Docket File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

December 30, 1992

Docket No. 52-002

APPLICANT: ABB-Combustion Engineering, Inc. (ABB-CE)

PROJECT: CE System 80+

SUBJECT: PUBLIC MEETING OF DECEMBER 2 and 3, 1992, TO DISCUSS THE CE SYSTEM 80+ SEVERE ACCIDENT SUBMITTAL

On December 2 and 3, 1992, a public meeting was held at the ABB-CE facilities in Windsor, Connecticut, between representatives of ABB-CE and the U.S. Nuclear Regulatory Commission (NRC). The purpose of the meeting was to discuss ABB-CE's submittal entitled, "System 80+ Severe Accident Phenomenology and Containment Performance," dated August 1992. The meeting focused on questions resulting from the staff's review of the submittal. Enclosure 1 provides a list of attendees. Enclosure 2 is the staff's detailed meeting summary. Enclosure 3 lists the questions discussed that were originally included in the associated meeting notice. Enclosure 4 contains ABB-CE's preliminary responses to the staff's severe accidents questions.

The majority of the staff's questions (Enclosure 3) were either answered during the meeting or will be addressed when the next revision of the document is submitted on January 21, 1993.

Fuel coolant interaction (FCI) and debris coolability were identified as the two areas of greatest concern because an obvious path to resolution was not clearly defined. Because first-of-a-kind-engineering (FOAKE) has not been performed for the System 80+ design, a detailed structural analysis will not take place until that time, which ABB-CE believes is a post design-certification action. Therefore, it was unclear as to whether ABB-CE will be able to determine the reactor cavity's structural ability to accommodate ablation resulting from core-concrete interaction (CCI) and loads associated with a FCI. At the conclusion of the meeting, ABB-CE agreed to evaluate what could be done to define the forcing function and determine the ability of the reactor cavity to withstand ablation with static and dynamic loads resulting from these phenomena.

The high ultimate strengths calculated for the System 80+ containment did not consider the potential effects of the containment penetrations. Since the System 80+ does not have finalized penetration designs, ABB-CE has assumed for analytical purposes that System 80+ will have penetrations that have greater strength than the containment shell. ABB-CE stated that containment penetrations performance parameters will be addressed in the procurement specifications that will require penetrations to be qualified to the ultimate capacity of the containment. This is a different approach over existing plants with steel containments. If the penetrations are as strong as the containment shell, catastrophic failure of the metal shell could become a credible event.

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ABB-CE provided the staff with MAAP computer code input files on a floppy-disk during the meeting. By letter dated December 15, 1992 (LD-92-117), ABB-CE submitted documentation declaring this information as proprietary in accordance with the provisions of 10 CFR 2.790. In addition, it appeared that a significant number of MAAP computer code cases to supplement the probabilistic risk assessment have not yet been performed.

The staff commented that the number of hydrogen igniters appeared to be low. Representatives of the NRC's containment systems and severe accidents branch (SCSB) will meet with members of Duke Engineering & Services, Inc. (DESI) to gain a better understanding of igniter placement and obtain the volume and vent capacity for the compartments in containment.

The following is a synopsis of commitments that were made and other concerns that were raised. A more complete and detailed summary is provided in Enclosure 2.

(1) ABB-CE agreed to define and determine the forcing function and the reactor cavity's capability to withstand ablation and static and dynamic loads resulting from severe accident phenomena (e.g. CCI and FCI).

(2) ABB-CE committed to address the staff's concern of the unique susceptibility of the sumps (e.g. reactor cavity sump) to CCI.

(3) ABB-CE committed to go through the various sequences and identify the equipment and instrumentation that are necessary to monitor and mitigate the consequences of a severe accident. ABB-CE's initial response to exactly what the System 80+ plant operators will be looking at to make this determination will not be provided to the staff until the January 21, 1993, submittal.

(4) ABB-CE committed to meet with the staff after the January 21, 1993, submittal to discuss accident management.

(5) ABB-CE committed to provide the staff with a list of station battery loads during station blackout and other severe accident conditions.

(6) ABB-CE committed to provide the staff with the calculation justifying that the containment can withstand 50-percent entrainment from a high-pressure melt ejection (HPME) event.

(7) ABB-CE committed to document their conclusion that the most probable mode of reactor vessel failure is via the instrument tubes.

(8) ABB-CE committed to provide the hand calculation that determines the maximum pressure spike resulting from adiabatic, isochronic complete combustion (AICC) of hydrogen.

(9) ABB-CE committed to evaluate the accident handling capability of the System 80+ containment purge system. Use of the purge system lends the potential to reduce over 50 percent of the total risk if the system has the capability to purge under certain severe accident conditions. In response to the staff's request, ABB-CE agreed to revise the severe accidents mitigation design alternatives (SAMDAs) submittal accordingly. ABB-CE will also provide any previous assessments performed by DESI for use of a filtered vent.

(10) ABB-CE agreed to provide NRC's analysis group with the design-basis accident mass and energy release rates as listed in the Combustion Engineering Standard Safety Analysis Report (CESSAR-DC) on a floppy-disk.

Enclosure 3 contains the information requested by the analysis group to assist them in their performing MELCOR analyses for CE System 80+.

Sincerely,

(Original signed by)
Michael X. Franovich, Project Manager
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Associate Directorate for Advanced Reactors
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Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See next page

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MEETING ATTENDEES

DECEMBER 2 and 3, 1992

NAME	ORGANIZATION
M. Franovich	NRR/PDST
R. Mitchell	NRC
A. El-Bassioni	NRC
J. Kudrick	NRC
M. Snodderly	NRC
J. Monninger	NRC
R. Palla	NRC/NRR/SPSB
R. Schneider	ABB-CE
L. Gerdes	ABB-CE
S. Ritterbusch	ABB-CE
M. Jacob	ABB-CE
T. Oswald	Duke Eng. & Svcs.

Enclosure 1

MEETING WITH ABB-CE
ON THE CE SYSTEM 80+ DESIGN
FOR SEVERE ACCIDENTS

MEETING OBJECTIVE

This meeting represented the first severe accidents working meeting between the staff and ABB-CE personnel. As a result, it is important that the staff obtain a better understanding of how ABB-CE has concluded that the design is able to adequately accommodate a significant and credible portion of the severe accident sequences. In addition, the current System 80+ submittal refers to various analyses which have led ABB-CE to substantiate design adequacy. The meeting was intended to address, in more detail, these analyses. Finally, ABB-CE has applied the results of generic analyses throughout their severe accident evaluations. The staff needed to understand how ABB-CE concluded that the generic analyses were applicable to the System 80+ design.

DISCUSSION

To focus the discussions, the staff generated a series of questions which were forwarded to ABB-CE. These questions were subsequently used as the meeting agenda. As a result, this meeting summary also used the questions as a basis for the meeting findings.

Computer Analyses

The staff transmitted a series of questions concerning the MAAP runs which have been conducted in support of the design. In addition, the staff also wanted to perform some MELCOR computer runs to aid in the staff's understanding of the System 80+ design.

ABB-CE indicated that MAAP code was used to implement a series of computer runs. The results were then used as input for probabilistic risk assessment (PRA) analyses. Items of interest included timing for early core interactions, time to vessel failure, fission product releases, and late containment failure modes (i.e., containment pressure, temperature, and non-condensible production).

ABB-CE declared the System 80+ MAAP computer code input files as proprietary information. By letter dated December 15, 1992, the applicant has provided appropriate documentation attesting to the proprietary status of this information in accordance with the provisions of 10 CFR 2.790.

In addition, ABB-CE agreed to send on a floppy disk the design-basis accident (DBA) mass and energy release rates as listed in the CESSAR.

As part of these discussions, both parties decided that a meeting was needed to continue the MAAP discussions relative to how it was used in the PRA analyses. ABB-CE was interested in defining the specific MAAP runs which were needed in this regard. This level of detail, however, proved to be impossible to produce during the current meeting. It was agreed that a two-day meeting should be held January 4 and 5, 1993, in Windsor, Connecticut.

Enclosure 2

Severe Accident Chapter Questions

Questions 1, 2, and 3

The discussion focused on the methods used to develop the containment fragility curve. The approach was to generate the curves assuming a plain sphere without any penetrations. Although, the methodology has not been discussed with members of our structural staff, ABB-CE indicated that the method is described in NUREG/CR-4870. The NUREG addresses penetration strains and the immediate area of the shell around the penetration. To account for each penetration, procurement criteria would be established in such a manner to ensure that the penetration would not be less robust than the shell. The staff indicated that if ABB-CE were to adopt this approach, such performance parameters must be specified in the associated ITAAC.

The staff expressed a concern that unless the staff would agree completely with this approach, both the Service Level C and ultimate pressure values of the containment could change. As a result, the staff said that we would be in contact with our structural staff and set up a telecon with the appropriate members of ABB-CE staff.

The use of expert solicitation in the development of the fragility curve was also discussed. Expert solicitation from NUREG-1150 indicated that a 50-percent failure probability would be reached at strains between 2 and 3 percent. ABB-CE reduced the strain to 0.5 percent to account for the presence of the penetrations as well as additional safety margin. This information would also be forwarded to our structural staff for their evaluation. In order to determine the sensitivity of Level 2 PRA results to the containment fragility characteristics, ABB-CE agreed to consider performing a sensitivity analysis assuming different containment fragility.

The discussion also included how seismic loads should be considered in the calculation of the conditional containment failure probability (CCFP) value. The staff indicated that when one considers whether the design meets the 0.1 value specified in SECY-90-016, only internal events need be considered. As a result, seismic loads need not be considered, but seismic loads should still be discussed in the overall evaluation of the design to cope with various events.

Question 4

Penetrations will be designed consistent with the fragility curve assumptions. This means that the penetrations would be stronger than the shell. It was noted that this is not consistent with existing containment designs. In fact, past analyses have taken credit for the fact that penetrations would fail prior to shell failure. As a result, catastrophic failure of the metal shell need not be considered as a credible event. ABB-CE noted the unique consideration for penetration designs and indicated they would get back to the staff on this matter.

The question of what shell temperatures were considered in the development of the curves was raised. It was indicated that for early failures, a temperature

of 280 °F was used while 460 °F was used for later failures. MAAP results were used to compare the validity of these values. It was found that MAAP predictions of late containment temperatures ranged between 380 °F and 430 °F. This information substantiated the selection of 460 °F temperature.

The staff requested that ABB-CE reconfirm these temperature values, based on the revised MAAP 3.B analyses that will be performed in support of the Level 2 PRA update. ABB-CE agreed to provide the reconfirmation results once the analyses are available.

Question 5

The discussion concerning the question of what is needed for demonstrating equipment survivability was much needed. The initial position of ABB-CE was that all equipment would be used early in the transient. As a result, no additional consideration was necessary since the environment would be no worse than DBA conditions.

The staff discussed the sources from which survivability criteria are identified. They are DBA, 50.44, 50.34(f), and SECY 90-016. It was acknowledged that neither 50.49 nor Appendix B of Part 50 were applicable in events which were beyond DBA conditions. DBA requirements contain substantial margins to cover the unknown. However, beyond design conditions should consider best estimate calculations without this added safety margin.

It was suggested that ABB-CE look into the approach taken by both ice condenser and Mark III designs to show that the needed equipment would perform their specified function. ABB-CE indicated that they would go through the various sequences and identify the appropriate equipment and instrumentation and show why the equipment will function. Although they were unable to commit to a schedule, ABB-CE did indicate they would look to Duke Engineering & Services for the selected approach. ABB-CE will respond in the December 15, 1992, submittal with both an approach and commitments on when their future submittal will be provided to the staff.

Accident management and how the System 80+ equipment could be used were also discussed. ABB-CE felt that identification of equipment and a discussion of how it would be used should be separated from accident management. The staff indicated that this issue and its resolution could be delayed until the more substantial PRA analyses are completed. As a result, agreement was reached that a separate meeting be scheduled after January 21, 1993, to further discuss this issue and other related items.

Question 6

ABB-CE indicated that they intend to take credit for fission product removal by the annulus ventilation system and secondary containment building for both DBA, as well as severe accidents. It was pointed out that in spite of a severe event occurring within containment, the secondary containment could well be unaffected. The approach to establishing the leakage value above design was also discussed. The staff agreed that penetration failure need not be considered, especially in light of the design criteria discussed in response to

questions 1-4. However, increases due to the pressure should be considered. The staff suggested that a ratio of the actual pressure difference to the design pressure difference be used to increase leakage. ABB-CE had intended to use the square root of the squares of the pressures. It is the intent of ABB-CE to select the more conservative approach, if at all possible.

Question 7

The intent of the hold-up volume is to reduce the possibility of debris entering the in-containment refueling water storage tank (IRWST) during the post-loss-of-coolant-accident (LOCA) recirculation phase of operation. In addition, it was clarified that the containment sprays have the same initiation signals as more recently licensed CE plants. There is a scram permissive with a 2 out of 4 hi-hi containment pressure.

Question 8

There were several clarifications of the intent of the words provided in the ABB-CE submittal.

Question 9, 10, and 11

To determine the water levels throughout the containment during the course of the event, ABB-CE developed a separate hydraulic computer program. The program is capable of computing the transient depth of water in all possible water pools within the containment during an event. A simplified version of this program was incorporated into the MAAP program via a contract with Advanced Reactor Severe Accident Program (ARSAP).

ABB-CE indicated that the IRWST does not completely surround the containment, but only 300 degrees due to the presence of the refueling canal. The hold-up volume is rectangular in shape measuring about 37 feet by 11 feet. The maximum water capacity of the hold-up volume is about 60,000 gallons.

ABB-CE indicated that earlier PRA results indicate that a dry or partially flooded cavity is expected to occur less than 1.0 percent of the time. In addition, the design was established so that at no time will the water reach the bottom of the reactor vessel.

The interconnecting valves which are required to open during the event to provide a flow path between the hold-up and other volumes are normally in a dry environment. However, it was noted that post-LOCA conditions will have the valves submerged. ABB-CE will check to ensure this condition is properly recognized in the valve specifications.

The staff noted that these flood valves should be included as part of the pre-operational test program, as-well-as ITAAC for valves. These valves should undergo actual dynamic testing to verify their capability.

Question 12

NUREG-1465 is being used as the reference. ABB-CE will try to be consistent throughout the discussions since it was noted that there may be some incorrect references in the current writeup.

Question 13

ABB-CE believes that the control room operators will know long before actual core damage that the event is going beyond DBA conditions. The initial response to exactly what plant information/instrumentation the operators will use to make this determination will not be provided to the staff until the January 21, 1993, submittal. But, ABB-CE said that this information would not be explicitly included in the emergency procedure guidelines (EPGs). The staff indicated that the NRC will evaluate MAAP runs to ensure that the response with respect to timing as well as the actions are fully supported. If not, the staff expects the differences to be addressed in the response.

Question 14

ABB-CE acknowledged that it is possible that the operator could unnecessarily flood the reactor cavity. However, there does not appear to be any serious downsides due to this operation. It would mean more cleanup. When questioned, ABB-CE stated that adequate net positive suction head (NPSH) would exist for safety injection pump operation with the reactor cavity flooded. In addition, there is a dry-cavity analysis which shows acceptable results of core-concrete interaction (CCI), and therefore, there is no minimum water level is not an issue.

An outside water source for the containment sprays has also been provided. An external tee on piping outside the containment has been provided to allow the use of fire water. The tee would be normally blind flanged; however, a spool piece would be provided to allow alignment with the fire water.

Current System 80+ documentation indicates the hydrogen ignition system (HIS) will not be initiated until an indication of core damage. The staff indicated that under this criterion, a significant quantity of hydrogen may be generated prior to system activation. Based on the ensuing discussion, ABB-CE agreed that this may be too late. ABB-CE committed to review the HIS actuation criteria used in operating ice condenser and boiling water reactor Mark III plants, and modify their criterion accordingly.

Question 15 and 16

Either the question was already addressed, or the clarification was straight forward.

Question 17

The threshold pressure value is still under discussion at ABB-CE. ABB-CE believes 250 psi can be the final value. In fact, it was stated that most, if not all the calculations to date, actually use this value. However, ABB-CE is

not ready to commit to such a conservative value unless they are positive that the revised analyses show that this value yields acceptable results. ABB-CE expects to respond to this question more fully in the January 1993, submittal.

Question 18

Depressurization of the reactor coolant system (RCS) using the SDS pathway will take about one hour while MAAP shows that it will take about 4 hours before you get into trouble. ABB-CE also indicated that the shortest possible time for this operation is approximately 40 minutes. In light of the relatively short time needed for this operation, ABB-CE believes there is sufficient margin to take this action. The event that ABB-CE uses for making this conclusion is a loss of feedwater. A depressurized RCS is necessary to initiate feed and bleed operation.

The staff committed to identify those MAAP results that they would like ABB-CE to submit in support of the PRA and severe accident closure analysis. This response will probably be provided during the scheduled January 4 and 5, 1993, meeting.

Question 19

ABB-CE indicated that the station batteries are designed to provide power for only 2 hours. Four-hour capability per battery bank is achieved through use of a load management program. This was in accordance with the CESSAR-DC Chapter 8 description of the dc system. The severe accident documentation also indicated that the design has a cumulative 8-hour battery capability. Discussions did not provide any clarification in this area as to what equipment is on the batteries. The staff will speak with electrical engineering reviewers to ascertain the battery adequacy for station blackout (SBO) and accident capability. ABB-CE will provide loads for severe accidents including SBO.

Question 20

Results of MAAP runs were used to determine the timing of the various key events in the accident progression, such as time to core damage and time to reactor vessel failure. To allow everyone in the meeting to understand the models contained in MAAP, a brief discussion of the basic nodes in MAAP was provided.

The RCS includes a crude reactor model, a detailed pressurizer, two steam generators, a PORV, ECCS systems, and all critical pumps. The principle input parameters involve mass and energy.

All the sumps as well as the containment are also modeled. The containment is divided into four nodes: an upper containment volume above the crane wall; a volume above the reactor vessel and up to the crane wall; the reactor cavity; and the volume outside the cavity region and up to the crane wall. Input consists of typical containment parameters such as volumes, heat capacitance, etc.

In addition, heat removal systems such as sprays, IRWST, and RHR are also modelled as well.

For the January 4 and 5, 1993, meeting, ABB-CE will discuss the MAAP base case and sensitivity analyses. This will include discussion of the Electric Power Research Institute's (EPRI's) recommendations for MAAP 3.B sensitivity analyses and how ABB-CE complied with these recommendations.

Question 21

Mainly straight forward clarifications.

Question 22

ABB-CE's initial response was that the pool seal would eliminate the upward pathway thereby assuring the 90-percent assumption. However, further discussion called into question whether or not the seal could withstand the anticipated pressures without rupturing. In addition, a second possible flow path around the reactor vessel hot and cold leg nozzles was acknowledged by ABB-CE. ABB-CE indicated that calculations have shown that the design could accommodate up to 50 percent rather than the assumed 10-percent upward flow. Since the impact of these two additional debris flow paths could not be confirmed during the meeting, ABB-CE agreed to get back to us on their position on this issue.

Question 23

ABB-CE would get back to us the following morning.

Question 24

ABB-CE did not have a referencable document concerning the BETA last test evaluation. The staff indicated will check with the NRC staff, but the staff did not believe such a reference existed. In any case, ABB-CE indicated that the writeup would be revised to use only acceptable references.

Question 25

After a brief discussion, the staff agreed with the comment.

Question 26

Straight forward clarification.

Question 27

Detailed review will be delayed until the meeting scheduled for January 4 and 5, 1993.

Question 28

The staff indicated that it was essential that an analysis be performed which would show the structural capability of the support structure. ABB-CE agreed

with this assessment and planned to discuss what could be provided to the staff. They indicated that internal meetings were necessary before they could have any meaningful discussions with the staff. As a result, ABB-CE could only indicate that they would get back to the staff.

Question 29, 30, and 31

ABB-CE plans to get back to the staff on these items.

Question 32

ABB-CE believes that they have performed the necessary analysis to adequately address this issue. ABB-CE indicated that all that was necessary was to locate the analyses and transmit the information to the staff. ABB-CE will get back to the staff when this information could be provided, but they did not consider this to be an issue.

Question 33

ABB-CE indicated that the containment sprays can also be powered by the combustion turbine. In addition, if sprays are not available, the containment will fail eventually because operation of the sprays is the only way to remove energy from the containment via a heat exchanger within the flow loop.

The containment sprays are modelled in MAAP as either on or off.

Question 34

The selection of the one hour was based on expert judgment. ABB-CE believes that whether the value is one or four hours, the categorization of events would not change. However, the amount of combustible gases present would be greater if a 4-hour period were used. ABB-CE indicated they will reassess the use of a 1-hour versus a 4-hour time period, and identify an approach for dealing with additional combustible gases if the one hour is retained.

Question 35

ABB-CE indicated that they intend to comply with the EPRI Utility Requirements Document (URD). However, it must be realized that the EPRI requirements are in a sense a living document and are constantly changing. By December 24, 1992, ABB-CE intends to submit an update to the staff on the design comparison with the EPRI requirements (URD). Currently, they believe the design is in general compliance with the EPRI criteria with approximately 25 departures from the EPRI URD.

Question 36

ABB-CE believes that the steam generator tubes are preserved when the reactor pressure falls below 1000 psia. In addition, ABB-CE has concluded that below 200 psi, direct containment heating (DCH) is eliminated. It was noted that the current thinking of our research staff is slightly more conservative. Below

150 psi, there is complete agreement. Between 150 and 250 psi, it is felt that it could be acceptable, but it is questionable. At 300 psi and above, the elimination of DCH would be unacceptable.

In addition, the applicant had referenced an industry report from the Nuclear Management and Resources Council on severe accidents. The industry report suggests that creep in steam generator tubes is not credible with 9 feet of water above the core. ABB-CE will attempt to provide the report or provide hand calculations to support this assumption.

Question 37

Discussion centered about a better understanding of the assumptions which went into the development of the figure in question. ABB-CE indicated that the underlying assumption was that everything reacts, ejects, and burns. They believe that this is very conservative. They also considered no entrapment, but no steel reaction was included in the calculations.

With respect to the general divisions in the figure, the following added information was provided. For the dry cavity with no RCS water means that only steam is considered. The values used in the figure were EPRI numbers. In addition, a dry cavity means that no hydrogen burns are considered with RCS water meaning only left over water. The region of full cavity with RCS water means that about 227,000 kg of water was assumed available.

ABB-CE considers that DCH should only be considered for a dry cavity. This opinion is believed by ABB-CE to be consistent with NUREG-1150. However, it was again noted that if it could be shown that the design could tolerate 50-percent involvement, this capability should resolve the issue. However, the staff would still question the assumption of the 10-percent assumption, as presented by ABB-CE.

Question 38

Current documentation references the EPRI Technical Basis Report (TBR) developed in support of the industry accident management program report. This document is not available to the staff. However, ABB-CE indicated that a simple calculation can be performed to prove the same thing. ABB-CE agreed to attempt to make the reference available or provide an acceptable basis to resolve this issue (e.g., hand calculation).

In addition, ABB-CE will identify the differences between System 80 and System 80+. For each feature that is provided in the System 80+ design, there will be a discussion of the risk improvement of the feature. ABB-CE believes that, for the most part, they have provided the staff with the necessary information. It is documented in the staff's draft safety evaluation report in Section 19.1 starting on page 19-13 and continuing for about 12 pages. However, the staff indicated that this information is relevant only to severe accident prevention features and their impact on core damage frequency. The staff recommended that ABB-CE consider developing comparable information for

the severe accident mitigation design alternatives (SAMDAs) and existing features for System 80+. This would include an assessment of the reduction in risk or CCFP for each feature. For purposes of Chapter 19.2, the staff indicated that the section appeared to be sufficient to describe why the design is an improvement over existing designs. It was further indicated that the detailed information would be nice. However, the staff indicated that the staff would get back to ABB-CE if the added information would be needed for design certification. The staff will discuss this matter further with ABB-CE as part of the January 4 and 5, 1993, PRA meeting.

Question 39

ABB-CE intends to review several key reports as well as performing some structural calculations. The applicant recognized the importance in establishing the structural capability of the structures. However, it was unclear during the meeting as to what was possible in this short period of time. In conjunction with the structural calculations, ABB-CE also agreed to perform related forcing function calculations.

The staff recognized the large amount of work necessary to fully resolve this issue. As a result, they agreed to provide guidance to the ABB-CE staff. In this spirit, the staff agreed to accept final documentation of this effort no later than March 21, 1993. If received by this date, the staff will evaluate the information in the System 80+ final safety evaluation report. However, it was noted that meetings of the type of this current meeting would be necessary to keep the staff informed of the progress in the severe accident analysis area.

Question 40

ABB-CE indicated that their calculations assumed a 20 kg mass participating with 100-percent efficiency for fuel-coolant interaction (FCI). In response to the staff's concerns, ABB-CE agreed to consider significant core involvement (about 20-percent core) with lower conversion efficiency (greater than 2 percent).

It was noted that this effort needed frequent communication between the structural staffs of both organizations. To this end, the staff agreed to initiate these discussions so that an acceptable approach could be identified. The intent will be to establish criteria or considerations to be placed on the structures, so that when the actual calculations are performed, the necessary FCI information will be properly considered.

Question 41

The structural capability of the reactor cavity to withstand steam explosions reported in the System 80+ documentation was based or developed through design comparisons of existing plant structures and not System 80+ design-specific structural calculations. More importantly, ABB-CE did not expect to perform calculations of this level of detail prior to design certification. The staff indicated that this will require involvement of the NRC structural staff.

However, the staff indicated that ABB-CE must specify and support performance goals for static and dynamic load capability of the reactor cavity structure. ABB-CE agreed to do this and attempt to define the forcing function.

ABB-CE committed to provide an evaluation; however, ABB-CE stated that a detailed calculation for the structure would not be achievable at this time. This position was based on the premise that the finalized structural design of the reactor cavity would be accomplished through first-of-a-kind-engineering (FOAKE).

The adequacy of both the number and location of the hydrogen igniters was identified as an area that needed to be discussed. This meeting will take place sometime in January of 1993, at the Duke Engineering & Services, Inc. (DESI) facilities in Charlotte, North Carolina. During this meeting, the staff will audit design information from the PACE system which yields 3-D computer visuals of the interior of the containment and nuclear annex.

Question 42

For the case involving dynamic loads from ex-vessel steam explosion (EVSE), the staff suggested that a rectangular load should be considered with the appropriate time of the applied load to be determined by ABB-CE. ABB-CE should perform a sensitivity study for timing in the case of dynamic load analysis for the reactor cavity. It was noted that per NUREG-2462, this profile represents the most conservative profile. As a result, ABB-CE would consider the staff's recommendation, but would not commit to using this profile without further study.

Question 43

The staff expressed concern that recent TMI studies indicated that lower reactor vessel head failure/slump could occur prior to instrument tube failure. ABB-CE committed to document their basis for concluding that the instrument tube failure is the most probable. It was also noted that there are no other bottom penetrations in the reactor vessel.

Question 44

Our previous discussion on FCI more or less addressed this issue. NUREG-4896, page 27 further documents the basis of the design. However, it was noted that ABB-CE agreed to forgo the Zion probabilistic safety study as their basis and will undertake their own study.

Question 45

The reference which was used to justify the selection of the upward heat flux was R. Henry. It was taken from the Westinghouse AP600 submittal. In the context of the System 80+ submittal, it represented a bounded number which is based on a combination of both ACE and SWISS experiments. However, this information was not really used by ABB-CE, since a rate dependent load is not used.

Question 46

The pressure spikes for rapid steam generation event were hand calculated. It was assumed that the spike would occur at the same time that the peak LOCA containment pressure occurred. This is a conservative assumption and not a mechanistically-derived conclusion. The spike was based on 70 percent of core involvement with no added chemical reaction energy. ABB-CE committed to provide the staff with a copy of the hand calculation.

Question 47

The reference for the calculation of the pressure rise associated with a hydrogen burn was stated by ABB-CE to be NUREG/CR-5567, page 38. The staff stated that a further check by the staff was needed to determine the acceptability of this approach. It was indicated that a NUREG/CR report does not necessarily indicate staff approval, since it is a contract-produced document.

Of particular note was the fact that ABB-CE indicated that their investigation has shown that caution should be taken by the operator in turning on the sprays when steam inerting is suspected. ABB-CE indicated that hydrogen instrumentation should be checked prior to taking action to determine if hydrogen concentrations would indicate a basis for a delay in operator action. As an aside, ABB-CE verified that the design has safety-grade and redundant hydrogen instrumentation which is capable of monitoring hydrogen concentrations up to 15-percent hydrogen concentration. The staff believes that this operator information on containment spray actuation during a potentially steam-inerted environment should be appropriately relayed to a combined license applicant through the EPGs or System 80+ accident management guidelines (AMGs).

It was noted that in several instances, ABB-CE has used 100-percent metal-water reaction to mean all zircoloy clad interacts with water. This translates to 130 percent of the fuel clad surrounding active fuel. The staff clarified that 10 CFR 50.34(f) requires 100-percent fuel clad metal-water reaction of only active fuel. ABB-CE used hydrogen control information from the Department of Energy (DOE) report, "Technical Support for Hydrogen Control Requirements for EPRI Advanced PWR." The staff suggested that a common reference be used to avoid any confusion. Since the majority of references refer to active fuel, it was suggested to use this as the reference rather than the total amount of zirconium within the core. ABB-CE agreed with this suggestion and will modify all references to this common base.

Question 48

ABB-CE supported their approach by comparing their calculated values against experimental data. They concluded that their calculated values bounded all experiments and was therefore conservative. Figure 4.1-2 was the focus of the discussion. It is used by ABB-CE as confirmation of the actual calculations performed by them. It was noted that the presence of steam seems to indicate that the peak pressures would be about 15 psi higher than dry air.

Question 49

Either previously answered by other question responses, or the clarification was straight forward.

Question 50

ABB-CE reported that the approach used the guidance of a advanced light water reactor document cited in Question 48. In any case, ABB-CE intends to redo the MAAP calculations. The staff pointed out that local hydrogen concentrations predicted by lumped parameter codes such as MAAP are a strong function of the volume of the nodalization scheme.

While ABB-CE did not agree nor disagree with the limitation, they indicated that the maximum hydrogen concentration was a minor point since System 80+ relies on igniters to limit concentration. The staff reiterated the point that the 2-percent value is an artifact of the node chosen in the MAAP run. In any case, hydrogen igniters will be located immediately outside the IRWST. In addition, ABB-CE will provide a discussion of why high concentrations in the vicinity of the IRWST are not possible. As part of this, ABB-CE agreed to review the mass and energy releases from the IRWST for selected sequences to identify potentially rich hydrogen concentrations. These hydrogen conditions include loss-of-feedwater, station blackout, and feed-and-bleed sequences. This will be submitted as part of the January 21, 1993, package.

The staff raised the question of the number and location of the hydrogen igniters in the design. It was noted that an ice condenser has about 100 igniters, while System 80+ has 42. This was puzzling since System 80+ is both larger in volume and core size and appears to be more compartmented. In addition, System 80+ has used the same placement criteria as an ice condenser. Clarification of quantity and location of System 80+ hydrogen igniters will occur during the DESI meeting.

ABB-CE should be prepared to discuss the differences between an ice condenser and the System 80+ relative to the number and location of hydrogen igniters during the DESI meeting.

Question 51

The calculation of the pressure spike due to a hydrogen burn was performed by hand using the adiabatic, isochronic complete combustion (AICC) method. It is based on assuming adiabatic boundaries and uniform mixing within the selected volume. In addition, only global burns have been considered. ABB-CE has considered three different ranges of metal water reaction for their PRA studies: 50 percent for unrecovered events, 75 percent for recovered events, and 100 percent including CCI non-condensibles. The lowest value, which is assigned to unrecovered events, is based on the lack of water in the core.

The initial containment pressure is assumed to be either 20 or 30 psia. These two values correspond to having sprays or without sprays, respectively.

Using the above methods, ABB-CE calculated an overpressure of 110 psi due to a global burn of 130-percent metal-water reaction, as discussed in question 47. Therefore, for a spray case, the peak containment pressure would be 120 psia. This compares with an overpressure of 140 psi reported in the NUREG/CR report. ABB-CE believes the difference to be attributable to the variation of the specific heat at constant volume (Cv) with temperature.

ABB-CE agreed to clarify the methods and approaches used to calculate hydrogen combustion loads for early and late deflagrations since different approaches have been referenced (e.g., AICC curve, Sandia report, other sources).

Question 52

The containment pressure value of 30 psia represents the maximum steam concentration without entering into the steam inerted condition.

Question 53

Combined with other responses.

Question 54

ABB-CE indicated that the steam generator (SG) represents only one of several bypass paths. In the spring of 1992, they submitted a response to the staff's questions regarding the three specific improvements of the SG design. Their response was consistent with the EPRI views as to why there is no valid reason to upgrade the SG, interfacing system LOCA (ISLOCA), etc.

The staff raised the question as to why the containment bypass issue was not addressed in the SAMDAs submittal. ABB-CE indicated that a response to this question was beyond the agenda of this meeting and should be deferred to a later time. However, the staff noted that this issue will need to be appropriately addressed for SAMDAs.

Other containment bypass pathways were discussed with respect to ISLOCA and pipe pressure capability. High-pressure/low-pressure pathways, such as component cooling water (CCW) interface with the RCS through the high-pressure seal cooler for RCP seal, and the shutdown cooling system were mentioned. ABB-CE indicated that an ISLOCA submittal for SECY-90-016 purpose has recently been submitted to the staff for review.

Question 55

The staff asked why ABB-CE did not consider venting of the containment. A look at the most significant risk contributors shows that over 50 percent of the total risk is associated with core melt sequences which are a result of containment failure due to over pressure, prior to vessel breach.

It was acknowledged by ABB-CE that a containment vent would totally eliminate these sequences. In other words, total risk could be reduced in half if a vent were available. Since the purge line dumps into the containment annulus, this may be sufficient to completely address the venting issue. The annulus would

serve as a large volume to attenuate the pressure to low enough values such that the secondary containment would not be damaged due to overpressure conditions.

ABB-CE agreed to evaluate the accident handling capability of the containment purge system. ABB-CE's evaluation will include assessment of isolation valves operability and purge line piping capability under accident conditions. The staff iterated that the System 80+ SAMDAs submittal must be revised accordingly on the issue of a filtered vent. In addition, ABB-CE committed DESI to provide any previous assessments for use of a filtered vent as a System 80+ severe accident design feature.

ABB-CE will get back to the staff on this issue.

Question 56

By letter dated May 8, 1992 (LD 92-042), ABB-CE indicated that two of the three features were addressed. The third feature was addressed in a November 18 or 24, 1992, submittal. Their identifier to the letter is either LD 92-113 or 115.

Question 57, 58, 59, 60

ABB-CE indicated that from a PRA viewpoint, CCI and core debris coolability were being adequately covered by the ranges of values of the individual parameters. However, from a deterministic view, additional studies needed to be performed.

The staff presented their views on the topic. First of all, it must be recognized that the experimental data is scant at best. Therefore, if ABB-CE does not want to be dependent upon future tests to close this issue, a strong effort is needed to determine the capability of the structures (reactor cavity) to withstand the EVSE. Structural analyses, as well as an effort to define the forcing function on the cavity, will be required to resolve this issue.

ABB-CE indicated that their detailed structural analyses are not planned until FOAKE. However, ABB-CE understood and agreed with the approach to closure. ABB-CE needs some time to decide what could be done in the near future to address the structural capability issue. ABB-CE will get back to the staff by January 4, 1993.

Another question raised was what constitutes containment failure relative to erosion of the base mat. For the reactor cavity area, ABB-CE presented the view that the containment shell is encased on both top and bottom with concrete. Therefore, if the bottom portion of the shell in the region of top and bottom concrete were to be penetrated due to CCI, it would not constitute failure. In fact, for PRA purposes they will continue to assume no failure under these conditions. However, for the deterministic approach, ABB-CE will reconsider their assessment.

The staff pointed to containment integrity leak surveillance tests (e.g., Appendix J) that have pulled the steel liner from the concrete, thereby

creating a leak path due liner deformation and concrete fracture. The staff also pointed out that it is hard to justify not failing the containment when the containment shell is breached. It was also pointed out that it did not appear that the placement of the sump in the reactor cavity was well thought out with respect to a severe accident environment. The reactor cavity sump is located near the edge of the containment boundary where the top layer of concrete is of minimum thickness. Simply moving the sump toward the center should improve the ability to accommodate concrete erosion.

ABB-CE committed to reevaluate the location of the sump location and determine if suitable alternatives exist. The alternatives included: (1) evaluating an off center line location of sump; (2) reducing the size of the sump; and (3) adding a false floor in cavity (as stated in Chapter 5 of the EPRI URD).

Questions on the Structural Analyses

Due to an apparent miscommunication, ABB-CE had not received in advance the detailed structural questions generated by the NRC structural staff. As a result, ABB-CE had less than one day to look over the request. Based on this limited time, the following preliminary observations were provided to the staff.

Most, if not all, of the questions appeared to focus on the theoretical bases for the approach. Therefore, a response to these specific questions should not take much time nor put into question the basic approach taken by ABB-CE. However, what will take a significant effort will be the response to structural issues raised during this meeting. ABB-CE will attempt to answer all our questions no later than March 15, 1993.

QUESTIONS RELATED TO CE'S SYSTEM 80+ SEVERE ACCIDENT PHENOMENOLOGY AND CONTAINMENT PERFORMANCE SUBMITTAL DATED AUGUST 1992

1. Explain the development of the containment fragility curve and the accompanying calculations.

2. 3-3 ASME Service Level C loading conditions allow material strains representative of incipient yield... 3-5 For peak strains typical of incipient yield conditions, the probability of containment failure is less than 0.05. 3-7 ASME Service Level C failure probability 0.03. Explain discrepancy.

3. 3-7 Basis for assigning 5% failure probability to Ultimate Capacity ASME pressure?

4. 3-9 Penetrations designed to withstand ASME ultimate pressure and severe accident temperature. ASME ultimate pressure (169-147 psig) assigned failure probability of .05. Why not design to 1.0% strain (220-204 psig) where failure probability is equal to 1? Credit is taken for no failure of penetrations between ASME ultimate and 1.0% strain.

5. 3-9 Discuss the Equipment Survivability section of SECY-90-016 in relation to the penetrations, seals, any equipment, and instrumentation relied upon in severe accident prevention or mitigation. Provide a list of all equipment, instrumentation, seals etc., that would be relied upon for severe accidents or be exposed to severe accident conditions, and the accompanying parameters for which they are designed to tolerate. Does the containment spray system, cavity flooding valves etc., have any special requirements for severe accident operation?

6. 3-10 What credit for DBAs and severe accidents is taken for holdup and filtration in the containment annulus of the shield building? Is the annulus filtration system included in other CE plants (Palo Verde)? What was the impedus for its installation? AVS designed for design basis fission product loadings, what about severe accidents? Can the AVS be powered from the combustion turbine generator?

7. 3-13 What is the inter-relationship between the CFS and containment spray system for providing an inexhaustible continuous supply of water?

8. 3-13 Provide preliminary indications of the accident management guidance for when to manually actuate the cavity flooding system. Credit can not be taken for a pre-flooded cavity unless there is assurance of the indications used for flooding.

9. 3-13 What is the purpose of the holdup volume tank? Why not flood directly from IRWST to reactur cavity?

10. 3-15 Provide timing for filling the HVT, reactor cavity, and final level in each. Do any accident sequences result in a core mass in dry cavity or only partially filled cavity?

11. 3-15 Any equipment qualification requirements for the CFS valves?

12. 3-15 What features are discussed in "Section 4.3.2"? Is this the correct reference?

13. 3-15 How will prolonged and irreversible core uncovery be determined? Will the cavity ever be flooded when core damage is not known to have occurred?

14. 3-15 What amount of water is required in the cavity prior to introduction of the core melt, for the assumptions made regarding coolability and core concrete interaction to be true?

15. 3-15 What is the impact on containment performance and core debris coolability if the CFS works but the containment spray system does not?

16. 3-16 Provide the 100% metal-water reaction pressurization calculation. Does it assume one burn, intermittent burns, continuous burn, etc. What pre-existing containment pressures are assumed and why?

17. 3-19 Reactor coolant system pressure, where direct containment heating is no longer a concern (anticipated corium dispersal threshold value).

18. 3-20 RD capability to depressurize the RCS from 2500 to 250 psia prior to reactor vessel melt-through. How long is this? How long to core melt and vessel melt-through in representative sequences?

19. 3-20 Battery sized to power loads for 4 hours. What about SBO coping period of 8 hours? Sequences and timing for when depressurization is needed.

20. 3-20 Provide results of the MAAP runs referenced.

21. 3-23 Is the instrument shaft referenced, the same as the ICI chase in Fig. 3.6-2?

22. 3-24 DCH steam exists through louvered vents under the refueling pool. Are there any paths where the pressures associated with a HPME would force any material (insulation) out of the way creating a different flow path?

23. 3-24 Provide reactor cavity floor area and representative core debris depths for various sequences or amounts of core debris.

24. 3-27 Provide a copy of Reference 3.12 and basis for conclusion that supports coolability in the long term.

25. 3-27 How will cooling of the upper layers of corium retard any concrete attack?

26. 3-29 What type of concrete is used in the basemat?

27. 3-29 Provide analysis for 30 hours for corium to contact the containment liner.

28. 3-29 Provide discussion on how the reactor vessel and RCS are supported and what structures are required for this.

29. 3-29 What in-cavity structures might be damaged by an ex-vessel steam explosion?

30. 3-29 Discuss the area above the refueling pool not prone to missiles. What amount of concrete is below the core debris chamber and HVT sump? Are any sumps located within the reactor cavity?

31. 3-32 What is the external source of water for the containment spray system? (fire water, tee provided?)

32. 3-33 Are there dead-ended regions of the cavity where water from the containment spray system would not be recirculated to the HVT and IRWST?

33. 3-33 Can the containment spray system be powered from the combustion turbine generator? What is the impact on containment performance with no containment spray systems available? How is the containment spray system used in the MAAP analyses?

34. 4-1 What is the significance of defining early containment failure as 1 hour after the core debris penetrates the reactor vessel?

35. 4-1 Are there any areas where the System 80+ design departs from the EPRI Requirements for severe accidents?

36. 4-3 Discuss the SECY-90-016 criteria for high pressure melt ejection, such that a depressurization system should provide a rate of RCS depressurization to preclude molten-core ejection and to reduce RCS pressure sufficiently to preclude creep rupture of steam generator tubes.

37. 4-6 Is this a generic graph or specific to the System 80+ design?

38. What are the specific details of the cavity/containment design that will influence the consequences of an EVSE event and how does this relate to the statement on page 4-21, "Proper location of support structures and cavity wall design can effectively eliminate the containment threatening potential of steam explosions,"? (4.1.2.2.2, page 4-20)

39. What is the basis for statement, "the actual mass of corium expected to be involved in any one explosion is small (under 20 kg)"? (4.1.2.2.2.3, page 4-21)

40. Why was the applicability of the BETA V6.1 experiment not addressed?

41. Provide the analysis that establishes the cavity design strength to be approximately 225 psid. (4.1.2.2.4, page 4-22)

42. The cavity design strength mentioned above is for a static load yet dynamic loads are more likely in the case of a steam explosion. What is the cavity design strength for a dynamic load associated with a pressure impulse lasting 5 ms?

43. Basis for stating the most likely RV failure mechanism will be via instrument tube failure? (4.1.2.2.4, page 4-22)

44. The submittal states, "The energetics of this type of an event (FCI) were estimated in (DOE/ID-10271 "Prevention of Early Containment Failure due to High Pressure Melt Ejection and Direct Containment Heating for Advanced Light Water Reactors," March 1990.) to produce localized cavity loads in the vicinity of 10 bar. DOE/ID-10271 refers to a part of the Zion Probabilistic Safety Study that determined this cavity load. How is this analysis directly applicable to CE80+? (4.1.2.2.4, page 4-22)

45. What experimental data on corium quenching indicates that the quenching process exhibits maximum heat fluxes of up to 30 Mw/m^2 for short time periods? (4.1.2.3.2.3, page 4-23)

46. We would like to review the 21 psi pressure spike calculation and why it was assumed that the initial containment loading was at design limits (49 psig) at the time of vessel breach. (4.1.2.3.3, page 4-24)

47. Are there any other means for compensating for loss of steam inerting besides the igniters? It is not clear how the 80+ design will compensate for a loss of steam inerting once the containment sprays are activated. (4.1.3.1.2.1.1, page 4-27)

48. Further discussion of Figure 4.1-2: ALWR Combustion Potential.

49. We would like to review the analysis that establishes for 100% Metal Water Reaction complete AICC hydrogen burns result in peak containment pressures of about 140 psia. (4.1.3.1.4, page 4-32)

50. We would like to review the analysis that the vented IRWST hydrogen concentrations are only 2 v/o greater than the overall containment concentrations. (4.1.3.1.4, page 4-32)

51. What is the basis for the statement, "Igniter burns should produce pressure spikes less than that associated with a 50% core wide oxidation"?

52. What is the basis for assuming the hydrogen burn will be initiated from a 30 psia base pressure? (4.1.3.1.5, page 4-34)

53. Further discussion of Table 4.1-4: Summary of PRA Assumptions for System 80+ Hydrogen Deflagration Induced Loading and 4.1-5: Summary of System 80+ Containment Failure Probability Due to Hydrogen Deflagration.

54. Section 4.1.4.6.1 describes the containment bypass phenomena. What about failure of containment penetrations such as the personnel and

equipment hatches? (4.1.4.6.1, page 4-54)

55. Containment failure before core melt represents 62% of the containment failure frequency and 55% of the total risk for the CE System 80+. Provide an analysis for inclusion of a filtered venting system and direct venting system for the CE System 80+. A direct venting system could be considered, if scrubbing though the IRWST is expected. Provide insights to scenarios were this scrubbing may be effective and the expected decontamination factors. A filtered venting system should be considered for sequences that release directly to the containment or if the IRWST is determined to be ineffective in scrubbing.

56. Containment bypass represents 28% of the containment failure frequency and 40% of the total risk. Containment bypass consists of interfacing system LOCA and steam generator tube rupture with unisolable path to the atmosphere. Provide an indication of the benefits of inclusion of the SECY-90-016 criteria for addressing interfacing system LOCA and areas where the criteria has not been met. Provide an analysis for directing the steam from secondary side relief valves back to the containment and through the IRWST.

57. Provide the analysis of basemat melt-through including assumptions of heat fluxes, amount of core, temperature, ablation rates etc.

58. Provide a best-estimate analysis of the impact on containment performance of continued core-concrete interaction for 24 hours.

59. What are the DCH assumptions in Figure 4.1-1 for the best estimate dry cavity with RCS water case?

60. How much radial and axial ablation can the reactor cavity withstand without failure?

RAIs on CE System 80+ for Containment Performance

1. The statements under the item "Design Basis Pressure Capacity" section in page 3-3 of Reference 1 should be revised in accordance with the responses of RAIs 220.45 through 220.48 and 270.44.

2. From the Response to RAI 270.42 for general membrane stress for 3-D finite element model for the steel containment vessel with openings, the maximum stress intensity calculated for the testing load condition (load combination is $D + L + P_t + T_t$) at 53 psig is 24,614 psi and the allowable stress intensity for Level C Service Limit is 52,480 psi at design basis accident temperature of 290 °F. Since the stresses are at or below yield, a linear calculation for the allowable Level C Service Limit pressure can be determined as a check from the testing load condition as follows; 24,614 : 53 = 52,480 : X, X = 113 psig for 3-D model. The temperature is assumed to be 290°F. From Figure 3.1-2 of Reference 1 for 2-D axisymmetric thin shell model, the Level C Service Limit pressure is 145 psig. Since the internal pressure is dominant in the resulting stresses, the difference in the results (145 psig vs. 113 psig) needs to be explained.

3. In page 3-5 of Reference 1, it states, "The material properties were represented by a bilinear stress-strain curve which was assumed to be essentially elastic-perfectly plastic in nature." which means the stress is maintained at yield, while the strain is increased. However, use is made of a 5% strain hardening modulus in SA-537 Class 2 Stress-Strain Curve (Ref. 2). Provide the reasons why a bilinear stress-strain curve with a 5% strain hardening is chosen.

4. In page 3-5 of Reference 1, it states, "the strain at the maximum pressure of 193 psig is approximately 0.003 in/in." Explain how this strain and pressure can be obtained beyond the yield point using von Mises theory when is valid up to the yield point.

5. In page 3-5 of Reference 1, it states, "The exact value varies depending upon element location and whether the midsurface or inner/outer surface is examined." Explain how the membrane strain can be varied with location and across the plate thickness?

6. Assuming a bilinear stress-strain curve, the stress calculations at 0.003 in/in strain with 5% strain hardening can be performed as $\sigma_{0.003} = \sigma_y + (0.003 - \sigma_y/E) \times 0.05 \times E$, and $P_{0.003} = \sigma_{0.003}(2t)/r$ which give the values in following table.

	σ_y (psi)	E (psi)	$P_{0.003}$ (psig)	P_{80+} (psig)
110°F	59,500	29.00E6	177.5	193
290°F	52,480	28.35E6	157.8	169
350°F	51,100	27.80E6	153.7	161
450°F	48,800	27.30E6	147.1	147

   Provide the reasons why System 80+ analyses produce higher pressures.

7. In page 3-6 of Reference 1, it states, "the 0.02 in/in actual tensile failure point of SA537 Class 2 material used in the containment shell construction." Is the 0.02 in/in strain true tensile failure point for SA537 Class 2, or should it be 0.2 in/in

8. Explain how the extrapolation method in page 3-6 of Reference 1 be can used to get the ultimate pressures of 0.003 in/in strain at higher temperatures using a bilinear stress-strain curve?

9. Provide the bases for 3% and 5% of probabilities of failure at Level C and ultimate pressures, respectively, in page 3-7 of Reference 1.

10. Provide the following information:

    a. Material strength uncertainty, modelling uncertainty, and pressure distribution for containment fragility curve.

    b. Material characteristics for penetration seals to ensure minimal containment leakage at higher pressures and temperatures. It should be noted the containment should fulfill not only structural integrity function but also the leaktightness function. Structural integrity is necessary but not sufficient, because a 3/8"ϕ hole in the containment may not fulfill its function of restricting the release of radioactive materials in case of a reactor severe accident even though it is structurally sound. Therefore, it is essential to establish the leakage through the seals of various penetrations.

11. Typographical errors:

    a. In Page 3-1 of Reference 1, the material is SA537, not SA357.

    b. In the column title of membrane strain in page 3-7 of Reference 1, the unit is in/in, not % in/in.

References:

1. ALWR-FS-DCTR-33, Rev. 0, "System 80+ Severe Accident Phenomenology and Containment Performance," Combustion Engineering, Inc., August 1992.

2. Meeting handout, "System 80+™ Steel Containment Vessel Code Design Activities," Combustion Engineering, Inc., April 22 and 23, 1992.

PRELIMINARY RESPONSE TO ITEM 1/2/3:

The fragility curve is developed from shell membrane stress-strain curves based on structural analyses performed by DESI and information from NUREG-1150 (SAND-1309) "Experts Determination of Structural Response Issues", Issues 5 and 6.

Based on Issue 5:

EXPERT A

1. Membrane failure considered likely to occur between 1 and 5 % strain.

2. Onset of general yield was taken to have a containment failure probability of .02.

EXPERT B

1. Onset of general yield taken to have a cont. failure probability of .05

2. At 2% strain containment failure has a failure probability of .5

Based on Issue 6:

EXPERT C

1. Experiments show that general failure of steel containments will not occur until a global strain of 2% has been reached.

PRA Application

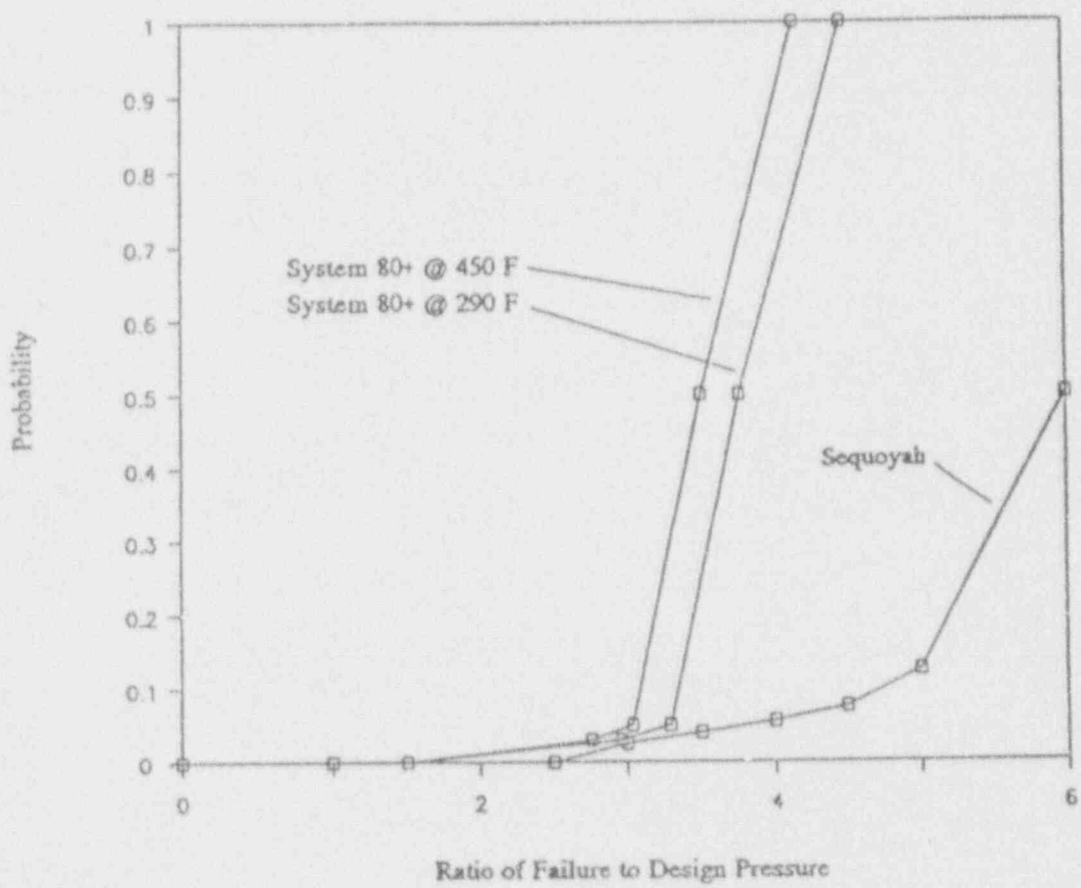
ASME LEVEL C taken a onset of general yield with a containment failure probability of .03. (Between experts A and B of issue 5)

ASME Ultimate calculation taken as a failure probability of .05 (strain = .003). Based on expert A (issue 1) and assuming the failure probability in this range is linear with strain, the failure probability for a .3% strain is between .03 and .15. A value of .05 was selected as being consistent with the Sequoyah fragility curve.

Median failure point conservatively selected based on Sequoyah fragility curve and expert judgement as 0.5% strain.

Enclosure 4

Figure 3.1-3: Comparison of Fragility Curves for System 80+



PRELIMINARY RESPONSE TO ITEM 4:

Penetrations should be designed to be consistent with the PRA assumed fragility curve. Requirements in excess of this are considered out of scope of this effort.

PRELIMINARY RESPONSE TO ITEM 5:

See response to RAI 410.141.

Also see page 41 of NUREG/CR-5567 which discusses the excellent survivability of hydrogen ignitors and other equipment in severe accident environments.

Q. 410.141 Your response to RAI 440.20 lists, in part, the hydrogen mitigation system igniters and cabling, as well as valves for the reactor cavity flooding system, as equipment that is relied upon to mitigate consequences of severe accidents. SECY-90-016 requires that there be high confidence that this equipment will survive severe accident conditions for the period that is needed to perform its intended function. However, SECY-90-016 has concluded that it is not necessary for redundant trains to be qualified to meet this goal.

With this general background, there are several areas where information is missing in your response to RAI 440.20. Therefore, please provide the following:

a. Provide the results of the calculations used to establish the environmental conditions for severe accident mitigative equipment. These conditions should include pressure, temperature, and radiation, as a function of time. In addition, provide the basis for concluding that the above conditions are bounding for the range of severe accidents.

b. In addition to the environmental conditions, provide any further criteria that will be imposed on the mitigative equipment. Indicate if these added criteria are to justify that there is reasonable assurance that this equipment will perform its function. Provide and justify the seismic design of this equipment.

c. Describe the electric power supplies for post accident mitigative equipment, including train and bus configurations supplying class 1E and alternate power sources. Describe the provisions for switching between the power sources, if required in the course of a severe accident.

Response to 410.141 a:

The equipment used in severe accident mitigation include:

(1) hydrogen mitigation system igniters and cabling,
(2) reactor cavity flooding system (CFS) valves, and
(3) safety depressurization system (SDS) valves.

The capability of igniters to function in harsh environment has been demonstrated via a number of NRC and EPRI sponsored test programs. For System 80+ application the igniters and associated cabling are expected to be available to perform their intended function if they survive the environment corresponding to the most limiting containment environment during a design basis LOCA or Main Steam Line Break. Since hydrogen combustion is not a significant threat to System 80+, the primary intent of the igniters is to minimize potential containment combustion loadings. The design basis accident (DBA) qualification range is sufficiently restrictive to encompass most severe accident scenarios. Because the low likelihood of exceeding DBA limits a more restrictive qualification criteria is considered unnecessary.

Response to 410.141 a (Cont'd)

The CFS valves are intended for operation prior to a reactor vessel breach. Therefore, these valves are not required to be qualified to extreme temperature, pressure, and radiation conditions representative of the later portions of a severe accident scenario. Thus, acceptable operation of the CFS valves is obtained during a severe accident scenario by qualifying them to design basis accident containment environmental conditions.

The SDS valves are expected to be employed for severe accident mitigation prior to, or immediately following, core uncovery. Therefore, no additional qualification testing (other than that is required for design basis accident containment environment) is considered necessary.

Response to 410.141 b:

Cavity Flooding System (CFS) and Safety Depressurization System (SDS) piping and components are designated in accordance with ASME Section III and ANSI/ANS 51.1. The CFS piping and components are Code Class 2 and Safety Class 2. The SDS piping and components that are part of the RCS pressure boundary are Code Class 1 and Safety Class 1. The remaining portions of the SDS are Code Class 2 and Safety Class 2. ASME Code and ANSI Safety Class designations for these piping and components are specified in CESSAR-DC Section 3.2 and Tables 6.7-2 and 6.8.2-1. As described in CESSAR-DC Section 3.2.1, all components in Safety Classes 1, 2, and 3 are Seismic Category I. Use of the specified classifications is intended to provide reasonable assurance that the CFS and SDS equipment will appropriately perform their functions.

Response to 410.141 c:

The major severe accident mitigative equipment that requires electric power supplies consist of (1) the hydrogen igniters, (2) the cavity flooding system, and (3) the safety depressurization system.

The hydrogen igniters are powered from the Class 1E 120V AC Vital Instrumentation and Control (I&C) Power system as described in CESSAR-DC Section 8.3.2.1.2.1 and Table 8.3.2-3. This system normally receives power from offsite power sources, with the Diesel Generators, Alternate AC Source (combustion turbine generator), or the emergency batteries supplying power if offsite power is unavailable. As described in CESSAR-DC Section 6.2.5.2.2, each igniter location consists of two igniters, one powered from each electrical division.

The cavity flooding system valves are powered from the Class 1E DC Vital Power System. Each of the four holdup volume flooding valves are powered from separate Class 1E channels and each of the two cavity flooding valves are powered from separate Class 1E divisions as seen from CESSAR-DC Table 8.3.2-4. The power to the Class 1E buses is normally supplied by either of two offsite power sources. Upon loss of both offsite power sources, the Class 1E Diesel Generators and the Class 1E batteries supply power to the buses. The diverse Alternate AC source combustion turbine generator can power these buses if power from all other sources is lost.

Response to 410.141 c (Cont'd)

The source of power for the rapid depressurization valves of the safety depressurization system is the Class 1E DC Vital Power System. The power to the Class 1E buses is normally supplied by either of two offsite power sources. Upon loss of both offsite power sources, the class 1E diesel generators and the Class 1E batteries supply power to these buses. The diverse Alternate AC Source (combustion turbine generator) can power these buses if power from all other sources is lost.

PRELIMINARY RESPONSE TO ITEM 6:

PART 1:

YES. This system is not considered necessary if realistic source terms are used.

PART 2:

NO. Palo Verde is a reinfrced concrete containment.

PART 3:

The AVS can be helpful in removing fission products following a severe accident where the containment remains intact.

This feature was not previously credited in the PRA.

PART 4:

Yes. (?)

PRELIMINARY RESPONSE TO ITEM 7

The relationship is very close.

1. CFS is indirectly replenished by sprays
2. without sprays cont. will fail even if corium or core is cooled

PRELIMINARY RESPONSE TO ITEM 8

SEE ATTACHED SHEET FOR HRA ISSUES ASSOCIATED WITH CFS ACTUATION

U..FFCFSMOVS: OPERATOR FAILS TO INITIATE CAVITY FLOODING SYSTEM

A severe design basis event has occurred. After monitoring core level, temperature and pressure indicators and realizing that regardless of corrective actions taken, reactor vessel rupture is likely to occur, the operator must initiate the Cavity Flooding System (CFS). The CFS is used to quench the debris beds in the reactor cavity by injecting water into the reactor cavity to flood the cavity. The source of this water is the In-Containment Refueling Water Storage Tank (IRWST), which delivers water first to the Holdup Volume Tank (HVT) and then to the reactor cavity. The operator must initiate this process.

There are four pathways (called "spillways") from the IRWST to the HVT. Each spillway contains one manual isolation valve which is normally open, and one motor-operated valve which is normally closed. There are two spillways from the HVT to the reactor cavity. Each spillway contains one motor-operated valve which is normally closed.

For this analysis, it is assumed that there is a procedure that requires that all the motor-operated valves be opened at the same time. In order to initiate the Cavity Flooding System, the following actions must occur:

1. Operator must recognize the symptoms of the onset of core damage. These include: loss of water level above the core which is monitored by the Reactor Vessel Level Monitoring System (RVLMS), superheated steam temperatures at the core exit which is monitored by the Core Exit Thermocouples (CETs), and core voidage which can be determined by a review of the Self Powered Neutron Detectors (SPNDs).

2. Operator must open HVT spillway motor-operated valves SI-390, SI-391, SI-392, and SI-393. This is done from the control room.

3. Operator must open reactor cavity spillway motor-operated valves SI-394 and SI-395. This is done from the control room.

For the purpose of this analysis, it is assumed that the operator must initiate Cavity Flooding before vessel failure. This gives the operator approximately 1 hour from the time that the initial indications described in item #1 occur. The time required to open the valves is less than 5 minutes. The stress level is considered to be high. It is also assumed that at least 2 Senior Reactor Operators (SROs) and 1 Shift Technical Advisor (STA) are in the control room.

The inclusion of this event in the fault tree model represents failure of the operator to open motor-operated valves SI-390, SI-391, SI-392, SI-393, SI-394, and SI-395 from the control room upon recognizing the need to actuate the Cavity Flooding System.

PRELIMINARY RESPONSE TO ITEMS 9,10 AND 11

System 80+ Cavity Flood System
Cavity Height vs. Time

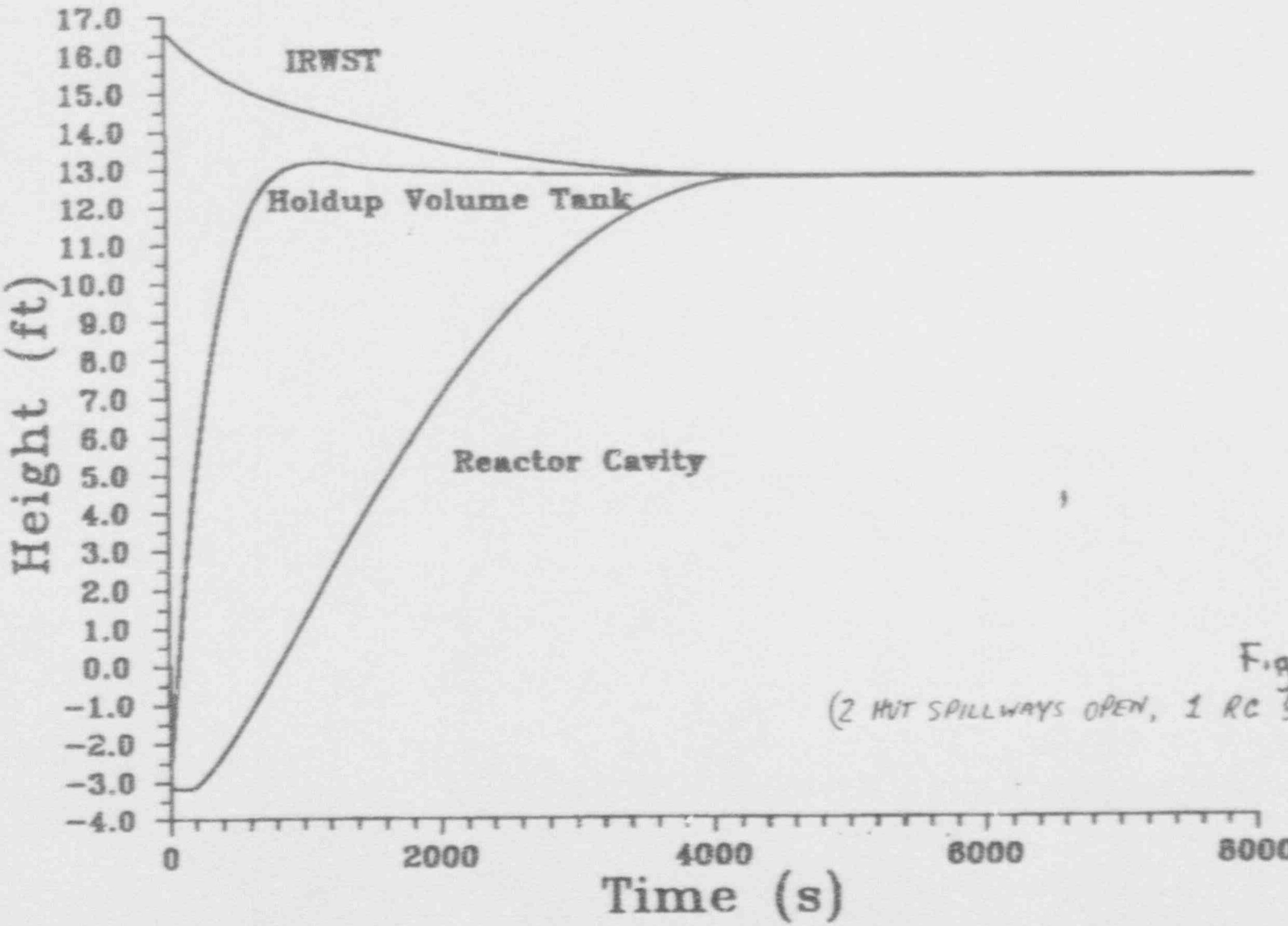


Figure 1
(2 HVT SPILLWAYS OPEN, 1 RC SPILLWAY OPEN)

3/5

System 80+ Cavity Flood System
Cavity Height vs. Time

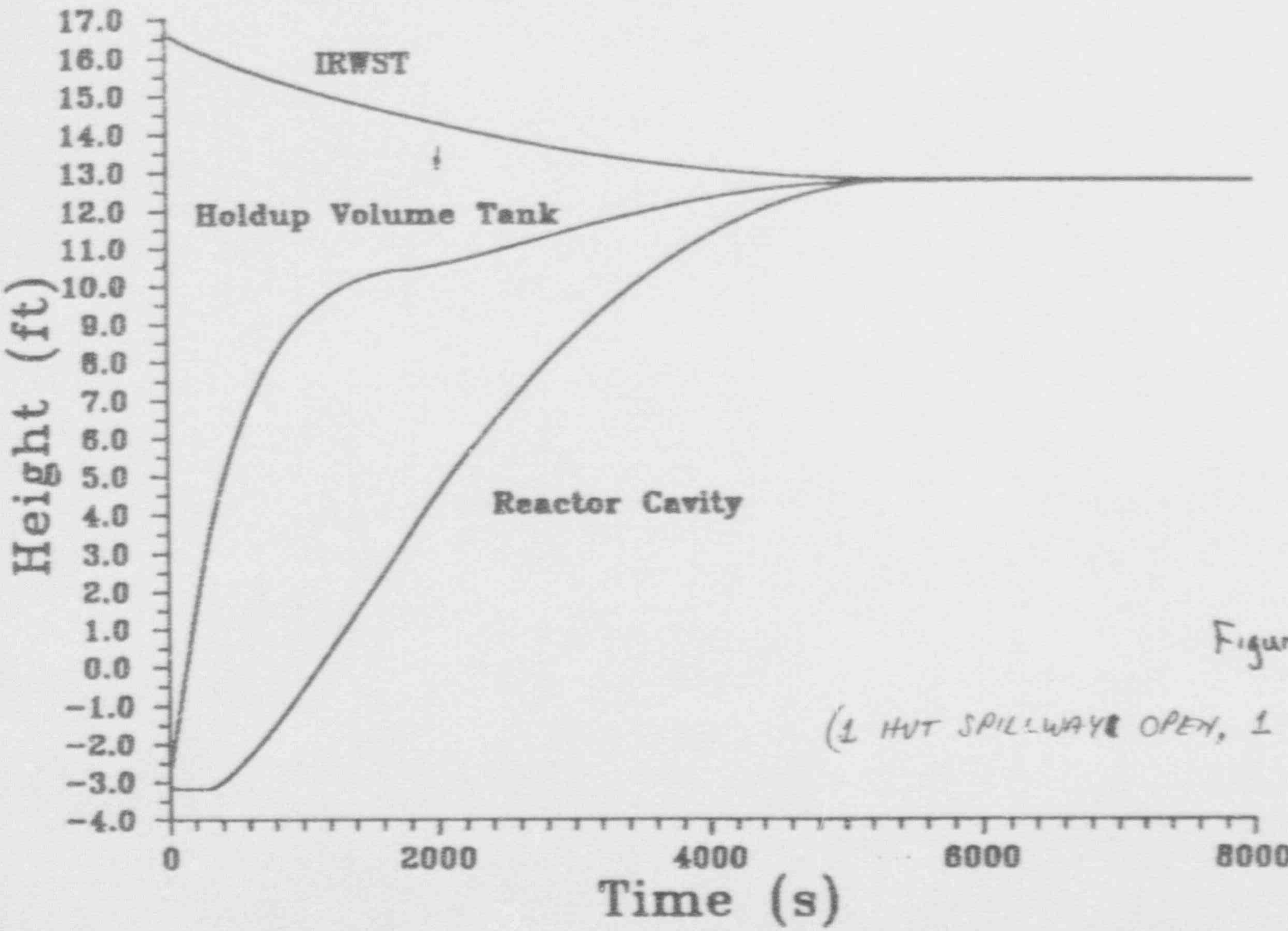


Figure 2

(1 HVT SPILLWAYE OPEN, 1 RC SPILLWAY OPEN)

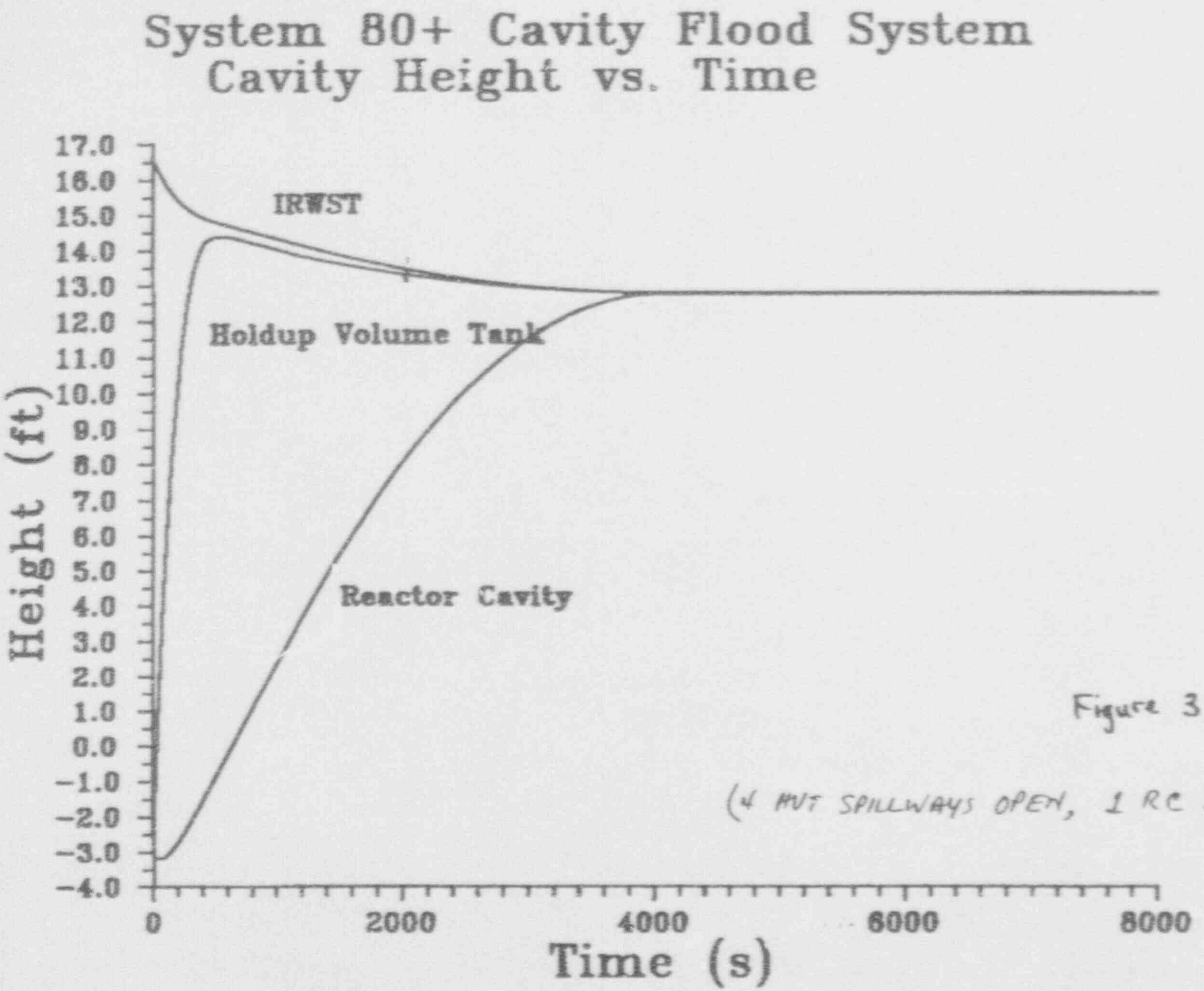


Figure 3

(4 HVT SPILLWAYS OPEN, 1 RC SPILLWAY OPEN)

5/5

PRELIMINARY RESPONSE TO ITEM 12

Section 4.3.2 discusses scrubbing capability of a water pool. An expanded discussion of the pool scrubbing feature can be added to the report.

PRELIMINARY RESPONSE TO ITEM 13

SEE RESPONSE TO ITEM 8:

There are no serious consequences of early CFS actuation.

PRELIMINARY RESPONSE TO ITEM 14

Water in the reactor Cavity prior to breach is desirable. However, the system serves a similar purpose even if the CFS actuation is delayed until after VB.

PRELIMINARY RESPONSE TO ITEM 15

If CFS steaming cannot be condensed the containment will ultimately reach failure pressure.

PRELIMINARY RESPONSE TO ITEM 16

Discussion.

PRELIMINARY RESPONSE TO ITEMS 17/18

DCH threshold is about 250 to 350 psig. See DOE/ID-10271

For a typical TLOFW event it will take about 1 hour to depressurize to 250 psig

RV failure time for this sequence will be about 4 hours.

Opening of SDS according to EOPs will afford sufficient time to establish low pressure conditions in the RCS prior to VB

PRELIMINARY RESPONSE TO ITEM 19

EXCERPT FROM CESSAR -DC PAGE 8.3-29

8.3.2.1.2.1.2 125V DC Vital Instrumentation and Control Power Batteries

Each of the independent load group channels and divisions of 125 Volt DC Vital Instrumentation and Control Power is provided with a separate and independent 125 volt battery.

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 SBO 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.

ITEM 20

DISCUSSION

RESPONSE TO ITEM 21

YES.

PRELIMINARY RESPONSE TO ITEM 27

Discussion

PRELIMINARY RESPONSE TO ITEM 23

SEE ATTACHED.

PRELIMINARY RESPONSE TO ITEM 24

SEE ATTACHED.

ITEM 25

DISCUSSION

RESPONSE TO ITEM 26

SYSTEM 80+ IS DESIGNED FOR CONSTRUCTION WITH EITHER BASALTIC OR LIMESTONE/COMMON SAND CONCRETE

PRELIMINARY RESPONSE TO ITEM 31

SEE ATTACHED FIGURE

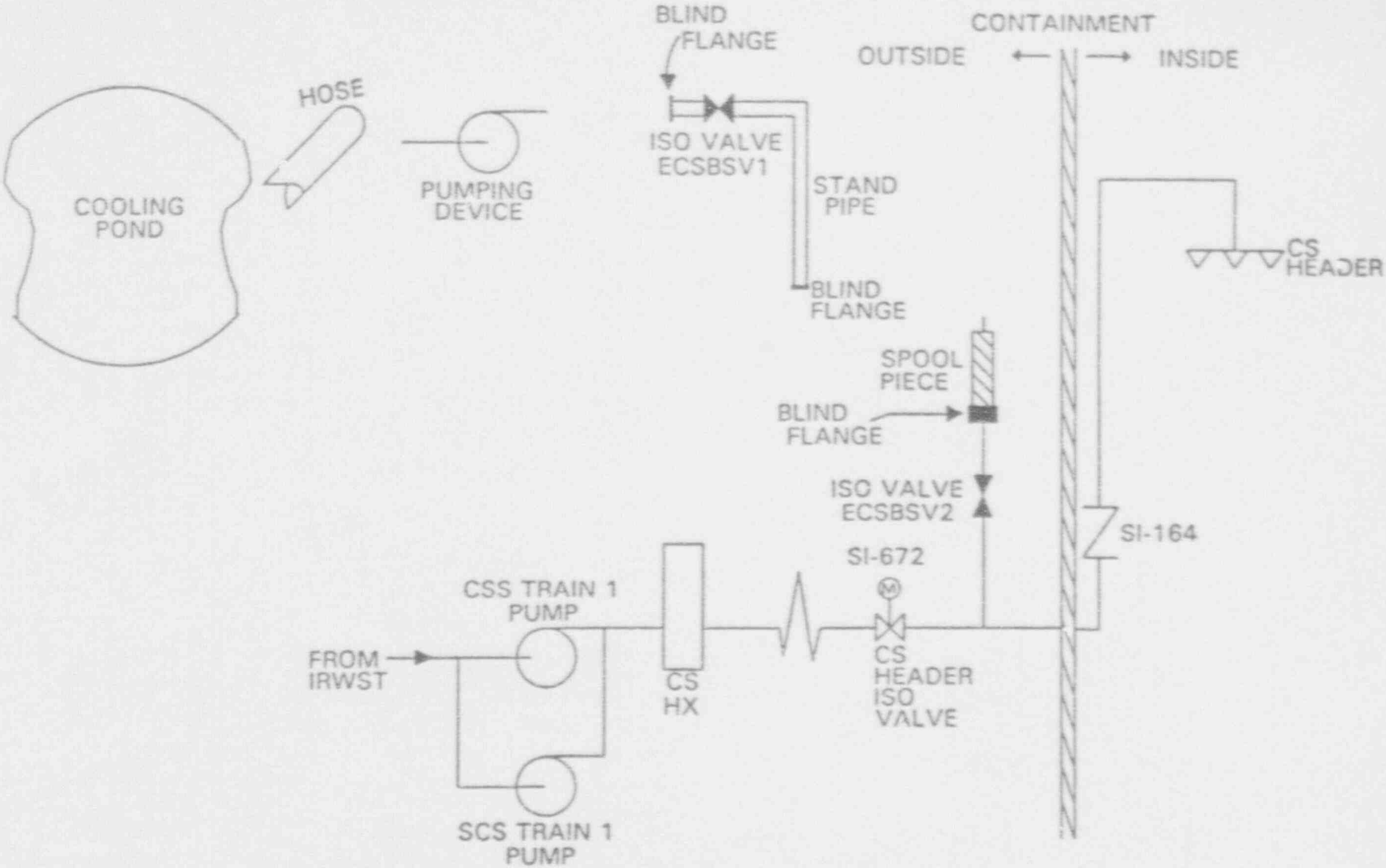


FIGURE : CONCEPTUAL DESIGN OF EMERGENCY CONTAINMENT SPRAY BACKUP SYSTEM

PRELIMINARY RESPONSE TO ITEM 37

THIS CALCULATION IS APPLICABLE TO THE SYSTEM 80+ DESIGN. SEE ATTACHED SHEET FOR DETAILS. WORK IS FROM APPENDIX A OF DOE/ID-10271.

A.0 DESCRIPTION OF BOUNDING ANALYSES

The assumptions regarding the bounding analyses are described in Section 2.1 of the main report. This appendix provides a description of the methodology and printouts of the results, which were obtained on a Lotus 123 spreadsheet.

A.1 Methodology

A.1.1 Dry Case

This case addressed the situation where no consideration was given to any liquid water in the RCS or the reactor cavity. (Steam in the RCS was assumed to be present). All the heat from the cooldown of the core debris materials and from chemical reactions was assumed to be transferred to the containment atmosphere.

The initial composition of the containment atmosphere was determined by starting with the pressure and temperature obtained from the MAAP code for the station blackout sequence, just prior to reactor vessel meltthrough. The initial amounts of oxygen and nitrogen were determined assuming a temperature of 80F prior to the start of the accident sequence. The amount of water vapor in the containment was estimated by taking the difference between the total pressure and the sum of the partial pressures of oxygen and nitrogen in containment at the point in the sequence just before meltthrough.

The heat sources were computed by summing the sensible heat of cooling from the initial temperature of the melt to the final temperature of the core materials, and adding the heats of chemical reactions. The sources of the sensible, or stored, heat were the fuel, the zirconium oxide from the in vessel oxidation fraction, the unoxidized zirconium, and the steel. The heats of reaction of zirconium oxidation in steam to form hydrogen and of subsequent hydrogen burning in oxygen were considered. Oxidation of the steel was not considered in this case because of the assumed absence of water

available to oxidize it to any great extent. It was assumed that the energy from the in-core fraction of the zirconium oxidation contributes to the initial debris temperature, taken as 2533 K, and is not counted as an additional heat source. The heat sinks were considered to be the nitrogen and water vapor initially in the containment and the oxygen remaining after the hydrogen burn. The process was assumed to be a constant volume process in thermodynamic equilibrium.

The final temperature was obtained by a trial and error solution as the temperature where the heat sinks equal the heat sources. The final pressure was obtained by summing the partial pressures of the constituents in the gas phase at the final temperature.

A.1.2 Cases With Water Present

Two cases were analyzed where water was present: one where water and steam in the RCS were assumed to codisperse with the core debris at the time of meltthrough, but where no water existed in the reactor cavity; and another identical case, except that 227,000 kg of water were assumed to exist in the reactor cavity at the time of meltthrough. In each case, the RCS water was assumed to be the liquid in the lower head and the steam in the remaining RCS volume. Water in the pressurizer was not considered because it would follow later in the depressurization sequence.

The heat sources were identical to those of the dry case, except that the oxidation of steel was allowed. The heat sinks were also identical, with the addition of the effect of the additional water. If the heat balance indicated that liquid water remained in the system at the equilibrium condition, saturation properties were used to determine the final temperature and internal energies. If the heat balance indicated that no water remained, a separate area of the spreadsheet was used to calculate the final condition, using superheat properties that were calculated starting with the saturation properties at the given pressure. The solution was obtained by assuming an initial temperature, which affected the pressure, which affected the steam

saturation properties. These were used to compute the heat sources and heat sinks and interated until a convergent solu''on was reached.

A.2 Properties

The thermodynamic properties were as follows:

A.2.1 Properties

(Values at 1500K, unless otherwise stated, from MAAP User's Guide).[A1]

- C_v UO_2 = 333 J/Kg-K
- C_v Z_r = 356 J/Kg-K
- C_v O_2 = 660 J/Kg-K
- C_v steam = 1760 J/Kg-K[A2]
- C_v ZrO_2 = 645 J/Kg-K
- C_v N_2 = 750 J/Kg-K
- C_v Steel = 663 J/kg - k
- Heat of zirconium reaction with water = 5.51E8 J/Kg-mol ZrO_2 (at 2500 K)
- Heat of H oxidation = 2.40E8 J/Kg-mol H_2O (at 2500 K)
- Heat of iron oxidation = 4.113E8 J/kg-mol Fe_2Q_3[A3]
- Steam properties from Reference A2 and A4

A.2.2 Physical Parameters

- o Initial containment temperature = 340 K
- o Mass UO_2 in core = 112,000 Kg
- o Mass zirconium in core = 33,000 Kg
- o Mass steel in ejection = 10,000 Kg
- o Containment volume = 3.3E6 ft.^{3} = 99,400 m^3
- o Mass of water in reactor vessel bottom head = 43,300 kg
- o Mass of water vapor in RCS and 200 psi = 29,800 kg

A.3 Calculations Performed

The spreadsheet program has the capability of varying the following:

Core debris initial temperature
Containment initial temperature
Fraction of core ejected
Amount of steel ejected
Fraction of ex vessel hydrogen reacting
Fraction of steel oxidizing
Initial amounts in containment
Amount of water in the cavity
Amount of water in the reactor vessel
Amount of steam in the reactor vessel.

The term "fraction of core ejected" means the fraction of the core which is participating in the energy transfer to the containment atmosphere. The core could be ejected from the vessel, but its participation in further

energy transfer could be inhibited by the cavity configuration, access to reactants, particle dynamics, reentrainment, and other factors identified in more depth in other parts of this report. The three cases discussed in Section 2.1 of the main report were run for varying fractions of the core ejected to illustrate sensitivity to the performance of the reactor cavity. The other parameters were set at best estimate values consistent with other analyses.

A.4 Results

Table A1 shows the results for the dry case. Table A2 for the case where RCS water codisperses with the core debris during the high pressure melt ejection. Table A3 is similar to Table A2, except 227,000 Kg water was assumed to be present in the reactor cavity. These results are summarized in Figure 1 of the main report.

Table A-1
Case 1. Containment Pressure vs. Core Fraction

COPC10A3 co dispersal. P vs FRACTION CORE, 10000 KG STEEL (X FRCORE)OX.
CONTAINMENT PRESSURE CALCULATION, NO CAVITY WATER, RCS STEAM ONLY, VARY CORE FRACTION
2533K, .245%Zr, 89%H

	Baseline Col Do not use						
. ..ore hot initial Th (K)	2533	2533	2533	2533	2533	2533	2533
T containment initial Ti (K)	350	350	350	350	350	350	350
Fraction of core ejected	1	1	0.8	0.6	0.4	0.2	0
Amount of steel (kg)	10000	10000	10000	10000	10000	10000	10000
Fraction of Zirc reacted in vessel	0.245	0.245	0.245	0.245	0.245	0.245	0.245
Ex vessel hydrogen, fraction of max	1	1	1	1	1	1	1
Fraction of steel oxidizing	1	0	0	0	0	0	0
INITIAL AMOUNTS IN CONTAINMENT							
Moles of O_2 = p(tot)(.2)V/RT Kg-mol	719	719	719	719	719	719	719
Moles of N_2 = p(tot)(.8)V/RT Kg-mol	2877	2877	2877	2877	2877	2877	2877
Moles of H_2O init as vapor	313	313	313	313	313	313	313
Water in cavity, Kg	217000	0	0	0	0	0	0
Fraction of cavity water participating	1	1	1	1	1	1	1
WATER IN RCS							
Volume = (4/3)pi*r3/2 (m3)	43.31982	43.31982	43.31982	43.31982	43.31982	43.31982	43.31982
Mass Liquid = Vol*1000Kg/m^3 (kg)	43319.82	0	0	0	0	0	0
u sat RCS water at 13.61 MPa (J/kg)	1672701.	1672701.	1672701.	1672701.	1672701.	1672701.	1672701.
T sat RCS water at 2000 psia (K)	608.8888	608.8888	608.8888	608.8888	608.8888	608.8888	608.8888
Mass RCS vapor (kg) =	29800	29800	29800	29800	29800	29800	29800
Cavity water boiling point K	435	467.7777	463.4917	457.3133	448.3032	435.9892	0
u sat RCS vapor at 2000 psia (J/kg)	2357000	2357000	2357000	2357000	2357000	2357000	2357000
HEAT SOURCES (Rows don't match A1 spreadsheet from here on)							
Cooldown of UO_2 to heat water to Tfinal	7.8E+10	-5.0E+09	3.5E+09	9.4E+09	1.2E+10	9.2E+09	0
Cool init reacted ZrO_2 to ht wat to Tf	1.5E+10	-9.5E+08	6.6E+08	1.8E+09	2.2E+09	1.7E+09	0
Cool init unreacted ZrO_2 to ht wat to Tf	4.5E+10	-2.9E+09	2.0E+09	5.4E+09	6.8E+09	5.3E+09	0
Cool steel	1.4E+10	-9.0E+08	6.3E+08	1.7E+09	2.1E+09	1.6E+09	0
Heat of reaction, initially unreacted Zr	1.5E+11	'.5E+11	1.2E+11	9.0E+10	6.0E+10	3.0E+10	0
Heat of reaction H burn	1.7E+11	.7E+11	1.5E+11	1.2E+11	9.5E+10	6.9E+10	4.3E+10
Ht of reaction Fe to H to H_2O	7.4E+10	0	0	0	0	0	0
Cool H_2O from burn(No, already in Q burn)	6.5E+10	-3.1E+09	2.3E+09	6.7E+09	9.8E+09	1.1E+10	1.0E+10
Sum of heat sources	6.1E+11	3.1E+11	2.8E+11	2.4E+11	1.9E+11	1.3E+11	5.3E+10
. ..ESS final temp (K) (input at row 77)	450	2668	2415	2114	1751	1306	747
u(liq) at initial conditions 80F (J/kg)	111385.8	111385.8	111385.8	111385.8	111385.8	111385.8	111385.8
u(liquid) at final condition X psia	677037.9	814278.4	796332.7	770464.0	732738.4	681179.6	591540.2
u(sat vapor) at final psia (J/Kg)	2564164.	2583290.	2582617.	2578548.	2572905.	2565006.	2544497.
Sat vapor sp. vol. at final psia (m3/kg)	0.294100	0.142575	0.155437	0.178173	0.215537	0.287587	0.493994
Mass H_2O initially in vapor + from burn	18658.8	18658.8	16692.05	14725.31	12758.56	10791.82	8825.076
Mass water evap from cavity	217000	0	0	0	0	0	0
Mass water evap from RCS (incl init vap)	73119.82	29800	29800	29800	29800	29800	29800
Total water evaporated	290119.8	29800	29800	29800	29800	29800	29800
u final at T,p final, J/kg	2443125.	6147811.	5730450.	5232601.	4634116.	3901626.	2981354.
U gain all water vapor to Tfinal	5.4E+11	1.4E+11	1.2E+11	1.0E+11	8.2E+10	5.6E+10	2.3E+10
NO MORE CAVITY WATER, NO ENTRY							
T	450	2668	2415	2114	1751	1306	747
ADDITIONAL HEAT SINKS							
Heatup of O_2	1.5E+09	3.5E+10	3.1E+10	2.7E+10	2.1E+10	1.5E+10	6.0E+09
Heatup of N_2	6.0E+09	1.4E+11	1.2E+11	1.1E+11	8.5E+10	5.8E+10	2.4E+10
Sum of heat sinks (incl. U tot H_2O)	5.5E+11	3.1E+11	2.8E+11	2.4E+11	1.9E+11	1.3E+11	5.3E+10
Heat sources minus heat sinks	6.4E+10	94813858	-1.1E+08	-4.8E+07	61708452	53864909	21909863
Moles O_2 left after combustion	222.9117	357.2	411.8318	466.4636	521.0954	575.7272	630.359
temp R	810	4802.4	4347	3805.2	3151.8	2350.8	1344.6
FINAL PRESSURE AT WATER B.P.							
P N_2 English units psi	16.66861	98.82638	89.45492	78.30546	64.85944	48.37603	27.66990
P O_2 left after combustion	1.291494	12.26999	12.80513	12.69608	11.74763	9.680708	6.062556
" H_2O sum	99.38811	92.47688	80.31018	67.32659	53.30253	37.91896	20.63786
& PRESSURE TOTAL PSI	117.3482	203.5732	182.5702	158.3281	129.9096	95.97570	54.37032
After initial guess, cell ref to Row 179							
Guess pressure		0	0	0	0	0	0
INPUT Guess superheat Temp (K)	450	2668	2415	2114	1751	1306	747
Heat sources minus heat sinks	6.4E+10	94813858	-1.1E+08	-4.8E+07	61708452	53864909	21909863

TABLE A-2

Case 2. Containment Pressure vs. Case Fraction

COPC6A3 co dispersal. P vs FRACTION CORE, 10000 KG STEEL (X FROOREICX
CONTAINMENT PRESSURE CALCULATION, NO CAVITY WATER, RCS WATER AND STEAM, VARY CORE FRACTION
2533K, .245FrZr, 89kH

	Baseline Col Do not use						
T core hot initial Th (K)	2533	2533	2533	2533	2533	2533	2533
T containment initial Ti (K)	350	350	350	350	350	350	350
Fraction of core ejected	1	1	0.5	0.6	0.4	0.2	0
Amount of steel (kg)	10000	10000	10000	10000	10000	10000	10000
Fraction of Zirc reacted in vessel	0.245	0.245	0.245	0.245	0.245	0.245	0.245
Ex vessel hydrogen, fraction of max	1	1	1	1	1	1	1
Fraction of steel oxidizing	1	1	1	1	1	1	1
INITIAL AMOUNTS IN CONTAINMENT							
Moles of O2 = p(tot)(.21)V/RT Kg-mol	719	719	719	719	719	719	719
Moles of N2 = p(tot)(.81)V/RT Kg-mol	2877	2877	2877	2877	2877	2877	2877
Moles of H2O init as vapor	313	313	313	313	313	313	313
Water in cavity, Kg	217000	0	0	0	0	[illegible]	0
Fraction of cavity water participating	1	1	1	1	1	1	1
WATER IN RCS							
Volume = (4/3)pi*r3/2 (m3)	43.31982	43.31982	43.31982	43.31982	43.31982	43.31982	43.31982
Mass Liquid= Vol*1000Kg/m^3 (kg)	43319.82	43319.82	43319.82	43319.82	43319.82	43319.82	43319.82
u sat RCS water at 13.61 MPa (J/kg)	1672701.	1672701.	1672701.	1672701.	1672701.	1672701.	1672701.
T sat RCS water at 2000 psia (K)	608.8888	608.8888	608.8888	608.8888	608.8888	608.8888	608.8888
Mass RCS vapor (kg) =	29800	29800	29800	29800	29800	29800	29800
Cavity water boiling point K	435	467.7777	467.7777	462.8696	452.7848	437.9658	0
u sat RCS vapor at 2000 psia (J/kg)	2357000	2357000	2357000	2357000	2357000	2357000	2357000
HEAT SOURCES (Rows don't match A1 spreadsheet from here on)							
Cooldown of UO2 to heat water to Tfinal	7.8E+10	9.5E+09	1.5E+10	1.8E+10	1.7E+10	1.2E+10	0
Cool init reacted ZrO2 to ht wat to T f	1.5E+10	1.8E+09	2.9E+09	3.4E+09	3.3E+09	2.2E+09	0
Cool init unreacted ZrO2 to ht wat to Tf	4.5E+10	5.5E+09	8.9E+09	1.1E+10	1.0E+10	6.8E+09	0
Cool steel	1.4E+10	1.7E+09	2.7E+09	3.2E+09	3.1E+09	2.1E+09	0
Heat of reaction, initially unreacted Zr	1.5E+11	1.5E+11	1.2E+11	9.0E+10	6.0E+10	3.0E+10	0
Heat of reaction H burn	1.7E+11	1.7E+11	1.5E+11	1.2E+11	9.5E+10	6.9E+10	4.3E+10
Ht of reaction Fe to H to H2O	7.4E+10	7.4E+10	5.9E+10	4.4E+10	2.9E+10	1.5E+10	0
Cool H2O from burn(No, already in Q burn)	6.5E+10	8.0E+09	1.3E+10	1.7E+10	1.9E+10	1.7E+10	1.2E+10
Sum of heat sources	6.1E+11	4.2E+11	3.7E+11	3.1E+11	2.4E+11	1.5E+11	5.4E+10
GUESS final temp (K) (input at row 77)	450	2279	2020	1720	1372	965	481
u(liq) at initial conditions 80F (J/Kg)	111385.8	111385.8	111385.8	111385.8	111385.8	111385.8	111385.8
u(liquid) at final condition X psia	677037.9	814278.4	814278.4	793727.9	751503.0	689456.0	579060.0
u(sat vapor) at final psia (J/Kg)	2564164.	2583290.	2583290.	2582155.	2576020.	2566588.	2541284.
Sat vapor sp. vol. at final psia (m3/kg)	0.294100	0.142575	0.142575	0.157250	0.196300	0.274667	0.531519
Mass H2O initially in vapor + from burn	18658.8	18658.8	16692.05	14725.31	12758.56	10791.82	8825.076
Mass water evap from cavity	217000	0	0	0	0	0	0
Mass water evap from RCS (incl init vap)	73119.82	73119.82	73119.82	73119.82	73119.82	73119.82	73119.82
Total water evaporated	290119.8	73119.82	73119.82	73119.82	73119.82	73119.82	73119.82
u final at T,p final, J/kg	2443125.	5496044.	5062090.	4566583.	3994562.	3328478.	2537605.
U gain all water vapor to Tfinal	5.4E+11	2.8E+11	2.4E+11	2.0E+11	1.6E+11	1.1E+11	4.4E+10
NO MORE CAVITY WATER, NO ENTRY							
T	450	2279	2020	1720	1372	965	481
ADDITIONAL HEAT SINKS							
Heatup of O2	1.5E+09	2.9E+10	2.5E+10	2.1E+10	1.6E+10	9.3E+09	2.0E+09
Heatup of N2	6.0E+09	1.2E+11	1.0E+11	8.3E+10	6.2E+10	3.7E+10	7.9E+09
Sum of heat sinks (incl. U sat H2O)	5.5E+11	4.2E+11	3.7E+11	3.1E+11	2.4E+11	1.5E+11	5.4E+10
Heat sources minus heat sinks	6.4E+10	1.3E+08	-1.9E+08	-5.8E+07	1.2E+08	13484289	19915827
Moles O2 left after combustion	222.9117	222.9117	304.4011	385.8906	467.3800	548.8695	630.359
temp R	810	4102.2	3636	3096	2469.6	1737	865.8
FINAL PRESSURE AT WATER B.P.							
P N2 English units psi	16.66861	84.[illegible]1729	74.82357	63.71116	50.82076	35.74492	17.81690
P O2 left after combustion	1.291494	6.540703	7.916713	8.545548	8.256035	6.819361	3.903734
P H2O sum	99.38811	149.5099	129.7656	108.0739	84.27771	57.91942	28.19302
[illegible]AL PRESSURE TOTAL PSI	117.3482	240.5679	212.5059	180.3306	143.3545	100.4837	49.91366
After initial guess, cell ref to Row 179							
Guess pressure		0	0	0	0	0	0
INPUT Guess superheat Temp (K)	450	2279	2020	1720	1372	965	481
Heat sources minus heat sinks	6.4E+10	1.3E+08	-1.9E+08	-5.8E+07	1.2E+08	13484289	19915827

Table A-3
Containment Pressure vs. Core Fraction

OPC7A4 co dispersal. P vs FRACTION CORE, 10000 KG STEEL (X FRCORE)OK
CONTAINMENT PRESSURE CALCULATION, CAVITY WATER, RCS WATER AND STEAM, VARY CORE FRACTION
533K, .245FrZr, 8PWH

hot initial Th (K)		2533
ainment initial Ti (K)		350
raction of core ejected		1
mount of steel (kg)		10000
raction of Zirc reacted in vessel		0.245
x vessel hydrogen, fraction of max		1
raction of steel oxidizing		1
NITIAL AMOUNTS IN CONTAINMENT		
Moles of O2 = p(cor)X.21V/RT Kg-mol		719
Moles of N2 = p(cot)X.81V/RT Kg-mol		2877
Moles of H2O init as vapor	155	313
Water in cavity, Kg		227000
Fraction of cavity water participating		1
/ATER IN RCS		
olume = (4/3)pi*r3/2 (m3)		43.31982
lass Liquid= Vol*1000Kg/m^3 (kg)		43319.82
nt RCS water at 13.61 MPa (J/kg)		1672701.
sat RCS water at 2000 psia (K)		608.8888
lass RCS vapor (kg) =		29800
Cavity water boiling point K		450.1009
sat RCS vapor at 2000 psia (J/kg)		2357000
EAT SOURCES (Rows don't match A1 spreadsh		
ooldown of UO2 to heat water to Tfinal		7.6E+10
ool init reacted ZrO2 to ht wat to T f		1.4E+10
ool init unreacted ZrO2 to ht wat to Tf		4.4E+10
ool steel		1.3E+10
eat of reaction, initially unreacted Zr		1.5E+11
eat of reaction H burn		1.7E+11
t of reaction Fe to H to H2O		7.4E+10
ool H2O from burn(No, already in Q burn)	0	6.4E+10
ium of heat sources		6.1E+11
UESS final temp (K) (input at row 77)		505
(kg) at initial conditions 80F (J/Kg)		111385.8
(liquid) at final condition X psia		740265.7
sat vapor) at final psia (J/Kg)		2573892.
at vapor sp. vol. at final psia (m3/kg)		0.207495
lass H2O initially in vapor + from burn		18658.8
lass water evap from cavity		227000
lass water evap from RCS (incl init vap)		73119.82
otal water evaporated		300119.8
final at T,p final, J/kg		2544377.
gain all water vapor to Tfinal		6.0E+11
IO MORE CAVITY WATER, NO ENTRY		
		505
DDITIONAL HEAT SINKS		
Heatup of O2		2.4E+09
Heatup of N2		9.4E+09
um of heat sinks (incl. U sat H2O)		6.1E+11
eat sources minus heat sinks		-9568325
Moles O2 left after combustion		222.9117
emp R		909
NAL PRESSURE AT WATER B.P.		
P N2 English units psi		18.70589
P O2 left after combustion		1.449344
P H2O sum		115.1477
NAL PRESSURE TOTAL PSI		135.3029
fter initial guess, cell ref to Row 179		
uess pressure		0
NPUT Guess superheat Temp (K)		505
ources minus heat sinks		-9568325

Table A-3. (Continued)

COPC7A4 (RCS vap incl co dispersal P vs FRACTION CORE, 10000 KG STEEL (X FRCORE)(K, GO TO K1 FOR RESULTS
CONTAINMENT PRESSURE CALCULATION, CAVITY WATER, RCS WATER AND STEAM
DO NOT INSERT COLUMNS. KEEP COL B AS SOURCE. DO NOT "MOVE" COLUMNS
CAVITY DRIES OUT, USE SHEET K1 ROUTINE. Contains props to .2 psi
.133K, .245FrZr, 89GH

T core hot initial Th (K)	2533	2533	2533	2533	2533
T containment initial Ti (K)	350	350	350	350	350
Fraction of core ejected	0.8	0.6	0.4	0.2	0
Amount of steel (kg)	10000	10000	10000	10000	10000
Fraction of Zirc reacted in vessel	0.245	0.245	0.245	0.245	0.245
Ex vessel hydrogen, fraction of max	1	!	1	1	1
Fraction of steel oxidizing	1	1	1	1	1
INITIAL AMOUNTS IN CONTAINMENT					
Moles of O2 = p(satK.21V/RT Kg-mol	719	719	719	719	719
Moles of N2 = p(satK.81V/RT Kg-mol	2877	2877	2877	2877	2877
Moles of H2O init as vapor	313	313	313	313	313
Water in cavity, Kg *****	227000	227000	227000	227000	227000
Fraction of cavity water participating	1	1	1	1	1
WATER IN RCS					
Volume = (4/3)pi*r3/2 (m3)	43.31982	43.31982	43.31982	43.31982	43.31982
Mass liquid = Vol*1000Kg/m^3 (kg)	43319.82	43319.82	43319.82	43319.82	43319.82
u sat RCS water at 13.61 MPa (J/kg)	1672701.	1672701.	1672701.	1672701.	1672701.
T sat RCS water at 2000 psia (K)	608.8888	608.8888	608.8888	608.8888	608.8888
Mass RCS vapor (kg) =	29800	29800	29800	29800	29800
Cavity water boiling point K	425.9	417	405.1	392	374
u sat RCS vapor at 2000 psia (J/kg)	2475000	2475000	2475000	2475000	2475000
HEAT SOURCES					
Cooldown of UO2 to heat water to Tfinal	6.3E+10	4.7E+10	3.2E+10	1.6E+10	0
Cool init reacted ZrO2 to ht wat to Tf	1.2E+10	8.9E+09	6.0E+09	3.0E+09	0
Cool init unreacted ZrO2 to ht wat to Tf	3.7E+10	2.8E+10	1.8E+10	9.3E+09	0
Cool steel	1.1E+10	8.4E+09	5.6E+09	2.8E+09	0
Heat of reaction, initially unreacted Zr	1.2E+11	9.0E+10	6.0E+10	3.0E+10	0
Heat of reaction H burn	1.5E+11	1.2E+11	9.5E+10	6.9E+10	4.3E+10
Ht of reaction Fe to H to H2O	5.9E+10	4.4E+10	2.9E+10	1.5E+10	0
Cool H2O from burn	5.5E+10	4.5E+10	3.4E+10	2.3E+10	1.2E+10
Sum of heat sources	5.0E+11	3.9E+11	2.8E+11	1.7E+11	5.5E+10
GUESS final temp (K) (input at row 77)	425.9	417	405.1	392	374
Estimated final vapor pressure MPa	3.503552	0.394557	0.282701	0.190476	0.107123
Estimated final vap pres (psi)	74.02222	58	41.55714	28	15.74712
u(liq) at initial conditions 80F (J/Kg)	111385.8	111385.8	111385.8	111385.8	111385.8
u(liquid) at final condition X psia	638936.2	601671.9	551846.6	496996.9	421630.9
u(sat vapor) at final psia (J/Kg)	2556148.	2547118.	2535681.	2521639.	2500610.
Sat vapor sp. vol. at final psia (m3/kg)	0.368103	0.463712	0.634534	0.932654	1.735980
Mass vap. at final psi (kg)	270032.9	214356.9	156650.3	106577.4	57258.70
Mass H2O initially in vapor + from burn	20559.55	17625.93	14692.31	11758.69	8825.076
Mass wat evap from cavity & rx (ladion)	249473.3	196731.0	141957.9	94818.78	48433.62
Mass water remaining, cav + RCS (-=dry)	50646.45	103388.7	158161.8	205301.0	251686.1
Correction for RCS energy	1.4E+11	1.4E+11	1.4E+11	1.4E+11	1.4E+11
Correction to not double count					
U gain all water (J)* to Tsat	5.0E+11	3.9E+11	2.8E+11	1.7E+11	5.6E+10
T sat (K)	425.9	417	405.1	392	374
ADDITIONAL HEAT SINKS					
Heatup of O2	1.2E+09	1.0E+09	8.4E+08	6.4E+08	3.6E+08
Heatup of N2	4.6E+09	4.0E+09	3.3E+09	2.5E+09	1.5E+09
Heatup of H2O init in containment	7.5E+08	6.6E+08	5.5E+08	4.2E+08	2.4E+08
Sum of heat sinks (incl. U sat H2O)	5.1E+11	4.0E+11	2.8E+11	1.7E+11	5.8E+10
Heat sources minus heat sinks	-4.8E+08	-4.9E+09	-4.3E+07	-5.4E+09	-3.1E+09
Moles O2 left after combustion	304.4011	385.8906	467.3800	548.8695	630.359
temp R	766.62	750.6	729.18	705.6	673.2
FINAL PRESSURE AT FINAL T					
P N2 English units psi	15.77592	15.44625	15.00546	14.52021	13.85347
P O2 left after combustion	1.669172	2.071798	2.437696	2.770144	3.035336
H2O sum (Psat)	74.02222	58	41.55714	28	15.74712
FINAL PRESSURE TOTAL PSI	91.46731	75.51805	59.00030	45.29036	32.63593
Original guess temp (K)	425.9	417	405.1	392	374
Heat sources minus heat sinks	-4.8E+08	-4.9E+09	-4.3E+07	-5.4E+09	-3.1E+09

A.5 REFERENCES

A1. Fauske and Associates, Inc., Technical Report 16.2-3, MAAP (3.0) Modular Accident Analysis Program User's Manual - Vol II, Atomic Industrial Forum, Bethesda, MD, February, 1987.

A2. E. A. Avallone, and T. Baumeister, Eds., Marks' Standard Handbook for Mechanical Engineers, Ninth Edition, McGraw-Hill Book Company, New York, 1978.

A3. R. H. Perry, et al., Eds., Chemical Engineers' Handbook, Fourth Edition, McGraw-Hill Book Company, New York, 1963.

A4. J. H. Keenan and F. G. Keyes, Thermodynamic Properties of Steam, John Wiley and Sons, New York, 1936.

PRELIMINARY RESPONSE TO ITEM 39

S[illegible] explosions are millisecond phenomena and are typically triggered once a mass of corium reaches a solid surface (floor).

Since the t[illegible]al failure mode of the RV is via failure of small instrum[illegible]rations, the loading associated with any given steam exp[illegible] be that associated with the mass in the water pool from [illegible]r" of corium with the diameter equal to the ICI tube diame[illegible]a. the depth of the pool. For a 1" diameter tube and a 14 foot p[illegible] the volume of corium contributing to the EVSE would be .076[illegible] This ie equivalent to a mass of 41.9 lbm or 19 kg.

RESPONSE TO ITEM 40

BETA V6.1 information was unavailable.

PRELIMINARY RESPONSE TO ITEM 41

The containmnet strength qu[illegible]d in the document i[illegible] epresentative of cavity wall strenghts typ[illegible]l of recent C-E d[illegible]ned PWRs.

PLANT	Cavity Wall Design Strength
WSES	240 psid
SCE	229 psid
Millstone 2	247 psid

PRELIMINARY RESPONSE TO ITEM 43

The RV lower head failure map of NUREG/CR-5642 demonstrates that the most likely mode of RV failure will be caused by either instrument tube ejection or tube rupture. These maps are generally applicable to the System 80+ lower head. See attached sheet.

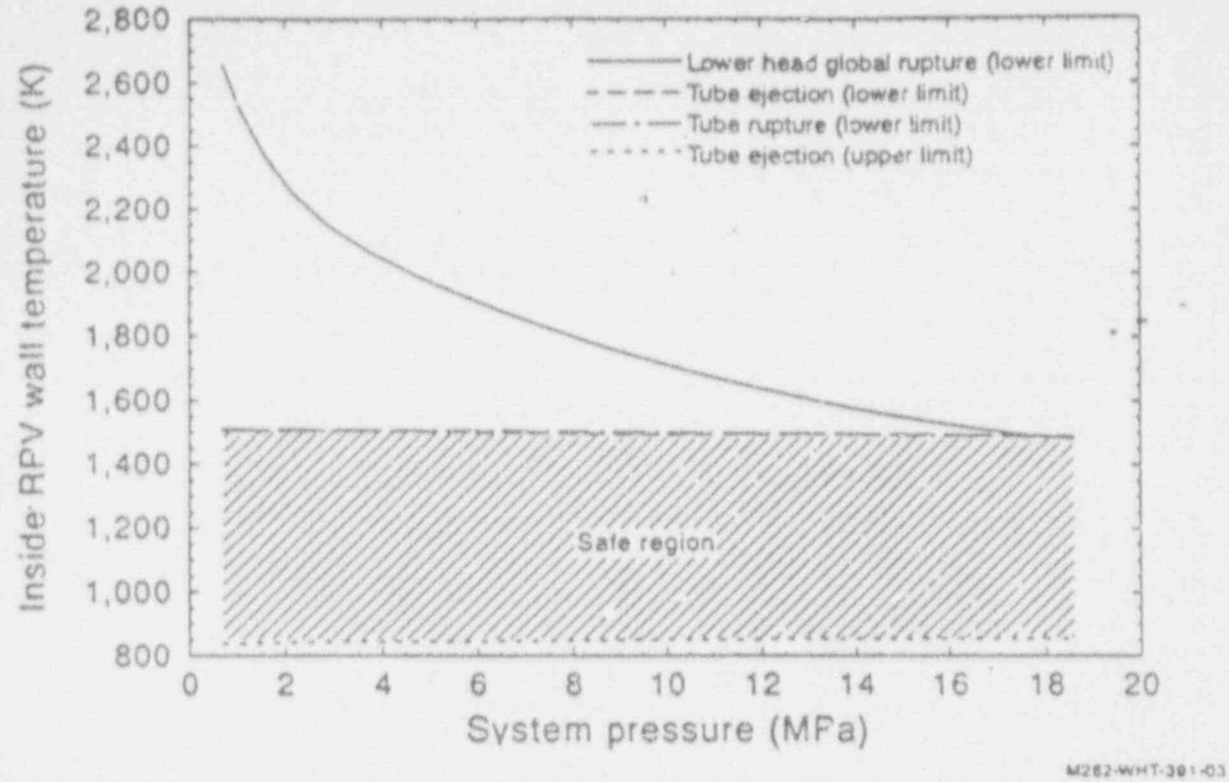


Figure 4-28. Westinghouse instrument guide tube failure map at maximum radial gap (0.008 cm).

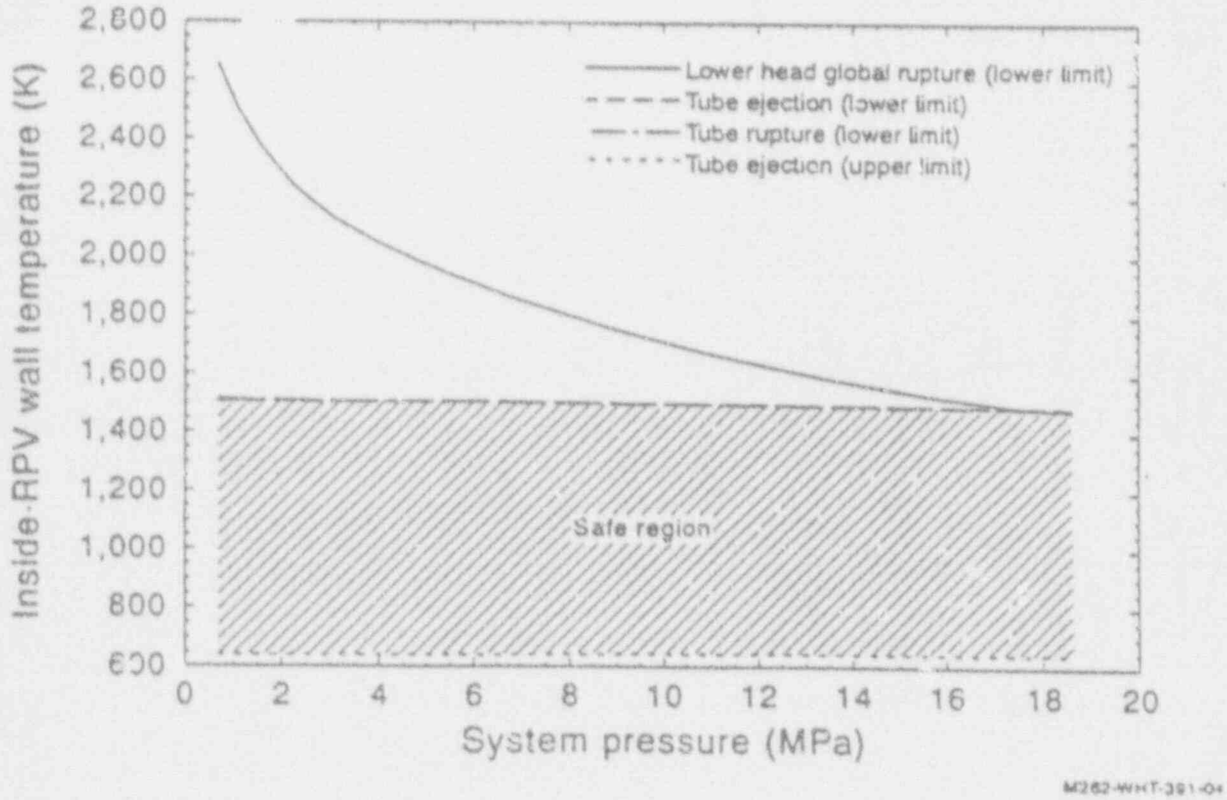


Figure 4-29. Westinghouse instrument guide tube failure map at minimum radial gap (0.002 cm).

REF: NUREG/CR-5642 LIGHT water Reactor lower head Failure Analysis (DRAFT), Oct. 1991

PRELIMINARY RESPONSE TO ITEM 44

Calculations employed a water filled cavity, and a PWR with an instrumented lower head.

The CE design has a larger cavity, lower projected water depth and similarly instrumented lower head.

PRELIMINARY RESPONSE TO ITEM 45

See attached sheet.

AP-600 submittal

One can provide a theoretical basis for heat fluxes in the range of 10.4 x 10^6 Btu/h-ft^2 (30 MW/m^2) for a system with co-dispersed debris and water as depicted in Figure ■■. A steam velocity sufficient to levitate and separate the water droplets from the high temperature dense debris is given by

$$U_g = \frac{3.7 \sqrt[4]{g\sigma(\rho_f - \rho_g)}}{\sqrt{\rho_g}}$$

where g is the acceleration of gravity, σ is the steam-water surface tension and ρ_f and ρ_g represent the saturated water and steam densities respectively. If this is considered to be the maximum steam production rate which could exist without separation of the water droplets from the co-disperse configuration, then the heat flux associated with the vapor production rate is given by

$$q/A = 3.7\ h_{fg} \sqrt{\rho_g} \sqrt[4]{g\sigma(\rho_f - \rho_g)}$$

where h_{fg} is the latent heat of vaporization. Substituting the appropriate values for steam and water at 1 atm into this expression results in a value of 10.4 x 10^6 Btu/h-ft^2 (30 MW/m^2); a value in agreement with those observed in the various experiments. Hence, the major ramification of an explosive interaction could be the co-dispersion of melt and water which then continues to transfer energy and vaporize water into the containment atmosphere at a rate limited by the ability of the water droplets to remain as part of the co-dispersed medium.

PRELIMINARY RESPONSE TO ITEM 48

This information can be found in DOE/ID Report "Technical Support for the Hydrogen Control Requirement for the EPRI Advanced PWR", pages 37 to 40 (attached).

2.2.4 Deflagration Analytical Basis

The above example can be generalized to find the maximum post-combustion pressure for an ALWR containment. As steam is added to an atmosphere of air and hydrogen, the initial pressure and final post-combustion pressure both increase. However, eventually, the mixture becomes inert due to steam addition when the flammability limit is reached. An approximate method is derived in Appendix A, which can be used to determine final pressures resulting from complete combustion of 13% H_2 on a dry basis at various system steam pressures. A calculation using steam table values is presented here. The flammability limit of diagram of Figure 2-2 can be used to determine whether these mixtures are flammable. The maximum final pressure for a 13% dry basis H_2 plus steam mixture which is possible in a containment under initially saturated conditions is defined for the flammability limit. Containment conditions will not be superheated because such a condition is only possible when dry core debris exists in the containment, a situation precluded by the EPRI ALWR debris coolability requirement (see Section 3.1).

The results of this thermodynamic equilibrium calculation for possible containment atmospheres are shown in Table 2-5 and Figure 2-9. The final pressure increases as the initial temperature increases because the initial pressure must increase due to steam addition, but the pressure ratio decreases and is seen to be highest for the dry case (300°K). The maximum theoretical pressure in a containment following a burn is 6.6 atm based on the flammability limits. However, also shown in Figure 2-9 is the anticipated boundary between the complete and incomplete combustion regimes. The maximum probable pressure is thus about 6.4 atm. When more steam is present, incomplete combustion would be expected and the final pressure would be less than this value.

Table 2-5

PRESSURE RISE AND FLAMMABILITY RESULTS FOR

13% H_2 IN DRY AIR WITH SATURATED STEAM ADDITION

T(°K)	X_{HW}[a]	X_S[b]	P_W[c]	P_f[d]	P_f/P_w	Flame[e]
300	0.127	0.026	1.2	5.8	4.8	Y
325	0.117	0.097	1.4	5.9	4.2	Y
350	0.099	0.240	1.8	6.0	3.3	Y
375	0.074	0.430	2.5	6.6*	2.6*	Y
380	0.069	0.470	2.8	-	-	N

[a] wet H_2 mole fraction

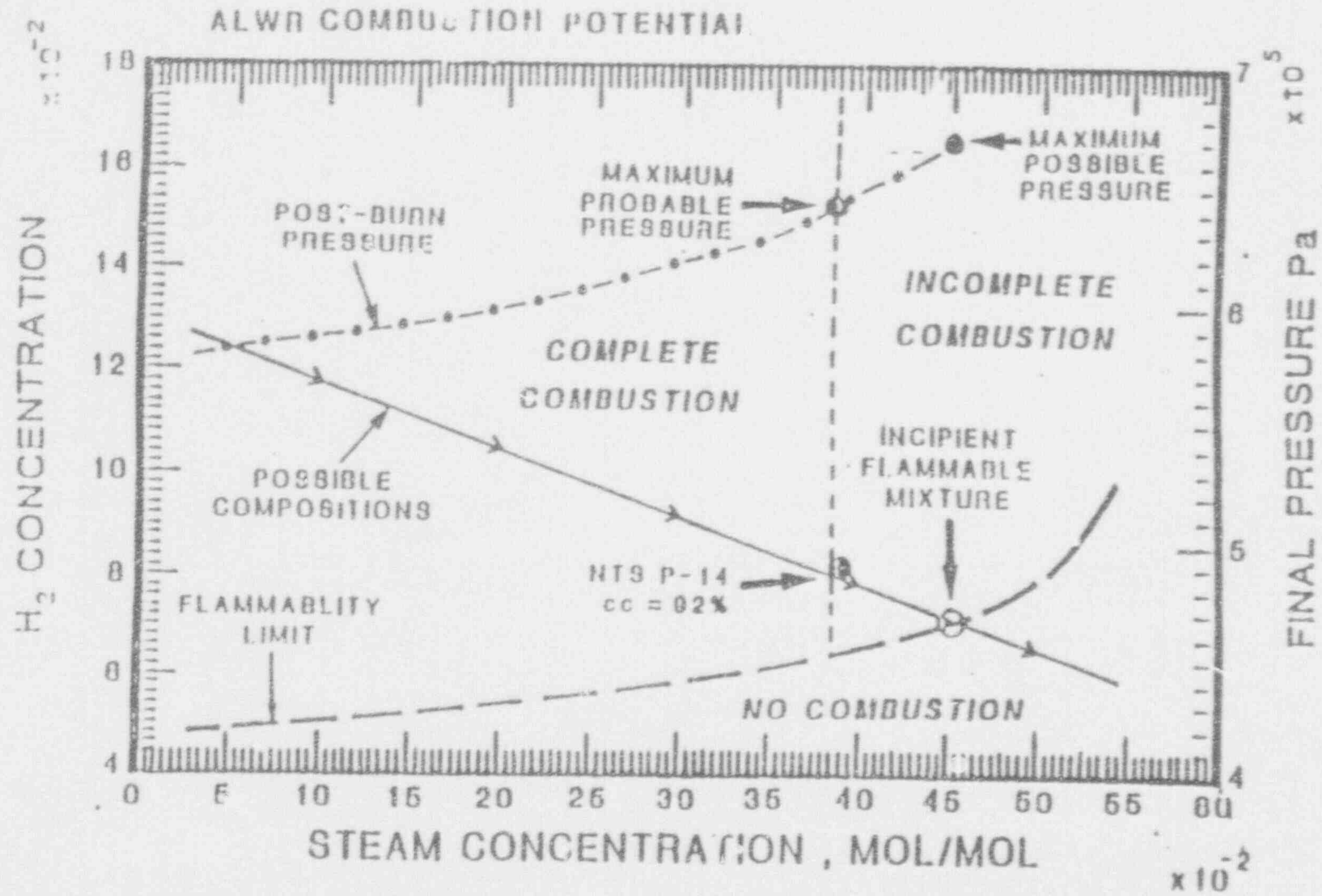
[b] H_2O mole fraction

[c] initial pressure (atmosphere)

[d] final (postburn) pressure (atmosphere)

[e] Y = flammable, N = not flammable

*overestimate due to incomplete combustion



MOP.880221 A.A

Figure 2-9. ALWR Combustion Potential.

PRELIMINARY RESPONSE TO ITEM 49

CALCULATION OF 140 PSIA BASED ON NUREG/CR-5567 PEAK PRESSURE CALCULATION SCALED TO A 100% BURN (PAGE 38)

$$Pb = 0.22 + (1.42/.75)\ Mzr/V$$

where Pb is pressure in Mpa gauge

For System 80+ Mzr=32653 kg and V=94650 m3

pb= 0.873 Mpa guage = 126.6 psig = 141.3 psia

Note: this estimate is generally conservative. More realistic AICC burns from sat. conditions will give pressures closer to 110 psia

PRELIMINARY RESPONSE TO ITEM 50

See Attached Figures form DOE/ID-

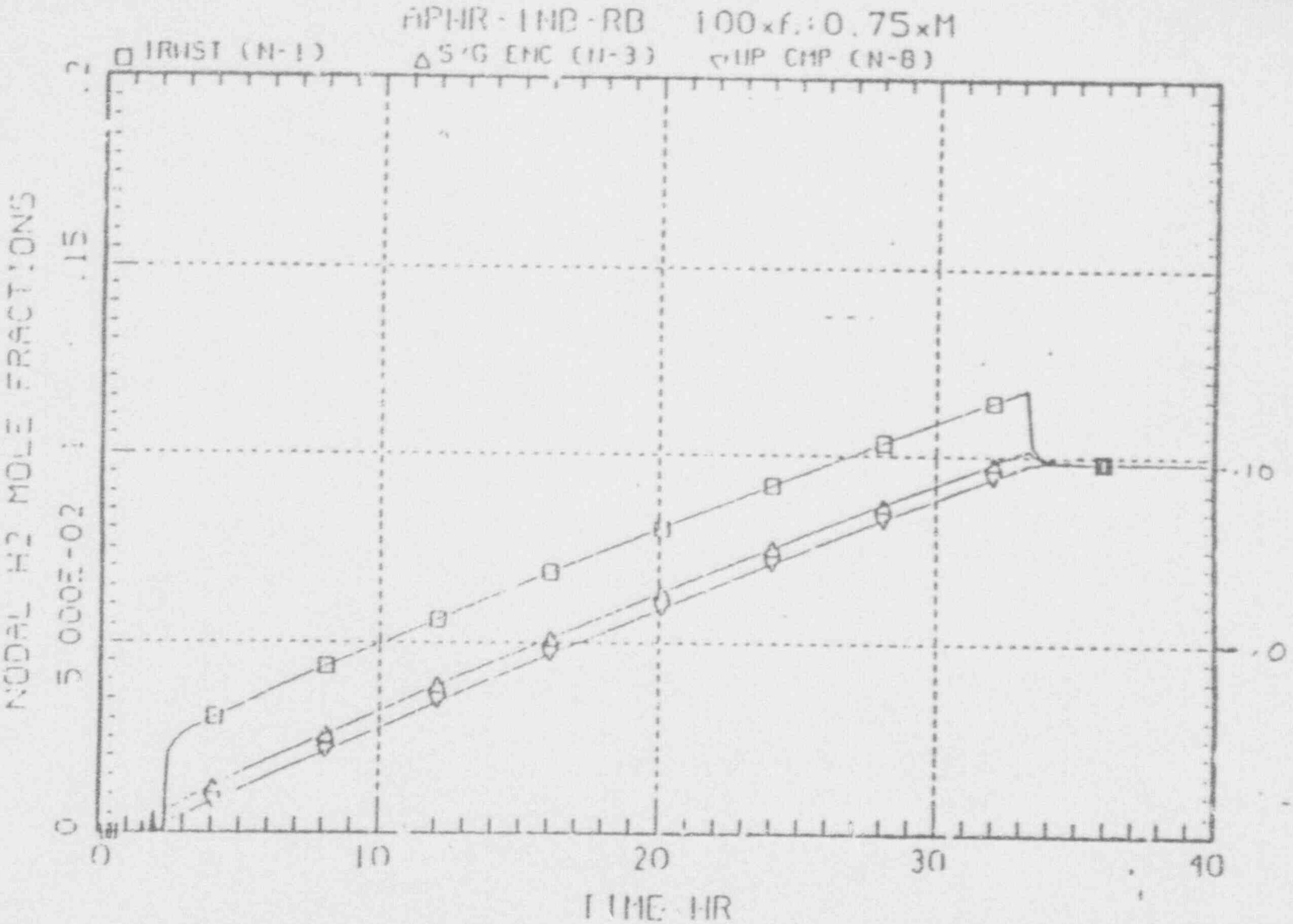
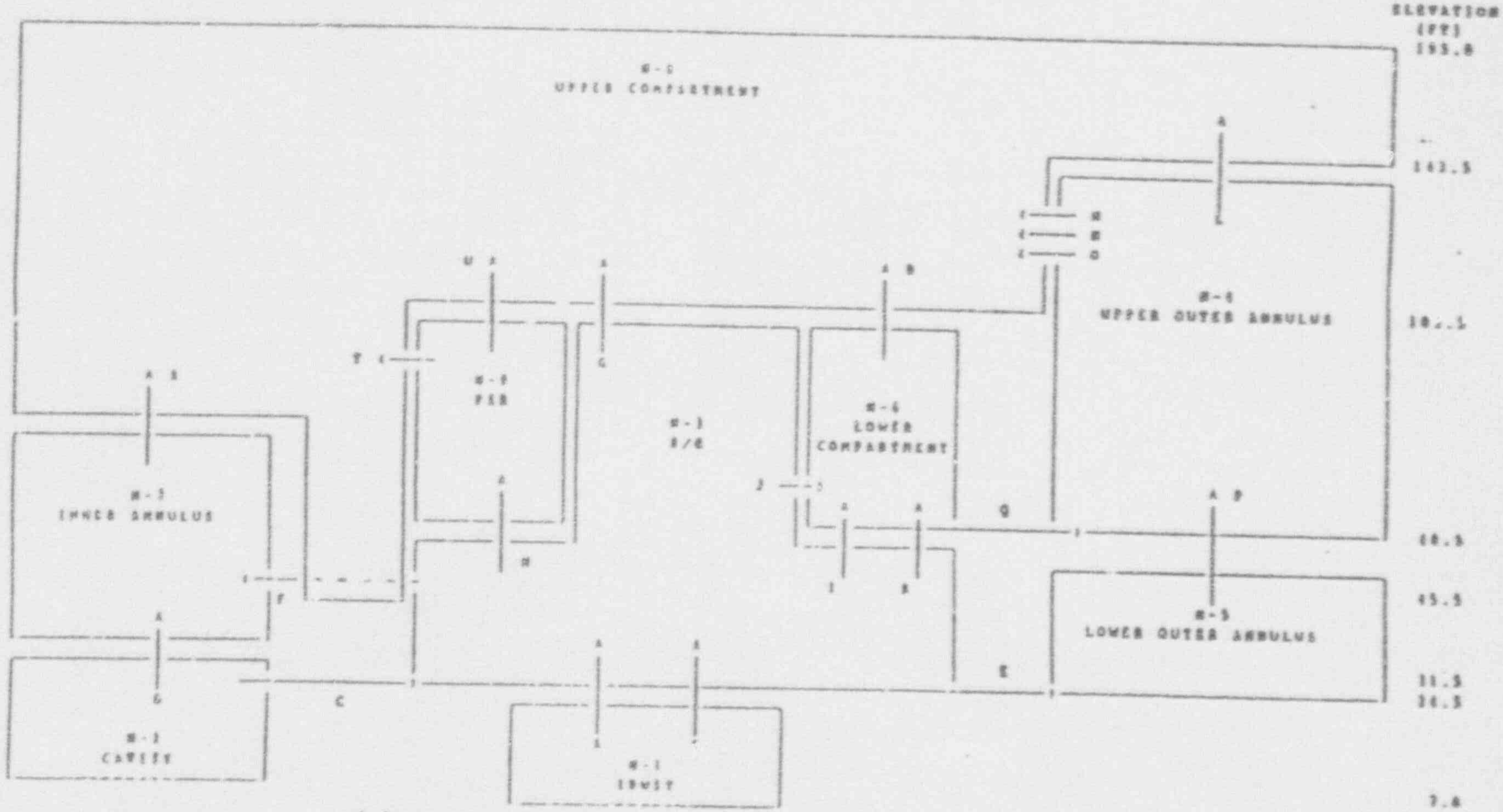


Figure 2-22 Hydrogen concentrations in the modified IRWST. (Steam generator enclosures and upper compartment during a high-pressure SBO with blockage and no burns.)



NOTE: MEANS FLOW PATH IS NORMALLY OPEN BETWEEN NODES

MEANS FLOW PATH IS NORMALLY CLOSED BETWEEN NODES

Figure 2-20 Nodalization of System 80+ Spherical Containment used for Hydrogen Mixing Studies.

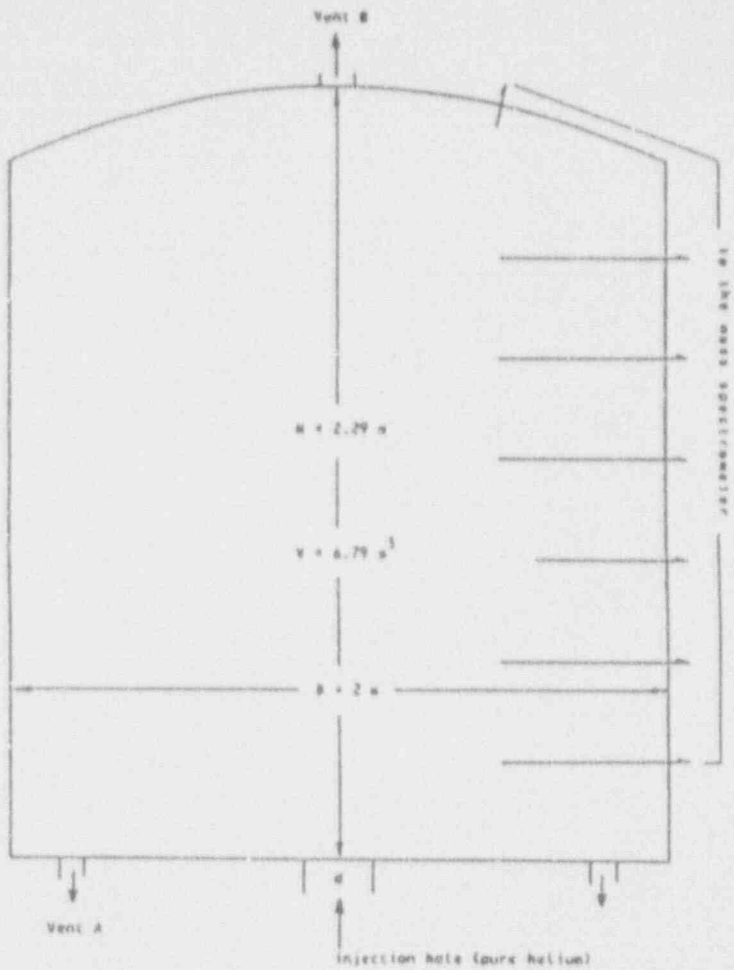


Fig. 1 Sketch of the facility

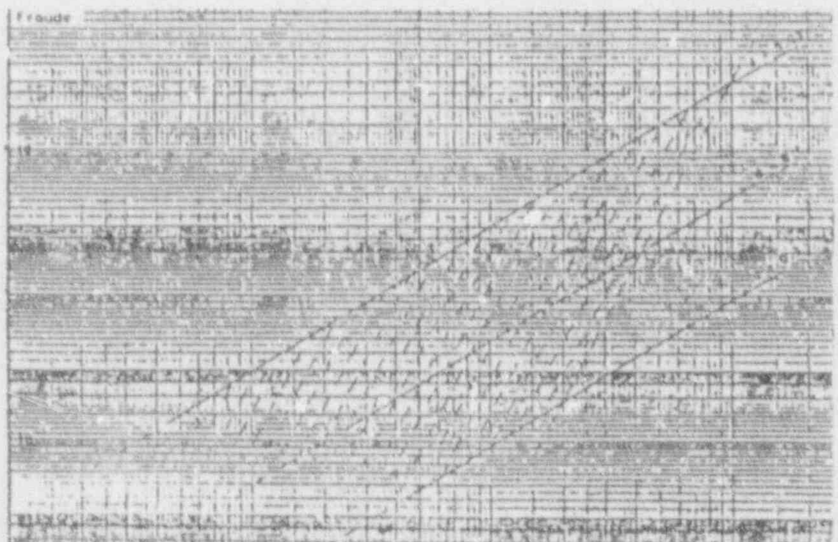
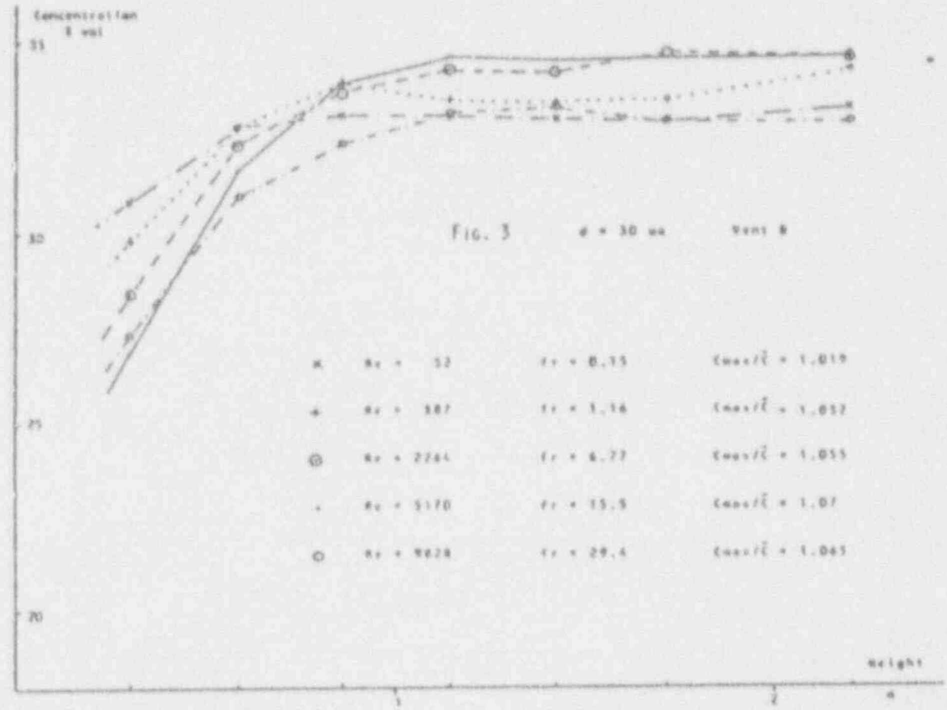


Fig. 2. Area in the Fr(Re) representation covered by the stratification tests


Concentration
% vol
35
30
25
20
Height
1
2
FIG. 3 d = 30 mm Vent B
Re = 52 Fr = 0.15 Cmax/C̄ = 1.019
Re = 587 Fr = 1.16 Cmax/C̄ = 1.052
Re = 2264 Fr = 6.77 Cmax/C̄ = 1.055
Re = 5170 Fr = 15.5 Cmax/C̄ = 1.07
Re = 9828 Fr = 29.4 Cmax/C̄ = 1.045

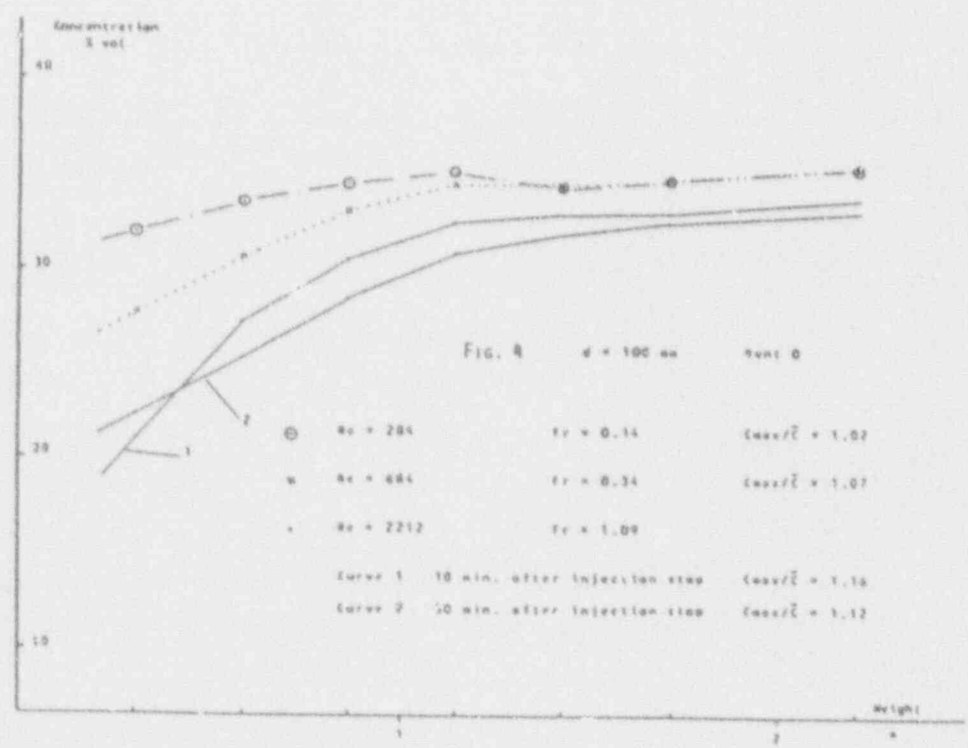

Concentration
% vol
40
30
20
10
Height
1
2
FIG. 4 d = 100 mm Vent B
Re = 284 Fr = 0.14 Cmax/C̄ = 1.02
Re = 684 Fr = 0.34 Cmax/C̄ = 1.07
Re = 2212 Fr = 1.09
Curve 1 10 min. after injection stop Cmax/C̄ = 1.16
Curve 2 30 min. after injection stop Cmax/C̄ = 1.12

Figure 5

Figure 6

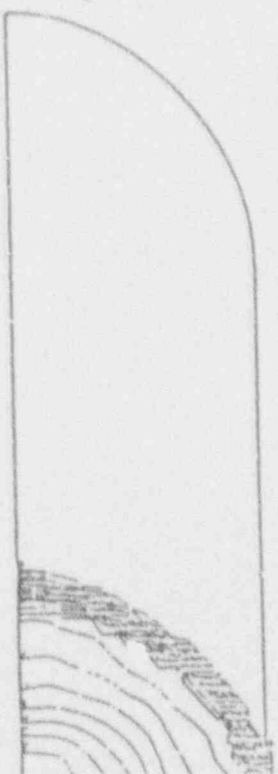


Figure 7

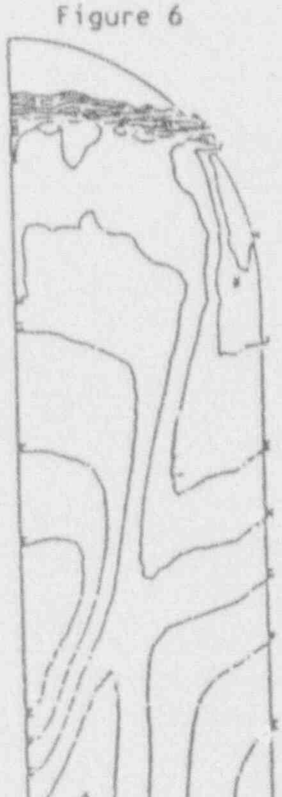


Figure 8

preliminary response to item 59

See response to question 37

REQUEST FOR ADDITIONAL INFORMATION: SYSTEM 80+

SEVERE ACCIDENT PHENOMENOLOGY
AND
CONTAINMENT PERFORMANCE

NOTE: it may be helpful for the preparer to know, that NRC will be using CONTAIN and MELCOR codes.

1. Please, provide sufficient information needed to perform multinode containment analysis:

- subcompartment volumes and elevations,
- inter-compartment connections: junctions flow areas, elevations, flow resistance coefficients (forward and reverse),
- heat structures data associated with each compartment: surface areas, thicknesses, liners and coating (if any), materials type and physical properties
- detailed description of the cavity

2. Please, provide the following:

- nominal core power,
- decay heat curve, if different from standard PWR ANS,
- initial fission product inventory (primarily Cs, I, Te, La, Sr, Ru and Ce), for a given burnup (e.g. end of cycle, or 2/3 of the cycle)
- if possible, decay heat curve for CsI,
- total masses of the fuel, zirconium and steel in the core region (i.e. steel potentially melted during core degradation)

3. Please, describe the methodology used in the severe accident evaluation including:

- containment event tree with appropriate split fractions,
- list of the risk dominant accidents

4. For the risk-dominant (beyond DBA) accident(s) please provide the following:

- mass end energy release rates to the containment,
- fission product release rates to the containment
- hydrogen generation rate and total mass generated,
- melted core fractions,
- timing of melt start, vessel failure
- assumed core melt temperature
- rate of core melt injection to the cavity
- composition of the core melt
- containment pressures and temperatures
- assumed or calculated heat fluxes ("up" and "down") during core-concrete interaction
- rate of ablation (radial and axial)