PMSTPCOL NPEmails

From:	John Lai
Sent:	Thursday, April 03, 2008 8:46 AM
To:	George Wunder
Cc:	Glenn Kelly; Hossein Hamzehee
Subject:	FW: Information from ABWR SSAR
Attachments:	20_03_09.pdf

George,

Do we still have all the old RAI responses from the ABWR SSAR review? The COL applicant (NRG-STP) would like to have a copy of the CAFTA input deck submitted in a 5 and 1/4 inch diskette (see response 725.44) if NRC still has it.

Thanks for the help.

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From: Stillwell, Daniel [mailto:dwstillwell@STPEGS.COM]
Sent: Thursday, April 03, 2008 7:43 AM
To: John Lai
Cc: Gibson, Gregory T
Subject: Information from ABWR SSAR

John,

The attached file contains GE responses to the Ninth set of RAIs on the ABWR SSAR. Questions 725.43, 725.44, and 725.45 reference the transmittal of CAFTA and MAAP input files to the NRC. This is all of the additional information I have right now. I'm not sure of the dates of the RAI or the responses, although it was probably in the mid 90s.

Thanks for your help.

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Hearing Identifier: SouthTexas34NonPublic_EX
Email Number: 161

Mail Envelope Properties (1FA53ADF29758448974A8AC1118E627E2946F0B0D7)

Subject: FW: Information from ABWR SSAR
Sent Date: 4/3/2008 8:46:07 AM
Received Date: 4/3/2008 8:46:08 AM
From: John Lai

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Files	**Size**	**Date & Time**
MESSAGE	1113	4/3/2008 8:46:08 AM
20_03_09.pdf	285285	

Options
Priority: Standard
Return Notification: Yes
Reply Requested: Yes
Sensitivity: Normal
Expiration Date:
Recipients Received:

20.3.9 Response to Ninth RAI — Reference 9

Question 725.1

In most of the currently available BWR PRAs, the loss of offsite power sequence with successful recovery of offsite power within 30 minutes (i.e., TM sequence in Fig. 19D.4-4) is transferred to the MSIV closure (i.e., isolation events) event tree. Please provide the basis for transferring it to the reactor shutdown tree (i.e., Fig. 19D.4-1) instead.

Response 725.1

If offsite power is recovered within one half hour, no core damage will have occurred. Following the recovery of offsite power, all safety systems become available, and the result is very similar to a reactor shutdown event. Therefore this outcome is transferred to Figure 19D.4-1.

Question 725.2

Should not the event tree top event, Q (Feedwater), appearing in the reactor shutdown event tree (Fig. 19D.4-1) be replaced by “Feedwater and PCS”? Otherwise, a branch should be added to the uppermost sequence (with an end state of OK) to determine the success or failure of the top event, W. Note that condenser problems (hardware or others) can lead to a manual shutdown.

Response 725.2

Yes, the value Q represents feedwater and PCS - also in Figures 19D.4-2, 19D.4-3, and 19D.4-11.

Question 725.3

Please provide the basis of not crediting automatic depressurization for the safety function, X, in the reactor shutdown event tree (Fig. 19D.4-1).

Response 725.3

There is no automatic signal for this sequence since a high drywell pressure signal is required but not present.

Question 725.4

Does ABWR have a design feature which allows the reactor operator to utilize RCIC in steam condensing mode to transfer reactor decay heat to the ultimate heat sink? If yes, why is no credit given to such a feature in evaluating the safety function W (containment heat removal)?

Response 725.4

The ABWR RHR system does not have a steam condensing mode. Earlier BWR RHR designs had this mode, which utilized the RCIC steam supply line to remove steam from the vessel and condense it with the RHR heat exchangers. Field experience with this feature has not been entirely satisfactory, and it was not included in the ABWR design. Furthermore, the ABWR has an additional high pressure inventory makeup system compared to previous BWR ECCS networks. This improved high pressure makeup reliability further supported elimination of the steam condensing mode of the RHR. The net result has been a simplification of the overall RHR system.

Question 725.5

In essentially all of the event trees shown in Fig. 19D.4-1 through Fig. 19D.4-14, failure of the W function (long term heat removal) is assigned a probability of failing to run RHRA or RHRB or RHRC rather than failing to start and run RHRA or RHRB or RHRC, if the preceding V function (RHR injection or condenser) is a success. This would be correct if one of the RHR pumps was successfully started and run to accomplish the mission of the V function, and then switched to a long term heat removal mode. Notice, however that the success of the V function can also be achieved, as indicated in Table 19.3-2, by using one condenser pump and one condenser transfer pump. In such a case, the approach taken in the ABWR FSR will underestimate the failure probability of W since the RHR pump has to be started and then run throughout the mission time. Also, can one RHR pump alone always accomplish the missions of both the V and the W functions for all the transients including a large LOCA?

Response 725.5

All three pumps are automatically started upon the receipt of either a low vessel level or high drywell pressure signal and can be transferred to other operating modes while running. Assessment of the ABWR modeling simplification showed it to underestimate the failure of W by a negligible amount which was considered insufficient to warrant added model complexity.

To determine that 1 RHR pump in the low pressure flooding mode is sufficient to perform the V function (i.e. core cooling) for any LOCA it was necessary to consider the complete spectrum of possible breaks. For the small and medium breaks, it was assumed that 3 ADS valves were also available to depressurize the vessel. For the large breaks, it was assumed that only the RHR pump is available since sufficient vessel depressurization occurs through the break. LOCA analyses for the success criteria were based on best-estimate predictions using the GE licensing approved computer models.

For small and medium breaks, it was determined that the maximum vessel bottom head drainline break (0.00202 m^2) was the most limiting case. This is the lowest possible break location on the vessel resulting in a high break flow throughout the transient. The LOCA analysis for this case resulted in a calculated peak cladding temperature (PCT) of 966°C which is well below the 1204°C limit. Important variables from this analysis are shown in Figures 20.3.9-1 through 20.3.9-7.

For large breaks, it was determined that a 0.0279 m^2 break in an RHR vessel shutdown suction line was the most limiting case. This is the lowest possible break location on the vessel for large breaks which again, results in a high break flow. The results of this break are worse than breaks with larger break areas because the RHR injection is delayed due to the relatively slow vessel depressurization through the smaller break. The LOCA analysis for this case resulted in a calculated PCT of 591.1°C. Important variables from this analysis are shown in Figures 20.3.9-8 through 20.3.9-14.

Based on the results of the above study, it was concluded that 1 RHR pump provides sufficient core cooling for large breaks, and also for small and medium breaks with the equivalent of at least 3 ADS valves available.

To determine that 1 RHR system is sufficient to perform the W function (i.e., long term heat removal), an evaluation study was performed to determine bounding containment pressure and temperature response during the long-term decay heat removal period. Among various potential transient and LOCA events, liquid breaks are expected to be more bounding, since significant pool drawdowns could occur in liquid breaks. Pool drawdowns substantially reduce the pool heat sink capacity for long term cooling, since pool drawdowns result in a substantially reduced pool volume available for long term containment cooling. An instantaneous guillotine rupture of a feedwater line (large liquid break) was chosen for the long-term heat removal analysis. Containment response analysis for this feedwater line break case was performed using the GE licensing approved computer models.

The initial conditions and analysis assumptions were consistent with those used for earlier analyses (defined under Subsection 6.2.1.1.3.3 of the SSAR). With regard to the availability of ECC systems, only one RHR system was assumed to be available for the long-term containment cooling. The RHR has four modes of operation and heat is removed from the containment in each of these modes. During the core cooling mode which is initiated automatically, the RHR heat exchanger is in the loop and the containment heat removal is established.

The containment pressure and temperature responses for a feedwater line break are shown in Figures 20.3.9-17 and 20.3.9-18, respectively. The calculated maximum pressure and temperature values are about 0.366 MPaA and 158.9°C, respectively. These maximum values are substantially lower than the ultimate pressure capability of the containment structure (0.62 MPaG at 371.1°C) defined in SSAR Appendix 19F.

Based on the results of the above analysis, it was concluded that 1 RHR system provides sufficient long-term heat removal capability to assure containment structure integrity.

Question 725.6

In both the non-isolation event tree (Fig. 19D.4-2) and the isolation / loss of feedwater event tree (Fig. 19D.4-3), the uppermost sequence (with an end state of OK) should branch out at the top event, W, since success of Q (feedwater alone) does not automatically warrant the success of W. The same comment also applies to the IORV event tree (Fig. 19D.4-11).

Response 725.6

The value Q represents feedwater and PCS. See also Response to Question 725.2.

Question 725.7

In Table 19D.4-1 through Table 19D.4-17, the branch point value of the safety function V (LPFLA or LPFLB or LPFLC available) was assigned a value of 1.27E-03, with the source of the data given as Table 19D.4-1. No such data, however, can be found in Table 19D.4-1. Also, for the loss of offsite power event trees, failure of V (LPFLA or LPFLB or LPFLC or one condensate and one condensate transfer pump) is given a value of
7.37E-03. Again, no such data can be found in the tables. Please explain how this value was calculated.

Response 725.7

The values for the safety function, V, are conditional probabilities derived from Tables 19D.4-1 and 19D.4-2, given failure of the high pressure injection functions. This notation needs to be made on the tables of input values for each event tree (Tables 19D.4-4 through 19D.4-18).

The value of V is determined from the values in the two tables as follows:

V = (ECCS / HPCC) (Condensate Unavail.)

As an example, for transients

V = (1.21E-06 / 9.52E-05)(1.0E-01)

= 1.27E-03

Question 725.8

For isolation / loss of feedwater events, successful RHR operation using the PCS requires reopening of the MSIVs and the recovery of feedwater if it is initially lost. In Fig. 19D.4-3, which event tree top event takes into consideration the reopening of MSIVs? Also, will the chance of reopening the MSIVs be smaller if there are stuck open SRVs?

Response 725.8

See Table 19D.4-6 (Symbol Q and Note 1) for treatment. No credit is taken for feedwater recovery.

The presence of a stuck open SRV will not reduce the possibility of reopening the MSIVs. The differential pressure across the MSIVs and a spring holds them closed. To open the MSIVs, the turbine stop valves must be closed and the MSIV drain / bypass lines are opened to pressurize the steam line downstream of the MSIVs and thus reduce the differential pressure across the valves. The MSIVs can be opened once the differential pressure across the valves is reduced to the point where the valve pneumatic actuator pressure can overcome the differential pressure across the valves and the spring force holding them closed. A stuck-open SRV would tend to reduce vessel pressure and the differential pressure across the valve. Thus, in this case the operator would be able to open the MSIVs slightly sooner.

Question 725.9

In the loss of offsite power and station blackout event tree (Fig. 19D.4-4), the probability of failing all three diesel generators (7.99E-04) is used to sort out station blackout sequences (i.e., BE2, BE8, and BE0) from the loss of offsite power sequences (i.e., TE2, TE8, and TE0). Note, however, that "all DG not fail" could mean: (1) one DG is available, (2) two DGs are available, or (3) all three DGs are available. In Figs. 19D.4-5 and 19D.4-6, the unavailability of U_h (HPCF B or C with a probability of 1.58E-02) was computed based on the assumption that two diesel generators are available. If only one DG is available at the onset of loss of offsite power, this unavailability could become larger. It appears that some kind of weight-averaging should be applied to modify this value based on the probabilities of having either one or two DGs when the loss of offsite power occurs. Also, in Fig. 19D.4-4, the failure probability of opening SRVs following an ATWS event was taken to be 1.0E-06. For ATWS events, a large number (15) of SRVs need to be opened for pressure relief, and hence, the failure probability of opening the required number of SRVs can be expected to be larger.

Response 725.9

It is recognized that there are many combinations of diesel generator failures that play a role in the LOOP event tree, as follows:

(1) Failure of diesel generators one at a time

(2) Failure of diesel generators two at a time

(3) Failure of diesel generators three at a time

The last combination (3), identified as EDG in Figure 19D.4-4, represents the station blackout event.

All of the above diesel generator failure combinations are factored into the ABWR PRA analysis and properly accounted for in the event tree modeling. The purpose in modeling the failure of all three diesel generators separately is that the station blackout accident had been shown to be a dominant contributor to core damage frequency in past BWR PRAs. We therefore decided to include it as a separate branch (EDG) in Figure 19D.4-4 to highlight its importance. The LOOP event tree models in

Figures 19D.4-5, -6, and -7 reflect all remaining cutsets containing diesel generator failure to start probabilities. These trees are not developed with any precondition that a specific number of diesel generators are available. The only precondition, given loss of offsite power, is that all three diesel generators have not failed. Evaluated in sequence and collectively, Figures 19D.4-4 through 19D.4-10 properly account for all cutsets containing diesel generator failure probabilities.

The failure probability for the ATWS event was intended to be 1.0E-04 but entered as 1.0E-06. The event tree will be corrected. The impact of this correction on the overall core damage frequency is negligible.

NOTE: The unavailability for$U\eta$ of 1.58E-02 mentioned in above is the Amendment 4 value. The Amendment 8 value for $U\eta$ is 4.52 E-03.

Question 725.10

In all of the loss of offsite power event trees (Figs. 19D.4-5, 4-6, and 4-7), the failure probability of HPCF (U_h) is taken to be the same irrespective of the offsite power recovery time and regardless of whether there are stuck-open SRVs. Can the heating up of suppression pool for a prolonged period of time due to stuck-open SRVs adversely affect the availability of HPCF?

Response 725.10

In the loss of offsite power event trees of Figures 19D.4-5, 19D.4-6, and 19D.4-7, all success paths include success of the heat removal function, W.

Figure 19D.4-5 is based upon recovery of offsite power within two hours and Figure 19D.4-6 is based on recovery between two and eight hours. In each of these two trees, the failure probabilities for the heat removal function are based upon the availability of offsite power. For this assumption as well as for the use of a single failure probability for HPCF to be appropriate, regardless of whether or not there are stuck open relief valves, the HPCF pumps must be capable of operating with increasing pool temperatures due to decay heat and blowdown depressurization energy until the time that credit is taken for the W function. Therefore, a bounding analysis was performed to determine whether HPCF can remain operational for the resulting suppression pool temperature, assuming an eight-hour period with a stuck open relief valve before the heat removal function becomes available.

Evaluations indicate that the RHR and HPCF pumps can remain operational to 182°C in response to emergency core cooling and heat removal demands. Calculations were performed to determine suppression pool temperature rise from integrated decay heat for eight hours plus that due to stuck open relief valve blowdown depressurization energy. Results of these analyses indicate a temperature rise from an assumed initial value of 35.0°C to approximately 148.9°C from these two contributors over the eight-hour period. Therefore, use of the single HPCF failure probability is justified.

In Figure 19D.4-7, as in the previous two figures, all success paths include success of the heat removal function, W. In this tree, however, the offsite power recovery time is not bounded. Consequently, failure probabilities for W are taken from the loss of offsite power column of Table 19D.4-1 which is based upon the availability of power from diesel generators only. Therefore, with the heat removal function available, use of a single value for HPCF, regardless of SORV status, is appropriate.

Question 725.11

Please provide the basis of not considering stuck-open SRVs in the station blackout event tree (BE2, Fig. 19D.4-8).

Response 725.11

Sufficient steam is generated following transient initiation to drive the RCIC pump and provide adequate core cooling for at least two hours with a stuck open relief valve.

Question 725.12

In the same event tree cited above (Question 725.11), the failure probability of W (RHRA or RHRB or RHRC) is taken to be 5.19E-04, which does not correspond to that (1.59E-03) shown in Table 19D.4-1 for the case of loss of offsite power. Are the values shown in the column under the heading of “Loss of Offsite Power” in Table 19D.4-1 also applicable to station blackout? If not, please explain.

Response 725.12

The value for W is taken from the “All Transient and Small LOCA” column of Table 19D.4-1 since the heat removal function (to prevent containment failure) is not needed for many hours after the recovery of offsite power.

The values shown in the column under the heading of “Loss of Offsite Power” in Table 19D.4-1 are not applicable to station blackout, with the exception of the RCIC value of 4.01E-02 to start and run for 8 hours. RCIC is capable of starting and running for 8 hours with only battery power for control. All other ECCS systems require AC power for successful operation.

Question 725.13

In the station blackout event tree (BE8, Fig. 19D.4-9), why does the sequence with success of RCIC need to be branched out for testing the success of HPCF? According to the success criteria listed in Table 19.3-2, successful core cooling using a high pressure system can be achieved by using either RCIC or one train of HPCF for all transients including loss of offsite power. Furthermore, both HPCF and LPFL require AC power which, in this case, is not available for nearly eight hours. Please explain why both HPCF and LPFL are included as event tree top events.

Response 725.13

The branch probability is that RCIC will run for eight hours without electric power other than battery power for control. Following recovery of electric power (at times greater than eight hours), other ECCS systems must pick up the load to provide both additional core cooling and the heat removal function.

Either HPCF or LPFL can provide the necessary additional core cooling following the recovery of electric power.

Question 725.14

For IORV transients, there is no immediate automatic scram signal, and the operator may be required to manually scram the reactor and start the makeup system before the suppression pool temperature exceeds the heat capacity temperature limit. Please provide the basis of not including “timely manual scram” as an event tree top event in the IORV event tree (Fig. 19D.4-11).

Response 725.14

Manual scram of the reactor was conservatively ignored. If included, it would have negligible effect on calculated overall CDF. The event tree is based upon automatic scram initiation on high drywell pressure.

Question 725.15

Please explain why feedwater (Q) was not credited as a viable means of core cooling in the small LOCA event tree (Fig. 19D.4-12). Note that, according to the success criteria shown in Table 19.3-2, feedwater can be used to successfully cool the core in the event of a small steam LOCA.

Response 725.15

No distinction was made between steam and liquid LOCAs in developing the LOCA event trees. A conservative approach was taken, and success criteria were taken to be those defined for liquid LOCAs.

Question 725.16

Please explain why HPCF is given credit in the large LOCA event tree (Fig. 19D.4-14) despite the high degree of depressurization caused by the large LOCA.

Response 725.16

HPCF utilizes a motor-driven pump which is designed to operate over the entire range of possible vessel pressures.

Question 725.17

Please provide justification of not considering vapor suppression in the large LOCA event tree.

Response 725.17

Since ABWR is designed to withstand a large LOCA, structural failures which could lead to LOCA steam entering the pool air space and defeating the pressure suppression function were judged too unlikely to merit treatment in the event tree. An inadequate pool water level could result in loss of pressure suppression, but since the level is covered by technical specifications, this possible cause was not treated. The potential for stuck open vacuum breakers causing a loss of pressure suppression is discussed in Subsection 19E.2.4.6.

Question 725.18

In constructing the ATWS event tree (Fig. 19D.4-15), no distinction was made between ATWS events with MSIV closure (isolation) and those with bypass available (non-isolation), although the former is generally more severe and limiting. Please explain why the same branch point probabilities were used in quantifying the ATWS sequence frequencies despite differences in the success criteria, such as the time available for the operator to inhibit ADS or the unavailability of normal heat removal system for containment heat removal (see Table 19.3-3).

Response 725.18

This question is addressed in Subsection 19D.4.15. Success criteria for the isolation case were applied in a single event tree to conservatively assess ATWS. Since the frequency of ATWS is low, this simplified treatment does not significantly alter the results of the PRA.

Question 725.19

It appears that the low core damage frequency (9.1E-09/RY) found for ATWS sequences is mainly driven by the low initiating event frequency (9.34E-09/RY), which was obtained by taking scram failure probability, C, to be 1.0E-08. Please explain in detail how this scram failure probability was calculated. From the fault tree developed for a single control rod drive (Fig. 19D.6-17a, Figure 1), the probability of failure to insert an individual control rod can be estimated to be roughly 3.0E-06. No explanation, however is given as to how this probability is used to generate the probabilities of the basic events shown in the fault tree of control rod drive system (Fig. 19D.6-19a, Figure 1). Also, no probability data is given for the event RPS (RPS fails to initiate scram) appearing in the fault trees for reactivity control (Fig. 19D.6-16b).

Response 725.19

The probability estimate, C, of 1.0 E-08 for failure to successfully control reactivity with control rods was developed by evaluating gate RPSTOP of the functional fault tree for reactivity control presented in Figure 19D.6-16a, page 19D.6-202. The reactivity control functional fault tree and bases for assessing its contributing components are presented in Sections 19D.6.5.1 through 19D.6.5.5, pages 19D.6-5 and 19D.6-6.

As stated in Subsection 19D.6.5.3, development of the control rod drive system fault trees, reactivity control success criteria, and component failure rate estimates relied substantially on extensive earlier analyses performed to assess BWR scram system reliability. These analyses are documented in Reference 1 of Subsection 19D.6.6.

Because of the extremely high redundancy built into these systems, random failures are insignificant contributors to control rod insertion failure (FCR). Only common cause failures are expected to be significant contributors and therefore, the evaluation focuses on common cause failures only.

Based upon the prior analyses and the design features of the ABWR CRD mechanical system, four modes of common CRD System failure which might result in a failure to achieve hot shutdown were identified:

(1) Failure of more than three HCUs in one reactor quadrant,

(2) Failure of more than one of the three electrical division power supplies to the motors,

(3) Failure of five or more facially adjacent rods, and

(4) Failure of fifty-five or more randomly situated rods.

Probabilities for events (1) and (2) in the reactivity control functional tree were obtained by applying a common cause factor of 0.01 to the single unit failure probabilities for application in Figure 19D.6-19a, resulting in values of 4.0E-08 and 3.16E-06 for events (1) and (2), respectively. In order for either of these events to cause a failure to control reactivity, the corresponding motors or HCU backups would also have to fail. The failure probability resulting from this “and” operation is negligibly small.

The “Foreign Object in Drive” component (FOID) in Figure 19D.6-17a was judged to be the only basic event with credible potential for event (3), i.e., mechanical common cause failure of five or more adjacent control rods. A common cause factor of 0.01 was applied to the individual control rod value to obtain the failure probability for component FARFLOC in Figure 19D.6-19a. The 1.0E-09 probability for random failure of five or more adjacent control rods (FARFRAND) was taken directly from the Reference 1 of Subsection 19D.6.6, analyses and judged to be applicable to ABWR.

For event (4), the probability of failing fifty-five or more randomly situated rods was judged to be negligible. A value of <1.0E-09 was used for 55RANROD in Figure 19D.6-19a.

The basis for the RPS value is provided in Subsection 19D.6.5.2.

Question 725.20
In Table 19.3-3, the time available for the operator to initiate one train of SLC is given to be 10 minutes for both isolation and non-isolation ATWS events. Should not the time available for the former be shorter because the suppression pool is heated up sooner?

Response 725.20
Yes. The 10 minute time presented is the time available for operator action for the ATWS events with isolation. Without isolation, the heatup of the suppression pool is slower and there is more time for operator action. However, for simplicity, the 10 minute time is used as a bounding value for both cases.

Question 725.21
For an ATWS event which is initiated or accompanied by closure of all MSIVs or loss of condenser, can adequate core coolant inventory be maintained by RCIC alone (as indicated in Table 19.3-3)? For some BWRs of current design, such an event requires HPCI or a combination of HPCI and RCIC.

Response 725.21
Analyses of this sequence using TRAC have been performed by GE and by INEL which indicate that severe fuel damage will not occur in an ATWS event where no boron is injected with RCIC providing the only core cooling. The INEL analysis is documented in DOE/ID-10211 “Analysis of a High Pressure ATWS with Very Low Make-up Flow”, dated October 1988. Both analyses showed that the vessel reached an equilibrium condition with the water level below the top of active fuel, but adequate steam cooling in the core to prevent fuel damage. The peak clad temperature (PCT) in the INEL study was approximately 825°K. The calculated PCT in the GE study was somewhat lower than that calculated by INEL.

Question 725.22
In quantifying ATWS sequence frequencies, the same branch-point value was used for W (containment heat removal) regardless of whether there are stuck open SRVs. Was suppression pool heating due to stuck-open SRVs taken into account in estimating the failure probability of W?

Response 725.22
In an ATWS event, if boron is successfully injected, the success criteria are the same whether or not an IORV occurred. If boron is not injected in time, it is conservatively assumed that W does not matter and the events lead to containment failure and core damage.

Question 725.23
Is there any reason why the event tree top event “ADS Inhibit” in the ATWS event tree is placed before “Feedwater or HPCF” and “RCIC” although it appears more logically correct to place it after the latter top events?

Response 725.23
Success criteria for the ATWS event following isolation require ADS inhibit within five minutes following initiation of the event. It is therefore a critical action which must be immediately accomplished, regardless of whether injection success has been verified by this time.

Question 725.24
Was any functional event tree or fault tree developed to analyze the unavailability of feedwater, condensate, and condenser system? How was the unavailability of feedwater (Q), for example, evaluated for different transient initiators?

Response 725.24
No functional trees were developed for feedwater, condensate, and condenser system.

The basis for the unavailability of feedwater for each transient initiator is specified on the branch point value table accompanying each event tree, and in Table 19D.4-2.

Question 725.25
In the event tree quantifications, the frequency of a particular accident sequence was obtained by multiplying together the initiating event frequency and the branch point probabilities of the failed safety functions, such as U, V, or W, appearing in the sequence description. This approach is proper if the branch point probabilities were evaluated by properly accounting for the common-mode failures among the event tree top events by linking together the relevant fault trees. Were these fault tree linkings done in the ABWR analyses to obtain the upper-bound of minimal cut sets for safety function failures such as UV, QUV, or UVW? If not, please explain how the branch-point probabilities were calculated for the individual safety functions such as U, V, or W.

Response 725.25
Functional fault trees were developed and evaluated to properly account for commonalities between systems. For example, the core cooling functional fault tree combined all injection functions as well as support systems (electrical, service water, and instrumentation) to properly account for commonalities and system dependencies.

Question 725.26
Were all the system failure probabilities (except for RCIC) listed in Table 19D.4-1 obtained by quantifying the fault trees shown in Section 19D.6? Were the probabilities of failing all ECCS systems computed by linking the high pressure and low pressure system fault trees? If so, which mode of the low pressure system was used? Also, were these values actually used in the event tree quantifications?

Response 725.26
All of the system failure probabilities (including RCIC in Amendment 8) in Table\ 19D.4-1 were obtained by quantifying the fault trees shown in Section 19D.6.

The probabilities of failing all ECCS systems were computed by linking the high pressure, low pressure, and support system fault trees and evaluating them concurrently in single functional fault trees.

The LPFL mode of the low pressure system was used in the ECCS functional fault trees.

Conditional failure probabilities calculated from the functional fault tree evaluations were actually used in the event tree quantifications.

Question 725.27

Were the fault trees for the support systems, such as electric power system, service water system and instrumentation system individually quantified? Are the results of such fault tree quantifications (in terms of minimal cut sets) available for comparison with BNL calculations?

Response 725.27

Yes, the support system fault trees were individually quantified. These results are reflected in the front line system and functional fault tree results, but were not separately recorded. These minimal cut sets can be obtained by evaluating sub-trees of interest from the functional fault tree CAFTA files provided in response to the Question 725.44 information request.

Question 725.28

What modifications to the fault tree input data were made to obtain the system failure probabilities corresponding to loss of offsite power (last column of Table 19D.4-1)? Was the failure probability of switchgear taken into consideration when the failure probability of the W function (for example, in Figure 19D.4-7) was calculated?

Response 725.28

The functional fault trees were modified by setting the loss of both the normal preferred and alternate offsite power sources (ELOOP1 and ELOOP2 in the electric power trees) to be “True” in the fault tree program for these evaluations. Station blackout cutsets were subtracted from each of these evaluations to obtain the values in the last column of Table 19D.4-1.

The failure of switchgear was taken into account when the failure probability of the W function was calculated.

Question 725.29

Please briefly describe the possible impacts of omitting the development of system fault tree for plant air system on the frontline and the support systems.

Response 725.29

The plant air system is not safety grade, reflecting that its performance is not expected to adversely affect the functioning of ECCS. Safety grade air and nitrogen accumulators are explicitly treated in the ADS and CRD fault trees.

In the short term, loss of plant air will result in MSIV closure and scram, a scenario which is addressed in the isolation/loss of feedwater event tree of Figure 19D.4-3. The loss of plant air is considered to be subsumed in the initiation frequency of this event.

Long term SRV operability is addressed in Subsection 19E.2.1.2.2.2.

Question 725.30

It was noted that a very small fraction of the failure data shown in Table 19D.6-2 through 19D.6-7 are inconsistent with those shown in the relevant fault trees (for example, DIV2MUX, HMV14BHW, and HXV032CQ in Table 19D.6-2). Which values were actually used in the fault tree quantifications?

Response 725.30

All values actually used in the fault tree quantifications are provided in the basic event data files furnished in response to the Question 725.44 information request.

Question 725.31

The break areas for the various LOCAs (large, medium, and small) are defined to be significantly larger than those used in, for example, the Limerick PRA. Do the initiating event frequencies used in the event tree quantification reflect these changes in the definition of break sizes?

Response 725.31

Medium and large LOCA break areas defined for ABWR, while larger than those used in the Limerick PRA, are smaller than those used for BWR/6 in GESSAR. ABWR has fewer pipes whose failure can lead to a LOCA than previous BWR designs. For example, recirc pipes have been eliminated. Consequently, ABWR LOCA initiation frequencies are judged to be smaller than those for earlier BWR plant designs, and use of the GESSAR LOCA frequencies represents a conservative application.

Question 725.32

How does the RWCU (reactor water cleanup) system work to remove decay heat? What suction lines are used? What is the heat sink? Does the non-generative heat exchanger have enough capacity to remove decay heat?

Response 725.32

Figure 20.3.9-15 is a schematic of the reactor water cleanup (RWCU) system. During normal operation, the RWCU takes suction from both the vessel bottom head drain line and the residual heat removal system B suction line. The flow passes through the regenerative heat exchanger (RHX) and the non-regenerative heat exchangers (NRHXs). It is then pumped through the filter-demineralizers, through the shell side of the RHX and returned to the vessel via the feedwater lines.

Calculations for the success criteria assume that the RWCU heat removal capability is maximized by having the return flow from the filter-demineralizers bypass the RHX. This is accomplished by closing valve F012 and opening valve F013 (refer to Figure 20.3.9-15). In this configuration, decay heat is transferred by the NRHXs from the vessel to the reactor building cooling water system and then to the reactor service water system which is the ultimate heat sink. The success criteria calculations for this case show that if the reactor system is maintained at high pressure, the RWCU can remove all the decay heat generated in the core at any time greater than 4 hours after scram.

Question 725.33

For RHR shutdown cooling mode, suction is taken from RPV. Where are the points of suction for the three suction lines? Also, where are the discharge points for the core cooling subsystem return lines?

Response 725.33

In the shutdown cooling mode, all three RHR systems take suction from the vessel (Figure 3.9-1) through the RHR shutdown lines at the locations specified in Table 20.3.9-1. In both the shutdown cooling and low pressure flooding mode, RHR-B and RHR-C inject directly into the vessel (see Figure 3.9-1) via their own separate distribution spargers. In these modes RHR-A injects into the feedwater line “A” which in turn injects into three segments of the feedwater sparger. This information is summarized in Table 20.3.9-1.

Question 725.34

Questions on Table 19D.4-1.

(1) What modifications were made to the fault trees to obtain the failure probabilities corresponding to large or medium LOCAs?

(2) Are the RCIC failure probabilities calculated by quantifying the revised fault trees in Amendment 8?

(3) What are the failure probabilities corresponding to station blackout?

Response 725.34

(1) ■RCIC was removed from both LOCA functional fault trees since it is not a success path

■ The medium LOCA will initiate ADS automatically and is so modeled.

- ADS was removed from the large LOCA fault tree since it is not required.
- Service Water System gates WILA(B,C) (Figure 19D.6-14g) are included only in the large and medium LOCA fault trees.

(2) RCIC failure probabilities documented in Table 19D.4-1 were calculated by quantifying the revised fault trees in Amendment 8.

(3) The only system failure probability corresponding to station blackout is the RCIC value of 4.01E-02 for failure to start and run for 8 hours under the "Loss of Offsite Power" heading. RCIC is capable of starting and running for 8 hours with only battery power for control. All other systems require AC power for successful operation, and their appearance in any SBO event tree success paths is predicated upon prior recovery of AC power.

Question 725.35

What modifications were made to the fault trees to obtain the core damage frequency corresponding to incorporation of (a) gas turbine generator; and (b) fire system water connection?

Response 725.35

To develop the estimate of gas turbine generator impact, fault trees were modified by adding the gas turbine, as a component, to "and" gates EAC69SC(D,E). These gates represent the power sources to each of the three 6.9 kv divisional buses.

Fault trees were not modified to incorporate a fire water system connection. The estimate was made by applying the assumptions described in Subsection 19.3.1.5.2 to the base case results presented in Table 19.3-4 and detailed in Section 19D.4

Question 725.36

Following loss of offsite power, feedwater pumps (motor driven) are tripped and MSIVs are likely to be closed. Are the FW pumps or the RWCU pumps connected to DG power source? Is re-opening of MSIVs considered in calculating the probability of NHR for the W function? In other PRAs, feedwater is considered unavailable following LOOP.

Response 725.36

Neither the FW pumps nor the RWCU pumps are connected to a DG power source.

Re-opening of the MSIVs is considered in calculating the probability of NHR for the W function.

Question 725.37

Class II sequence frequency was calculated to be 4.29E-06. The input to the Class II containment event tree, however, is 2.5E-06. Please explain the difference. Was the CDF for Class II sequences (4.29E-10) obtained by taking 0.01% of 4.29E-06?

Response 725.37

The Class II sequence frequency of 4.29E-06 is that documented for the base case in Table 19D.4-3. Containment event tree initiating frequencies are based upon the inclusion of a gas turbine generator (See Table19.3-6.) The 2.5E-06 value input to the Class II containment event tree reflects the estimated impact on the base case of incorporating a gas turbine generator.

Yes, the CDF for Class II sequences (4.29E-10) was obtained by taking 0.01% of 4.29E-06. The 0.01% value is documented in Subsection 19D.5.12.2.

Question 725.38

ATWS transient scenarios vary significantly depending on whether MSIV are closed or whether offsite power is available. How can a single ATWS event tree properly handle all ATWS events of different initiators?

Response 725.38

This question is addressed in Subsection 19D.4.2.15. Success criteria for the isolation case were applied in a single event tree to conservatively assess ATWS. Since the frequency of ATWS is low, this simplified treatment does not significantly alter the results of the PRA.

Question 725.39

In the ATWS event tree, failure to initiate SLCS is given a probability of 0.2 (time available for the operator = 10 min.) A typical value used for this action in most other BWR PRAs is 0.87 (with time available for the operator = 8 min.). Please explain the difference.

Response 725.39

There are substantial differences in the values presented in various PRAs to represent human failure to successfully respond to postulated ATWS events. Bases for the 0.87 value cited above are discussed in Section 3.4.3.3 of the Shoreham PRA (Report SAI-372-83-PA-01, June 24,1983). It references, among others, Swain and Guttman (NUREG/CR-1278) and WASH-1400.

It is our judgement that these are very conservative values. With advances in human error probability evaluation techniques, recent PRAs estimate significantly lower probabilities for these operator actions.

The two human failure probabilities in Figure 19D.4-15 (initiate SLC and inhibit ADS) were evaluated using the Operator Action Tree/Time Reliability Correlation approach described in NUREG/CR-3010 (BNL-NUREG-51601), dated November 1982. High dependence was assumed between the two required actions, as well as an initial reluctance to act. The assumed time line for required human response was as follows:

(1) Five minutes to observe and diagnose the need for SLC and ADS inhibit.

(2) Requirement to inhibit ADS without additional delay.

(3) Five additional minutes to manually initiate SLCS.

Strong dependence between the two events was treated according to the methods of Swain and Guttmann (NUREG/CR-1278, Chapter 10).

The above treatment yielded failure probabilities of 0.2 and 0.1 for SLCS initiation and ADS inhibit, respectively. These values are conservative relative to the most recent Grand Gulf and Peach Bottom PRAs (NUREG/CR-4550, Vol. 6, Rev. 1, September 1989 and NUREG/CR-4550, VOL. 4, Rev. 1, August 1989, respectively) conducted as part of the NUREG-1150 effort for the Nuclear Regulatory Commission.

Question 725.40

In the ATWS event tree, the probability of failing to inhibit ADS is taken to be 0.1. A typical value used in other PRAs is 0.5 if high pressure core injection is a failure, and 0.005 if HPCI is a success. To be able to make such a distinction, the order of the event tree top events for “HPCI” and “failure to inhibit ADS” must be interchanged.

Response 725.40

There are substantial differences in treatment as well as the values presented in various PRAs to represent human failure to successfully respond to postulated ATWS events. The treatment and values cited in the above question are those discussed and applied in the Shoreham PRA (Report SAI-372-83-PA-01, June 24, 1983). It is our judgement that these are very conservative values. With advances in human error probability evaluation techniques, recent PRAs estimate significantly lower probabilities for these operator actions.

The distinction mentioned in the question was not made when developing the ABWR ATWS event tree and determining the probability of failing to inhibit ADS. The ABWR ATWS event tree structure is essentially equivalent to other BWR PRAs, including the most recent Grand Gulf PRA (NUREG/CR-4550, Vol. 6, Rev. 1, September 1989) conducted as part of the NUREG-1150 effort for the Nuclear Regulatory Commission.

The basis for the value used in the ABWR for failure to inhibit ADS of 0.1 is described in Question 725.39 regarding failure to manually initiate SLCS. These two actions were treated together, since there was considered to be strong dependence between them. The ABWR value for failure to inhibit ADS is conservative, relative to the most recent Grand Gulf and Peach Bottom PRA revisions.

Question 725.41

For loss of offsite power initiators, stuck open relief valves (SORVs) were considered in Amendment 4, but were eliminated in Amendment 8. Please explain why.

Response 725.41

The event developed in Figure 19D.4-5 is for loss of offsite power for less than 2 hours, and RCIC is adequate for core cooling for at least this length of time with or without an SORV. Therefore, the Amendment 4 lower branch of this figure becomes redundant and unnecessary within this time frame, and the event tree was simplified accordingly.

SORV considerations were not eliminated for loss of offsite power events of duration greater than two hours.

Question 725.42

For isolation / loss of FW events, the unavailability of feedwater is taken to be 0.43 (= 40%(1) + 60%(0.05)). Is not the value 0.05 too optimistic for the MSIV closure initiators?

Response 725.42

The basis for the 0.05 value is presented in Table 19D.4-2, and is judged to be a reasonable estimate.

Question 725.43

In order to expedite the staff's review, please provide a copy of the MAAP code and requisite input information that was used in the ABWR evaluation.

Response 725.43

The MAAP-ABWR code was provided on floppy disk. Also included on the disk were the parameter file, which contains the plant description input, and an input file which describes the LCLP-PF-D-M sequence. Finally, sample output for this problem was included to allow verification of correct installation of the code.

Question 725.44

Please provide a copy of the magnetic medium containing all system level fault trees and functional level fault trees modeled for the initiating events applicable to the ABWR.

Response 725.44

System and functional level fault trees and data files required for their evaluation are provided on the floppy disk labeled "ABWR FAULTREE". System level trees are imbedded in the functional trees, and can easily be extracted for individual evaluation using the CAFTA fault tree program. The following files are included:

Core Cooling Fault Trees

TRAN1989.CAF	Transients and small LOCAs
LLOC1989.CAF	Medium and large LOCAs
LOOP1989.CAF	Loss of offsite power
ECCS.BE	Basic event data files
ECCS.GT	
ECCS.TC	

Heat Removal Fault Trees (Suppression Pool Cooling)

SPTRS89.CAF	Transients and LOCAs
SPLOOP89.CAF	Loss of offsite power
RHRSP.BE	Basic event data files
RHRSP.GT	
RHRSP.TC	

Question 725.45
Please provide the input files for the MAAP calculations.

Response 725.45
The parameter file which contains the plant description input to MAAP has been provided on floppy disk as described in the response to Question 725.43.

Question 725.46
The probability of containment failure resulting from loss of heat removal is given as 3.4E-6 in Section 19.1.2. However, the frequency of containment structural failure resulting from loss of containment heat removal is given as 2.5E-7 per reactor year in Section 19D.5.12.4. Please clarify.

Response 725.46
In Subsection 19.1.2, the frequency of containment failure resulting from both the internal events and the seismic events is provided. The value in Subsection 19D.5.12.4 refers to results from internal events PRA only.

Question 725.47
Is the failure pressure of the upper drywell (UDW) head above 500 F independent of the UDW temperature? If it is a function of temperature, please provide the function. Please also provide the leak area for the high temperature failure. Is high temperature failure considered to be P (penetration) or D (drywell head) failure in the release mode from containment when binning the accident sequences?

Response 725.47
The failure pressure of the UDW head is a function of temperature as shown in Figure 19F.3-7. The leak area for high temperature failure is given in Subsection 19F.3.2.2. For calculational purposes a stairstep approximation to this curve was used. High temperature failure is considered a penetration failure followed by drywell head failure when the pressure reaches the failure pressure given in Figure 19F.3-7. In order to clarify this, the description for P in Subsection 19E.2.2 and in Table 19E.2-3 will be modified.

Question 725.48
What are the locations and sizes of the passive flooders? Please describe the melting process of the passive flooder fuse including the temperature distribution in the fuse. What is the reliability of these flooders? Are there any examples of their use in other industries?

Response 725.48
Further details about the location and size of the passive flooders is included in Subsection 9.5.12.

The passive flooder fuse is designed to open fully when the outer metal temperature of the valve reaches 260°C. The temperature distribution in the fuse is a detailed design item that has not been completed at this time. The flooders are expected to be highly reliable. Qualification tests will be performed before the final material selection to ensure that performance and reliability requirements are met.

Similar devices have been employed in fire protection. One example is the Thermal Control Valve (Model F430) made by the Grinnell Fire Protection Systems Company, Inc. which uses a fusible-type solder assembly to regulate flow.

Question 725.49
The CET for Class IV accidents was not developed because of negligibly low occurrence frequencies (Section 19D.5.11.1). However, CETs for accident classes with similar or lower frequencies (Classes IB-3 AND IIIA) were developed. Please explain.

Response 725.49
We agree that the frequencies for all the three classes are negligible and CETs are really not needed. The event tree structures for classes IB-3 and IIIA are identical to classes

IB-1 and IA, respectively. CETs for IB-1 and IA had already been developed, and it was decided that CETs for IB-3 and IIIA could be developed without much effort. No such CETs were available to develop the Class IV CETs.

Question 725.50a

With respect to Firewater Addition (FA), is it necessary to have a separate “FA” category for a mitigating feature? It appears that “FA” is included in “IV” (e.g., Figures 19E.2-6 describe a sequence SBRC-FA-D0. However, this sequence is binned as SBRC-IV-D0 in CET IB-2, Figure 19D.5-8). The CETs do not show any sequences with “FA”.

Response 725.50a

The “FA” mitigating feature is used to describe the use of the firewater system to prevent core damage when adequate time is available for the operator to use the system before core damage occurs. For internal events, this is assumed possible only for sequences in which an ECCS injection system was initially available. Sequences 1, 2 and 3 were incorrectly labeled IV in Figure 19D.5-8, the current amendment corrects this figure.

Question 725.50b

Withdrawn.

Question 725.50c

How is the firewater addition or spray handled in the CETs?. It appears that it is included sometimes in “ARV” (e.g., Seq. 3 of CET ID.2) and sometimes in “ARC” (e.g., Seq 6 of CET IA-1 (sic)). Would it not simplify and clarify the CETs if firewater is designated as a separate heading? Firewater spray appears to play a major role in reducing the release fractions by scrubbing in the case of containment failure. (A suppression pool loses its scrubbing function once the vessel fails). Therefore, it is important to know if firewater is available for a particular sequence.

Response 725.50c

In principle, the firewater system is included in two nodes of the CETs. In the ARV node, firewater could be used as an ECC system to inject water into the vessel, in order to prevent vessel failure. For this case, any possible release of fission products is directed to the suppression pool which results in significant fission product scrubbing. However, there is limited time available for the operator to perform this function, therefore, this mode has been conservatively neglected and only ECC systems are used in this capacity. In the ARC node, the firewater is used to deliver spray to the upper drywell after vessel failure has occurred.

After vessel failure fission products which had been held up on surfaces in the vessel are slowly revaporized and released to the drywell. The suppression pool maintains its scrubbing function as fission products are swept into the wetwell via the vent system. However, because of the large drywell volume, the sweeping of fission products to the wetwell occurs very slowly. The net effect of the slow revaporization of fission product from the vessel and the sweeping of fission products into the wetwell is a quasi-equilibrium condition in which about 10% of the volatile fission products remain in the drywell. Furthermore, after drywell head failure occurs, very little of the fission products revaporized from the vessel are swept into the wetwell. Thus, nearly all of the volatile fission products remaining in the vessel at the time of drywell head failure bypass the suppression pool.

As observed, the firewater system operating in spray mode is effective in removing the volatile fission products. The use of the RHR spray function would have the same result. In addition, for sequences in which there is low pressure failure and no containment failure (i.e. when heat removal is obtained either by the recovery of RHR or by use of the overpressure relief rupture disk, any ECC system can perform the ARC function. Therefore, it is not appropriate to call out the firewater system explicitly.

Question 725.51

It is repeatedly stated that corium cools in the LDW after vessel failure by the water which was retained in the lower plenum in many of the accident descriptions. Why did this water not cool corium in the vessel before vessel failure? How much water is available in this manner? Would accidents progress differently if the water cooled the core in vessel?

Response 725.51

The response to this question is provided in new Subsection 19E.2.4.9.

Question 725.52

Questions on Figures 19E.2-2 (Accident sequence LCLPPFDM)

(1) In Figure C, why does the upper drywell temperature continue to increase throughout the accident?

(2) In Figure E, why does the drywell water level change between the PF opening and the DW head failure?

(3) In Figure B, why does the drywell pressure decrease after water boils away? (The gas temperature does not show any corresponding drop during this period.)

Response 725.52

(1) The drywell temperature continues to increase during the transient due to radiation heat transfer from the vessel.

(2) The small, periodic changes seen in the lower drywell water level after the passive flooder opens are due to an instability caused by the small pressure and density differences between the lower drywell and the wetwell.

Consider an instant in time at which there is a small pressure drop between the wetwell and the lower drywell at the elevation of the flooder. Relatively cool water from the suppression pool flows through the passive flooder into the lower drywell. This causes the temperature of the lower drywell pool to drop slightly. The pressure in the lower drywell also decreases at this time (Figure 19E.2-2B) since it is governed by the saturation pressure of steam. This causes more water to flow into the lower drywell. When the elevation of water in the lower drywell is sufficient to prevent further flow, the water level stops increasing. The temperature in the lower drywell pool then begins to increase as decay heat is added. This generates steam causing the pressure in the lower drywell to increase, which could result in reverse flow through the flooder.

While this instability is based on physical phenomena, there are damping mechanisms (such as mixing in the lower drywell pool), which may overdamp the system, and prevent the water level from oscillating. It is not possible to model these mechanisms. Therefore, since the oscillating system could overpredict the steam generation and scrubbing rates, the logic used by MAAP to calculate flow through the passive flooder prevents this reverse flow of water. This causes the partial pressure of steam to increase gradually, and steam passes into the suppression pool via the vent system until the cycle repeats.

To determine sensitivity to this phenomenon, calculations were performed using very small time steps. This influences the magnitude of the oscillations. No significant effect on the transient response was observed as a result of the cyclic behavior.

(3) The drywell pressure decreases after water is initially boiled off in the lower plenum because there is no steam generation during this time, but steam condensation continues on the containment heat sinks.

Question 725.53

Referring to Figures 19E.2-5 (Accident Sequence LCHPPFPH): Figure A shows a pressure drop at about 17 hours. This was explained in the text as being due to the flow of water from the suppression pool into the drywell (A similar phenomenon was shown in Figure 19E.2-11.). Please clarify. It appears that the DW pressure should be higher than the WW pressure during this period. This pressure drop appears to delay the DW head failure by about 10 hours. What impact will this have on the final release fraction?

Response 725.53

The pressure in the drywell is higher than the WW pressure during the period in question. However, the water head of the suppression pool over the passive flooder causes water flow from the wetwell to the lower drywell. The phenomenon responsible for the pressure drop in this sequence is the same as that described in the response to Question 725.52b which referred to the change in the water mass in the lower drywell.

As discussed in the response to Question 725.52b, the changing water level, and the resulting pressure drop, are the result of a physical instability. The magnitude of the pressure drop (also observable in Figure 19E.2-2B) is amplified in this case because the partial pressure of noncondensible gases is lower in this case due to the leakage from the containment through the penetration seals in the drywell. This leakage also causes duration of the pressure correction to be increased.

Extrapolating the pressurization rate shown in Figure 19E.2-5 in order to determine the drywell head failure time is incorrect. If drywell head failure had not occurred, a second pressure correction would have been observed, resulting in a sawtooth pattern as shown by the sketch of undamped system response in Figure 20.3.9-16. Clearly, extrapolating the pressure using the slope of the curve during the time pressure is overly conservative. The pressure of a damped system would lie between the peaks of this response, as indicated by the sketch of the overdamped system response. One cannot extrapolate the pressurization rate based on a single incomplete cycle. Furthermore, the actual containment failure time will be dependent on the degree of damping in the system.

To estimate the uncertainty in the time to containment failure and the fission product release, a modification to MAAP was made which limited the flow of water from the passive flooder. It was assumed that the flow rate is equal to the mass of water which can be boiled away by decay heat, this serves to overdamp the pressure response. The pressurization of the containment using this assumption was very smooth. Rupture of the drywell head occurred at 35.7 hours compared with 37.0 hours for the undamped case shown in Figure 19E.2-5. The release rate of fission products is very similar to that of the base case. The final release fraction of CsI was about 12%, the same as the base case.

Question 725.54

The suppression pool bypass due to stuck open WW-DW vacuum breakers is of concern only for cases involving wetwell venting. Please explain the consequence ratio of 825 used in the equation on Page 19E.2-40. In the same equation, the fire water unavailability of 1.5% was assumed, which is considerably lower than 10% used elsewhere. Please explain.

Response 725.54

The consequence ratio is described in Subsection 19E.2.3.3.1(4). The firewater unavailability appropriate for this case is that used for containment analysis, which allows the operator ample time to diagnose the problem and begin operation of the firewater system. This must be contrasted with the probability of successful firewater injection in the vessel, where a limited amount of time is available. Considering both internal and external events a firewater unavailability of 1.5% was judged to be appropriate.

Question 725.55

The CET top event “ARC” (core melt arrest in containment) can occur if any of the following conditions exist, RHR is available, or RHR is recovered, or firewater is available, or PF operates.

Except for firewater, other features are already designated as top events of CETs (CHR, RCH, PF). Is it necessary to have “ARC” as a separate heading? It appears to be duplicative and confusing regarding how “ARC” occurred. (It is confusing since some of the top events are operation / availability of systems while some of them are events caused by operation of the same system.)

Response 725.55

The CETs model three types of core melt arrest:

- ARV - arrest in the RPV, generally within 1 hour
- ARC - arrest in the containment within about 20 hours
- P - arrest due to PF actuation.

With the current value of P=0.0 (i.e., PF is always successful), the node ARC is redundant. However, the CET was structured this way to enable sensitivity studies to be carried out by varying the values of P. Additionally, as discussed in the response to 725.50c, if a drywell spray is initiated prior to, or at the time of, a drywell head failure, the fission product release is significantly reduced. The release categories associated with successful ARC reflect this reduction.

Question 725.56

High temperature failure (HTF) occurs if corium is carried to the UDW and no spray is available. Does the probability 0.01 include the probability of both of these occurring? Wouldn’t it be clearer if this heading is replace by “Corium in the UDW” and “Spray Available”? (See also Question 725.57a)

Response 725.56

We believe that your question should read ‘... “Corium in UDW” and “Spray Unavailable” ...’ since the results of the PRA show that if either condition were not present there would be no HTF.

HTF occurs when the temperature in the UDW exceeds 533°K and the pressure exceeds 0.515 MPa. The temperature and pressure conditions are calculated by MAAP based on the systems available. With the passive flooder incorporated in the design, these conditions were met only for the sequences when there is corium in the upper drywell as a result of a high pressure melt sequence and the drywell sprays fail to operate. However, before the adoption of the passive flooder in the design, several other sequences resulted in HTF. Therefore, the use of HTF is more appropriate.

Question 725.57

Questions on Class IA/IA.1 and IIIA/IIIA.1 CETs

(1) High temperature failure probability is identical whether RHR is available or not in these CETs. However, if RHR is available, the probability to have UDW spray appears to be higher and, therefore, the probability of high temperature failure smaller. (See the previous question.)

(2) Why isn't the probability for "ARC Yes" 1.0 when RHR is available (i.e., what does the probability of 1.e-5 represent in Sequence 4 of CET IA?)

(3) Sequence 3 of CET IA is binned as ..FSNN. Does this imply that core melt is arrested in the containment due to FW? Why not RHR?

(4) How is core melt arrested in the containment without RHR for Sequences 4 and 6? Is this due to FW?

(5) What is the basis for the containment failure probability at the time of vessel failure, 0.001, or high temperature failure probability, 0.01? What is the sensitivity of the final consequence to uncertainty in these numbers?

Response 725.57

(1) For Class IA events, RHR is initially available, but operator action is required to align it in the UDW spray mode. Firewater, which is available, also requires operator action. The spray function is thus dominated by operator action and HTF value reflects the value assigned for operator action. This results in a somewhat conservative value for Class IA CET compared to the Class IA.1 CET. For the IA.1 CET, only firewater is available and the value selected is based on operator action to align firewater in the UDW spray mode.

(2) In CET IA, in sequence 4 RHR is available (i.e., RHR pumps are running and pipes are under pressure) but vessel injection had not taken place until the node ARC. When the reactor depressurizes following RPV failure, RHR injection is virtually assured. However, for conservatism, a value of 1.0E–05 was selected to model failure of all injection valves to open upon demand.

(3) In this sequence, there is a spray operating, as indicated by the lack of high temperature failure following a high pressure melt sequence. This function could be provided either by the firewater system or by the RHR system. In

addition, the RHR is providing containment heat removal so no containment failure occurs. The FS designator was used to indicate the presence of drywell spray.

(4) It is assumed that this question refers to CET 1A.1. In sequence 4 the core melt is arrested via either the firewater spray or RHR spray and containment heat removal is recovered before containment structural failure. In sequence 6 the firewater spray arrests the core melt, but drywell head failure occurs because the containment heat removal function is not recovered.

(5) The value of CI was assigned based on engineering judgment. It was judged that for a high pressure core melt sequence, the probability of CI was not high and at the same time was not negligible, and a value of 0.001 was selected. Increasing this value by a factor of 10 in all the CETs increases CET bin LCHP00EH directly by a factor of 10 (from 4.5E-11 per year to 4.5E-10 per year) and decreases frequencies of other sequences (in most of which core melt is arrested) by a small amount. This will not have significant impact on risk.

The HTF value is assigned based on the availability of UDW spray either from RHR or FW. As discussed in a) above, the value of 0.01 reflects operator action required to align the systems in the spray mode. Increasing this value by a factor of 10 is all CETs increases the CET bin LCHPPFPH value by a factor of 10 (from 4.5E–10 per year to 4.5E–09 per year) while decreasing frequencies of other sequences (in which core melt is arrested) by a small amount. This would result in a small increase in the CCDF curve for frequencies below 4.5E–09 per reactor year.

Question 725.58

Questions on Class IB-1 / IB-1.1 and IB-3 / IB-3.1 CETs

(1) How is the core melt arrested in the containment for Sequences 2 and 4 of these CETs? Are these probability same for IB-1 and IB-3 because they are solely due to FW?

(2) Why isn’t the RHR recovery probability 100% for Sequences 2 and 5 for IB-1?

(3) Why is probability of the RHR recovery failure significantly higher for Sequence 7 than for Sequence 4 in IB-1?

(4) Why is the probability of RHR recovery failure 5 times higher for Sequence 4 of IB-3 than Sequence 4 of IB-1, while they remain the same between Sequences 7 of IB-1 and IB3? (Incidentally, the “RCH No” branch probability for Sequence 7 of IB-3 appears to be misprinted. It should be 0.1, not 0.01.)

(5) Sequence 7 of IB-1 is binned as PFDH while Sequence 7 of IB1.1 as PSDN. This implies that the consequence of the low pressure vessel failure is more significant than that of high pressure. Please explain. (The same question for IB-3.)

Response 725.58

(1) In CETs IB-3 and IB-3.1, the core melt is arrested in sequences solely due to FW. For CETs IB-1 and IB-1.1 it is also possible for recovered ECC systems to inject water to the containment; however, since the failure probability of ARC is very small, the same value was used for all trees.

(2) In CET IB-1, at node RCH, onsite power becomes available which means that there is a high probability (but not 100%) that RHR will become available. It was judged that some operator action would be required to align the system in the correct mode, and a value of 0.01 was assigned for RCH in Sequence 2 to reflect the operator error probability. In Sequence 5, a higher value of 0.1 was assigned to RCH because of an additional operator error earlier in the sequence at node ARC when he could not get the FW aligned successfully.

(3) Answered as part of b) above.

(4) We agree that RCH value in Sequence 7 of CET IB-3 is misprinted as 0.01 instead of 0.1 and Figure 19D.5-9 has been corrected.

There is a difference between the RHR recovery probabilities in Sequence 4 of IB-1 and IB-3. In CET IB-1, at node RCH, power had been recovered and RHR has a high probability of recovery. However, as explained in b) above, a value of 0.01 was used to take into account operator action that was judged to be necessary. In CET IB-3, at node RCH power had not been recovered, but there was a conditional probability associated with its recovery. Since IB-3 event frequency accounts for power recovery up to 8 hours (i.e., at the entry to CET IB-3 the frequency 6.4E-10 represents all sequences in which power was not recovered in 8 hours) at node RCH conditional probability of recovering power in 20 hours, given it was not recovered in 8 hours, was calculated to be 0.022. The RCH value was then calculated by combining 0.022 with the operator error probability of 0.01 to obtain 0.032 which was rounded up to 0.05 in Sequence 4.

In Sequence 7 also there is a similar difference in the RCH values of IB-1 and IB-3. However, this value of 0.022 was considered negligible compared to the high operator value of 0.1 used for this node and the same value of 0.1 was used for both CETs.

(5) There is a typographical error in the bin names for Sequence 7 of CETs IB-1 and IB-3. The sequence name should be LCLPPFDM as shown in corrected Figure 19D.5-6.

Sequence 7 of IB-1 results in higher releases than Sequence 7 of CET IB1.1 because there is a containment spray available in the high pressure melt sequence, as designated by the sixth character in the sequence name. The containment spray serves to “scrub” the fission product release which might result from drywell head failure.

Question 725.59

Questions on Class IB-2 CET

(1) The core damage frequency for this class is not the same as that of Table 19.3-6. Please clarify which is correct.

(2) The probability of failure to depressurize the reactor is 3 times lower for Class IB-2 compared to Class IB-1/3 (0.002 vs. 0.006). Is this due to the time available before depressurization? Does this probability depend on how much time is available before the demand of this equipment? (i.e., what action can be taken to improve availability of this equipment before challenge regardless of how much time is available?)

(3) Please provide the basis for the “ARV No” branch probability of 0.006 for Sequences 4 to 7 and 0.6 for Sequence 12.

(4) Why is the “ARC No” branch probability of Sequence 7 significantly higher for this CET than others (0.05 vs. 0.01)? Why isn’t this branch further divided depending on the RHR recovery? (This is done for cases which have even smaller probabilities.)

(5) Sequence 6 is binned as FSDH. This is the only place where a sequence is binned as “High” when FW scrubbing is available. Please explain.

(6) Why is RHR unavailability significantly lower for Sequence 11 compared to the similar sequences for other CETs (0.01 vs. 0.05 for IA)?

(7) Why isn’t Sequence 12 further branched like the similar sequences of IB-3.1?

Response 725.59

(1) The correct value of 1.55E-8 is given in Table 19.3-6. The value of 1.8E-8 used in the CET IB-2, Figure 19D.5-8, has been corrected to read 1.6E-8 (rounded). The incorrect value used gives higher value for the sequences in CET IB-2.

However, since the probability of fission product release is very low, there will be negligible impact on risk.

(2) The difference in probability of failure to depressurize the reactor between Class IB-2 and Class IB-1/3 CETs is attributable only to the time available for operator action. The equipment unavailability was considered to be negligible compared to operator error probabilities.

(3) In the IB-2 CET, RCIC fails after 8 hours of operation. If power is recovered in 2 hours, core melt can be arrested in the RPV. The conditional probability that power is recovered in 10 hours given that it was not recovered in 8 hours is calculated to be 0.6. Once the power is recovered, many high pressure systems become available and the system unavailabilities are judged to be negligible compared to the value of 0.6. Therefore, a value of 0.6 was used for Sequence 12.

(4) For Sequences 4 through 7, the reactor has been depressurized and FW also becomes available. The combined probability of recovering power and failing FW is calculated to be 0.006 (i.e., 0.6 for power recovery times 0.01 for FW).

(5) Sequence 7 of CET IB-2 models the failure of ARC. In this sequence, SBO event has lasted more than 8 hours without recovery of offsite or onsite power. RCIC has operated for 8 hours and then failed. The operator successfully depressurized the reactor, core melt started leading to RPV failure when core melt was not arrested in the RPV. Corium was discharged on the lower drywell floor with containment intact. Core melt can be arrested in the containment if power is recovered within 20 hours. Earlier at node ARV, recovery of power in 10 hours has been accounted for. At this node the conditional probability of recovering power in 20 hours, given it was not recovered in 10 hours is calculated to be 0.04. Once power is recovered, many systems become available and the system unavailability is judged to be negligible. The 0.04 was rounded off to 0.05 for use in the CETs. No additional credit was taken for recovery of FW which had failed earlier at node ARV.

We agree that this branch should be further divided depending upon the RHR recovery. Such RHR recovery was modeled in the CET. The event tree nodes were mislabeled and Figure 19D.5-8 has been corrected. No changes to the tree structure or calculations are necessary since RHR recovery was factored in the CET logic. Note that an additional correction was made to the bin names for Sequences 1-3 as discussed in Question 50(a).

(6) There is a typographical error in the bin name for Sequence 6 of IB-2. The correct sequence name is SBRCFSDL, as shown in corrected Figure 19D.5-8.

(7) Sequence 11 was computed with a value of 0.01 for RCH. However, this value has been wrongly entered as 0.25 in the CET. This correction has been made in the copy of the IB-2 CET provided in response to question d) above. No changes to the results are necessary since they have been evaluated using the correct value of 0.01.

In this sequence SBO event lasts more than 8 hours, RCIC operates and fails after 8 hours. However, power is recovered in about 10 hours and core melt is arrested. With power recovery, there is a high probability that RHR becomes available; however, a value of 0.01 is used on the assumption that some operator action may be required.

(8) In Sequence 12, the reactor is at high pressure, core melts leading to RPV failure and the power has not been recovered in about 11 hours. Since the probability of this case is very low, it was decided to conservatively model this event sequence as one leading to high release.

Question 725.60

Questions on Classes ID and IIID CETs.

(1) How is core melt arrested in RPV? In this solely due to FW? (This branch existed in Amendment 4 which did not have FW.)

(2) Why is the probability of RHR recovery failure significantly higher in this CET than in others?

Response 725.60

(1) Core melt is arrested in RPV if any one of the failed systems is recovered in about 1 hour or if the FW is aligned by the operator. Numerically, however, the recovery probability of core cooling systems is negligible compared to the FW availability.

(2) In most SBO CETs, if offsite or onsite power is recovered within about 20 hours, RHR can be recovered. The comparison of RHR recovery probabilities for CETs ID & IIID should therefore be made with non-SBO events (e.g., CET IA-1) where RHR is not available at the beginning of the event. In these events, the RHR recovery is calculated by dividing the CET frequency into three parts: 1) percentage of events with system failure; 2) percentage of events with loss of offsite power and 3) percentage of events with SBO. Each of these parts have a different recovery probability associated with it and a weighted average is used to estimate the RHR recovery

probability. This process yields a higher value for the Class ID events than for Class IA.1 events because of the differences in the relative proportion of these three parts.

Question 725.61

Questions on CET II.

The “CC No” branch fraction is significantly reduced from Amendment 4 to Amendment 8 (0.001 from 0.1). Besides the availability of firewater, what else contributed to this reduction?

Responses 725.61

On Figure 19D.5-12, the “CC NO” branch fraction is 0.001 as stated in the question. The basis for the value is stated in Subsection 19D.5.11.3, the last paragraph of which concludes: “A value of 0.01 was used in the analysis for loss of conventional cooling. Since the fire water addition system is much less vulnerable (to containment failure), it was assumed available 90% of the time, even if conventional cooling was lost.” Thus “CC NO” = 0.01 x (1-0.9) = 0.001 in Figure 19D.5-12.

Table 20.3.9-1 RHR Vessel Nozzle Locations

	Elevation - cm from RPV bottom)	Azimuth (Degrees)
Vessel Outlet -		
RHR-A	1091	10
RHR-B	1091	310
RHR-C	1091	170
Vessel Inlet-		
RHR-A (Injects via FW line)	1161	30, 90, 150
RHR-B	1091	240
RHR-C	1091	60

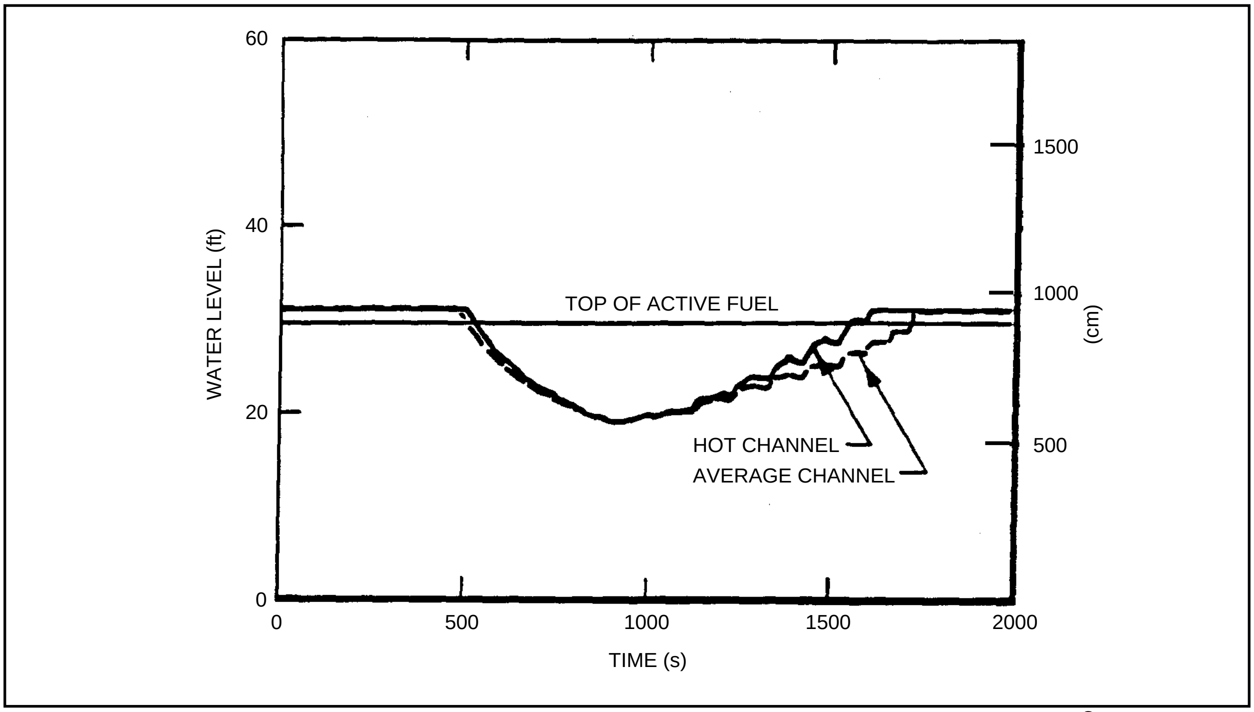

Figure 20.3.9-1 Water Level in Fuel Channels Following a 0.00202 m^2 Vessel Bottom Head Drainline Break:1 RHR + 3 ADS Available

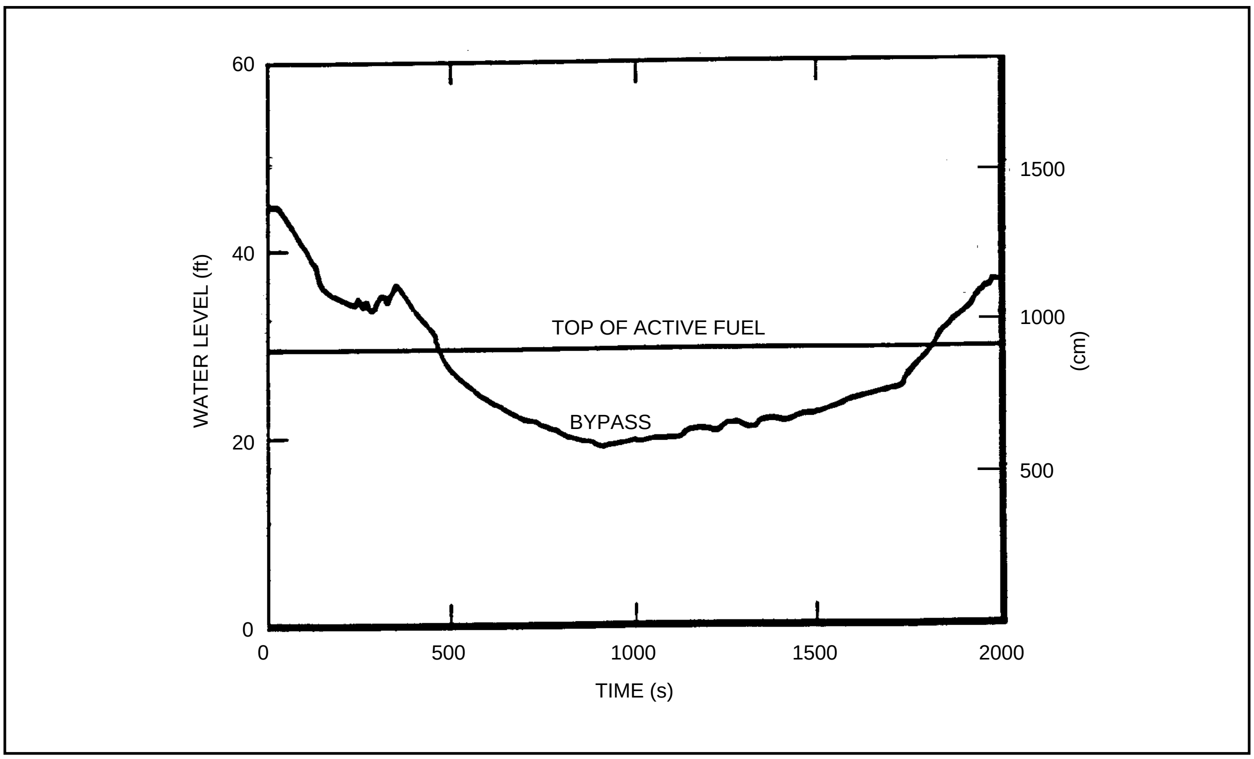

Figure 20.3.9-2 Water Level Inside Shroud Following a 0.00202 m^2 Vessel Bottom Head Drainline Break:1 RHR + 3 ADS Available

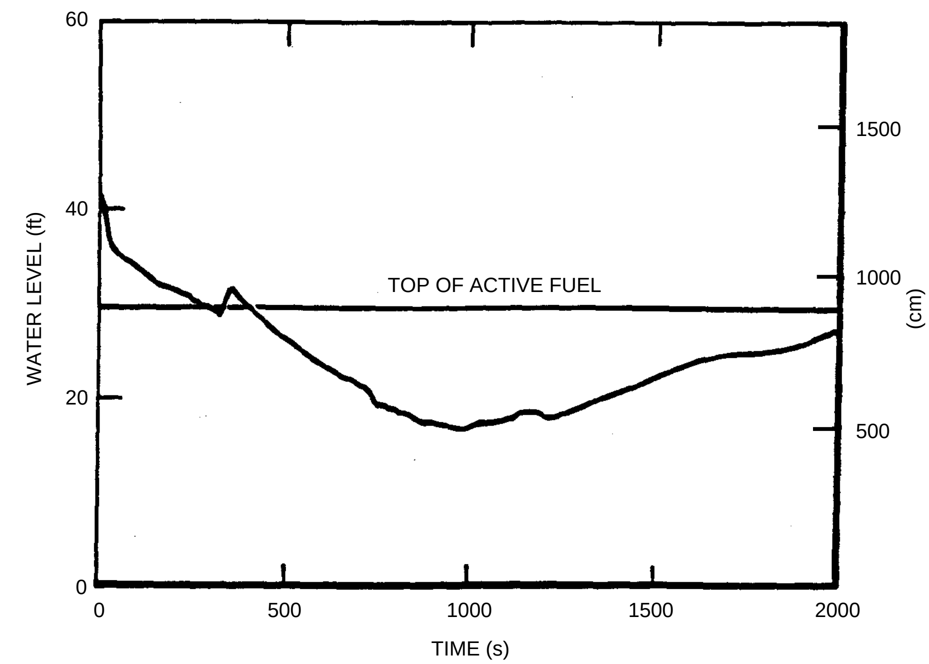

Figure 20.3.9-3 Water Level Outside Shroud Following a 0.00202 m^2 Vessel Bottom Head Drainline Break:1 RHR + 3 ADS Available

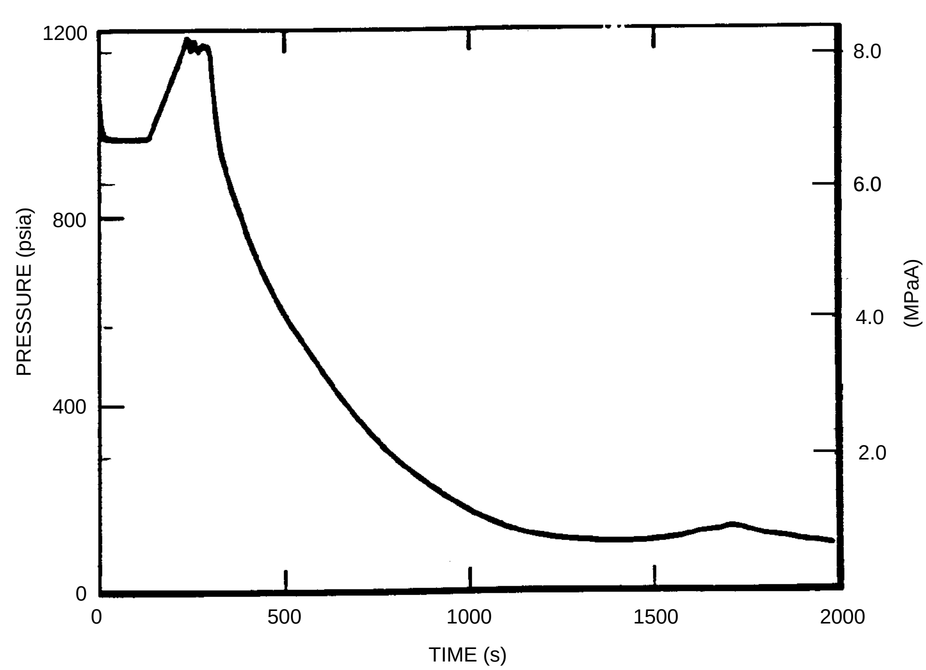

Figure 20.3.9-4 Vessel Pressure Following a 0.00202 m^2 Vessel Bottom Head Drainline Break:1 RHR + 3 ADS Available

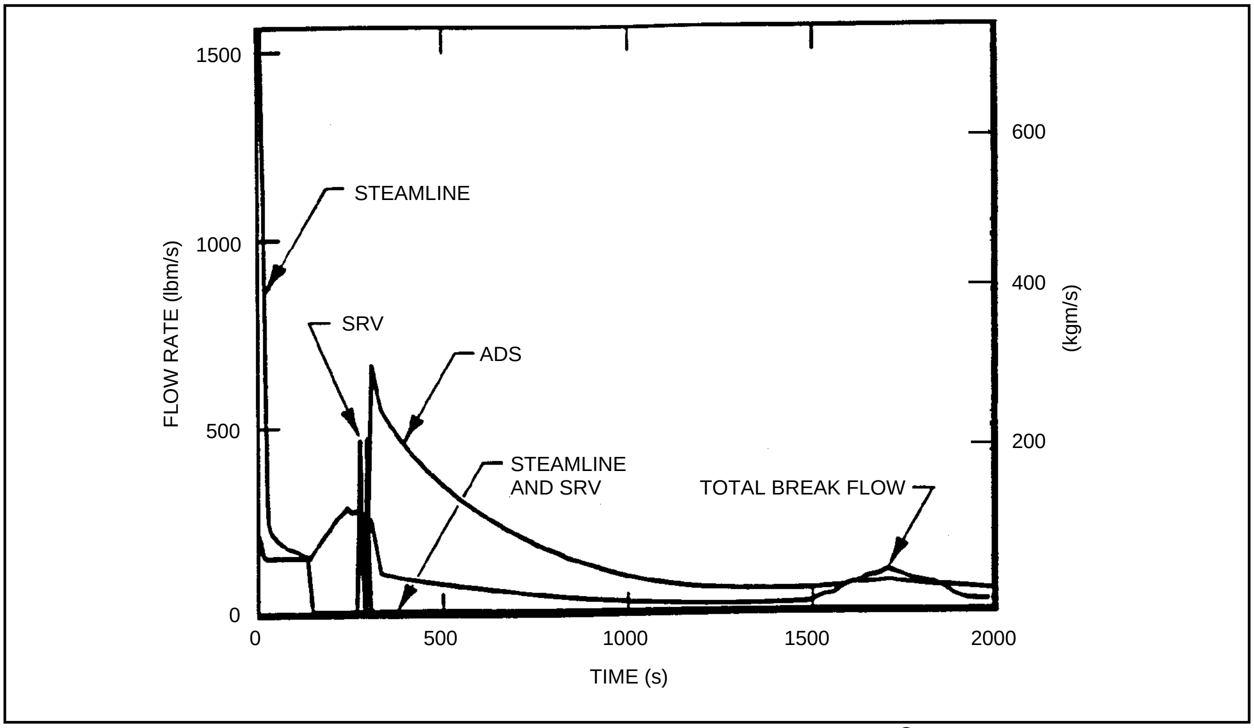

Figure 20.3.9-5 Flow Out of Vessel Following a 0.00202 m^2 Vessel Bottom Head Drainline Break:1 RHR + 3 ADS Available

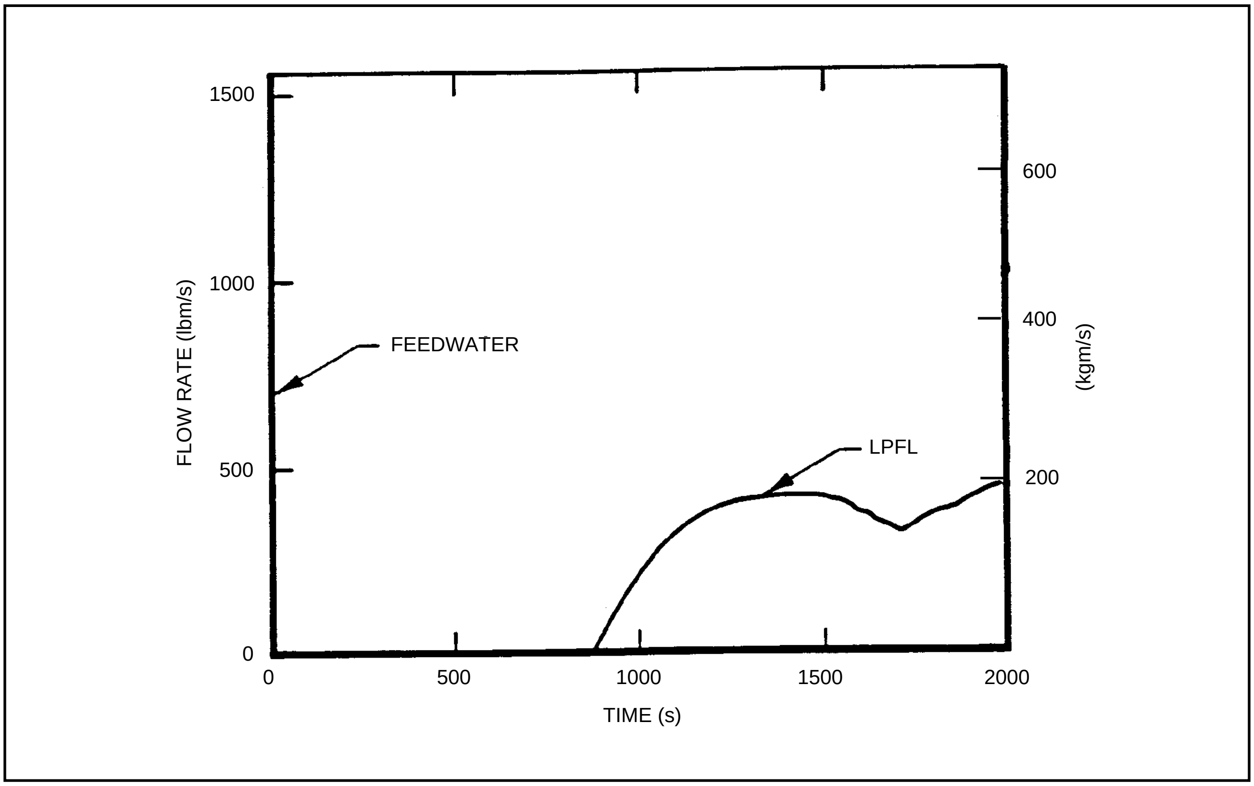

Figure 20.3.9-6 Flow Into Vessel Following a 0.00202 m^2 Vessel Bottom Head Drainline Break:1 RHR + 3 ADS Available

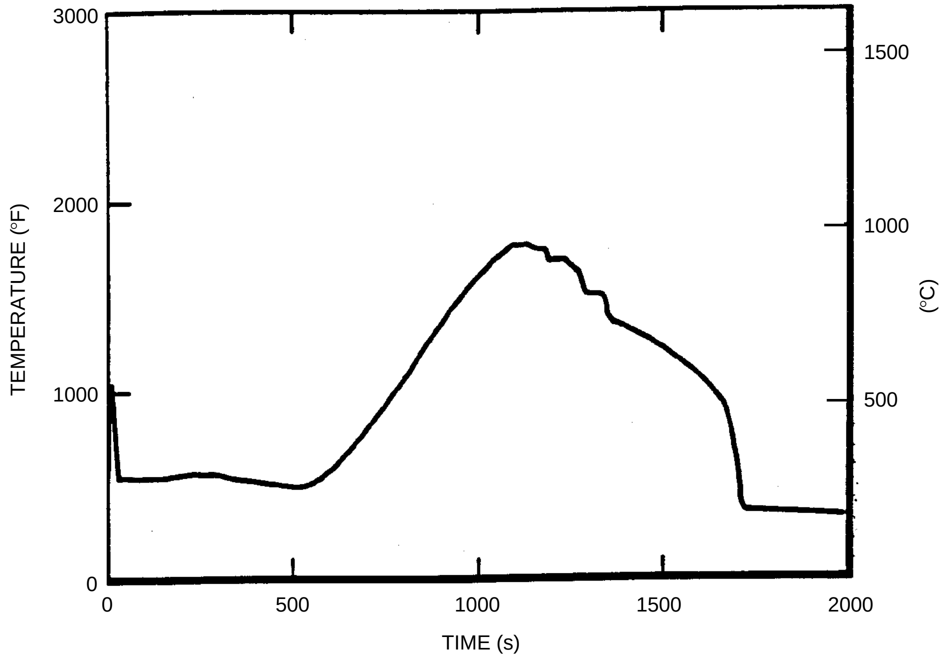

Figure 20.3.9-7 Peak Cladding Temperature Following a 0.00202 m^2 Vessel Bottom Head Drainline Break:1 RHR + 3 ADS Available

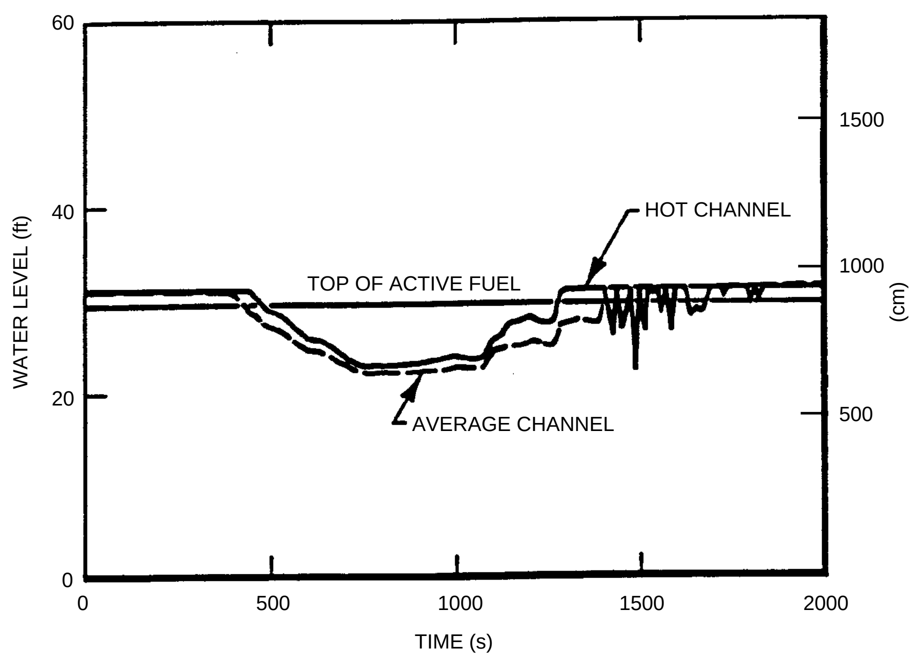

Figure 20.3.9-8 Water Level in Fuel Channels Following a 0.0279 m^2 Break in the RHR Vessel Shutdown Suction Line: 1 RHR Available

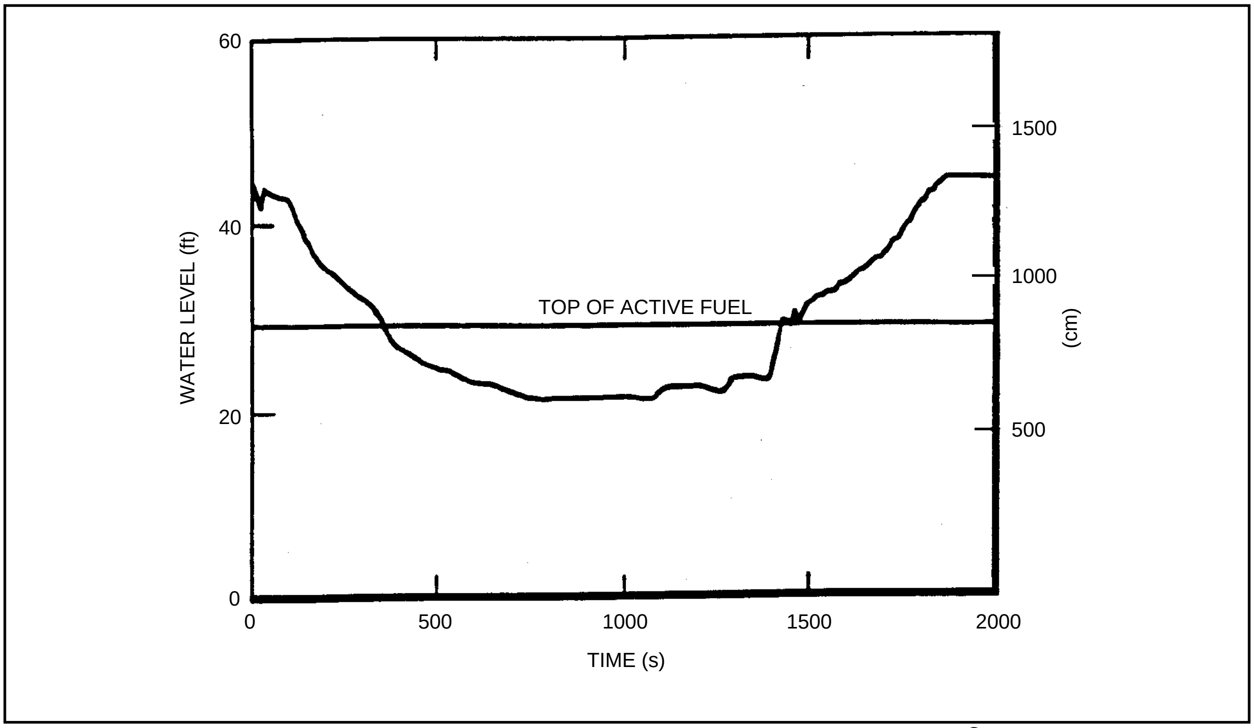

Figure 20.3.9-9 Water Level Inside Shroud Following a 0.0279 m^2 Break in the RHR Vessel Shutdown Suction Line: 1 RHR Available

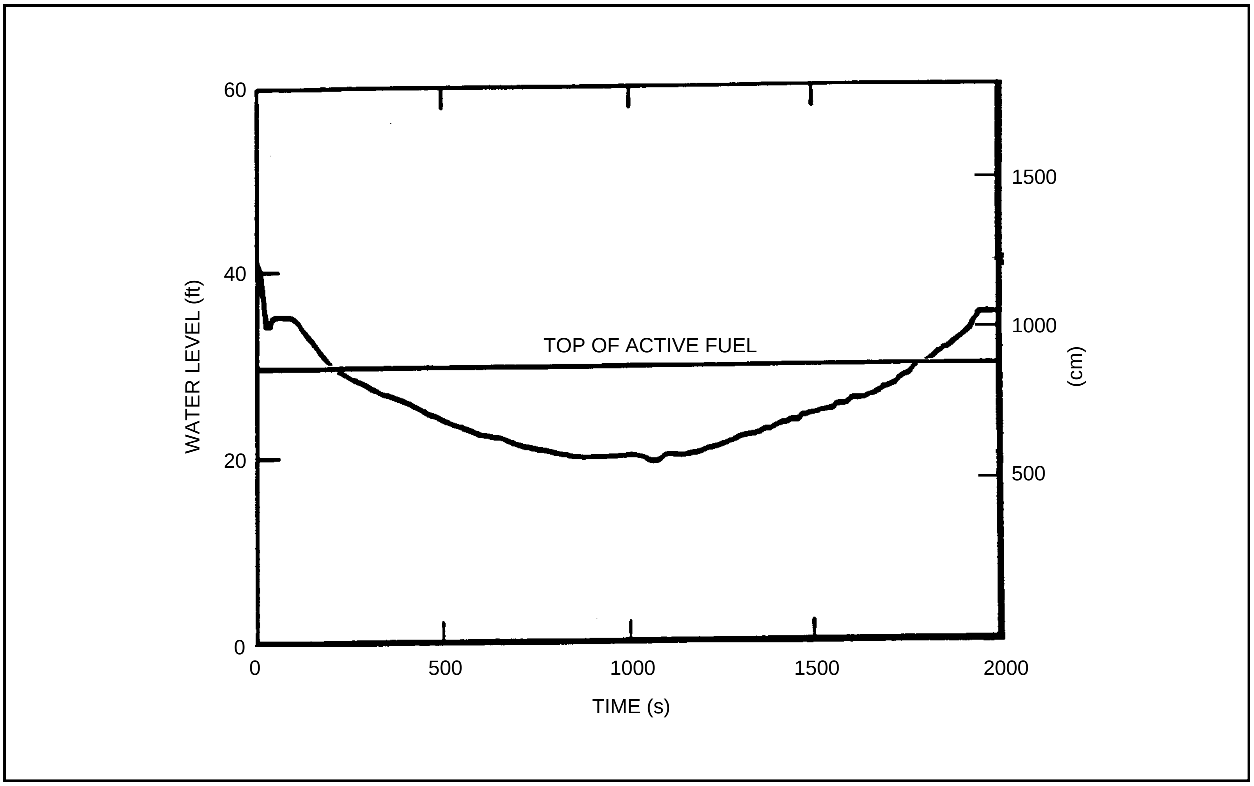

Figure 20.3.9-10 Water Level Outside Shroud Following a 0.0279 m^2 Break in the RHR Vessel Shutdown Suction Line: 1 RHR Available

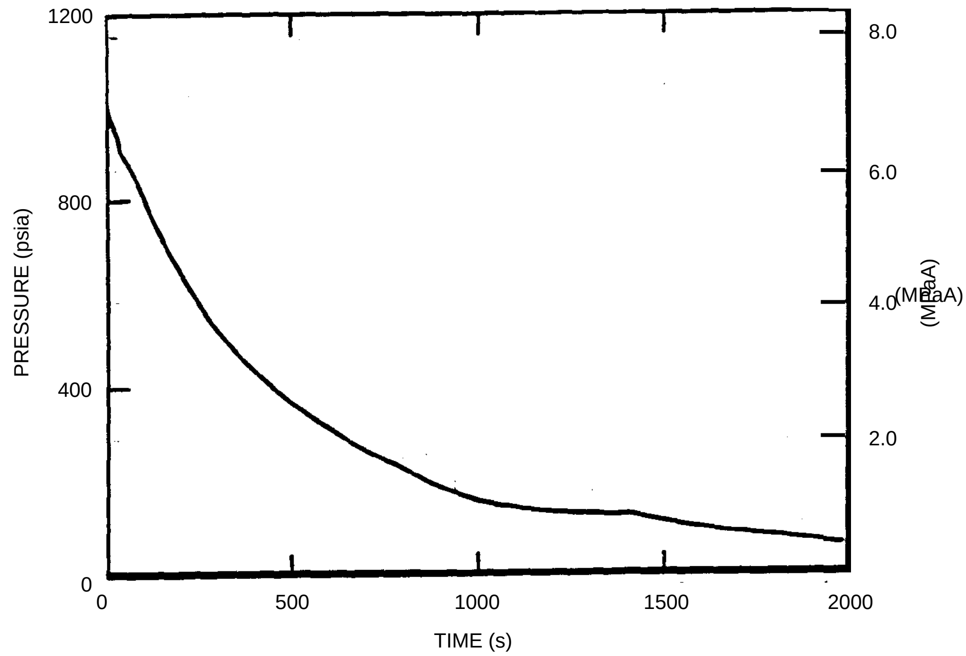

Figure 20.3.9-11 Vessel Pressure Following a 0.0279 m² Break in the RHR Vessel Shutdown Suction Line: 1 RHR Available

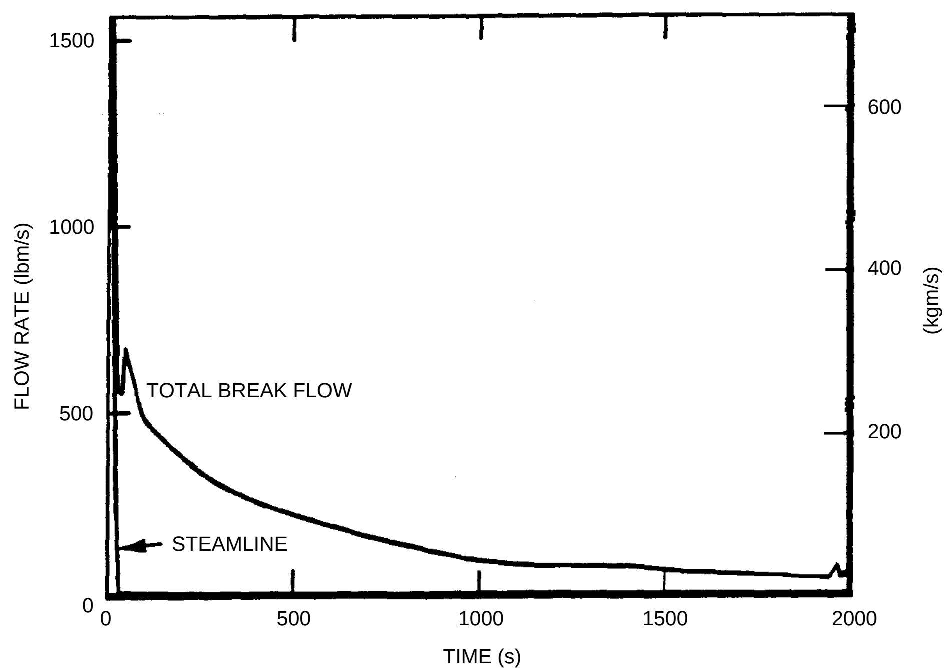

Figure 20.3.9-12 Flow Out of Vessel Following a 0.0279 m² Break in the RHR Vessel Shutdown Suction Line: 1 RHR Available

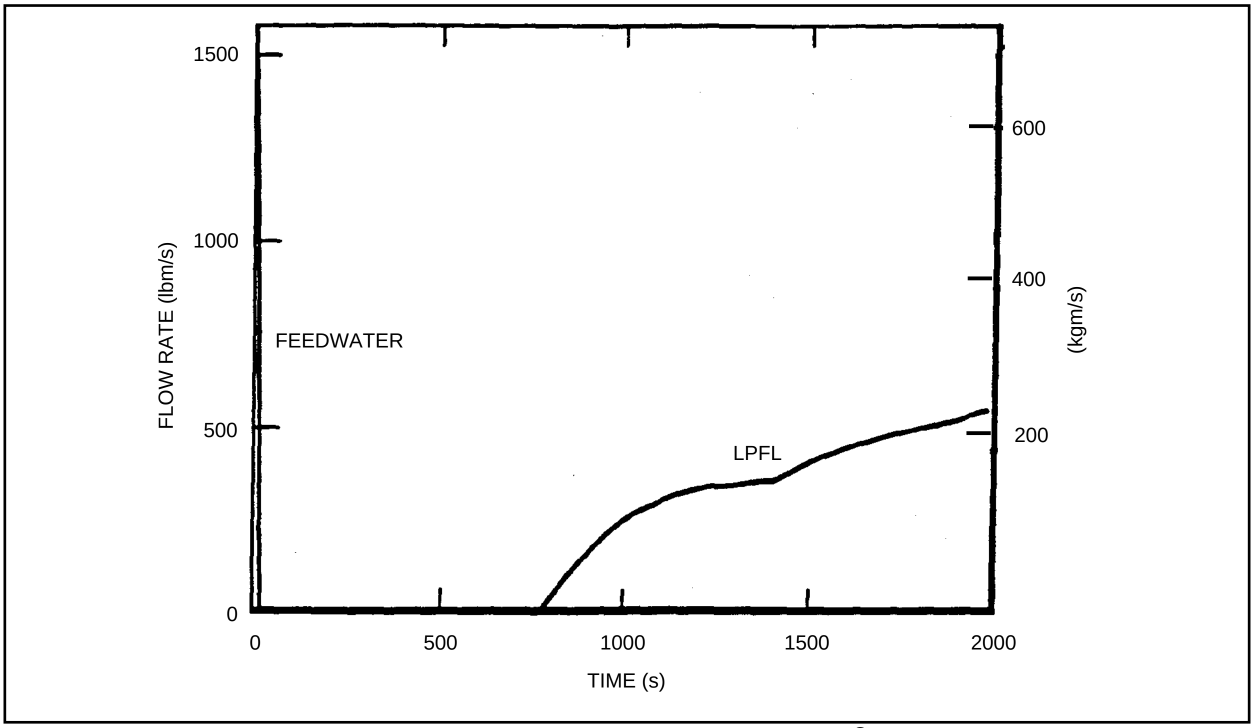

Figure 20.3.9-13 Flow Into Vessel Following a 0.0279 m^2 Break in the RHR Vessel Shutdown Suction Line: 1 RHR Available

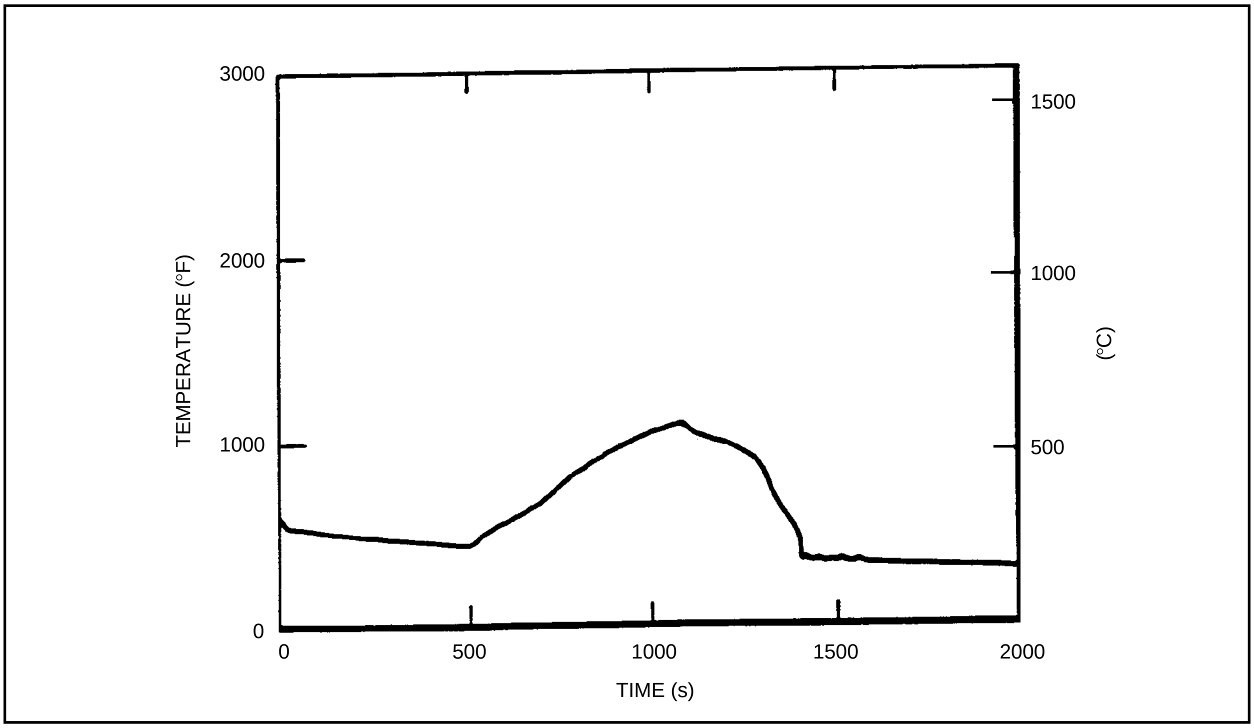

Figure 20.3.9-14 Peak Cladding Temperature Following a 0.0279 m^2 Break in the RHR Vessel Shutdown Suction Line: 1 RHR Available

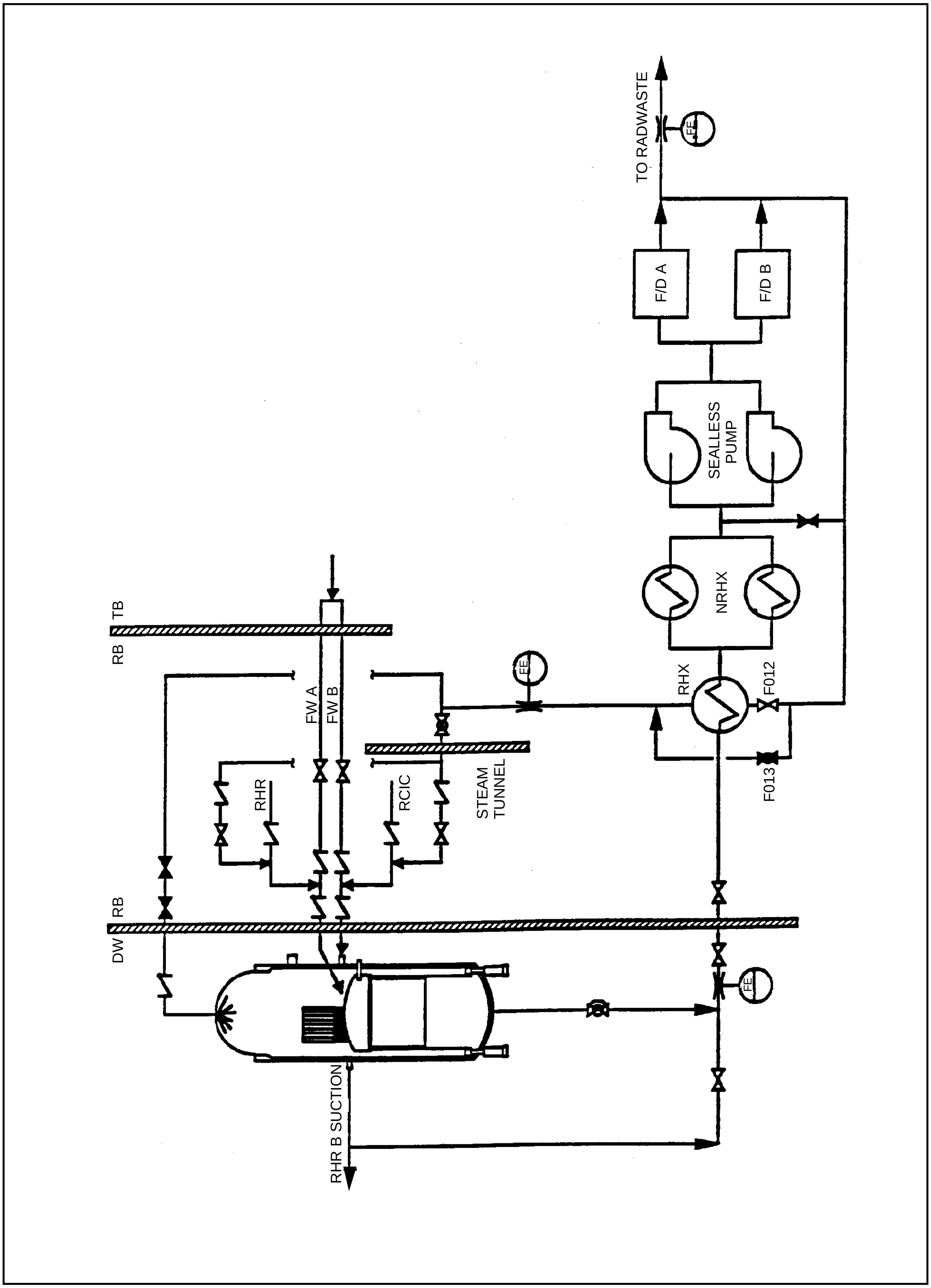

Figure 20.3.9-15 Reactor Water Cleanup System (Response to Question 725.32)

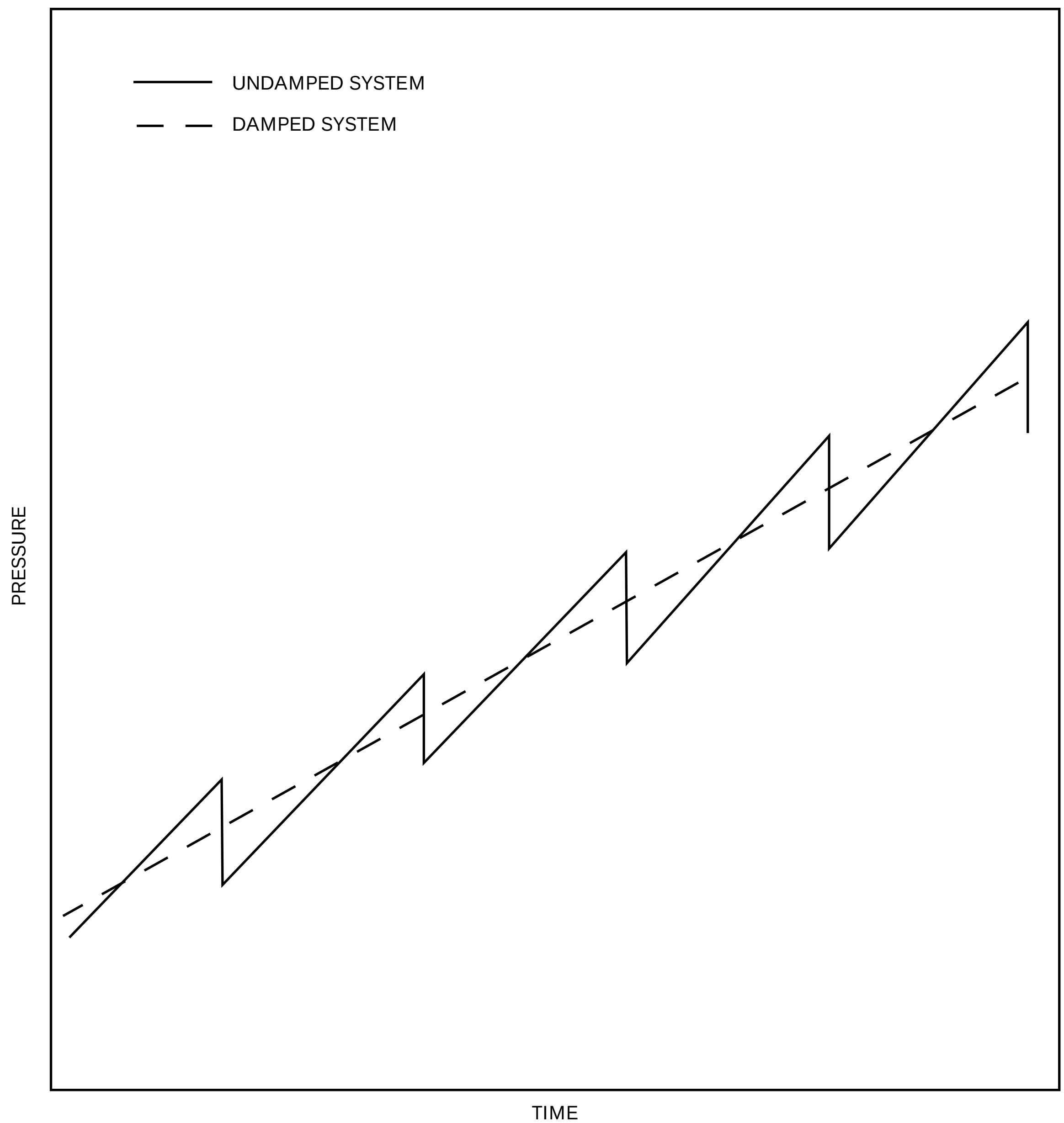

Figure 20.3.9-16 Characteristic Response for Damped and Undamped Systems (Response to Question 725.53)

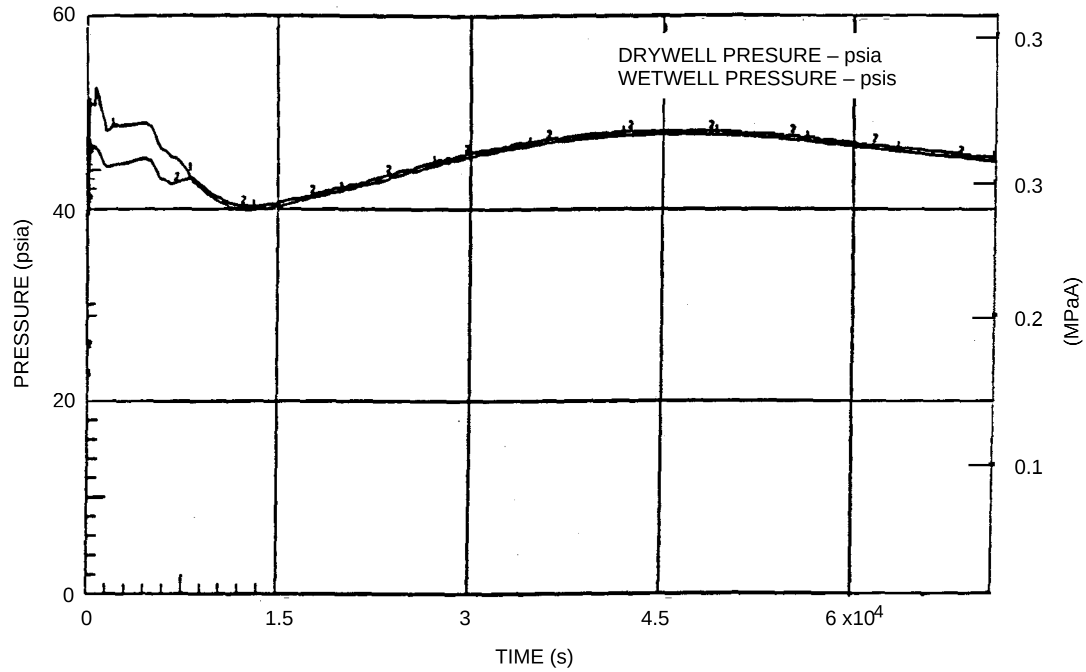

Figure 20.3.9-17 Pressure Time History After a Feedwater Line Break Available ECCS: 1 RHR System

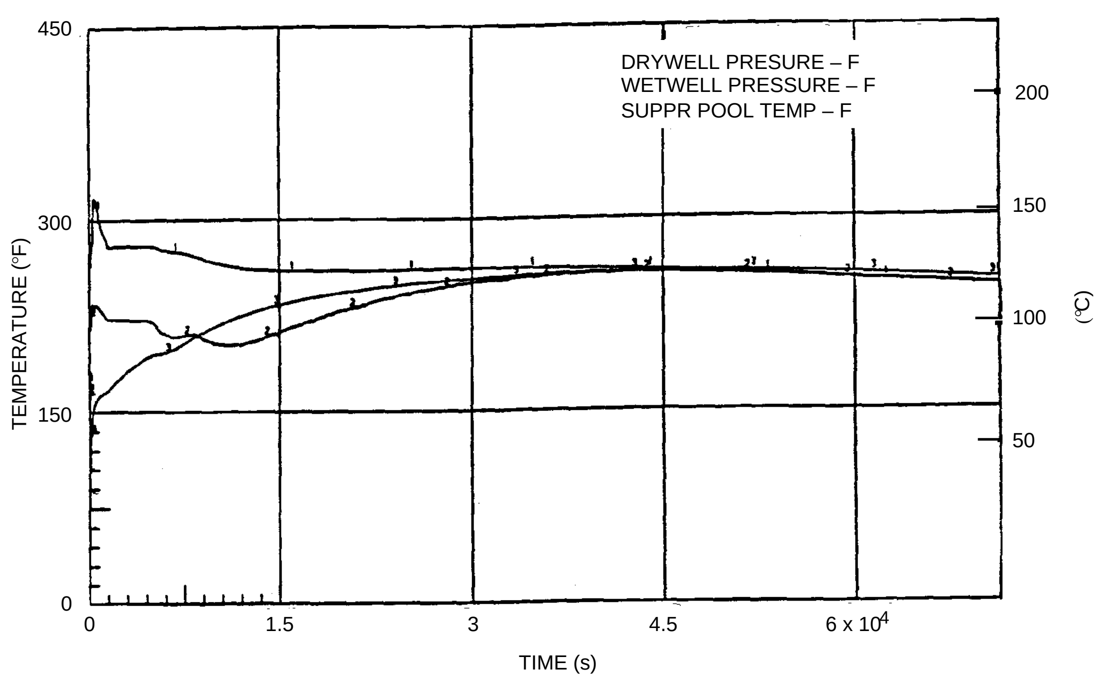

Figure 20.3.9-18 Temperature Time History After a Feedwater Line Break Available ECCS: 1 RHR System