 9.2-4 and 9.2-5) are essentially identical.

Response 722.53

See the response to Question 722.52.

Question 722.54

Please provide a discussion of how ATWS accidents are handled in the CETs.

Response 722.54

There are felt to be no special properties unique to ATWS sequences that would affect the progression of a severe accident. Therefore , the ATWS accident sequences are mapped into Plant Accident Sequences and then Plant Damage States in the same manner as the core damage sequences for the other accident initiators.

QUESTION 722.55

Containment liner melt-through due to direct contact of debris with the containment vessel is not addressed as a potential containment failure mechanism in the CETs. Please provide justification for excluding this challenge from consideration in the PRA. Include a discussion of CE System 80+ design features which minimize its potential (e.g., concrete walls which separate the liner from the cavity region), and an assessment of the effectiveness of these features. Specifically address the challenge represented by the mass of debris assumed to escape the cavity during DCH events.

RESPONSE 722.55

The containment liner melt-through due to direct contact of debris with the containment vessel in the upper compartment is not included as a potential containment failure mechanism due to its low probability of occurrence. However, basemat melt-through in the cavity region was considered as a failure mechanism in the PRA.

Containment liner melt-through in the upper compartment of the containment was not considered since the potential for escape of sufficiently energetic debris projectiles from the cavity region was considered to be negligible. The debris dispersion into the upper compartment was assumed to be minimal due to the debris deentraining and retention features of the cavity design (e.g. core debris chamber designed to minimize the flow rate of the debris, and the tortuous pathway that the debris would have to traverse in order to communicate with the upper compartment). In the System 80+ design, any debris leaving the reactor cavity will be delivered to the Refueling Water Pool (RFWP) region via the seal table. The concrete walls of the RFWP surround the seal table and extend vertically for 20 feet. The RFWP volume provides substantial area to decelerate any steam and debris exiting the reactor cavity. From this position the decelerating debris would have to travel more than another 100 feet before contact with the upper containment sphere. Therefore, the containmnet missile challenge represented by ejected corium debris can be considered negligible. CE is currently updating the System 80+ PRA. The potential for containment melt-through in the upper compartment region will be re-evaluated, and if warranted, would be considered at that time.

Question 722.56

In calculating the probability of late containment failure due to steam overpressure, a non-recovery probability of 0.01 was assigned for all internal event cutsets involving failure of components outside containment based on the assumption that containment failure would not occur for at least 48 hours. A value of 0.1 was assigned for seismic events. While 48 hours is a significant amount of time in which to implement recovery actions, these values appear optimistic recognizing that the ability to recover might be hampered by several factors. These include lack of spare/replacement parts and components, the ability to cross-connect all necessary systems (due to lack of pre-existing cross-connect capability), the ability to access areas of containment in order to perform recovery/repair, and the ability to obtain offsite support during seismic events. Please justify credit taken for recovery of containment heat removal in view of these concerns. Include in your response a description of provisions in the System 80+ design which would facilitate long term recovery actions to recover heat removal (e.g., ability to connect containment spray headers to external water sources and to remove water from the containment sump). Also, identify components/parts that are assumed to be available (in stock or accessible on short notice). Describe commitments/interface requirements related to assuring a high recovery factor for containment heat removal.

Response 722.56

C-E understands the NRC concern about the non-recovery factors assigned for the recovery of containment cooling systems after 48 hours in severe accident late-overpressurization scenarios. C-E is currently updating the System 80+ PRA. The recovery of containment cooling will be reassessed as part of this update.

Question 722.57

In Section 9.2.1.1.2, containment isolation failure is defined as leakage exceeding 200 volume percent per day. Please provide the basis for selecting 200 percent per day leak as criterion of containment isolation failure. Clarify how this leakage rate was calculated, e.g., was it based on normal containment pressure or accident pressure, were flow restrictions/loss coefficients accounted for. Describe the expected leakage rate due to severe accident challenges, such as containment temperatures and pressures beyond the design values, and how leakage between the design basis leak rate and 200 percent per day were categorized.

Response 722.57

The containment isolation leak rate was established based on a review of NUREG/CR-1150. It is intended to represent the leakage under design basis temperatures and pressures. This information was used to back calculate an effective leak area which was then used for determining leakage and releases over the temperature and pressure ranges calculated by MAAP when determining the releases. The effective leak area, based on the originally assumed leak area, was used to represent all containment isolation failures. C-E is currently updating the System 80+ PRA and will include a more mechanistic evaluation of containment isolation failures and their ramifications.

Question 722.58

Containment isolation failure probability of 2.0E-3 per demand was taken from WASH-1400. Justify this approach in view of the fact that isolation probability strongly depends on the specific design of the containment (such as number of penetrations), and the reliability of the isolation system. Clarify whether failure of containment isolation concurrent with the core damage due to a seismic event is included in this number.

Response 722.58

C-E concurs that the isolation probability is strongly dependent on the specific containment design. The WASH-1400 isolation failure value was used as a preliminary estimate, and design requirements have been established to ensure that it is achievable. C-E is currently updating the System 80+ PRA and will include a more mechanistic evaluation of containment isolation failures and their ramifications.

QUESTION 722.59

In Figure 9.2-2, DCH loads are combined with hydrogen combustion loads in estimating the pressure rise associated with the DCH event. Please discuss why rapid steam generation loads were also not combined in estimating the combined pressure rise for DCH events. Also clarify whether DCH and rapid steam generation are considered mutually exclusive in the PRA.

RESPONSE 722.59

This question addresses the structure of the Supporting Logic Model (SLM) for the event DCHFAIL on page 2 of Figure 9.2-2. In establishing the DCH containment challenge the deterministic peak pressure analysis was performed
using the Modular Accident Analysis Program (MAAP) code. In this context the DCH containment challenge included all non-combustion aspects of the ejected debris containment heatup process. Specifically, MAAP calculated contributors to DCH include the following:

(a) energy transfer from the fragmented corium to the containment atmosphere,

(b) energy transfer from the fragmented corium directly to the stainless steel
containment shell, and

(c) rapid steam generation due to corium-water interaction in the reactor
cavity.

This approach was taken since water availability during DCH was low and its contribution to the limiting DCH pressure spike was small. Thus, rapid steam generation loads were also included in the calculation of the combined pressure rise during DCH events. However, DCH event and rapid steam generation event were considered to be mutually exclusive containment challenges in the PRA. A rapid steam generation containment challenge implies the existence of sufficient water in containment to preclude the conditions necessary for a DCH challenge. Conversely, a DCH containment challenge implies that the amount of water in contianment would be insufficient for a rapid steam generation challenge.

QUESTION 722.60

In the quantification of containment failure probability (DCHFAIL) it appears that double credit is incorrectly taken for the cavity trapping core debris. Furthermore, no basis for establishing a baseline (pre-existing) containment pressure for this event is presented. The first credit is taken by assuming that the cavity design is effective in eliminating debris dispersion 90 percent ofthe time. The second credit is taken by assuming that for the 10 percent of the time the cavity is ineffective, the peak pressure will still be limited to 100 psia (partly due to debris holdup in the cavity). For that fraction of sequences in which the cavity is ineffective, the pressure rise should be based on ineffective holdup of debris (e.g., a pressure rise of 90 psi for NUREG-1150). Please address this apparent inconsistency. Also, provide a baseline containment pressure specific to the dominant high pressure sequences for the System 80+ containment design, rather than assuming that the NUREG-1150 value is applicable.

RESPONSE 722.60

The quantification of the containment failure probability (DCHFAIL) does not double credit the debris retention capability of the System 80+ cavity design. The 100 psia peak pressure is established from a pre-reactor vessel failure value of 20 psia followed by a DCH pressure spike of about 80 psi. The 80 psi DCH spike is realized from MAAP analyses which assume ineffective holdup of the core debris in the cavity region. This peak DCH spike value for the System 80+ design can be justified in relation to the NUREG-1150 value of 90 psi spike by assuming that the DCH pressure spike for large dry containment is directly proportional to the reactor power and inversely proportional to the containment free volume as follows:

$$[\text{DCH } \Delta P] = (90 \text{ psi}) \times \frac{(3800 \text{ MWt}) \times (2.6 \times 10^6 \text{ cu. ft})}{(3250 \text{ MWt}) \times (3.4 \times 10^6 \text{ cu. ft})}$$

$$= 80 \text{ psi.}$$

The high System 80+ containment failure pressure provides sufficient margin such that assumed DCH loadings can be accommodated by the System 80+ containment design without failure of the containment shell.

QUESTION 722.61

In Section 9.2.2.6.1, the reactor cavity is said to include a convoluted path that would disentrain corium before it reaches the upper containment. Based on this design feature, the probability of dispersing corium into cavity is assigned a probability of 0.1. Please provide a description and design details of the debris entraining features of the cavity, and the technical basis for the judgements that the probability of dispersing corium into upper containment is essentially zero and that a value of 0.1 for CAVTGEOM is a conservative estimate. Also provide an assessment of the effect of higher values of CAVTGEOM on the risk results.

RESPONSE 722.61

The cavity region of the System 80+ design is designed to minimize debris entrainment and subsequent debris dispersal in the upper compartment of the containment. System 80+ is equipped with an offset core debris chamber designed to de-entrain and trap the debris ejected during a reactor vessel breach. A discussion of the debris chamber design can be found in Reference 1. In summary, the reactor cavity debris chamber and exit shaft have been designed such that high inertia corium debris would de-entrain and collect in the debris chamber while the lower inertia steam/hydrogen/air mixture would negotiate a right angle turn and exit the reactor cavity via the seal table. Once deposited in the debris chamber, the debris would be difficult to re-entrain since the retention zone should exhibit a low velocity recirculation flow pattern. Any corium negotiating the 90 degree turn woud be de-entrained by the reactor cavity concrete ceilings and seal table structure.

The estimate of the fraction of the debris able to negotiate the turn into the vertical cavity shaft was established using a Sandia correlation for debris impingement determined from high pressure melt ejection tests (See Reference 1, Appendix D). Application of this model to the System 80+ cavity geometry results in a prediction that 90% of the corium debris would be de-entrained into the debris chamber and that 10% of the debris could potentially negotiate the turn into the reactor cavity shaft. Consequently, the probability of dispersing corium into the upper compartment was conservatively assumed to be 0.1. This estimate neglects the significant debris de-entrainment capability of the cavity ceiling and internal shaft structures and walls.

Reference 1. Advanced Reactor Severe Accident Program, "Prevention of Early Containment Failure Due to High Pressure Melt Ejection and Direct Heating for Advanced Light Water Reactors", Prepared by TENERA, SAROS, and Fauske & Associates for EG&G Idaho and the U.S. Department of Energy under Contract No. DE-AC07-76ID01570, March 1990.

Question 722.62

In estimating the probability of containment failure due to DCH, the study assumes a 90 percent probability that high pressure sequences (other than SLOCAs) will be reduced to low pressure sequences as a result of creep rupture in the hot legs or surge line. (SLOCAs, also high pressure sequences, are assigned a 0.01 probability of failure). While temperature-induced failures may be highly likely under idealized flow conditions, realistic accident sequences can be expected to involve perturbations which would disrupt or prevent the natural circulation flows needed to produce piping failure. Examples include actuation of PORVs or partially successful operator attempts to add water. These details are generally beyond the level of detail in modelling in the PRA. Accordingly, a lower value for probability of hot leg/surge line failure would appear warranted. Please provide an assessment of the effect of lower hot leg/surge line failure probabilities on the risk results.

Response 722.62

If a lower probability of hot leg/surge line failure was assigned for high pressure melt through sequences, the split fraction probability of an early containment failure due to DCH would increase and the split fraction probabilities for late containment failures would decrease. The total magnitude of the change in split fractions would depend on the probability assigned for hot leg/surge line failure. The overall plant risk would increase because the releases calculated for an early containment failure due to DCH tend to be greater than the releases calculated for the other early and late containment failure modes.

Question 722.63

Since the reactor cavity in the System 80+ design is configured to minimize the dispersion of the corium into the upper containment, the associated flow restrictions may tend to increase the potential for local pressurization of the reactor cavity due to DCH and rapid steam generation phenomena. This pressure may cause the relocation of the vessel, other RCS system components, and containment penetrations. Please provide an assessment of the loads on the cavity walls and vessel supports due to high pressure vessel failure.

Response 722.63

The response to Question 722.1 provides a description of the geometry of the System 80+ cavity and adjoining areas. As can be seen from this description, the System 80+ cavity design provides a convoluted path that will dis-entrain the corium without significantly impeding the steam flow. The cross-sectional areas of the flow path to the upper containment compartment are large enough to prevent overpressurization of the cavity and adjoining areas. Thus, the loads on the cavity walls and vessel supports will remain within acceptable limits.

QUESTION 722.64

The early hydrogen burn logic tree does not reflect the potential for large hydrogen burns coincident with reactor vessel failure at high pressure. In such sequences, a significant quantity of hydrogen would be released at a rate far greater than the igniters can handle at low concentrations. Please discuss how the challenge to containment from large hydrogen burns at vessel failure is addressed.

RESPONSE 722.64

The potential for large hydrogen burns coincident with reactor vessel failure is addressed within the DCHFAIL and EH2BFAIL elements of the early containment failure logic diagram.

The early containment failure logic diagram (ECFAIL) presented in Figure 9.2-2 Page 1 (Page 9-40 of System 80+ PRA Report) denotes early containment failure occurring due to "DCH with unconditional hydrogen burn", "early hydrogen burn", or "rapid steam generation". For the most part, large hydrogen burns coincident with reactor vessel failure at high pressure are modeled in the DCHFAIL containment logic (Page 9-41). DCH occurs coincident with RV failure. To assure that large hydrogen burns are considered, all hydrogen produced up to the point of reactor vessel failure was unconditionally burned coincident with the DCH pressure spike. For this event no reduction in the available hydrogen source due to igniters was credited.

For high pressure reactor vessel failure scenarios that do not result in DCH, the large hydrogen burn potential is modeled through the "early hydrogen burn" supporting logic model (EH2BFAIL, Page 9-44 of System 80+ PRA report). This diagram shows that the three conditions, (1) "igniters fail to burn at low flammability level (IGNITFL)", (2) "early H_2 burn occurs (EH2BURN)", and (3) "containment strength cannot withstand H_2 burn pressure spike (EH2BSTREN)", must be true to result in containment failure due to the early hydrogen burn logic. Success of item 1 would indicate that hydrogen is generated at a rate far greater than the igniters can handle. This will allow for large buildup of hydrogen within the containment potentially leading to a large hydrogen burn. Success of item 2 suggests that large early hydrogen burn occurs. Item 3 shows that a pressure spike due to large hydrogen burns is considered in relation to the containment strength.

Question 722.65

In assessing the potential for containment failure due to late hydrogen burn (Section 9.2.2.7.2), the probability of a pressure rise of sufficient magnitude to challenge containment (LH2BSTERN) was assigned a value of 0.0 for all PDSs in which containment sprays are available, and 0.5 for all other PDSs. The former value appears to be based on fact that baseline pressures would be low when sprays are available, combined with an implicit assumption that hydrogen ignition will always occur at low concentrations due to igniters or random ignition. This value does not appear to be valid for sequences in which an ignition source is unavailable until late in the sequence. Please justify a P(LH2BSTERN) value of 0.0 for sequences in which sprays operate but hydrogen burn does not occur until late in the sequence (e.g., due to late recovery of igniters). Also justify the value 0.5 for sequences without sprays. As part of this response provide a discussion of the containment pressure rise calculations which form basis of these probability estimates, and identify major input and modelling assumptions.

Response 722.65

C-E has reviewed the late hydrogen burn calculation and understands the NRC's concern regarding the split fractions assigned to LH2BSTREN. C-E is currently updating the System 80+ PRA and will re-evaluate the potential for a late hydrogen burn challenging containment.

Question 722.66

In assessing the threat from late hydrogen burn (Section 9.2.2.7.2.2), the probability of high hydrogen concentration existing late in an accident sequence (LHIH2CON) is modelled as a product of: (1) the probability that an ignition source was not available to burn hydrogen as it was generated, (2) the probability that hydrogen burn did not occur earlier (NOEH2BURN, and (3) the probability that core concrete interaction occurs to generate hydrogen (H2CCI). This logic (see Figure 9.2-3, page 2) appears to have several deficiencies as identified below:

- the probability IGNITFL is redundant with the probability NOEH2BURN, and in effect takes double credit for igniter system operation. IGNITFL is already imbedded in EH2BURN and does not appear appropriate as a separate event in determining LHIH2CON. The probability NOEH2BURN also appears to be redundant with the probability LSPARKX in the proceeding branch. Specifically, both NOEH2BURN and LSPARKX in effect are equivalent to the probability that an ignition source did not exist early (1. -ESPARK). Since this probability is already imbedded in NOH2BURN, the probability LSPARKX does not appear necessary as a separate condition.

- it is conceivable that greater than 75 percent of the fuel cladding can be oxidized in-vessel, and can pose a challenge to containment if it accumulates. Thus, it does not appear that core concrete interaction is a requisite condition for a large hydrogen burn. Sequences in which the containment is initially steam inerted and subsequently deinerted by sprays or natural processes are an example of such challenges.

Please address this apparent inconsistency in the analysis.

Response 722.66

The element, IGNITFL, appears in the branch, LHIH2CON, and as an independent element in figure 9.2-3 primarily to describe the conditions that lead to a high hydrogen concentration late in the sequence. The two occurrences of this element are redundant, but this does not affect the calculation. Although they do represent similar conditions, the element, NOEH2BURN is not completely redundant to LSPARKX as can be seen from the logic on page 3 of figure 9.2-3 and page 5 of figure 9.2-2.

After further review of the logic for late hydrogen burn and the logic for early containment failure, C-E concurs that core-concrete interaction may not be the sole requisite condition for generating sufficient hydrogen for a large hydrogen burn late in the sequence. As stated in the response to Question 722.65, C-E intends to re-evaluate the issue of late hydrogen burns as part of the current update to the System 80+ PRA. This issue will be covered as part of that re-evaluation.

Question 722.67

In Section 9.2.2.6.3, the probability that the RCS is at high pressure (RCSPHIGH) is said to be determined directly from the PDS definition, and is assigned a value of 1.0 for high, 0.5 for medium and 0.0 for low pressure PDSs. However, the associated logic tree (Figure 9.2-2, page 7) refers back to page 3 of Figure 9.9-2 for value of RCSPHIGH. When the basic event probabilities are propagated up through the latter, the values of RCSPHIGH obtained are 0.1 for high, 0.5 for medium, and 0.0 for low pressure PDSs. These values are inconsistent with the values presented on page 9-81 for high pressure PDSs.

Response 722.67

C-E concurs that the values presented for RCSPHIGH in section 9.2.2.6.3 are inconsistent with the values that are calculated based on page 3 of figure 9.2-2. The values presented for RCSPHIGH in section 9.2.2.6.3 should be based on page 3 of figure 9.2-2 and not determined directly from the PDS. C-E is currently updating the System 80+ PRA. Sections 9.2.2.6.1 and 9.2.2.6.3 will be corrected as part of this update.

QUESTION 722.68

Please provide details regarding the calculation performed to predict the containment pressure rise associated with rapid steam generation. Identify and justify major parameters and assumptions including: RCS pressure at vessel failure; mass, composition, and temperature of debris in the lower plenum at vessel failure (including fraction of total core and percent oxidized); rate of discharge into the cavity; mass of water in the reactor cavity and added over the period of interest; core debris quench assumptions; containment spray availability; and heat transfer to containment and structures over the period of interest.

RESPONSE 722.68

Rapid steam generation calculations were performed within the MAAP code using the EXVIN (Ex-Vessel Steam Explosion) subroutine. The MAAP Users Guide documentation on this subroutine describes the major assumptions, the modeling and implementation of the subroutine within the code. A copy of this description from the MAAP Users Guide is provided as Attachment 1.

The pressure spike due to rapid steam generation was established based on a low pressure failure of the reactor vessel (RV) and an initial RV lower head penetration radius of 0.052 ft which is the inside radius of a single In-Core Instrumentation (ICI) tube. Representative MAAP calculations indicate that the fraction of the core in the lower plenum is approximately 90 % and the amount of core oxidation at that time is between 55 to 60 %. In computing the debris temperature distribution, the melting temperature of the U-ZR-ZRO_2 eutectic mixture was assumed to be 4040 degree F, and the latent heat of fusion was assumed to be 107.48 Btu/lbm.

EX-VESSEL STEAM EXPLOSION

INTRODUCTION

For some accident sequences there may be water present on the reactor cavity or pedestal floor prior to vessel failure. In this case, as the degraded core material is released from the reactor vessel it would encounter water at a comparatively low pressure, and the potential would exist for a steam explosion. The experimental observations reported in References [1], [2], and [3] showed that a steam explosion could be triggered in water when the molten material contacts a wetted, solid wall. Contact with a solid wall would first occur when high temperature core material penetrates through the water and contacts the floor.

The purpose of the EXVIN is to calculate the mass of steam produced during the first interaction between debris and water. No structural effects are predicted because of the low potential for containment failure. Steam production after initial contact of debris with water is handled by subroutine PLSTM.

ASSUMPTIONS

It is assumed that an explosion occurs when debris contacts the floor. It is assumed that the maximum amount of debris involved is that which would be in a column extending from the floor to the water surface. This is the case for the first explosion. Succeeding explosions are ignored because much less debris would be involved. It is assumed that the energy transfer to water is complete over a single timestep and that the debris is quenched to water saturation.

MODEL

The potential for a steam explosion is evaluated after debris contact with the floor occurs. Contact occurs either when the time for debris fall has elapsed or when the defined interacting mass is accumulated, whichever comes first. After contact, the explosion occurs if more than 1 kg of debris and water are present, and if the debris is above the water saturation temperature.

The mass of corium interacting in the water is given by a conical frustum volume times the density. For a constant debris flow rate, the radius of the vessel failure increases linearly with time (see VFAIL).

Defining r_{po} = initial penetration radius and r_p = current penetration radius, x_w as the water height, and x_v as the vessel height, the radius of corium at the water surface is equal to

$$r_s = \frac{x_w}{x_v} \left(r_p - r_{po}\right) + r_{po} \qquad (1)$$

Therefore, the mass of corium available for the interaction is equal to

$$m_{cm} = \rho_{cm}\, x_w \frac{\pi}{2} \left(r_s^2 + r_{po}^2\right) \qquad (2)$$

where ρ_{cm} = the corium density.

The time for corium to fall is given by the free fall relation

$$t_f = \frac{-u_o + \sqrt{u_o^2 + 2g\, x_v}}{g} \qquad (3)$$

$$u_o = W_{cm}/\left(\rho_{cm}\, A_p\right) \qquad (4)$$

where

u_o = initial corium velocity,

g = acceleration of gravity,

W_{cm} = corium flow rate from the vessel, and

A_p = penetration area.

When either t_f is elapsed or the accumulated corium mass exceed m_{cm} given by (2), the resulting steam mass is calculated. m_{cm} is limited to be no more than the accumulated corium mass in containment.

The resulting steam mass is

$$m_{st} = \text{[illegible]} \frac{h_{cm} - h_{cmsat}}{h_{fg}} \quad (5)$$

where

h_{cm} = corium enthalpy,

h_{cmsat} = corium enthalpy at water saturation, and

h_{fg} = water latent heat.

IMPLEMENTATION

EXVIN is called by EVENTS if corium contact with the floor has not occurred yet. Contact is defined by either the elapsed time or accumulated mass as explained above. The steam explosion event code is only set if more than 1 kg of steam is produced. No steaming is calculated if there is either less than 1 kg corium or water.

EXVIN therefore only calculates contact and a steam explosion once. In EVENTS, the corium, water, and steam mass and energy variables are updated based on the energy and mass transfer from EXVIN.

REFERENCES

1. G. Long, "Explosions of Molten Aluminum and Water - Cause and Prevention", Metal Progress, May 1957.

2. P. D. Hess and K. J. Brondyke, "Cause of Molten Aluminum - Water Explosions and Their Prevention", Metal Progress, April 1969.

3. D. Buxton and W. B. Benedick, "Steam Explosion Efficiency Studies", NUREG/CR-0947, SAND 79-1399, November 1979.

Question 722.69

In estimating DCHSTERN, EH2BSTERN and RSGSTERN by comparing the pressure generated by DCH, hydrogen burn, and rapid steam generation with the containment strength, discuss whether and how the uncertainty ranges for the pressure estimates and the containment pressure capacity were taken into account.

Response 722.69

The uncertainty ranges for the pressure estimates and the containment pressure capacity were not taken into account beyond the use of the probability of containment failure versus pressure curve represented by equation 9.2-9 when estimating DCHSTREN, EH2BSTREN and RSGSTREN. C-E is currently updating the System 80+ PRA. As discussed in the response to Question 722.90, C-E intends to address uncertainties and sensitivities as part of this update.

Question 722.70

The logic tree associated with late containment failure (Figure 9.2-3, page 1) assumes that the core debris is coolable and core concrete interactions (CCI) are terminated in the cavity once the cavity is flooded (since it meets the requirement for the cavity floor area of the EPRI ALWR Requirement Document). Experimental studies indicate that core concrete interactions can continue in spite of the existence of an overlying water pool. Continued interaction would produce non-condensible combustible gases which can challenge long term failure of the basemat. Please provide an assessment of how the PRA results would be affected if, for some non-negligible fraction of the wet cavity sequences, core-concrete interactions were unmitigated by the overlying water pool. Specifically address the effect of continued core-concrete interaction on: the magnitude of late combustion events, the frequency of late containment overpressure failure, the frequency of containment basemat meltthrough failure, and the structural integrity of the load bearing concrete walls.

Response 722.70

If it were assumed that, for some non-negligible fraction of the wet cavity sequences, core-concrete interactions were unmitigated by the overlying water pool, the frequency of late containment overpressure failure sequences and the frequency of containment basemat melt-through sequences would increase while the frequency of intact containment sequences would decrease a corresponding amount. In addition, the isotopic content of the releases for some of the late containment overpressure failure sequences would also change due to the increased amount of material released by the ablation of the concrete althought the overlying water pool would scrub most of the fission products prior to reaching the containment atmosphere.

Question 722.71

Figure 9.2-3, page 1, shows that it is necessary for the corium to remain in the cavity (NODISPERS) for late containment failure to occur. Please discuss and justify the disposition in the PRA of those sequences where corium is dispersed into the upper containment but DCH does not result in failure of containment. In either case the core debris continues to generate decay heat and can possibly attack structures and generate additional combustible gases.

Response 722.71

These conditions were not covered in the System 80+ PRA. C-E is currently updating the System 80+ PRA. The potential impact of corium in the upper containment compartment will be re-assessed as part of this update. Because of the System 80+ cavity design, it is felt that little corium, if any, will be dispersed into the upper containment area.

Question 722.72

Discuss why the probability of melt-through given core concrete interactions (i.e., P(BMTCVM) in Figure 9.2-4, page 1) is not assigned a value of 1.0

Response 722.72

Section 9.2.2.8 in the System 80+ PRA Report, (DCTR-RS-02, Rev. 0, January, 1991), provides a basic description of the element BMTCVM. In general, given a dry cavity, ablation of the concrete of the base mat will proceed downward and, to a certain extent, laterally. The nearest point of entry to the subsphere area is slightly below and to the side of the cavity. As concrete ablation proceeds, the mass of corium will descend and will spread laterally. It was assumed that there was a 90% probability that the corium would ablate laterally into the subsphere area before the downward ablation had proceeded to a point where the corium in the ablated cavity was below the floor level of the nearest subsphere room. Thus, there was a 10% probability that the corium would not ablate into the subsphere and would continue to ablate downward through the full thickness of the basemat and the underlying stone or soil. It was assumed that in this case there would be no atmospheric releases.

Question 722.73

To determine revaporization release, the availability of steam generators late in the core-melt sequence (i.e., P(NOSSHR) in Figure 9.2-6) is checked based on the plant damage parameter "SGA (Steam Generator Availability)." Justify how the SG status is still relevant at the time of containment failure which in some cases occurs later than 48 hours.

Response 722.73

The parameter for Steam Generator Availability, SGA, is based on the availability of the steam generators at the time of core damage. There is no guarantee that the steam generator will continue to be available for long periods of time after core damage. C-E is currently updating the System 80+ PRA and will re-evaluate the appropriate means for establishing the value of NOSSHR for long time frame sequences.

QUESTION 722.74

It was argued that the holdup time for large and medium LOCAs is very short, therefore P(DEPOSIT)=0. Please justify this assumption in view of the fact that fission products are produced in the vessel for several hours and steam generators are available for some of these accident sequences during this period.

RESPONSE 722.74

P(DEPOSIT) is the probability of fission product plateout/deposition in the primary system. It is assumed to be zero for medium and large LOCA sequences since, for these scenarios, the primary system blowdown occurs so rapidly that the fission products have little time to plateout/deposit in the primary system. This assumption is considered to be conservative, since the fission products are transported from the primary system to the containment and are available for potential release to the environment following containment failure. This assumption is further supported by the fact that for significant plateout/deposition to occur the fission products must pass through the steam generators. For the vast majority of LOCAs (except perhaps LOCAs occurring in the pump suction line), the flow path of least resistance would be via the downcomer for a RCP discharge or the reactor vessel inlet nozzle break and via the hot leg nozzles for an outlet nozzle break. Thus, passage of significant quantities of fission products through the steam generator will be unlikely. While some amount of deposition is possible, increasing P(DEPOSIT) to a realistic value greater than zero will have a negligible impact on the results and conclusions of the PRA.

Question 722.75

Please identify the potential points of release from the containment into auxiliary or other adjacent buildings. Describe the fission product removal characteristics for each potential release path (e.g., flow paths, volumes, surface areas, and fire sprays if available). Discuss why only the interfacing system LOCAs are considered to have a release path through the auxiliary building sufficient to allow deposition of fission products.

Response 722.75

The potential points of release from the containment into the auxiliary building were not specifically identified except for the interfacing systems LOCA (ISL). For an ISL, the release point would be in the RHR pump room, low in the subsphere area. The path to atmosphere from this area is long, tortuous and restricted. In this case, it was assumed that the pathway to atmosphere was such that there would be deposition of fission products. For other containment failures, it was assumed that the containment failure point would, in general, be above the point at which the containment shell was imbedded in concrete. In these cases, it appeared that the release pathways would be such that there would be little deposition of fission products.

QUESTION 722.76

Provide the basis for the release energy values listed in Table 9.3-1.

RESPONSE 722.76

Table 9.3-1 of the System 80+ PRA Report (Document No. DCTR-RS-02, Rev 0) lists data for various release parameters for the System 80+ release classes. These data were obtained from analyses using the MAAP code. One of the parameters listed in the table is the energy release rate to the environment through a potential containment breach. The data for this parameter is given in the far right hand column of Table 9.3-1 under the heading "Release Energy". The MAAP code calculates this energy release rate in Btu/hr which is converted to yield the data in cal/sec. The quoted value in the table is the energy release rate at the start of the release for the most adverse accident scenario within a particular release class.

QUESTION 722.77

The sequence selected for RC6.2 was a large LOCA with vessel failure at 1.8h and basemat meltthrough at 330h, whereas the sequence selected for RC6.4 was a station blackout with vessel failure at 19h and basemat meltthrough at 180h. Intuitively, the LOCA sequence should result in more rapid failure of the basemat. Please explain why this is not the case and why the rates of meltthrough are significantly different for RCs 6.2 and 6.4.

RESPONSE 722.77

The difference in the basemat failure time for representative sequences for release classes 6.2 (large LOCA) and 6.4 (loss of offsite power, was largely due to assumptions made in the analysis which set the amount of concrete erosion necessary to fail the basemat For RC 6.4 the basemat failure distance was conservatively set at 10 feet. For release class 6.2 basemat failure was taken at a concrete erosion level of 15 feet. If RC 6.4 was assumed to fail at an erosion level of 15 feet of concrete, the meltthrough time would be extended from 180 hours to 350 hours. Thus, for a similar basemat failure criterion the LOCA sequence results in faster basemat meltthrough than that predicted for the station blackout sequence.

Question 722.78

In calculating source terms, releases were tracked for 24 hours following vessel failure for all release categories except RC3.1 (containment failure before core melt) where they were tracked for 24 hours following containment failure. However, in the consequence calculation, it is mentioned that "the fission product release fraction are based on the total release over a period of 24 hours from the onset of the release for each of the release classes." Please explain this discrepancy. It is noted that the EPRI ALWR Requirement Document states that the frequency of exceedance of 25 rem at 0.5 miles in the consequence calculations should be calculated for 24 hours following the release. (If "24 hours after vessel failure" is used, very small frequencies will be calculated to exceed 25 rem for many late containment failure sequences since there will be no containment failures, and therefore, no release before 24 hours after vessel failure for these sequences. It appears that the CE results indicate this is the case).

Response 722.78

In calculating source terms for the System 80+ PRA, the releases were tracked for 24 hours following initial releases. C-E is currently updating the System 80+ PRA. The text describing the release classes will be revised to clarify the time over which the releases were tracked.

QUESTION 722.79

Explain why the vessel failure time is longer for RC 5.2 than RC5.1. It appears that accidents in these classes are similar except that RC 5.2 is low pressure vessel failure and RC 5.1 is high pressure vessel failure. The vessel failure time for high pressure sequences (i.e., RC 5.1) should be later than for low pressure sequences since high pressure sequences lose water slower. Please explain why the containment failure times of these two sequences are substantially different.

RESPONSE 722.79

A review of the primary sequences for Release Classes 5.1 and 5.2 shows that the only difference between them is the presence of an induced hot leg failure prior to vessel failure for RC 5.2. Both sequences start as loss of offsite power accidents with the turbine driven auxiliary feedwater pump flow available for eight hours. For RC 5.2, the induced hot leg failure will cause the RCS to depressurize earlier than that for RC 5.1 and subsequently lead to an earlier actuation of the passive accumulators resulting in the delivery of additional water to a reactor vessel that is still essentially intact. For RC 5.1, the accumulators will not begin to discharge to the RCS until the RCS has begun to depressurize through the vessel breach. This timing is very important because, for RC 5.2, the accumulator water adds to the RCS inventory thereby extending the ultimate vessel failure time in comparison to that for RC 5.1 for which no accumulator water is available until the vessel breach.

Containment failure for RC 5.2 occurs earlier in comparison to RC 5.1, since (1) the induced hot leg failure for RC 5.2 adds more mass and energy to the containment rapidly, and (2) the actuation of the passive accumulators in RC 5.2 provides more water to the RCS to generate additional steam which is discharged to the containment atmosphere.

QUESTION 722.80

Table 9.3-2 shows that the release fractions for melt-through sequences (RCs 6.2 and 6.4) are generally lower than those of late containment failure sequences (RCs 5.1 and 5.2). If this is true, explain why is it necessary to flood cavity.

RESPONSE 722.80

RCs 6.2 and 6.4 represent dry sequences (no cavity flooding) for which containment failure occurs late in the scenario due to basemat melt-through. For these sequences containment failure occurred at 303 and 184 hours respectively with the resulting radiological releases of 31 and 628 rems. For RCs 5.1 and 5.2 sequences the cavity flood system is actuated. However, the cavity is dry well in advance of containment failure which occurs at 128 and 94 hours respectively. The dryout of the reactor cavity following actuation of the cavity flood system was a feature of the cavity flood system considered at the early stages of the System 80+ design. In light of this early design feature, which was employed in the RCs 5.1 and 5.2 sequences, the radiological release are in fact consistent with the time to containment failure (i.e., the earlier the containment failure time, the larger the radiological releases). The cavity flood system has since been improved to provide a continuous supply of water to the reactor cavity which is sufficient to permanently quench and scrub the corium debris. This improved design feature will be fully addressed in an updated System 80+ PRA submittal.

QUESTION 722.81

Explain why the release fraction for Group 3 "iodine" is less for RC 5.2 than for RC 5.1. These classes appear to include similar sequences except for reevaporation in RC 5.2.

RESPONSE 722.81

Table B5.3-2 of CESSAR-DC Appendix B contains a typographical error in the column heading "Group 3 Iodine" release fraction for RC 5.2. The table reads a value of 3.26E-2. The correct value is 3.26E-1. Comparison of this value against the "Group 3 Iodine" release fraction for RC 5.1 suggests that the release fraction for RC 5.2 is larger than that for RC 5.1. The "Group 3 iodine" release fraction consists of iodine as cesium iodide (CSI) and cesium as cesium hydroxide (CSOH). A corrected Table B5.3-2 will be submitted as part of a future CESSAR-DC amendment.

A comparison of release classes RC 5.1 and RC 5.2 is provided in response to Question 722.79. There are two reasons for the higher "Group 3 Iodine" release fraction for RC 5.2. First, RC 5.2 is a low pressure melt-through scenario caused by a late break in the RCS due to a hot leg creep failure. However, since the creep failure occurred after substantial core damage, an efficient path to the containment was established to release the iodine and cesium that was generated in the core subsequent to the damage. Second, the containment failure time was a full day sooner than that for RC 5.1, as explained in response to Question 722.79. This results in a shorter duration for any deposition or settling of CSI and CSOH in the containment structures, leading to a higher release fraction at containment failure in comparison to the release fraction for RC 5.1.

QUESTION 722.82

Comparing RC 4.1 to 4.2 and RC 2.2 to 2.4, fission product scrubbing by containment spray appears to be very effective even for early containment failure and isolation failure sequences. The release for these sequences should be very fast and the release duration short. Justify that there is sufficient time for spray to be effective for these sequences.

RESPONSE 722.82

The sprays are indeed very effective for scrubbing radionuclides from the containment atmosphere. The results of above release classes, for which fission product scrubbing was very effective, were obtained through MAAP analyses. Results typical of those cited above are supported by findings of the Source Term Expert Group in their report DOE/ID-10298, "Licensing Design Basis Source Term Update for the Evolutionary ALWR", September, 1990. In this study the effective spray induced aerosol removal constant was identified to be greater than 100 per hour for the first ten minutes of the release and 50 per hour for the next half hour. These removal rates will yield even faster and more effective fission product scrubbing by the sprays than that predicted by MAAP.

QUESTION 722.83

It is stated that "if containment heat removal is not reestablished within approximately 42 hours, the containment maximum pressure capacity will be reached at this point." However, the containment failure time for late containment failure sequences RC 5.1 and RC 5.2 are given as over 90 hours in Table 11.3-1. Please explain this discrepancy.

RESPONSE 722.83

In CESSAR-DC Appendix B, the statement is made that "if containment heat removal is not reestablished within approximately 42 hours, the containment maximum pressure capacity will be reached at this point." This statement applies to a representative event within release class RC 3.1 category of accidents where the containment overpressurizes and fails prior to sustained core damage. This occurs due to a loss of containment heat removal (resulting from a loss of component cooling water) in conjunction with an initiating event that involves a loss of RCS coolant with RCS makeup available. For such a scenario the secondary side heat sink is effectively unavailable, and if adequate containment heat removal is not reestablished within about one and a half to two days, the containment will overpressurize and potentially fail prior to occurrence of any core damage.

With regard to release classes 5.1 and 5.2, the containment failure times pertain to different accident scenarios such as a loss of offsite and onsite power for which vessel failure occurs prior to containment failure. For RC 5.1 and 5.2, credit is given for the presence of 8 hours of station batteries supplying power to the steam generator auxiliary feedwater pumps. The auxiliary feedwater removes a significant portion of the RCS energy via the secondary system in contrast to direct transfer of the RCS energy to the containment atmosphere in the case of release class 3.1. Consequently, for release class 5.1 and 5.2 significant heatup and pressurization of the containment atmosphere is delayed until after vessel failure which is prolonged due to the availability and use of the secondary side heat sink.

QUESTION 722.84

Please explain why the containment failure time for RC 3.1 (36 hours) is significantly shorter than those of late containment failure sequences, RC 5.1 (128 hours), or 5.2 (90 hours). The rate of heat addition to the containment is similar in both cases. Please also explain why it takes 15 hours after loss of core cooling to fail the vessel in RC 3.1.

RESPONSE 722.84

The accident sequence for RC 3.1 leads to a containment failure prior to any RCS damage or reactor vessel failure. The failure of the containment occurs due to a lack of containment heat removal while RCS heat removal is facilitated via feed and bleed with safety injection available. A total loss of feedwater scenario is postulated for RC 3.1 leading to a manual actuation of feed and bleed for removing the energy generated within the RCS. No containment heat removal occurs since the containment sprays are assumed to be unavailable. As a result the containment pressure rises to its ultimate failure point at 36 hours with the RCS still intact.

RCs 5.1 and 5.2 represent release classes for which the dominant sequences are conventional loss of offsite power transients where the reactor vessel failure occurs prior to the containment failure. The containment failure times are significantly longer for these release classes because RCS heat removal is successfully accomplished via the use of the steam generators and the turbine-driven auxiliary feedwater pump flow for the first eight hours of the transient. This substantially reduces the steaming challenge to the containment.

The vessel failure for RC 3.1 occurs at approximately 15 hours after containment failure. Safety injection is assumed to fail just after containment failure resulting in the loss of core cooling. The long period of time to fail the reactor vessel following the loss of core cooling is attributed to the small amount of decay heat present in the RCS after 36 hours of adequate RCS heat removal.

QUESTION 722.85

Explain why the release fractions for RC 2.4 are significantly less than for RC 3.1, especially for iodine and cesium. This is surprising since in both sequences the containment failed before vessel failure, but in RC 2.4 fission products are not scrubbed by the spray.

RESPONSE 722.85

A review of representative transient results for RC 3.1 and RC 2.4 indicate that the major difference in the two event sequences is associated with the length of time water is available in the cavity to scrub the fission products. Both transients actuate the cavity flood system prior to reactor vessel (RV) failure. However, for sequence RC 3.1 the RV failure occurs at a system pressure sufficiently high to remove much of the water from the reactor cavity (For this transient the reactor cavity is dry within 300 seconds of RV failure.). For RC 2.4 the RV failure occurs at a sufficiently low pressure so that most of the water delivered to the reactor cavity remains. In this case water is available for fission product scrubbing for about 80,000 seconds. While both RC's deposit most of the corium in the cavity region, fission product scubbing is more effective for RC 2.4 than for RC 3.1. Hence the observed lower releases of iodine and cesium for RC 2.4.

QUESTION 722.86

Discuss why the release fraction of Groups 1 and 2 (noble gases) for RC 2.2 are significantly smaller than for other classes.

RESPONSE 722.86

RC 2.2 events are representative of a small LOCA in an unisolated containment. For this event, containment sprays and the spray cooling heat exchangers are assumed to be available for containment cooling (as well as fission product scrubbing). This combination of conditions has been predicted by MAAP to result in a subatmospheric condition developing within the containment approximately five hours after event initiation. In essence the small LOCA produces a steam pressurized evacuation of air from the containment, thus lowering the partial pressure of the noncondensible containment gases. This process is accompanied by steam condensation and spray cooling of the containment atmosphere which in effect reduces the steam partial pressure and cools the steam/air containment mixture to below atmospheric pressure. Once subatmospheric, any fission products remaining within the containment will be trapped via the negative pressure gradient. This results in a smaller release fraction of Groups 1 and 2 (noble gases) for RC 2.2 in comparison to the other release classes.

Question 722.87

Large fractions of the containment melt-through sequences (RCs 6.2 and 6.4), and late containment failure sequences (RCs 5.1 and 5.2) were shown to result in a dose less than 1 rem in Figures 10-2 through 10-5. However, a dose of more than 2 rems was shown even for most of the "no containment failure" case RC 7.1 (Figure 10-1). Please discuss the reasons for this. Can this be because releases were tracked for 24 hours after vessel failure, rather than after the initial release?

Response 722.87

As discussed in the response to Question 722.78, the releases for all release classes are tracked for 24 hours following initial release. In most cases, this means from containment failure. As shown in table 9.3-1, the releases for Release Classes RC6.2, RC6.4, RC5.1, and RC5.2 all occurred relatively late, allowing time for radioactive decay and deposition of fission products inside containment.

Question 722.88

Please provide comparisons of CE System 80+ risk results to NRC health safety goals, including individual risk of early fatality, individual risk of cancer fatality, and probability of one or more early fatality.

Response 722.88

Consistent with the guidance provided in Appendix A to Chapter 1 of Volume 2 of the EPRI ALWR Utility Requirements document, only the dose at 0.5 miles was calculated for the System 80+ PRA. C-E is currently updating the System 80+ PRA and will provide a comparison of the System 80+ risk results to NRC health and safety goals, including individual risk of early fatality, individual risk of cancer fatality, and probability of one or more early fatalities.

Question 722.89

Please identify and discuss those areas in which the System 80+ design and PRA deviates from the EPRI ALWR requirements related to a severe accident (Chapter 5) and PRA (Chapter 1, Appendix A), if any.

Response 722.89

In general, the System 80+ PRA conforms to the guidance provided in the 1990 version of Appendix A to Chapter 1 of Volume 2 of the EPRI ALWR Utility Requirements document. The primary areas in which the System 80+ PRA differs from these guidelines are:

- the System 80+ PRA includes a numerical/data uncertainty analysis as part of the level 1 analyses,
- the System 80+ PRA used lower sequence truncation values than recommended,
- the System 80+ PRA used slightly different initiating event frequencies for some initiators based on a C-E evaluation of pertinant operating experience data.

C-E is currently updating the System 80+ PRA. This update will discuss conformance to the version of Appendix A to Chapter 1 of Volume 2 of the EPRI ALWR Utility Requirements document in effect as of January, 1992.

Question 722.90

Please provide an assessment of the sensitivity of risk results (core damage frequency and conditional containment failure probability) to key parameters and assumptions which are recognized to contain large uncertainties. For the Level 2 portion of the analysis, include as a minimum, treatment of the sensitivity to the following:

- SDSFAIL
- CAVTGEOM/NODISPERS
- HSINTACT
- IGNITFL
- ESPARK
- NCHREOV
- NOCOOLG

Also provide an assessment of the sensitivity of risk results to different strategies for actuating the cavity flood system, e.g., before versus after vessel failure; independent of containment heat removal status versus conditional on heat removal being available.

Response 722.90

Consistent with the guidance provided in Appendix A to Chapter 1 of Volume 2 of the EPRI ALWR Utility Requirements document, the System 80+ PRA does not include uncertainty or sensitivity analyses. C-E is currently updating the System 80+ PRA. It is C-E's intent to incorporate uncertainty and sensitivity analyses in the System 80+ PRA as part of this update. The phenomenological uncertaincies in the severe accident analysis will have a significant role in these uncertainty and sensitivity analyses. Based on discussions with the NRC staff during the System 80+ review meeting in January, 1992, C-E plans to meet with the NRC staff to establish an acceptable scope and approach for performing these analyses.

Question 722.91

Uncertainty in the analysis of phenomena involved in core melt, containment responses and source terms during severe accidents are considerable. The uncertainty in estimation of the availability and reliability of various systems to prevent or mitigate severe accidents also adds to the uncertainty of evaluation of the final risks. Please discuss your plans to account for this uncertainty in your final risk analysis.

Response 722.91

C-E is currently updating the System 80+ PRA. It is C-E's intent to incorporate uncertainty and sensitivity analyses in the System 80+ PRA as part of this update. Please see the response to Question 722.90.

Question 722.92

Accident management, as defined in SECY-89-012, involves actions taken by plant staff to: (1) prevent core damage, (2) terminate progress of core damage and retain the core within the vessel, (3) maintain containment integrity, and (4) minimize offsite release. The present focus of the CE PRA is on the first of these objectives (with the exception of a few operator actions to actuate designed systems such as the igniter system and cavity flood system). However, the PRA can also be used as a tool to identify and assess potential risk reduction measures aimed at the latter three objectives of accident management. If identified at the design stage, specific provisions can be made in the plant design to facilitate (or eliminate the need for) such measure (e.g., automation of otherwise manual actions, or use of remote manual rather than local manual valves). Please describe your plans to use System 80+ PRA to identify and assess additional accident management measures, and to expand the scope of the study for this purpose.

Response 722.92

C-E is currently updating the System 80+ PRA. A severe accident management plan will be developed in parallel with this update. It is C-E's intention that the PRA will provide input to the severe accident management plan in terms of the potential severe accident progressions and the assumed operator responses. The PRA results will also be used to identify potential areas for risk reduction. The items in Generic letter 88-20 and in SECY-89-012 will be specifically addressed in the severe accident management plan. This plan, when completed, will form the basis for developing severe accident management guidelines for System 80+. These guidelines, the severe accident management plan and the System 80+ PRA will be provided to the owner-operator. The owner-operator will use this information to develop the severe accident management procedures.

Question 722.93

Please discuss the applicability and significance of each of the accident management strategies identified in Generic Letter 88-20, Supplement 2 to the CE System 80+ design. Specifically identify any design features which eliminate the need for a strategy, or facilitate implementation of a strategy. Identify and discuss any other unique measures or strategies for dealing with potential severe accidents in the System 80+ design.

Response 722.93

See the response to Question 722.92.

Question 722.94

Provide a description of the design of the equipment hatch and its ability to be rapidly reclosed during shutdown, if necessary. Include a discussion of the need for AC power or any other support systems in order to effect closure, and the pressure seal arrangement, i.e., whether the hatch is pressure-seating as opposed to pressure-opening (which would require full bolting to accomplish sealing under pressure). Discuss any strategies/procedures for rapidly closing major penetrations during shutdown.

Response 722.94

The equipment hatch will be pressure-seating as opposed to pressure-opening. In view of the expected nature of System 80+™ outage activities, the need to open and reclose the equipment hatch will be infrequent. There are no foreseeable reasons to maintain the equipment hatch in a continual open position, such as for routing of welding cables or eddy current testing cables, since cabling for such support functions will utilize dedicated penetration(s) instead of being routed through the equipment hatch. During periods of Reactor Coolant system reduced inventory or midloop operation, the equipment hatch will normally remain bolted closed. The mechanical means for moving of the equipment hatch will be powered from the Permanent Nonsafety buses, which are capable of being fed from offsite power, the Alternate AC Source, or the Diesel Generators.

Question 722.95

Please discuss CE's planned approach for assuring that each of the five elements of accident management defined in SECY-89-012 will be appropriately addressed by the vendor/licensee. Identify the respective responsibilities of CE and of the licensee for addressing each of the elements, and any methods and/or guidance that are expected to be used in this process (e.g., the "Process for Evaluating Accident Management Capabilities" developed by NUMARC, the "Severe Accident Management Guidance Technical Basis Report" developed by EPRI, or the accident management guidelines now under development by each of the reactor vendors as part of the industry Accident Management Program).

Response 722.95

See the response to Question 722.92.

Question 730.11.a

TMI Action Item II.E.4.2, Position 5, states that the containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions. Clarification 6 of the same action item states that the pressure setpoint for initiating containment isolation should be far enough above the maximum expected pressure inside containment during normal operation so that inadvertent containment isolation does not occur during normal operation due to instrument drift or fluctuation due to the accuracy of the pressure sensor. A margin of 1 psi above the maximum expected containment pressure should be adequate to account for instrument error. What is the maximum expected containment pressure under normal operating conditions? At what pressure is containment isolation initiated?

Response 730.11.a

Based on safety analysis requirements, containment isolation must occur prior to +4.0 psig. The typical expected instrument uncertainty error is ±1.5 psig, which reduces the actuation setpoint to a +2.7 psig nominal value. This allows a worst case early actuation to occur at +1.4 psig (2.7 psig-1.3 psig).

A 1.4 psig actuation value provides an additional +1.1 psig margin from the maximum normal containment pressure value of +0.3 psig. This additional margin was established to conservatively bound expected fluctuations in containment pressure due to such factors as instrument air leakage, containment air temperature changes, and changes in differential pressure between inside and outside containment.

Since this high containment pressure setpoint actuates both the CIAS and SIAS, it was conservatively established to minimize spurious challenges to the safety injection system. The methodology employed in establishing this setpoint is consistent with other C-E operating plants.

Question 730.11 (II.E.4.2)

a. TMI Action Item II.E.4.2, Position 5, states that the containment setpoint pressure that initiates containment isolation for non-essential penetrations must be reduced to the minimum compatible with normal operating conditions. Clarification 6 of the same action item states that the pressure setpoint for initiating containment isolation should be far enough above the maximum expected pressure inside containment during normal operation so that inadvertent containment isolation does not occur during normal operation due to instrument drift or fluctuation due to the accuracy of the pressure sensor. A margin of 1 psi above the maximum expected containment pressure should be adequate to account for instrument error. What is the maximum expected containment pressure under normal operating conditions? At what pressure is containment isolation initiated?

b. In accordance with TMI Action Item II.E.4.2, Position 3, all systems labelled as non-essential must be automatically isolated by the containment isolation signal. Provide a statement on the compliance with this requirement.

Response 730.11

b. All containment penetrations not used for accident mitigation or safe shutdown will be automatically isolated by a Containment Isolation Actuation Signal (CIAS) unless:

- The valves are normally locked closed
- The penetrations are normally sealed (i.e., fuel transfer tube)
- The lines are needed for RCP operation (RCP seal injection and component cooling water to RCP seal coolers, motors, and lube oil coolers)

The exception for RCP operation allows the RCPs to be available for accident mitigation or safe shutdown if offsite power and non-essential support systems are available. These lines are continuously monitored for radiation and can be manually isolated from the control room. This clarification will be added to CESSAR-DC, Section 6.2.4. This revision is attached.

6.2.4.1.2 Design Features

The following is a summary of Containment Isolation System design features. Incorporation of these features into the Containment Isolation System results in a design where the design criteria for containment isolation barriers given above are met.

A. Containment isolation valves and interconnecting piping are designed and constructed to Safety Class 2 and Seismic Category I standards as defined in ANSI N18.2-1973 and Regulatory Guide 1.29, respectively.

B. The design pressure and temperature of all piping and connected equipment comprising the isolated boundary is greater than the design pressure and temperature of the containment.

C. Containment isolation valves and interconnecting piping are protected against missiles.

D. Containment isolation valves and interconnecting piping are protected against the effects of pipe whip and jet impingement.

E. The maximum allowable particle size entrained in water taken from the containment sump is limited. This ensures that the proper operation of ESF systems and CIS valves will not be inhibited by debris introduced into the containment following a LOCA. E

F. Containment isolation valves are designed to operate under normal environmental conditions and to fulfill their safety related function under post-accident environmental conditions, consistent with the requirements of Section 3.11.

G. Containment isolation valve and associated penetration piping are qualified in Section III of the ASME Code, as Class 2 components, as described in Section 3.9.3.

H. Maximum allowable actuation times are imposed on containment isolation valves consistent with their required safety function and ANSI/ANS 56.2-1984. I

I. Valve operators and power sources are selected for containment isolation valves consistent with their required safety function. E

← Insert A

<u>Attachment to CESSAR-DC (Page 6.2-37)</u>

<u>Insert A</u>

J. All containment penetrations not used for accident mitigation or safe shutdown are automatically isolated by a Containment Isolation Actuation Signal (CIAS) unless:

- The valves are normally locked closed
- The penetrations are normally sealed (i.e., fuel transfer tube)
- The lines are needed for RCP operation (RCP seal injection and component cooling water to RCP seal coolers, motors, and lube oil coolers)

The exception for RCP operation allows the RCPs to be available for accident mitigation or safe shutdown if offsite power and non-essential support systems are available. These lines are continuously monitored for radiation and can be manually isolated from the control room.

Question 810.1

Please explain why the CESSAR document (Section 1.8) lists NRC's Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," as "withdrawn" and not applicable to the System 80+ design.

Response 810.1

Regulatory Guide 1.101 (Revision 1) was identified as "withdrawn" based on a published status report. Table 1.8-1 will be revised in a future amendment to refer to Revision 2 as "not applicable". Revision 2 endorses Revision 1 to NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," which was published in November 1980 to provide specific acceptance criteria for complying with the standards set forth in § 50.47 of 10 CFR Part 50. These criteria provide a basis for NRC licensees and State and local governments to develop acceptable radiological emergency plans and improve emergency preparedness and are, therefore, not applicable to the System 80+ design.

TABLE 1.8-1 (Cont'd)

(Sheet 13 of 19)

REGULATORY GUIDES

Document/Title GDC References	Original or Revision Issue Date	Reference CESSAR Section	
Reg. Guide 1.101 - Emergency Planning for Nuclear Power Plants	Revision 2 ~~Withdrawn~~ 10/81	Not Applicable	B
Reg. Guide 1.102 - Flood Protection for Nuclear Power Plants	Revision 1 9/76	2	E
Reg. Guide 1.103 - Post Tensioned Prestressing Systems for Concrete Reactor Vessels and Containment	Withdrawn		B
Reg. Guide 1.104 - Overhead Crane Handling Systems for Nuclear Power Plants	Withdrawn		
Reg. Guide 1.105 - Instrument Setpoints for Safety-Related Systems	Revision 2 2/86	7.1.2.27	E
Reg. Guide 1.106 - Thermal Overload Protection for Electric Motors on Motor-Operated Valves	Revision 1 3/77	7.1.2.28	
Reg. Guide 1.107 - Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures		Not Applicable (Concrete containment)	B
Reg. Guide 1.108- Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants	Revision 1 8/77	8.1	E
Reg. Guide 1.109 - Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I		Not Applicable	E

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