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6	6-69	720.029 (R1)
24	24-17	720.046 (R1)
24	24-22 and 24-23	720.046 (R1)
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30	30-26a through 30-28	720.029 (R1)
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Appendix A	A-103c and A-103d	720.012 (R1)
Appendix A	A-151 and A-152	720.013 (R1)
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REVISION 2 CHANGE ROADMAP (Cont.)

Section	Page No.	Type of Change[1]
Appendix D	D-13 and D-14	720.080 (R1)
Appendix D	D-35 and D-36	720.080 (R1)

1. Changes incorporated as a result of Westinghouse responses to NRC Request for Additional Information (RAI) identified by RAI number.

MLOCAs are primary system breaks with a diameter from 2 inches to 9 inches. Breaks that are located on the hot leg or cold leg are modeled in the MLOCA event tree. CMT line breaks and DVI line breaks are modeled in separate event trees, but have break sizes within the MLOCA spectrum.

The CMT line break is defined as any break in the CMT balance line or CMT injection line up to the check valves that prevent reverse flow from the DVI line. The CMT line break is very similar to an 8-inch MLOCA on the cold leg, except the affected CMT cannot inject into the RCS until it is almost completely depressurized. This renders the affected CMT ineffective, because it is not capable of providing automatic ADS signals and by the time the RCS is depressurized to such a low pressure the IRWST is capable of providing sufficient injection. Therefore only 1 CMT is considered available to provide injection to the RCS for this break. However, the break location does not affect the accumulators. Both accumulators may be available, since there are check valves in the discharge of the CMTs that are between the postulated break location and the accumulator tee.

A DVI line is an 8-inch pipe, but the effective area of the DVI line break initiating event depends on the location of the break. For all locations on the DVI line, the initial effective break area cannot be greater than 4 inches because there is a flow restrictor in the DVI nozzle where it connects to the reactor vessel downcomer. For a double-ended break, a second pathway for coolant loss from the RCS can be created after the CMT isolation valve is opened. The second pathway can be equivalent to a 4.4 inch or 6.8 inch inside diameter (ID) break depending on the location. The second pathway allows coolant loss from the RCS via the cold leg and CMT. Whether this second break pathway occurs depends on the opening of the faulted CMT isolation valve, which is not explicitly modeled on the event tree. While the second pathway has the potentially adverse impact of additional coolant loss, the associated draining of the faulted CMT leads to earlier ADS actuation signals, which is a benefit. Ignoring the draining of the faulted CMT causes the DVI line break to be very similar to a 4-inch MLOCA, except there can never be more than 1 CMT and 1 accumulator to provide injection to the RCS.

Thus the CMT line break and DVI line break are differentiated from the MLOCA event tree due to equipment loss that occurs as a result of the initiating event. However, the success paths still require the same operating equipment as the MLOCA event. For example, 1 out of 2 CMTs may be part of a MLOCA success path, while 1 out of 1 CMT would be the success criterion on the equivalent CMT line break or DVI line break success path. Thus, analyzing a spectrum of hot leg breaks from 2 inches to 9 inches is typically representative of the CMT line break and DVI line break too.

LOCAs smaller than 2 inches diameter but larger than 3/8-inch diameter are represented in the small LOCA event tree. When stage 4 ADS valves are part of the success criteria for these smaller breaks, a stage 2 or stage 3 ADS valve must first open to reduce the RCS pressure below the stage 4 interlock pressure. Thus the small LOCAs are sometimes referred to as a high pressure event.

There are additional initiating events that have high pressure success paths including the actuation of ADS. For example, a PRHR tube rupture is a specialized small LOCA. It is considered in a separate event tree since it is feasible for the operator to terminate the event

by isolating the break. In addition, the operation of the CVS could reduce the net loss from the break. If the break is not isolated, the PRHR tube rupture accident progression transfers to the SLOCA event tree. Likewise, there is a transfer to the SLOCA event tree for the RCS leak event, if the CVS fails or if the operator fails to take actions that will keep the CVS injecting. The smaller break size of the RCS leak would cause the timing of the accident to be slower than for a larger break, but can generally be represented by the same accident progression as the SLOCA.

Another special small LOCA is the possible sticking open of a pressurizer safety valve during an ATWS accident. In this situation, the turbine has been tripped, reactor heat removal has been established (PRHR HX or SFW), the pressurizer safety valves have successfully opened and limited the peak reactor pressure. In a couple minutes, the reactor power stabilizes at the PRHR HX capacity and the reactor pressure drops below the pressurizer safety valve setpoint. At this time, the pressurizer safety valves should re-close. If one of the safety valves does not re-close, the event becomes a small LOCA. In this case one CMT is required to operate to shutdown the reactor and to provide high pressure reactor makeup. By the time ADS occurs on low CMT level, the reactor would be shutdown and its temperatures and pressures would have returned to levels at or below full power values. As a result, this event is bounded by other small LOCAs.

The SGTR initiating event is also a specialized small LOCA. It is modeled in a PRA event tree that first considers possible sequences for successful core cooling that include non-safety related systems, operator actions, and automatic isolation of the faulted steam generator. Actuation of ADS valves is only a part of the success path when some of these systems or actions have failed. Although the rupture of a steam generator tube creates concerns of large releases directly to the environment if there is core damage, the location also generally makes it easier to prevent core damage. However, if the event progresses to the need for ADS valves, the general plant behavior is that of a high pressure event.

In addition to small LOCAs, there are transient initiating events that can also lead to ADS actuation. The similarity in the transient initiating events is that there is some failure or power excursion that causes a reactor trip. All the success paths for transient initiating events that include ADS actuation also have failure of main feedwater, failure of startup feedwater, and failure of PRHR. With the failure of these systems, the steam generators remove decay heat until the secondary side empties. When the secondary side heat sink is lost, the RCS heats up and pressurizes until the pressurizer safety valves open. The inventory loss through the safety valves causes the CMTs to recirculate and eventually drain. CMT draining actuates ADS to mitigate the event. This is classified as a high pressure accident scenario, and the accident progression is very similar to the SLOCA initiating event.

Within the transient initiating events, separate alpha-designators have been defined for success paths with loss of offsite power, station blackout, or a steamline break initiating event. The loss of offsite power and station blackout are separate due to differences in the fault trees, however the RCS thermal-hydraulic response is not different. The steamline break event with failures of the PRHR and CVS is also similar to the other transients. However, the steamline break can cause a faster depletion of the steam generator inventory than some other events. The steamline break event is one of the few events where a CMT actuation signal occurs when there is not a loss of primary fluid. If the CMT and PRHR function, the accident

A3.5 Post ADS Long Term Core Cooling

In the AP600, long-term core cooling success criteria was justified by conservative analysis methods using the WCOBRA/TRAC long-term cooling model. Success for long-term cooling was based on the ability of the passive core cooling system to provide sufficient gravity injection from the containment recirculation flow paths and sufficient venting from the ADS valves. For long-term cooling, the analysis was performed with conservative assumptions (i.e., minimum safeguards flow rates and vent size, Appendix K decay heat, etc.) and were included in Reference A-4. For AP1000, the capacities of the ADS-4 valves, IRWST injection and containment recirculation lines have been significantly increased relative to the AP600. The following changes contribute to this increase in capacity.

- IRWST injection lines increased from 6″ to 10/8″
- Containment recirculation lines increased from 6″ to 8″
- ADS-4 common lines increased from 14″ to 18″ and the ADS-4 valves were increased from 10″ to 14″
- Initial IRWST water level increased from 130′ to 131.58′
- Post accident containment water level increased from 106.2′ to about 109′
- Check valves added to refueling cavity drain to prevent it from flooding initially
- RNS water supply changed from IRWST to spent fuel cask loading pit (increases IRWST drain down time)

The increase in AP1000 relative to AP600 flow capacity is 89 percent for ADS-4, 84 percent for IRWST injection and 130 percent for containment recirculation; the core power has increased 76 percent (Reference A-21). Therefore, this increase in ADS Stage 4 vent capacity is judged to be sufficient for justification of the long-term core cooling success criteria assumed in this revision of the AP1000 PRA.

A4 Anticipated Transient Without Trip

A4.1 ATWS Background

Failure of the reactor trip function could result from several causes:

- Reactor trip signal from the protection and monitoring system (PMS) fails
- Reactor trip breakers fail to open
- Rod cluster control assemblies (RCCAs) fail to fall into the core after power to the gripper coils is removed

For AP1000, if the reactor trip function fails, the diverse actuation system (DAS) would provide a backup method for tripping the reactor. Failure of DAS reactor trip could result from failure of the DAS signal, or failure of the RCCA motor-generator (MG) sets to trip.

There is no credible mechanism for mechanical binding of multiple RCCAs once power is removed from the gripper coils (except possibly as a result of a seismic event, which the internal events PRA and models are not intended to address). Further, even with all of the RCCAs stuck out of the core, AP1000 core characteristics and plant features are available to mitigate the event consequences and avoid an overpressure in excess of the ASME service level C limit.

ATWS analysis was performed for the AP600 plant (Reference A-20). This analysis demonstrated that the AP600 plant could successfully ride out an ATWS event without inserting the control rods, considering that:

- Loss main feedwater is the most limiting initiating event
- PRHR HX provides an adequate heat sink
- The core reactivity feedback is sufficient to limit the peak RCS pressure to less than 3200 psig for more than 95 percent of full power core life

This analysis showed that the AP600 response to ATWS is comparable to existing Westinghouse PWRs.

The AP1000 employs a low-boron core. One of the benefits of such a core design is that the total reactivity feedback properties of the core, including moderator temperature coefficient, are more negative throughout core life than in conventional cores. As a result, as shown by the following plant analysis, the AP1000 has a zero unfavorable exposure time (UET) for equilibrium core cycles. For about 40 percent of the first core cycle the allowable maximum RCS pressure may be exceeded. As a bounding assumption, this would only result in a UET of 1.5 percent, assuming a plant life of 40 years. It would be even less for the plant design life of 60 years.

The AP1000 also includes automatic CMT actuation on the same DAS signal that starts the PRHR on low wide-range steam generator water level. If CMT operation is credited, the UET is zero even for the first core cycle. Additional ATWS analysis is provided in subsection A4.2.3, which demonstrates this performance.

A4.2 ATWS Analysis

ATWS analyses for the Westinghouse PWR plants have historically indicated Loss of Normal Feedwater as the most limiting ATWS event.

Specific analyses, performed for the AP600 plant, confirmed that this conclusion is also applicable to the passive plant designs.

The main reasons for this result are the following:

- Component sizing approach
  - Pressurizer and steam generator safety valve sizing
  - Pressurizer sizing

- Core and fuel characteristics
  - Fuel pitch
  - Moderator to fuel volume ratio
  - Moderator coefficient
  - Doppler coefficient

- Steam generator design
  - Vertical U-tube steam generator
  - Significant secondary side water inventory

The same conclusions can also be drawn for the AP1000 that compares closely to the AP600 plant.

In particular, the following considerations apply:

- Pressurizer and steam generator safety valves are sized to be able to relief the steam resulting from a full-power mismatch. This means that, even considering the worst loss of load transient, pressurizer pressure cannot exceed Condition II acceptance criteria (110 percent of the design pressure).

- The pressurizer in the AP600 and AP1000 is about 50 percent larger than those in traditional PWRs with the same power rating. Hence, a larger coolant swelling (that is, larger temperature changes) must occur to overfill the pressurizer. This, in turn, results in a larger negative reactivity insertion from the moderator reactivity feedback at the time of overfilling.

- Steam generators are essentially sized according to the same rules. Water inventory in the steam generators is sufficient for about 80 to 90 full power seconds both in the AP1000 and AP600, and in addition, following a loss of normal feedwater event, there are roughly the same full power seconds from normal steam generator water level to the reactor trip setpoint on low steam generator level.

- Kinetics parameters are close enough since, apart for the different fuel length, the same fuel is used for the two plants. In addition, the AP1000 is characterized by a low boron core design that provides significant benefits in terms of moderator reactivity coefficient.

As was

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done for the AP600 ATWS analysis (Reference A-20), the LOFTRAN computer code is used to perform these analysis. The AP600 ATWS analysis (Reference A-20) determined that the most limiting initiating event for an ATWS is a complete loss of normal feedwater.

Analysis has been performed for the AP1000 plant to verify that the peak RCS pressure is less than the ASME emergency stress limits, which occurs at greater than 3200 psia. In these analyses, the control rods are not inserted, even though DAS automatically de-energizes the motor generator set power. All of the mitigating system actions are modeled as being actuated by the DAS. DAS uses a low wide range SG level signal to actuate the following:

- Automatic trip of the turbine
- Automatic trip of the reactor coolant pumps
- Automatic trip of the CMTs
- Automatic start of PRHR HX

Both of the pressurizer safety valves are assumed open when the pressure exceeds their setpoint.

A4.2.1 Equilibrium Core ATWS Analysis

For the equilibrium core cycle analysis, the limiting moderator temperature coefficient (MTC) was used. The least negative MTC is -12.5 pcm/°F, which occurs at a core burnup of about 2.0 GWD/MTU. At any other time in an equilibrium core cycle the MTU is more negative. This MTC is evaluated at hot, full power conditions with no xenon in the core.

The sequence of events is as follows:

ATWS for Equilibrium Core Cycle	
Time (sec)	**Event**
0 to 4	All feedwater flow to SGs is lost
62.6	DAS low wide range SG level is reached
66.6	Turbine is tripped on DAS signal
70.0	Pzr SV open
72.6	PRHR HX is actuated on DAS signal
118	Max RCS pressure reached
178	Pzr SV re-close

The results for this analysis are shown in Figures A4.2.1-1 through A4.2.1-4.

As seen in Figure A4.2.1-1, the peak RCS pressure is about 3000 psia. This provides margin to the pressure limit of 3200 psig.

A4.2.2 First Core ATWS Analysis

For the first core cycle analysis, the MTC used was based on the limiting the peak RCS pressure to less than 3200 psig. An MTC is -10.0 pcm/°F results in a peak RCS pressure of

about 3200 psig (Figure A4.2.2-1). The MTC during the first core cycle is less than this value about 60 percent of the time; the UET for the first cycle is then 40 percent. As discussed in Section A4.1, the overall UET over a 40-year plant life would be 1.5 percent. Section A4.1 also discusses several significant conservatisms in this analysis, that if they were replaced with more realistic methods would reduce the overall UET to essentially zero.

The sequence of events is as follows:

ATWS for First Core Cycle	
Time (sec)	Event
0 to 4	All feedwater flow to the SGs is lost
57.9	DAS low wide range SG level is reached
61.9	Turbine is tripped on DAS signal
66.0	Pzr SV open
67.9	PRHR HX is actuated on DAS signal
121	Max RCS pressure reached
182	Pzr SV re-close

The results for this analysis are shown in Figures A4.2.2-1 through A4.2.2-4.

A4.2.3 ATWS Analysis with PRHR Operation

The core MTC is –5.0 pcm/°F, which occurs at the beginning of the first core cycle. At any other time in the first core cycle or in an equilibrium core cycle, the MTC will be more negative.

This case assumes operation of the PRHR heat exchanger and CMTs; the PRHR heat exchanger is assumed to be available. Only one CMT is assumed to be operable.

The sequence of events is as follows:

ATWS with PRHR Operable	
Time (sec)	Event
0 to 4	All feedwater flow to SGs is lost
50	PZR SV open
62.1	Turbine bypass fully open
64.1	Turbine is tripped on DAS signal
66.1	RCP trip on DAS signal
74.1	PRHR HX is actuated on DAS signal
74.1	CMTs are actuated on DAS signal
141	Max. RCS pressure reached
230	PZR SV reclose

The results for this analysis are shown in Figures A4.2.3-1 through A4.2.3-4.

As seen in Figure A4.2.3-1, the peak RCS pressure is 2818 psia. This provides margin to the pressure limit of 3200 psig.

A4.2.4 ATWS Analysis with SFW Operation

The core MTC is –5.0 pcm/°F, which occurs at the beginning of the first core cycle. At any other time in the first core cycle or in an equilibrium core cycle, the MTC will be more negative.

This case assumes operation of the SFW pumps and CMTs; the PRHR HX is assumed not to be available. One SFW pump is feeding both steam generators, and one CMT is operable.

The sequence of events is as follows:

ATWS Analysis with SFW Operation	
Time (sec)	Event
0 to 4	All feedwater flow to the SGs is lost
62.1	DAS low wide-range SG level is reached
64.1	Turbine is tripped on DAS signal
50.0	PZR SV open
66.1	RCPs trip on DAS signal
74.1	CMT is actuated on DAS signal
141	Max. RCS pressure reached
363	PZR SV reclose

The results for this analysis are shown in Figures A4.2.4-1 through A4.2.4-4.

A5 Thermal-Hydraulic Uncertainty Evaluation

A5.1 Approach

A5.1.1 Objective

The objective of this evaluation is to determine the low thermal margin, risk-important sequences in the AP1000 PRA. These sequences will then be used to define a set of cases that will be analyzed to bound the T/H uncertainty. Analyzing these cases using DCD methods is considered sufficient to bound the T/H uncertainty. The approach used in this evaluation is the same used for the AP600 (Reference A-4), which was accepted by the Nuclear Regulatory Commission (Reference A5-14).

A5.1.2 Evaluation Process

The method developed for the AP600 is used for the AP1000 in identifying the dominant success paths that should be examined for T/H uncertainty. This process includes the following steps:

1. The event tree success paths developed for the AP1000 PRA are "expanded" to quantify the probability of sequences with fewer failures than are assumed in the success criteria. Note that success paths using nonsafety-related features are not included in these expanded event trees.

2. The event trees that should be "expanded" are identified based on insights from the AP1000 success criteria analysis and the AP600 T/H uncertainty evaluation.

3. The expanded event tree end states are binned into categories that distinguish the accident progression. Two basic groups of end states are used: OK and UC categories. The OK category includes sequences that do not have low thermal margin. The UC category includes sequences that have low thermal margin. These two basic groups of end states are further divided into subcategories that include similar equipment availabilities (refer to Table A5.1-1). The "OK" and "UC" end-state category definitions used in this evaluation are the same as was used for the AP600.

4. The frequencies of success paths with UC end states are quantified, as shown in Figures A5.1-1 through A5.1-5.

5. The success paths with UC end states are sorted by their possible contributions to core damage frequency (CDF) and large release frequency (LRF), as shown in Table A5.1-2.

6. The risk-important sequences are identified using the acceptance criteria defined in subsection A5.1.4. These sequences, shown in Table A5.1-3, are subject to further examination to bound their T/H uncertainty.

7. The contribution of the "residue" – namely, the total CDF and LRF contribution of those UC success paths not selected as dominant – is calculated and monitored to make sure that this contribution is relatively small.

8. A smaller number of more limiting cases are selected for T/H analysis to minimize the number of analyses needed. These cases, shown in Table A5.1-6, have additional failures, although not as many as the success criteria. They are analyzed using the detailed DCD T/H computer codes and methods to show adequate core cooling. This analysis demonstrates that the T/H uncertainty has been bounded.

A5.1.3 Scope of Expanded Event Trees

In the AP600 T/H uncertainty analysis, 10 expanded event trees were developed. These expanded event trees included:

- Large Loss-of-Coolant Accident (LLOCA)
- Medium Loss-of-Coolant Accident (MLOCA)
- Core Makeup Tank (CMT) Line Break
- Direct Vessel Injection (DVI) Line Break
- Intermediate Loss-of-Coolant Accident (LOCA)
- Small LOCA with passive residual heat removal heat exchanger (PRHR HX)
- Small LOCA without PRHR HX
- Steam generator tube ruptures (SGTRs) with PRHR HX and Automatic Depressurization System (ADS) operation
- SGTRs without PRHR HX and with ADS operation
- Transients with ADS operation

Based on the results of the AP600 quantification of these expanded event trees, the Small LOCA, SGTR, and Transient trees did not affect the selection of cases analyzed for T/H uncertainty. There were two reasons for this outcome:

- First, only 5 of the 24 sequences identified as dominant accident sequences were from these event trees. Note that three of these five sequences were included in subcategory UC6, which does not apply to the AP1000 since its defining equipment availability was fewer ADS Stage 4 valves (two of four) than assumed in the PRA success criteria. Since the minimum AP1000 success criteria is three of four ADS Stage 4 valves, this subcategory does not apply to AP1000.

- Second, these sequences were less limiting from a T/H perspective because they had more equipment available and ADS occurred later with lower decay heat.

Based on the work done in Reference A-4, the Westinghouse designer-PRA team has obtained insights about which success paths in which initiating event categories are candidates for potential T/H uncertainty evaluation. Since the AP1000 design is based on the AP600 design, and the AP1000 retained the same passive safety systems and their configurations, it is reasonable that the insights obtained from the AP600 T/H uncertainty work are applicable to AP1000.

Further work has been done and documented in this report to provide analytical support for this assertion, for justification of dominant success paths studied for the AP1000 T/H analysis.

Based on these considerations, expanded event trees were not developed for AP1000 small LOCAs, SGTRs, or transients.

The following event trees were expanded for the AP1000:

- Large LOCA (LLOCA)
- Spurious ADS Actuation (SPADS)
- Medium LOCA (MLOCA)
- Core Makeup Tank Line Break (CMTLB)
- Safety Injection Line Break (SI-LB) (DVI Line Break)

Note that there are two differences between this list and the corresponding AP600 events:

- The AP1000 has two large LOCA categories (LLOCA and Spurious ADS Actuation). Whereas, the AP600 has only one large LOCA category.

- The AP600 medium LOCA and intermediate LOCA categories are combined into a single category in the AP1000 PRA, which bounds the same LOCA size range.

The expanded event trees for the AP1000 PRA event trees are developed and quantified in Section A5.2. The dominant success paths subject to further investigation are identified in Section A5.3, based on their percentage contribution to either CDF or LRF. The cases to be evaluated for T/H uncertainty are identified in Section A5.4.

A5.2.7 Calculation of Success Path Frequencies

The frequencies of the success paths classified with a UC end state are quantified using the system models already developed for the AP1000 PRA. The results are shown in Figures A5.1-1 through A5.1-5. The same modeling assumptions as in the AP600 T/H uncertainty analysis are used to provide the percentage contribution of each of these sequences to plant CDF and LRF if these sequences were assumed to be core damage. The dominant sequences are collected for further evaluation in Section A5.3.

A5.3 Risk-Important Success Paths

A5.3.1 Sorted Success Paths

In Section A5.2, the frequencies of success paths with UC end states are calculated, as shown in Figures A5.1-1 to A5.1-5. These paths are collected and sorted by their frequencies. The path frequency is tentatively assumed to be core damage to allow comparison with the base case CDF. The resulting list is shown in Table A5.1-2.

Next, the risk-important success paths are identified. For this purpose, the acceptance criteria in subsection A5.1.4 are used. The risk-important success paths are shown in Table A5.1-2 with their CMF and/or LRF percentage numbers shown in bold letters and outlined; in addition, these sequences are marked with a ">>" in the left column. There are 13 sequences identified as being risk-important. This table also shows which sequences are bounded by the short-term and long-term analysis cases in Table A5.1-6.

Table A5.1-3 lists these 13 risk-important sequences; these sequences are subject to further examination for thermal-hydraulic uncertainty considerations. Note that each of these paths has adequate ADS actuation, and at least one CMT or one accumulator injecting. Not all of these paths need to be low-margin cases from a T/H analysis point of view; see Section A5.4 for further discussion of these sequences.

The residual contribution of the remaining UC sequences are approximately 4 to 5 percent of CDF or LRF. This percentage is too small for modeling uncertainty considerations. Thus, the residual sequences need not be further examined.

A5.3.2 Role of Passive Residual Heat Removal

In the MLOCA (MLOCA, SI-LB, and CMTLB) success paths where both CMTs fail to inject, automatic actuation of PRHR is required to provide adequate time for the operators to actuate the ADS manually. In Table A5.1-3, six sequences had both CMTs failed. It is expected that almost all of the frequency of one of these sequences is attributable to cases where PRHR is available. To show this, the frequencies of these sequences is recalculated assuming that PRHR failed. These success paths are labeled with the additional letter "p". The results are shown in Table A5.1-4.

From Table A5.1-4, it is observed that postulating the additional failure of PRHR in the risk-important sequences with both CMTs inoperable results in a small CDF and LRF. Thus, the corresponding sequences in Table A5.1-4 will be labeled as PRHR available; the results

are shown in Table A5.1-5. These are the risk-important sequences further discussed for T/H uncertainty considerations in Section A5-4. Table A5.1-5 shows which analysis case in Table A5.1-6 bounds each of the risk important sequences.

A5.4 Cases for Thermal/Hydraulic Uncertainty Analysis

In the previous section, 13 risk-important UC sequences are identified for further evaluation. These sequences are listed in Table A5.1-5. These sequences have the following general characteristics:

- All of these sequences have four of four ADS stage 4 valves open and several of the ADS stage 2/3 valves open.
- Eight of the sequences have containment isolation successful, and five have containment isolation failure.
- Six of the thirteen sequences have only one tank (CMT or accumulator) available; the remaining sequences have two to four tanks (both accumulator and CMT).
- The sequences with no CMTs have the PRHR HX available.

Five short-term and two long-term cases have been selected to be analyzed to bound the T/H uncertainty of the 13 cases in Table A5.1-5. Table A5.1-6 lists these seven AP1000 cases. These cases were selected to minimize the total number of cases required to be analyzed while ensuring that the T/H uncertainty of the cases in Table A5.1-5 are bounded. As shown in Table A5.1-2, the cases in Table A5.1-6 bound more cases than those in Table A5.1-5.

For each of these cases, Table A5.1-6 indicates the initiating event, the available equipment, and which risk-important case(s) that it bounds in Table A5.1-5.

A5.4.1 Description of Thermal/Hydraulic Uncertainty Analysis Cases

Short-Term Case A, Reactor Coolant System Hot-Leg Medium LOCA (3.0")

This case bounds risk-important sequences 3, 10, 12, and 13 in Table A5.1-5. The specific case was selected because MAAP success criteria analysis indicated potential core uncovery with a LOCA break size that results in the accumulator just starting to inject at the time the operators were manually actuating ADS. The operator action time in the PRA for this sequence is 20 minutes. Since the CMTs are assumed to have failed and the reactor coolant system pressure remains above the accumulator pressure, there is no RCS injection until the operators actuate ADS. This sequence can lead to core uncovery before ADS actuation. Note that the PRHR HX is included because it is required by the AP1000 success criteria for MLOCAs with failure of CMTs.

Short-Term Case B, Double-Ended Core Makeup Tank Balance Line LOCA

This case bounds risk-important sequences 4 and 5 in Table A5.1-5. The specific case was selected because MAAP success criteria analysis indicated potential core uncovery with a

LOCA break size that results in rapid accumulator injection such that the accumulator is mostly empty by the time the operators manually actuate ADS. Since the CMTs are assumed to have failed, there will be essentially no RCS injection during the depressurization to IRWST injection. The PRHR HX is included because the AP1000 success criteria require the PRHR HX for MLOCAs with failure of CMTs.

Short-Term Case C, Double-Ended Direct Vessel Injection LOCA

This case bounds risk-important sequences 1, 7, 9, and 11 in Table A5.1-5. The specific case was selected because MAAP success criteria analysis indicated potential core uncovery with a DVI break and injection from only one CMT. Since the accumulators are assumed to have failed, the injection will be reduced in the time frame just after ADS actuation.

Short-Term Case D, Large Cold-Leg LOCA

This case bounds risk-important sequence 8 in Table A5.1-5. The specific case was selected because large LOCAs all have significant core uncovery and the design basis DCD analysis (Reference A-26) has successful containment isolation. In the PRA, successful core cooling can be accomplished even with failure of containment isolation. The case analyzed for the PRA also assumes that offsite power remains available until the reactor coolant pumps are tripped when the CMTs are actuated several seconds into the event. If offsite power is also assumed to be lost, this sequence would not be risk-important. As discussed in the AP1000 DCD, having offsite power available results in somewhat lower PCTs for the AP1000.

Short-Term Case E, Spurious ADS Stage 4 Large LOCA

This case bounds risk-important sequences 2 and 6 in Table A5.1-5. The specific case was selected because large LOCAs have significant core uncovery, and the design basis DCD analysis (Reference A-26) is a cold-leg break location, has injection from two accumulators, and has successful containment isolation. In the PRA, only one accumulator is considered to be required for these hot-leg LOCAs. In addition, successful core cooling can be accomplished even with failure of containment isolation. Note that the case analyzed has successful containment isolation. Analysis of the same case with failure of containment isolation is considered unnecessary because:

- There is a large margin between the PCT calculated for this case compared with the PCT limit (2200°F – 1061°F = 1139°F margin with uncertainties).
- The effect of a lower containment pressure on a hot-leg break will be smaller than for a cold-leg break because the steam/water mixture leaving the core does not have to go through the steam generator to get to the break. Bypassing the steam generator eliminates the steam binding effect that occurs in large cold-leg LOCAs.

Long-Term Case F, Double-Ended Direct Vessel Injection LOCA

This case bounds risk-important sequences 1 through 5, 7, 9, and 10 in Table A5.1-5. The specific case was selected because DVI LOCAs have lower containment water levels/driving heads available during recirculation operation. This accident is the same as Case C except

that containment isolation is available. Note that this case is a transient case that covers operation from ADS stage 4 opening through initiation of containment recirculation.

Long-Term Case G, Double-Ended Direct Vessel Injection LOCA

This case bounds risk-important sequences 6, 8, and 11 through 13 in Table A5.1-5. The specific case was selected because DVI LOCAs have lower containment water levels/driving

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heads available during recirculation operation. This accident is the same as Case C except that four of four ADS stage 4 valves open. Note that this case is a transient case that covers operation from ADS stage 4 opening through initiation of containment recirculation. The case continues until leakage from the containment is terminated when the passive containment cooling system is able to remove decay heat with the containment pressure at atmospheric pressure.

The analysis results of these 7 cases are reported in Sections A5.2 and A5.3.

A5.5 T/H Uncertainty Analyses for Short-Term Cooling

Section A5.5 identifies the thermal/hydraulic analyses that are performed to support the low-margin, PRA-important accident scenarios. The scope of these analyses is short-term cooling, from the initiation of the event until IRWST gravity injection is established. The analysis methodology is consistent with design basis methods, codes, and assumptions. The conservative assumptions used in the analyses bound the T/H uncertainties identified in Reference A-4, Section 2, providing a robust basis for the success criteria that have been credited in the AP1000 PRA. Section A5.5.1 documents the small LOCA analyses performed with the NOTRUMP and LOCTA codes. Section A5.5.2 documents the long term cooling analyses performed with the WCOBRA/TRAC code. Details of the analysis methodologies used are provided within each subsection.

A5.5.1 NOTRUMP/LOCTA Analyses of Small LOCAs

The potentially risk-significant accident scenarios identified in Section A5.5 were analyzed using the design basis NOTRUMP small LOCA analysis code and the LOCTA cladding heat-up code. Assumptions from Appendix K to 10CFR50 were used in both the NOTRUMP and LOCTA code calculations. Sections A5.5.1.1 and A5.5.1.2 identify the analysis methodology, and Section A5.5.1.3 provides analysis results.

A5.5.1.1 NOTRUMP Analysis Methodology

The methodology presented in DCD Section 15.6.5.4B for the application of NOTRUMP to the AP1000 design was used in T/H uncertainty analyses with some exceptions, as follows:

1. Equipment failures/assumptions were based on the potential for risk significance for the AP1000, as defined in Chapter 6.

    a. The PRHR was modeled in the T/H uncertainty analyses for cases with no CMTs.
    b. More than one failure was considered in the T/H uncertainty analyses.

2. The break discharge coefficient was assumed to be 1.0.

3. Credit for containment isolation was modeled in two of the three cases. A containment backpressure of 25 psia was used in the containment isolation cases. This is the same value used in the DCD small-break LOCA long-term cooling analyses (Section 15.6.5.4C). When applied to short-term cooling, this pressure is conservatively

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

Initially, the only injection comes from the IRWST into the reactor vessel through the intact DVI injection line (Figure A5.3-14). Since at the beginning of the analysis, the level in the PXS-B room is below the DVI injection nozzle elevation, only steam from the downcomer is vented out through the break (Figure A5.3-13). Water starts to flow back into the downcomer through the broken DVI line about 2 hours into the transient. This is the time when the level in the PXS-B room becomes high enough to provide sufficient driving head. At the onset of this event, the additional amount of water supplied into the downcomer through the DVI break supplements the IRWST injection. This leads to enhanced core cooling, and momentarily, faster depressurization occurs at about 2.05 hours into the transient (Figure A5.3-11). Consequently, the IRWST injection is increased even further, and as a result, the levels in the downcomer (Figure A5.3-1), the reactor core (Figure A5.3-2), and the upper plenum (Figure A5.3-8) are also increased. The effect of this injection flow increase can also be seen on Figure A5.3-4, which shows a sharp void fraction decrease in the upper half of the fuel region.

The available three out of four ADS Stage 4 valves provide enough venting capacity to assure adequate depressurization and successful performance of the passive safety systems (Figures A5.3-9 and A5.3-10). The fuel remains covered throughout the transient and adequate core cooling is provided to remove the decay heat. The hot rod cladding temperature is about 20°F above saturation (Figure A5.3-12) and is steadily decreasing.

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

A5.6.2.2 Case G – DEDVI Line Break in the PXS-B Room with Four of Four ADS Stage 4, Containment Isolation Failed

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

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

boundary conditions taken from the MAAP4 calculation, but with the adjusted IRWST level decrease.

Initially, the only injection comes from the IRWST into the reactor vessel through the intact DVI injection line (Figure A5.3-28). Since at the beginning of the analysis, the level in the PXS-B room is below the DVI injection nozzle elevation, only steam from the downcomer is vented out through the break. Water starts to flow back into the downcomer through the broken DVI line about 2 hours into the transient. This is the time when the level in the PXS-B room becomes high enough to provide sufficient driving head for this to happen. This time, unlike the Case F DVI break scenario, the transition into reversed injection of water through the break into the downcomer occurs a little earlier, and is somewhat softer. As a result, the increased depressurization rate observed in Case F does not occur. Still, the levels in the downcomer (Figure A5.3-15), the reactor core (Figure A5.3-16) and the upper plenum (Figure A5.3-22) are maintained high enough by the available DVI injection.

The availability of all ADS Stage 4 valves provides enough venting capacity to assure adequate depressurization and successful performance of the passive safety systems (Figures A5.3-23 and A5.3-24). The fuel remains covered throughout the transient, and adequate core cooling is provided to remove the decay heat. The hot rod cladding temperature is about 20°F above saturation (Figure A5.3-26) and steadily decreasing.

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

A6 References

A-1 AP600 Standard Safety Analysis Report

A-2 AP600 Probabilistic Risk Assessment

A-3 MAAP4/NOTRUMP Benchmarking to Support the Use of MAAP4 for AP600 PRA Success Criteria Analysis, WCAP-14869, April 1997

A-4 AP600 PRA Thermal/Hydraulic Uncertainty Evaluation for Passive System Reliability, WCAP-14800, June 1997

A-5 AP600 Adverse Systems Interaction Report, WCAP-14477

A-6 AP600 Shutdown Report, WCAP-14837

A-7 "Evaluation of the AP600 Conformance to Inter-system LOCA Acceptance Criteria," WCAP-14425, July 1995

A-8 "Operational Assessment for AP1000," WCAP-15800

A-9 AP600 Human Factors Engineering Operational Experience Review Report, WCAP-14645

Table A5.1-2 (Sheet 1 of 6)

LOW MARGIN (UC) SEQUENCES SORTED BY CORE DAMAGE FREQUENCY

	End State	Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2,3	Bounded By Short/Long-Term		
>>	UC5	silb06	8.96E-07	5.37E-08	371.66	275.60	Yes	Yes	1	0	4	2-4	C	FG	
>>	UC4	sad06	4.58E-07	2.75E-08	190.05	140.93	Yes	Yes	2	1	4	2-4	E	FG	
>>	UC1	silb11	3.05E-07	1.83E-08	126.76	94.00	Yes	Yes	0	1	4	2-4	A	FG	
>>	UC2B	mlo31	2.89E-07	1.73E-08	119.85	88.88	Yes	Yes	0	2	4	2-4	AB	FG	
>>	UC2B	cmt31	1.34E-07	8.05E-09	55.67	41.28	Yes	Yes	0	2	4	2-4	AB	FG	
>>	UC8	sad25	9.12E-08	5.47E-09	37.82	28.05	No	Yes	2	2	4	2-4	E	G	(2)
>>	UC3	mlo11	3.01E-08	1.81E-09	12.48	9.26	Yes	Yes	2	0	4	2-4	C	FG	
>>	UC8	llo15	8.51E-09	8.51E-09	3.53	43.63	No	Yes	2	2	4	2-4	D	G	
>>	UC3	cmt26	6.42E-09	3.85E-10	2.67	1.98	Yes	Yes	1	0	4	2-4	C	FG	
>>	UC2A	mlo36	2.44E-09	1.47E-10	1.01	0.75	Yes	Yes	0	1	4	2-4	A	FG	
	UC5	silb08	2.09E-09	1.25E-10	0.87	0.64	Yes	Yes	1	0	3	2-4	C	F	
	UC5	silb07	1.64E-09	9.83E-11	0.68	0.50	Yes	Yes	1	0	4	0-1	C	FG	
>>	UC5	silb23	1.52E-09	1.52E-09	0.63	7.77	No	Yes	1	0	4	2-4	C	G	
	UC2A	cmt36	1.14E-09	6.85E-11	0.47	0.35	Yes	Yes	0	1	4	2-4	A	FG	
	UC4	sad08	1.07E-09	6.42E-11	0.44	0.33	Yes	Yes	2	1	3	2-4		F	
	UC4	sad07	8.40E-10	5.04E-11	0.35	0.26	Yes	Yes	2	1	4	0-1	E	FG	
	UC4	sad30	7.77E-10	4.66E-11	0.32	0.24	No	Yes	2	1	4	2-4	E	G	(2)
	UC1	silb13	7.21E-10	4.32E-11	0.30	0.22	Yes	Yes	0	1	3	2-4		F	
	UC2B	mlo33	6.92E-10	4.15E-11	0.29	0.21	Yes	Yes	0	2	3	2-4		F	
	UC4	sad17	6.76E-10	4.05E-11	0.28	0.21	Yes	Yes	1	1	4	2-4	E	FG	

Table A5.1-2 (Sheet 2 of 6)

LOW MARGIN (UC) SEQUENCES SORTED BY CORE DAMAGE FREQUENCY

	End State	Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2,3	Bounded By Short/Long-Term		
	UC2B	mlo32	6.44E-10	3.86E-11	0.27	0.20	Yes	Yes	0	2	4	0-1	AB	FG	
	UC1	silb12	6.15E-10	3.69E-11	0.26	0.19	Yes	Yes	0	1	4	0-1	A	FG	
>>	UC1	silb28	5.16E-10	5.16E-10	0.21	2.65	No	Yes	0	1	4	2-4	A	G	
>>	UC2B	mlo73	4.88E-10	4.88E-10	0.20	2.50	No	Yes	0	2	4	2-4	A	G	
	UC2B	cmt33	3.17E-10	1.90E-11	0.13	0.10	Yes	Yes	0	2	3	2-4		F	
	UC2B	cmt32	2.70E-10	1.62E-11	0.11	0.08	Yes	Yes	0	2	4	0-1	AB	FG	
	UC2B	cmt73	2.27E-10	1.36E-11	0.09	0.07	No	Yes	0	2	4	2-4	A	G	
	UC8	sad27	2.13E-10	1.28E-11	0.09	0.07	No	Yes	2	2	3	2-4			
	UC8	sad26	1.67E-10	1.00E-11	0.07	0.05	No	Yes	2	2	4	0-1	E	G	(2)
	UC8	sad36	1.34E-10	8.04E-12	0.06	0.04	No	Yes	1	2	4	2-4	E	G	(2)
	UC3	mlo13	7.00E-11	4.20E-12	0.03	0.02	Yes	Yes	2	0	3	2-4	C	F	
	UC3	mlo12	5.49E-11	3.30E-12	0.02	0.02	Yes	Yes	2	0	4	0-1	C	FG	
	UC3	mlo53	5.08E-11	5.08E-11	0.02	0.26	No	Yes	2	0	4	2-4	C	G	
	UC3	mlo26	4.42E-11	2.65E-12	0.02	0.01	Yes	Yes	1	0	4	2-4	C	FG	
	UC8	llo17	1.99E-11	1.99E-11	0.01	0.10	No	Yes	2	2	3	2-4			
	UC8	llo16	1.56E-11	1.56E-11	0.01	0.08	No	Yes	2	2	4	0-1	D	G	
	UC3	cmt28	1.50E-11	8.97E-13	0.01	0.00	Yes	Yes	1	0	3	2-4	C	F	
	UC8	llo21	1.25E-11	1.25E-11	0.01	0.06	No	Yes	1	2	4	2-4		G	
	UC3	cmt27	1.17E-11	7.04E-13	0.00	0.00	Yes	Yes	1	0	4	0-1	C	FG	
	UC3	cmt68	1.09E-11	6.51E-13	0.00	0.00	No	Yes	1	0	4	2-4	C	G	
	UC2A	mlo38	5.86E-12	3.51E-13	0.00	0.00	Yes	Yes	0	1	3	2-4		F	

Table A5.1-2 (Sheet 3 of 6)

LOW MARGIN (UC) SEQUENCES SORTED BY CORE DAMAGE FREQUENCY

End State	Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2,3	Bounded Short/Long-Term		
UC2A	mlo37	5.45E-12	3.27E-13	0.00	0.00	Yes	Yes	0	1	4	0-1	A	FG	
UC2A	mlo78	4.13E-12	4.13E-12	0.00	0.02	No	Yes	0	1	4	2-4	A	G	
UC5	silb09	3.85E-12	2.31E-13	0.00	0.00	Yes	Yes	1	0	3	0-1	C	F	
UC5	silb25	3.53E-12	3.53E-12	0.00	0.02	No	Yes	1	0	3	2-4	C		
UC5	silb24	2.77E-12	2.77E-12	0.00	0.01	No	Yes	1	0	4	0-1	C	G	
UC2A	cmt38	2.69E-12	1.62E-13	0.00	0.00	Yes	Yes	0	1	3	2-4		F	
UC2A	cmt37	2.30E-12	1.38E-13	0.00	0.00	Yes	Yes	0	1	4	0-1	A	FG	
UC4	sad09	1.97E-12	1.18E-13	0.00	0.00	Yes	Yes	2	1	3	0-1		F	
UC2A	cmt78	1.93E-12	1.16E-13	0.00	0.00	No	Yes	0	1	4	2-4	A	G	
UC4	sad32	1.81E-12	1.09E-13	0.00	0.00	No	Yes	2	1	3	2-4			
UC2B	mlo34	1.78E-12	1.07E-13	0.00	0.00	Yes	Yes	0	2	3	0-1		F	
UC1	silb14	1.57E-12	9.45E-14	0.00	0.00	Yes	Yes	0	1	3	0-1		F	
UC4	sad19	1.57E-12	9.44E-14	0.00	0.00	Yes	Yes	1	1	3	2-4		F	
UC4	sad31	1.42E-12	8.51E-14	0.00	0.00	No	Yes	2	1	4	0-1	E	G	(2)
UC4	sad18	1.23E-12	7.40E-14	0.00	0.00	Yes	Yes	1	1	4	0-1	E	FG	
UC1	silb30	1.22E-12	1.22E-12	0.00	0.01	No	Yes	0	1	3	2-4			
UC2B	mlo75	1.17E-12	1.17E-12	0.00	0.01	No	Yes	0	2	3	2-4			
UC4	sad41	1.14E-12	6.85E-14	0.00	0.00	No	Yes	1	1	4	2-4	E	G	(2)
UC2B	mlo74	1.09E-12	1.09E-12	0.00	0.01	No	Yes	0	2	4	0-1	A	G	
UC1	silb29	1.04E-12	1.04E-12	0.00	0.01	No	Yes	0	1	4	0-1	A	G	
UC2B	cmt34	6.91E-13	4.15E-14	0.00	0.00	Yes	Yes	0	2	3	0-1		F	

Table A5.1-2 (Sheet 4 of 6)

LOW MARGIN (UC) SEQUENCES SORTED BY CORE DAMAGE FREQUENCY

End State	Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2,3	Bounded By Short/Long-Term		
UC2B	cmt75	5.35E-13	3.21E-14	0.00	0.00	No	Yes	0	2	3	2-4			
UC2B	cmt74	4.56E-13	2.74E-14	0.00	0.00	No	Yes	0	2	4	0-1	A	G	
UC8	sad28	3.91E-13	2.35E-14	0.00	0.00	No	Yes	2	2	3	0-1			
UC8	sad38	3.12E-13	1.87E-14	0.00	0.00	No	Yes	1	2	3	2-4			
UC8	sad37	2.45E-13	1.47E-14	0.00	0.00	No	Yes	1	2	4	0-1	E	G	(2)
UC3	mlo14	1.29E-13	7.71E-15	0.00	0.00	Yes	Yes	2	0	3	0-1	C	F	
UC3	mlo55	1.18E-13	1.18E-13	0.00	0.00	No	Yes	2	0	3	2-4	C		
UC3	mlo28	1.03E-13	6.17E-15	0.00	0.00	Yes	Yes	1	0	3	2-4	C	F	
UC3	mlo54	9.24E-14	9.24E-14	0.00	0.00	No	Yes	2	0	4	0-1	C	G	
UC3	mlo27	8.04E-14	4.83E-15	0.00	0.00	Yes	Yes	1	0	4	0-1	C	FG	
UC3	mlo68	7.46E-14	7.46E-14	0.00	0.00	No	Yes	1	0	4	2-4	C	G	
UC8	llo18	3.66E-14	3.66E-14	0.00	0.00	No	Yes	2	2	3	0-1			
UC8	llo23	2.92E-14	2.92E-14	0.00	0.00	No	Yes	1	2	3	2-4			
UC3	cmt29	2.75E-14	1.65E-15	0.00	0.00	Yes	Yes	1	0	3	0-1	C	F	
UC3	cmt70	2.52E-14	1.51E-15	0.00	0.00	No	Yes	1	0	3	2-4	C		
UC8	llo22	2.29E-14	2.29E-14	0.00	0.00	No	Yes	1	2	4	0-1		G	
UC3	cmt69	1.97E-14	1.18E-15	0.00	0.00	No	Yes	1	0	4	0-1	C	G	
UC2A	mlo39	1.36E-14	8.13E-16	0.00	0.00	Yes	Yes	0	1	3	0-1		F	
UC2A	mlo80	9.01E-15	9.01E-15	0.00	0.00	No	Yes	0	1	3	2-4			
UC2A	mlo79	8.10E-15	8.10E-15	0.00	0.00	No	Yes	0	1	4	0-1	A	G	
UC5	silb26	6.28E-15	6.28E-15	0.00	0.00	No	Yes	1	0	3	0-1	C		

Table A5.1-2 (Sheet 5 of 6)

LOW MARGIN (UC) SEQUENCES SORTED BY CORE DAMAGE FREQUENCY

End State	Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2,3	Bounded By Short/Long-Term		
UC2A	cmt39	5.47E-15	3.28E-16	0.00	0.00	Yes	Yes	0	1	3	0-1		F	
UC2A	cmt80	4.33E-15	2.60E-16	0.00	0.00	No	Yes	0	1	3	2-4			
UC2A	cmt79	3.58E-15	2.15E-16	0.00	0.00	No	Yes	0	1	4	0-1	A	G	
UC4	sad33	3.20E-15	1.92E-16	0.00	0.00	No	Yes	2	1	3	0-1			
UC4	sad20	2.78E-15	1.67E-16	0.00	0.00	Yes	Yes	1	1	3	0-1		F	
UC4	sad43	2.62E-15	1.57E-16	0.00	0.00	No	Yes	1	1	3	2-4			
UC2B	mlo76	2.30E-15	2.30E-15	0.00	0.00	No	Yes	0	2	3	0-1			
UC1	silb31	2.26E-15	2.26E-15	0.00	0.00	No	Yes	0	1	3	0-1			
UC4	sad42	2.02E-15	1.21E-16	0.00	0.00	No	Yes	1	1	4	0-1	E	G	(2)
UC2B	cmt76	9.92E-16	5.95E-17	0.00	0.00	No	Yes	0	2	3	0-1			
UC8	sad39	5.15E-16	3.09E-17	0.00	0.00	No	Yes	1	2	3	0-1			
UC3	mlo56	1.51E-16	1.51E-16	0.00	0.00	No	Yes	2	0	3	0-1	C		
UC3	mlo29	1.35E-16	8.09E-18	0.00	0.00	Yes	Yes	1	0	3	0-1	C	F	
UC3	mlo70	1.25E-16	1.25E-16	0.00	0.00	No	Yes	1	0	3	2-4	C		
UC3	mlo69	9.54E-17	9.54E-17	0.00	0.00	No	Yes	1	0	4	0-1	C	G	
UC8	llo24	5.09E-17	5.09E-17	0.00	0.00	No	Yes	1	2	3	0-1			
UC3	cmt71	3.23E-17	1.94E-18	0.00	0.00	No	Yes	1	0	3	0-1	C		
UC4	sad44	0.00E+00	0.00E+00	0.00	0.00	No	Yes	1	1	3	0-1			
UC2A	cmt81	0.00E+00	0.00E+00	0.00	0.00	No	Yes	0	1	3	0-1			

Table A5.1-2 (Sheet 6 of 6)

LOW MARGIN (UC) SEQUENCES SORTED BY CORE DAMAGE FREQUENCY

End State	Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2,3	Bounded By Short/Long Term
Totals/% of total		2.24E-06	1.45E-07	100.00	100.00	<< all 102 cases listed above						
		2.23E-06	1.44E-07	99.86	99.85	<< for 58 cases bounded by analyzed T/H uncertainty cases						
		2.22E-06	1.44E-07	99.44	99.40	<< for 13 cases bounded by dominant cases						

Notes:

1. Sequences with >= 1-percent contribution to either CDF or LRF are highlighted by outline of percentage CDF/LRF. The remaining sequences are to be treated as "residue."
2. Spurious ADS 4 case was analyzed with containment isolation; because of large PCT margin, case without CI is expected to be OK.

Table A5.1-3

RISK-IMPORTANT UC SEQUENCES

End State	Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2, 3
UC5	silb06	8.96E-07	5.37E-08	371.66	275.60	Yes	Yes	1	0	4	2-4
UC4	sad06	4.58E-07	2.75E-08	190.05	140.93	Yes	Yes	2	1	4	2-4
UC1	silb11	3.05E-07	1.83E-08	126.76	94.00	Yes	Yes	0	1	4	2-4
UC2B	mlo31	2.89E-07	1.73E-08	119.85	88.88	Yes	Yes	0	2	4	2-4
UC2B	cmt31	1.34E-07	8.05E-09	55.67	41.28	Yes	Yes	0	2	4	2-4
UC8	sad25	9.12E-08	5.47E-09	37.82	28.05	No	Yes	2	2	4	2-4
UC3	mlo11	3.01E-08	1.81E-09	12.48	9.26	Yes	Yes	2	0	4	2-4
UC8	llo15	8.51E-09	8.51E-09	3.53	43.63	No	Yes	2	2	4	2-4
UC3	cmt26	6.42E-09	3.85E-10	2.67	1.98	Yes	Yes	1	0	4	2-4
UC2A	mlo36	2.44E-09	1.47E-10	1.01	0.75	Yes	Yes	0	1	4	2-4
UC5	silb23	1.52E-09	1.52E-09	0.63	7.77	No	Yes	1	0	4	2-4
UC1	silb28	5.16E-10	5.16E-10	0.21	2.65	No	Yes	0	1	4	2-4
UC2B	mlo73	4.88E-10	4.88E-10	0.20	2.50	No	Yes	0	2	4	2-4
Totals =		2.22E-06	1.44E-07								
		1.26E-08	8.62E-10	5.22	4.42	=	Residue from UC Sequences not selected				

Table A5.1-4

SEQUENCES WITH CORE MAKEUP TANK AND PASSIVE RESIDUAL HEAT REMOVAL FAILURE

Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	PRHR
silb11p	6.54E-11	3.92E-12	0.03	0.02	Yes	Yes	0	No
silb28p	1.12E-13	1.12E-13	0.00	0.00	No	Yes	0	No
mlo31p	6.29E-11	3.77E-12	0.03	0.02	Yes	Yes	0	No
mlo36p	5.31E-13	3.19E-14	0.00	0.00	Yes	Yes	0	No
mlo73p	1.07E-13	1.07E-13	0.00	0.00	No	Yes	0	No
cmt31p	2.87E-11	1.72E-12	0.01	0.01	Yes	Yes	0	No
Sum =	1.58E-10	9.67E-12	0.07	0.05				

Table A5.1-5

AP1000 THERMAL/HYDRAULIC UNCERTAINTY LOW MARGIN/RISK-IMPORTANT SEQUENCES

Case	Initiating Event	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2/3	PRHR	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	Short-Term	Long-Term
1	SILB	Yes	Yes	1	0	4	2-4	N/A	8.96E-07	5.37E-08	371.7	275.6	C	F
2	SADS	Yes	Yes	2	1	4	2-4	N/A	4.58E-07	2.75E-08	190.1	140.9	E	F
3	SILB	Yes	Yes	0	1	4	2-4	Yes	3.05E-07	1.83E-08	126.8	94.0	A	F
4	MLOCA	Yes	Yes	0	2	4	2-4	Yes	2.89E-07	1.73E-08	119.9	88.9	B	F
5	CMT	Yes	Yes	0	2	4	2-4	Yes	1.34E-07	8.05E-09	55.7	41.3	B	F
6	SADS	No	Yes	2	2	4	2-4	N/A	9.12E-08	5.47E-09	37.8	28.0	E	G
7	MLOCA	Yes	Yes	2	0	4	2-4	N/A	3.01E-08	1.81E-09	12.5	9.3	C	F
8	LLOCA	No	Yes	2	2	4	2-4	N/A	8.51E-09	8.51E-09	3.5	43.6	D	G
9	CMT	Yes	Yes	1	0	4	2-4	N/A	6.42E-09	3.85E-10	2.7	2.0	C	F
10	MLOCA	Yes	Yes	0	1	4	2-4	Yes	2.44E-09	1.47E-10	1.0	0.8	A	F
11	SILB	No	Yes	1	0	4	2-4	N/A	1.52E-09	1.52E-09	0.6	7.8	C	G
12	SILB	No	Yes	0	1	4	2-4	Yes	5.16E-10	5.16E-10	0.2	2.6	A	G
13	MLOCA	No	Yes	0	2	4	2-4	Yes	4.88E-10	4.88E-10	0.2	2.5	A	G
								Totals =	2.22E-06	1.44E-07				
				Residue from UC Sequences not selected					1.26E-08	8.62E-10	5.2	4.4		
				Residue from sequences with PRHR failure					1.58E-10	9.67E-12	0.1	0.0		

Table A5.1-6

LOW MARGIN (UC) SEQUENCES ANALYZED FOR THERMAL/HYDRAULIC UNCERTAINTY EVALUATION

Analysis Case	Initiating Event[1]	Cont. Isol.	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2/3	PRHR	Bounds Dominant Case
Short-Term Cooling									
A	Reactor coolant system hot leg (3.0")	No	Yes	0	1	4	0	Yes	3,10,12,13
B	Double-ended CMT balance line (6.8")	Yes	Yes	0	2	4	0	Yes	4,5
C	Double-ended DVI line (4")	No	Yes	1	0	3	0	No	1,7,9,11
D	Double-ended cold-leg LLOCA	No	Yes	2	2	4	0	Yes	8
E	Spurious ADS-4[2]	No	Yes	1	1	4	0	Yes	2,6
Long-Term Cooling									
F	Double-ended DVI	Yes	1/1&1/1[3]	1[4]	0	3	0	No	1-5,7,9,10
G	Double-ended DVI	No	1/1&2/1[3]	1[4]	0	4	0	No	6,8,11-13

Notes
1. Break sizes are effective sizes (inside diameter or orifice; not outside pipe diameter).
2. Spurious ADS assumes all four ADS stage 4 valves open at same time as initiating event.
3. Indicates number of valves open/number of flow paths open.
4. These cases bound cases with one accumulator and no CMTs because the difference in CMT – accumulator volumes is small relative to the total containment flood volume.

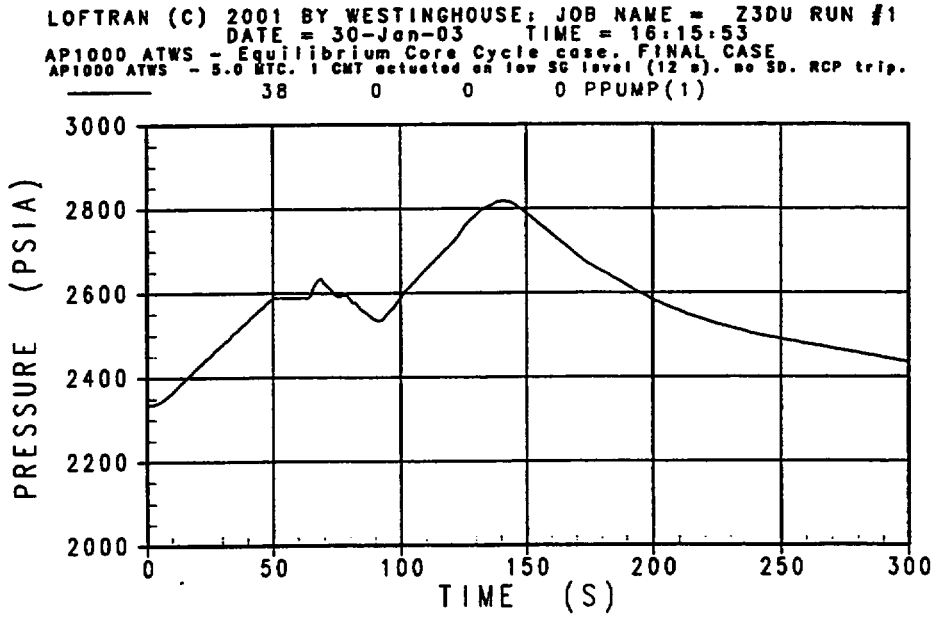
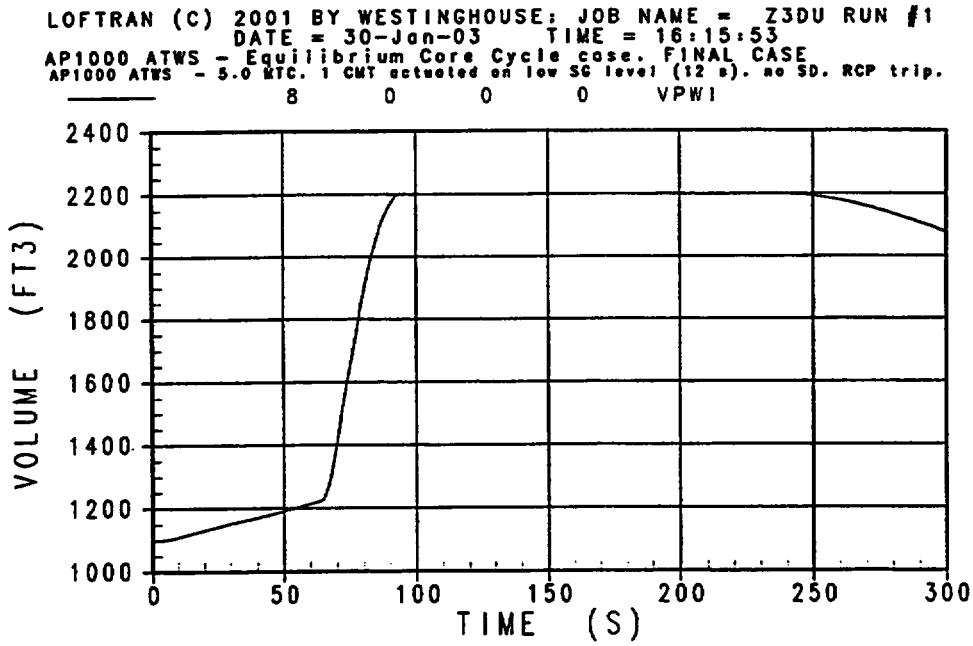


Figure A4.2.3-1

ATWS PRHR Operable – RCS Pressure



Note: The pressurizer volume includes the volume of the surge line.

Figure A4.2.3-2

ATWS PRHR Operable – PRZ Volume

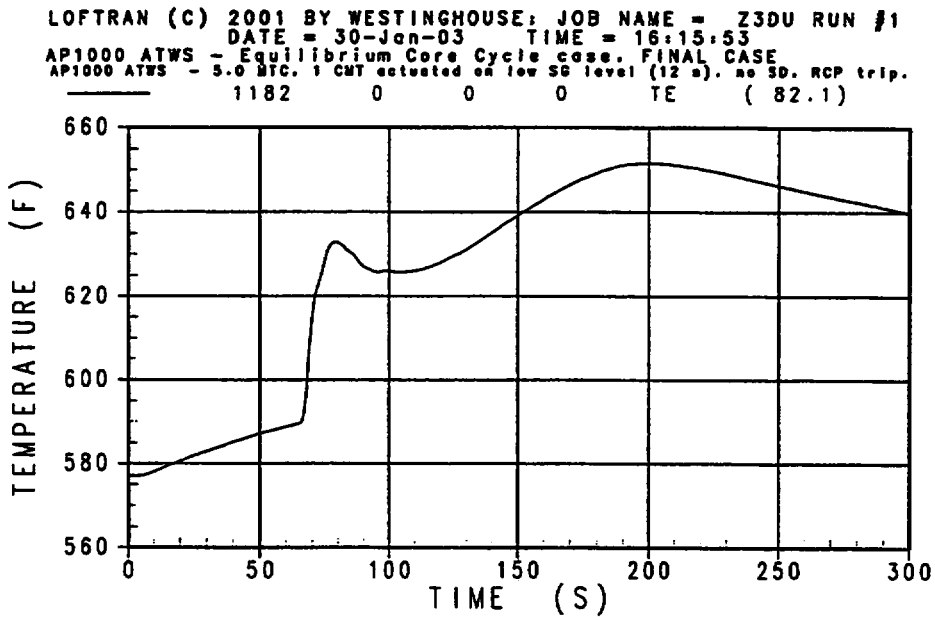


Figure A4.2.3-3

ATWS PRHR Operable – Core Average Temperature

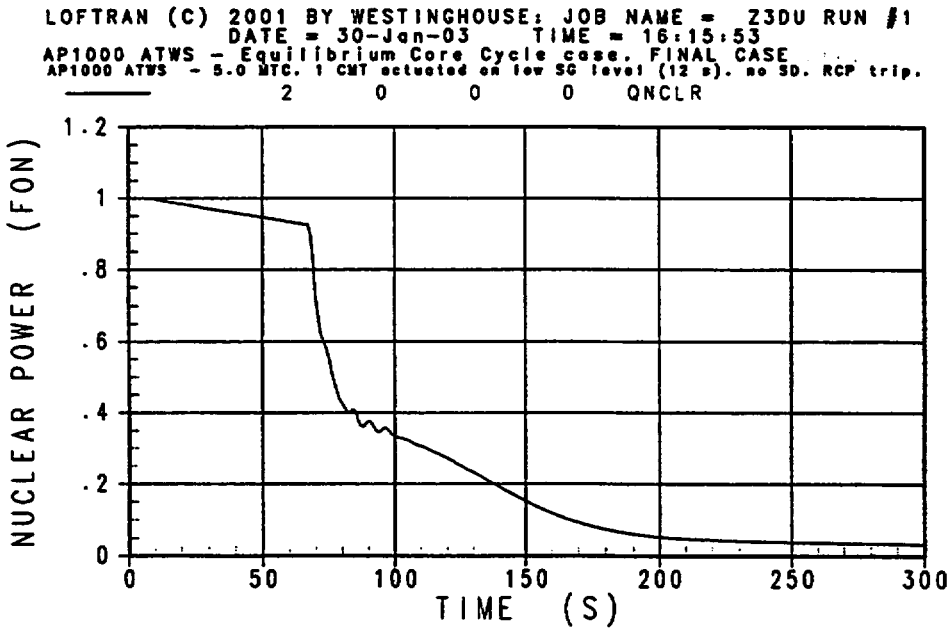


Figure A4.2.3-4

ATWS PRHR Operable – Nuclear Power

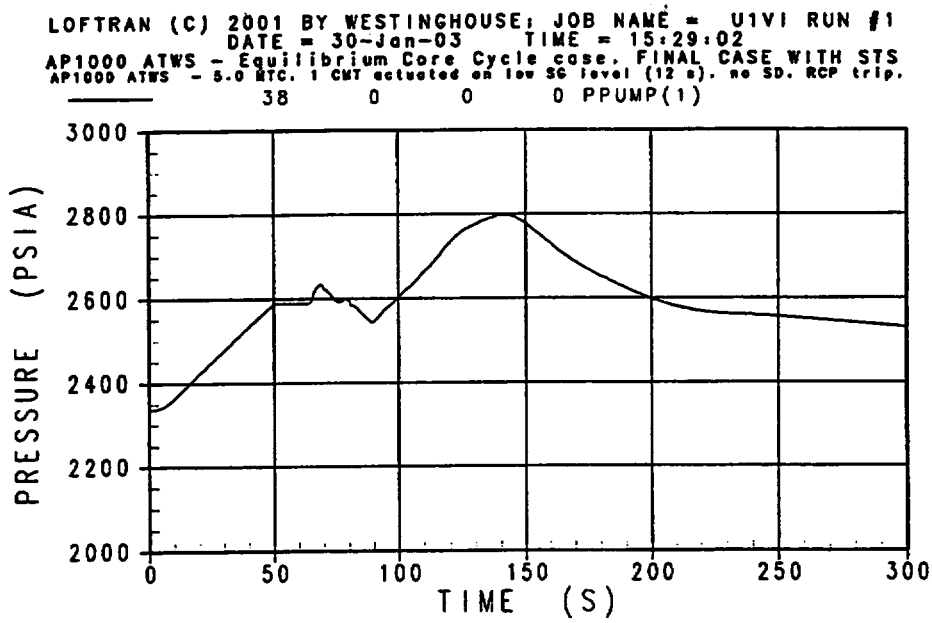
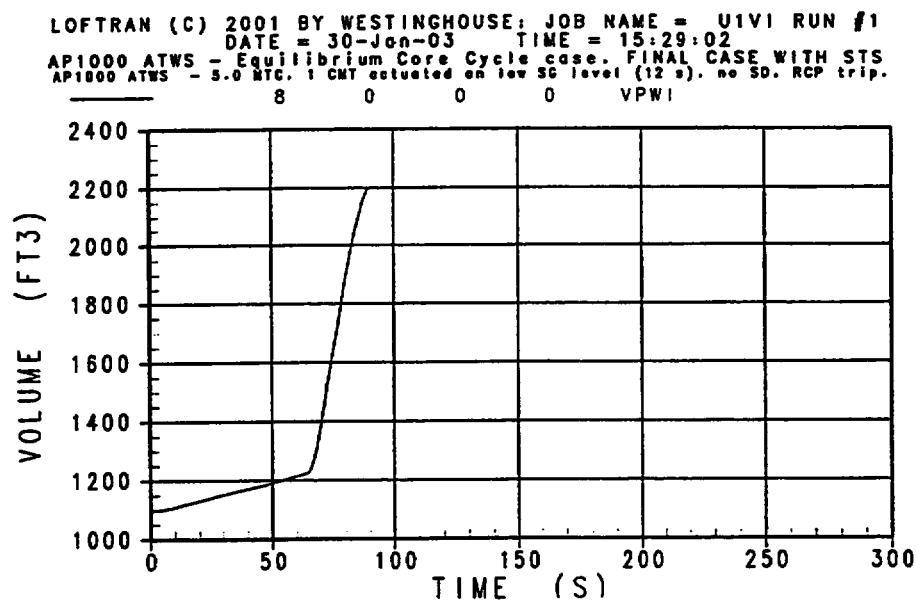


Figure A4.2.4-1

ATWS with SFW – RCS Pressure



Note: The pressurizer volume includes the volume of the surge line.

Figure A4.2.4-2

ATWS with SFW – PZR Volume

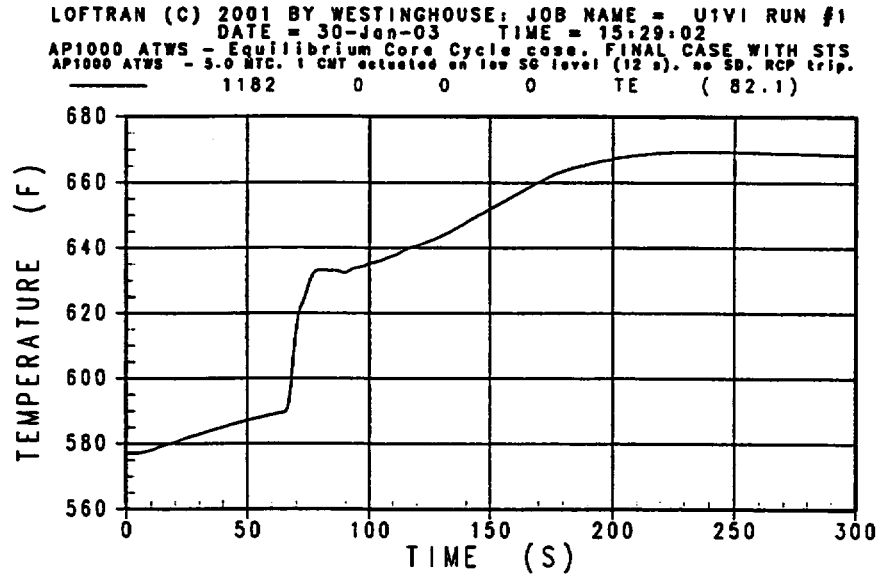


Figure A4.2.4-3

ATWS with SFW – Core Average Temperature

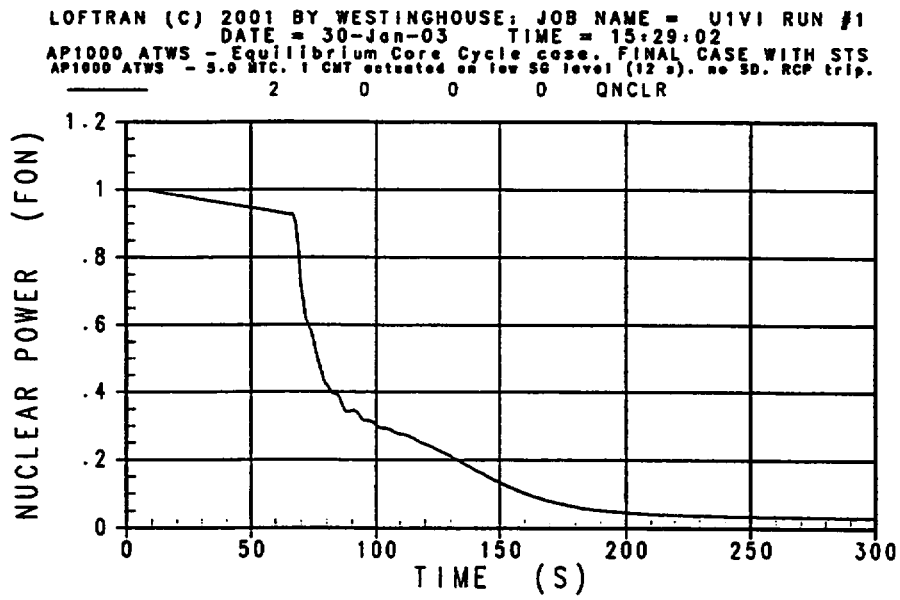


Figure A4.2.4-4

ATWS with SFW – Nuclear Power

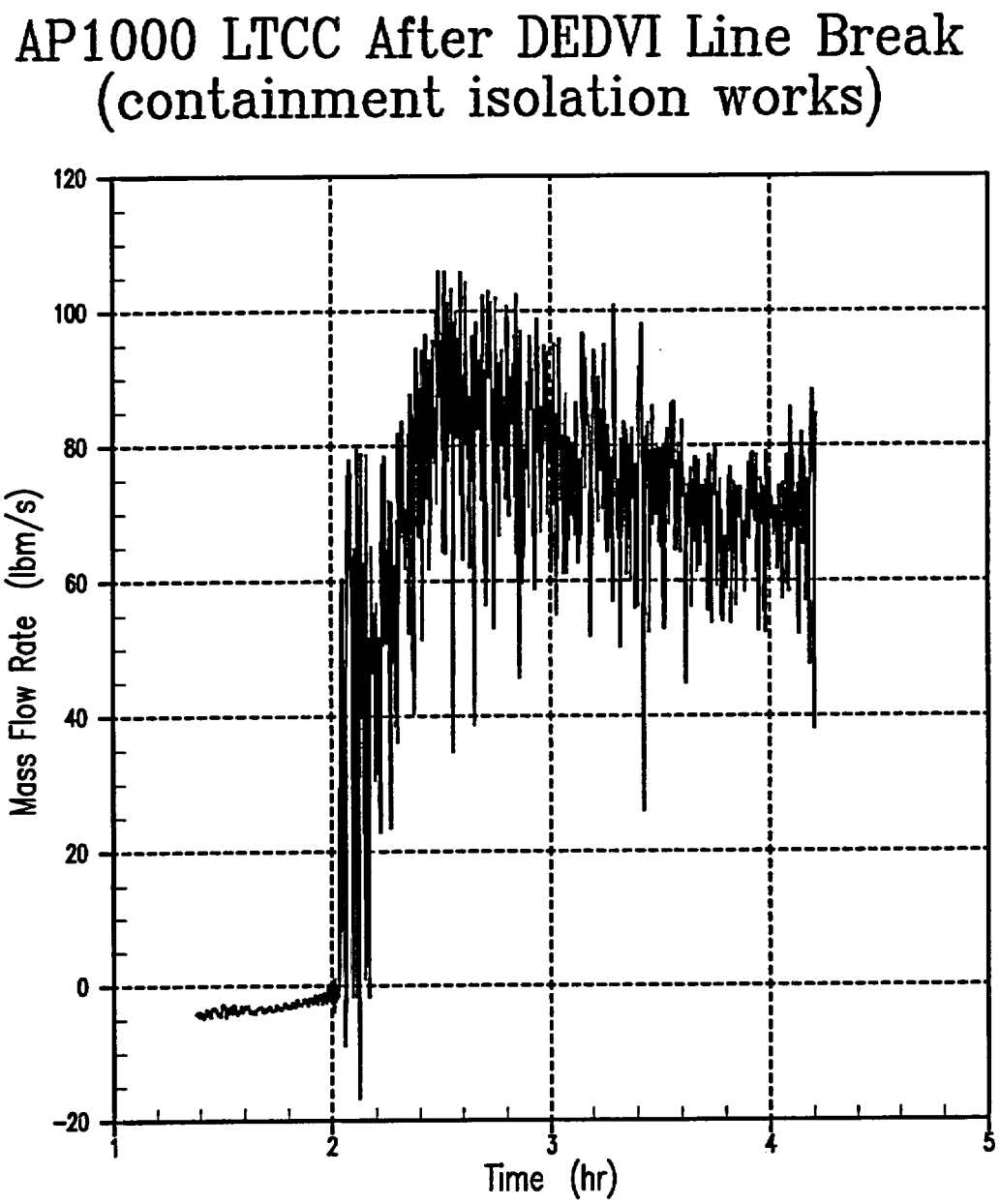


Figure A5.3-13

Case F – DVI-B Mixture Flow Rate

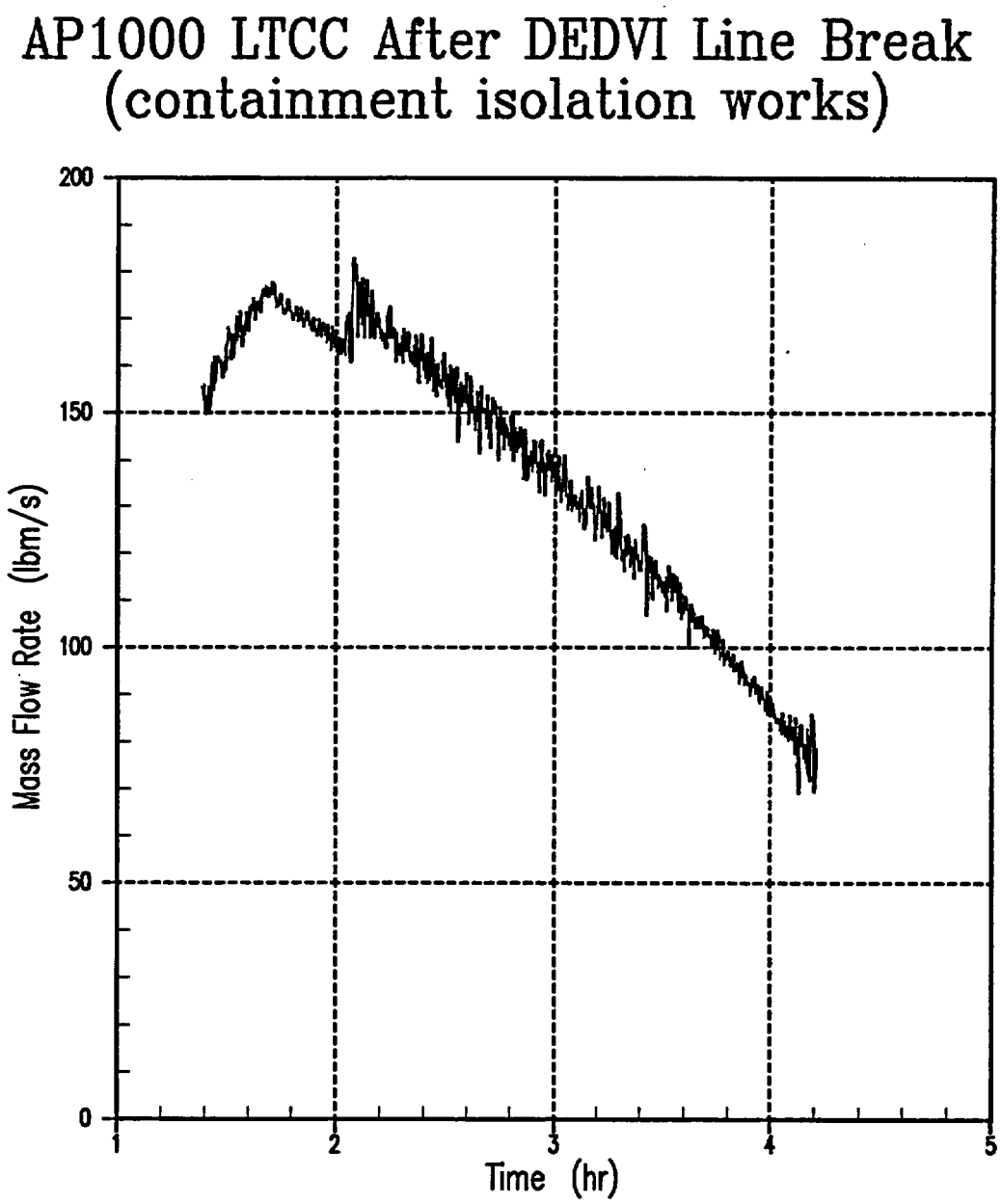


Figure A5.3-14

Case F – DVI-A Mixture Flow Rate

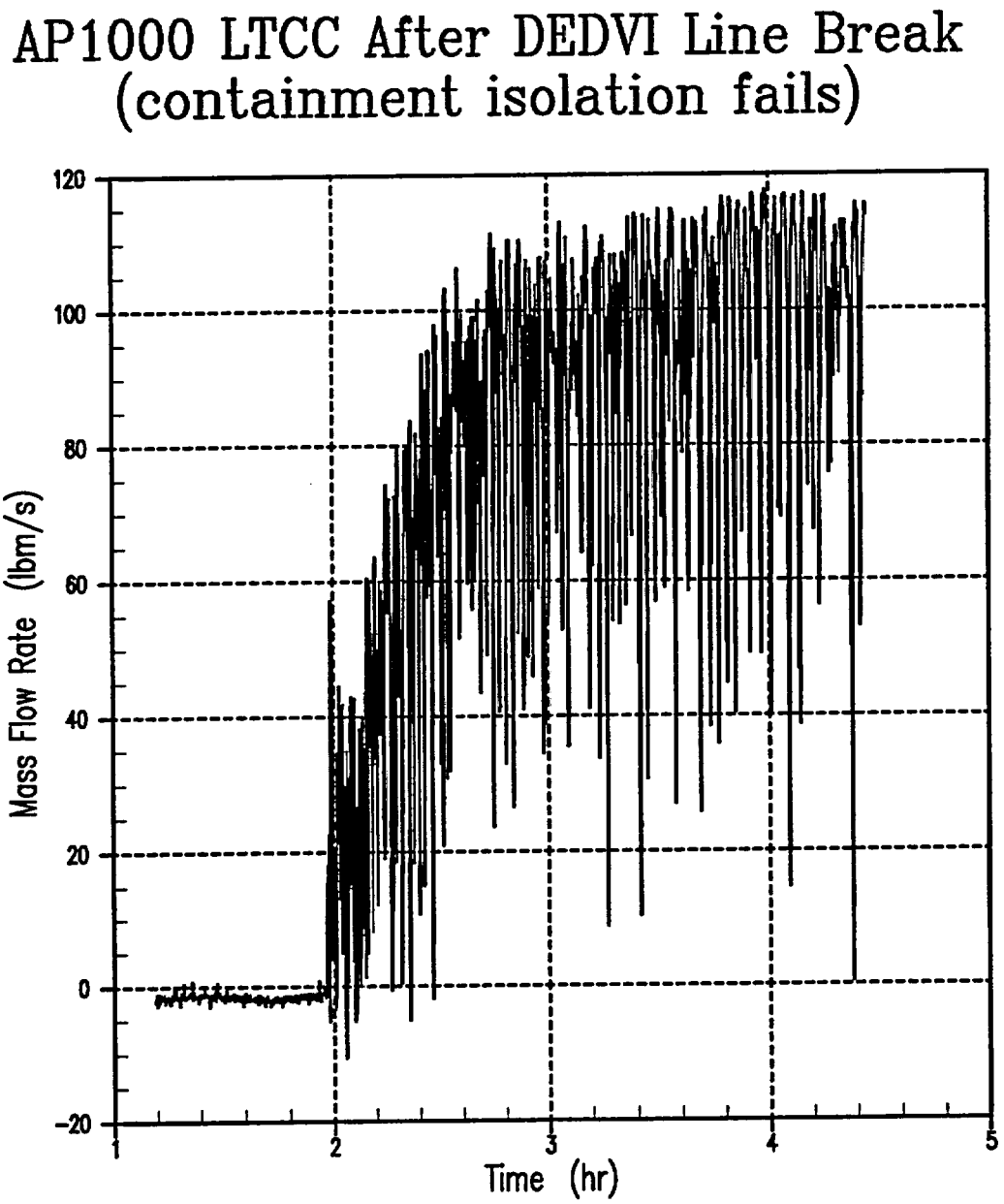


Figure A5.3-27

Case G – DVI-B Mixture Flow Rate

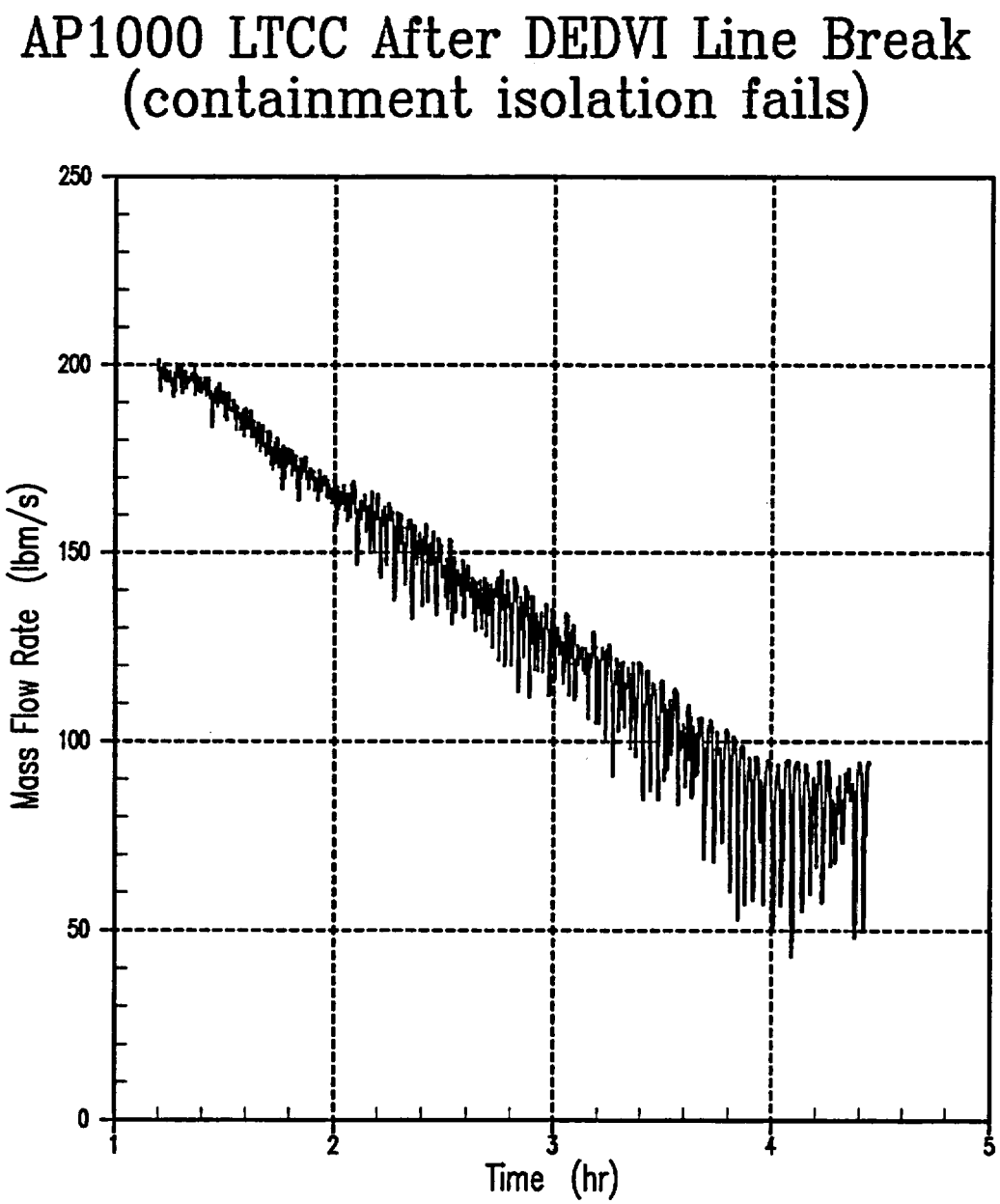


Figure A5.3-28

Case G – DVI-A Mixture Flow Rate

D.6.4 Summary of Equipment and Instrumentation

The equipment and instrumentation used in achieving a controlled, stable state following a severe accident, and the time it operates are summarized in Tables D-3 through D-5.

D.7 Severe Accident Environments

D.7.1 Radiation Environment – Severe Accident

The radiation exposure inside the containment for a severe accident is conservatively estimated by considering the dose in the middle of the AP1000 containment with no credit for the shielding provided by internal structures.

Sources are based on the emergency safeguards system core thermal power rating and the following analytical assumptions:

- Power Level (including 2% power uncertainty) ..... 3,468 MWt
- Fraction of total core inventory released to the containment atmosphere:

Noble Gases (Xe, Kr)	1.0
Halogens (I, Br)	0.75
Alkali Metals (Cs, Rb)	0.75
Tellurium Group (Te, Sb, Se)	0.305
Barium, Strontium (Ba, Sr)	0.12
Noble Metals (Ru, Rh, Pd, Mo, Tc, Co)	0.005
Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am)	0.0052
Cerium Group (Ce, Pu, Np)	0.0055

The radionuclide groups and elemental release fractions listed above are consistent with the accident source term information presented in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants - Final Report."

The timing of the releases are based on NUREG-1465 assumptions. The release scenario assumed in the calculations is described below.

An initial release of activity from the gaps of a number of failed fuel rods at 10 minutes into the accident is considered. The release of 5 percent of the core inventory of the volatile species (defined as noble gases, halogens, and alkali metals) is assumed. The release period occurs over the next 30 minutes, that is, from 10 to 40 minutes into the accident. At this point, 5 percent of the total core inventory of volatile species has been considered to be released.

Over the next 1.3 hours, releases associated with an early in-vessel release period are assumed to occur, that is, from 40 minutes to 1.97 hours into the accident. This source term is a time-varying release in which the release rate is assumed to be constant during the duration time. Additional releases during the early in-vessel release period include 95 percent of the noble gases, 35 percent of the halogens, and 25 percent of the alkali metals, as well as the fractions of the tellurium group, barium and strontium, noble metals, lanthanides, and cerium group as listed above.

There is no additional release of activity to the containment atmosphere after the in-vessel release phase.

The ex-vessel and late in-vessel periods commence after the early in-vessel release period. After 2 hours, the early in-vessel period is assumed to have completed. The late in-vessel period continues for an additional 8 hours. At the completion of the late in-vessel period, the total fraction of the core inventory defined above has been released to the containment atmosphere.

D.7.2 Thermal-Hydraulic Environments

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

The five cases are:

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

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

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

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

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

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

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

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

The case is initiated by a DVI line break. The cavity is flooded, but the PXS compartment is not. The break is above the water level in the compartment. Water is unable to get back into the vessel

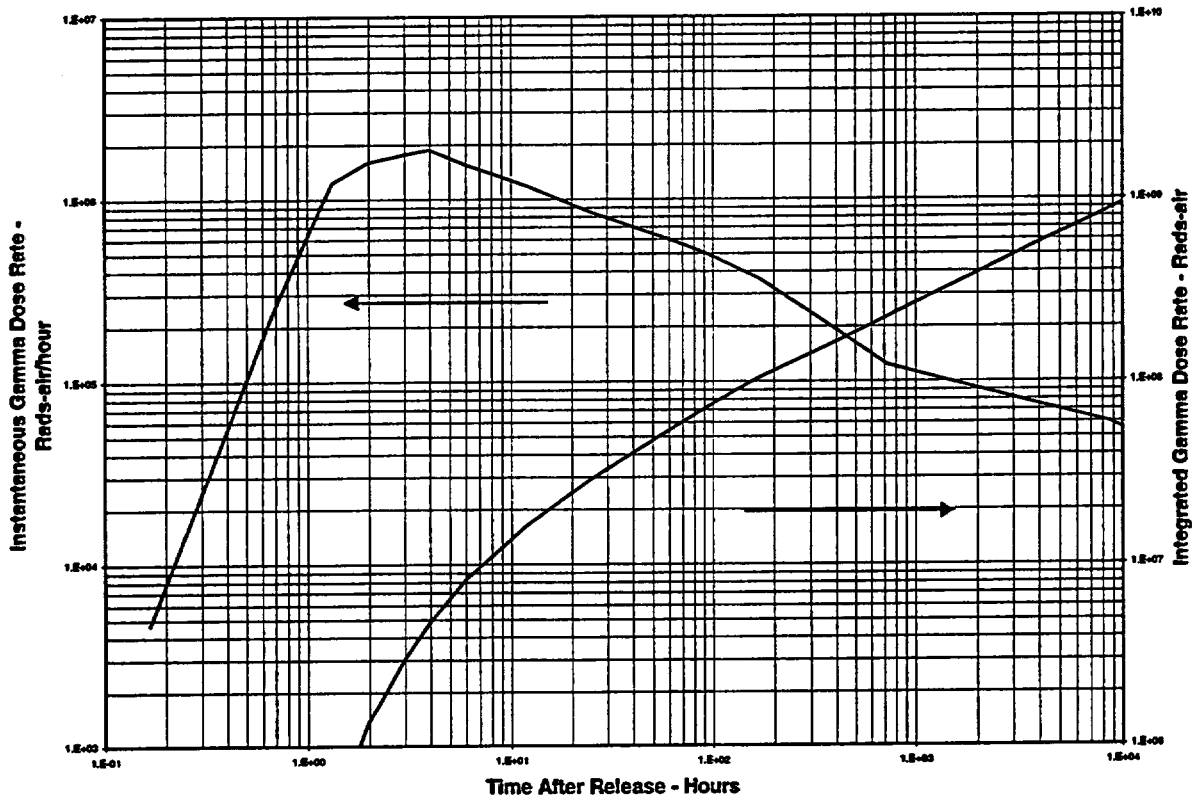


Figure D-1

Post-LOCA Gamma Dose and Dose Rate Inside Containment

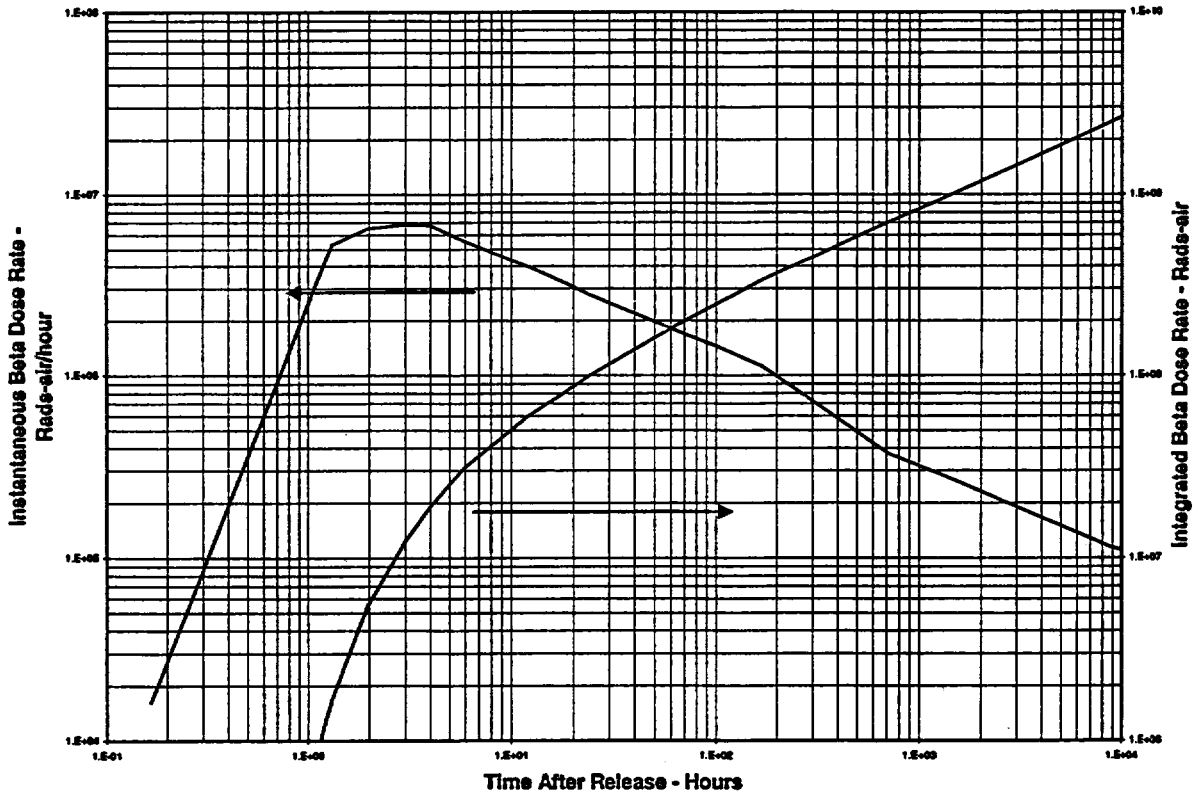


Figure D-2

Post-LOCA Beta Dose and Dose Rate Inside Containment